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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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PLANT LICENSE RENEWAL SUBCOMMITTEE

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TUESDAY

NOVEMBER 17, 2015

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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., Gordon R. Skillman, Chairman, presiding.

COMMITTEE MEMBERS:

GORDON R. SKILLMAN, Chairman

RONALD G. BALLINGER, Member

DENNIS C. BLEY, Member-at-Large

CHARLES H. BROWN, JR. Member

HAROLD B. RAY, Member

PETER RICCARDELLA, Member

STEPHEN P. SCHULTZ, Member

JOHN W. STETKAR, Member

DESIGNATED FEDERAL OFFICIAL:

KENT L. HOWARD, SR.

ALSO PRESENT:

PAUL AITKEN, Dominion

SHERRY BERNHOFT, EPRI

STEVEN BLOOM, DLR

ANGIE BUFORD, NRR

ROBIN DYLE, EPRI

JERUD HANSON, NEI

HEATHER JONES, NRR

MARK KIRK, RES

KEITH LEONARD, Oak Ridge National Laboratory

MARVIN LEWIS, Public Commenter*

ANDREW MANTEY, EPRI*

JIM MEDOFF, NRR

CHRISTOPHER MILLER, Director, DLR

DARRELL MURDOCK, RES

BEN PARKS, NRR

JACOB PHILIP, RES

JOSE PIRES, RES

SRI RAO, RES

SHEILA RAY, NRR

RICHARD REISTER, NEI

JASON REMER, NEI

MOHAMMAD SADOLLAH, NRR

JOHN SIMONS, MPR ASSOCIATES

MADHUMITA SIRCAR, RES

MAKUTESWARA SRINIVASAN, Public Commenter

BRIAN THOMAS, RES/DE

JAY WALLACE, RES

*Present via telephone

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

2 8:30 a.m.

3 MEMBER SKILLMAN: Ladies and gentlemen,
4 good morning. This meeting will now come to order.
5 This is a meeting of the Plant License
6 Renewal Subcommittee. I'm Dick Skillman. I'm Chairman
7 of this Subcommittee. ACRS Members that are in
8 attendance are Pete Riccardella, Harold Ray, Stephen
9 Schultz, our ACRS Chairman, John Stetkar to my left, Ron
10 Ballinger, Charlie Brown, and Kent Howard to my right
11 of the ACRS Staff is the Designated Federal Official for
12 this meeting. Dr. Bley is also with us this morning.
13 Today we will hear presentations from the
14 NRC Staff, the Nuclear Energy Institute, the Electric
15 Power Research Institute, and the Department of Energy.
16 We have not received written comments or requests for
17 time to make oral statements from members of the public
18 for today's meeting.
19 The entire meeting will be open to public
20 attendance. The Subcommittee will gather information,
21 analyze relevant issues and facts, and formulate
22 proposed positions and actions, as appropriate, for
23 deliberations by the Full Committee. The rules for this
24 participation in today's meeting have been announced as
25 part of the Notice in the Federal Register previously

published.

A transcript is being kept and will be made available as stated in the Federal Register Notice. Therefore, I ask that participants in this meeting use the microphones located throughout the meeting room when addressing the Subcommittee. I also request that all attendees please silence their personal electronic devices.

When addressing the Subcommittee, I ask that the participants please identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

We will now proceed with the meeting and I call upon Chris Miller to begin the presentation. Thank you.

DIRECTOR MILLER: Thank you, Chairman Skillman. Good morning. I'm Chris Miller. I'm the Director of NRR's Division of License Renewal, and we really appreciate the opportunity to come forward and update you on the status of Staff and industry activities related to several technical issues involving subsequent license renewal.

I'm going to quickly recap where we've been over the last year or so. Early 2014, the Staff submitted SECY Paper 14-0016, Ongoing Staff Activities to Assess

1 Regulatory Considerations for Power Reactor Subsequent
2 License Renewal. Subsequent to that, members of the
3 Department of Energy, Electric Power Research
4 Institute, Nuclear Energy Institute, and the NRC Staff
5 met with the Subcommittee on April 8th, and with the ACRS
6 Full Committee on May 8th of last year to discuss the
7 SECY paper and to share information on current
8 activities and research in a number of technical areas.
9 Following these meetings, the ACRS issued
10 a letter to the Commission on May 22nd reaffirming that
11 the license renewal rule, as it's currently structured,
12 is the appropriate option for subsequent license
13 renewal, that it maintains a well understood process of
14 life extension, and that it preserves regulatory
15 lessons learned. The ACRS letter also stated that the
16 Generic Aging Lessons Learned Report, the GALL Report,
17 once updated to reflect the latest operating
18 experience, Lessons Learned, and evolving research is
19 the appropriate guidance for SLR.
20 On August 29th of 2014, the Commission
21 issued its SRM regarding SECY-14-0016. Among other
22 things, the SRM directed the Staff to keep the
23 Commission informed on the progress in resolving
24 several issues related to SLR, including reactor
25 pressure vessel neutron embrittlement at high fluence,

1 irradiation assisted stress corrosion cracking of
2 reactor vessel internals, concrete and containment
3 degradation, and electrical cable qualification and
4 condition assessment.
5 In August of this year the Staff sent the
6 Commissioners' Assistant note to update them on the
7 status of these activities. In light of this, I know that
8 you're aware that it's the Applicant's responsibility
9 to provide sufficient information to demonstrate
10 there's an adequate technical basis to justify
11 continued operation beyond 60 years, but both the NRR
12 and the Office of Nuclear Regulatory Research have
13 worked diligent to revise and update both the GALL and
14 the Standard Review Plan for SLR to insure that the Staff
15 can perform an adequate review of subsequent license
16 renewal application, to confirm that an Applicant has
17 provided an adequate technical basis for SLR, and are
18 on a path to issue -- and we are on a path to issue the
19 GALL and the SRP in time to support the first SLR
20 applications.
21 The purpose of today's meeting is to
22 discuss the research activities which are ongoing to
23 address the technical issues from the SRM and the CA
24 note. This will help inform the members of the ACRS on
25 the status of these activities, and how they relate to

1 the various revisions that are being made to GALL and
2 the Standard Review Plan, which we will be discussing
3 with you in February of next year.
4 At the February meeting, we will be provide
5 specific information on what the Staff expects to
6 address, issues for which research findings are not yet
7 available. This would include submittal of
8 plant-specific evaluations of the technical issues, as
9 well as plant-specific programs to address the issues,
10 and to insure adequate management of aging structures
11 and components. So, that's what we look forward to in
12 February.
13 That being said, I'll turn it over to Mr.
14 Brian Thomas for his opening remarks. Brian.
15 MR. THOMAS: Thank you, Chris. Good morning.
16 My name is Brian Thomas. I'm the Division Director for
17 the Division of Engineering in the Office of Research.
18 I do want to express on behalf of the Staff and the
19 management team in the Office of Research appreciation
20 for this opportunity to brief you on the research
21 efforts in support of SLR.
22 As my colleague, Chris, has just mentioned,
23 the Commission asked the Staff to inform them on the
24 progress in resolving the issues on the four major
25 topical areas related to SLR, as Chris just identified.

1 Although the burden is on the Applicant to
2 demonstrate that it can safely manage the effects of
3 aging during the subsequent license renewal, the NRC
4 must be prepared to review applications in a timely
5 manner. This will require a robust and firm technical
6 basis for confirming the adequacy of aging management
7 programs for safety-significant structures, systems,
8 and components. Thus, our Staff has undertaken the
9 necessary confirmatory research to support the Agency's
10 timely reviews of potential subsequent license renewal
11 applications.
12 Several materials, degradation issues were
13 identified in the 5-volume EMDA, the Expanded Materials
14 Degradation Assessment study. Volumes 1-5, that's cited
15 as NUREG CR-7153 through a joint effort undertaken by
16 both the NRC and the DOE LWRS Program.
17 EMDA documents provided a list of sub-issues that fall
18 under the four major technical areas. The NRC Staff
19 grouped these sub-issues under three categories. The
20 shorter term -- one category is the shorter term, which
21 ranges from the time period of two to three years. These
22 are sub-issues that require better knowledge and
23 technical understanding to develop regulatory guidance
24 or revise existing positions that could potentially be
25 non-conservative, or highly uncertain. The other

1 category is the longer term -- our longer term needs
2 which we look at over a period of three to eight years.
3 These are sub-issues that require the development of
4 technical bases to support future regulatory guidance
5 on aging and monitoring requirements.
6 And the third category is the category for
7 which we thought there was no additional research
8 needed. Within this category, these are sub-issues
9 where the technical basis is judged to be adequate to
10 support the regulatory positions that have been
11 developed, and there is no expectation that industry
12 will attempt to demonstrate that a less conservative
13 position can be accepted.
14 When relevant research findings may not be
15 available in time to be incorporated into the GALL or
16 the SRP for the Staff's review of the lead SLR
17 applications, the Staff will expect the Applicants to
18 perform plant-specific evaluations, as was mentioned by
19 Chris, of the technical issues and submit
20 plant-specific programs to insure adequate aging
21 management of the applicable SSEs.
22 The NRC Staff continues to work with the
23 industry, DOE, and EPRI to align on research priorities
24 to support the resolution of major technical issues for
25 SLR. To this end, the Staff from these organizations are

1 engaged in active discussions to identify, prioritize,
2 and execute research activities so that they are better
3 aligned with the shorter and longer term needs for
4 resolving the major technical issues.
5 These discussions will foster mutual
6 understanding of the technical issues that need to be
7 addressed by industry, mitigate the impacts of reduced
8 funding for research, and provide opportunities for NRC
9 to harness relevant data, information, and knowledge
10 gained through LWRS research activities to better
11 inform the Commission's SLR process.
12 The mutual sharing of the data and the
13 information does not imply identical interpretations or
14 conclusions. The NRC Staff would utilize the results to
15 independently interpret the regulatory implications to
16 aid the Staff's review of SLR applications.
17 As part of today's meeting, our Staff will
18 present the specific research activities outlining the
19 objectives, execution of the activities, and the
20 expected outcomes. They will also provide updates on the
21 progress made towards the current status of resolution
22 of the technical challenges.
23 I will now turn it over to Jerud for his
24 presentation.
25 MR. HANSON: Good morning, everyone. I'm

1 Jerud Hanson with the Nuclear Energy Institute. I'm the
2 Industry Lead for SLR. I'd like to start off by first
3 thanking the members of the NRC Staff, the Division of
4 License Renewal, and I'd like to particularly thank
5 Butch Burton with the Division of License Renewal for
6 all the work and the effort he's put into preparing for
7 this meeting. It's a big effort and his coordination was
8 very helpful.
9 I'd also like to extend my thanks to the
10 members of EPRI and the Department of Energy that are
11 going to be presenting throughout the day. Not only for
12 today, they provided a lot of support, and they continue
13 to be dedicated for SLR, but even aside from today, they
14 do an excellent job, and I want to recognize them for
15 that, so thank you.
16 As Chris was going over earlier, we've
17 accomplished a lot in the last year since the last
18 meeting, nearly a year, and we're going to be going over
19 those different items, different tasks in greater
20 detail as we move through our presentations, so I'm
21 going to keep my introduction short and just say thank
22 you. And at this point, I believe I'm going to turn it
23 over to Steve.
24 MR. BLOOM: Good morning, thank you. My name
25 is Steve Bloom, and I'm the Chief of the Subsequent

License Renewal Branch in NRR. I'd like to go over the agenda and structure of today's presentations. Next slide.

EPRI and DOE will begin with an overview of EPRI's program for long-term operations for nuclear power plants and DOE's Light Water Reactor Sustainability Program, along with a discussion of metals research and development related to SLR.

Following that, the NRC Staff will have several presentations starting with the presentation on the fluence evaluations for SLR. Fluence was not specifically identified in the Commission's SRM as one of the four technical issues, but it impacts all of the other technical issues. And as such, we felt that giving you a status of our fluence work is important to understand the other technical issues.

After that, there will be discussions on vessel embrittlement, irradiated assisted stress corrosion cracking of internals, concrete degradation, and cable qualification and assessment. And starting with the fluence discussion, the NRC Staff, DOE, and EPRI will present together. We believe this will help to better focus both the Staff and industry presentations, and to be better respond to any questions the Subcommittee may have. We've also allowed

1 additional time after the presentations to answer any
2 additional questions you may have. Next slide, please.
3 This slide provides the schedule for the
4 upcoming actions to issue the NRC GALL and SRP for SLR.
5 We hope to issue the draft GALL and SRP for public
6 comment by the end of December of this year. There will
7 then be a 60-day public comment period starting in early
8 January, and ending in late February. During that
9 period, the NRC plans to host at least two public
10 meetings to help address and clarify issues and address
11 questions that stakeholders may have on these draft
12 documents. Our hope is that holding these meetings will
13 help stakeholders formulate meaningful and valuable
14 comments on the documents.
15 After the comment period, the Staff will
16 review the comments and revise the GALL and SRP, as
17 needed. As discussed earlier, we have another meeting
18 scheduled in February of next year, and at that meeting
19 we will again show you how the GALL and SRP have been
20 revised for SLR, including revisions associated with
21 the technical issues that you will hear discussed today.
22 We plan to hold two technical sessions
23 during the regulatory information conference in March.
24 The first will be hosted by NRR and will focus on the
25 licensing-related issues associated with SLR. The

1 second will be hosted by research and will focus on the
2 same technical issues. There may be a third session
3 devoted specifically to activities associated with ASR
4 or alkali-silica reaction.
5 After the GALL and SRP are revised based on
6 the public comments, the Staff will be ready to issue
7 the final GALL, SRP, and the technical bases document.
8 The bases document provides the Staff's resolution of
9 all comments that were received, and explain the bases
10 for these revisions that were made. We plan to again
11 address the ACRS in February of 2017 before they're
12 issued as final documents. And, therefore, we plan to
13 issue them finalized in May of 2017, well before the lead
14 plants submit their applications, which we believe will
15 be in either late 2018 or early 2019.
16 Issuing the GALL and SRP this early should
17 help Applicants prepare and submit SLR applications
18 that address all the relevant issues which will help
19 reduce the number of follow-up questions from the Staff,
20 and help to make an efficient and effective review of
21 the applications.
22 Unless there are any questions, I'll turn
23 it over to EPRI and DOE for their first presentation.
24 MEMBER STETKAR: Yes, Steve, there is a
25 question.

1 MR. BLOOM: Yes, sir.

2 MEMBER STETKAR: And I know we're going to
3 hear the Subcommittee in February of next year about the
4 draft GALL report, but I'm just curious why did the Staff
5 feel it's necessary to issue a separate GALL report for
6 subsequent license renewal? Why isn't this simply Rev.
7 3 of the GALL report?

8 MR. BLOOM: Because Rev. 2, which is the
9 document that have GALL and SRP that are still allowing
10 plants to come in for their initial licenses, has the
11 backing of technical basis that goes out to 60 years.
12 That same bases does not allow licensees to go out to
13 80 and, therefore, we felt it was necessary to come out
14 with a document that addresses the issues that go out
15 to 80 years, while at the same time not installing these
16 new programs for plants that only are coming in for their
17 initial licensing out to 60.

18 MEMBER STETKAR: That's -- I guess we'll
19 learn more about that in February, but it's just
20 curious.

21 MEMBER SCHULTZ: Steve, on your topic, you
22 mentioned the public interactions that are going to
23 happen over the next few months.

24 MR. BLOOM: Yes, sir.

25 MEMBER SCHULTZ: After the public comment

1 period is completed you've got a long time before 2017.
2 What public interactions are you planning over that year
3 and a half --
4 (Simultaneous speaking.)
5 MR. BLOOM: We haven't yet finalized it, but
6 the plan is that once we've done our initial assessment
7 of the public comments and determined how we're going
8 to make changes, we plan to hold additional sessions to
9 kind of explain to everyone how we made determinations
10 for the four different areas that you'll hear about
11 today. So, we'll have a session just on concrete and say
12 hey, based on our -- based on the public comments, this
13 is how we're determining that we think we should go in
14 the direction of concrete, or cabling, or whatever it
15 is. So, we will have additional public sessions at that
16 point, too.
17 MEMBER SCHULTZ: On a topic basis.
18 MR. BLOOM: Correct.
19 MEMBER SCHULTZ: Thank you.
20 MEMBER BLEY: Steve, one more follow-up on
21 what John asked you. I don't want to completely wait
22 until February. During the day today if it's possible
23 to point out places where you see requirements actually
24 changing, we would appreciate that, or if you can find
25 the places that will focus on the technical basis, as

1 well. That would be helpful.

2 MR. BLOOM: Okay, thank you. And we'll try

3 to do that.

4 MR. HANSON: Hello. Jerud Hanson with NEI.

5 Before we move into EPRI's technical presentations, I

6 just wanted to provide a brief introduction, a summary

7 of why we're here today, so please, next slide.

8 As we're all aware at this point, by 2040

9 we're anticipating if we move -- go to 2040, half of the

10 nation's nuclear power plants will have operated for 60

11 years. By 2030, we could experience electricity

12 shortages if a significant number of plants are retired.

13 And based on what we're seeing with current economic

14 positions, in the next four years we could very likely

15 see that. And although new plants are currently under

16 construction in regulated states, there's little

17 economic incentive to continue building new plants.

18 It's the industry's position that pursuing

19 license renewal is the surest way to avoid this outcome

20 and insure that nuclear power continues playing a role

21 in the U.S.' electricity portfolio. Next slide, please.

22 So, I'm just going to recap a little bit on

23 some of the activities that Chris mentioned earlier.

24 What we've done since the last meeting, first of all,

25 we've completed and published an SLR roadmap, and this

1 provides a good pathway for SLR, as well as a generic
2 overview of the SLR process that could be used with
3 essentially any audience, whether it be the industry or
4 members of the public. The Committee has been provided
5 with some bound copies of this book, if you'd like to
6 review it.
7 Also as a lead for the industry in the GALL
8 review, we've coordinated a strategy and a team of
9 stakeholders who stand ready to review and provide
10 comments to the draft GALL upon issuance. As we move into
11 2016 and continuing SLR, this is the next big step we
12 need to take, and we're ready and excited to get our
13 hands on the GALL, and review and provide comments.
14 One of the other things we've done in the
15 last year that's been very helpful is we've conducted
16 site visits. We've done there, one at Salem-Hope Creek,
17 one at EPRI's main research and development laboratory
18 in Charlotte, and also at Oak Ridge National Laboratory.
19 Here both industry and the NRC Staff participated on
20 tours and received overviews from the facility staff on
21 ongoing research and development in the areas of
22 concrete, cable, and metal aging management programs.
23 And most importantly recently, Dominion recently
24 announced its intention to pursue second license
25 renewal for Surry Station, and we continue -- we

1 anticipate to hear more of this as we move into 2016.
2 MEMBER SKILLMAN: Jerud, let me ask you, why
3 the site Salem-Hope Creek was chosen when there are some
4 very aging plants still operating very well? Why weren't
5 those very old plants also visited?
6 MR. HANSON: I wasn't participating in SLR
7 at that time, so I can't answer that question right now,
8 but I believe Jason can.
9 MR. REMER: Jason Remer, NEI. We just saw an
10 opportunity with Salem-Hope Creek. Salem-Hope Creek
11 volunteered, they were getting ready for an assessment
12 from the NRC, so they were -- had some plant equipment
13 that was open, some trenches that had been dug so we
14 could actually see the inspection. So, it was more of
15 an opportunistic opportunity, and they were gracious to
16 host us, so that was really the only reason we did
17 Salem-Hope Creek.
18 MEMBER SKILLMAN: Thank you.
19 MR. REMER: Sure.
20 MR. HANSON: Thank you, Jason. Next slide,
21 please.
22 Okay. So, this is our org chart for SLR at
23 NEI, and there's just a few different points I'd like
24 to point out about this. One, as you can see, we have
25 an industry working group task force or focus group

1 dedicated to every technical area of SLR. Some recent
2 changes we've made that I'd like to point out is, first
3 of all, we now have Bill Pitesa, the Chief Nuclear
4 Officer of Duke Energy, who's been assigned as Chairman
5 of the SLR Executive Working Group. And this is
6 important because we want to insure that SLR continues
7 to be a priority and receives the appropriate resources
8 necessary. This will also be helpful due to Bill
9 Pitesa's sitting on the Nuclear Strategic Initiatives
10 Advisory Committee, or NSIAC, to insure this remains a
11 topic of discussion. Also a recent change we've made is
12 we now have Nancy Ranek with Exelon assigned to lead the
13 SLR Environmental Task Force. Those are some recent
14 changes I wanted to point out with the org chart at NEI
15 assigned to SLR. Next slide, please.
16 MEMBER SKILLMAN: Would you go back to that
17 slide?
18 MR. HANSON: Sure.
19 MEMBER SKILLMAN: What is the difference in
20 that image, the assignment of Nancy Ranek to SLR
21 Environmental Task Force versus SLR LR Task Force, Jerud
22 Hanson? What's the difference?
23 MR. HANSON: Okay. The SLR Environmental
24 Task Force strictly addresses the environmental portion
25 of SLR. And the SLR LR task -- License Renewal Task Force

basically addresses everything other than environmental. Nancy Ranek was chosen specifically because of her extensive background working specifically in environmental SLR issues.

MEMBER SKILLMAN: So, she is really birds, bees, and bogs for your pots and pans?

MR. HANSON: You could put it that way, yes.

MEMBER SKILLMAN: Okay, thank you.

MR. HANSON: So, why are here today, and what are we going to hear? EPRI and DOE have continued to conduct scientific research to understand the technical issues associated with long term operation of nuclear power plants, and that's what we're going to hear a lot of today.

What this research demonstrates is that there are no generic technical issues that would prevent a well maintained nuclear power plant from operating safely during the next -- the second license renewal period. We think learnings from this research, development combined with the updates to industry guidance, continuous monitoring, updating and replacement of parts and systems, and operating experience will help insure nuclear power plants continue to operate safely until 80 years. That's why we're here. Are there any questions?

1 MEMBER SKILLMAN: Thank you.

2 MS. BERNHOFT: Okay. While she's pulling up
3 the next presentation, it will be a joint introductory
4 presentation provided by Rich Reister and myself. I know
5 I've talked to the group before, but for those that don't
6 know me, I'm Sherry Bernhoft. I work for the Electric
7 Power Research Institute, and I am the Program Manager
8 for Long-Term Operations. The Long-Term Operations
9 Program formally started in 2010 to address the
10 additional research that was going to be needed to
11 transition the plants from thinking about an end of life
12 of 60 years, and taking that out to 80 years of
13 operations. You can go to the next slide, please.

14 I'm going to move through this fairly
15 quickly. For those who are not familiar with EPRI, we
16 have three aspects. We're independent, even though the
17 funding source does come from the utilities, our
18 research is independent. You know, we report what comes
19 out of the research findings. Not for profit, is
20 chartered to serve for public benefit, and we're very
21 collaborative. You can go on to the next slide, please.

22 This is an example, and you're going to see
23 throughout the presentations that will be made by EPRI
24 and DOE Staff over the next few hours, a lot of the basic
25 research and development is done by the National Labs,

1 funded with the DOE LWRS funding. EPRI's role is really
2 into the collaboration of the technology and a lot of
3 the applied applications of that, so we work together
4 quite well.
5 You'll hear throughout today that we had
6 developed joint roadmaps, we have frequent technical
7 interactions between our technical staff, and we're
8 always questioning, you know, how is this going to lead
9 to building the technical case for safe, long-term
10 operations of the plants within the U.S. and
11 internationally.
12 We also brought in, as needed, we'll talk
13 to suppliers and the vendors, as well, to make sure it's
14 all coordinated. We're not in competition, we're in
15 collaboration.
16 We mention two, that really should change
17 now to 100 reactors, the Watts Bar Unit 2 now, but we
18 talked about OE is very important. And this is just a
19 snapshot that we have global participation with our EPRI
20 membership, so we collect a lot of operating experience,
21 and we use that combined with the research to talk about
22 our guidance documents, how they can be updated and
23 enhanced, and used by the members for implementation of
24 aging management programs. Next overhead, please.
25 This is just a snapshot --

1 MEMBER SKILLMAN: Back a slide, that slide,
2 please. What is the extent for foreign countries of the
3 collaboration, particularly for countries like Russia,
4 Germany, and France that have extensive experience?
5 MS. BERNHOFT: Okay. With the Russians, we
6 do have some export controls issues that, you know,
7 prohibit other than publicly available information for
8 exchange with them. With France, EDF is our largest
9 single member, and we collaborate quite a bit with EDF.
10 We joint fund the Materials Aging Research Institute in
11 France. The two primary funders are EPRI and EDF, and
12 we do quite a bit of joint research with EDF there, and
13 we learn from their experience, are active members of
14 many of our committees.
15 MEMBER SKILLMAN: Deutschland, Germany?
16 MS. BERNHOFT: Germany is not as active
17 right now due to some of their political considerations.
18 They've joined some of our programs over the year, so
19 we learn as we can from their publicly available
20 information.
21 MEMBER SKILLMAN: Sherry, thank you.
22 MEMBER STETKAR: As long as -- I have to be
23 fair to Asia, China, have you had any outreach with China
24 at all?
25 MS. BERNHOFT: China is our largest growing

1 member right now.

2 MEMBER STETKAR: They are --

3 MS. BERNHOFT: Actually, the entire Asian

4 sector we've -- we've actually put technical boots on

5 the ground now in Asia, because that whole sector

6 between Japan's restarting and China with their new

7 build is our largest growth area right now for EPRI.

8 MEMBER STETKAR: Thank you.

9 MS. BERNHOFT: Tina, did you have anything

10 to add? Okay, thank you.

11 And this is just a snapshot right now of

12 some of our major players with where we do our

13 international research. We have LTL projects going on

14 with KNHP right now on environmentally assisted

15 fatigue. As I talked about at the EDF, we co-own or we

16 co-fund the Materials Aging Institute there. UJV, we do

17 -- we're doing some digital electronic research with

18 them right now. CEA, I'll be talking later about a

19 project on boric acid spent fuel pools that's actually

20 occurring at CEA. So, this is just a snapshot of some

21 of the international collaboration and research we have

22 going on. And, of course, Department of Energy is a very

23 significant partner with us. You can just keep going.

24 This is --

25 So, what I wanted to say, too, for EPRI, for

1 just the nuclear sector, a total of four sectors, for
2 just the nuclear sector, we spend about $170 million of
3 research a year. These are the major program areas
4 within EPRI, and what's really fun about working on
5 long-term operations is I have the ability that I draw
6 on all of these program areas, some of the best subject
7 matter experts. It's a very matrixed organization to
8 support long-term operations, so you've got about 40
9 years of research here. You've got $170 million of
10 research. We're drawing on the best for pulling together
11 the incremental needs for long-term operations. Our
12 goals and objectives is --
13 MEMBER BROWN: Where does the money come
14 from? Is that all industry money, or is it
15 Congressionally appropriated money? I'll let you answer
16 the question instead of me trying to --
17 MS. BERNHOFT: Yes. We have several sources
18 of money. Most of the money does come from our industry
19 utility members. We do get some money from Department
20 of Energy for specific projects, but the majority of the
21 funding will come from the utility industry members.
22 MEMBER BROWN: Okay. And the next question
23 that goes with that, based on the -- all the fundamental
24 long-term operation stuff you all identified in the
25 other documentation, the fluence, the pressure vessel

1 materials, how -- do you have full control over how that
2 is managed so that it can be focused on the specific
3 things you need and it doesn't get vectored off to -- I
4 don't know the right term for it, but other interested
5 activities that don't directly affect or help getting
6 down the pipeline whether these materials are okay for
7 the longer term?
8 MS. BERNHOFT: Okay. I'll give you the
9 5-minute answer instead of the 8-hour answer.
10 MEMBER BROWN: About 30 seconds --
11 MS. BERNHOFT: I understand. We have in each
12 of the major program areas, we have utility advisors.
13 They're advisors. At the end of the day, the strategic
14 plan, the directions, and the need to support the
15 research will be set by the -- I'm sorry, the industry
16 -- some of the industry executives to get their input,
17 but it's really going to be the EPRI Executive Team
18 that's going to appropriate the research to support. And
19 we do keep money carved out for long-term operations
20 within that funding scenario. So, we do have advisors,
21 we get input from our advisors, but the ultimate
22 decision is made by EPRI.
23 MEMBER BROWN: Okay. So, I'm going to make
24 an assumption based on your comment that based on the
25 specific material, concrete, other type technical

1 research that you need to do that you can control and
2 make sure you get sufficient breadth of research to
3 insure you've covered the bases for the various types
4 of plants that you have to support for extended
5 operations.
6 MS. BERNHOFT: Like every organization
7 -- the short answer is going to be yes, but like every
8 organization, I do have to prioritize within that. So,
9 as Brian said, you know, some things are going to be one
10 to two years. I set out the need to complete the research
11 for those, and it's going to be three to five years. So,
12 it's up to me to prioritize, and I take input from the
13 advisors, with DOE, with NRC Research.
14 MEMBER BROWN: The reason I ask that is a
15 comment was made that Surry is looking like it's going
16 to make an application, and I guess if I had to
17 prioritize money, I would try to look at what specific
18 issues do I have to deal with there from a materials,
19 all these other type -- and try to make sure I can cover
20 that first plant which ought to help really set the pace
21 for the rest of them.
22 MS. BERNHOFT: Yes.
23 MEMBER BROWN: Is that -- am I thinking
24 properly, or --
25 MS. BERNHOFT: That is actually going on

1 right now.

2 MEMBER BROWN: Okay, thank you.

3 MS. BERNHOFT: Yes, because the first plants

4 -- and it's not just the first plant. I'll just a minute

5 here to -- we're asking who is the next wave? That first

6 grouping of plants coming through totally agree they

7 need a sound technical basis and we need to support

8 success.

9 MEMBER BROWN: Thank you.

10 MEMBER SKILLMAN: Sherry, as you say that

11 would it be accurate for us to assume that those

12 potentially identified lead plants are having active

13 discussions now, not a negotiating thing but a technical

14 discussion? What are the big hitters, that type of

15 thing?

16 MS. BERNHOFT: I'm going to put somebody on

17 the spot. Paul, would you mind addressing that question?

18 MR. AIKEN: So, Paul Aiken from Dominion.

19 So, as we've been talking about, you know, we've been

20 thinking about what call it second license renewal, but

21 subsequent license renewal for the last year at least

22 within Dominion, and we've been actively engaged with

23 the industry through NEI, and through EPRI, through the

24 ongoing meetings that we have on research, so I'm a Vice

25 Chairman on one of the committees. So, we help Sherry

1 set priorities for long-term operations for second
2 license renewal, so our input, as well as that of the
3 various executives representing the various utilities
4 are all coming to see the light for second license
5 renewal. We've all been dealing with Fukushima, GS-191,
6 so we're going to be rolling into second license renewal
7 also. It's going to be -- the focus is certainly
8 transforming, and I think the right attention is being
9 put on the right research right now.
10 MEMBER SKILLMAN: Let me ask -- while you're
11 up, Paul and Sherry.
12 MR. AIKEN: Sure.
13 MEMBER SKILLMAN: It would be easy to think
14 that this discussion is solely around major technical
15 issues, but to a timid public the idea that one of these
16 machines might be allowed to operate 20 additional
17 years, when some of the public is hostile to the existing
18 addition of 20, from 40 to 60. In my view, this becomes
19 somewhat of a public perception issue and, hence,
20 communication with the public about the idea of a second
21 license renewal from 60 to 80 is a major issue. How is
22 that being addressed?
23 MS. BERNHOFT: I'll let Jerud handle that.
24 MEMBER SKILLMAN: Okay.
25 MR. HANSON: What I can tell you to answer

1 that question is NEI as an organization has a division
2 within NEI that is specifically dedicated to
3 communicating the beneficial uses of nuclear
4 technology, communicating with the public through the
5 technical means and opportunities necessary.
6 As you'll see from the SLR roadmap, this is
7 just one of the tools that NEI is using in that form to
8 communicate the beneficial uses, not only of nuclear
9 technology, but the benefits of extending our plants
10 from 60 to 80 years. So, we have at NEI what we believe
11 is a fairly aggressive communication campaign to
12 communicate specifically SLR, what the objectives are,
13 why it's a good idea, and why it's -- will continue to
14 be a safe, reliable form of electricity for this
15 country. So, we are doing that and that is how we are
16 doing it at NEI.
17 MR. AIKEN: And I would just add to that, I
18 think it was last year we had a focus group, NEI
19 sponsored some focus group discussions for the citizens
20 around Philadelphia, different backgrounds,
21 diversity, educational, and they talked to them about
22 the concept of going forward. And the general consensus
23 from the public was they were okay with it, as long as
24 the NRC was involved and engaged, and they had the
25 authority to shut down a plant if they weren't operating

appropriately. So, that was the overwhelming response from these focus groups. So, I thought that was -- you know, that might be just an indicator of what the public perception is.

MEMBER SKILLMAN: I thank both of you. Jerud.

MR. HANSON: I'd like to just make one more comment. I'm just trying to push this SLR roadmap, so I'm going to take one more opportunity. It's a very good document in that it contains the information that we are specifically communicating with the public. It's all kind of wrapped up into that tool, but if you read through that handbook, you'll see a lot of information in there. It's very good information, and we are using that to communicate the benefits of SLR to the public. So, I just want to emphasize it again. Thank you.

MEMBER SKILLMAN: Thank you. Thank all of you.

MS. BERNHOFT: Okay, thank you. I just have a couple of more quick overheads for the introduction. We're having good dialogue, so I appreciate the questions.

So, one of the two focuses when we formed the LTL program; one is that the utilities need to make the decision internally, is this good for their

1 business? And we know that there are lots -- local
2 economics that are driving some possible shutdowns, so
3 some of the questions that we ask, you know, from the
4 technical standpoint, you know, the lifetime of the
5 cables, even the lifetime of the digital equipment,
6 things like that.
7 A lot of the research we're doing to help
8 support them to make that type of a risk decision that
9 it's a good investment to continue for 60 to 80 years
10 of operation. And then the second part of it is, is as
11 they make that decision to continue to extend their
12 license, that the technical basis exists to inform the
13 Aging Management Programs that we're talking about. And
14 it's also risk management and, you know, safety
15 performance and cost become the criteria.
16 And when we set up the program, we wanted
17 to have deliverables to support these two goals in the
18 2014 to 2019 time frame, and we're on track in supporting
19 this deliverables.
20 Very quickly, these are some of the major
21 -- the three major categories that we support under the
22 LTL Program, the Aging Management research. I already
23 talked about how we do prioritization. This is the
24 highest priority area is the Aging Management research.
25 But I think Dick might have touched on this, as well,

is that if the plants are going to continue to operate, there are other considerations, and you need to look at your overall I&C system, enhancements in safety and risk analysis tools. We're not going to talk about any of those today. And we also look at what we call some enabling technologies. And that last bullet is, we are going to support the plants with demonstration projects or what is needed, and these will be specifically focused over the next few years to support the lead plants.

With that, I'll turn over to Rich.

MR. REISTER: Good morning. My name is Richard Reister. I'm the Program Manager for the Light Water Reactor Sustainability Program at the Department of Energy. I'm going to provide a very brief overview of the program to provide some context, so this afternoon as you hear the more technical information you'll understand how it fits into the program.

The goal of the LWRS Program is to provide a sound scientific foundation for long-term operation, so we develop technologies related to the long-term operation, again, to improve the reliability and sustain the safety of the plants as they continue to operate.

The program is divided into four major

areas. The first area that you're going to be hearing most about today is the materials aging and degradation area, and that's -- funding-wise that's about half the program goes toward materials aging. But I also want to mention the other aspects of the program that you really won't hear today, but we work in advanced instrumentation and controls area. It's kind of essentially moving from the analog to the digital technologies in order to be able to continue operating these plants as those analog systems age.

Looking at risk-informed safety margin characterization area, and this is really using advanced tools and simulation methods to get a better handle on safety margin. Again, as both plants age, but also as changes to the plants occur to understand safety margin both for the operators and regulator. And then, finally, we have an area called reactor safety technology that's really focused on the Fukushima Lessons Learned, which is primarily looking at some of the severe accident areas.

MEMBER SKILLMAN: Richard, what are the attributes of this risk-informed safety management or safety margin characterization? Is this going to be a group of people comparing plant safety margins? Is this going to be some kind of hybrid PRA activity? What is

1 this activity going to be?

2 MR. REISTER: It's using simulation tools,

3 so RELAP-7 is one of the primary tools, but we also have

4 a code we call RAVEN, which is a simulation-based tool.

5 It's really kind of trying to marry the deterministic

6 and probabilistic methods together, so you essentially,

7 for an issue that you might want to address, you run a

8 simulation of the plant many times, and a simulation is

9 not -- you don't have margin built into the simulation.

10 You try to simulate what actually occurs in the plant.

11 But that simulation is -- and you run it many times with

12 that simulation driven by the probabilities of what's

13 going to occur, so that's the probabilistic aspect of

14 it.

15 And what you get out of that is a

16 distribution of likely outcomes, basically. And you

17 look at those outcomes in terms of success criteria, you

18 know, did you -- are you going to meet your peak cladding

19 temperature if you wanted to use that as a success

20 criteria? And then you can -- as you make changes, or

21 if the plant ages, or you make a change to the plant,

22 you can understand how that margin, those outcomes might

23 change. So, how does that probability distribution of

24 outcomes change? And that's really the -- what we're

25 looking at.

1 MEMBER SKILLMAN: I certainly understand
2 the concept, but what I'm kind of reaching for in my mind
3 is, what is the model that you use with the consideration
4 that that model is representative of 99 plants?
5 MR. REISTER: Oh. We're not looking at
6 trying to model all of the plants in a comprehensive
7 model. We're --
8 MEMBER SKILLMAN: You're in Model 1.
9 MR. REISTER: -- looking at individual --
10 yes.
11 MEMBER SKILLMAN: Which one? And why is that
12 one okay?
13 MR. REISTER: Well, we're not going to run
14 -- the idea is that we're not going to be the ones
15 running that model. We're going to -- we're providing
16 the technology, the capability for running these
17 models, and provide that to the industry so that they
18 can run the models on their own plants.
19 MEMBER SKILLMAN: I see.
20 MR. REISTER: So, we're really about
21 developing the capability and the technology, not
22 actually doing the analysis. Although, we do work -- the
23 idea is we will work with pilot plant projects with
24 industry to demonstrate, you know, here's how it can be
25 done to transfer the technology.

1 MEMBER SKILLMAN: Okay, thank you.

2 MEMBER RAY: You talked today about some
3 things you're going to talk about today, and some things
4 you won't talk about today. One of the things that I
5 wonder about, is the basic assumption that any changes
6 in the, I'll call it the external environment related
7 to the plant, external threats or whatever, are kept up
8 to date continuously on an ongoing basis and, therefore,
9 there's no additional consideration of those hazards
10 associated with SLR?

11 MR. REISTER: Well, we have been involved in
12 some of the analysis looking at seismic and flooding
13 threats, and part of the safety margin characterization
14 pathway is looking at better techniques for doing the
15 analysis of those --

16 MEMBER RAY: Will that be addressed at a
17 different time, or how is that factored into this SLR
18 question? It was one of the comments we made in the past,
19 and I --

20 MR. REISTER: It involves long-term
21 operation of the plants, but it's really not something
22 that's addressed related to the second license renewal
23 application.

24 MEMBER RAY: Well, that's what I was asking.
25 Your assumption is, or the assumption that's being made,

I believe, would be that this is kept up to date and not part of SLR, specifically.

MR. REISTER: Yes.

MEMBER RAY: Correct?

MR. REISTER: Right.

MS. BERNHOFT: Yes, that is correct. You conform with your current licensing basis and you keep that up to date.

MEMBER RAY: And so there's no reason because we're talking about beyond 60 years to look at that particularly the way you're looking at instrumentation, or corrosion, or fluence, or whatnot.

MR. REISTER: Yes.

MEMBER RAY: I just want to be clear that that's not going to come up at any point along the way.

MR. REISTER: Right. And that's probably a good example of what I'm trying to explain, that there's work we're doing that certainly influences the long-term operation of plants, but it's not part of second license renewal, and that's the subject of this meeting.

MEMBER RAY: Okay. Well, we'll ask the Staff when they come up what their view is on it.

MR. REISTER: Okay. So, this is just a summary of the federal government role of Department of

1 Energy. We certainly believe that these plants, as Jerud
2 mentioned, are in the national interest to continue to
3 operate, so we're addressing the fundamental scientific
4 questions related to long-term operation.
5 Primarily, that's through using our
6 National Laboratories that have this large theoretical
7 and expertise in these areas. I'd also mention that the
8 benefits in terms of materials, understanding also
9 benefit other technologies that DOE is interested, like
10 Advanced Reactors. And the bottom line is, we feel the
11 federal program by having this fundamental scientific
12 understanding, we reduce the uncertainty and risk
13 related to long-term operation. And that provides
14 industry with the comfort that they can make the
15 investments that they need in terms of maintaining the
16 plants to operate them as long as it can be done safely
17 and reliably. Next slide.
18 Just briefly mention that our -- we have a
19 detailed program plan, as well as a joint program plan
20 with EPRI. EPRI has its own program plan, but we have
21 a document that really puts those two plans together to
22 make sure that we're well coordinated. So, in each of
23 the technical areas, it describes how those -- our
24 work and the work of EPRI fits together.
25 In addition, each of those four pathways

1 that we mentioned, materials, I&C, safety margin, and
2 reactor safety technology, they also have detailed
3 program plans. All these program plans are publicly
4 available on our website as shown here, so we try to be
5 as open as we can about what work we're doing, and we're
6 certainly open to feedback, if people feel we're missing
7 an area, or our priorities are different.
8 MEMBER RAY: Thank you.
9 MR. REISTER: Just briefly in the materials
10 area, which you'll hear mostly about, these are the
11 technical areas we're working in, and you'll be hearing
12 a lot more about this this afternoon. Next slide.
13 So, again, trying to understand what the
14 DOE role is, for each of those materials areas we look
15 at trying to provide high quality data in terms of
16 measuring the actual degradation of the material. And
17 then we look at that data to get an understanding of the
18 underlying mechanisms. So, we're trying to move from
19 perhaps an empirical, more of an empirical
20 understanding to an actual physical or mechanistic
21 understanding of degradation of the materials. And we
22 use that understanding to develop models of
23 degradation, and hopefully be able to incorporate those
24 models into a simulation type model of degradation so
25 we can project out into the future, you know, longer term

1 how that material will degrade.

2 And then we also want to look at monitoring,

3 being able to monitor that degradation to both

4 understand how it degrades, detect any degradation that

5 we might not be aware of if something different is

6 occurring, and also to validate our models of

7 degradation.

8 And, finally, we're looking at mitigation

9 strategies, how can we mitigate degradation? Certainly,

10 you can replace but there might be other ways to reduce

11 the rate of degradation through different temperatures,

12 or fluences, or whatever. So, that's -- essentially,

13 for each of those material areas, these are the things

14 we're looking at. We don't do everything on this list

15 for every material, or every mode of degradation, but

16 we try to prioritize what is needed most, what gap needs

17 to be filled for each of these areas.

18 MEMBER SKILLMAN: Richard, could you

19 describe what other industries you reach into for

20 information?

21 MR. REISTER: Well, I'd say certainly in the

22 concrete and the cables area, we've looked at what

23 information is out there in industry. Certainly,

24 there's a lot of concrete out there, for example, you

25 know, roads, bridges, dams, so we're trying to

1 understand -- collect as much information as we can from
2 those industries and not duplicate any work. So,
3 certainly, we look -- you know, that's an example.
4 In the cables area, there's a lot of cables
5 in use, like in the oil and gas industry they have cables
6 that they use for like their offshore rigs and such that
7 are, again, subjected to a lot of harsh environments,
8 but not exactly the same as the nuclear environment. So,
9 a lot of our -- we try to collect that information from
10 those industries that we can. And our research is really
11 focused on what's unique to the nuclear environment that
12 we can't get from those industries? So, like the, you
13 know, cables in a radiation environment would be an
14 example of you're not going to get that in other
15 industries.
16 Same thing with concrete. That's why we
17 focus on the radiation effects of concrete, not because
18 it's necessarily the most important degradation mode,
19 but that's what's unique at a nuclear power plant. The
20 other modes of degradation we can get from other
21 industries.
22 MEMBER SKILLMAN: Okay, thank you.
23 MR. REISTER: And that's all.
24 MS. BERNHOFT: Okay. So, to summarize, Rich
25 talked about kind of the fundamental aspects. I won't

go into detail. I think I used this slide in my last presentation when I talked to your group, Dick. But this shows now how the research ties or maps into the Aging Management Programs.

And in summary, we've talked about this before, but there is a technical basis established and it's what's being used right now for 40, 60 years, so fundamentals are there. The researches you'll hear about from these next few presentations is we do the research for some continuous improvement areas, but fundamentally we've got a good fundamental understanding of the materials aging and degradation. It's all based on the success of the coordination and collaborations. And Jerud said this also, but a different way to say it is, what we've identified from the research so far for a well maintained plant is we're not seeing the show stoppers.

MEMBER BLEY: Let me follow-up on that one. You have no show stoppers, but are there some significant questions that still remain? And where is that focusing on what you do next?

MS. BERNHOFT: I will not say there's significant questions, no. I mean, what we can do --

MEMBER BLEY: We got this pinned down.

MS. BERNHOFT: Yes.

1 MEMBER BLEY: No problems.

2 MS. BERNHOFT: But what I would ask, you
3 know, we're going to be going through some detailed
4 presentations on the metals, the cables, and the
5 concrete, and then you have an opportunity to ask me that
6 question again at the end of the day.

7 MEMBER BLEY: I'm sure we'll ask the people
8 presenting. Thanks.

9 MS. BERNHOFT: Okay. Going to bring up your
10 first round of experts.

11 MEMBER SKILLMAN: Okay.

12 MR. DYLE: Good morning.

13 MEMBER SKILLMAN: Good morning.

14 MR. DYLE: My name is Robin Dyle. I'm from
15 EPRI, and I'm here along with Dr. Keith Leonard from Oak
16 Ridge National Lab. We'll talk to you about some of the
17 metals research.

18 I've been keeping track of the time. I have
19 a lot of information that I plan to have in the package
20 but not speak to. It's there for background information
21 to help flesh out what we'll speak to, so I'll go through
22 this rather quickly as I talk about the reactor coolant
23 system research that we're doing at EPRI, and would
24 welcome questions as we go along.

25 So, the first thing I'd like to talk about

is the programs we have at EPRI, and kind of tell you where we are, and then explain why we're going where we go with the research, so you kind of understand the baseline program.

I've listed here the MRP Program, which manages the materials reliability project activities for PWRs except for the steam generator. Then there is a steam generator management program that focuses solely on steam generator management issues. Then we have the BWR VIP, which is a BWR program that manages reactor coolant system, reactor vessel, and reactor internals. So, that's the scope of those three programs, and they develop the aging management tools for the activities.

Also list WRTC which is our welding program. I've got a couple of slides on that, and Dr. Leonard does, too. And then primary systems corrosion research is our fundamental group that really looks at the mechanistic ways of degradation and things of that nature, so we've got a few slides in the back of the presentation for that, as well. And then the other programs would include NDE, water chemistry, and things of that nature, so all of that goes into supporting the development of the materials programs.

And we do more than just testing from a

research perspective. We have programs in place that allows us to assess the plants, to do inspections to understand how the aging is occurring. We have groups focusing on mitigation and testing, things such as hydrogen water chemistry. They can mitigate stress corrosion cracking in a BWR environment.

The NDE Group helps develop improved inspection techniques. Over the years we've gone from rather rudimentary volumetric exams to some rather robust phased array type examinations now, which again better helps us understand how the plants perform, and the way things work.

Just for your reference, and this is a takeaway here, the GALL AMPs that are covered by these three major programs that I'll talk to you about, so that's there for your reference. We have an integrated plan that we use to try to get through things. And when we put this together, we tried to figure out how can we identify the problems that we have, assess the condition of where the plants are so that we can do the inspections and evaluation. And, again, how can we mitigate the problems to forestall the issues? And then we also needed a plan in place that should we find something, how do you do the repairs or replacements? So, all that's built in. And we came up with an approach that we call

1 the Degradation Matrix and the Issue Management Tables,
2 and I'll talk about those real quickly. I believe Sherry
3 mentioned those when she was here last year.
4 But I will say, when we started down this
5 path in the area around Davis-Besse, we were a little
6 bit reactive trying to catch up with things. When we did
7 a self-assessment in 2010, we had got all the base
8 programs put in place and we were able to truly be
9 proactive and look forward, which made it a timely point
10 to look at LTO and address that in our degradation
11 matrix. So, the timing was good, so we've been truly
12 really proactive since 2010.
13 When you look at the approach, there's two
14 documents listed here, and I've got the reference
15 numbers. These are available to the public. They can be
16 downloaded, they can be looked at. The degradation
17 matrix, the idea there is identify every material in the
18 primary loop and every degradation mechanism, and what
19 our knowledge state. The parallel document is one that
20 Brian mentioned earlier in his presentation, the NRC's
21 EMDA. They're very much alike as far as what they do.
22 But then we then take that in what we call
23 the Issue Management Tables, and at a sub-component
24 level evaluate that degradation mechanism for the
25 components and determine what we know, what the safety

1 implications are, and then based on that develop two
2 things, our aging management program, and our research
3 plan going forward. So, we really use that tool to help
4 guide our thinking as we go forward.
5 An example of what the degradation matrix-
6 MEMBER BLEY: Is --
7 MR. DYLE: Yes?
8 MEMBER BLEY: Is there -- in your whole
9 presentation, is there any difference between what
10 you're telling us as it applies to let's say GALL 2, and
11 what was developed for it, and what is perhaps
12 forward-looking to the GALL for subsequent license
13 renewal?
14 MR. DYLE: There are technical things that
15 we're doing for subsequent or second license renewal to
16 enhance what we know, and I'll be talking about that,
17 but this is the base program that we put together to
18 satisfy the GALL criteria, even prior to the GALL.
19 MEMBER BLEY: Okay, thank you.
20 MR. DYLE: Okay. This is just an example of
21 the degradation matrix, and I'm not going to go through
22 it, but you can see across the top, the degradation
23 modes. You can see across the lefthand column the
24 materials that are involved. This is a hyperlink
25 document. You can go in and search, and look at the

1 different aspects. And then within the matrix, it
2 identifies whether we do have a problem with yes, or N
3 for no. And then it's color coded based on what our
4 knowledge state is, and then you'll notice in some of
5 the squares there's a flag that says LTO. Those have
6 specific LTO aspects related to them.
7 The next slide is just an example of the
8 sub-component level for the IMT. So, you can see there
9 for the reactor vessel we identify, in this case it's
10 the BWR feedwater nozzle safe ends. The materials are
11 included, the degradation mechanisms, those both come
12 from the MDM. And then we have the consequences of
13 failure, the different tools that we know about
14 mitigation, inspection, repair. And then the last
15 column is the gaps that we identify, and those are the
16 needs, whether they're better inspection tools, more
17 frequent inspections, or research. And those gaps are
18 what we use to prioritize where we go forward, and what
19 we need to do. And that's what we're using to guide what
20 we're doing now in SLR.
21 The BWR VIP program, just real quick,
22 living program. It focuses on the BWRs. It's been in
23 place since 1994. There's 32 different AMPs that
24 reference BWR VIP documents in the current GALL that are
25 used as aging management tools. This is a list of the

1 guidelines, and I think this was presented at the last
2 presentation, but it has all the components that are
3 safety-related. It has the inspection and flaw
4 evaluation guidance, mitigation, repair, replacement
5 criteria. You'll note that some of the documents have
6 a dash-A. Those are NRC-reviewed and approved
7 documents. You'll see some that are Rev. 2, Rev. 3,
8 several of these documents are under revision right now
9 based on research and they'll need to be updated. And
10 they can address what we need to do for SLR.
11 We're working with the lead plants to
12 understand what their needs might be so that we can
13 update and prioritize the documents as an appropriate
14 tool for SLR.
15 Similarly, there's the MRP. They take care
16 of the PWR materials issues. The primary document they
17 have is on the next slide, is MRP 227, but I will say
18 the work that was done by this group is also related to
19 PWSCC. There are two code cases that are currently
20 endorsed and required in the regulations. It's based on
21 the work that this group did. One of them has to do with
22 the PWR head inspections, and the other was piping
23 inspections for PWSCC. And those are code cases N-729
24 and N-770.
25 As far as MRP-227 goes, it's the internals

1 guidance document, and 227-A is currently being used.
2 Revision 1 was just published this month, and it
3 addresses NRC comments and licensee action items that
4 the NRC had identified. It closes those, and we're
5 already starting work on Rev. 2 of the document which
6 will specifically focus on SLR needs that are unique,
7 that are not covered in the current document.
8 We'll give you a status of where we are with
9 the PWR internals inspections. We started those in
10 implementing 227, and as of December 2014, 15 reactors
11 have done inspections, and the observations to date,
12 virtually no stress corrosion cracking. The IASCC has
13 been very limited, it's been limited to bolts, and that
14 may be due to the irradiation stress relaxation that
15 occurs at the same time that you're seeing the damage
16 occur. We haven't had any measurable void swelling, and
17 the only failure for the bolts is these high-strength
18 nickel alloy clevis bolting; however, they weren't in
19 the safety path.
20 I will mention that there were problems
21 identified with the guide cards for the rod insertion,
22 and the industry identified that and wrote interim
23 guidance, and issued that to the industry to use, and
24 that has now been incorporated in Rev. 1. So, that's an
25 example of how when the industry finds a problem, we then

1 try to deal with it immediately, and then put it in an
2 updated document at a later point.
3 MEMBER SKILLMAN: Robin, when you say no
4 real findings through all of this inspection, why should
5 we take any comfort from that? What were the states of
6 irradiation for those plants that were inspected, such
7 that we should conclude this is really good news?
8 MR. DYLE: That's a good question. Part of
9 it was the inspections are identified conservatively to
10 occur at different fluence thresholds where you might
11 anticipate the onset of some of the problems, so you
12 inspect early so that if the problem does occur, you have
13 the time to deal with it. So, for those thresholds we
14 have identified, we haven't seen the problem to date.
15 I'm not going to tell you it's not going to occur because
16 we anticipate it, and that's what the science would tell
17 us, but to date the program seems to be robust and
18 conservative that we're looking before the problem
19 occurs.
20 MEMBER SKILLMAN: Are these 15 reactors
21 moldy oldies, or are these generally new plants?
22 MR. DYLE: Yes, they are required to
23 implement 227 as they come into the renewal period, so
24 the first 15 reactors would have been the first 15 in
25 the sequence. Now, there are some things they do before

1 they go to the renewal period, but those are generally
2 the older plants.
3 MEMBER SKILLMAN: Okay. Thank you, Robin.
4 MEMBER BALLINGER: I have a question. When
5 you project out to 80 years, there's an uncertainty band
6 on all of these things. Do you get overlap between that
7 uncertainty band and expected behavior, or initiation
8 of some of these mechanisms like swelling, for example?
9 Would you expect swelling in 80 years? Is it within the
10 uncertainty band of what we now know?
11 MR. DYLE: Do I expect? Not, necessarily.
12 MEMBER BALLINGER: The royal you.
13 MR. DYLE: The -- okay. It is anticipated
14 and accounted for, because you can't assume it won't
15 occur, so the inspection program was to look for it
16 before it becomes problematic. And the other is
17 uncertainty in when it would occur, and that's the
18 reason we're continuing to do more research.
19 MEMBER BALLINGER: But is it based on an
20 assumption that it will occur, or expected intersection
21 of the uncertainty band of the behavior and what you see
22 --
23 MR. DYLE: Based on the assumption that it
24 will occur, and a defined threshold has been
25 conservative to look for it early.

1 MEMBER BALLINGER: Thank you.

2 MEMBER RAY: Robin, I always say, you know,
3 prediction is difficult, especially of the future. You
4 know, what's -- could you brief us on sort of the living
5 aspects of this program, how you're continuing to
6 monitor and add things as we discover new mechanisms --

7 MR. DYLE: Okay.

8 MEMBER RAY: -- and new phenomena?

9 MR. DYLE: And this would apply to BWRs and
10 PWRs both, but the inspections that are conducted are
11 conducted to satisfy the program. We do this underneath
12 the materials initiative so we have INPO verifying that
13 all utilities do everything that the documents require.
14 They then have to report the inspections results back
15 to the programs. The programs then compile that
16 information to determine whether the guidance is still
17 adequate, too conservative, not conversation enough,
18 and then make adjustments. The guide card wear is an
19 example. We started seeing the operational experience
20 that say we're getting more wear than was anticipated,
21 put together a program for inspections to manage that,
22 and people started doing inspections. Then when MRP-227
23 Rev. 1 was produced, that information was included in
24 there.

25 We do the same thing on the BWRs. Every six

1 months, the BWRs report the inspection results from the
2 outage one year previous. We compile that in a document
3 and provide it to the Staff, so every six months we're
4 updating NRC on what those inspections are. Then we
5 trend the data, look at the data, we compare the
6 inspection results, for example, to our ability to
7 perform volumetric exams. And I'll talk later in the
8 Crack Growth document how we've measured our model
9 against field experience and laboratory data.
10 MEMBER SCHULTZ: Robin, using the guide card
11 wear issue as an example, what does the guidance say for
12 these plants in terms of next time inspection? What is
13 the period for the next time inspection for these
14 plants?
15 MR. DYLE: I don't remember the specifics.
16 I'd have to pull the document out and look, but it
17 identified which plants are more susceptible, and
18 identified what they needed to do. I just don't have the
19 recall on what that was.
20 MEMBER SCHULTZ: Okay. And for these plants
21 that are coming up into their license renewal period,
22 then they're looking at the next 20 years, and we're
23 trying to make a case for 20 years subsequent. So, why
24 should we feel comfortable about the result that says
25 no excessive bolting failure that was recognized? What

is the program saying about what is going to be the, again, future inspections related to these aspects for these plants that are going to be the first into the subsequent license?

MR. DYLE: Well, the current guidance would have anticipated -- again, it was written conservatively to go look -- would have potentially anticipated more cracking, so we're seeing less than the document would have anticipated first. The second thing is now that we get the operational experience, you look at the guidance, and then expand out for another 20 years for additional fluence, reduction in toughness, and say how do we need to adjust those inspections based on what the science anticipates, and what the field experiences are telling us? And that's what will be in the next version of MRP-227.

MEMBER SCHULTZ: Maybe I'm overreacting to the phrase "no excessive bolting failure."

MR. DYLE: Right. And, you know, as the plants age you would expect some due to stress corrosion cracking because of the fluence and the water chemistry. And there's also a fatigue aspect, but it's been very limited, less than what we tried to account for.

MEMBER SCHULTZ: So, that's a better phrase.

MR. DYLE: Yes.

1 MEMBER SCHULTZ: Limited failure.
2 MR. DYLE: Limited failure.
3 MEMBER SCHULTZ: Thank you.
4 MR. DYLE: Less than what we would
5 anticipate.
6 MEMBER SCHULTZ: Thank you.
7 MR. DYLE: The Steam Generator Management
8 Program is the oldest program. We started in the '70s
9 and '80s, and it does focus on steam generator issues.
10 Beginning in the 1990s, they were able to be more
11 proactive and look at other things looking out into the
12 future, and we've got good performance from the 690
13 tubing. This is the AMP for the GALL for the Steam
14 Generator Program, and for time's sake, I'll go ahead
15 and go through that.
16 I guess the current status of where we are
17 today with the materials programs, we did a review of
18 the aging management programs back in 2013. We looked
19 at a total of 50, we identified eight aging management
20 programs that we felt like we needed to do some
21 additional work on, and I believe Sherry presented those
22 at the last meeting. And work is underway to enhance
23 those AMPs for longer operation, but the current
24 standing and the current belief in the industry is that
25 with the existing aging management programs we have in

1 place, the plants would be safe from now on. I wouldn't
2 tell you plants would be able to necessarily operate for
3 80 years, because there would be limits they would bump
4 up against and have to stop. But the first thing we've
5 done with these aging management programs is guarantee
6 that the plants stay safe, and then we look and try to
7 figure out how to enhance and increase operability, so
8 we're making sure safety is there with the existing
9 aging management programs. Now, the research that we're
10 going to do will try to expand our knowledge to give us
11 more margin and better operability, and improve safety
12 for 60 to 80 years.
13 MEMBER SCHULTZ: Robin, you used a term that
14 we don't hear very often around here, guarantee safety.
15 Was that slip of the tongue?
16 MR. DYLE: Well, I probably shouldn't say
17 guarantee because there are no guarantees in life,
18 particularly when it comes to materials. But to the best
19 we can do it, we've got programs in place that will give
20 us high confidence that the plants will be safe.
21 MEMBER SCHULTZ: I understood that that's
22 what you were trying to say, but guarantee is a very
23 strong word.
24 MR. DYLE: It is. There are no guarantees in
25 life.

1 MEMBER SCHULTZ: Thank you.

2 MR. DYLE: You're correct.

3 Okay, for time sake, so Keith gets to talk
4 to you, I'm going to go through some of these research
5 programs at a very quick pace, very high level. I know
6 one of the first things you're interested in is the
7 reactor pressure vessel. And we have programs in place
8 to look at the vessel, there's a coordinated
9 surveillance program, a supplemental surveillance, and
10 then at the bottom of the page I have some documents
11 listed that describe those.

12 Here's the five areas that we're looking at
13 in the future. There's the BWR integrated surveillance
14 program, the two I just mentioned. We're doing some
15 atomic probe tomography work on irradiated samples, and
16 I've got a table that shows that. We're working with DOE
17 and Oak Ridge on the ATR program, and Keith will discuss
18 that. We've recently started an effort to develop a
19 revision to a code case that would allow for better use
20 and expanded use of the master curve technology to
21 monitor vessel integrity.

22 Embrittlement trend correlations are used
23 to predict the temperature transition shifts in the
24 vessels, and we need to take care of that. There's a lack
25 of data, so to support the 60 to 80 years we've come up

with a couple of programs that will add to our database so we can better understand what's going on. And the two programs are listed here. I'll talk about each one of them.

As far as the coordinated surveillance program, the idea was how can we get as much data as possible that would exceed three times ten to the 19th neutrons per centimeter squared by 2025? So, what we did was we worked with different plants and had them extend the period of time they would operate before they would pull the capsule and test the surveillance specimen so we could maximize the data we would get at higher fluence levels to help us have an understanding how the plants were performing.

We originally started with 13 capsules. We've lost two of the capsules due to early shutdowns, but the other 11 are still in place, and we plan to get that information between now and 2025 which helps us greatly. And the program is in MRP-326.

MEMBER SKILLMAN: Robin, before you proceed, will the integrated fluence on at least one of the capsules provide basis for an application in 2019? In other words, will the application fluence --

MR. DYLE: Yes. And I've got a table, a chart that shows that shortly.

1 MEMBER SKILLMAN: Thank you. Okay.

2 MR. DYLE: The supplemental surveillance
3 program is one that we designed where we've actually
4 gone out and collected samples from Westinghouse that
5 had previously been tested, reconstituted surveillance
6 specimens, reinserted them in capsules and we're
7 putting the two capsules in Shearon Harris and Farley
8 1 in 2016, and we're going to irradiate those capsules
9 for an additional 10 years to get additional fluence at
10 much higher levels than we would have been able to get
11 otherwise. The nice thing is, is these have been
12 irradiated in existing PWRs so we're adding to the
13 database, and the materials were selected because they
14 really filled a gap in the embrittlement trend
15 correlation when you look at the chemistry. So, there
16 was thought that was put into the selection process.

17 I think this answers your question as far
18 as the fluence goes. If you look on the lefthand side
19 of this plot you can see the brown circles that are
20 enclosed. That's the data that's been collected so far.
21 If you look at the light blue squares, that's the 40-year
22 fluence. The diamonds that are open is the 60-year
23 fluence, and the diamond -- the triangles are the
24 projected 80-year fluence. So, you can see where the
25 fluences fall out over time, and along the bottom it

shows the years.

The supplemental surveillance program is represented by the green triangles that are at year 2025. So, you can see the vertical line for the different data points that we will get at the different fluence levels that will help fill in that gap of what's going on. And you can see that those fluences exceed what the 80-year fluence will be for some of the PWRs. The BWRs are much lower fluences, as well.

And then the diamonds that are filled in are from the coordinated surveillance program, so again you can see some of those data are up at the levels that would exceed or equal the 80-year fluences for the plants. So, that's what we hope to get out of the program.

MEMBER BALLINGER: Robin, are these acceptable fluences tied to specific reactor vessels?

MR. DYLE: I'm not sure I understand.

MEMBER BALLINGER: All right. Every vessel has an allowable NVT.

MR. DYLE: Right.

MEMBER BALLINGER: Okay.

MR. DYLE: And what these will be used, is these data will be used to further enhance and develop and expand the embrittlement trend correlation. Now, the samples did come from operating plants, so when the

1 material is tested 10 years from now, those plants will
2 have to take that data and then determine what that means
3 for their plant, as well, because it did come from a
4 plant.
5 MEMBER BALLINGER: But the reinserted
6 material will go into Shearon Harrison and Farley --
7 MR. DYLE: Right.
8 MEMBER BALLINGER: -- at a particular
9 location. Those reinserted materials didn't come from
10 Shearon Harris on Farley. They came from another --
11 MR. DYLE: Came from other plants, and they
12 were selected for material --
13 MEMBER BALLINGER: And those capsules were
14 in a certain location and expected a certain fluence
15 rate.
16 MR. DYLE: Right.
17 MEMBER BALLINGER: How comparable are they?
18 MR. DYLE: The flux is not that much
19 different. It was deemed as they developed the program
20 not to be an issue.
21 MEMBER BALLINGER: Temperature?
22 MR. DYLE: Temperature should be the same
23 because there's two different temperatures and they
24 tried to match those.
25 MEMBER BALLINGER: Okay.

1 MEMBER RAY: There's a chemistry element of
2 all this --
3 MR. DYLE: Yes.
4 MEMBER RAY: -- that doesn't appear on
5 this. Right?
6 MR. DYLE: Right. And that's the reason the
7 specific materials that were selected were selected for
8 the chemistry that they had, they filled gaps in the data
9 that we currently didn't have.
10 MEMBER RAY: Thank you.
11 MEMBER BALLINGER: Are the reinserted
12 -- keep going around in circles. Are the reinserted
13 capsules from reactors that might also submit SLRs?
14 MR. DYLE: I believe so. I'd have to go back
15 and look at the list of where they came from.
16 MEMBER BALLINGER: So, why not -- if not,
17 why not put them in those reactors?
18 MR. DYLE: Well, the thing was to find two
19 host plants that were willing to do this and made the
20 commitment, so you get the two hosts. That's critical.
21 MEMBER BALLINGER: Some volunteers.
22 MR. DYLE: That's not an inexpensive
23 proposition to reinstall capsules to become a host
24 reactor for a research activity.
25 Okay. What we end up with, and this is a real

1 quick chart. The first column shows what we would have
2 had prior to starting these programs. We would have had
3 50 total data points above three with the 29, with 14
4 above six, and two above eight. You look at the
5 right-hand column, we'll now have 83 data points above
6 three to the 19, and we'll end up with 16 above eight,
7 and eight above 90 to the 19th which previously would
8 have had no data, so that expands our database. And this
9 is an open program. We're sharing this with the Staff,
10 and sharing the data, as well. For time sake --
11 MEMBER RICCARDELLA: If I go back to the
12 previous chart, you were going much higher than 90 to
13 the 19th. Right? There's some data --
14 MR. DYLE: Right. 1E to the -- that shows 1.2
15 E to the 20th, but we at least know it's above 9.
16 MEMBER RICCARDELLA: Oh, I see, it's above.
17 Okay.
18 MR. DYLE: Above 9.
19 MEMBER RICCARDELLA: I understand.
20 MR. DYLE: For time's sake, I'm going to skip
21 through this real quick, because Keith will talk about
22 the ATR Experiment. I'll just tell you that some of the
23 samples that we've got that are in the PSSP are also in
24 the ATR Experiment, so we're getting additional testing
25 done there. And we're also doing work in taking these

1 materials and using them in a CRIEPI Program where we're
2 doing atom probe tomography to look at the structural
3 changes of what's going on. So, this table here is a
4 takeaway for you. You can look at the heat numbers. It
5 shows which one of those heats are in the ATR-2
6 Irradiation Experiment in the fourth column, the sixth
7 column shows the surveillance fluence expected, whether
8 they're in the PSSP, and then the CRIEPI-EPRI column,
9 the next to the full right-hand side are the plants that
10 are the samples that we'll do atom probe tomography on.
11 MEMBER BROWN: How do those reflect the
12 fluence levels you think you have to get to? I'm trying
13 -- I'm an electrical guy, so I'm trying to connect the
14 dots on these expected -- are these expected fluence
15 levels that they would expect to see after 80 years, or
16 --
17 MR. DYLE: Some are expected, and some
18 exceed that, so we have some conservatisms --
19 MEMBER BROWN: So, your data is going to
20 exceed the expected levels. Okay.
21 MR. DYLE: I've tried to bound what we would
22 expect the behavior to be.
23 MEMBER BROWN: Okay.
24 MR. DYLE: And by testing all these
25 materials and all these different programs we're

1 getting a good matrix of multiple ways to look at the
2 materials and how they're behaving.
3 And then the last item is, the BWRs have an
4 integrated surveillance program that is going to be
5 updated for second license renewal. So, that's it on
6 reactor vessels. And I've got a whole lot more slides,
7 and we don't have time to go through those.
8 The next thing I do want to talk about is
9 the crack growth model that we've developed. We took
10 data from all over the world, we used expert panels to
11 look at the data. I will tell you that we started with
12 1,600 data segments from six labs, 37 different heats,
13 800 of those segments or more were quasi-static loading
14 so they gave us a chance of having good data. The experts
15 ranked them from 1 to 5. We kept everything from 1 to
16 3 to use in the model. This is what the final data set
17 looked like, For the low ECP environment that were
18 ranked greater than 3 we ended up with 187 data points
19 that we deemed acceptable from 22 heats. And then for
20 the normal water chemistry which would represent the
21 unmitigated BWR water chemistry we had 282 crack growth
22 observations.
23 This is what the data scatter looked like.
24 You can see why we skipped everything that was above 3.
25 MEMBER RICCARDELLA: That's a log scale.

1 MR. DYLE: Yes, that's a log scale. It was
2 huge scatter. This is what the model looks like. And
3 then, I guess, the thing that we can take away from this
4 is one thing we determined was there was really not a
5 difference between the crack growth rates in the bore
6 60 in the normal light water reactor environment looking
7 at the data. We also determined that the data for the
8 low ECP environment could be used for BWRs and PWRs, as
9 well, if you adjusted for temperature. So, that means
10 we only have to have two models to model crack growth
11 rate.
12 The data was representative out to 47 dpa,
13 so it far exceeds anything a BWR would ever see, and gets
14 to most places that a PWR would see as far as the
15 irradiation itself. And so this is a good model that gets
16 us headed down the path for longer operation.
17 The other thing I will say from this slide,
18 there's a good data fit. We also did a comparison of the
19 crack growth rates that this model generates with the
20 existing laboratory data and field data, and it was
21 conservative even for the recent measured field data
22 that we got on some shrouds that have seen significant
23 irradiation and significant cracking. So, we believe
24 it's a good, sound model and we're now trying to work
25 through as the last bullet here says, with ASME to get

1 these crack growth rates endorsed in the ASME code as
2 opposed to trying to go through some other review and
3 approval process. And that way they'd be available for
4 use internationally, as well as domestically, and can
5 be endorsed by the Staff, if they so choose, when they
6 update the regulations. So, that's a real quick run
7 through on the crack growth model.
8 MEMBER RICCARDELLA: Is there cast
9 materials in that --
10 MR. DYLE: No. It was all just stainless,
11 wrought material, no cast. We've got a separate program
12 looking at cast.
13 Irradiation materials testing, Sherry
14 mentioned roadmaps. Here's the roadmap that can show you
15 what we're doing in this area, and it's broken down
16 between the BWR VIP, Primary System Corrosion reserves
17 and the Materials Reliability Program. I'll leave that
18 for you to look at. And then I'm going to go through these
19 real quick, so Keith can speak. But we have -- the
20 program was Zorita. The Staff will also talk about it
21 today because it's a joint project between the industry,
22 EPRI, and the NRC on testing material that was taken out
23 of the Zorita plant. And we took some baffle plates, this
24 is just a picture of the material. We've got a joint
25 program.

1 The box on the bottom were testing of the
2 Zorita weld and heat effected zone materials which we
3 have limited data on. So, that's important. We've got
4 a lot of data on wrought material but less on welding
5 -- on welds and heat affected zone material. We're doing
6 fracture toughness and crack growth rate testing both.
7 We're also looking at IASCC crack growth rates and
8 initiation with some of the material from there. We also
9 even hope to take some of the baffle plates and do
10 additional irradiation on them, if we can find a
11 suitable place to do that so we can extend the data. And
12 these crack growth rates will be added to the model, so
13 it's additional data that we'll use to update that crack
14 growth model that I just previously discussed.
15 We're also looking at thermal aging
16 analysis of the stainless steel weld materials. Again,
17 we've got less material and less data on welds than we
18 do other things, so that program starts next year trying
19 to look at the combined effects of thermal aging and
20 irradiation on cast and welded stainless metals, and do
21 that work.
22 And then we're going to do additional crack
23 growth tests at Halden both for PWRs and BWRs, but this
24 is a PWR chart. And then we are participating in the
25 Gondole Void Swelling Program, as well, to look at the

1 void swelling. The early results show little swelling
2 that's greater than .5 percent, and actually some of the
3 samples have showed negative swelling. I'm not sure how
4 that works out, but at least that's what the data shows.
5 We're also looking at the effect of lithium
6 on IASCC initiation, and we have seen some correlation
7 to lithium rates and lithium concentrations in the
8 initiation data, continuing to track that and do a
9 little more work in that area. And also looking at the
10 dynamic strain effects on IASCC initiation rates, as
11 opposed to static loads looking at front end loads and
12 see what effects they would have on crack initiation.
13 MEMBER SKILLMAN: Robin, what is that image
14 in the upper left, please?
15 MR. DYLE: That's the test room for how they
16 had the samples included in there.
17 MEMBER SKILLMAN: Okay, thank you.
18 MEMBER BROWN: Can you go back to that one?
19 It says that the lithium enhances initiation rate. Does
20 that mean it gets worse, or it gets better?
21 MR. DYLE: Gets worse.
22 MEMBER BROWN: Oh, okay. So, I read the words
23 correctly.
24 MR. DYLE: Yes.
25 MEMBER BROWN: So that lithium is bad.

1 MR. DYLE: Well, it has an effect. The
2 question is how much, and how significant, so you can
3 monitor that and know what the impact is. That's the
4 reason we're doing the work. When you start looking at
5 longer operation and trying to figure out why things
6 occur, you start pulling the thread on everything, and
7 that's one of them.
8 MEMBER BROWN: Thank you.
9 MR. DYLE: And I've already discussed that.
10 Real quick, the BWR testing from Zorita materials that
11 took welds from these locations, and again that's what
12 they look like, and there's a weld with 2 dpa, and then
13 there was a baffle plate at 10 dpa. And we're doing
14 fracture toughness testing, the material will be
15 characterized, they'll do fracture toughness testing to
16 update our database for that. And then there will be
17 crack growth rate testing on the heat affected zone and
18 weld models, as well. So, again, trying to collect more
19 data, enhance the models, and make them more aware. And
20 this is the test protocol that we'll have for testing
21 those materials.
22 One example I'll show you real quick is BWR
23 shroud boat sample. And this is just an example where
24 the industry on its own decided to better understand
25 what was going on. We have what we call off-axis

1 tracking, the classic IGSCC and IASCC is normally in the
2 heat affected zone parallel to welds, and we're seeing
3 cracking now that goes perpendicular to the welds. The
4 question is, you know, what could that be? What is it?
5 Could it be IASCC or something else?
6 This is the science project. This is what
7 a BWR shroud weld looks like. You can see in the middle
8 of it there's a 1-inch layer, the width of that weld is
9 1-inch, that gives you some idea of magnification. But
10 you can see three vertical cracks that go across the weld
11 metal, so we looked at that and said that's not normal.
12 What should we do? So, we decided to take boat samples.
13 The plant volunteered to do it, spent several million
14 dollars extracting the boat sample and doing the safety
15 evaluation on their own, and now EPRI is doing the
16 testing of that.
17 I just -- I have these construction photos
18 in here just to give you something to look at later.
19 Besides IASCC, there is the potential that these are
20 locations that significant cold work from construction.
21 The picture shows alignment lugs where the shroud shells
22 were aligned, and then spiders that were installed on
23 the inside that kept the structure intact while it was
24 being shipped. In all cases before the plant operated,
25 those had to be ground off, so every one of those

1 locations has the high likelihood of significant cold
2 work, which would be a source of crack initiation.
3 MEMBER RICCARDELLA: These cracks are all
4 located in areas where the --
5 MR. DYLE: We don't know that. That's what
6 we're trying to determine. And this is just a schematic
7 of how they actually did the alignment where they
8 actually hammered wedges in place to get the things
9 aligned, and then welded them intact. So, there's lots
10 of abuse that went on in the construction of these.
11 This is what the boat sample looked like
12 just for your information. It's 3-inches by 2-inches,
13 it did capture the flaw that we looked at, so you've got
14 the cracked tip both in the axial direction and the depth
15 direction so you can look at it. And we're evaluating
16 that now, again try determine whether it's IASCC or some
17 other mechanism. And there's specific objectives. And
18 then just a picture for you to look at of here's the
19 specimens that are being taken out of that sample, both
20 tensile specimens, fracture toughness specimens, and
21 all so we can really understand what that material tells
22 us. And this was in an area that was about 2E to the 21st
23 neutrons per centimeters squared as far as fluence goes.
24 So, we found a location that was high fluence and had
25 this cracking. We've seen that cracking in other places

as early as 1993 in lower fluence areas, so it's not necessarily tied to fluence.

MEMBER SKILLMAN: Robin, would you back up a slide, another one, one more.

MR. DYLE: Okay. One more?

MEMBER SKILLMAN: One more. One more.

MR. DYLE: Okay.

MEMBER SKILLMAN: That one. Has there ever been a -- this is really a winging it question. I spent time in the shops for the vessels for the steam generators, for the reactor coolant pumps. It always impresses me that there's a leather mallet, maybe a leather mallet that's bigger than a smaller leather mallet where a craftsman can impart an awful lot of force to make a big piece of metal do something. Is there any data of fabrication process induced failures? I mean, we have this body of knowledge where we think we really understand these machines, but I know of no action that was ever taken to determine whether or not hammering a steam generator tube into its tube hole or pounding two wedges to achieve concentric alignment has even been considered. And this is a perfect example where you would use a wedge, and a dog, and a hammer to create a millimeter worth of alignment on a 12-foot diameter piece of machinery.

1 MR. DYLE: And in 14 or 15 places around the
2 diameter then welded in place.
3 MEMBER SKILLMAN: Yes.
4 MR. DYLE: I don't know of any database that
5 exists. I do know some of the analyses we've done in the
6 past where you look at cracks in laboratory situations,
7 you can see evidence of significant cold work which
8 would come from that manufacturing process, but I don't
9 know that there's ever been an effort to really try to
10 collect and correlate that type data from a
11 manufacturing perspective.
12 I do know early days of the BWR program, we
13 actually took two shrouds from cancelled plants and cut
14 them up and looked at the stresses that were imparted
15 and how they relieved themselves, and all of that, and
16 how that might play into crack growth rates and things
17 of that nature, so you could see significant stresses
18 that were there from construction, not really looking
19 at the cold work and the use that occurs in the shop.
20 We do have criteria that says, for example,
21 in the BWR, if you know your jet pump has -- where they
22 use draw beams to pull the jet pump in alignment to focus
23 on and look at those draw beam locations for the
24 potential for cracking. So, we've tried to account for
25 that where we were aware of it.

1 MEMBER SKILLMAN: Thank you.

2 MR. DYLE: We're running out of time, and I
3 want Keith to be able to talk. I just -- there are some
4 slides in here. The Staff asked me to put some
5 information on on X-750 and XM-19, so there is a test
6 that we're doing currently where we've sectioned X-750
7 material. GE Global Research did it, Martin Moore did
8 the characterization for us, and we've now got that in
9 the ATR reactor, and we're going to do fracture
10 toughness and crack growth rates on those materials
11 which are important because they're used as repair
12 materials. And these are the conclusions so far that
13 we've seen that you get a variation in crack growth
14 almost by a factor of 2 depending on micro structure and
15 orientation. But the fracture toughness is actually
16 higher than what we anticipated, and that's just the
17 early results. That will probably finish in mid-2017,
18 but that's the work that's underway there.

19 Primary System Corrosion Research, I just
20 say here, here's a list of the 2015 projects based on
21 topics, irradiated material stress corrosion cracking,
22 environmentally assisted fatigue, and for time's sake
23 I'll go on. Here's the model they use where they try to
24 focus on looking at irradiation induced micro
25 structures and the impact it would have at a mechanistic

level on yield strength and fracture toughness, void swelling. So, that's the way the program is constructed to look at the deep science there to try to do that. It's more like the work that Keith is doing at Oak Ridge Labs. And those products will be developed.

One that I've highlighted here, we're trying to use irradiation -- simulate irradiation damage using heavy ion irradiation. If we can use that and it reasonably simulates what we see from neutron damage, then we have a different way to generate additional data to do testing. So, we're spending some money to see if that's a plausible way to do that kind of work to help us in the future. And I'd be glad to answer questions later this afternoon on the details of that.

Here's a model of how the program works. You start with material at zero, 50, and 75 dpa and then we're going to irradiate it with ion irradiation, and then look at the difference in the micro structures from the ion irradiation versus the neutron irradiation and see how they compare, and if we can build some kind of model that would simulate that behavior.

And here's corrosion research deliverables. Also, the ARRM program is a program where we're working with DOE to try to develop

1 radiation-resistant materials for use and repair
2 materials in new plants. And I'll just skip that for the
3 time being. It's there for your review.
4 And then welding on irradiated material,
5 here's a roadmap. I'm going to leave it to Keith to talk
6 about that. He's got some slides on this, but the main
7 thing we tried to figure out is there's place in a
8 reactor that you can't weld, that may need to be
9 repaired, and it wasn't readily apparent how you would
10 do a mechanical fix. So, how can we do welding on
11 irradiated stainless knowing the problems that we have?
12 So, we're developing a couple of options, such as
13 multiple laser welding, or friction stir welding, and
14 there's pictures of that. And I'll leave it to Keith to
15 discuss that.
16 And then finally, the other areas I don't
17 even have slides on that we're working on to support LTO
18 is environmentally assisted fatigue. Alloy 690 PWSCC
19 that we're working with the NRC on an expert panel
20 looking at information. And piening is a mitigation tool
21 to protect the vessel heads, and then also the BWR
22 chemistry program. So, that's all I've got. I'm sorry,
23 Keith. Any questions real quick?
24 MR. LEONARD: Okay, thank you very much.
25 Okay. Let me introduce myself. I'm Keith Leonard. I'm

1 from the Oak Ridge National Laboratory. I'm the new
2 pathway lead for the materials aging and degradation
3 part of the LWRS program. I'm taking over for Jeremy
4 Busby who's now one of the Division Directors, or the
5 Division Director for the Material Science and
6 Technology Division at Oak Ridge. So, our roles have
7 kind of changed here a little bit, but -- and this has
8 been actually within the past couple of months.
9 So, as Rich kind of described, DOE LWRS
10 areas of research are shown, and we really kind of focus
11 on the five Ms that Rich talked, measurement,
12 mechanisms, modeling, monitoring, and mitigation. So,
13 we're going to touch briefly upon some of the research
14 work that's being conducted by DOE through the LWRS
15 program at the various laboratories, universities that
16 are working with us.
17 Okay. So, the first area to talk about is
18 high fluence effects on RPV steels. So, there's three
19 areas of work that's being done. The first is looking
20 at high fluence properties, the mechanical performance
21 of the materials. And we're also developing multi-scale
22 modeling of the phased development and properties of
23 these materials.
24 Another area of work is development of new
25 test methodology. This includes small fracture

1 toughness specimens that you can see in the lower left.
2 This is utilizing what they call mini compact tension
3 specimens machined from broken shroud piece specimens,
4 so this is increasing the amount of material that we have
5 capable of testing, and tests have been verified with
6 CRIEPI on these ORNL conducted materials.
7 We're also looking at small angle neutron
8 scattering or sands, and we've had some recent
9 comparison between sands between National Institute of
10 Standards and ORNL data, and this is data that's looking
11 at the -- what they sometimes term as light lumen plate
12 phases or copper rich, or manganese-nickel-silicon
13 precipitates in RPV steel, so average size that the
14 sands determine for an aged material for 1,100 hours at
15 350 degrees C was 1.36 nanometers. So, this allows us
16 another tool to better determine what's happening
17 within the micro structure.
18 So, as Robin had discussed, the ATR 2
19 Experiment, so understanding the mechanical properties
20 in plant life extension is really a critical component.
21 And high fluence data is really need as reactor
22 surveillance data becomes sparse above 3 times 10 to the
23 19th neutrons per centimeter squared. So, current RPV
24 embrittlement models can under-predict some of the
25 transition temperature shift data, and that's shown in

1 the upper right corner between the predicted versus
2 measured. And this is -- the predicted is based on the
3 IONI model, which is not fully incorporating some of the
4 late blooming phases into the design.
5 So, the work that's being done in the ATR
6 2 Experiment is really to do some high fluence
7 irradiations on a number of different RPV steel alloys,
8 so there's 180 different RPV steel alloys, 1,600
9 specimens that are being tested. The material is now at
10 Oak Ridge, came in a few months ago, and some of the
11 testing is underway.
12 Another area is modeling support in RPV
13 steels, so this is kind of multi-part area. So, the
14 formation of copper rich precipitates and
15 manganese-nickel-silica precipitates is one of the main
16 reasons that -- one of the leading causes for
17 embrittlement of reactor pressure vessel steels,
18 although it's been observed in experiments that even in
19 copper free or relatively copper free alloy
20 compositions that you can have manganese-nickel-silica
21 precipitation. So, this is the developing a cluster
22 dynamics model to calculate the evolution of these
23 manganese-nickel-silica precipitates. And then the
24 upper right, this is just an example of some of the data.
25 This is growth of volume fraction versus fluence for a

1 particular alloy composition. And this is kind of
2 assuming various preexisting nuclide densities. This
3 was established from parameters for evaluation of
4 copper rich precipitates, but also takes into account
5 other conditions in which no preexisting or homogenous
6 nucleation can occur. So, the next phase is really
7 incorporating into -- integrating both the copper rich
8 precipitate development, as well as the other model with
9 the manganese-nickel-silica precipitates.
10 MEMBER SKILLMAN: Keith, in the upper right
11 hand image there is neutrons per square meter. Is that
12 accurate?
13 MR. LEONARD: I believe so. That was -- I can
14 double check that, but that was off of one of the
15 milestone reports from the University of Wisconsin.
16 MEMBER SKILLMAN: Okay, thank you.
17 MR. LEONARD: Okay. So, cast austenitic
18 stainless steels is another area, and a large amount of
19 cast materials are used in LWRR coolant systems.
20 Relatively few critical degradation modes of concern
21 are expected within the current design lifetime, but
22 what we want do is address some of the longer term aging
23 issues, identifying any kind of concerns that may come
24 up with extended lifetime addressing unknowns within
25 the EMDA evaluations. So, we're looking at a number of

1 different models, CF-3, CF-3M, CF-8, CF-8M materials
2 that are static cast, as well as EPRI has provided
3 centrifugal cast alloys, bendage alloys, also compared
4 them to wrought 304 and 316L.
5 So, our purpose here is to be able to
6 develop an understanding within a limited evaluation
7 time, so we want to be able to perform and get some
8 results back within a shorter period of time, five
9 years. So, we're doing accelerated thermal aging
10 research. So, kind of the schematic on the right shows
11 kind of the pathway in which we're doing this work. So,
12 we have -- at the start of the work, we've gone back and
13 looked through existing data collection and performed
14 analysis. We've done computational studies to identify
15 phased present at the end of life, and that's shown in
16 the upper left plot. And each one of those in the X axis
17 is a particular alloy composition and an aging
18 temperature. And you can see if you take the first four,
19 you have aging temperature of 290, 330, 360, and 400
20 degrees C, and you can see that there's some differences
21 in the type of precipitates that show up. We've also
22 looked at non-aged baseline material properties, and
23 that's shown in the bottom lower left corner.
24 So, currently, we're performing at Pacific
25 Northwest National Laboratory aging materials studies.

1 The first side of materials have come out and that's the
2 1,500 hour material specimen, and I'll show you an
3 example of that. So, how do we determine temperatures
4 for accelerated aging? That goes back to looking at the
5 Arrhenius equation and plotting out aging time versus
6 aging temperature. So, we want to be able to perform
7 these tests within a 5-year program, and so your plot
8 on the left is service for 60 years at 300 degrees C.
9 And the box that's shown is your aging temperature range
10 for a 5-year experiment based on activation energies for
11 diffusion control processes leading to precipitation.
12 So, for aging temperatures that we're
13 considering, we're looking at 330 degrees C to simulate
14 300 degree C aging times, 360 degrees C aging
15 temperatures to simulate 330 degrees temperatures.
16 We're also doing aging at 290 just to help establish a
17 baseline also, which were not -- you know, which was not
18 accelerated aging. But we're also duplicating some
19 previous work by Argonne National Lab that had aged at
20 400 degrees C to make sure that our results compare with
21 them, whether that is an over-aging condition based on
22 what we see from the ERANEUS diagram.
23 So, following is just some preliminary
24 1,500 hour aging impact data for a couple of the model
25 alloys. This is Model CF-3 alloy that has 15 percent

ferrite, which shows some changes in the upper shelf energy and transition temperature with aging at various temperatures between 290 and 400 degrees C. And in the lower left is the Model CF-8 alloy which has 5.6 percent ferrite initially, and the influence of aging. So, the aging effects appear significant from the SHARPI impact data; aging effects on tensile properties are less significant compared to the as-received material for both alloys.

So, modeling of environmentally assisted fatigue is another area that we're looking at. Environmentally assisted fatigue has been identified as a possible knowledge gap for long term operation. To date, the fatigue assessment is largely empirical based on experimental data for a particular material load environment. It also encompasses an issue that sometimes safety factors are used in assessing the structural integrity, and those safety factors may be overly conservative.

So, in this work that's being conducted by Argonne National Lab through the LWRS program, we're modeling EAF mechanisms through mechanistic-based approach to support experimental studies. So, eventually, we want to get to a model that will provide capability to extrapolate severity of the mode of

degradation under realistic reactor environmental loading cycles.

So, to do this the plan is to conduct tensile and fatigue experimental tests under air and LWR environments, look at the cyclic plasticity based on evolutionary material modeling and parameters, and that can be plugged into a finite element model. So, what you're seeing on this slide is just some of the experimental data that's being generated, and looking at how the material hardens or softens as a function of accumulated plastic strain over time. So, some of these parameters that are time-dependent and material-dependent are going into these finite element models.

So, what you're seeing here is kind of the initial start of taking what's known experimentally and putting it into these FE models. And this is a simplified example of just a tensile specimen with precision pole rods developed. We also have finite element models that are being developed for other system-level components, but this is just kind of a stepping stone in incorporating the data. And we have good correlation between the model and experimental data. So, by 2018 we hope to have a model for environmentally assisted fatigue in specific LWR components that can be utilized.

1 Crack initiation in nickel-based alloys,
2 this is work that's being done by Steve Bruemmer and
3 Mychailo Toloczko at Pacific Northwest National
4 Laboratory. The key here is identifying the mechanisms
5 and precursor state to predict the causes for crack
6 initiation leading to more focused inspections,
7 informed regulatory guidelines, and aging management
8 programs. So, the next few slides are just kind of
9 examples of some of the work.
10 The image that you see on the right is an
11 example of a constant load test sample and this is one
12 of the assemblies that goes into a test chamber to
13 simulate LWR reactor environmental conditions. In the
14 lower left is some examples when -- for Alloy 600
15 testing. Alloys are generally tested for a period of
16 time, taken out, inspected, and looking at the cracks
17 that are generated. So, this was crack morphology and
18 crack stress intensity for stress corrosion crack
19 initiation, so this is for stress intensity which is the
20 K value there is determined from finite element modeling
21 for cracks initiated on these tensile specimens, so we
22 see a difference between a more severe effect from the
23 cold work where you have a shorter length crack, but a
24 deeper length crack, and that can create a stress
25 intensity factor that's a little bit higher at the

1 surface than in the middle of any of the material.
2 So, work at PNNL continues on with current
3 work that's now within its first year of evaluating long
4 term response of cold work Alloy 690 materials, and 690
5 is a replacement alloy for Alloy 600. So, this is a
6 multi-year test with the early surface exams similar to
7 what's been done. Examinations have reached one year of
8 exposure without really establishing a SEC initiation.
9 Now, those were constant load tests, but
10 what they're seeing in some new tests, not SEC
11 initiation tests on cold work 690 reveal that internal
12 grain boundary creep cavitation which is shown there,
13 those are grain boundaries which you see IG creep
14 cavitation. And this is an issue that's showing up in
15 dynamic straining in which the level or load is
16 increased during the testing. So, irradiation --
17 MEMBER SKILLMAN: Keith, where would that
18 phenomenon be occurring in an operating reactor? What
19 is the location?
20 MR. LEONARD: Probably anywhere you have a
21 significant stress concentration or concentrator, any
22 kind of what you would say, like a keyed material, or
23 a -- what would be a good example? Anywhere where you
24 have like a corner cut, and it's -- the issue is really
25 to evaluate while we're not seeing the under static

load, you know, issues, issues do show up. And I believe they're showing up in dynamic straining, not just from the blunt notch, but it's also a possibility if you have changes in loading conditions within just, you know, a non-notched material, that that may be an issue, as well.

MEMBER SKILLMAN: I was just trying to imagine a location where that phenomenon would be occurring.

MR. LEONARD: Okay. That's a good question.

MEMBER BALLINGER: To expand on that a little bit, where is 690 used other than in the steam generators and in weld like 52 and 152 weld metal. Where is 690 used other than those places?

MR. DYLE: This is Robin Dyle. The new PWR heads have 690 penetrations in them instead of the 600 penetrations.

MEMBER BALLINGER: Oh, okay. But they're using 52 or 152 weld. Right? So, that's where the stresses would be high. Yes.

MEMBER RICCARDELLA: Well, what do you mean by dynamic loading? Is that fatigue loading?

MR. LEONARD: No, it's not fatigue loading. And I have to check in the duration of -- but after a period of time they step up the amount of loading on the

1 material or the train of samples that are contained
2 within the --
3 MEMBER RICCARDELLA: So, they hold and then
4 increase the load, hold and then increase, and then
5 hold.
6 MR. LEONARD: Okay. So, this is work on
7 looking at mechanisms of irradiation assisted stress
8 corrosion cracking. This is work that's being done with
9 the University of Michigan, as well as some people at
10 Oak Ridge. And the kind of checked areas are some of the
11 work that's been done looking at specifically
12 mechanistics of the -- of some of the effects that are
13 happening in the material relative to different
14 materials, whether it's the composition and other
15 factors. I'll show you some examples.
16 So, one of the things that has been
17 developed is a four-point bend test that was developed
18 to study IASCC initiation, and this allows them to
19 perform this work on high irradiated materials. You have
20 less volume that you have to deal with from a safety
21 perspective. The bend tests on the same alloys and test
22 conditions were performed as previous continuous
23 extension rate tensile tests, and revealed very good
24 correlation between the two. The bend test allows for
25 initiation of the crack formation, but without going to

1 taking the material to failure.

2 Other tools that are being utilized shown

3 on the right is information that was relatively new.

4 This was taken off of FEI Talos Transmission Electron

5 Microscope. This allows for tremendously improved

6 compositional mapping capabilities and resolutions

7 looking at the chemistry nature associated with the

8 crack in the material.

9 So, understanding mechanisms by SEC, this

10 is, again, the University of Michigan with some work at

11 Oak Ridge. Four point bend tests reveal stress threshold

12 similar to what the continuous extension rate tests were

13 seeing, so we have good agreement. U of M has looked at

14 alloy purity, they've done some tests looking at

15 commercial versus high purity 304 and the role of

16 inclusions that you see up there on the right. They've

17 also looked at alloy chemistry looking at higher nickel

18 concentration alloys, as well as grain boundary

19 orientation. So, micro chemistry, grain boundaries,

20 grain boundary orientation relative to applied stress,

21 they've seen that in higher nickel alloys. You have

22 better slip transmission across the grain boundaries,

23 and that leads to a little bit more resistance to IASCC.

24 MEMBER SKILLMAN: Keith, let me ask this. Is

25 this a good time, or will there be a good time very soon

1 to take a break? I'm thinking of two things; we've got
2 -- you've got some good material remaining. We've got
3 people in the room that may need a bio break.
4 MR. LEONARD: Okay.
5 MEMBER SKILLMAN: We have the NRC Staff
6 coming up with all of their preparation, and I don't want
7 to short change any of our participants.
8 MR. LEONARD: It's really -- I have a little
9 bit on modeling, a little bit on the mitigation with the
10 welding technology, and then a few slides on Zion
11 harvesting.
12 MEMBER SKILLMAN: Can we take 10 minutes
13 right now?
14 MR. LEONARD: Yes, sure.
15 MEMBER SKILLMAN: Ladies and gentlemen,
16 let's take a 10-minute recess, it's not 15, 10 minutes,
17 return please at 15 minutes to 11 on that clock. We are
18 recessed.
19 (Whereupon, the proceedings in the
20 above-entitled matter went off the record at 10:33 a.m.
21 and resumed at 10:45 a.m.)
22 MEMBER SKILLMAN: Let the record show we
23 have a quorum. We are back in session. Keith, please
24 proceed. Ten minutes.
25 MR. LEONARD: Thank you. Once again, my

1 name is Keith Leonard. So to kind of finish up the talk
2 here on DOEL/WRS sponsored research, we have a number
3 of modeling task that are complimentary to some of the
4 experimental work, and this is really developed to
5 incorporate, you know, projecting out further time
6 lines, to where we have kind of a lack of experimental
7 data and eventually get that and compare that with
8 reactor, experimental reactor data.
9 So this is a phase prediction of
10 development, looking at different effects. We have
11 precipitation models that are looking at various
12 steels, looking at what phases are going to be stable
13 or not as stable, and/or kinetically inhibited.
14 modeling of radiation-induced changes. We have mild
15 predictions that look at radiation through segregation
16 models.
17 So enrichments and grain boundaries,
18 radiation-induced precipitation models, prediction of
19 gamma prime within the material, as far as its
20 formation. That's shown there on the lower right, and
21 we also have some high fluence swelling models that are
22 being developed, to look at material swelling.
23 Robin had talked briefly about some of the
24 mitigation areas, and so this is the advanced welding
25 R&D that's being developed. This is a joint project

with EPRI, and it's really to demonstrate how we can utilize advanced welding techniques on irradiated materials for cost-effective repair and replacement.

So you're probably familiar with the underlying mechanism. You have helium and a little bit of hydrogen generation in the materials when they're heated up, say through a welding process, that those gases are going to go to grain boundaries that are going to cause areas where you can have cracking.

So friction stir welding appears to be a possible options to stress voids and cracks due to the solid state low welding temperature. So this is an example of the welding hot cell that's being developed. This is currently at EPRI at their Charlotte facility, and this is going to be shipping to Oak Ridge very soon.

Within the hot cell, this is actually -- what you're seeing there is actually a liner. It's not the actual cell that actually goes into the hot cell, and within it you have a laser subsystem for laser welding, as well as a friction stir welding subsystem. Kind of the closer examples of some of the tests that were conducted with the friction stir welding block, as well as the laser weld test block.

So all of this is designed to be able to be utilized in the hot cell environment, utilizing

1 manipulators. It's also kind of a complicated process
2 that has to be kind of evaluated. Another area of work
3 is finite element, thermal and mechanical modeling and
4 experimental validation.
5 So this is looking at kind of hybrid
6 technology, in which you're using the laser to do
7 post-weld thermal treatments. This is to basically
8 reduce the amount of internal stress within the
9 material, and avoid future issues.
10 To kind of conclude the talk this morning,
11 we have a little bit of work that's on harvested
12 materials. So one area that we're looking at is the
13 decommissioning of the Zion Unit 1 and 2 as an
14 opportunity to examine service-aged materials
15 degradation.
16 I'll skip ahead. This is an example.
17 This is Zion Unit 1 RPV Harvesting Plan. There is a
18 couple of sections that we're interested, and Section
19 1 that you see there in the lower right contains a
20 beltline weld and the nominal compositions are shown and
21 high dose region is contained within that section, which
22 peaks at about 7 times 10 to the 18th neutrons per
23 centimeter squared.
24 We have Section 8 or Segment 8, which
25 contains the beltline vertical axis weld. So this

allows us to get a lot of information that -- an opportunity to evaluate the welds, as well as the baseline materials, as well as through thickness, while attenuation, chemistry and mechanical properties, microstructural studies and would be a great source of information.

So Dr. Tom Rosseel is going up there I believe it's next week for the harvesting of those sections.

Another area that we've been working with has been with Westinghouse and people from Ginna Exelon, and this is retrieving some baffle former bolts. There's two specific bolts. They're Bolt 22 and 23, and they're kind of highlighted in that table that shows the fluences at the head, mid-shank and mid-thread.

These -- all the bolts that are -- were taken out of the reactor did not show any signs or indications of cracking, but we will investigate them further once we get them into the hot cell. They're going to be sectioned up, so that we can perform tests, mechanical characterization testing as well as microstructural examinations.

The two bolts that were selected were the highest, one of which was the highest fluence bolt. The second one was a fairly high fluence bolt, but it allows

1 kind of an overlap of some of the fluences at different
2 locations, so we can get more information on fluence
3 effects at different locations as there's a fluence
4 profile, a temperature profile and structural profile
5 going through the material.
6 So they'll be -- this is planned for
7 retrieval at the end of February of next year through
8 the summer. They're going to be at Westinghouse for
9 some initial sectioning, and then those test pieces come
10 to Oak Ridge for testing. That concludes this portion
11 of the LWRS talk on Metals Research.
12 MEMBER SKILLMAN: Keith thank you, and
13 Robin thank you. Let me ask my colleagues. Any
14 questions?
15 MEMBER SCHULTZ: Keith, one question. As
16 you described the welding, I guess I'll call it a
17 research program associated with the welding
18 techniques, what is the time scale for that? You
19 mentioned it was an EPRI-related program too. Robin,
20 you may have some perspective on this.
21 But you've got a lot going on there that
22 looks fairly attractive in terms of its possibilities.
23 What's the time schedule for the overall program?
24 MR. LEONARD: So for this next year, it's
25 really going to be the welding cubicle in at the facility

at Oak Ridge, and testing that out, getting all the safety reviews done. During that time, there's going to be a number of side projects that are looking at the friction stir welding on simulated materials, and eventually gearing up for the 2017 time frame or fiscal year 2017 for actually looking and performing test runs on irradiated materials.

Those materials are available. They're basically waiting for the program to start.

MEMBER SCHULTZ: So it sounds like it's two, three, four year horizon?

MR. LEONARD: It's a three year horizon.

MEMBER SCHULTZ: Sure, thank you.

MEMBER RICCARDELLA: Keith, you know, looking at the design of the RPV harvesting plan, those radiation or those end of life dose levels are relatively low, if I look at where they fall on that big chart there.

MR. LEONARD: Yes.

MEMBER RICCARDELLA: Why is that? Is that just a low flux vessel or something?

MR. LEONARD: I believe that was -- with the Section 1, that is the section that contains kind of the highest flow dose area.

MEMBER RICCARDELLA: I see that. But it's

7 times 10 to the 18th.

MR. LEONARD: It still provides us -- we felt that it still provides us a good opportunity to look at attenuation effects, changes to the reactor pressure -- reactor pressure vessel wall, as well as some of the welding. So it's a valuable material, but yes, ideally it doesn't go quite as high in fluences as that we would really like to see.

MEMBER RICCARDELLA: Okay, thank you.

MEMBER SKILLMAN: Colleagues, any other questions?

(No response.)

MEMBER SKILLMAN: Robin and Keith, thank you very much, and we'll turn the presentation over to the NRC staff.

MR. BLOOM: This is Steve Bloom. I just wanted to get us going on the agenda. So we will now have members of the staff, NRR and Research talking about subjects like the fluence, the vessel and the vessel internals, and I won't interrupt anymore.

MEMBER SKILLMAN: Thank you, Steve.

MR. PARKS: Good morning. My name is Benjamin Parks. We're not starting with this presentation. We're starting with the fluence one. I'm going to speak about fluence. In the interest of

1 time, we'll cut it to fast neutrons, so this is quick.
2 This is my colleague, Jay Wallace from the Office of
3 Research. I'm in the Office of Nuclear Reactor
4 Regulation, Division of Safety Systems.
5 I'd like to acknowledge also my colleague,
6 Matthew Hardgrove, who could not be here, also in
7 NRR/DSS. He has done a lot of the coordination with the
8 SLR expert panels and just had some travel commitments
9 that he couldn't rearrange today.
10 Next slide, please Heather. Just a brief
11 overview. I have about a dozen slides. We're going to
12 talk about the current regulatory framework and
13 technical aspects relating to calculating or estimating
14 neutron fluence, the technical issues that we're
15 observing as we are entering or considering subsequent
16 license renewal, what we're doing to resolve those
17 technical issues, the status of that work, schedule and
18 milestone, and in the meantime, what we're doing to
19 address the technical issues, barring the completion of
20 additional research.
21 Slide 3 please. I'm not going to go
22 through all of the regulations, but basically the point
23 is there isn't a regulation that says you need to
24 calculate fluence or a regulation that tells you how to
25 do it. We do know that we need to account for the

1 effects of neutron irradiation damage, and here are some
2 GDCs and the sections of 10 C.F.R. Part 50 that require
3 accounting for neutron irradiation damage,
4 particularly on the reactor vessel.
5 There are also requirements pertaining to
6 reactor vessel material surveillance programs, and
7 those also require estimates of neutron fluence.
8 So what we do is we provide guidance for
9 calculating pressure vessel neutron fluence in
10 Regulatory Guide 1.190, Calculational and Dosimetry
11 Methods for Determining Reactor Pressure Vessel
12 Fluence.
13 That was devised through the 1.990's. We
14 issued it in final form in 2001, and it's been largely
15 useful in calculating or calculating the fluence
16 associated with temperature shifts for reactor vessel
17 irradiation embrittlement, and it's also been very
18 useful in evaluating license renewal applications for
19 the first 20 year extension. Slide 4 please.
20 MEMBER SKILLMAN: Ben, before you move
21 ahead, why should we believe that these regulations
22 really are good for 60 to 80? What I mean by "good" is
23 acceptable, appropriate, technically defensible?
24 MR. PARKS: So these -- I could read your
25 question two different ways. The first way I interpret

1 your question is the regulations just say accounting for
2 the effects of neutron radiation damage, okay, and so
3 as long as you're providing a reliable and believable
4 fluence estimate, you have the tools to go and do the
5 further evaluation.
6 Now when you talk about the efficacy of
7 regulations, why is it good, let's talk about 50.61, the
8 PTS rule, and say there's an absolute fluence threshold
9 that you would reach and above that, you need to take
10 some pretty drastic measures to reduce how much flux you
11 emit from your core, in order to protect your reactor
12 vessel.
13 So what we did a while ago is we promulgated
14 50.61 Alpha, which gives people a different means to
15 provide the assurance that they have the fracture
16 toughness prevention that is necessary for -- well, from
17 40 to 60 years certainly, but then also from 60 to 80
18 years.
19 So let me pause there and say am I answering
20 your question.
21 MEMBER SKILLMAN: Ben, yes you are. What
22 I was really getting to is there are folks in the room
23 I'm sure, including members of the Committee, that think
24 back to the late 50's and 60's and say to ourselves we
25 never would have anticipated we'd be having this

1 discussion. Some of the regulations really have their
2 roots back in those days.
3 Here we are in 2015, talking about life
4 beyond 80, with this body of regulation, that has been
5 kept up to date. But I think one could make the case
6 it's kind of dated. Unless we are able to say even
7 though it was dated, its fundamentals are sound today
8 and how we apply it in the future is fully technically
9 acceptable because.
10 MR. PARKS: Okay, and if I could just
11 comment on that, I mean I speak for the neutrons
12 primarily. That's more my job. A lot of other folks
13 will tell you about how they've evaluated the overall
14 framework or the regulations that exist for license
15 renewal.
16 MEMBER SKILLMAN: I was going to the
17 neutrons, not to the words in the regulation. I'm
18 talking about the technical outcome, such that an
19 assessment can be depended upon.
20 MR. PARKS: Sure, okay. So I guess when
21 we're talking about the neutrons, you know, not having
22 an explicit prescriptive regulation that says this is
23 how you need to do it, we're starting to look more and
24 have more slides to kind of show how we're shifting where
25 we look to make sure that we're providing industry with

1 the tools, so that they can consistently and reliably
2 predict the effects or estimates where the neutrons are
3 going and how to count them, and then from there assess
4 what the effects are.
5 MEMBER SKILLMAN: Fair enough. Thank
6 you, Ben. You've answered the question.
7 MR. PARKS: Okay.
8 MR. MEDOFF: Ben, may I interject for a
9 second?
10 MR. PARKS: Sure Jim Medoff.
11 MR. MEDOFF: This is Jim Medoff of the
12 staff. We're responsible -- there's a
13 two-part -- there are two parts to these types of
14 analyses.
15 The reactor systems staff does the analysis
16 of the neutron fluence projections, but it's the --
17 either the Division of Engineering or Division of
18 License Renewal tech staff that do the calculations that
19 are required by those regulations that are up there on
20 the slide, which is 50.60 or 50.61 for the pressurized
21 thermal shock assessments, or Appendix H for the upper
22 shelf or PT limit assessments.
23 As Ben said, those rules require you to
24 project those analyses and use the -- take in the
25 accounts of neutron irradiation out to the end of the

license life. So for a 40-year program initial licensing, they would use the 40 year fluence.

The way we account for it in license renewal space, whether it be the first license renewal or subsequent is these analyses come in as time-limiting aging analyses.

So if they project them out to the end of the period, they're going to use either the 60 year fluence for an initial license renewal, or we're going to require them to use the 80 year fluences for the subsequent.

What Ben is going to get into is how the conservatisms in the modeling for the projected fluences.

MEMBER SKILLMAN: Thank you.

MR. PARKS: Okay. So the next slide, Slide 4 please, explains the way Reg Guide 1.190 works. It provides the guidance for calculating, and it keeps saying the words "pressure vessel," pressure vessel fluence and I'll share some pictures later.

The guidance basically explains how nuclear data need to be collapsed, so that you can use -- solve a tractable problem with respect to your core neutron source, and then modeling the core neutron source. Then finally to perform transport

1 calculations out to the reactor vessel and beyond if so
2 needed.
3 The guidance, I will point out, is
4 specifically tailored to discrete ordinance transport
5 calculations and Monte Carlo methods. However, we can
6 review and accept other types of methods. It's just
7 that the guidance isn't exactly written for those
8 methods.
9 And then another very important part of Reg
10 Guide 1.190 is qualifying the fluence calculations that
11 are then used for embrittlement analyses, because
12 fluence calculations, by and large, pursuant to this Reg
13 Guide, are intended to be realistic. Provide your best
14 estimate.
15 But that means that you also need to show
16 that your estimate is believable by comparing to some
17 in-plant data, and you also have to estimate your
18 analytic uncertainty and show that you're meeting
19 certain uncertainty requirements, and that your
20 solution agrees with actual physical tangible data.
21 So we have some standard benchmarking
22 experiments that we recommend comparing against some
23 computational benchmark problems, so that we eliminate
24 real world uncertainties and variabilities in a
25 calculation. Then we also require plant and

problem-specific qualification. Usually that's a capsule dosimetry analysis for the type of reactor you're estimating fluence on, and then, as I mentioned, uncertainty requirements.

Okay. So what are the technical issues when you go to subsequent license renewal? Well, here I've shown a blue cloud of my neutrons, and Reg Guide 1.190 is intended for beltline region fluence calculations. When we said beltline in the 1950's or the 1960's, what we really thought is that's the area near the core.

So in this blue cloud you can see we're primarily concerned with the neutrons that are accumulating on the reactor vessel, pretty close to the core, near the active fuel height region of the core.

For an original 40 year license, that coincided reasonably well. However, what we're seeing in going from 40 to 60 years, although we're not identifying any inadequacies right now, is that that beltline region is starting to expand.

I need to point out right here that beltline doesn't mean active fuel height region of the core; it actually means those areas where the vessel surface fluence exceeds 10 to the 17 neutrons per centimeter squared, or is estimated to do so.

1 So what kinds of problems do we identify?
2 Here, this yellow cloud of neutrons shows that as you
3 increase that area, you start to become concerned about
4 the neutron transport that would occur probably let's
5 say up to the vessel nozzle region, and when you
6 propagate neutrons up that far, what happens is it's not
7 just the neutrons that emit from the core that become
8 important because they're traveling such a far
9 distance; you're also looking on my left-hand side arrow
10 here at neutrons that come more directly out of the core
11 through the vessel and then reflect back.
12 So we've primarily been concerned with
13 perhaps surface, vessel inner surface or quarter T
14 neutron fluence estimates, which are then attenuated
15 using a certain formula that's in a different Reg Guide.
16 However, now we're starting -- as we start to broaden
17 where we're considering, we need to consider that the
18 neutrons might behave differently.
19 So in this example, you might estimate at
20 a certain exposure that the vessel outer surface nozzle
21 weld might have a different fluence, and in fact in some
22 cases, it could even be higher than the inner surface
23 fluence, and that's because of what's called a cavity
24 streaming effect that I'm illustrating here.
25 Okay. So on Slide 7 we have a different

1 type of problem, and that is that we're not just looking
2 at reactor vessel embrittlement. We're also
3 considering other damage mechanisms that happen not
4 just to the vessel but to other internals.
5 So two examples of that would be
6 irradiation-assisted stress corrosion cracking and
7 irradiation embrittlement, and in particular where you
8 might be concerned about these is above the core.
9 So let's say an upper core plate or below
10 the core, the lower core plate. Our challenge is here,
11 in Reg Guide 1.190, we've provided guidance and we've
12 focused on simplifying the fluence problems so that you
13 can accurately estimate a vessel fluence.
14 Well that means that you can simplify
15 what's happening in the middle of the core. If you want
16 to estimate what's happening in the middle of the
17 vessel, the geometric center of the vessel, then you
18 probably shouldn't be simplifying what's in the middle
19 of the core as much as you are for calculating the vessel
20 fluence.
21 So what we're starting to see is fluence
22 methods and fluence estimates being applied in broader
23 uses than just on the vessel.
24 So in Slide 8, just to kind of sum this up,
25 we've always focused on this blue cloud, the

1 quote-unquote beltline region of the vessel. So I
2 think in Slide 9 we summarize.
3 What are we seeing? So as we increase the
4 neutronic exposure from 60 to 80 years, we're looking
5 at calculating fluence over greater transport distances
6 from the core. So that's, you know, looking at the
7 right -- you know, if we're going to calculate fluence
8 at a nozzle, what kind of problem should we be solving,
9 and with what fidelity.
10 So we're also considering fluence in other
11 regions besides the core periphery, upper internals in
12 particular are an example there. The other thing is the
13 state of the art has probably evolved beyond discrete
14 ordinance and Monte Carlo.
15 We kind of viewed in the early 2000's Monte
16 Carlo is a very computationally expensive state of the
17 art type method, and in fact some methods that we've
18 approved that are based on discrete ordinance, were
19 compared to MCNP calculations. They were viewed as the
20 gold standard.
21 Well, discrete ordinance has been around
22 for a long time, longer than I've been alive perhaps.
23 So we're going to look at other types of methods. Are
24 there other types of methods to calculate fluence that
25 are more appropriate or more economical, you know, more

precise, more accurate than these two?

Then finally we're going to consider fluence methods for other than just reactor vessel embrittlement. So we're going to look at the appropriate methods to feed those IASCC or IE type problems. That's primarily where you're looking in the core. You're starting to shift to the fluence on internals.

So on Slide 10, what are we doing to resolve these technical issues?

MEMBER BROWN: Can you go back a slide?

MR. PARKS: Sure.

MEMBER BROWN: You say other methods you're going to investigate. I guess I'm a little bit baffled. You sounded questionable, like nobody knows what other methods are --

(Simultaneous speaking.)

MR. PARKS: But to give you an example, we have one vendor who offers a method of characteristics method instead of purely discrete ordinance. But we're also considering whether you might hybridize a calculation and use both discrete ordinance and Monte Carlo transport methods.

MEMBER BROWN: Okay. So it's not like you have to go out and invent something. That's what it

1 sounded like, that there aren't any other methods.
2 (Simultaneous speaking.)
3 MR. PARKS: I didn't mean that all.
4 MEMBER BROWN: There are other methods.
5 It's a fairly simple question that I made complicated
6 and I apologize for that. Thank you.
7 MR. PARKS: Not at all. Okay. So what
8 are we doing? NRR issued a user need request earlier
9 this year, and we're basically looking for a technical
10 basis, either to revise Reg Guide 1.190 or to issue
11 additional guidance. We might issue additional
12 guidance because there's a clear regulatory basis for
13 Reg Guide 1.190.
14 It's based on the regulations that require
15 a very robust reactor vessel. So there might be a
16 different regulatory basis that we identify for
17 calculating fluence on internals, so that might push us
18 into a different regulatory guide.
19 Once we're pretty well into the research,
20 we'll understand what we're going to do. But basically
21 the idea is we want a basis for new guidance, for broader
22 guidance.
23 I've put in this table, it spans two slides,
24 basically what we're asking research to provide us with
25 to that end, and where we're starting, when I say study

1 and determine highest importance phenomena, what I mean
2 is if you are familiar with a phenomena importance in
3 ranking table or a PIRT type approach, I want to make
4 sure that Research and their consultants are focusing
5 on what the most important pieces of a fluence problem
6 are, so that's where they spend their resources
7 gathering the data.
8 So their plan for that, for example, is to
9 do sensitivity studies and see where, if they make a
10 small adjustment, they see a lot of variance in the
11 result or stuff like that, just to make sure that they're
12 focused on the right things.
13 Then determine the best suited types of
14 methods. Is a discrete ordinance transport problem
15 still appropriate, because it's certainly the workhorse
16 of the industry right now? Are there other easily
17 attainable methods beyond the two that were specified.
18 We already know that there are some acceptable ways to
19 do it. So we want to kind of fold those into our
20 guidance.
21 The appropriate calculational rigor. So
22 Reg Guide 1.190 specifies how rigorous your
23 calculations need to be for us to generally believe what
24 the results are or, you know, have confidence in those
25 results.

1 So especially if we're calculating neutron
2 transport over greater mean-free paths from the core,
3 for instance, we might need to tighten up our precision.
4 We might need to put more work into the computation than
5 we say is necessary for the vessel today.
6 So we're going to look into that, and then
7 Heather, the next slide please. The other really
8 important part of Reg Guide 1.190, because we're
9 providing guidance for best estimate calculations and
10 then asking you to quantify your uncertainty and
11 demonstrate your capabilities, we want to look for
12 better benchmarking data than just, you know, a
13 database, a capsule dosimetry, our existing
14 computational benchmarks and the experiments that were
15 performed that formed the basis of Reg Guide 1.190.
16 Things like the Venus 3 benchmark or the Oak
17 Ridge full critical assembly, those types of
18 experiments are the ones that I'm talking about. So we
19 want to see if we could get data that are in different
20 locations of a reactor vessel or perhaps maybe exposed
21 to a different neutron energy spectrum or stuff like
22 that.
23 We want to see what's out there first before
24 we go and create this stuff, because creating such
25 benchmarking data is probably very expensive and cost

1 prohibitive. But there could be things like X-vessel
2 dosimetry or, you know, campaigns of dosimetry that span
3 a lot of height inside a reactor vessel that would be
4 valuable for this. So there might be stuff out there
5 that we just don't quite know about right now.
6 MEMBER SKILLMAN: Ben, let me ask this
7 question please. When I think of a reactor vessel, I
8 think of the device that ships on a barge five to six
9 hundred tons. I don't think about internals. Within
10 the reactor vessel appurtenances, core capture lugs.
11 MR. PARKS: Sure.
12 MEMBER SKILLMAN: Other weld mints that
13 are essential for certain events for the design of that
14 vessel. How are those other appurtenances addressed?
15 MR. PARKS: I can give you an example based
16 on the 40 to 60 year license renewal period, and what
17 EPRI has done with the Materials Reliability Program.
18 They've devised ways to screen all of these different
19 components.
20 They estimate in a very bounding, generic
21 nature what the fluence is for the internals, the
22 appurtenances, etcetera, and then they determine what
23 the susceptibility of those components would be to a
24 certain type of irradiation damage, IASCC or
25 irradiation embrittlement.

1 Once they do that, they decide how are we
2 going to manage the aging of that component? Are we
3 going to inspect it more often? Are we going to replace
4 it? And so based on that, they're not -- they're using
5 fluence as a tool to tell us how concerned do we need
6 to be about this component, you know, the fluence being
7 one of those screening criteria, the other one being how
8 safety significant would a failure of this component be.
9 Then once they do that, they decide how they
10 need to treat it, to make sure that the component, my
11 own words, behaves safely during the license renewal
12 period.
13 So those programs aren't exactly in my area
14 of expertise, but I can say I would anticipate that from
15 60 to 80 years for subsequent license renewal, a similar
16 management program would be used.
17 MEMBER SKILLMAN: Thank you Ben.
18 MEMBER BALLINGER: I have a question. How
19 much of this kind of work is being done to determine the
20 neutron dose and fluence to concrete structures that are
21 much further out? Doesn't that overlap here a little
22 bit?
23 MR. PARKS: You know, I do see a lot of
24 overlap, especially when I was talking about the cavity
25 streaming problem.

1 MEMBER BALLINGER: Yeah.

2 MR. PARKS: Concrete comes into that,
3 because you're transporting neutrons out into the, I
4 guess, the buyer shield, you know, some concrete pieces.
5 So certainly you need to make sure your problem includes
6 concrete materials.

7 I am focused, because of our area of concern
8 on primarily the vessel, but also the internals.
9 However, I could see this work dovetailing reasonably
10 well with efforts to understand better the behavior of
11 concrete.

12 MR. THOMAS: Yeah. If I can respond to
13 that, Brian Thomas. Yeah. The work that staff is
14 doing to look, take a proper look at the calculational
15 approaches and methods for fluence under Reg Guide 1.190
16 is also being viewed with regard to its relevance to
17 concrete.

18 In other venues in the research of
19 concrete, the staff is looking at highly irradiated
20 concrete. However, the calculational methodology that
21 the staff is working on, based on a user needs request,
22 is we are taking a very broad look at it. That's how
23 does it apply to concrete also?

24 MEMBER BALLINGER: Okay, and to follow up
25 on that, I can understand the science, the exciting

1 science that would go with this kind of thing. But as
2 a practical matter, what's the bang for the buck with
3 regard to subsequent license renewal and extending the
4 fluence calculations out to 80 years, beyond what's
5 already available to be done?
6 Maybe I'm not writing it right, wording it
7 right, but why do this if it's -- how is it necessary?
8 MR. PARKS: Okay. So maybe I can answer
9 your question in reverse, Dr. Ballinger. If we go --
10 what's on the next slide, Heather? Okay yes, right
11 here, and let me talk about the problems I'm confronting
12 today and how we had to do this on a case-specific basis,
13 and then I'll explain what the bang for the buck is,
14 okay.
15 So we're already seeing fluence
16 assessments for the 40 to 60 year license renewal
17 period, for other than what we would envision as the
18 traditional active core height beltline of the reactor
19 vessel, and what we're seeing is an industry that's
20 pretty well prepared.
21 They've done a lot of research and they've
22 made some changes and advancements in their methods to
23 prepare themselves to calculate areas in those red
24 clouds and the yellow clouds where I showed you our
25 guidance isn't really intended to be applied.

1 Our challenge as staff is we have to assess
2 those on their merits on a case-by-case basis, because
3 our guidance isn't tailored to it. So the bang for the
4 buck is being able to be a standard and predictable and
5 consistent regulator with regards to how we evaluate
6 those methods when we see them being used.
7 MEMBER BALLINGER: I guess I understand.
8 MEMBER BROWN: I guess my question is
9 somewhat relative to that. You talk about the expanded
10 volume in which the neutron fluence must be evaluated
11 relative to material performance. The licensees have
12 to provide their analysis to you all and you all do your
13 confirmatory analyses. So you do your method stuff.
14 Is all this being coordinated or your
15 thought processes out to the licensee, so they know what
16 to expect? The enhanced expectations, they're going to
17 have to deliver analyses and how do their methods then
18 benchmark against yours, in terms of how they're
19 developed --
20 MR. PARKS: Okay. I think to answer that
21 question, I would give you a couple of examples that
22 we've seen recently, where the broadened use of a
23 fluence method is being applied, and what we're seeing
24 is applicants produce the calculations under the
25 presumption that they're using an NRC-approved method

1 or a Reg Guide 1.190 adherent method.

2 When we look at it we say benchmarking in

3 1.190 is for this area, and your methods as approved are

4 intended to used in that area; they're not intended to

5 be used over here, they're surprised.

6 Now fortunately, okay. So let me put

7 credit where credit is due. When we ask questions and

8 they go through their round of being shocked and

9 surprised, we find out that the vendor that's supplying

10 those calculations as tightened up their precision.

11 They're doing pin by pin modeling in the core, in the

12 central region of the core, and they're providing a

13 better axial representation of the height dependence of

14 the problem.

15 Okay. But still, we're not really being

16 consistent and predictable if by and large, when our

17 applicants produce new calculations like this, they're

18 surprised that we're asking questions. We're running

19 the risk, because we haven't standardized our guidance,

20 of having reviewer to reviewer discrepancies in our

21 review.

22 So it really is our responsibility to

23 provide durable guidance for this type of problem.

24 MEMBER BROWN: Well doesn't the -- I mean

25 you talked about your yellow region, where you've got

1 now reflected irradiation external to the normal
2 beltline or core area you're talking about. I mean are
3 the methods compatible with the reflected? I mean is
4 that different than you do when you're doing the
5 internal calculations?
6 MR. PARKS: So to answer that one, I'm
7 actually going to have to draw on some work that's been
8 published by some folks in the industry, and they
9 studied the problem with varying levels of complexity
10 in their modeling.
11 So what they're saying is they're using Reg
12 Guide 1.190 recommended approaches, and then they're
13 comparing it. If they step up the computational
14 intensity that they apply to the problem, they finally
15 get a different answer.
16 Okay. So if you're familiar with
17 specifying a computational domain based on geometry, to
18 do a grid refinement study or something like that to
19 increase the order of their approximations, so on and
20 so forth, and they find that they're getting a different
21 answer.
22 It's telling us okay, so Reg Guide 1.190 was
23 studied at great depth and we understood that what we
24 specified for the vessel beltline was okay. But when
25 you step away, you need to apply more resources to your

1 problem.

2 MR. DYLE: Ben, if I could, this is Robin

3 Dyle. To go with what Ben's saying, Dr. Ballinger you

4 asked a question about application.

5 The BWRs years ago developed a

6 three-dimensional fluence model that's been reviewed

7 and approved by the staff, and it was necessary because

8 as fluence increases on stainless steel, your crack

9 growth rate and fracture toughness changes.

10 So you need to be able to monitor that. So

11 we saw the need early on to do that. That model was

12 developed and the staff has reviewed and approved it.

13 I think what we're seeing now is with the expanded

14 beltline in some of what the PWRs are doing, the staff's

15 view is they would like to have a more prescriptive

16 definition of how that should be done.

17 And you know, we think the models we've

18 developed are pretty good, because they meet what I

19 would call the performance criteria of Reg Guide 1.190,

20 as far as uncertainty. And I think that's where it's

21 going on.

22 The other thing I would add is the BWR model

23 was benchmarked against four different internals

24 locations, and I showed earlier a picture of the boat

25 sample. One of the things we're going to do there is

1 retrospective dosimetry and then benchmark that
2 location to the fluence model for that plant.
3 So we're constantly trying to do things, to
4 see how good these models are. We're sharing that with
5 the staff.
6 MR. PARKS: Now I need to point out though
7 that -- because that's developed under the auspices of
8 perhaps EPRI-funded work or by a different vendor. If
9 those data were generated or developed by say a
10 different vendor, that vendor could choose to make them
11 proprietary, and the reason that the NRC is going back
12 and looking at this stuff.
13 One thing we're asking Research to do for
14 us and our contractor to do is to see what data are
15 available, because we'd like to use that. We'd like
16 access to it, and we'd like to make it available more
17 broadly to the entire industry if it's not.
18 So all of that helps us standardize, and the
19 other thing I want to clarify is we wouldn't necessary
20 be prescriptive, because we're issuing -- our goal is
21 to issue regulatory guidance, meaning it's not
22 obligatory to follow it. It's just explaining what an
23 acceptable approach is.
24 MEMBER BALLINGER: I just look at the
25 scatter in the data, and then I ask myself the additional

1 modeling effort that's being done, does that collapse
2 this in any way? In other words, when you do the
3 extrapolation or interpolation or however you want to,
4 calling it out to 80 years, you've got this error band
5 which is pretty wide. That's all. I'm just --
6 MEMBER SCHULTZ: Well let me expand that a
7 bit. With regard to the pressure vessel fluence
8 uncertainty, it was stated that and has been applied
9 that plus or minus 20 percent is considered reasonable
10 and acceptable related to those predictions. What is
11 being found as you've gone into these location-specific
12 calculations?
13 MR. PARKS: Okay. That's actually a
14 really good question, because that's my second bullet.
15 We have to consider other things, because the reason
16 plus or minus 20 percent was put into Reg Guide in part
17 -- into Reg Guide 1.190 was in part because there's at
18 least 20 percent margin in the fluence term and the
19 temperature shift calculation, okay.
20 So the Reg Guide's compatible with other
21 regulatory guidance and other requirements. Now when
22 we look at a different location-specific fluence
23 evaluation, we consider whatever the threshold is
24 judged to be for that particular damage mechanism, which
25 is -- it's a little bit downstream of the fluence

1 calculation, but you need to figure out am I over 10 to
2 the 22nd or 10 to the 23rd neutrons per square
3 centimeter, and if I am, then I need to consider these
4 other damage mechanisms.
5 The other thing is, you know, for a
6 mechanism like IASCC, it's been offered to us well we're
7 using this threshold, and that's a really conservative
8 threshold, because it's for a very highly stressed
9 component, and the component that we're seeing that
10 fluence on is not that highly stressed.
11 So it's a bit more qualitative, but we're
12 okay with that because we don't have explicit regulatory
13 requirements to account for radiation damage on vessel
14 internals, the same way that we do on the reactor vessel.
15 MEMBER SCHULTZ: You mentioned that other
16 methods have been applied and we heard the example
17 related to the BWR vessel internals program. With
18 regard to -- how many of these -- how much review has
19 been done of these models to date by the staff? What
20 has already been submitted and reviewed?
21 MR. PARKS: Okay. So I think there's
22 about maybe four or five different Reg Guide 1.190
23 adherent, NRC-approved transported methods, about
24 that many, okay, and those have been reviewed insofar
25 as they have been found to produce vessel fluence

1 calculations within the required accuracy and
2 uncertainty.
3 In terms of what we're seeing that's new,
4 I think that we have probably two or three different
5 applications where we have either, in a plant-specific
6 calculation, we have limited our review just to those
7 locations that are important to us, or if a vendor is
8 requesting a more detailed review, a more thorough
9 review, then the staff, you know, tailors its review
10 efforts accordingly.
11 However, there are three such examples of
12 more broadly applicable fluence calculations. They
13 have received incredibly more detailed review. We
14 usually hire a consultant to help us out with the review,
15 whereas when we're doing a review of a fluence estimate
16 that's using a Reg Guide 1.190 method and it's for the
17 vessel, it's more or less a checklist.
18 We can just pull the Reg Guide and say did
19 you meet this? Yes, yes, yes. Okay. So I can't
20 provide you an hours and hours comparison, but I can tell
21 you that the three recent applications that we've had
22 have received considerably more detailed review.
23 MEMBER SCHULTZ: And in the applications
24 that have been received, you mentioned the approach has
25 been to provide more detailed modeling in the extension

1 to the new application. In each case, have
2 benchmarking information been provided to demonstrate
3 the quality of the improved modeling?
4 MR. PARKS: If benchmarking hasn't
5 specifically been provided, like for example, I've
6 calculated fluence on my core plate, and here's a
7 scraping of the core plate that I have tested the
8 activity, they have made a case by there is other similar
9 benchmarking that's available, or the method is
10 otherwise believable as it's presently qualified.
11 So the answer to your question is no, not
12 always. That's another reason why we're doing this
13 research, to provide the guidance. But if they don't
14 have that, they have to make a justification why we still
15 would believe their number or increase their margins,
16 so that they have more room for error.
17 MEMBER SCHULTZ: Thank you.
18 MR. PARKS: This is my final slide. So if
19 there are no further questions.
20 MEMBER SKILLMAN: Thank you, Ben.
21 MR. PARKS: I thank you all for your time.
22 (Pause.)
23 MR. KIRK: Okay. So my name is Mark Kirk.
24 I'm from the Office of Nuclear Regulatory Research. My
25 colleague Benjamin just filled you in on life from the

1 neutron's perspective. I'd like to give you the
2 counter-perspective, from the viewpoint of the target,
3 the reactor pressure vessel. Yes, go ahead. Can we go
4 full speed?
5 There you go. Okay. So if we can go back
6 to the title slide. I'd just like to acknowledge that
7 this is -- even though there is just one person sitting
8 here in the target, thank you. Thank you very much.
9 I'll be here all night.
10 (Laughter.)
11 MR. KIRK: This is the product of a
12 collaborative effort between Office of Research, which
13 my colleague Rob Tregoning, and NRR with Carolyn
14 Fairbanks and Allen Hiser, and of course many other
15 staff included. So next slide. I'm going to talk
16 about the regulatory framework for a reactor pressure
17 vessel aging management during the period of subsequent
18 license renewal.
19 I will talk about ongoing work concerning
20 formula to predict embrittlement effects, talk about
21 our generic evaluation criteria used to assess
22 embrittlement effects and summary, and of course
23 answer any questions that should arise.
24 So this is the overall framework for the
25 presentation, and you'll see this at the end is a

1 summary. So first off, we're going to talk about the
2 overall regulatory framework, which talks to how we
3 project the change in the mechanical properties of the
4 reactor pressure vessel steel, be it transition
5 temperature shift, upper shelf energy drop, change in
6 JR curve properties and so on as the vessel ages.
7 This is the particular area where we'll
8 talk about the impact of SLR and answer the question of
9 whether there's a regulatory framework in place
10 adequate to address RPV aging management through 80
11 years.
12 The blue and the red areas talk about work
13 that's certainly related and topical, about the
14 predicted formula that we use and the generic evaluation
15 criteria that we use. But I'd just like to point out
16 to the Committee that these are things that we're
17 certainly not assessing for the first time for SLR.
18 These are long, ongoing topics within the
19 reactor safety community. So these started before SLR
20 and they'll continue as long as we're operating vessels.
21 So next slide.
22 MEMBER SKILLMAN: Mark, back up.
23 COURT REPORTER: Microphone please.
24 MEMBER SKILLMAN: So in this image --
25 excuse me. In this image, we have this sea of blue bulk

1 of data. So that's database stuff.

2 MR. KIRK: Yep.

3 MEMBER SKILLMAN: And then projections are

4 the dots. Is that what that is?

5 MR. KIRK: Well the -- and I'll show you

6 some real data later on, so I don't want to focus too

7 much on the graphic. But the idea was that we have --

8 the bulk of our experience of course at lower exposures,

9 lower ages, lower fluences, what have you, and of course

10 as you're aware there's a lead factor. So the

11 surveillance data leads the vessel.

12 As we go out to longer ages, higher

13 fluences, we get progressively less data. What you'll

14 see on the slide with the real data is it's actually not

15 as sparse as this, which is fortunate moving forward in

16 time.

17 MEMBER SKILLMAN: I was challenging in my

18 mind what is meant by "limited data at long exposures."

19 So those blue dots are actual data.

20 MR. KIRK: Our actual data.

21 MEMBER SKILLMAN: That it's not a whole lot

22 of data?

23 MR. KIRK: Yeah, and we'll get more

24 specific on that in Slides 11 and 12, so that I could

25 punt that question, yeah.

1 MEMBER SKILLMAN: Now I understand.
2 Thank you.
3 MR. KIRK: So this slide talks about the
4 regulatory framework that we have in place. It
5 discusses the two things: the provisions of Part 54 and
6 the provisions of Part 50. Of course Part 54 requires
7 applicants to do a time-limited aging analysis.
8 Embrittlement of the reactor pressure
9 vessel is clearly a time-limited aging phenomena, and
10 the main point of this whole slide is that within Part
11 50, we have a very mature regulatory framework to
12 collect data on time-limited aging through 10 C.F.R. 50
13 Appendix H, requirements for a surveillance program,
14 and also to do assessments of reactor vessel safety
15 through Appendix G for a normal heat up and cool downs,
16 50.61 and 50.61(a) for PTS events, and of course we have
17 related regulatory guides.
18 So because we have the benefit of this
19 mature and well-developed framework, we expect that the
20 methods of Part 50 will be what licensees will be using
21 to provide the required time-limited aging analyses
22 that are required by Part 54.
23 Next slide. So what I've said, Part 54
24 requires a TILA. Methods to demonstrate management and
25 compliance are expected to mirror the well-established

1 practices of Part 50 and just note that Part 50
2 incorporates ASME guidance and ASTM standards,
3 certainly ASTM standards by reference or they're
4 conditioned if the staff finds them to be not quite
5 thorough enough.
6 We'll note here that there are some
7 documents like certainly equivalent margins analysis
8 documents may need to be updated to demonstrate for 80
9 years, but the methodology is in place to do that.
10 So our overall conclusion is that from sort
11 of a legal framework and technical methodology
12 perspective, there's nothing particularly unique or
13 different needed for SLR. Therefore, the staff
14 believes that the existing regulatory framework is
15 adequate to 80 years.
16 Now that's not to say we're never going to
17 talk to you about reactor pressure vessels again. I
18 certainly hope not. So we'll go on to talk about some
19 of the areas.
20 What are the mechanical properties that we
21 have to make predictions for as Ben's neutrons continue
22 to hit my target? What are we predicting and talk about
23 documents containing those prediction formula, both NRC
24 documents, codes and standards documents, and what are
25 we doing to keep them current and up to date?

1 So next slide. So where are we making
2 predictions? Ben pretty much covered this already.
3 Traditionally, radiation effects have been monitored
4 for materials where the fluence is projected to exceed
5 10 to the 17th. In the under 40 year time frame, that
6 was pretty much limited to the area around the active
7 core.
8 As Ben has explained in much more detail
9 than I'll cover here, in extended operations the region
10 above 10 to the 17th, and by the way, 10 to the 17th is
11 where we begin to see effects on mechanical properties
12 when we do our SHARPI test or our fracture toughness
13 test.
14 That region enlarges to encompass
15 materials outside those adjacent to the core, and Ben
16 has addressed that, I think, pretty well from a fluence
17 perspective.
18 MEMBER BLEY: Mark.
19 MR. KIRK: Yep.
20 MEMBER BLEY: So far, including this
21 slide, nothing here would suggest to me that there would
22 be any new requirements in a GALL as 60. We already have
23 the requirement on the fluence, that if you go above that
24 fluence you have to dig more deeply.
25 MR. KIRK: Yes.

1 MEMBER BLEY: That's still true, where we
2 are, as far as you've gone?
3 MR. KIRK: Yes.
4 MEMBER BLEY: I don't know if you're going
5 to get --
6 MR. KIRK: Yeah, yeah. No, the -- we'll
7 talk a little bit about potential changes and screening
8 limits at the end, but there's nothing envisioned at
9 this time. The effort at this stage, as I see it, is
10 more to keep our predictive formula up to date, both from
11 a physical understanding perspective and also from an
12 empirical perspective.
13 MEMBER BLEY: Thank you.
14 MR. KIRK: Yeah. So next slide.
15 MEMBER BROWN: I had a question about the
16 last one.
17 MR. KIRK: Oh sorry.
18 MEMBER BROWN: Based on Ben's comments and
19 the pictures that you all have shown, why aren't there
20 modifications that could be applied to reduce the
21 neutron fluence at the nozzles, shielding of some sort?
22 MR. KIRK: Well --
23 MEMBER BROWN: Has anybody considered
24 that, or even annealing vessels --
25 MR. KIRK: I can't.

1 MEMBER BROWN: Put the annealing aside.

2 MR. KIRK: Thank you, thank you.

3 MEMBER BROWN: But the shielding is a

4 relatively mechanical approach to doing things, that

5 would take you out of this analysis regime and simplify

6 going forward.

7 MR. KIRK: Yeah, and I can't speak

8 specifically to shielding of the nozzle area, but

9 certainly many licensees have seen fit to shield

10 selected areas of the traditional beltline area,

11 because it's getting too much dose and would be

12 projected to exceed a regulatory limit.

13 That's always been the motivation. Say if

14 you've got a weld with higher copper in the traditional

15 core, in the traditional beltline region and that's

16 getting too much dose, you may employ management

17 strategies to reduce the dose. You may shield it so as

18 to stay well within regulatory limits.

19 Perhaps some of the other staff in the room

20 can add something to this. I would expect the same kind

21 of strategy would be taken in the case of the nozzle,

22 should it become necessary. But right now, just to

23 point out, the -- when you get up into the nozzle course,

24 you're looking at ID fluences.

25 They're starting to creep above 10 to the

17, but not very much, and you'll see later on when I show you some real data, 10 to the 17 is really where the material starts to the respond. So we're still talking about small changes in transition temperature shifts.

MEMBER RICCARDELLA: Even at 80 years?

MR. MEDOFF: Mark, I'm going to help you out here.

MR. KIRK: Let me address this question. Well, if 80 years corresponds to 10 to the 17, then yeah. I mean as you know, once you get up to 10 to the 18, now we're starting to talk about measurable levels of embrittlement. But even so, still pretty much well within our limits. Jim.

MR. MEDOFF: Just in answer to your question, could we require shielding, the regulatory framework in Part 50 to address these assessments, if you don't meet the acceptance criteria in the rule, you have to take appropriate corrective action.

Some are very specific. Like if it's an upper shelf energy assessment, the rule would require you to send in an equivalent margins analysis for staff approval. But if it's something like a PTS assessment, they have options they can take on the corrective actions.

1 They could certainly reduce power; they
2 could modify the design and put in the shielding as you
3 said. But also we have the alternative PTS rule. They
4 could submit that as a license amendment. So we
5 couldn't force them into a specific course of action.
6 They do have options to take.
7 MEMBER BROWN: Thank you.
8 MR. KIRK: Okay. So this slide is just a
9 very high level slide showing you what are the
10 mechanical properties we're predicting and where do
11 they come into our regulatory framework? So certainly
12 -- no. Can you stay there? Thank you.
13 So certainly, our main mechanical
14 properties that we're looking at are sharp transition
15 temperature shift, as measured by the shift at 30 foot
16 pounds or 41 joules. I'll probably go back and forth
17 between those, and the drop in the upper shelf energy.
18 All of our current regulations and the ASME
19 code are expressed in terms of those values. In the
20 event those values start to push up against regulatory
21 limits or your PT window starts to close, you may need
22 to move into fractured toughness space to try to get some
23 relaxation from the margins that are implicit to the
24 SHARPI approach.
25 So there you'd be looking at JR curve

1 properties for tearing resistance on the upper shelf
2 and/or transition temperature shift as measured by the
3 master curve. All of those type of methodologies are
4 undertaken on a case-by-case basis, where the SHARPI
5 screening limits are -- have been transgressed or are
6 likely to be transgressed.
7 But these are all the things. We need to
8 know the effect of radiation damage and consequence, the
9 chemistry on each and every one of these properties.
10 I'm going to show you where that gets into both our
11 regulatory framework and our codes and standards
12 framework.
13 So next slide. So the two slides now where
14 these properties appear. First where they appear in
15 our regulations and guidance, and then the next slide
16 will be where they appear in consensus codes and
17 standards.
18 So Delta T30, we have not one, not two but
19 three places where it appears in our regulations, in Reg
20 Guide 1.190 and in both the PTS rule and the alternate
21 PTS rule. Also in Reg Guide 1.99 is a formula to predict
22 the effect of radiation dose and chemistry on upper
23 shelf energy.
24 In Reg Guide 1.161, we have formulas to
25 predict the effect of radiation on JR curves, and of

1 course at the bottom, it's not a property but it's worth
2 -- certainly worth mentioning is the data source for all
3 of this is a required 10 C.F.R. 50 Appendix H
4 surveillance programs.
5 The second column is the dates on which
6 these documents were last revised by the NRC, and the
7 final column shows when they were last reviewed by the
8 NRC, and in some cases indicates where there's effort
9 underway. So things you should be probably expecting
10 to hear from the staff about in the near future.
11 MEMBER RICCARDELLA: What does that 2016
12 note mean?
13 MR. KIRK: We are -- every so often, and I'm
14 not going to state a number of years because it's not
15 my area. But we review regulatory guides for their
16 continued adequacy. We did that last in 2014 for Reg
17 Guide 1.99. At that point, and this is on the website,
18 we said that Reg Guide 1.99 was adequate for continued
19 use.
20 But we also noted in the letter that's
21 posted on the website that we would be doing a thorough
22 and complete review of the provisions of Reg Guide 1.99
23 relative to the current state of knowledge, and would
24 be publishing that sometime in 2016.
25 So that's what that reflects, is you should

be seeing or we've committed to on our website to do that in 2016, and that would be in the form of a report.

MEMBER RICCARDELLA: But it's not a revision to the Reg Guide?

MR. KIRK: Not at this stage. No. This would be the technical basis that would be needed to either say Reg Guide 1.99 is good for continued use, or we should change Reg Guide 1.99 within some time frame, yep. But not a new -- we've not committed to a new version of Reg Guide 1.99 at this stage.

MEMBER RICCARDELLA: Thank you.

MR. KIRK: Okay. So next slide. This shows the parallel places, in some cases parallel, where these mechanical properties appear in consensus codes and standards. There's a predictive formula for Delta T30 in the ASTM E900 standard, which was recently updated, in fact republished in 2015 of this year.

That's not yet been reviewed by the NRC. Of course K1C and K1A appear in ASME Section 11, as everybody knows. There are parallel requirements for surveillance in ASTM standard E185 on surveillance program design and ASTM E2215 on surveillance program conduct.

I should note revision and updating as plants go to numbers of years that we never anticipated

1 they would go to before.

2 Both of those new ASTM standards are being

3 reviewed by the NRC as part of its current Appendix H

4 rulemaking, and also reflecting on something that Robin

5 pointed out in his slides, there is a code case N830

6 within ASME Section 11, Working Group on Flaw

7 Evaluation.

8 We're working together with our industry

9 colleagues. We've put forward a comprehensive set of

10 fracture toughness models that account for -- allow you

11 to predict both bounds and statistical distributions on

12 fluidic steel fracture toughness all the way from the

13 lower shelf to the upper shelf, and that's something

14 that's being put forward in ASME at the current time as

15 an alternative to current procedures, which are based

16 on RTM DT.

17 So next slide. So here's the real data,

18 and we'll have two slides of this. So let me -- there's

19 a lot on this slide. So let me orient you to what you're

20 seeing. In the upper graph is a plat of the current USA

21 surveillance database. This is all from 10 C.F.R. 50

22 Appendix H programs, and the horizontal axis is the log

23 10 of the fluence.

24 So we go from data -- actually there's some

25 data that haven't been plotted that go down as low as

1 times 10 to the 16, although they're not too interesting, all the way up to 10 to the 20th.

The vertical axis that you're seeing is the difference between the measured transition temperature shift value at 41 joules or 30 foot pounds, depending on what unit system you like. The difference between the measured value and the Reg Guide 1.99 prediction.

So if this was a statistical conference, we'd call this a residuals plot, and what you want to see to be happy is no residual trends. So you want to see basically a horizontal scattering of points, and really we're not doing so bad, considering that Reg Guide 1.99 was first put forward in the mid-1980's, based on a fit of only about 176 data points.

I should note there is a little bit -- there is a bit more data above the line towards the curve conservative prediction side than under the zero line. So there's a small conservative bias and a regulatory prediction formula. I don't think that should come as a terrible shock to the system.

We also see when we get up to the high fluences the beginning of possibly a non-conservative trend, and I'll talk a little bit more about that in a minute.

Getting to a question that I think Member

1 Bley asked earlier about the distribution of fluences,
2 I think the middle and the bottom graphics illustrate
3 this pretty well, at least that was my hope.
4 So the sources of the data are shown. The
5 PWR data comes from a 2011 EPRI report, so that's pretty
6 up to date. These are the ID fluences on all the vessels
7 that are now operating. So 2011 data for PWRs.
8 The data on BWRs is more out of date. There
9 is a new EPRI reference that I haven't incorporated, but
10 since they don't get out to high fluences I didn't bother
11 for this presentation.
12 MEMBER SKILLMAN: Mark, this is ID to
13 Quarter T?
14 MR. KIRK: ID.
15 MEMBER SKILLMAN: ID, not Quarter T.
16 MR. KIRK: Not Quarter T, ID.
17 MEMBER SKILLMAN: ID.
18 MR. KIRK: ID.
19 MEMBER SKILLMAN: Okay, thanks.
20 MR. KIRK: I'm trying to remember the
21 knockdown factor, but you can figure that out. But it's
22 important here to have, in my view, the ID fluences. I
23 know we used Quarter T in the integrity calculations.
24 The ID fluences, of course, correspond to the fluence
25 at which the surveillance data has been collected.

1 So you see here the distribution of data in
2 the fleet at 40 years in blue, 60 years in red and 80
3 years in green. Just so you don't have to do inverse
4 logarithms in your head, as a kindness to the Committee
5 from -- I have trouble --
6 At 40 years, the PWR fluences range at the
7 low end of 1 times 10 to the 19th, and at the high end
8 5 times 10 to the 19th. When we get up to the 80, the
9 projected 80 year fluences, they range from 2 times 10
10 to the 19th up to 1 times 10 to the 20th.
11 I just drew a few vertical lines, again so
12 we don't have to do it in our minds and we start to see
13 that when we get out to the 80 year fluences, that some
14 of the fleet may be getting into the region where the
15 surveillance data, which of course has a lead factor as
16 we all know. So we want to have advance information on
17 the fleet, good design, is getting into the region where
18 we might be seeing some non-conservatisms.
19 So now if we go to the next slide, it will
20 be the same thing, but you'll see a lot more data now.
21 Now I should say -- should have said before, both the
22 USA surveillance data and now I'll call it the non-USA
23 surveillance data; this is data from other nations.
24 The biggest data sets come from France, Germany,
25 Belgium, Korea, Japan and others here and there.

1 All these data were compiled by ASTM
2 Committee E1002, in support of its development of its
3 new transition temperature shift standard, which was
4 passed earlier this year as E915. I should point out
5 that while these data have been QA'd, reviewed and
6 accepted by the ASTM committee, they've not yet been
7 reviewed by the staff, and that's something that we'll
8 of course be looking at as part of the Reg Guide 1.99
9 effort.
10 But taking that as it is, these are the data
11 that are coming from overseas. Of course, what we see
12 first is we're getting more data at higher fluences, and
13 that larger amount of data is continuing to show the
14 somewhat non-conservative trend that we saw in the USA
15 data.
16 MEMBER RICCARDELLA: Is that because of
17 different chemistries?
18 MR. KIRK: All the chemistries are plotted
19 here. So everything --
20 MEMBER RICCARDELLA: I understand, but I
21 mean is that because they have worse chemistry overseas?
22 MR. KIRK: No, no, not particularly. They
23 have higher fluences overseas. Well, they have higher
24 fluences in their surveillance data. The chemistries
25 overseas are broadly, very broadly consistent with what

1 we have here. The major holdouts, which are the --
2 Can I point? The major holdouts, which are
3 some of these points down here, are the high nickel welds
4 that you see in one particular plant that everybody
5 probably knows the name of in Sweden, and a number of
6 German PWRs. But other than those plants that have a
7 higher nickel content than we're used to in the United
8 States, the chemistries are broadly consistent.
9 I think the major difference that you're
10 seeing here is simply that for a combination of reasons
11 I'm sure, the data from overseas tends to occupy higher
12 fluences than we have here. Which actually is good for
13 us, because the reactor designs are very similar.
14 There aren't any significant spectrum differences. So
15 this is again giving us a window into the future.
16 MEMBER RICCARDELLA: But looking at that,
17 you could say okay, we could ignore some of those
18 really bad points that are down at minus 150. But it
19 certainly seems that that trend toward non-conservatism
20 is --
21 MR. KIRK: Right, and this is why it's very
22 timely for the staff to be doing its assessment of Reg
23 Guide 1.99 at this time. Now of course the -- let's say
24 it. The staff hasn't looked at it, so I don't want to
25 say anything that we haven't deliberated and decided on.

1 But let's say at the end of the day
2 everybody agrees with the data. It's still true that
3 the bulk of the fleet is at lower fluences, due to the
4 lead factors. So then it becomes a question of when's
5 the best time to update the regulations.
6 One other thing I wanted to note, because
7 it's been brought out before and getting back to
8 Chairman Skillman's question about the amount of data.
9 This is the real plot to which I was referring. So in
10 some of the previous presentations it has been pointed
11 out, you know, savings have been made like data are
12 sparse, about 3 times 10 to the 19th.
13 You can't count the number of points on this
14 plot because fortunately there are so many. But what
15 I can share with you is that in the entire domestic and
16 international database, we have approximately 420
17 transition temperature shift values above 3 times 10 to
18 the 19th.
19 That works out to about 22 percent of the
20 total database. The EPRI programs will add about 33
21 values to that database. So that's about an eight
22 percent increase. But I should also note that we're not
23 relying in terms of future data strictly on those 33 data
24 points. Robin's slide also pointed out that there are
25 50 data points coming in, already as part of

1 surveillance programs that are above 3 times 10 to the
2 19th.
3 We'll also be getting more data from
4 international surveillance, and also from research
5 programs such as the ATR. So one of my colleagues, Bob
6 Hardies, once told me that the best kind of data is ABT,
7 and I wasn't familiar with that acronym.
8 He explained "already been tested." It's
9 the cheapest data available, and we have a lot of already
10 been tested data on which we can inform our current
11 decisions, in view of the fact that we'll be getting even
12 more data later.
13 So I focused a lot on the data. Just some
14 other things to point out. The existing work that's
15 been done within ASTM that the staff has yet to review
16 but will be reviewing, has focused exclusively on the
17 transition temperature shift.
18 ASTM has never had an upper shelf energy
19 drop relationship. That's one of the things that my
20 colleague Mike Gordon and I are working on in Research,
21 is using these data to develop an upper shelf energy drop
22 relationship and also to assess, of course, the adequacy
23 of the current relationship.
24 And as I pointed out, the two programs that
25 industry -- that Robin spoke of will be providing very

high value data to add to this information.

MEMBER BALLINGER: I have a question.

MR. KIRK: Yep.

MEMBER BALLINGER: I should have remembered this from a previous slide, but what's the maximum beltline fluence that you're going to see out to 80 years?

MR. KIRK: It's the end of the green line. It's projected at 1 times 10 to the 20th.

MEMBER BALLINGER: Okay. So it's just about where it starts to tail off? Just a little past, just little past.

MR. KIRK: Yeah. And then at the bottom, not to give short shrift to real fracture toughness data, but as I pointed out, there's an effort underway and Rob and I also reflected on this, within ASME to use the existing models, many of which came out of the NRC effort on pressurized thermal shock, in an integrated way to provide ASME code users with a complete set of fracture toughness models for use in code space.

That work is also ongoing and takes full account of the radiation damage effects. So if there aren't any further questions on this one, I'll go to the next slide.

Just talk a little bit about generic

evaluation criteria. Of course, the NRC's assessment of the adequacy of current evaluation methods, you know, for example the PTS limits -- I shouldn't say limits -- the PTS screening criteria, the methodologies for developing pressure temperature limits, will inform the need for potential future modifications.

And I think this is pretty common sense. We would only proceed generic revision if we saw an issue coming forward that's affecting many plants. If only one or two plants has an issue with the evaluation methodology, then that would probably be dealt with on a plant by plant basis. Right now, we don't see anything but we wouldn't rule it out either.

This approach is consistent with past efforts by both the NRC and the industry. The NRC examples were briefed a lot with this Committee, with 10 C.F.R. 50.61(a), and the industry examples, many of them involve the master curve.

Babcock and Wilcox's approach, as explained in their topical Report 2013 has allowed many plants with Linde 80 welds to use fracture toughness data in lieu of SHARPI data, and of course there's the Kiwanis example.

We have some ongoing efforts that might factor into needed revisions. At the NRC, we're

1 continuing to evaluate the possible risk-informed
2 revision of 10 C.F.R. 50, Appendix G on heat-up and
3 cool-down limits, and I've already mentioned that
4 numerous times the work in ASME on fracture toughness
5 models.
6 So next slide. So just to summarize, in
7 terms of direct SLR impact, we feel like we have an
8 adequate regulatory framework for 80 years, consistent
9 with the efforts in the blue and the red that I've talked
10 about, to keep our predictive formula up to date and
11 informed by that and understand if there comes a time
12 when we might need to take a look at generic evaluation
13 criteria.
14 So that's the end of my prepared comments.
15 If there are any further questions?
16 MEMBER SKILLMAN: Colleagues, colleagues.
17 Any further questions or questions for Mark please?
18 (No response.)
19 MEMBER SKILLMAN: Hearing none, Mark thank
20 you very much.
21 MR. KIRK: Thank you.
22 MEMBER SKILLMAN: Steve, back to you.
23 MR. BLOOM: Actually, we have one more
24 presentation by Sri Rao on vessel internals.
25 MEMBER SKILLMAN: Okay.

1 MR. BLOOM: In addition to Sri, we have Jim
2 Medoff from the Office of NRR.
3 MR. MEDOFF: Hello, this is Jim Medoff.
4 I'm a member of --
5 MEMBER SKILLMAN: Microphone on please.
6 There you go, the green button. Thank you.
7 MR. MEDOFF: Good afternoon. I'm Jim
8 Medoff of the Division of License Renewal staff. I've
9 been working with members of our division, the Division
10 of Engineering and members of the Office of Research to
11 look into what we need to do to update the aging
12 management guidance for reactor internals, both for the
13 boiling water reactors as well as those for the PWRs.
14 I have to my left Dr. Sri Rao. He's going
15 to do most of the presentation. I'm going to present
16 the first two slides, because it relates to what we're
17 doing for updating the two aging management programs for
18 those type of components.
19 For the existing GALL, which is GALL
20 Revision 2, we have one aging management program.
21 That's AMP XIM-9, boiling water reactor vessel
22 internals. It relies on a number of EPRI BWRVIP
23 documents for its aging management recommendations.
24 It's a pretty comprehensive program, and we
25 rely on the VIP inspections for managing things like

1 cracking, even lost material and some other effects in
2 the boiling water reactor internals. We're going to
3 continue to use those documents for renewal. So we will
4 that AMP in the guidance document.
5 The one thing that we will need the industry
6 to assess is because those reports are currently based
7 on 60 year reviews, if they relate to any aging
8 management of things like irradiation effects like
9 irradiated assisted stress corrosion cracking, you've
10 heard Robin talk about it earlier, we may need to think
11 about whether to go back to look at the guidelines, to
12 see whether they need to be updated.
13 So there will be a further evaluation
14 recommendation for those types of effects. But other
15 than that, we will continue to use XIM-9 in the guidance.
16 That's about all we're prepared to say right now,
17 because we have to put out the updates of the GALL for
18 commentary.
19 For the PWR internals, we have the current
20 GALL. GALL Revision 2 has a program, XIM-16A. It's
21 based on the guidelines and the EPRI MRP report,
22 MRP-227-A. This is improved methodology. It's a
23 little different than that for the boilers.
24 When EPRI developed the guidelines for
25 that approach, they did not -- it's not as comprehensive

1 as those for the BWRs. They used a more risk-informed
2 approach to aging management. They developed what they
3 called some functionality analyses, some failure nodes
4 and effect analyses to determine which were the most
5 safety-significant internals to look at from an aging
6 standpoint and from a consequence standpoint.
7 Those are the reckoning, the ones that were
8 the highest consequences of risk sort of were the ones
9 they were looking to for in terms of inspection of the
10 internals. We did approve that. But it's not quite as
11 comprehensive as those for the boilers.
12 An issue is because it was based on more of
13 a risk-informed approach, the assessments were based on
14 60 years. So we don't know that whether the categories
15 would change for the inspections going out to 80. So
16 unfortunately at this point, because the 80 year
17 report's not coming in until seven years down the road,
18 we can't approve that.
19 So in the updated GALL, we will be removing
20 that AMP and we're going to tell the industry that when
21 they come in for the PWR internals to propose a
22 plant-specific program and to justify it, and that's
23 what we're going to go out with the industry for
24 commentary.
25 So that's sort of the approach we're

1 taking. What Dr. Rao will talk to after this, we'll go
2 into some of the slides of the research they're doing
3 to support what we can do for the internals.
4 MR. RAO: I am Sri Rao from Research. I
5 have my own commercial. I'm hard of hearing. If you
6 have a question, please ask it a little bit loud. Then
7 only I can answer it. That's my commercial.
8 We have -- we're doing. I mean, I am from
9 Research, and at Research we -- our response will be is
10 to assist the regulated side of NRC, the staff, to
11 provide them with technical -- adequate technical basis
12 which they can use which can be helpful for them in their
13 decision making or provide for their regulatory
14 concerns.
15 So basically the key technical issues that
16 we are actually pursuing right now under this -- I had
17 removed the word "irradiation-assisted stress
18 corrosion cracking because it's a narrow. So we made
19 it a broad, say irradiated-assisted degradation.
20 Particularly, we are looking at stainless steel, which
21 are for the baffle plate material and also high-strength
22 bolting material. Particularly our concentration is
23 on at high fluences, which is actually associated with
24 the external operation.
25 Our research basically focuses on

1 microstructure changes including the void swelling
2 which may impact the mechanical properties such as the
3 crack growth rate and fracture toughness, and of course
4 the chemistry sides, stress corrosion cracking.
5 And the second major thrust we have is loss
6 of fracture toughness for cast austenitic stainless
7 steel. So we are concentrated looking at the effect of
8 both thermal and neutron embrittlement, when you cannot
9 do at the same time, but at least we try to do first
10 saturated with thermal aging. Then we do irradiation
11 and then look at -- as a function of for those what are
12 the effects?
13 And this is -- and also, the second topic
14 is thermal embrittlement for prolonged exposure at
15 operating temperatures. This morning DOE has
16 presented they just -- they started. We have actually
17 archival experience, about 40 years on this thermal
18 embrittlement of cast stainless steel. Actually right
19 now we are machining CF8 CASS material, which has been
20 cooked for about 20 to 25 years at 350 degrees. We are
21 machining to find out the fracture toughness of that
22 one. So we are continuously doing it from long time.
23 The latest knowledge, this is how we put
24 this one, is what we have -- already have it in the
25 supporting SLR. I just want to briefly give you a

1 review of that one. We identified how the material
2 chemistry will affect the IASCC. In fact, such as the
3 -- we establish about the oxygen content, the
4 high-oxygen, low-sulfur, high-sulfur, low-carbon or
5 even the criticality at which you will get very good
6 resistance from the source corrosion cracking. And so
7 this has been published. It's available in
8 NUREG/CR-6965.
9 We have established the effect of neutron
10 fluence on the neutron embrittlement, the degree for
11 which the neutron embrittle. It has come in
12 NUREG-7027. And in fact we have two papers published
13 Journal of Nuclear Materials, that one.
14 We also establish the difference between
15 the effect of fast reactor irradiation and light water
16 reactor irradiation for the core internal materials.
17 In fact, the importance of this when I will talk about
18 it later on and it is actually published NUREG-7128.
19 These are the things that we have completed during the
20 last five years time period.
21 So the recent activity to support present
22 SLR, particularly the aging degradation of reactor
23 vessel internal -- our object is to assess the effect
24 of IAD and void swelling during the subsequent license
25 period under representative plant conditions,

1 including irradiation and also the temperature. We
2 would like to also continue our investigation on the
3 combined effect of thermal and neutron irradiation,
4 particularly the embrittlement of CASS under
5 representative temperature and irradiation conditions.
6 Right now the research projects, NRC's IAD
7 research focus on -- we are doing both cooperative and
8 also our own independent research, particularly with
9 the ex-plant material, Zorita, the Spanish reactor,
10 which is a small reactor. It is beaten up like hell.
11 And we have extracted the material and with EPRI
12 cooperate we are actually working investigating the
13 material degradation mechanisms involved at high
14 fluence. In fact, right now the fluence we are talking
15 about is about 55 dpa. I will talk about later on a
16 little bit more. NRC cask research focuses on
17 independent testing of irradiated casks to understand
18 the combined effects.
19 The reason why we are trying to do the
20 cooperative research is to maximize the resources. But
21 at the same time we keep our own interpretation of the
22 results so that we'll have an independent assessment of
23 what the test results are.
24 So this is our broad the IAD research plan
25 taking up to 80 years of operation. The red line what

1 we think one dpa per one operational year for LWR to take
2 effect. And if we're talk like that one, at the end of
3 80 years of period we assume the baffle plates will
4 receive about 80 dpa. At the same time the welds we have
5 near the cold metal welds or -- they could receive
6 maximum of about 8 to 10 dpa. So that's what we had it.
7 So there is a hyperlink of the -- I'm sorry.
8 PARTICIPANT: This one?
9 MR. RAO: Yes. So this Zorita program is
10 actually -- EPRI is operating it. It's an
11 international program with which we have utility
12 partners and also regulators. So the main thing is how
13 with ex-plant materials and test the fracture toughness
14 and crack growth rate, crack initiation and also
15 microstructure characterization at 3 dpas, from 10 to
16 50 dpa at 320 -- anywhere from 290 to 320 degrees.
17 Go back. So the weld -- as I said, when we
18 harvested the Zorita material, we also had some cold
19 barrel run-through at Zorita reactor. Typically
20 they're one to two dpa. After -- I mean, when the baffle
21 plates receive 55 dpa, the welds we have only 1 to 2 dpa.
22 That's why we thought that 80 dpa maximum we could
23 receive 8 to 10 dpa. That's how we're thinking about
24 it.
25 We harvested the welds and we are actually

1 making samples for testing the welds and also
2 heat-affected zone for both CGR, fracture toughness
3 testing. And this is also for partners, collaborating
4 partners, EPRI. And this testing too is being done at
5 Studsvik in Sweden.
6 The second thing is for the subsequent
7 license period we went up to eight dpa, which take a
8 little bit of time. So we have partners under Halden
9 Research Project. So they actually would like to
10 participate with them for a collaborative effort in
11 testing these welds. So there at Halden we are going
12 to irradiate it to eight dpa and then we'll test it. And
13 right now some samples of that are currently tested in
14 Halden and some of the welds will be tested
15 independently at our contractor, the national
16 laboratory. That's what we are planning right now.
17 The testing of two dpa will be started in 2016.
18 The biggest thing, it is the big wild best,
19 the high fluence plate, how to get 80 dpa. There are
20 different people have different methodologies they're
21 proposing. And some people -- there are pluses and
22 minuses in the different approaches. We want to take
23 a jump board, 55 dpa Zorita material and try to go into
24 80 dpa. Some people said fast reactor irradiations can
25 reach very quick, but the thing is what we found is they

have not considered enough and their results from the fast reactor completely the microstructure mechanical defect structure properties are different to the tough samples irradiated using LWR conditions.

Some people suggest why can't we do ion radiation or proton radiation. Unfortunately, for 30 years I worked in the same area. I worked on -- both on ion radiation and also proton radiation and neutron irradiation. To me, in my opinion it is like comparing apples to pineapples. The only commonality is apple in both terms. So I don't believe in that one. So are actually debating. Recently we presented it at a conference to show the technical experts these are the things that are there. What would you do if you are reading this program? So that's how we are trying to get the conscientious idea how to approach this.

Go back. So that is the -- yes, next slide. So right now basically that is the state with our effort.

The second one supporting is on CASS, CASS materials. Up to now the -- it's the most recent. We evaluated the combined effects and we are the first in the world to say that both thermal and neutron radiation effects the fracture toughness. And we published it. It's in NUREG/CR-7184. And we also proposed a bounding value for delta ferrite for thermal and neutron

radiation with acceptable fracture toughness values that are applicable to develop a CASS screening criteria, which is in NUREG/CR-7185. And in fact, we updated it. That is coming -- I will talk about it in a minute. In fact, this is how we support the SLR. It's right now the Grimes letter, Grimes criteria where that the regulator is using our recommendations based on the proposed mechanisms.

Not only that one, we updated the methodologies to calculate the fracture toughness and establish the mechanism in ASME code case. They accepted our proposed mechanism and they would like to implement the modification to that one based on NUREG/CR-4513, Revision 1. Particularly, this is for thermal embrittlement. In fact, we want to file for thermal and also radiation embrittlement.

Next slide, please. This is the slide we are proud of, right, as I said. Here blue is as we say un-irradiated material. So, the irradiation is pink. And that's how I see. So if you see like from -- as we receive the material loses fracture toughness. At the same time, if we age it, it loses fracture toughness. You start with the aged material, irradiate it, even to a small dose of 0.08 dpa. Very, very small dose. We have seen that one. We saw that one both in CF-3, CF-8

1 and CF-8M, all these three. We focus on these three
2 because most of the components, these are the ones that
3 are used in the reactors. So, this definite
4 demonstrates that with the combined effect thermal, as
5 well as radiation effects the fracture toughness even
6 at a low dose at 0.08 dpa.
7 Next please. So right now the CASS
8 Research Plan -- and I want to give you the -- what is
9 the status. We are actually doing CASS and also welds.
10 And we did gap analysis, a thorough gap analysis. We
11 have done literature review to update the guidance and
12 address the neutron and thermal effects. In fact,
13 although I put it expected completion is 2016, the new
14 NUREG/CR-4513, Revision 2 has completed its review
15 process. In fact, right now it is being editorial
16 review by all publications. So, we had actually the
17 letter. I think hopefully by end of January 2016 or
18 early February 2016 that NUREG/CR-4513 review, too,
19 will be available publicly. In fact, that one we
20 actually knew the effect of nickel, high-nickel,
21 low-nickel, how it effects the fracture toughness.
22 That's one thing.
23 And on the welds, we have actually
24 technical report which will come hopefully by March or
25 -- and we say it could be June 2016. However, we have

1 two papers, both on the methodology and the mechanistic
2 aspect of CASS material, and also the welds that has been
3 accepted in general pressure vessels technology. And
4 in fact, they may come any day. In fact, the electronic
5 version is already available in the database of the
6 journal.
7 The CASS three dpa and testing, we were
8 delayed with irradiation problems at Halden that's
9 there. We do the irradiation. And right now we
10 started working on that one and we hope to complete that
11 one in 2017. All of our testing is being done at Argonne
12 National Laboratory.
13 Next, please. So this is the actually
14 reactor vessel internals research timeline. See,
15 Zorita Internals Research Program hopefully will
16 complete by the end of 2016. The welds at Studsvik;
17 that is the just base one and two dpa welds we harvested,
18 will be -- the data will be available by the end of 2017.
19 And the Halden welds, they keep coming because the
20 irradiation takes about 80 years of time depending upon
21 the dose. And high fluence materials, we are in the
22 planning stage in what we want to do, that it will take
23 a little bit.
24 However, on the extreme right-hand side you
25 have the first subsequent license renewals applications

1 will start coming. My understanding from my SLR and the
2 staff colleagues is around somewhere end of '18 and
3 middle of 2019. But that time we will have some very
4 good understanding of it. And in the case of CASS, the
5 validation of thermal aging is completed. And 0.08
6 dpa, that was completed. The CASS weld gap analysis,
7 although we put 2016, this stripe should move almost
8 close to the present line. And the three dpa CASS
9 materials hopefully will be completed by 2017. So that
10 is the present timeline.
11 And I would like to summarize what I said.
12 We described the regulatory framework for the reactor
13 vessel internals and we identified important technical
14 issues from them, and then we have taken all that one.
15 We discussed about ongoing research to address the
16 degradation mechanisms of reactor vessel internal
17 during the subsequent license renewal period,
18 particularly the effect of IAD on stainless steel plates
19 and weld materials at high fluences. A loss of fracture
20 toughness in CASS materials because it's under thermal,
21 and also at neutron irradiation.
22 We continue the staff collaboration; that
23 clearly is one of our aim from research, to provide
24 technical basis for the regulatory staff to address an
25 disposition of any emergence issues identified through

1 this research. Thank you very much.

2 MR. MEDOFF: And I'd like to add something

3 to that last statement, because it does relate to the

4 way we're doing process here, especially what we're

5 doing for the GALL update. We are very well aware of

6 the research activities in these areas. We mentioned

7 a guideline for CASS, Chris Grimes' letter. Before we

8 ever had interim staff guidance documents for license

9 renewal, the license renewal is addressed as a project

10 directorate, and we put out these project director issue

11 letters. The Chris Grimes letter he is referring to is

12 the current bases for managing thermal aging

13 irradiation induced loss of fracture toughness in CASS

14 materials. It's what we have in the existing AMP.

15 If through their activities those

16 guidelines need to be updated; and I think there is a

17 current push for DE to start doing the technical work

18 for the Division of License Renewal, we will keep on top

19 of those. If it's in time for the GALL update, we'll

20 update the AMP and get it out for commentary to the

21 industry. Otherwise, if we need to update them, we'll

22 do it through an interim staff guidance document. So

23 I did want to say that.

24 CHAIRMAN SKILLMAN: Jim, thank you.

25 Colleagues, any questions for the staff at this point,

please?

MEMBER RICCARDELLA: Yes, I just have a technical -- on your slide No. 15 -- could we take a look at that, please? You say J interval. Was that J1C or JMAX or JR?

MR. RAO: Okay. It is basically J two millimeters.

MEMBER RICCARDELLA: J?

CHAIRMAN SKILLMAN: Dr. Rao?

MR. RAO: J at two millimeters.

CHAIRMAN SKILLMAN: Dr. Rao?

MR. RAO: Yes.

CHAIRMAN SKILLMAN: Please speak into your --

MR. RAO: Oh, sorry.

CHAIRMAN SKILLMAN: Okay. J at two millimeters extension?

MR. RAO: J at two millimeters, because the reason is most of the samples are 1/4T-CT sample. So that is only the maximum we can go.

MEMBER RICCARDELLA: Okay. So, it's basically an estimate of J1C?

MR. RAO: Yes.

MEMBER RICCARDELLA: And so what this says is that both aging and irradiation have an effect, but

once it's aged, the irradiation doesn't have as much of an effect, right?

MR. RAO: Yes. I mean, not that much. But I mean, at this -- it has to be recognized we are only doing at 0.08 dpa. If I increase the dpa, I mean, it will have more effect.

MEMBER RICCARDELLA: Yes.

MR. RAO: That's what we want to do it with three dpa.

MEMBER RICCARDELLA: I understand, but I'm just looking at the big effect of radiation for unaged material versus for aged --

MR. RAO: Yes, big difference with thermal. I mean, it is well established in CASS when you are -- in fact, most of the CASS where we thermally treated them at 400 degrees for about 10,000 hours. So more or less how much we increase the saturation and complete it down there.

MEMBER RICCARDELLA: Thank you.

MEMBER BALLINGER: This also I think is the same for Wallace?

MR. RAO: Yes.

MEMBER BALLINGER: So you see the same effect of aging and in situ testing in welds, I think.

MR. RAO: That is what it is from the

1 wisdom. And also, some of the data what we had the
2 paper, as I said. But we want to reserve it until we
3 get our weld test data, because as I said, we are right
4 now doing a series of welds in heat-affected zones. So
5 we want to resolve that. And so --
6 MEMBER BALLINGER: Thank you.
7 CHAIRMAN SKILLMAN: Colleagues, any
8 further questions for the staff at this point?
9 (No response)
10 CHAIRMAN SKILLMAN: Jim, Dr. Rao, Heather,
11 thank you very much.
12 Ladies and gentlemen, we are recessed until
13 1:30, 1330, on that clock.
14 (Whereupon, the above-entitled matter went
15 off the record at 12:30 p.m. to reconvene at 1:30 p.m.
16 this same day.)
17 MEMBER SKILLMAN: Ladies and gentlemen we
18 have a quorum, and we are back in session. Sherry,
19 please proceed.
20 MS. BERNHOFT: Okay, thank you. This
21 afternoon, we're going to start with a topic that I'm
22 sure is going to keep everybody awake after lunch.
23 We're going to talk about concrete. No, this is a real
24 story I have to tell you. Before I came to work for
25 EPRI, I really did not appreciate concrete. I really

1 thought it was, like everybody said, watching concrete
2 dry. But since I've come to work for EPRI, and John's
3 heard this before, I've really learned that concrete is
4 very fascinating. You get into aggregates, and how it
5 specs, and how it fails, so it's a very challenging,
6 technical topic. So what we're going to do--
7 [Off mic comment]
8 MEMBER SCHULTZ: Do you have your
9 microphone on?
10 MS. BERNHOFT: Okay, we're off to a good
11 start this afternoon. So the real issue with concrete
12 is it's a need to monitor and look at aging management
13 programs for concrete structures due to, as we've talked
14 about, the radiation and environmental exposure, and
15 Dr. Ballinger asked the exact correct question before;
16 we are going to be talking quite a bit about neutron
17 irradiation on concrete in this presentation. So that
18 was a good set up for earlier. Concrete research has
19 been going on for a while; we started it with a
20 literature survey. I won't go through all of these
21 reports here, but each of those is a technical report
22 that has been put out by EPRI; the report number that's
23 in parentheses is how you would find those on epri.com,
24 but you can see that we've put together an aging
25 management, we've looked at some different inspection

1 techniques, we've done some literature surveys on the
2 radiation effects of concrete, what we'd expect the
3 fluences to be at 80 years on the reactor vessel cavity,
4 and Keith will be going into quite a bit of work or follow
5 up conversation on what we've done with that. I'll talk
6 some about boric acid effects and some other things.
7 So the four topic areas that we're going to
8 talk about today is the effects of irradiation on
9 concrete, concrete degradation due to the alkali-silica
10 reactions that we've talked about some, and we'll give
11 a little more background on that, boric acid damage when
12 the spent fuel cools, and the creep impact on
13 post-tension containments. First topic area, the
14 problem statement here right now is what is the impact
15 of the vast neutrons that escape from the reactor core
16 and basically interact with the biological shield wall,
17 and understanding what that is going to do long term to
18 the capability of the biological shield wall for vessel
19 support. You can see the research duration that we
20 outline this over; it's going to from 2012 to 2018. So
21 I think this closes some of the gaps that Chris and Brian
22 were talking about in their earlier discussion to have
23 good information on these topical areas. We do have the
24 technical report that we talked about there that was
25 recently published.

1 This is--this whole topical area under
2 irradiation damage of concrete and where we are on it
3 right now, this is one that we've partnered I think
4 extremely well with Oak Ridge National Labs through the
5 DOE LWRS program on. This is a really good example of
6 what I was talking about before of the fundamental
7 research being supported by the national labs, and the
8 applied research being done by EPRI with our members.
9 And Keith's going to go through quite a bit of detail
10 on the work he's doing to understand the fundamental
11 mechanisms of concrete when exposed to radiation. What
12 we're doing from an EPRI standpoint is we're looking at
13 what the bounding fluences are, and we're also taking
14 down all this information and we're doing what I call
15 the so what. And the so what is what is the structural
16 significance of the concrete there to support the
17 reactor pressure vessel when it's been exposed to the
18 radiation impacts out to project at 80 years of
19 operations.
20 MEMBER SKILLMAN: Go back to slide 5,
21 please.
22 MS. BERNHOFT: Yes.
23 MEMBER SKILLMAN: The image on the right,
24 upper right there is a fascinating image; it shows
25 several things. It shows the box at port for the

1 reactor vessel nozzle, it shows the liner, which is the
2 water integrity barrier for refueling, but it also shows
3 the rebar spacing and size.
4 MS. BERNHOFT: Yes.
5 MEMBER SKILLMAN: So number 11 is
6 approximately as big as your wrist; there is a lot of
7 steel in that concrete, a huge amount of steel. And so
8 my request is, as you speak about concrete, you also
9 speak about the relationship of the steel in the
10 concrete, because the steel is susceptible to heating
11 if the neutrons get that far, and hence the potential
12 for delamination of whatever the mat is between the
13 rebar fabric and the concrete.
14 MS. BERNHOFT: Okay.
15 MEMBER SKILLMAN: If you would, please.
16 MS. BERNHOFT: What I'll let happen is when
17 Keith gets to his presentation, a lot of the work that
18 he's doing is actually looking at the attenuation of the
19 neutrons through the concrete, and what he'll show is
20 it only goes the first few inches through the concrete
21 itself. So you're going to have so much attenuation
22 when you get to that point.
23 MEMBER SKILLMAN: Should be.
24 MS. BERNHOFT: Yes.
25 MEMBER SKILLMAN: But as long as we can

back that up--

MS. BERNHOFT: Yes.

MEMBER SKILLMAN: --with data, fine.

MS. BERNHOFT: Okay.

MEMBER SKILLMAN: Thank you.

MS. BERNHOFT: Okay.

MEMBER BLEY: Sherry, one other thing, just to jump ahead; I didn't have a chance to go through your slides. Are you also going to be doing research on the steel plate construction that's in some of the new structures? Steel plate with concrete in the middle that you--

MS. BERNHOFT: Are you talking like the modular constructions?

MEMBER BLEY: Yes.

MS. BERNHOFT: We may have work going on with that with our advanced nuclear technology group; specifically in the LTO program, they're not working on that right now.

MEMBER BLEY: Okay, well it's a while before--

MS. BERNHOFT: Yes.

MEMBER BLEY: --if you're looking at aging effects, but--

MS. BERNHOFT: No, that's good, that's one

1 thing we do need to make sure that real time, we're
2 passing on our lessons learned to our advanced nuclear
3 technology group too, so valid point. Okay. So--and
4 so what I'm going to do is, Keith is going to pick it
5 up from here; he's going to talk about this curve, how
6 we generated it and some of the data that he's doing to
7 back that up, and then I'll talk about how we're doing
8 some of the structural evaluations.
9 MR. LEONARD: Okay. Thank you, Sherry.
10 So what you're seeing in front of you, and we'll see this
11 again in a later slide with reference to some of the
12 materials properties in terms of concrete, is the
13 neutron fluence and operating U.S. BWRs at 80 years
14 operation. So this would be based on extrapolations of
15 the current flux to what an 80-year operation would be,
16 and you can see that it's fragmented into two loop, three
17 loop, four loop. So there's quite a range that appears.
18 In general, we're looking at above one times 10 to the
19 19th fluence levels, neutrons per centimeter squared,
20 fluence levels. BWRs are in order of magnitude less
21 than a PWR. Thank you.
22 So one of the things that we're working on
23 currently, and a lot of my talk is going to be from the
24 molecular build it up into the structural components and
25 some of the modeling that's being done. And so one of

1 the subject matters that we're looking at is the
2 fundamental aspects of radiation damage in the
3 concrete. So one of the experiments that's ongoing
4 right now is looking at the components that make up the
5 concrete, performing the radiation experiments on that.
6 So we have single crystal mineral analogs of quartz,
7 calcite, dolomite, in addition to concrete and concrete
8 paste samples that have been irradiated at the high flux
9 isotope reactor. So these are a couple of different
10 temperatures; sealed, meaning it's a sealed capsule;
11 perforated meaning it's open to the hydraulic port fluid
12 so you're looking at about 50 degrees Celsius, and a
13 difference in fluences. So we're looking at what is the
14 fundamental behavior of the individual components.
15 So molecular dynamic simulation of
16 irradiated minerals, so this is looking at a couple of
17 examples of bionic and covalent materials and the
18 changes associated with irradiation damage. So in the
19 calcite, we see bond angle rotations, while in the
20 quartz we see disordered structure appearing. You have
21 progressive disordering of the alpha quartz and the
22 other quartz, the disordered silica under energetic
23 particle exposures. Okay, next slide. So this
24 changes a number of things. One, is it changes the
25 dissolution rate, so how the material behaves under

1 different solvent pH levels, and what you'll see on the
2 left is the change associated, the lower curve or line
3 is un-irradiated data; the upper line is irradiated
4 data.
5 So you see a pretty substantial increase in
6 dissolution rate for a given pH level for the silica,
7 whereas you're not seeing that for the calcite. This
8 dissolution rate is also kind of--goes along with the
9 change in atomic constraint, which is kind of the
10 nearest neighbor ordering structure, and it kind of
11 relates to the mechanical properties. One thing to
12 point out is the change in dissolution rate; it could
13 possibly, although it hasn't really been seen, it's a
14 possibility of radiation-assisted alkali silica
15 reaction. So as the number of atomic constraints goes
16 down, and you can see on the left side the accumulated
17 energy damage for the comparison between quartz and
18 calcite. So quartz undergoes a lot of changes, and this
19 is manifested in the volumetric slowing. So you're
20 going from a crystalline structure into an amorphous
21 structure, and it's--it accompanies about a 15 percent
22 volumetric slowing. So this is radiation-induced
23 volumetric expansion.
24 So how does this expansion affect your
25 concrete aggregate? So what you're seeing here is a

1 meso-scale finite element simulation that was
2 performed, and what you have is the blue is the
3 aggregate, the light grey is the pristine cement paste,
4 and the black is the cracked cement paste. So the
5 initial segment A is basically just a temperature hold
6 at elevated temperatures about 223 degrees Celsius, and
7 then it goes, starting in B is where we start introducing
8 radiation damage. So what we see is that you get
9 volumetric expansion, and then you get damage and
10 cracking in the material. So how does this manifest
11 itself in the overall properties of the material, and
12 you see a relative compression strength reduction at
13 very high fluences. Now there's a lot going on in this
14 plot. So originally, this plot was based on much
15 fewer data by Hilsdorf back in 1978, and recently a lot
16 of investigation into literature, reports by Kevin
17 Field at ORNL has bulked this up. So now we have as you
18 can see a large number of data, and it's segregated into
19 silicates, limestone, and miscellaneous heavy
20 aggregates. And this will be kind of important in the
21 next slide.
22 MEMBER RICCARDELLA: Excuse me, is the
23 green or the red indicated, the Hilsdorf versus the new
24 data?
25 MR. LEONARD: It's not specifically; I'd

1 have to go back and check. It's not specifically
2 stated; I'd have to go back to the original report for
3 that.
4 MEMBER SCHULTZ: Is what we're seeing
5 here, the curve that you're showing is Hilsdorf's curve?
6 Is that--
7 MR. LEONARD: It's pretty close to it;
8 Hilsdorf tapers off a little bit just because of the
9 fewer data points above.
10 MEMBER SCHULTZ: So what we're seeing is
11 the updated curve--
12 MR. LEONARD: It's an updated curve, and
13 really kind of the bound is just based on probabilistic
14 inaccuracies. I mean, there's going to be a good amount
15 of inaccuracy; one thing that in discussions with Kevin
16 and what was found in digging through these reports is
17 temperature data was not very well categorized.
18 Fluence data also not as well categorized as what you
19 would like to see; particularly, a lot of these reports
20 or information was from a considerable time ago. So
21 there's kind of some flexibility, and so we don't want
22 to necessarily limit--there's kind of an uncertainly
23 band associated with the data.
24 MEMBER SCHULTZ: So--all right. The
25 shaded area, does that represent some analytical

uncertainty range--

MR. LEONARD: Yes.

MEMBER SCHULTZ: --or is it just a picture to cover the data nicely?

MR. LEONARD: No, it's an analytical range based on just probability.

MEMBER SCHULTZ: Okay.

MEMBER BALLINGER: So some of the old--

MEMBER SCHULTZ: That's fine; it gives a picture of what you're describing. Thank you.

MEMBER BALLINGER: So some of the older data, the way out high fluence data that was at the wrong temperatures, is that--

MR. LEONARD: Well, I don't want to say it was at the wrong temperatures. There's a lot--going back through, whether it was for concrete or for metal, sometimes there's a lot of uncertainties associated with early test data. So you know, there's--we have to just be a little bit cautious.

MEMBER RICCARDELLA: And when you're talking temperature, that's irradiation temperature?

MR. LEONARD: Irradiation temperature, yes.

MEMBER BALLINGER: But we're confident that the downturn is real?

1 MR. LEONARD: That's one of the reasons why
2 we're doing some testing. Just confirmatory, but we do
3 know that, you know, so you can see a lot of the silicates
4 are starting to show that downturn, and that makes sense
5 based on even other data of silicon oxide from fusion
6 materials programs that's been known for quite some
7 time, so. Okay, so here's the same plot back with the
8 fluence data for 80-year lifetime. So again, some of
9 the same points to kind of point out is that we really
10 have to look at individual reactors based on what kind
11 of concrete aggregate structures they have, what is
12 their makeup, are they silicate--more silicate based,
13 more limestone based, and that's going to have some
14 effect. So while the fluence may go out to fairly high
15 values, there's still some of that work that has to be
16 done to really identify well what is the constituent
17 components in that material, and is that an admission.
18 MEMBER RICCARDELLA: And I assume this is
19 fluence right at the surface?
20 MR. LEONARD: Right at the surface,
21 correct.
22 MEMBER RICCARDELLA: So it attenuates?
23 MR. LEONARD: It attenuates, yes.
24 MEMBER SKILLMAN: Keith, let me ask this
25 question, and it's really a curiosity question. I

1 think most of us would know who made or designed four
2 loop plants and three loop plants, but there are two
3 vendors that provided two loop plants, and so there is
4 a family of very small two loopers, and a family of
5 fairly large power output two loopers. Does the span
6 of the green data have a bias of one of those particular
7 designs versus the other?
8 MS. BUFORD: So there is a bar, each plant
9 in the U.S. is represented by one of those bars; if you
10 go to one of the EPRI reports, there is actually a plant
11 name for each of those bars.
12 MEMBER SKILLMAN: For each bar.
13 MS. BUFORD: We redacted it to share this
14 in public.
15 MEMBER SKILLMAN: Okay. Fair enough.
16 Thank you.
17 MS. BUFORD: But every plant in the U.S.
18 has a bar.
19 MEMBER SKILLMAN: Okay, thank you.
20 MR. LEONARD: Okay, so this is neutron
21 attenuation and energy cutoff, so this is work that the
22 attenuation of neutron energy through the concrete
23 biological shield was cited by Eagle Remmick, and this
24 is for just a three loop pressurized water reactor. And
25 what you see is attenuation of about one order over 12

1 to 15 centimeters in drop. Eagle also did some
2 calculations on damage values associated with the
3 energy levels, and so this is--some care also has to be
4 taken into this account. So evaluating dpa or
5 displacements per atom, this was based on looking at the
6 simple crystalline forms of the concrete structure and
7 not--we can't really it as a whole, so it's kind of
8 taking it as the individual components. And so what was
9 shown was that about 95 percent of the total damage comes
10 in at neutron energies greater than 0.1 MEV.
11 So taking the data that we know from effects
12 of radiation on volumetric swelling, there's been some
13 preliminary buildup of modeling effects on the stress
14 and other components, the stress of the biological
15 shield I should say, and slowly developing that
16 components into something a little bit more mature. So
17 what this is, what you're seeing is the effect of
18 different radio--actual longitudinal stresses acting
19 throughout the thickness. So this does not take into
20 account rebar at this point. So this is just a
21 preliminary build up of what we're trying to develop.
22 MEMBER RICCARDELLA: You did the stress
23 analysis without putting rebar in place?
24 MR. LEONARD: At this--well, at this
25 point, there's--you'll see later there's some, as we

1 talk about some of the ASR, some of the ASR models are
2 now incorporating rebar and constraints, so.
3 MS. BERNHOFT: Okay, so that covers a lot
4 of the fundamental technical work that was done; I'm
5 going to talk about now some of the structural
6 evaluation that EPRI's going to do, as supported by the
7 research that's done by Oakridge National Labs. Before
8 I talk about that, one thing that's important to
9 understand is that looking at the Westinghouse design
10 plants, which are going to be at the higher end of that
11 distribution of curves that we showed you, there are
12 actually four different types of support for the reactor
13 pressure vessel. The type one is where the vessel rests
14 right on the concrete as a support structure; the type
15 two containments you can see there actually has a metal
16 frame, and the biological shield is not a support
17 structure. The type three is only used for the four
18 loop plants, and if you remember back to your fluence
19 curves, they're not going to have the high fluence that
20 the shield wall, and then the type four has a shield tank
21 that would provide some neutron shielding to the
22 biological shield wall. So there are 11 plants in the
23 United States that fit that type one category; that's
24 background information.
25 So what we've done is we've actually

1 brought in a contractor that's got a lot of experience
2 with concrete and with doing finite element structural
3 evaluations, and we've turned them loose; we have
4 typical drawings supplied by one 2-loop and one 3-loop
5 type one plant designs, and this shows you a picture of
6 the mesh that they're building right now to run those
7 structural calculations, and they're putting in the
8 effects from Keith's research, looking at kind of the
9 worst case of the worst case of the swellings that we
10 would get from the different types of aggregates and the
11 changes in mechanical properties, and we expect to
12 complete all of this research and publish a final
13 technical report that will go through the evaluation
14 like the four different type of structures, you know,
15 why we look at the type one and why the other ones are
16 not. So there's a technical basis to take the other
17 three out of the mix, and then for those type one plants,
18 like I said, we've got a 2-loop and a 3-loop plant that
19 we'll be doing this detailed finite element analysis
20 with.
21 And what we expect to--what we would like
22 to show on that is what is the margins, and if we need
23 to start bringing in the rebar at that point, we would
24 bring in the rebar. But we'll start with, you know,
25 what margin would there be based on the test results we

1 have right now.

2 MEMBER SKILLMAN: So the margin that we

3 speak of is simply the margin in the concrete

4 independent from the steel?

5 MS. BERNHOFT: Yes.

6 MEMBER SKILLMAN: Understand. Okay,

7 thank you.

8 MS. BERNHOFT: Just for completeness, I

9 wanted to bring up that there is an International

10 Committee on Irradiation of Concrete; we've mentioned

11 Dr. Rosseel here before, he chairs this committee.

12 They had a meeting just a few weeks ago, but it's an

13 international collaboration, an international exchange

14 on this topical area. Anything you wanted to add to

15 that, Keith? Okay. Next topic that we'll move on to

16 is--I'll introduce this and same as last time, I think

17 Keith will go into a lot of the technical work that

18 they're supporting right now through Oak Ridge National

19 Labs. There's a lot going on on ASR, and I think you'll

20 hear from the NRC speakers as well. This is a very

21 interesting topic, millions and millions of dollars of

22 research going into it right now. The team NRC, NRC

23 Research, DOE, EPRI, we all came together and we put

24 together this diagram to try and represent all the

25 different people that are working on it, and the goal

1 that we want to get out to it is understanding what the
2 structural significance is going to be from structures
3 that are experiencing ASR.
4 One of the--a very recent report we put out
5 that's shown at the box in the bottom, it just came out
6 last month, is that we wanted to understand based on the
7 Seabrook experience what really is the scope of ASR in
8 North America. Is it a large issue that we need to put
9 a lot of significant research in, or is it really as some
10 people would like to think a very isolated occurrence?
11 What we did is working with one of our vendors or one
12 of our contractors here, put together a very good report
13 that looked at publicly available information, mostly
14 from Department of Transportation; I think somebody had
15 posed a question earlier where we get data from; that
16 was one information source. The red circles are areas
17 that have reported ASR in North America; the blue
18 triangles on this curve are the power plants in North
19 America.
20 So the conclusion we can draw from it based
21 on publicly available information is that ASR does occur
22 fairly widespread, and it's not just in the United
23 States; it's been in Japan and it's been in Spain as
24 well. So it's an international phenomena, and it takes
25 a susceptible aggregate and it takes water or moisture.

MEMBER BLEY: Have there been any actual structural failures that you've tracked down associated with ASR?

MS. BERNHOFT: John, can you answer that question for us? Okay, so John--

MEMBER BLEY: We need it on the record for the transcript.

MR. SIMONS: John Simons, MPR Associates. I'm not aware of any structural failures due to ASR; we have looked at the literature and there have been structures that have been condemned due to ASR. When they have subsequently tested them, the performance was not degraded anywhere near the levels that they thought it was.

MEMBER BLEY: Okay thanks.

MS. BERNHOFT: Thank you, John. Don't go far. The other thing that we felt that was important when we saw these results was what tools can we put into the plants that are considering looking at their structures today for ASR. You know, everybody says well I have a spec, and I use my spec when I built my plant, but we wanted to go back and say you know, what is the sensitivity today with what we know about ASR versus some of those earlier specs and QA standards. So we did design a screening methodology for the plants to

1 start using that could be potentially--so they can look
2 at their potential susceptibility to ASR so they know
3 early and start taking appropriate actions.
4 We're also looking at possible NDE
5 techniques; we have two different projects with that.
6 The slabs you see in the picture are actually at our EPRI
7 NDE Center in Charlotte; we've built slabs with known
8 ASR, and you can visually see it when you see the dried
9 slabs there. We've invited several vendors to come in
10 with commercially available techniques for evaluating
11 those; not a lot of promise so far, and then we're also
12 working with one of the universities under a NEA project
13 to test non-linear UT for ASR as well. So again, it
14 still needs a lot of development.
15 Where we want to take all this is that we
16 need to work with all of the research partners, NRC, DOE
17 and come up with how we're going to develop an aging
18 management program in the ASR area, and Keith's going
19 to talk about some of the fundamental research that
20 they're doing to help support that.
21 MEMBER BLEY: Sherry, before you go on, it
22 looks like that report you pointed out to us two slides
23 ago is not available publicly.
24 MS. BERNHOFT: It's available publicly,
25 but for a fee.

1 MEMBER BLEY: $150,000. Okay.

2 MS. BERNHOFT: But talk to us afterwards;

3 we do have--we have MOUs and we can look at it--you know,

4 we'll have to have an NDA in place.

5 MEMBER BLEY: Okay, because the reference

6 doesn't help us if we can't see it.

7 MS. BERNHOFT: I understand. We--I'll

8 get you with Rich Tilley, who's there and he's the expert

9 on helping us get these things shared.

10 MEMBER BLEY: Thank you.

11 MEMBER SCHULTZ: Sherry, are the reports

12 that you referred to earlier in your presentation

13 publicly available?

14 MS. BERNHOFT: They're publicly

15 available; they may have a fee associated with them.

16 MEMBER SCHULTZ: They also may have a fee?

17 MS. BERNHOFT: Yes.

18 MEMBER SCHULTZ: And are they proprietary?

19 You mentioned that some information was redacted for

20 this presentation. Would he find proprietary

21 information or non-public information in those reports?

22 MS. BERNHOFT: You will, and that's why

23 we'd have you, if you need access to them, to work under

24 an NDA or a Non-Disclosure Agreement.

25 MEMBER SCHULTZ: Okay, so they're provided

1 for information, but they're not publicly available?
2 MS. BERNHOFT: That's correct. If you had
3 a need, we'll work with you on that.
4 MEMBER SCHULTZ: All right, thank you.
5 MR. LEONARD: Okay, thank you. So we're
6 going to talk a little bit about some of the modeling
7 work that's being done, particularly related to ASR and
8 some of the experimental work as well. So what you're
9 seeing here is just the simplified model of a kind of
10 square volume or rectangular volume that's injected to
11 sheer stress, and we're looking at it in terms of what
12 happens when there's no ASR versus ASR. With ASR, you
13 get a little bit of expansion in the center where we've
14 placed the affected material, and you can see the kind
15 of changes associated with the increasing sheer stress.
16 So that was a model that was based on not--of just
17 concrete itself. Now we're looking at developing this
18 further into different confinement levels or
19 geometries.
20 So parameters include being truncated,
21 being panels all associated with difference in
22 thickness and geometries. Boundary conditions, our
23 non-restrained versus restrained or partially
24 restrained. That's kind of a misspelling, it should be
25 ASR instead of AAR expansion percentages.

1 Reinforcement ratio percentages. So these are
2 starting to now build up some complexity in the models.
3 MEMBER BALLINGER: When you say
4 reinforcement ratio, what do you mean?
5 MR. LEONARD: So that I believe is just
6 based on, more on a volumetric--
7 MEMBER BALLINGER: So this is still not
8 rebar?
9 MR. LEONARD: It's assuming--I believe the
10 way in which they do it is assuming that there is
11 basically a structural confinement. So it may not be
12 as finally developed as individual rebar, but rather a
13 barrier to further expansion.
14 MEMBER RICCARDELLA: Is all this work just
15 analytical, or are you doing some parallel testing?
16 MR. LEONARD: So there is some
17 experimental work that is going on.
18 MEMBER RICCARDELLA: I was at a conference
19 at NIST back in August on it, and all the concrete
20 experts in the world seemed like they converged there,
21 and lots and lots of people are working on analytical
22 techniques, but there's almost a universal need for some
23 validation with testing.
24 MR. LEONARD: I agree. There has been
25 both in some of the proposals that have been our for the

1 CNR proposal, there's been a big focus on concrete.
2 There's also, I'll show you in a couple of slides, some
3 of the LWRS supported work going on at the University
4 of Tennessee in terms of ASR.
5 MEMBER RICCARDELLO: There was one very
6 interesting test in Canada at one of the universities,
7 I don't remember--I think Toronto maybe, but at any
8 rate, they made some concrete intentionally accelerated
9 the ASR in it, and then they ran the standard compression
10 tests and tensile tests and all that. And all the
11 properties degraded, and they ran a sheer wall test and
12 it exceeded the capacity of normal non-ASR concrete.
13 Kind of an interesting result.
14 MR. LEONARD: Yes, there are a number
15 of--I've seen a number of papers from different
16 locations or different universities, entities that have
17 been testing it, so--
18 MEMBER BROWN: Before you go on--
19 MEMBER RICCARDELLA: But I'd strongly
20 advise that you keep bench-marking your analyses
21 against real test data.
22 MR. LEONARD: Absolutely.
23 MEMBER BROWN: To enhance my understanding
24 of what ASR does, what the effect of it is in concrete
25 based on your pictures on slide 25, I guess I would

1 derive the ideas that the ASR reduces the even
2 distribution of sheer force from being distributed and
3 concentrates it more, because you damage certain parts
4 of the--I mean you reduce the cohesion of the concrete
5 in certain volumes?
6 MR. LEONARD: Right, right. So the
7 question also went in the slide that, you know, the next
8 slide that it was on when you start to reinforce that
9 expansion constraint, then you can start to see
10 additional effects of, or more pronounced failures or
11 reductions in strength. And so that kind of relates to
12 what the University of Tennessee is doing when they're
13 doing some advanced or accelerated ASR testing on some
14 large scale concrete blocks, and there's a slide coming
15 up on that.
16 MEMBER BROWN: How do they generate for the
17 test purposes the alkali silica reaction's internal--
18 MR. LEONARD: It's a starting mix on the
19 concrete, but also they have to accelerate it
20 through--basically the room that it's in is basically
21 a modified room, so you can hold it at a given
22 temperature, and--
23 [Cross talk]
24 MEMBER BROWN: Okay, I got it. Thank you.
25 MR. LEONARD: Okay, we'll go to the next

one. So one of the--so we're talking about different modeling, and essentially we're trying to bring together all these different effects, so what you're seeing here is a couple of moisture transport and heat transfer model in a simplified concrete structure. On the upper image, the image on the left is temperature and the image on the right is relative humidity, and the temperature distribution profile is shown for given conditions of heat and gamma heat. So this is tying in knowing the temperature profile, you're going to understand more effects of what kind of radiation induced swelling that you're going to have, because that's strongly derived by temperature in addition to obviously the dose, but the ASR effects are also going to be affected by temperature and relative humidity. So this is just developing these core models and further growing them into something a little bit more involved that takes into account a number of different components or factors.

MEMBER RICCARDELLA: Excuse me, I got distracted. What's the difference between the blue figure and the green?

MR. LEONARD: The blue one is temperature, the one on your right would be relative humidity.

MEMBER BALLINGER: I'm having trouble with

1 units. I can't see, I can't determine what the
2 dimensions, relative dimensions are here because--can
3 you give it to me in Seabrook units? In 80 years, how
4 much ASR are we going to have? What's the depth in
5 Seabrook units?
6 MR. LEONARD: I don't know if I can answer
7 that at this point.
8 MEMBER BALLINGER: I mean basically, this
9 is tongue in cheek, but you know, these concrete walls
10 are thick, and they're reinforced. So you get ASR on
11 the outside; how much of the wall is actually penetrated
12 in a long period of time?
13 MR. LEONARD: That's a good question.
14 MEMBER BALLINGER: Does it get to the
15 rebar, for example?
16 MR. LEONARD: Right, and that's a good
17 question, and I really don't know the answer at this
18 point, and that's another aspect of even
19 non-destructive examination. I mean, you can see
20 things on the surface, but underlying, what's
21 happening inside the material is also a difficult
22 assessment.
23 MEMBER BALLINGER: But we do have a data
24 point with Seabrook, and I'm sure they've taken core
25 samples, and we know how many years that is, so how far

1 has that penetrated? It's probably in a report
2 somewhere?
3 MR. LEONARD: I'd have to take an action
4 item on that.
5 MS. BERNHOFT: I think that would probably
6 have to be posed to Seabrook as the licensee; that's the
7 data that they've been developing. Or John can--
8 MR. SIMONS: John Simons, MPR Associates.
9 I've been supporting Seabrook on the ASR issue. The ASR
10 from looking at cores goes throughout the wall; it's not
11 a surface phenomena--
12 MEMBER BALLINGER: Throughout the wall?
13 MR. SIMONS: --yes, throughout the
14 thickness. It's not a surface phenomena like boric
15 acid attack or irradiation; it actually occurs
16 throughout the wall; the internal moisture can help
17 drive the reaction. That is what you see from the cores
18 that they've removed. The difference is what's the
19 level of expansion in that concrete, and what we've seen
20 at Seabrook, the implant expansion appears to have
21 plateaued around .1, .2 percent, something like that.
22 The through thickness expansion we estimate to be about
23 .3 to .5 percent, but we're working on that now.
24 MEMBER BALLINGER: Okay, thank you.
25 MR. LEONARD: So we kind of touched upon

1 this slide. So there's experimental mock up; this is
2 a large structure of blocks; one block is going to be
3 a non-reactive, one is a reactive with free expansion,
4 and the other is a reactive concrete mix to develop ASR
5 under constraints in one plane. So this is--we're
6 monitoring both internally what's happening in the
7 material, but we're also going to be able to utilize this
8 block for other NDE examinations. So kickoff meeting
9 was held this September, and they're developing the
10 concrete mix components and structure, what they're
11 going to do. This will be inside as I mentioned a basic
12 climate controlled room, we make relative humidity and
13 temperature for the work.
14 MEMBER BALLINGER: So back to the question
15 again. If it really is throughout, then now the rebar
16 becomes very important.
17 MR. LEONARD: That's true.
18 MEMBER BALLINGER: And how much of this
19 testing includes rebar?
20 MR. LEONARD: On this one, no. It's basic
21 constraint on the outside. So there's a box structure,
22 if you will, around it. So there's not an internal
23 rebar structure.
24 MEMBER RICCARDELLA: Are these going to be
25 structurally tested or just sampled?

1 MR. LEONARD: They can be. They will be
2 cut and then checked to see how much there's release of
3 expansion and so there's--
4 MEMBER RICCARDELLA: But you're not going
5 to load them?
6 MR. LEONARD: No, not--
7 MEMBER RICCARDELLA: They're doing a
8 similar program at NIST, and I'm sure you're aware of
9 that, and those do have rebar--
10 MR. LEONARD: They have rebar.
11 MEMBER RICCARDELLA: Those do have rebar,
12 but you know, and they're big blocks, like the size of
13 one of these tables, but I don't think they're going to
14 do any structural testing there, they're just going to
15 take samples and do their standard compression test and
16 things like that.
17 MS. BERNHOFT: John, you want to --
18 MR. SIMONS: John Simons, MPR Associates
19 again. The testing programs we've been doing for
20 Seabrook Station down at the University of Texas in
21 Austin, our specimens there 28-foot long beams, two foot
22 thick, some four foot thick. Those are all reinforced
23 similar to the plant; they have two-dimensional rebar
24 mats on both faces.
25 MEMBER RICCARDELLA: Two-by-two,

1 four-by-four?

2 MR. SIMONS: Most of them were about three

3 feet tall, two feet thick and 28 feet long, tested in

4 shear, outer shear and four point flexible test.

5 MEMBER BALLINGER: We are getting

6 prototypic data then? You're getting real data?

7 MR. SIMONS: Yes. Yes, we are.

8 MR. LEONARD: Okay, so as Sherry

9 mentioned, there's the ICIC Technical Committee that

10 involves the radiation effects on concrete; there's

11 also RILEM Technical Committee on internal swelling

12 reactions, and there's several working groups. Yum Le

13 Pape, who's part of the Oakridge team in the LWRS program

14 is involved in large scale testing. There's also other

15 working groups on NDE monitoring, advanced ASR models,

16 and estimation of residual expansion of concrete and

17 structures affected by ASR. So with that, we'll go to

18 the next subject now, hand off back to Sherry for boric

19 acid damage and concrete spent fuel pools.

20 MS. BERNHOFT: Okay. I just want to

21 make--and I'll go through these two topics fairly quick,

22 I promise. So first, concrete's been interesting so

23 far hasn't it? We've had some good dialogue on that.

24 One of the things I want to say on--

25 MEMBER SKILLMAN: Let me respond to that.

1 It is, except that in almost every area where we have
2 concerns about ASR, we've got tons of rebar, and so to
3 speak about one without the other seems to me to be not
4 delivering the entire punch. And I know if you do some
5 digging around in some of the major bridges in the
6 country, some of the major structures that are the
7 composite of 5,000 psi concrete plus rebar, there is
8 cracking everywhere, but those infrastructure items are
9 used daily by hundreds of thousands of people, and we
10 deem those structures to be good to go. So I just think
11 it's important to keep the discussion focused, because
12 what we're really talking about is this dynamite strong
13 bullet proof concrete steel matrix that's been
14 developed by thousands of very intelligent civil
15 engineers, structural engineers, steel engineers, and
16 we depend on it.
17 And so I fear that we get carried away, at
18 least in public space, that we have this chemical
19 reaction that's going to make the building dissolve, and
20 I think we've got to keep in balance that isn't going
21 to happen. These buildings are mighty strong.
22 MS. BERNHOFT: I'm glad you brought that
23 up, because exactly what I was going to say, I mean I
24 was thinking through my mind as I heard the conversation
25 to recap this, and that was a great introduction is--and

1 John knows this as well--you know, I started off with
2 there's a lot going on on ASR; we reacted to Seabrook
3 and that needed to be reacted to, and some of it was
4 understanding the fundamentals of what was going on with
5 it. But you are spot on, now that we've got that
6 information from an applied research standpoint, we've
7 got to bring in the common sense in how you do the
8 structural evaluation of it. And some people in this
9 room, one of them being John, knows I have a total
10 passion about this, and he's--when he talks about an
11 environmentally controlled room, it's a greenhouse in
12 Austin, Texas that was what 135 degrees when we were in
13 there in July, when you drug me through it? And he does
14 have big steam girders out there, and also a lot of work
15 going on with the Department of Transportation.
16 So I say all that to say I totally agree with
17 you; the point is now that we've got a lot of good
18 fundamental work going on, we've got a lot of really
19 smart people working on it, we need to take and we need
20 to apply the lessons that are coming out of the
21 Department of Transportation and other industries that
22 rely on major concrete infrastructures and put the
23 definition that we need around this. I totally agree
24 with you, and I've got a passion about it, so--
25 MEMBER SKILLMAN: Thank you.

1 MS. BERNHOFT: --we'll make that happen.
2 So that aside, I'll move through these next few
3 overheads fairly briefly. This actually started with
4 some operating experience in the United States where
5 they were picking up leakage of spent fuel pools, and
6 of course the big issue it starts is a groundwater
7 consideration issue. So it starts with protection of
8 groundwater, but it did show up as one of the gaps for
9 looking at long term operations, so we built on some of
10 that earlier research that was put out in the 2012, 2013
11 time frame; that's the next overhead. I'm going to show
12 you a couple of pictures; we're coming to the end of a
13 three-year study that we've done at CEA. We did quite
14 a bit of testing with different aggregate type solutions
15 and modeling of boric acid corrosion on the concrete so
16 that we could start developing some of those models, and
17 this was also done in collaboration with the Materials
18 Aging Institute as well.
19 Our goal is that bottom bullet is to come
20 out long term what is going to be the impact of the boric
21 acid, you know, how this corrosion rate of it, will that
22 be an impact and if it is, do we need to develop an aging
23 management type framework for that. What we've come
24 out of the testing with is that it is a--I'll
25 characterize it as saying it's a very slow corrosion

rate on the spent fuel pool, so you can see some of the plots from one of our earlier reports; we are working with NPR right now to update this report based on the recent testing that has come out of CEA, and by next year we'll have put out--we're going to put out our final report on the corrosion rates and what structurally that will have the significance on the spent fuel pool wall. John, anything to add to that?

MR. SIMONS: No, this is an issue that actually turned out to be a non-issue structurally, once we got to the bottom of it, and I think with the additional CEA data, it's just going to confirm that for different concrete mixes.

MS. BERNHOFT: So really then the concern becomes groundwater protection, so.

MEMBER SKILLMAN: And within that topic of groundwater intrusion, I would think that there are some permutations and combinations in terms of the chemistry of that water, some water being more aggressive and some being less aggressive. So if we talk about groundwater, we have to talk about what is the chemistry of that groundwater.

MS. BERNHOFT: Yes.

MEMBER SKILLMAN: Okay, thank you.

MS. BERNHOFT: Good. The last area, I

1 just have a few overheads on. This was also listed in
2 in the EMDA, and it's the possibility of a creep in the
3 containments structures. We started--we did a
4 literature survey, just recently published the results,
5 available to the public but I'm sure for a fee, but we
6 can work on that. What we did is we looked at what is
7 known about creep. What came out of that in summary is
8 that creep has a very non-linear effect, that's what's
9 shown on the curve down there in the takeaway. Most of
10 your creep is going to occur early in life, then it kind
11 of levelizes out. The important points to understand
12 about it is the creep and creep concept is well known,
13 and it's included in the design basis of the containment
14 structures. We did not find any areas where we're aware
15 of issues due to creep of containment structures causing
16 failures. There is a surveillance program existing
17 right now to go out and inspect for your tendons and
18 re-tension as needed, and we haven't seen any tendon
19 failures due to creep. There have been tendon
20 failures, but not due to creep. So that's the summaries
21 of those reports.
22 MEMBER RICCARDELLA: Is the concern creep
23 of the concrete, or creep of the tendons?
24 MS. BERNHOFT: It's creep of the concrete;
25 this is focused on the concrete right now. You can

1 re-tension the tendons, so--other ASME programs will
2 look at the tendons, so this study was focused strictly
3 on the concrete.
4 MEMBER RICCARDELLA: Are you sure you
5 tested it long enough? In steels you have primary
6 creep, secondary creep and tertiary creep; maybe you
7 haven't just gotten to the tertiary creep yet.
8 MS. BERNHOFT: That's a good point; we can
9 look at that. And then the last project that we are
10 working on too, this was with Georgia Tech, is is there
11 any non-linear type UT testing that we could do to look
12 at concrete creep. Results are slightly promising, but
13 a lot more work would need to be done, and that report
14 will be coming out soon. So with that, if you have your
15 side package, there's several references to some of
16 Keith's more technical work in here as well. So we will
17 turn it over to NRC.
18 MEMBER SKILLMAN: Well let's just hold
19 here for a minute. Colleagues, any questions for
20 Sherry or for Keith? Hearing none, then Steve, back to
21 you, please.
22 MR. BLOOM: Thank you.
23 MS. BUFORD: Okay, my name is Angie Buford,
24 I work for the Division of License Renewal in the Office
25 of Nuclear Reactor Regulation. To either side of me,

1 I've got Mita Sircar and Jake Phillip; they both work
2 in the Office of Nuclear Regulatory Research. We are
3 going to be presenting today on the aging management
4 of concrete for subsequent license renewal, and before
5 we get into the presentation, I did just want to make
6 a comment, because Dr. Skillman, you hit on a point
7 that's very important to us in NRR as we try to develop
8 guidance for reviewing subsequent license renewal
9 applications.
10 It's one thing to have the small scale
11 testing for ASR, irradiate concrete, other issues, but
12 we're really interested in, like Sherry said, the so
13 what, the structural context, and how the reinforcing
14 with the concrete, what is the significance of that; at
15 what point is there an issue with the structural
16 performance. And so a lot of the focus of our research
17 is geared toward helping us to develop guidance to
18 determine at what point do you have an issue with the
19 structures actually performing their functions. So
20 with that, I'll just go through the outline of our
21 presentation today. Firstly, I'm going to just be very
22 quick, I'm going to present the regulatory framework and
23 technical items that we use from a licensing standpoint,
24 then I'm going to go ahead and turn the presentation over
25 to Jake and Mita, and they're going to discuss some of

1 our research activities.

2 MEMBER SKILLMAN: Excuse me Angie, if I

3 could. We're due for a break--

4 MS. BUFORD: Oh, sure.

5 MEMBER SKILLMAN: --not right now, at

6 1430; at 2:30.

7 MS. BUFORD: Okay.

8 MEMBER SKILLMAN: And so I would ask the

9 presenters if in the handoff between the several of you,

10 you say, you know now is a good time, if you'd please

11 tell us that so we can let everybody go for a few minutes

12 at the right time, but only at the right time.. So we'll

13 be sensitive to your schedule, but if you find among your

14 presentations a good time to take a time out, tell us

15 that and we'll break at that point. But I don't want

16 to break in the middle of your work.

17 MS. BUFORD: Okay.

18 MEMBER SKILLMAN: Thank you.

19 MS. BUFORD: Okay, that works. So go on to

20 the next slide. There are no regulations that directly

21 apply to concrete, but there are a series of regulations

22 that indirectly apply to concrete, and then also some

23 technical items that I'm going to walk through. 10 CFR

24 50.55a is the regulation that requires in service

25 inspection of concrete containments. The other

1 regulation that we have that applies to aging management
2 of concrete containment is Appendix J, which is the
3 Reactor Containment Leakage Testing. The maintenance
4 rule is addressed in Reg Guide 1.160, and that requires
5 plants to implement a structures monitoring program to
6 manage aging of structures. Of course Appendix B for
7 quality assurance, and then Part 54 of the license rule,
8 where we incorporate the regulations to extra
9 activities that ensure aging management through the
10 first period of extended operation and then obviously
11 now the second 60 to 80-year period.
12 So there's significant guidance that we use
13 that applies to both nuclear structures and also
14 concrete for other industries. We are lucky that for
15 most of our aging effects, there is a significant wealth
16 of knowledge that is applicable to nuclear reactors, but
17 also that was obtained from different industries. So
18 for most aging effects, we're covered, and there were
19 a few that we'll get into probably after the break we've
20 talked about earlier, alkali silica reaction,
21 irradiation of concrete, where there's no real ACI codes
22 that we can go to directly to get guidance. This is just
23 a list here of the GALL Report recommended guidance
24 that's incorporated and recommended for applicants in
25 license renewal. That's really all I needed to say from

1 the regulatory standpoint, and this is actually a good
2 time to break; we'll get into the research when we come
3 back if that's--
4 MEMBER SKILLMAN: Angie, thank you.
5 Ladies and gentlemen, we are recessed for 15 minutes;
6 please return at 20 minutes to 3.
7 (Whereupon, the above-entitled matter went
8 off the record at 2:24 p.m. and resumed at 2:40 p.m.)
9 MEMBER SKILLMAN: Ladies and gentlemen, we
10 have a quorum. We're back in session and member, past
11 chairman, Harold Ray, would like to make a statement,
12 please?
13 MEMBER RAY: Yes. I was reflecting on the
14 -- from the discussion previously and thought may as
15 well interrupt here and say a couple of things.
16 I was reflecting during the break on things
17 we had talked about earlier in the session preceding and
18 I thought this would be a good time for me to make a
19 comment that's been on my mind.
20 We just -- we meaning the committee -- just
21 recently went through a quite thorough review of a
22 concrete cracking case in a safety-related structure at
23 a power plant in which aging phenomena was very central
24 to the discussion. And in that respect, and it was
25 unique. In that respect,

1 I agree with the comment I think you made,
2 Angie, about margin being a main theme of what you're
3 trying to assure in anything like this where there's an
4 on-going phenomena taking place, and you don't know
5 precisely what it may entail or how it may progress, and
6 so on. It wasn't something that's exactly what we're
7 talking about here, but close enough.
8 There are two things I want to say though
9 about our review of that case. One was the important
10 role that testing played in determining -- this was
11 reinforced concrete structure -- in reaching the
12 conclusion that we ultimately reached, which was that
13 indeed this was something that was being adequately
14 managed going forward. This is in the first 20 years,
15 not the second 20 years. So that part is consistent
16 with what we're talking about here except, as I say, the
17 licensee in that case did quite extensive testing to
18 demonstrate that what was being assumed wasn't merely
19 transferred, and this is my second point, from other
20 experience that we have with civil structures where the
21 ratio between the design basis and the operating loads
22 are much closer than it is in this case here.
23 How well we do under normal circumstances
24 is a big stretch from that to the design loading in a
25 nuclear plant, unlike in a civil bridge structure or

1 something. And so I see this issue that we're dealing
2 with here quite differently than I think was implied by
3 some of the comments about this. On the one hand, we
4 did conclude that this was a case that we could accept
5 and it was well within our requirements as far as
6 assuring safety and so on. But it was with a lot of
7 deliberation that we did so. And I just didn't feel
8 that was quite as emphatic in what was discussed earlier
9 as I thought it should be. The objective testing, the
10 assurance that we've got margin against design loading,
11 not just operating loads and so on.
12 MS. BUFORD: So with the situation that
13 you're referring that was the laminar cracking issue?
14 MEMBER RAY: Yes.
15 MS. BUFORD: Okay, so I think I believe
16 that you're referring to ASR issue at Seabrook or
17 potentially at other plants.
18 MEMBER RAY: It was Davis-Besse, but --
19 MS. BUFORD: No, I mean this --
20 MEMBER RAY: No, I'm talking about just the
21 general comments that were made about let's not
22 overreact to degradation that takes place in concrete.
23 Well, I agree with that. But let's not disregard it
24 either is the point.
25 MS. BUFORD: Right, so in the particular

1 case that we're looking at here, the plant that has
2 identified the ASR cracking went back to their
3 calculation and applied for every limit state that
4 applies to structures. So that would be shear stress,
5 capacity, compressive stress capacity, lap splice, so
6 anchorage to rebar.
7 MEMBER RAY: Lap splice is critical.
8 MS. BUFORD: That's critical. They were
9 able to go to the literature and apply what they could
10 find to be the largest -- for lack of a better word
11 penalty due to ASR. So the largest say reduction in
12 shear stress, it's shear capacity, that they could find
13 would say 35 percent. So they went ahead and applied
14 that 35 percent reduction to the shear calculations for
15 the most critical structures and were able to show that
16 there was still margin in the capacity to perform those
17 functions.
18 MEMBER RAY: Without question. I'm not
19 questioning Seabrook or any other licensee case here.
20 MS. BUFORD: So with the massive
21 structures that are massively reinforced like we've
22 discussed, ASR is a very slow process. It's something
23 at least to the extent that small-scale testing has been
24 done and then applying that to larger testing. We've
25 got confidence that at any -- at least at this juncture,

1 the structures have more than enough margin to perform
2 their function.
3 But our concern is at what point is there
4 an impact to the structural capacity. And that's why
5 that's where we are in terms of needing to understand
6 in terms of formulating our guidance. We need that
7 validation and right now it doesn't exist in the
8 literature in terms of large-scale structures and
9 physical testing.
10 MEMBER RAY: Well, again, I think we're
11 aligned here. I'm simply saying that there could be an
12 inference that maybe this is not something as
13 potentially challenging as at least it was for us
14 recently. And we were only talking about the 40 to 60
15 year and not 60 year and beyond.
16 So in the case of splices, for instance, of
17 course it's the crack gap that's critical. And that has
18 to be shown by some testing that's conducted as to what
19 is -- how much is too much. But in any event, I just
20 wanted to make that point that we're talking about
21 something which I don't believe is -- there aren't easy
22 analogs for how to resolve the questions that it
23 creates.
24 MS. BUFORD: Agreed.
25 MEMBER RAY: Okay.

1 MS. BUFORD: I wanted to make one more
2 point before I turn the presentation over to Mita. We
3 have determined, in terms of the immediate time frame
4 and also in receiving applications for subsequent
5 license renewal, that the aging management programs
6 that are informed by guidance that is on this slide and
7 also by the inspections that are done for the
8 requirements, we do believe that the programs that we
9 amplified for subsequent license renewal would be
10 adequate and that right now there's nothing preventing
11 -- there's -- any of these concrete issues would not
12 prevent us to be able to do a plant-specific review of
13 a subsequent license renewal application.
14 So someone used the word "show stoppers"
15 for concrete. There aren't any of those. But we'll
16 talk about the research that we're doing to confirm.
17 MS. SIRCAR: From research last year,
18 September, we briefed on the concrete related issues.
19 And today we are going to update on those.
20 In the report has identified four technical
21 areas where there are some more knowledge and updating
22 is needed. Alkali-silica reaction, irradiation effect
23 on concrete, creep and potential for creep-fracture in
24 the post-tensioned containment, and effects of
25 potential boric acid attack concrete and steel of spent

1 fuel pool.

2 The technical issues related to

3 alkali-silica reaction is what is the type of

4 degradation due to alkali-silica reaction in concrete.

5 And more importantly and significantly what is the

6 structural significance and what is long-term overall

7 impact. And finally, the programmatic aspects like how

8 to monitor that and how to manage that.

9 For that irradiation effect on concrete the

10 knowledge is less and it's dated 1978. There's not much

11 recent data available, so there is a need for

12 characterization of degradation for irradiation

13 effect; understanding the structural significance for

14 long-term operations; and again, the programmatic

15 aspects like how to manage and monitor those structures.

16 The third one is creep and creep-fracture

17 for potential in post-tensioned concrete containment.

18 Because of the creep, creep is a non-linear and it

19 flattens out with age. However, there are gradual loss

20 of pre-stress for different reason for the pre-stressed

21 concrete containment. So periodic re-tensioning

22 sometimes is required and that activates the primary

23 creep and also sometimes new and old material also gets

24 mixed up because of some repair or some work in the

25 containment. Creep-fracture interaction under loading

1 is another area where some more study is needed.
2 Spent fuel pool and boric acid attack
3 basically kinetics and the extent of the attack and how
4 that varies with the different composition of the
5 concrete.
6 Now the plan, how we address those
7 technical issues. In this slide, I'm going to mention
8 some of the significant activities within the United
9 States and abroad. Information from this -- all these
10 activities will be factored in NRC's research and the
11 initiatives are also going -- the research plans are
12 going to be presented in the next slides.
13 NRC is monitoring and collecting
14 information on industry and DOE research. NRC will do
15 mainly the confirmatory research. NRC is
16 participating in the OECD activities. NRC is
17 participating in ICIC International commenting on
18 irradiated concrete activities. NRC is also in touch
19 with IRSN for the different concrete degradation
20 research.
21 The plan to address the technical issues.
22 NRC and industry research initiative underway for ASR,
23 and irradiation mechanism. And NRC research has
24 received requests for further research on ASR, May 2012,
25 and irradiation effects, this year May. NRC is

1 monitoring the other two activities like creep and
2 creep-fracture in post-tensioned containment and boric
3 acid effect on concrete spent fuel pool.
4 MR. PHILIP: Okay, next slide. All right,
5 we had given a briefing to ACRS last year in September
6 '14 on this program at NIST. We also have a site visit
7 by ACRS members and the committee to look at -- and where
8 we're giving a briefing and then showing the
9 experimentation hall where the block samples were
10 assembled.
11 In June and July of last year, of this year,
12 sorry, we had a workshop which was organized through
13 OECD, CSNI, NEA which basically looked at structural
14 performance of concrete due to different degradation
15 mechanisms. And based on that workshop which was held
16 for two and a half days, we have conclusions and
17 recommendations and developed a second phase of that
18 work. The workshop of the first phase and now we're
19 going to have a second phase that has been approved by
20 OECD headquarters in Paris and that is something that
21 we're looking at once our budgets are confirmed as to
22 our participation in that.
23 On this project, we just basically are
24 addressing the structural performance of ASR-affected
25 concrete for dynamic and static loading through its

service life. Also includes SLR period.

We have been approached by several entities outside of NRC who will be interested in testing out some of their nondestructive testing procedures on these three well-characterized beams that are going to be constructed at NIST under program. So we've been approached by the University of Nebraska, Penn State, Bonn in Germany and also I had one of the researchers from EPRI who came and I took her to NIST and we had discussion because they were also interested in testing some of their nondestructive testing procedures because in this program we don't have that. And NIST will be developing protocols and procedures for all these outside entities to do their NDE testing at no cost to us. And they want to actually establish timing as to when the test will be done. It should not interfere with their work under our program so that is under process and being developed.

MEMBER BALLINGER: I guess I may be a little confused. I was talking to Pete. It says "Current structural capacity to resist static and dynamic loads." Are they going to test these specimens under load?

MR. PHILIP: Yes. And that's the next slide. There are six tasks to this projects. Three of

1 them are structural tasks and three of them are
2 materials. And that comes to the next slide. So this
3 is the description of the tasks and it pertains to
4 ASR-affected concrete.
5 So in the first task which is being done
6 right now in which three samples have already been --
7 the skeletons of three samples have been constructed,
8 we are going to assess the in-situ mechanical properties
9 of those of ASR-affected concrete.
10 The second task which is again a structural
11 task, we assess the bond and lap splice lengths of
12 reinforcing affected by concrete, of ASR-affected
13 concrete.
14 The third task you're going to look at the
15 seismic characteristics of ASR-affected concrete.
16 There are three materials tasks. One of
17 them is to estimate the degree of reaction due to the
18 ASR. We try to predict the future expansion and the
19 ultimate expansion due to ASR. And these are closely
20 connected to the structural tasks. For instance in
21 task 4 which is under progress right now, they are
22 developing the mixed designs for the blocks that will
23 be cast. And if you go to the next slide, it shows a
24 sample -- it gives you an idea of what these blocks look
25 like. It's a 16 foot long concrete block. It's 6 feet

high and about 3.5 feet wide. It's going to be divided into three sections and there were some comments here from the ACRS members on reinforcement. These are reinforced beams or blocks.

One site to the right is heavily reinforced, sort of similar to a nuclear reactor reinforcement. The one on the left, the one third on the left is moderately reinforced. In the center, there is no reinforcement. Basically, we want to look at the effects of ASR concrete expansion and the confinement that is afforded by different levels of reinforcement and in one case no reinforcement. So that's a figure -- three of these beams have been cast so far, but the concrete hasn't been poured.

Skeletons of reinforcement and all of the wires coming out from these blocks which can measure the strain, the reinforcement, the three dimensional expansion of the concrete, etcetera, have been constructed and these specimens will be poured in the next month or so. So that's the progress on this.

MEMBER RICCARDELLA: Excuse me, I think what Ron was asking about, if you go back, there's no plans to physically test these in a 3-point bend or anything like that?

MR. PHILIP: No, no, no.

1 MEMBER BALLINGER: It's not mechanical --
2 MEMBER RICCARDELLA: It is, but it's
3 mechanical testing of samples and it's a very
4 scientific, organized test program, but they're not
5 going to actually take this thing and bend it. They
6 couldn't. It would fail where there's no
7 reinforcement.
8 MR. PHILIP: And if you go to the next slide
9 is the status. Two of these skeletons of these block
10 specimens have been constructed so far with all its
11 wires that comes out from all of the strain gauges,
12 etcetera. The work on the third beam is almost
13 completed. So the specimens are ready to be poured.
14 The mix designs for these concrete
15 specimens -- and one thing I failed to mention was that
16 the three beams have different levels of reactivity.
17 So one beam will have highly reactive aggregates. The
18 second one will have moderately reactive aggregates.
19 The third one will have lower reactive aggregates. And
20 they have a quarry out in New Mexico which has high
21 reactive aggregates that they're going to use and based
22 on the proportion of aggregates they put in each of the
23 beams, they can tell which is heavily, moderately, and
24 lightly reactive.
25 The mix design has been pretty extensive.

1 It has taken a year and a half, actually, to get these
2 results because they had to do a lot of testing. Here
3 we are trying to accelerate the reaction over two years,
4 two and a half years. So it requires a lot of testing
5 and analysis to actually tell you what the right mix
6 design would be so that hopefully in these big beams they
7 can predict the reactions and the rate of the reactions
8 enough so that you can get enough data, because you want
9 to get expansions that on that period that's
10 accelerated.
11 And the acceleration is not just having
12 reactive aggregates. They're also going to spike up
13 the alkalinity of the cements, to some extent.
14 The duration of this project is from 2014
15 to 2018. Next slide.
16 MS. BUFORD: Jake, could I jump in? I
17 wanted to give a correction. When you asked were there
18 going to be any structural tests, the other --
19 MR. PHILIP: The second task. One of the tasks
20 --
21 MS. BUFORD: Yes, so there are going to be
22 load tests.
23 MR. PHILIP: On the second task.
24 MS. BUFORD: The set up that you saw was
25 just for task 1 that was looking at the mechanical --

1 MR. PHILIP: That one does not have any.
2 MEMBER BALLINGER: That's the thing.
3 When we were there, we saw the initial parts of that.
4 MR. PHILIP: Right. The second task does
5 have two-point loading. So they're looking at the bond
6 strength and the lap splice. So that does -- and that
7 particular part has been designed by NIST and we are in
8 the process of going over and commenting and discussing
9 with them the design that they have come up with.
10 MEMBER BALLINGER: Is there any
11 coordination between the NIST program and the other
12 industry programs?
13 MR. PHILIP: Our understanding about the
14 other industry -- well, we don't have any interaction
15 with the testing at NIST -- at the University of Texas.
16 But we get insights into what they are doing through our
17 folks, through our colleagues at NRR as to what's
18 happening there. But other than that, there's no
19 interaction or anything of that sort. We just keep it
20 completely separate.
21 MEMBER BALLINGER: Does that run the risk
22 of losing the potential for reinforcement -- or pardon
23 the pun -- of the two programs in other words having the
24 two add up to more than the sum of each?
25 MR. PHILIP: I think both of the approaches

are different. If you look at the UT program, it's very specific to Seabrook. And the testing is basically to say we've done this testing here and how does that apply to Seabrook and can we make that connection. And that's something that NRR is going to evaluate. As far as the research that we are doing here for ASR, under the research program and based on the user need was to look at it from a generic sense, not particularly looking at any plants.

MEMBER BALLINGER: I would just think that it would be a mutual interest because these are monster specimens and expensive.

MR. PHILIP: Our experiments are very heavily instrumented, quite different from the ones at the University of Texas. I guess that is why a lot of other entities do want to come and test our beams because they are well characterized and instrumented. So we can make that connection when we do NDE because you have all that good instrumentation, strain gauges, and three dimensional concrete expansion stuff and taking surface measurements and doing modeling in addition that we can make that connection.

MEMBER BALLINGER: Thank you.

MS. BUFORD: I'll add on to what Jake is saying. So those of us in NRR, who are looking at the

1 testing from a licensing standpoint specifically for
2 Seabrook and that's from a license renewal standpoint,
3 we have been engaged with the region, the regional
4 inspectors who physically go to the testing site to
5 understand the program and the results that are intended
6 to come from that.
7 We have brought insight into our
8 interactions with research in terms of what the NIST
9 testing program is going to look like, so we haven't
10 officially integrated in terms of collaborating in any
11 way with the University of Texas program, but we have
12 taken insights that we've gained from our inspections
13 and visits to their site and understanding of their
14 programs to make sure that our research is confirmatory
15 to what we're going to learn from other areas.
16 MEMBER BALLINGER: Thank you.
17 MR. PHILIP: Next slide. We have NRC
18 grant projects. Now these grant projects are sort of
19 unique in that they are innovative research which hasn't
20 been done before. They're awarded based on NRC -- NRC
21 actually gives out in the Federal Register notice as to
22 what are the areas that they would like applicants to
23 submit grants. And one of them was on ASR, so we -- in
24 fact, two of them were on ASR. So I reviewed one and
25 Mita reviewed one. And I'm going to speak to the one

1 that I reviewed.

2 And the grant, by itself, is not part of the

3 research program as such, it's something that we look

4 at. We do not actually manage the contract as such.

5 The grantee is free to do their innovative research

6 which is generally new. The outcomes may or may or may

7 not be useful for us, but in some cases it may be very

8 useful for us. It's not really part of the research

9 program.

10 So this program was a request from

11 Northwestern University. This was from Northwestern

12 University and the professor there, Gianluca --

13 Professor Gianluca Cusatis had a novel, new method. He

14 calls it LDPM which is Lattice Discrete Particle Model

15 approach. The concrete where he looks at this

16 heterogeneous mixture of concrete and when it's

17 subjected to load, he looks at the interaction and the

18 displacement of the different particles within the

19 concrete to formulate some type of a meso scale model

20 of concrete behavior that he can probably apply to macro

21 scale. Of course, he just started the project. We

22 thought it was innovative. He showed us in some of the

23 stuff that he included in his proposal that he has had

24 good relationships between some of the modeling he did

25 using this particular technique with what he has seen

1 in some compressor strengths of concrete samples and
2 also some structural strength testing of beams after
3 that good relationship. So he wants to use this
4 technique to look at the effects of ASR on concrete and
5 try to predict its behavior in the future using
6 stochastic methods, using -- including parameter
7 uncertainty and model uncertainty and trying to come up
8 with a technique that he can have a stochastic frame of
9 the prediction of concrete degradation as a function of
10 time. So that's something that we have accepted as a
11 very innovative project and which has just started a few
12 months ago.
13 MS. SIRCAR: The next grant, the second
14 grant is to the University of Colorado and it is
15 experimental and numerical investigation of ASR,
16 specifically in nuclear power plant type of structures.
17 The first objective is to understand the
18 degradation of the shear strength due to ASR and why is
19 it shear strength. Because most of the nuclear
20 structures are shear wall type of structures where shear
21 strength is very critical and specifically for seismic
22 and also for internal pressure.
23 Here stain degradation due to ASR is one of
24 the objectives, the second objective is having a future
25 expansion prediction for ASR. And the third objective

is to develop numerical simulation to predict structural performance. And the outcome is basically the shear strength predict ASR expansion and develop a numerical simulation platform for ASR degradation.

The status is mixed design is almost complete and they are trying to cast specimens in this year or early next year. The curing facility is also almost ready. Experimental setup is also ready and numerical simulation, the finite element analysis has been started. PI is Professor Saouma. He's the RILEM Committee chairman on the Task Committee for ASR. And he is using his work and coordinating with the RILEM Committee.

The main objective for RILEM Committee is the kinetics of the ASR-related degradation and the structural performance.

Now we'll move to the second topic which is irradiation effect on concrete. The main objective for this research is for the SLR, like 60 to 80 years estimate bounding fluence on the concrete structure which is close to the reactor pressure vessel. And next is what is the degradation and what is the structural significance how it can be managed and monitored.

The research is split into two phases. First, there is some amount of work has been done,

1 already performed and completed by industry and DOE.
2 And we'll do the confirmatory review for the estimation
3 of fluence and dose on concrete, understanding the
4 various technical and scientific basis for the
5 screening of irradiation threshold that is the limit for
6 60 to 80 years. And data compiled by DOE and EPRI we
7 already have seen the plots that has been -- now there
8 are more data points on the plot and we'll review those
9 kind of work done by industry.
10 And then we extend that knowledge to more
11 on structural side like examine the sufficiency of dose
12 data to account for the bound and that will also
13 incorporate the concrete variability, the structural
14 layout and the details. Also, there are general --
15 different design has different type of elements. We'll
16 look into that.
17 Irradiation environment also varies from
18 plant to plant. Then the combined effect of neutron,
19 gamma, temperature, and also if there is such in future
20 shows there is a correlation between
21 irradiation-assisted ASR, we'll go into that, too.
22 Then once we have some understanding about
23 the concrete degradation, we'll try to characterize and
24 screen the degradation in the viewpoint of structural
25 performance that is the effect on strength, stiffness.

1 As you mentioned, the interaction with the
2 reinforcement of bond via the lap, via the anchorage.
3 When the reactor pressure vessel is supported on the
4 concrete, a good amount of anchorage goes into the
5 concrete and the embedment. So if that interaction is
6 affected, meaning it's serious.
7 So the first phase of the work will start
8 soon, the contracting process is going on. We are
9 expecting the proposal middle of this month, middle of
10 next month. And the duration of this project is 2016
11 to 2018.
12 The next phase of the project is, as we've
13 said, all data we saw are all data compiled by Kevin on
14 his data. So, conduct confirmatory testing using
15 harvested concrete from the commissioning plant. For
16 that, a lot of coordination and information collection
17 is under way and we are coordinating with the plan that
18 that is possible for harvesting concrete. Once that
19 plant is in Spain, planning on testing and
20 characterizing, accelerated irradiation one way to
21 expose a specimen to estimated level of fluence for SLR,
22 that is 80 years because this plant has run for 36 years.
23 So if we want to see the effect for 80 years
24 then further irradiation would be needed for that type
25 of work. It's very expensive and for that type of work

1 a collaboration is one way to accomplish. We are
2 looking into that and there are discussions within U.S.
3 and also internationally.
4 Also planning for numerical simulation of
5 reactor pressure vessel support structure to verify and
6 validate the structural integrity and performance for
7 the intended function. From mid-field perspective,
8 micro-modeling in different scale atomic level, micro
9 level, meso level is being developed in DOE. Once those
10 works get maturated, from there that information
11 probably will be feeding to the more structural finite
12 element model, the degradation aspect of it. And then
13 we'll evaluate the structural integrity and the
14 performance for the intended purpose that is the load
15 guiding, the shielding, both type of performance.
16 MEMBER SKILLMAN: Mita, let me ask this
17 question about your first entry there. Confirmatory
18 testing using harvested specimens is that bullet points
19 to. And I can understand that a cubic foot or a boat
20 sample or a slice of concrete can be effectively
21 analyzed for ASR, extent of ASR, extent of cracking, but
22 it strikes me that that sample does not provide other
23 information that one would wish to have. For instance,
24 if I want to know what the status is of a cable that is
25 in tension, I need to somehow interrogate that cable

1 while it remains in tension, once I take it out of its
2 intentioned state, I no longer have a representative
3 sample. I have a piece of wire.
4 So what is lost in a discussion about
5 concrete when you harvest a sample and you analyze it?
6 What don't you get that you need?
7 MS. SIRCAR: So basically, the harvested
8 sample and especially the irradiation or the
9 accelerated irradiation specimens are small. So of
10 course, the degradation in that and the radiated effect
11 in the type of strength loss from the point of
12 compression, modulus, tension, fracture, this kind of
13 basic material property would be asserted. Those would
14 be evaluated through the models that DOE is working on,
15 the atomic level, micro level, meso level model and
16 those will be a benchmark.
17 Once that material degradation part is
18 understood, then that degradation model will be fed into
19 the final element structural model to access the
20 structural performance and in that case, all the
21 enforcement, the anchorage thing will also go there.
22 And the basic properties of the degradation also and not
23 only that, mechanical testing, also physical properties
24 test, and the micro structural change also will be in
25 that plan. And once that understanding is clear, we'll

be able to understand better how that interface between the steel and concrete is working together. So once that understanding and bench marking is established through small scale specimens, then it will be fed into the bigger structure.

And definitely, there will be the reliability and I will say we have to keep some margin because in this case we cannot -- the fluence estimation is also protected and also we cannot test a big piece of concrete which is under the idea that with all these components inside. So in this case, whatever decision we make, we have to have sufficient margin.

MEMBER SKILLMAN: What confidence do you have that the specimen that you're harvesting is a specimen for which you know its time and temperature, you know its NVT, you know its other exposure characteristics?

MS. SIRCAR: For the plant that we are now looking at, we have those data. This is a Westinghouse plant and they had gathered all those data. So they have the historical data and plant operation data. So at least for this plant, we hope we'll be able to gather those, but this will be again the question maybe that this is just one type of concrete. But then there are variability of concrete also would be factored in.

1 MEMBER SKILLMAN: Thank you.

2 MEMBER STETKAR: Mita, I'm way past my
3 knowledge level here. I don't know anything about
4 finite element analysis and I don't know much about
5 concrete. I know a little bit about steel.

6 What kind of confidence do you have that
7 this wonderful finite element model that you build out
8 of piece parts of perhaps irradiated information about
9 rebar separately from irradiated information about
10 concrete, separately from irradiated information about
11 steel liner, separately from irradiated about
12 reinforcing cables in tension, has any relevance to the
13 irradiated integrated system?

14 MS. SIRCAR: The integrated system is the
15 whole model.

16 MEMBER STETKAR: No, no. You're right.
17 You said it's a model. I can build a model of a ship.
18 It might sink.

19 MS. SIRCAR: But the material degradation
20 --

21 MEMBER SKILLMAN: Do you have confidence?

22 MS. SIRCAR: The material degradation is
23 developed separately through all these tests. It's a
24 degradation model, not the finite element model I'm
25 talking about. The degradation model is a separate

1 thing. That degradation model will be used as an input
2 because whatever material model we have for good
3 concrete, or new concrete, or existing concrete that's
4 not degraded, cannot be used for this purpose. This has
5 a degradation for irradiation effect. So what is that?
6 If we have the knowledge for that, what degradation can
7 do on concrete, that knowledge will be given as input
8 in the bigger model. This is only a piece of that.
9 MEMBER SKILLMAN: That's my whole point.
10 But maybe Pete can help can help because he speaks --
11 MEMBER RICCARDELLA: But the message from
12 the NIST conference in August was whatever you put all
13 this model together, you better benchmark it against
14 some real structural, physical tests. Don't trust just
15 the modeling, but --
16 MS. SIRCAR: That would be the plan
17 specific. This is a framework will be developed on this
18 general test. But when a plant specific issue that one
19 plant is in that threshold range, then that plant
20 specific concrete will be characterized for that model.
21 MR. PIRES: This is Jose Pires from the
22 Office of Research. You know you have a slightly
23 different situation depending whether it is ASR or if
24 it's irradiated concrete. If you have the ASR like was
25 discussed before, you affect a large volume of the

1 concrete. So a large volume of the concrete is
2 affected, so in the laboratory it's feasible to
3 reproduce the degradated conditions of the concrete.
4 So the NIST research project includes in task 3,
5 actually testing walls and a simulation of seismic
6 loads. So it's part of that task 3, if I understand,
7 we are actually discussing what should be the size of
8 those specimens and so forth.
9 At the University of Toronto, they have
10 already done that and as part of the agreement that the
11 research project that is the OECD, we are going to do
12 actually a round robin analysis of shear walls and that
13 type of flow. So it's part of the activity that are
14 going to be done and each member country that
15 participates on the CS&I project can participate on
16 those analyses, they'll have the test data to verify the
17 model.
18 The irradiated concrete is a little
19 different. The mechanisms are also different, but you
20 only are going to affect a few inches of the concrete,
21 the thickness of the concrete, so you don't have to have
22 all these understandings of the behavior. We also
23 don't anticipate that the rebars are going to be much
24 affected by the fluence flow that is going to be there.
25 It's going to mostly on the crystal of the concrete.

1 So the mechanism of the expansion is
2 different also than they are in the ASR. That's what
3 DOE is researching. What is the -- how does that
4 expression take place? In the meantime in the concrete
5 area, the models of the concrete interaction with the
6 rebar is when we put all together are very advanced.
7 This is not a new area of research. It's been going on
8 for easily 50 years. And actually, there was already
9 studies done in the '70s. People were studying high
10 temperature gas reactors and things like that for
11 concrete and the irradiation conditions.
12 Actually, the company -- EPRI is now eying
13 to do some of these studies. So it is --- but it's only
14 those five few inches at the beginning that we are
15 worried about. And so we have confidence that we have
16 models that at least can bound response.
17 The harvested tests what we are doing there
18 is trying to see, for example, if what the DOE is doing
19 in the laboratory for that type of concrete under those
20 conditions is actually close, within reason of what is
21 happening in the field. We may lose a little bit
22 because of the confining stress.
23 MEMBER SKILLMAN: Thank you, Jose. Thank
24 you. Mita, please continue.
25 MS. SIRCAR: The second phase of this

1 research time line is 2016 to 2020. The outcome of this
2 research, the result of the research will form the
3 technical basis for the regulatory guidance on
4 structural integrity performance under the loading
5 demand and the shielding function. The methods and the
6 process for monitoring and managing the aging
7 degradation are due to effects on irradiation.
8 The summary of our presentation today, the
9 resolution of the technical issues. So we have a
10 near-term plan for which will do plant-specific
11 evaluation and GALL Report and SLR related information
12 have been updated and new further evaluation has been
13 added for radiation effect on concrete and for ASR the
14 further evaluations have been updated.
15 Long-term research plan is out of all this
16 confirmatory research, whatever knowledge we'll gain
17 we'll put back in our standard review document and the
18 regulatory documents, the GALL Reports, the guides and
19 meanwhile, if we need an inter-knowledge, then we'll
20 develop some ISG, interim staff guidance.
21 So with that, I conclude our presentation
22 on concrete today. Any more questions?
23 MEMBER SKILLMAN: Thank you very much.
24 Colleagues, any questions for the concrete team from the
25 NRC?

1 Hearing none, thank you very much. Steve,
2 back to you.
3 MR. BLOOM: The schedule calls for a break
4 after they were done, but I know we're kind of running
5 behind. Do you want to go right into electrical?
6 That's up to you.
7 MEMBER SKILLMAN: I want to go right into
8 electrical. I want to go back to electrical or go into
9 electrical because Charlie Brown is still here.
10 MR. BLOOM: Oh, okay, well, then let's get
11 going and we'll turn it back over to industry, Sherry
12 and Keith.
13 MEMBER SKILLMAN: Please continue.
14 MS. BERNHOFT: Okay, we will continue. We
15 are the same team, but we changed our hats, so we can
16 now talk about cables for us.
17 How I want to start the presentation is
18 somewhat similar to what I do with the concrete is that
19 for your background and your knowledge, EPRI has been
20 working on electrical cable research for well over 20
21 years. And we have over 50 documents that are available
22 on epri.com. A lot of the work was driven out of need
23 based on some forensics testing, issues that were
24 identified early on in plant life with electrical cable,
25 cable performance.

1 The punch line out of all of that is if you
2 look at the take away box, what we learned from over 20
3 years of gathering cable data is that in power plants,
4 the cable systems are performing well. What we're
5 finding is yes, there are cable failures, but they're
6 usually attributed to installation issues, the use of
7 the wrong cable design and the wrong location, maybe
8 some local environmental type situations, but your
9 cable system installed properly in a benign environment
10 or a normal expected operating environment is operating
11 well. So we start with that basis.
12 We put together several reports that are
13 listed here to help the utility engineers develop their
14 cable inspection programs, the tools, the handbooks,
15 and some testing techniques that we're going to talk
16 about in a little more detail throughout this
17 presentation.
18 So to lay out the presentation and how we're
19 approaching the research path on cables, this was
20 actually put together by Keith's organization. I think
21 it's a great overhead to start the presentation with
22 because you need to understand what our cable stressors
23 are. You can see those, thermal, radiation, and water
24 that leads to the aging and the degradation that we're
25 seeing. You look at the changes that are happening in

1 polymer materials, understanding -- yes?
2 MEMBER BALLINGER: Sherry, could you go
3 back a slide, please? How come I'm missing that?
4 MEMBER SKILLMAN: We have a different
5 slide set.
6 MS. BERNHOFT: Yes, I reordered two
7 slides.
8 MEMBER BALLINGER: Where is the slide with
9 the little cable tights on it?
10 MS. BERNHOFT: That will be next slide.
11 MEMBER BALLINGER: Okay, I'll wait then.
12 MS. BERNHOFT: Sorry, I caught that over
13 the weekend and had those two switched around. It's
14 going to flow better, trust me.
15 MEMBER SKILLMAN: Since we kind of take a
16 break here, would you go back a slide?
17 MS. BERNHOFT: I'd be happy to.
18 MEMBER SKILLMAN: On the upper right hand
19 image, is that an EPRI recommended cooling system?
20 MS. BERNHOFT: No.
21 MEMBER SKILLMAN: All right, what are we
22 looking at? What I see there is a heavily loaded, or
23 heavily ladened --
24 MS. BERNHOFT: Cable tray?
25 MEMBER SKILLMAN: Cable tray, probably not

safety because they're not individually tied and it actually looks like something that is next to a block wall that might have been wet. So what I'm looking at there is a pretty dismal state of cables. Is that what you're telling us?

MS. BERNHOFT: I'm not trying to project that. It was probably open for inspections and that's why we grabbed the picture.

MEMBER SKILLMAN: But the bottom line is it's performing well in spite of it.

MS. BERNHOFT: You make a good point. That's not a good picture.

MEMBER STETKAR: It wasn't open for inspection. That tray has never been covered in its life.

MEMBER SKILLMAN: I'm expecting mouse droppings and rat droppings on top of that cable.

MS. BERNHOFT: Thank you for bringing that up. So with the flow, you look at the changes in the properties and then how we can assess the cable conditions. There are visual techniques. There are some NDE techniques and as Keith will talk about in his presentation, we continue -- part of the research for the cables is to really push for additional condition monitoring and NDE techniques to assess the condition

of the cables. And looking at changes over time, what's important is the ability to give the insights to determine what is the remaining useful life of the cable with an adequate margin of safety.

This is the overhead that you had talked about. When we talk about cable aging, before we go into the discussion on that, this is where we felt it was important that you draw the distinction between low voltage cables and medium voltage cables. What's shown on the overhead is the typical definition ranges for low versus medium voltage cables.

MEMBER BROWN: I guess my question is because I'm not as familiar with commercial plants as I am Navy plants, but you show the low voltage ones as all be XLPE which is -- is that even the case for the old, old plants for 450 to 600 volts cables that were typically run? I'm only questioning the type of cables that you're using as a representation or that should be evaluated for the subsequent life renewal in that XLPE is relatively modern relative to the stuff, PVC cables that were installed 60 years ago.

MS. BERNHOFT: So to back up on how the cables that are included in the testing program right now for the cable types that we have a lot of our data for, there are five major cable vendors in the United

1 States. And so what we've done is within the five major
2 cable vendors, cable types, that represents about 90
3 percent of the cables that are installed in the United
4 States and we have developed our test matrix that Keith
5 will be talking about for any of the accelerated testing
6 to make sure that we encompass those five major types
7 and vendors that are in the United States right now. So
8 this is probably a typical picture from one of those to
9 show --
10 MEMBER STETKAR: Sherry, when you say 90
11 percent of the cables, do you mean safety-related
12 cables?
13 MS. BERNHOFT: Yes. And installed in the
14 United States.
15 MEMEMBER STETKAR: Thank you for that
16 clarification.
17 MS. BERNHOFT: Safety-related cables that
18 are installed in plants in the United States. And then
19 with that, I'll go on to say that because we're an
20 international organization, we're also working -- not
21 important to this organization, we are working to
22 broaden our international understanding of cable types,
23 too. It is very much country dependent is what we're
24 finding with a few exceptions.
25 MEMBER BROWN: Well, I just emphasize that

1 it talks about fire resistance, flame resistance,
2 etcetera, in here. And if you go back in the early --
3 if you go back to the 1960s, the cables weren't
4 necessarily flame resistant. In fact, they ignited
5 rather easily if they were exposed to flames, based on
6 witnessing numerous fire tests at the Naval Research Lab
7 in Annapolis.
8 So I'm just trying to emphasize that if
9 you're going to go test cables, and try to provide some
10 evaluation of their utility going out from 60 to 80
11 years, I don't know why we didn't think of it for 40 to
12 60, but that's besides the point right now. You ought
13 to make sure you include cables that are in plants that
14 are going to be extended today which are probably not
15 even being made these days in many circumstances.
16 MS. BERNHOFT: Right. And you'll see part
17 of what we're doing that's one of the reasons that we
18 are working on -- we're working with a lot of field
19 return to harvested cables also in the testing program
20 to capture some of those other things. And what we
21 actually work to do, and I'll get to the harvesting, we
22 have a publication, actually put out on harvesting so
23 that we get to the consistency of it. But what we try
24 and do when we harvest is we also talk to the plant and
25 we see if they have warehouse cables from that original

1 lot, that original batch, that goes back to the type of
2 time frame we're talking about. And then we also take
3 the cable that's been installed in the field so that we
4 have the two of them to put into the test plan.
5 And a lot of our research right now that
6 we're building on for the condition, qualified
7 condition of the cables is based off of some of the
8 earlier Brookhaven data that was done to develop some
9 of the EQ rules and equations that we live with right
10 now. So we've actually gone back through and mined a
11 lot of that earlier Brookhaven data to build into our
12 understanding of the whole test matrix. But your point
13 is well taken to make sure that we're complete and we
14 have all generation.
15 MEMBER BROWN: The only reason for the
16 comment.
17 MS. BERNHOFT: Yes, it's a valid point.
18 We need to make sure the population is covered.
19 MEMBER BROWN: Good. Thank you.
20 MS. BERNHOFT: Thank you. So understand,
21 again, there's low voltage versus medium voltage and the
22 reason this is important is on to the next overhead is
23 that when we talk about aging, there are different
24 drivers or stressors for the aging. So for example with
25 the low voltage cable, you have less of a polymer

1 insulation. So our major areas of concern there are
2 going to be the temperature and the radiation exposure.
3 And that's an ambient temperature and there's a
4 cumulative radiation exposure.
5 With the medium voltage cables, much
6 thicker insulation materials, but what we see is the
7 primary failure mechanism there is the water treeing and
8 that's due to the submerged or the wet and dry type of
9 environment.
10 So what we've done for an EPRI standpoint
11 is through the research and through the publications is
12 put out guidelines for the plants as a strategy for aging
13 management of their cable systems. And it has these
14 basic elements here, these five basic elements. And to
15 summarize those, the first thing you really need to do
16 is you need to identify where your adverse environments
17 are in your plant, the areas where you're going to be
18 susceptible for cable failures, temperature radiation,
19 exposure to steam, any -- and you give special
20 consideration to your critical components and your
21 safety considerations.
22 You supplement that with a walk-down.
23 You're looking for any type of visible indications.
24 That is added on to as some type of a Tan Delta or
25 withstand testing to test for -- or insulation

1 resistance breakdown for your low voltage cables.
2 You're looking for your areas of some of your worst
3 degradation followed up with some local specific NDE
4 things like an indenter, UT, Lira, any type of an NDE
5 technique. And then of course, repair or replacement
6 based on that and extent of condition is very important.
7 What's not shown on this overhead are very
8 two important aspects of this. One is that with
9 operating experience right now, the plants are not
10 waiting to Year 40 to implement these cable aging
11 programs. The plants are implementing these cable
12 aging programs and these EPRI guidelines today. So
13 we're starting to collect data on those right now.
14 The other thing is through EPRI, we have
15 developed a cable users group and with that cable users
16 group, we do share operating experience and we invite
17 NRC research attends that meeting, INPO attends that
18 meeting, the major cable vendors attend that meeting as
19 well. And the first day of that meeting is to go through
20 that operating experience. I know Darryl is going to
21 come up and talk after us and he's attended probably the
22 last couple of years of those meetings as well.
23 MEMBER SKILLMAN: Sherry, when you say
24 that the plants are not waiting, should we interpret
25 that to mean that the owners are changing cable or they

are, if you will, substituting, they're just saying we're going to retire that cable set and we're going to run a whole new set?

MS. BERNHOFT: I think that would be condition dependent. If they see a cable degradation say with like a medium voltage cable, they can either replace that section of cable or they can replace that entire cable. But they are taking corrective action based on what their finding from the walk-down inspections.

MEMBER SKILLMAN: Thank you.

MS. BERNHOFT: The other important point that we want to talk about, as Keith is going to talk about, we do want to continue to try and improve the techniques available to the licensees for doing condition monitoring, but Tan Delta is a very important test that we have out there right now. And what's important really is kind of those last couple of points that I have on here is we have been collecting the results of using the Tan Delta tests now for over six years. We went through in June or in 2012, we did a validation that Tan Delta is a successful technique to tell us when we're finding localized degradation and breakdown in the cables.

And we recently went through this year and

1 completed another test that helps us understand that it
2 does -- it is effective in identifying degraded
3 conditions in localized conditions and you can confirm
4 your results and your degradation mechanisms using your
5 Tan Delta and importantly, what we've demonstrated with
6 a report that we just put out is that we didn't see cable
7 -- if you had a good Tan Delta reading, the other part
8 of the equation was true, too. If your reading was
9 good, we didn't see cable failures. So we do have a
10 technique that can be used for cable, localized cable
11 degradation and testing.
12 MEMBER BROWN: Is that report -- is that
13 available or is that another one of the $150,000
14 reports?
15 MS. BERNHOFT: It would be a fee for that
16 report. Our members pay for our research, so --
17 MEMBER BROWN: Is the ability, like they
18 did on the last one, that Dennis brought up that that
19 would be made available to the committee?
20 MS. BERNHOFT: Yes, what we can do is after
21 this, when I get with Kent, that's how we've done it
22 before, and Kent has worked with us before. Kent has
23 been able to get reports through us before. So what
24 we'll need to do is work through Kent to identify what
25 that need is that you would have and how we would start

setting up --

MEMBER BROWN: We just need it to look at if we're going to be evaluating this part of the overall SLR evaluation.

MS. BERNHOFT: Okay.

MEMBER BROWN: Cables are a critical part of that and we're going to have to address this next year and then probably the year after that.

MS. BERNHOFT: Absolutely. I don't disagree at all.

MEMBER BROWN: If I'm not alive at that point, it won't make any difference. You'll have to deal with the other.

MS. BERNHOFT: That's not what we want either. So --

MEMBER BROWN: Neither do I.

MS. BERNHOFT: Yes. So what we'll do is I'm just going to ask that you coordinate it through Kent and we'll work with Kent.

MEMBER BROWN: Thank you.

MS. BERNHOFT: And make that happen. Okay, so moving along. Another project, if you go back to what I talked about to the low voltage cable conditions that the stressors we looked at were to temperature and the radiation. What we wanted to do is

1 -- we have two phases to this project. Phase 1 is
2 complete and Phase 2 is well into execution right now.
3 We want to go through and look at actual
4 conditions in operating plants so we would understand
5 what the ambient temperature conditions would be for the
6 low voltage cable systems and we also want to understand
7 what the cumulative radiation would be out to 80 years
8 of operation so we can start seeing where they would
9 start fitting into the aging and where on the curves
10 those would be.
11 We've completed the first study. We were
12 able to gather data from 14 plants, mostly temperature
13 data, some radiation data. While it wasn't as robust
14 of a data collection effort as we wanted, we did find
15 that we would project from those 14 plants that you'd
16 have between 1 and 10 megarads of radiation exposure
17 over about an 80-year lifetime.
18 What we've done now for Phase 2 is we have
19 one plant, the PWR, which is going to go with a little
20 bit higher exposure and we have installed -- we just
21 completed installing 55 radiation temperature monitors
22 in their plant, around their critical cable systems in
23 the containment. And we'll collect data for an
24 operating cycle. So in 18 months, we'll have that data
25 to analyze and work on to further build into the case

1 of what will be the temperature and the radiation
2 exposure over the operating life of the cable system.
3 As I talked about too, another important
4 part of our project is we do want to have harvesting of
5 cables, similar to like the concrete. It starts giving
6 you something to validate on your points with the
7 accelerated testing. We'll be -- we're working with
8 NRC, DOE, and EPRI. We'll be harvesting cables here
9 probably in January from the Crystal River plant.
10 Keith has had a successful project working with NRC and
11 has harvested quite a few cables out of Zion.
12 We're working internationally to harvest
13 cables as well. We're talking to any opportunity that
14 we have, even San Onofre right now we've just started
15 talking with them, too, to get some cable harvesting.
16 To get some consistency, I think the question was posed
17 of how you would characterize the environment that you
18 remove those cables from and it is important to have that
19 history of the cable.
20 So we put out a publication, this
21 harvesting guideline last year, to help with the plants
22 that are going into harvesting. For any reason they're
23 going to be abandoning cables, we've put out the plea,
24 we'd like those cables. But with that, we need them to
25 capture all the environmental data from where that cable

1 came from, what its operating history has been and if
2 there's cable in the warehouse that's not been installed
3 in the plant that's from that same batch or lot that we
4 could have both of those cable samples to put into our
5 testing program. So that's a consistency in this
6 guideline.
7 With that, I will turn it over to Keith.
8 MR. LEONARD: Thank you, Sherry. So --
9 MEMBER BROWN: Before you go on, let me ask
10 one other question related to the set of slides there.
11 MS. BERNHOFT: Okay.
12 MEMBER BROWN: You talk about in your
13 earlier slide you mentioned three tests, the Tan Delta,
14 the withstand test, and an insulation resistance
15 testing.
16 MS. BERNHOFT: Yes.
17 MEMBER BROWN: When you talk about
18 withstand testing, I don't know what the definition is
19 in your world. All I can do is do is talk about it from
20 where I came from. That could be a damaging test. We
21 used to follow the "hipot" test. And you could damage
22 the cables under that, so we were reluctant to do that
23 in many circumstances. The insulation resistance test
24 on the -- how do I phrase this? We've only got -- the
25 Navy didn't use the higher volt, like 46 kV cables. The

1 most we've gone to is 15 kV on the latest aircraft
2 carriers.
3 And we struggled with the idea of how you
4 do insulation resistance tests on these because it's not
5 easy on an aircraft carrier to do that and we were not
6 using high voltage, what I call 15 kV or 20 kV mega ring
7 type instruments. I'm not even sure they were
8 available, but we didn't have to do it. So our testing
9 was done at lower voltages for that and we were always
10 questioning the validity of using the 5 kV tester for
11 that.
12 MEMBER STETKAR: Under all costs.
13 MEMBER BROWN: Under all costs. Exactly
14 right. So I mean I guess my question on the insulation
15 resistance and the withstand testing, I'm not so sure
16 how good the withstand testing -- you're liable to
17 damage some stuff, so I would be leery of that, but the
18 insulation resistance, whatever you're going to use, if
19 it's 26 kV, whatever the standard ratings are, a 5 kV
20 test is not a very robust test for determining anything.
21 I'm only questioning how you're going to do
22 that and what the test methods would be and what the
23 voltage levels would be for those testing. I don't know
24 if you're past that part of it, but at some place that's
25 got to be identified.

1 MS. BERNHOFT: Okay, that's a good
2 comment. I wouldn't do a very good job answering it for
3 you right now.
4 MEMBER BROWN: Okay, well, I just throw
5 that one out for information.
6 MS. BERNHOFT: Okay, good. Thank you.
7 MEMBER BROWN: For future discussion.
8 MEMBER SKILLMAN: Sherry, I'd like to make
9 a comment about cable harvesting.
10 MS. BERNHOFT: Yes.
11 MEMBER SKILLMAN: The experience I've had
12 is once a plant announces it's shut down, its warehouse
13 becomes a cheap place for people to pick up stuff,
14 including cable because cable is valuable.
15 MS. BERNHOFT: yes.
16 MEMBER SKILLMAN: And so for what could be
17 the more recent announcements of potential plant
18 closings it may be prudent to contact those owners and
19 say before you empty your warehouse, if you're really
20 going to shut the plant down, please give us a chance
21 to work with you on harvesting because once that cable
22 is out of the warehouse, it's gone.
23 MS. BERNHOFT: Yes. You make a very valid
24 comment. We got caught in that loop at Crystal River.
25 While we were talking to one organization, another

organization was out auctioning.

MEMBER BROWN: You made my point. Thank you.

MR. LEONARD: Okay, so DOE/LWRS research and electrical cables, this bullet point, some of the different topics that address gaps in the expanded materials degradation assessment. Research coordination and collaboration team established in 2013. And some of the initial work DOE/LWRS milestone reports and peer reviewed publications are becoming available on a variety of these subjects.

So just to kind of touch base here a little bit on some of the variable, multi-variable environmental effects on cable insulation you have environmental stressors, radiation (UV, gamma), thermal effects, mechanical and moisture and this all feeds into changes associated with the chemical structure of the material, chain scission, chain cross-linking, actually depending on the material type, depending on the temperature, you can actually get both mechanisms depending on the temperature for a given material type. But all of these really factor into property changes, mechanical, electrical, physical.

So this just outlines some of the fine time line going towards the conclusion of work. So we're

1 kind of finished up with the scoping study and we're
2 getting involved in a lot of the basic studies on thermal
3 effects, radiation effects, and the synergy between the
4 two. And all of this is going to eventually feed into
5 EPRI's development of a qualified condition, aging
6 curves, and acceptance criteria for cables.
7 So we kind of touched a little bit upon this
8 and I think that Member Brown's comments are valid. So
9 this -- we can kind of look at this a little bit further,
10 but I think some of this may be have been -- may have
11 been pulled from an IAEA report. And pretty much in
12 cables in U.S. plants, 36 percent of cables are XLPE,
13 36 are ethylene-propylene rubber. There's also
14 jacketing material, chlorosulfonated polyehtylenes,
15 neoprene, and PVC.
16 So our focus has primarily been on XLPE and
17 EPR and CSPE, the jacketing material which is the
18 Hypalon.
19 MEMBER BROWN: Excuse me again, but I mean
20 if your primary focus is on -- can you hear me? Sorry
21 about that.
22 If your primary focus is on XLPE and EPR,
23 it seems to me that that's outside the range of what you
24 really need to address. If you want to address plants
25 that are going to be extending -- from the first plants

that are going to be extending 6 to 80, because they're down in the lower categories of silicon rubber, PVC. I'm not even sure of the -- maybe neoprene. I don't -- the chlorosulfonated stuff, I don't remember any of that.

I'm not experienced with the commercial stuff. If you go back 60 years, I don't think that was used back then. I just think those were newer cable types.

MR. LEONARD: Okay.

MEMBER BROWN: I just don't think it's very useful for the old plants if you don't evaluate the actual cable materials that are in place.

MR. LEONARD: Sure.

MS. BERNHOFT: We'll confirm that.

MEMBER BROWN: Some of these guys that have actual plant experience can tell me if I'm wrong, but all I can do is deal with what my memory tells me.

MR. LEONARD: Okay. So again getting into some of the research work and we'll touch base on some of the experimental work that's ongoing. And this is kind of a coordinated approach with DOE and the various DOE labs and EPRI. And it's really to do a number of things. One is to evaluate the fundamentals associated with the changes that are occurring in the material.

1 And also involved in some of the NDE techniques and
2 technology development.
3 So this slide is kind of important and it
4 goes back to some of our discussion here and that is in
5 addition to some of the materials that are more modern
6 day, we are trying and actively obtaining materials that
7 were utilized in the reactors so that we do have some
8 control rod cable, and that cable, I'll just refer to
9 my notes, was placed in -- was ordered in 1979 and placed
10 in service in 1984. It was removed from service in 2014
11 during a condition monitoring cycle. So we do have this
12 material that's available and that's also being tested
13 for synergistic effects, thermal, and aging studies,
14 radiation data.
15 We also have some San Onofre power cable and
16 this was stored onsite since 2006 and not placed into
17 operation so we also have that available. Now that's
18 a little bit more modern time line, but we are trying
19 to actively look at -- or comparing some of these tests
20 with cables that have been harvested.
21 So this next slide goes back to the Callaway
22 control rod data and this is aging in air at 125 degrees
23 C. So this is a comparison with some literature
24 published data and really the take away is that there's
25 good agreement in the data between the aged material

that was from the Calloway and the literature data, that indicates a little evidence of aging since 1984 installation. Additional verification of the Gillen 2007 and Calloway is still underway.

The next few slides are kind of just a recap on what are the type of testing done currently as we speak a PNNL and ORNL. So this is just a sample of PNNL set up for the gamma radiation testing. You can see that that's the highest source -- inside the source cone, the highest dose samples that are set up there for testing. On the image on the right, you have adjacent to that basically a low temperature aging oven. And each one of those rows of cables is a distinct dose rate as you kind of separate it further from the -- away from the source.

So ORNL system is a little bit different. We rely on a -- we have a similar cobalt 60 irradiator. It's just a little bit different design. It's a drop-down chamber. It does have some flexibility in how we can set things up. It's been used before for different companies that have done in situ radiation testing, looking at gas evolution and so on and so forth. But it's a low-dose radiation experiment or capability that's available to us.

The HFIR-Gamma irradiation facility which

is shown on the bottom is basically a higher flux irradiation capable system and this utilizes the spent fuel portion of the HFIR reactor. So depending on the age of the spent fuel core is going to give you what you can tailor roughly what kind of dose rates that you want.

There's also a gamma profile within the depth or the actual length of the spent fuel core, so you can also further make some adjustments. Further on the lower right are sample holders that were designed. It was actually based on work for NEET project, nano-composite polymers, but it's been equally applied to the current work.

And the center hole that you see can be fitted for a rod heater. So for the gamma radiation facility, we can drop these down in the spent fuel core pool. We have umbilical that goes down. We can pass different types of sweep gas through there. We can also have heaters set up along with thermal couplers set up. So we can do the combination thermal and radiation conditions.

Another work that has begun is looking at diffusion limited oxidation and this is just identifying nucleation of degradation, the effects of sample geometry, what happens when you build up a rind of material that has definite oxidation effects. Does

that affect the internal material that isn't influenced?

Microstructure analysis and mechanical property analysis, so there's -- these are just a variety of techniques that are available to assess some of the damage mechanisms associated with the polymer radiations or thermal radiations that are being done.

So I'm going to hand this back to Sherry to talk about some of the submergence research at EPRI and NRC.

MS. BERNHOFT: Thank you. One thing I want you to notice, we've started each of these kind of topical areas you've seen that we've had -- what we're calling a road map here and these road maps have been worked on between EPRI, NRC research and you can see that there is test labs that are all funded by the DOE/LWRS program.

The reason I wanted you to note that is that we do have very definitive time lines for the cable research. When you have to go back through and look at the slide package, I'd ask you to look at these various road maps that we've developed and you can see that the milestones have been highlighted on there to help understand when some of those milestones will be accomplished.

1 A lot of times you go back to the goal is
2 in the 2018 to 2019 time frame to have completed
3 substantially those milestones and that's what we show
4 on these overheads right now. So I just want to bring
5 that up to your attention.
6 Talking about submergence in particular,
7 as I said before, submergence has been a topic that we've
8 been dealing with for about 20 years. You have
9 underground cable vaults. You have a humidity
10 environment. You have wetting of the cable insulation
11 material and drying. And this starts causing a
12 breakdown in the insulation materials. So a couple of
13 things in the degradation phenomena you see is what's
14 shown here in these pictures and that's a water tree
15 where you just slowly breakdown.
16 You need two things -- you need three things
17 actually for water tree to happen. You do need the high
18 voltage, the higher voltage. That's why we have not
19 seen it yet in the low voltage cables. We only see it
20 in the medium voltage cables. You need the wetting or
21 the high humidity environment. And you need a
22 susceptible polymer material. So these are the type of
23 conditions that we see that is causing this water tree
24 environment.
25 We've done quite a few publications on the

1 water tree. The most recent one came out August of this
2 year, pulling together about 20 years of data on the
3 mechanism. And what we wanted to prove -- or discuss
4 in this report is the validation of how you use the Tan
5 Delta testing to help you locate the water tree. Again,
6 it's not the entire cable. It's usually very much
7 location dependent on the cable. So this report does
8 show you that the Tan Delta will help you find localized
9 areas of the insulation breakdown so that you can effect
10 a repair on the cables with that.
11 The next area that we are looking at for
12 submergence is -- there are actually a couple cable
13 types that based on operating experience and that's the
14 brown ethylene EPR and the pink EPR that -- what we have
15 observed with insulation or operating experience to
16 date is we have not seen them failing in service. So
17 to help support the members, we are doing some
18 accelerated testing at EPRI right now on these two
19 insulation types to see if they are candidates for a
20 replacement-type material into the future.
21 Another research project that we're
22 working with submergence is the low voltage cables, not
23 necessarily that we would expect the water treeing type
24 breakdown that you'd see in the insulation materials,
25 but what we're doing is we are doing some accelerated

1 testing in a wetted environment is to see the long-term
2 stability of the polymer under a submergence
3 environment. So far we haven't seen any breakdown, but
4 we'll continue the testing on those.
5 MR. LEONARD: So again, another road map
6 that's available for review. This is assessment of
7 condition monitoring elements for electrical cables.
8 So we kind of begin with materials testing and aging,
9 setting up, determining key indicators for cable aging.
10 Are there new NDE techniques or older, established
11 techniques that need to be further developed in order
12 to get us to understanding or being able to predict
13 remaining use for life.
14 Nondestructive evaluation of cable
15 remaining useful life, just some examples. There's
16 monitoring during exposure that's being evaluated.
17 There's dielectric measurements, acoustic
18 measurements, and measurements are being done on short
19 segments and fairly long length cable segments as well.
20 Condition monitoring techniques for
21 electrical cables using NPPs. This is NRC Reg. Guide
22 1.218 and gives you a variety of testing, whether the
23 type of conductors that's applicable to and other
24 considerations, whether it's -- the test is full length
25 or local or whether you need to disconnect the ends. Is

there damage or potential damage to the cable.

So some long length cable measurements are being evaluated and tested. Frequency domain reflectometry, I believe that's what's been shown there on the upper right. It's kind of -- difficult to see, but that has been a technique that PNNL has been using for long length cables and has been showing some interesting results and good results.

Dissipation factor, Tan Delta, we've kind of mentioned, also the "hipot" which is potentially damaging because you're basically taking it up to a possible failure. So that was -- it's basically determining the adequacy of electrical insulation, so for normally occurring over voltages is my understanding of it that works. Partial discharge, localized dielectric breakdown of a small portion of the solid insulator system relating to high voltage or under high voltage stresses and another one is partial discharge which also can be lumped into a somewhat destructive analysis.

Local spot measurements, so this is measurements on small sections. Indenter testing which is kind of shown there. Sorry about the scale of the axis. As well as capacitance, acoustic, and dielectric constant.

1 So one of the goals is really can we use --
2 evaluate through these different techniques to really
3 get a better picture of remain cable use for life. And
4 while each one, each of the NDE technique can give you
5 some information, it's not -- none has really been set
6 up to give you the full picture.
7 So an example of aging model for
8 polyethylene is comparison of the elongation at break
9 which is the E over E over Cc. EE over EEO with three
10 other condition monitoring tests: NMR, percent gel,
11 and uptake factor. So it's really utilizing a lot of
12 these key variables together.
13 So at this point, trying to establish what
14 do we pull out of the NDE techniques and how useful is
15 it to really predict end of useful life. And at this
16 point, we really are relying on multiple techniques to
17 give us a bigger picture of the lifetime remaining of
18 the cable.
19 Materials degradation.
20 MEMBER RAY: Let me as a question at that
21 point. The lifetime remaining - we're always talking
22 here about the period 60 to 80 or so. It would be a real
23 coincidence, it seems to me like if this work that
24 you're describing were to only raise questions about the
25 potential end of useful life during that window of time.

1 In other words - saying it another way it
2 seems preferable or possible, at least, that it would
3 raise questions about an earlier than 60-year end of
4 life. Do you have some way of addressing that if that
5 were to occur?
6 MS. BERNHOFT: Yeah. I think what we've
7 said all along is that, you know, cable replacement
8 could be a possibility. And so it's understanding the
9 tools that you have to assess the cables and replacing
10 them in a timely manner.
11 MEMBER RAY: Then that would include
12 during a period sooner than 60 years -
13 MS. BERNHOFT: Yes.
14 MEMBER RAY: - I presume?
15 MS. BERNHOFT: Yes.
16 MR. LEONARD: We can go to the next one.
17 So this is just kind of a take on what's -
18 what we're doing with the harvesting of service stage
19 cables from Zion Unit 1 and 2.
20 So, again, ORNL has been placed as a
21 interface with Zion Solutions via the LWRS, EPRI, and
22 NRC and others have participated in multiple
23 discussions on materials in the interest of viability.
24 This is following the EMD recommendations
25 for service stage cables we can do further evaluations

1 on. So there's - in the spring of 2016 we're looking
2 to harvest additional cables.
3 There's current work out there that's now
4 testing of NRC cables at NIST. That is currently being
5 done. We're looking in the spring at also getting more
6 cables from Zion Unit 2. Some of the cables of interest
7 include thermal exposed materials which were in the
8 steam tunnels.
9 Thermal and radiation conditions and
10 benign environments to get - help establish some
11 baseline materials properties for that particular age
12 of cable.
13 So this kind of falls back to our - the
14 beginning of our topic and then providing a summary.
15 So, you know, we're doing a number of fundamental
16 understanding - detail of understanding of the cable
17 stressors and how they impact changes in the material
18 - what happens to the electrical properties, mechanical
19 properties physical properties.
20 We're looking at NDE techniques, always
21 looking for better transformational NDE techniques that
22 can give us better predictions and really the outcome
23 as we really want to be able to determine the remaining
24 useful life predictions in the end. So I think with
25 that, that's -

1 MS. BERNHOFT: Yeah, so understand one of
2 the things we're trying to do with cables is we're not
3 trying to say that they're going to make it to 80 years.
4 We're going to say the tools and the
5 techniques and the methodologies have to exist so people
6 can determine when cable replacements may be needed for
7 safe operations.
8 MEMBER RAY: And, of course, my only point
9 is they may disclose that it's not going to make it to
10 60 years either.
11 MEMBER SKILLMAN: Sherry and Keith, thank
12 you.
13 Colleagues, any further questions for this
14 team?
15 MEMBER BROWN: I'll just make an
16 observation, for what it's worth. I mean, I don't want
17 to be fired because of it but this the low voltage
18 cables. The Enterprise when to sea in 1961 and was
19 actually placed in service in the reactor compartment.
20 That's the equivalent of a containment, if
21 you want to call it that probably in '58 or '59 through
22 part of its test programs. It's all PVC cables, all low
23 voltage 600-volt rated cable.
24 We did an evaluation of that cable at one
25 of the refueling, roughly halfway - I call it roughly

1 halfway because nothing ever measures exactly halfway
2 when ships are operating - and to determine what did we
3 do with the cable. And we looked at it, had the guard
4 look at it.
5 We tested it with all the little, you know,
6 the tools we had which were not real, real good.
7 Determined we weren't going to replace of them and the
8 ship is finally brought in. I think it was, roughly,
9 one or two years ago.
10 So roughly, what is that, 50, 52 years of
11 operation, and when I retired we still had not -- that
12 was 15 years ago - they still had not had any failures
13 of the cable in the reactor compartment that had to be
14 replaced.
15 So our decision was -- since it's
16 non-dynamic and the shipboard environment is relatively
17 not benign at all on the earlier carriers. I mean, they
18 just suck salt air out of the ocean and blow it through
19 the reactor compartment and they survived.
20 So, I mean, one -- I have some -- I'm
21 probably sounding like I have a little bit of concern
22 about my - concern is the wrong word -- the little
23 examining details.
24 It's like looking at, you know, how much
25 extension or elongation can you have of a piece of

1 insulation before it breaks and using that as an
2 evaluation tool for the life of the cable when you put
3 it in.
4 We tested -- I mean, we did do Megger checks
5 and stuff like that but that was a one point -- like you
6 pointed in here, you can't trend Megger very well.
7 It's very sensitive to environmental conditions.
8 So I think you've got to look hard at - the
9 industry really has to look hard at how they want to
10 treat these and evaluate it on a couple of different
11 bases.
12 How good are our testing and the damage you
13 could incur if you go and start ripping cables out and
14 damage you might incur to other stuff because people
15 climb around on tables to repair this.
16 They're going to have scaffolding and other
17 things that people climb around on. You just got to be
18 careful with it.
19 So I think you got to do a lot of heart -
20 you know, soul searching to make sure we have a program
21 that is practically realistic and I'm just - cable was
22 pretty robust is what I'm - what I'm trying to - what
23 I'm trying to say and the reactor compartments on the
24 Navy ships are not benign radiation and temperature
25 wise.

1 They run quite toasty relative to -
2 probably toasty relative to the commercial plants even
3 and that the old insulations of PVC held up very, very
4 - pretty well and they -
5 MEMBER BLEY: They weren't ratcheted on
6 the other.
7 MEMBER BROWN: Well, I won't say they were
8 never wedded. There was a few circumstances where
9 MEMBER BLEY: They got wedded in general.
10 MEMBER BROWN: They were not wedded.
11 That's right. We did not have submerged cable,
12 although I will have to admit there was a case where the
13 bilges got filled.
14 There were some cables where they weren't
15 supposed to be because nobody caught it and they were
16 wedded for a while and we dried them out and everything
17 worked okay after that.
18 MEMBER RAY: But, Charlie, you -
19 MEMBER BROWN: I'm just giving some
20 experiences.
21 MEMBER RAY: Except I'm going to add to it,
22 if I may.
23 MEMBER BROWN: Okay.
24 MEMBER RAY: We never had that kind of
25 accident, to my knowledge, on the Enterprise and that's

1 the thing that we have to bear in mind is these things
2 are supposed to work at 60 years or 65 or 75 not under
3 normal conditions but under design conditions.
4 So it's hard for me to relate to those kinds
5 of experiences which I've had also and draw any
6 conclusions from them.
7 MEMBER BROWN: Well, if you look at - if you
8 look at the LOCA - if you look at the EQ testing we did
9 test those cables. They were in the reactor
10 compartment. They had special qualifications for
11 higher temperatures for short-term type operations.
12 And I agree with you. I'm not disagreeing
13 with your point. It's valid. I'd just make sure you
14 use some - I'm just emphasizing we ought to use some
15 common sense for this blacksmith technology approach to
16 doing business.
17 MS. BERNHOFT: Yeah. Thank you for that.
18 I know - I can speak for our main cable researcher. I
19 mean, he - I think he'd agree with your comment that,
20 you know, cable systems - if you go back to even my first
21 overhead, I mean, they are in principle robust properly
22 installed, you know, proper design.
23 MEMBER BROWN: That's the key point right
24 there.
25 MS. BERNHOFT: Yes. Absolutely. But,

1 you know, that's not an aging issue.

2 MEMBER BROWN: No.

3 MEMBER SKILLMAN: Colleagues, any other

4 questions or comments before we release this team?

5 Hearing none, Sherry and Keith, thank you,

6 and Steve, we're back to your team.

7 MR. BLOOM: Okay. We got one more

8 presenter, Darrell.

9 MEMBER SKILLMAN: Darrell, welcome to you.

10 MR. MURDOCK: Thank you.

11 MEMBER SKILLMAN: Yes, sir. Please go

12 ahead.

13 MR. MURDOCK: Good afternoon. My name is

14 Darrell Murdoch and along with my colleagues Sheila Ray

15 and Mohammad Sadollah we are going to brief you on the

16 condition assessment of EQ and non-EQ cables for SLR.

17 This presentation is going to be divided

18 into five sections, regulatory and technical framework,

19 technical issues, research to address those technical

20 issues, status of that research and, finally, schedules

21 and milestones.

22 The regulatory framework that govern the

23 condition assessment of EQ and non-EQ cables include 10

24 CFR 50.49, 10 CFR 50.65, 10 CFR 50 Appendix A, 10 CFR

25 50 Appendix B and 10 CFR Part 54.

1 Applicable regulatory guides include Reg.
2 Guide 1.89, which endorses IEEE standard 323, Reg. Guide
3 1.211, which endorses IEEE standard 383 and Reg. Guide
4 1.218, which uses SLR authority 10 CFR 50 Appendix B and
5 10 CFR 50.65.
6 The fifth volume of the expanded material
7 degradation assessment examined cable degradation
8 issues based on the following criteria: technical and
9 operating experience knowledge levels, expected
10 severity of the degradation and the likelihood of
11 occurring.
12 The EMDA also examined cable-aging issues
13 up to 80 years of operation and the EMDA formed a basis
14 for some of the technical issues that we will discuss
15 next.
16 The first technical issue we'd like to
17 bring to your attention is cable environmental
18 qualification for SLR and here we believe that
19 additional testing needs to be done to ensure that EQ
20 cables - environmental qualification - will be
21 maintained throughout the SLR period.
22 The second technical issue is condition
23 monitoring and for condition monitoring due to the lack
24 of testing data for the period of operation from 60 to
25 80 years we believe that additional research needs to

1 be done to develop acceptance criteria that can be used
2 to perform service life protection EQ and non-EQ
3 cables for the SLR period.
4 MEMBER SKILLMAN: Darrell, is there a case
5 that needs to be understood where licensees have
6 undertaken a very intense EQ program and could it be that
7 what they have already done is sufficient?
8 MR. MURDOCK: Well, next -
9 MEMBER SKILLMAN: I mean, some licensees
10 spend an awful lot of time telling the treasurer on their
11 EQ program -
12 MR. MURDOCK: Yes.
13 MEMBER SKILLMAN: - and they've got good
14 data. They know what the cables are qualified for, what
15 the instrumentation wiring and cable are qualified for.
16 MR. MURDOCK: So what we are asking is that
17 in determining the qualified life of the cables that you
18 are taking into account, you know, certainties that we
19 are discussing now to ensure that the environmental
20 qualification of the cables are not going to be
21 compromised. So we are not making a judgment either
22 way.
23 We are just saying that in determining how
24 long the cables are going to last that you take into
25 account these uncertainties to make sure that you don't

compromise the cable's ability of withstanding at the same basis event and performing its safety function.

MEMBER SKILLMAN: Yeah, fair enough. Okay. Thank you.

MR. MURDOCK: The next technical issue - and the next five technical issues the - that where discussed by the industry and they stem from performing aging in a highly accelerated manner, meaning high temperature and high radiation does rates, were the degradation mechanism that affect these cable samples and these highly accelerated conditions are different than the degradation mechanism that the cables see in their normal plant environment.

The first of those issue is the diffusion limited oxidation, which the industry already discussed, and for diffusion limited oxidation it takes place when the aging is done under a highly accelerated condition where the concentration of diffuse oxygen drops to zero or near zero, significantly reducing or eliminating the oxidation reaction process that takes place in the interior portion of the cable.

So if a cable sample that experienced the diffusion limited oxidation was put through a local test and will probably pass because it was not subjected to the right degradation mechanism because of the lack of

1 oxygen in the interior portion of the cable.

2 The next technical issue is activation

3 energy and the implication of activation energy is that

4 if you use the wrong values of activation energy to

5 perform a service life protection then you could be in

6 a situation where you could over estimate or under

7 estimate the lifetime of this material in question.

8 The next technical issue is synergistic

9 effects between thermal and radiation aging and here we

10 just want to highlight that aging done concurrently or

11 synergistically could be more severe than again done

12 sequentially as have been traditionally done in the

13 past.

14 The next technical issue is inverse

15 temperature and for inverse temperature for

16 semi-crystalline materials aging done at lower

17 temperature at a constant dose rate could be more severe

18 than aging done at a higher temperature at that same

19 constant does rate.

20 The next technical issue is dose rates -

21 does rate effect, and here age-related degradation done

22 at lower dose rate could be significantly more severe

23 than age-related degradation at higher dose rate for the

24 same sort of dose.

25 A final technical issue is cable

degradation due to submergence and the NRC completed a literature search at the end of 2014 that concluded that additional research was needed to systematically understand the stressors that come into play in a submerged environment and/or synergies.

Also, that there was not a consistent end of life criteria for cables in a submerged environment and that from a condition monitoring standpoint that additional research was needed to come up with a better - with a parameter that will give us a better indication of the cable as it ages in a submerged environment.

Next, I will talk about ongoing research.

MEMBER STETKAR: Darrell, before you get to the ongoing research, as I listen to this it strikes me that none of these seem uniquely associated with subsequent license renewal.

I could - I could raise the same issues if I'm looking at 20 to 40 years or 40 to 60 years or 60 to 80 years or 80 to 500 years.

So why are these now suddenly concerns for subsequent license renewal? Cable degradation due to submergence - I've been on the committee now for eight and a half years, and eight and a half years ago we were talking about failures of cables due to stressors under unexpected submergence conditions.

1 So this isn't necessarily uniquely
2 associated with subsequent license renewal, is it?
3 MR. MURDOCK: I think a lot of these issues
4 were raised through the EMDA, which was released in
5 2013, I believe, and actually the submergence issue,
6 yes, it could be something that could affect cables from
7 40 to 60 or even to first -
8 MEMBER STETKAR: Well, they - it affected
9 cables in the first five or six years at some plants.
10 So it's been known for a long time.
11 My point is that are we raising these issues
12 in the context of subsequent license renewal when we
13 should be just looking at them in general.
14 MS. RAY: This is Sheila Ray. I'm in NRR
15 electrical branch. Yes, you are correct. These are
16 issues that should be addressed in Part 50.
17 However, as Darrell will talk about, there
18 is some research that has been initiated and upon that
19 conclusion of that research staff will take the
20 appropriate action.
21 MEMBER STETKAR: Okay. Thank you.
22 MR. MURDOCK: So now I will discuss the
23 ongoing research to address some of the technical
24 concerns that I have just discussed.
25 The first project that we have ongoing is

1 called assessment of condition and monitoring methods,
2 and during this project our goal is to age new and
3 naturally aged cable samples to mimic 80 years of
4 operation.
5 Once we age the cable samples to - well, let
6 me just say this. So we have aged the cable samples to
7 the 60-year point, removed some of the samples, put
8 those samples through a LOCA test to make sure that they
9 pass and then we'll continue aging the cable samples to
10 the 80-year point and then subject the cable samples at
11 the 80-year point to a LOCA test to determine if these
12 samples would have had the ability of lasting throughout
13 SLR and have sufficient margin to be able to withstand
14 a design basis event and perform their intended safety
15 function.
16 In addition, during the aging process for
17 this project we are going to evaluate a number of
18 condition monitoring methods to determine their ability
19 to track degradation as a function of time.
20 And finally, during this project, the aging
21 for this project is going to be done synergistically at
22 low dose and low temperature.
23 Doing aging synergistically at low dose and
24 at low temperature closely mimicked the environment
25 that the cables see at the plant. And our goal is to

1 demonstrate that if you do the aging synergistically at
2 low dose and at low temperature you won't have to account
3 for the uncertainties listed in the EMDA such as
4 diffusion limited oxidation, activation of energy,
5 synergistic effects, inverse temperature, low dose
6 rates.
7 The second project that we have ongoing is
8 called evaluation of cable degradation in a submerged
9 environment and that was the project for which we
10 completed the literature search where the contractor
11 concluded that we didn't understand the stressors, that
12 there was not a consistent end of life material and that
13 we needed to do more research from a conditional
14 monitoring standpoint.
15 So the next step in this project is to
16 evaluate EPRI's approach to using Tan-Delta to perform
17 aging management of cables in a submerged environment,
18 and once we complete the evaluation of EPRI's approach
19 then if there is a need to do additional research we will
20 identify that research at that point.
21 Next, I will talk about the status of these
22 two projects. For the first project called assessment
23 of condition monitoring methods, the NRC hired the
24 National Institute of Standards and Technology and
25 carry out this project. NIST is in the process of

1 hiring a subcontractor to carry out the thermal and
2 radiation exposure portion of this project.
3 Once NIST hired a subcontractor - this
4 subcontractor - subcontractor will be tasked with
5 performing our functionality test of the chambers that
6 will be used to age the cable samples and once the
7 functionality - once the functionality test is done by
8 the subcontractor we should begin - we should be able
9 to begin the thermal and radiation portion exposure of
10 - terminal radiation exposure portion of the project.
11 For the second project, evaluation of cable
12 degradation in a submerged environment, as I mentioned
13 before, right now we are focused on evaluating EPRI's
14 approach for using Tan-Delta to perform aging
15 management of cables in a submerged environment.
16 Next, I will go over the schedules and
17 milestones and we hope to carry out the functionality
18 tests of the chambers by February of 2016. We are
19 working closely with Oak Ridge and EPRI to harvest
20 cables from Zion and Crystal River.
21 We hope to complete that by March of 2016.
22 We hope to complete the 24-month thermal and radiation
23 exposure by May of 2018 and we'll follow that up with
24 a simulated LOCA test that we hope to complete by October
25 of 2018.

1 And we hope to complete writing the - file
2 a report by December of 2018 and this report will be used
3 to update any applicable regulatory guidance.
4 The second project, evaluation of cable
5 degradation in a submerged environment, we hope to
6 complete evaluation of EPRI's approach by April of 2016.
7 In conclusion, though there are a number of
8 challenging technical issues that need to be addressed
9 for the SLR period, we believe that continuing
10 engagement between the NRC staff and the industry is
11 critical to achieve a satisfactory solution to these
12 technical issues. And that concludes my prepared
13 remarks.
14 MEMBER SKILLMAN: Thank you. Colleagues,
15 do you have any questions? Excuse me. Team, thank
16 you. Colleagues, do you have any questions for
17 Darrell?
18 MEMBER SCHULTZ: One question, Darrell.
19 You talked about quite a number of technical issues and
20 then two programs that you've identified for research.
21 And should we take from the - from that that
22 one of the programs is specific to one of the technical
23 issues, submergence, and the other one you seem to
24 indicate that it's somewhat integrated, the work that
25 NIST is doing, somewhat integrated and perhaps covers

1 the rest of the technical issues.

2 MR. MURDOCK: Yes.

3 MEMBER SCHULTZ: My question is are there

4 other research programs also planned to address some of

5 the other technical issues that are lower lying or is

6 this it?

7 MR. MURDOCK: This is it. With the

8 particle assessment of condition of monitoring methods

9 we are going to age new and naturally aged cable to

10 ensure that these cable - we'll age these cables to the

11 80-year point to make sure that they would have - they

12 have the ability of, first of all, making it to the

13 80-year point and making it through -- having enough

14 margin to withstand a descent basis event and perform

15 the safety functions. So that takes care of the first

16 issue that I mentioned, which was -

17 MEMBER SCHULTZ: That's fine. You don't

18 need to go through it one by one. But what you're saying

19 is that that's an integral test program that you feel

20 is going to demonstrate what is needed for the extended

21 period - license period.

22 MR. MURDOCK: Yes. So this - yes and no.

23 So for the five technical issues that were

24 raised in the EMDA official limited oxidation LRG -

25 synergistic effects, inputs temperature and those

rates, we have to take kind of a different approach.

Our approach is to say that if you do aging synergistically at low dose and low temperature that you don't have to account for these issues.

For the submergence issue we are going to evaluate EPRIs approach and then if there is additional research that we need to do then we'll identify that research at that point.

And for the first - the aging of naturally aged cable we are going to address the first technical issue that I brought to your attention.

So yes, for this project we think - with both these projects we think it will be comprehensive to address all those technical issues that are brought to your attention.

MEMBER SKILLMAN: Good. Colleagues, any further questions to the NRC team?

MR. THOMAS: Yeah, I do have a comment. I think it's in response to about Mr. Stetkar's question about

MEMBER SKILLMAN: Yeah. Please speak closely to the microphone.

MR. THOMAS: Yeah. Brian Thomas. This question about, you know, some of these research programs we've been doing research on. We've been

1 studying them for quite some time now.

2 That's true. As Mark Kirk mentioned

3 earlier, in the vessel area and in the reactor internals

4 area, it's the same, as well as in the cable area.

5 The thing is - and Sheila answered it

6 appropriately - I just wanted to make sure that we, you

7 know, I sort of expanded on her response - a lot of it's

8 - what we're doing now is punctuated by, of course, the

9 EMDA report, which said, you know, some of these areas

10 have, even though we have - we may have low knowledge

11 and therefore it may have a high potential for some sort

12 of a degradation during subsequent license renewal, in

13 addition to in some areas we have been doing a lot of

14 research we may have high knowledge but there still may

15 be a high potential for degradation during subsequent

16 license renewal.

17 And in addition to that, it was - it was

18 cited in the SRF from the commission - it said, you know,

19 these - this is one of the areas that you will focus on.

20 So although we may have done a fair amount

21 throughout the years, our interest and our focus got

22 heightened due to the subsequent license renewal.

23 MEMBER STETKAR: Yeah, and I appreciate

24 that. The reason I asked about a couple of the areas

25 was going back to a question I asked earlier this morning

about why is there a need for a separate GALL report distinctly associated with subsequent license renewal versus a natural update to the GALL - to the existing GALL report based on new information that we collect.

Why do you distinguish the need to have - you said earlier we need a new technical basis or something like that. Why is that necessary?

Why isn't it just the natural evolution of our learning of how things age and what phenomena we might be susceptible?

Whether - you know, there's no magic line that's drawn in the sand that says 59 years, 364 days and some-odd hours versus 60 years and one minute things suddenly change.

MEMBER RAY: Well, John, that's similar to the question I ask -

MEMBER STETKAR: Sure. Exactly. It's an evolving process.

MEMBER RAY: If something identifies an issue prior to 60 years are we going to be able to cross that line and say we got an issue we got to deal with sooner than 60 years and the answer was yes, we will, as I understood it.

MEMBER STETKAR: And the industry has actually - industry and the staff has actually

1 demonstrated that in spades, in my opinion, in a couple
2 of issues.
3 One is buried piping - buried and
4 undergrown piping where the industry has taken a lot of
5 actions to address that issue now for operating plants
6 within their current operating period.
7 Before pre-40 years any others is buried in
8 underground cables, wedded in particular where actions
9 are being taken now even under the current licensing
10 issues. So -
11 MR. THOMAS: And the thinking is - you
12 know, we would conduct the research in those areas. We
13 - you know, we're a little more focused and have taken
14 an integrated approach at it, if you will, looking
15 domestically with our partners as well as, you know, DOE
16 and EPRI and as well as internationally and basically
17 what we're doing from our research standpoint we're
18 developing technical bases to put the staff in a
19 position to do a much more efficient and effective
20 review of an application, if you will.
21 You know, we talked earlier about the near
22 term and the longer term work. We recognize some of
23 this work will continue. But there will be a point -
24 I want to say a line in the sand, so to speak - there
25 will be a point where we will say we have sufficient

1 technical basis that could be used to bear upon or would
2 have merit in our evaluation of that application at some
3 time in the process. If not the initial application,
4 you know, subsequent applications.
5 MEMBER STETKAR: I think a bit of my
6 concern, and we need to see what comes out in February
7 and see what it looks like. But a bit of my concern is
8 looking at how we've gone from GALL, I guess, Rev. 0,
9 which really predated my time, to GALL Rev. 1 to GALL
10 Rev. 2 and how that transitioned at least from 1 to 2
11 affected the period of license renewal that began,
12 again, before I was on the committee eight years ago and
13 continues up through now.
14 We've seen an evolution as we've learned
15 more. Some plants were renewed under GALL Rev. 1.
16 Bill, I'm being told, be held - I don't want to say held
17 accountable but the inspection process, once they
18 entered the period of extended operation will examine
19 them for what we know under GALL Rev. 2. Plants that
20 are being renewed right now are being held to GALL Rev.
21 2, by and large.
22 So I don't see - I just don't get why we need
23 new guidance - brand new guidance with a new cover on
24 it, especially if 95 percent of it looks the same as the
25 old guidance.

1 And do we - the concern is do we run a risk
2 of now having people come in and say now, I don't need
3 to meet the new thing, which I'd like to call GALL Rev.
4 3 because - today because I haven't passed that 59.99
5 year milestone.
6 I only need to meet that at 60.01 years,
7 which I think is Harold's issue also.
8 DIRECTOR MILLER: If I could, this is Chris
9 Miller, and you bring up some good points. I think
10 we'll have a much more robust discussion of that in our
11 next meeting.
12 MEMBER STETKAR: That's right. February
13 is when we actually see that guidance.
14 DIRECTOR MILLER: One of the issues that
15 you bring up is operating experience and we're certainly
16 learning that and you'll find lots of updates in the SLR
17 GALL that will reflect operating experience, you know,
18 that we've learned all along or better ways of
19 presenting the information or capturing the
20 information.
21 So we think that's important. So we'll
22 make sure that we have that.
23 But the second point is there will be some
24 things that weren't considered so important - shall I
25 say it that way - weren't considered as important. Not

that they weren't important from a safety perspective but that we weren't going to get to things, let's say, in the fluence area, degradation mechanisms that you say in 80 years.

We just weren't going to get there. It's not that we said at 59.99, you know, everything's cool and then at 60.1 it's going to start falling apart. It's that those mechanisms we knew that alpha 60 that all the plants in the fleet would likely be okay.

So those weren't addressed. And so you have to put some additional words in the SLR GALL that say okay, now you need to start considering these and if you don't have that information you need to come up with an aging management plan that considers these additional things and I think that's where you'll find the difference and why staff decided to go with one that was called an SLR GALL.

And we can go back and forth whether or not that's right but I think it would be more productive if we did it once the guidance is out.

But you do bring up some very good points and we'll definitely consider that as we put the guidance together.

MEMBER RICCARDELLA: Yeah. I mean, on that same point it seems like some of the new issues that

1 you have to consider are based on things like radiation
2 fluence or some other parameter that's important
3 whether you're in 40 years, 60 years or 80 years. I tend
4 to agree with John.
5 I just don't see the sense of it. And the
6 other thing you set up too is when you get new operating
7 experience, which we're going to get - we get some leaks,
8 we get something else - now you're going to have two
9 GALLs you're going to have to go back and revise and
10 you're setting yourself up for, you know, an
11 administrative nightmare.
12 MEMBER SKILLMAN: Colleagues, any other
13 discussion at this point? Hearing none, NRC team,
14 thank you. You're released.
15 Now, we have more admin in here. Nobody go
16 away. Any members that would like to come to the
17 microphone and make a statement or questions? I invite
18 you to do that now.
19 MR. SRINIVASAN: With due apology to Jason
20 Reimer, Mr. Chairman, I just wanted to address one thing
21 and that is the public perception and communication
22 issue.
23 MEMBER SKILLMAN: Sir, your name, please?
24 MR. SRINIVASAN: This is Makuteswara
25 Srinivasan, Co-format. I just wanted to address the

1 public communication perception issue and one of the
2 things personally that I have been advocating is to not
3 to use the word a show stopper.
4 For those who are not initiated in fine arts
5 and Broadway in this kind of a thing it makes it
6 difficult to understand. And also if there are no show
7 stoppers it begs the question why are you doing all of
8 this and so forth.
9 So that is one observation. The second one
10 pertains to the so-called inaccessible areas and I did
11 not see any coverage of what - I know it is a difficult
12 issue.
13 It is something that industries are
14 addressing on a daily basis and so forth. But it begs
15 the question that would those inaccessible areas be
16 subject to whatever it is, is it in the 59.99 or 35th
17 or whatever really.
18 And I know it is difficult - how you do
19 research of something. But again, for the public
20 communication and public perspective if they hear about
21 an access of areas and what NRC -- I mean, what industry
22 is doing or not doing it might be helpful.
23 Thank you for the opportunity given to me
24 to comment.
25 MEMBER SKILLMAN: Thank you for your

1 comment.

2 Now, on the bridge line, if anybody might

3 be there would you please just communicate that you can

4 hear us?

5 MALE PARTICIPANT: We can hear you.

6 MEMBER SKILLMAN: Thank you. Now,

7 anybody on the bridge line would you wish to make a

8 comment please?

9 MR. MANTEY: Yeah. Hi, this is Drew

10 Mantey from EPRI and I just wanted to - I'm the cable

11 aging management resource person. There were a couple

12 things that went on during the presentation I want to

13 make sure were reclassified.

14 With regard to the cable slice being

15 assessed, the predominant installations and jacket

16 materials are the hypalon, EPR and the XLPE TPE jackets,

17 neoprene jackets that we're evaluating.

18 There are some outliers there of older

19 plants that have silicone rubber and PVC installations

20 but, you know, for the most part those are not in

21 containment areas. They would be in other areas of the

22 plant.

23 There was another comment in regards to -

24 oh, what was the other one? The question on destructive

25 nature of high potential testing.

1 Withstand testing along with Tan-Delta is
2 the preferred method for evaluating medium voltage
3 cables and there were perceptions from other past
4 practices that make people nervous of the term withstand
5 or high potential tests.
6 But we're not doing anything to the cables
7 that they can't take by leaps and bounds, higher values
8 than what we applied during our diagnostic testing just
9 to make sure that there's no large single defect that
10 might cause a failure that Tan-Delta doesn't see.
11 I think they were the major issues I wanted
12 to address. So I'll stop there.
13 MEMBER SKILLMAN: Sir, thank you very
14 much. Is there -
15 MR. LEWIS: Marvin Lewis. I have a
16 question.
17 MEMBER SKILLMAN: Go ahead, Marvin.
18 MR. LEWIS: Thank you. Yeah, I don't know
19 what the problem was. I - probably my own problem about
20 which button to push.
21 Look, I'm listening to this and I hear ooh,
22 this passes and this passes and this doesn't show any
23 indication of failure and I don't - I haven't been in
24 this for 15 years. I saw no failures along the slide.
25 Well, this is all well and good. But I

1 don't hear anybody say hey, we're going to advertise
2 where there are failures, did they have a protocol where
3 they just fix things and went on and did not report it
4 or did not put a maintenance note in some kind of diary
5 or whatever.
6 And the failures are being hidden not
7 because - not purposefully but because it's just a way
8 things are done in a ship or a power plant or wherever.
9 I hope I'm making myself clearer than I sound. Thank
10 you. Bye.
11 MEMBER SKILLMAN: Marvin, thank you.
12 Members of the public, is anybody out there who else
13 would like to make a comment, please? Hearing none,
14 please close the bridge line.
15 Colleagues around the table, might any of
16 you have any final closing comments you would like to
17 make, please?
18 MEMBER RICCARDELLA: I have a few
19 thoughts. I think - I think the steel programs - I think
20 the aging management programs are very well understood
21 I think largely because of the operating experience
22 we've had.
23 We've had, unfortunately, numerous cases
24 of cracks and leaks and other problems in these
25 components. We've got our arms around them and I don't

1 see subsequent license renewal as being a quantum leap
2 to some new technology. It's just maintaining the
3 programs that we've been maintaining all along.
4 The concrete - aging management of
5 concrete, I think, is relatively new by comparison.
6 We've had a few well known cases of OE in the concrete
7 but they're relatively few and I don't think it's - that
8 topic is as well developed as the steel components.
9 Personally I question why we're spending a
10 lot of effort on the radiation effects. I think that,
11 you know, of all the safety-related concrete structures
12 in a plant we've got a couple of inches of concrete in
13 the shield wall that exceeds the threshold for damage
14 and everything else.
15 There's other issues that I think are
16 probably more important - ASR, chlorides, cracking -
17 things of that sort that affect a lot more structures
18 than the radiation.
19 But that's just my personal opinion, and as
20 for the cables that's so far from my area of expertise
21 I'll defer to Charlie on that. Cables.
22 MEMBER SKILLMAN: Pete, thank you.
23 Harold Ray, any comments you might have,
24 sir?
25 MEMBER RAY: Well, I think I've made the

1 principal comments I have. There is a policy issue and
2 that's not our domain but I want to acknowledge it, that
3 says that SLR is different than license renewal and that
4 decision is made elsewhere than here but it's something
5 that we reflect on in any event.
6 And I don't want to opine on it any further
7 except to say well, maybe LSR is different and that as
8 I was sharing offline with others there's a threshold
9 for challenging things that last for some period of time
10 and you don't want to have that threshold be such that
11 it's unreasonably low.
12 But at some point maybe you exhaust that -
13 the tenure of that assumption and you say all right,
14 we've got a new threshold to meet here and that may be
15 why we're talking about SLRs different than license
16 renewal, and I'll just leave it at that.
17 MEMBER SKILLMAN: Thank you, Harold.
18 Steve?
19 MEMBER SCHULTZ: Yeah, I have a general
20 comment that I've saved until the end because I think
21 it's applicable to each of the different areas that
22 we've discussed today and it is a general comment but
23 specifically applies perhaps more to the concrete issue
24 as well as the material performance at high fluence and
25 that is where we've pointed to a number of areas where

1 over the last few years and the next few years coming
2 there's going to be some substantial fundamental
3 research going - ongoing and with regard to both
4 experimental design and the measurements that are going
5 to be done, some advanced mechanistic and computational
6 models that are going to be proposed a lot of separate
7 effects work is going to be done as we know. And we've
8 done this before, we'll do it continuously.
9 But for both the NRC research and NRR that
10 is the challenge to assure that this is - all this
11 information is going to be assembled in an appropriate
12 representation for the facilities that we regulate
13 because there's a lot of uncertainty that we've seen on
14 the slides in many of these areas and if we don't
15 assemble that properly the research information is
16 going to be coming improperly.
17 We may just continuously go into a research
18 mode and that's not what we're about here. We talked
19 about bench marking and making sure that we've got
20 appropriate programs that are going to incorporate
21 bench marking into the research that's ongoing and we
22 talked about round robin evaluations at one point and
23 just - I know that research already understands that
24 there are sometimes cautions that need to be taken when
25 a round robin bench marking exercise gets started to

1 assure it gets finished appropriately.

2 The comment I wanted to make regarding the

3 - in the introduction we talked about - well, the public

4 the NEI talked about that there have been discussions

5 with the public and it was determined that the public

6 would be satisfied that NRC is appropriately monitoring

7 and has a program to do so and could take appropriate

8 actions if necessary in the extended period.

9 I think there's something else that we need

10 to identify and it came out in the public comments and

11 that is to be sure that we're identifying where it's

12 clear that we don't have cliff edge effects and that

13 we've got a continuum associated with the phenomena that

14 we're monitoring.

15 And where we can do that we ought to be

16 careful - not careful but clear what we have determined

17 from the research and if we can demonstrate that we're

18 working with a continuum here in terms of the effects

19 that we've identified that would be a comforting result.

20 Thank you.

21 MEMBER SKILLMAN: Steve, thank you.

22 John, I'm going to come back to you as our chairman.

23 Charlie Brown, might you have a comment before -

24 MEMBER BROWN: I just wanted to amplify -

25 I wanted to thank the EPRI - thank you very much - I

1 wanted to thank the EPRI gentlemen for talking about the
2 withstand voltage relative to the commercial plants.
3 And I didn't mean to imply that we never did
4 withstand voltage testing on our cables. Typically,
5 our initial popout was at twice rated plus a thousand
6 and our subsequent withstand voltages were done
7 typically at 60 percent of that value.
8 So we were still above the rating of the
9 cable but not at the twice rated plus a thousand. So
10 I appreciated his calibration on that and I had not made
11 that point during my initial comment.
12 The other observation I wanted to make was
13 that the NRC program that they were talking about was
14 focused - this is my opinion now - was focused on let's
15 do it on a performance basis as opposed to a first
16 principled basis.
17 In other words, let's take the cables,
18 irradiate them, then test them under the conditions we
19 expect them to operate and see if they work. I'm not
20 against the first principles - I mean, it would be nice
21 to have some other knowledge relative to some of the
22 materials and things like that.
23 But I do think that's a reasonable - a
24 reasonable approach to trying to get some feeling for
25 whether the cables are going to work. I'm a little bit

1 leery of the radius effect or applications when you
2 irradiate cable organic materials.
3 It's not - it's not like it's steel or rocks
4 or things like that. It's a little bit different and
5 you get the different performance and you can get bogus
6 results. So always been a little bit leery. You got
7 to be careful how you apply that and that's the - I'm
8 finished.
9 Thank you.
10 MEMBER SKILLMAN: Charlie, thank you.
11 Dr. Bley - Dennis Bley.
12 MEMBER BLEY: Yeah, thanks, Dick.
13 The main thing, and here I agree with John
14 and Pete from earlier, I've been listening - well,
15 first, I'm really impressed with the breadth of the
16 research we heard about and I've always, since I've been
17 here, been impressed with the amount of work and the
18 comprehensiveness of GALL 1 and GALL 2 and the EPRI
19 documents that provide support to the utilities in this
20 area.
21 I mean, we heard about a broad research
22 program that's been going on and continues to go on. We
23 heard only bits and pieces that are really aimed out at,
24 you know, beyond 60 years and most of that has to do with
25 increasing a radiation over time.

1 I didn't hear any really new failure modes,
2 no new requirements. The idea of having two separate
3 GALLs just worries me to death.
4 I mean, most of it's going to be the same
5 for both cases. The requirements of what you have to
6 do may shift as your radiation goes up but that's part
7 of the program.
8 I mean, that's easy to keep in there. If
9 they're separate, keeping them on track forever and ever
10 just makes me worry that they're going to diverge and
11 be kind of strange or eventually people will say you have
12 to do them both and leave it up to the poor licensee to
13 figure out what's common and not common between the two.
14 So that's my only down side of this. I'm
15 really worried about that idea of having two separate
16 programs.
17 MEMBER SKILLMAN: Thank you, Dennis.
18 Ron.
19 MEMBER BALLINGER: I think I'd like to echo
20 Pete's comments. The one thing I'd like people to think
21 about is the one place where we can spend a ton of money
22 in the black hole is starting to replace cables.
23 So I think that your comment on being very
24 careful about how we evaluate cable degradation and
25 being sure that we really need to do this because I

suspect that starting to rerun cables through things is potentially a very difficult and very expensive operation.

MEMBER SKILLMAN: Okay. Ron, thank you.

To our chairman, John.

MEMBER STETKAR: Thank you. I don't have anything to add.

MEMBER SKILLMAN: Thank you.

Kent, you've been a good soldier. Do you have any comments?

May I please thank NEI and EPRI and DOE, every participant for a thorough informative day. The objective of this meeting was to bring to the surface items that the ACRS should consider and I believe this meeting has accomplished that.

And so to the staff, Brian, Chris, Steven, thank you. I just want to remember one real important thing and that's what Yogi Berra said. He said it's amazing what you can observe when you take the time to see what you're looking at, and he was talking about observation.

This is an exercise of fine detail. We don't know what we don't know, and I think we have to be vigilant for what is it that we think we know but we really don't.

1 There are things we got to be looking for
2 in each one of these major programs and there are
3 probably one or two programs that we haven't thought of
4 that we should be thinking about.
5 It's amazing what you can observe when you
6 - it's amazing what you can see when you take the time
7 to see what you're looking at.
8 With that, thank you, everybody and safe
9 travels. We are adjourned.
10 (Whereupon, the above-entitled matter
11 concluded at 5:03 p.m.)
12
13

Subsequent License Renewal Update

Briefing for
Advisory Committee on Reactor Safeguards
November 17, 2015

Jerud Hanson
Nuclear Energy Institute

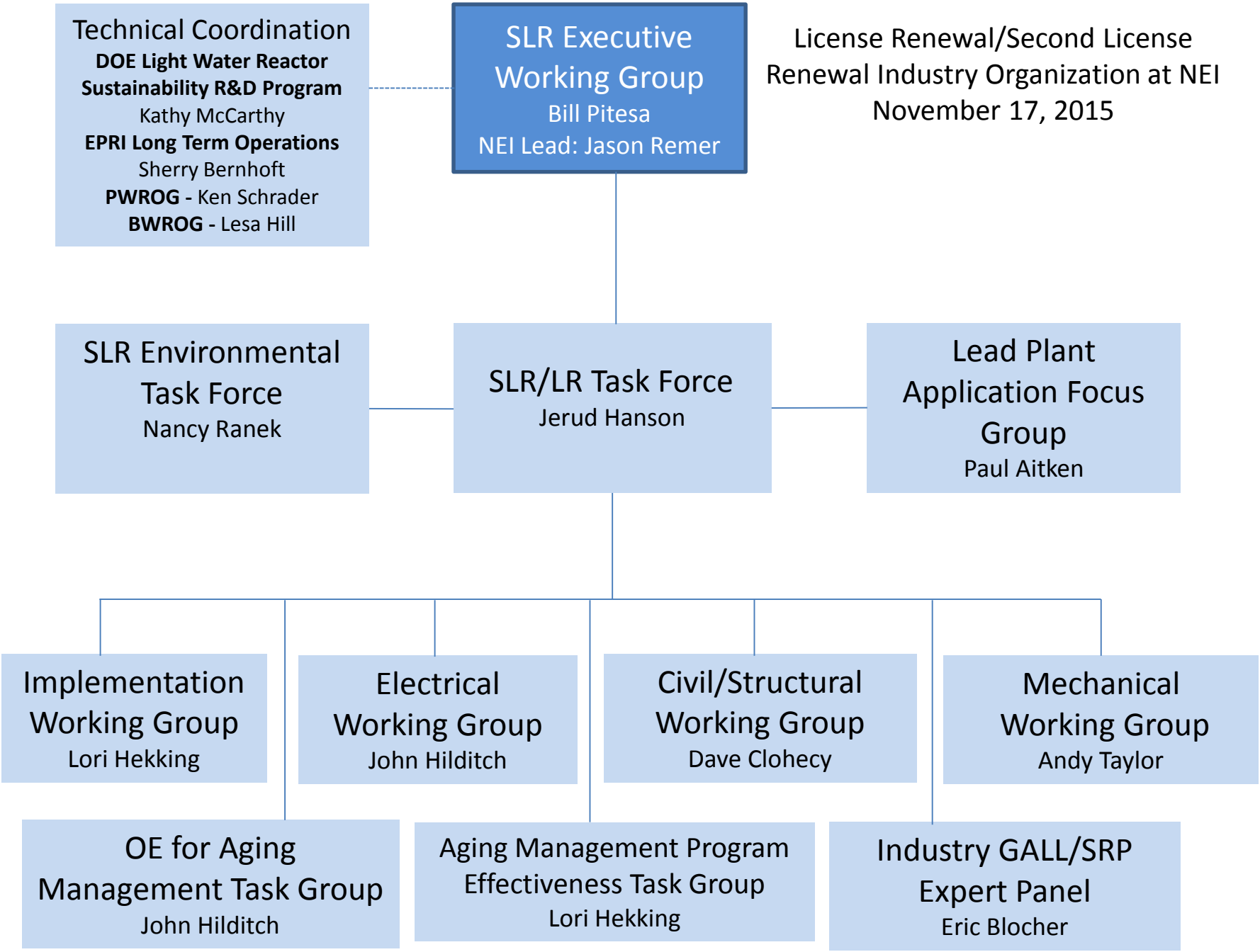
Summary of Initiative

- By 2040 half of the nation's nuclear plants will have operated for 60 years, by 2030 the U.S. could experience electricity shortages if a significant number of plants are retired
- Although new plants are currently under construction within regulated states, there is little economic incentive for building new plants
- Pursuing second license renewal is the surest way to avoid this outcome and ensure nuclear power continues playing a key role in U.S. electricity production

Update on Activities

- NEI continues to lead the industry in SLR activities:
  - Publication of SLR Roadmap
  - GALL review strategy
  - Site Visits
  - Deep-dive meetings
- Lead plant announcements

License Renewal/Second License Renewal Industry Organization at NEI
November 17, 2015



SLR Research & Development

- EPRI and DOE continue to conduct scientific research to understand the technical issues associated with long-term operation of nuclear power plants
- This research demonstrates that there are no generic technical issues that would prevent a well-maintained nuclear power plant from operating safely during the second license renewal period
- Learnings from this research and development combined with updates to industry guidance, continuous monitoring, updating and replacement of parts and systems and operating experience will help ensure nuclear power plants continue to operate safely

Questions?

Research for Safe Long-Term Operations

Joint EPRI-LTO DOE-LWRS Presentation to ACRS November, 2015

Sherry Bernhoft, EPRI Program Manager

Robin Dyle, EPRI Senior Technical Executive

Richard Reister, DOE Program Manager

Keith Leonard, LWRS Program Materials Aging and Degradation Pathway Lead

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Three Key Aspects of EPRI

Independent
Objective, scientifically based results address reliability, efficiency, affordability, health, safety and the environment

Nonprofit
Chartered to serve the public benefit

Collaborative
Bring together scientists, engineers, academic researchers, industry experts

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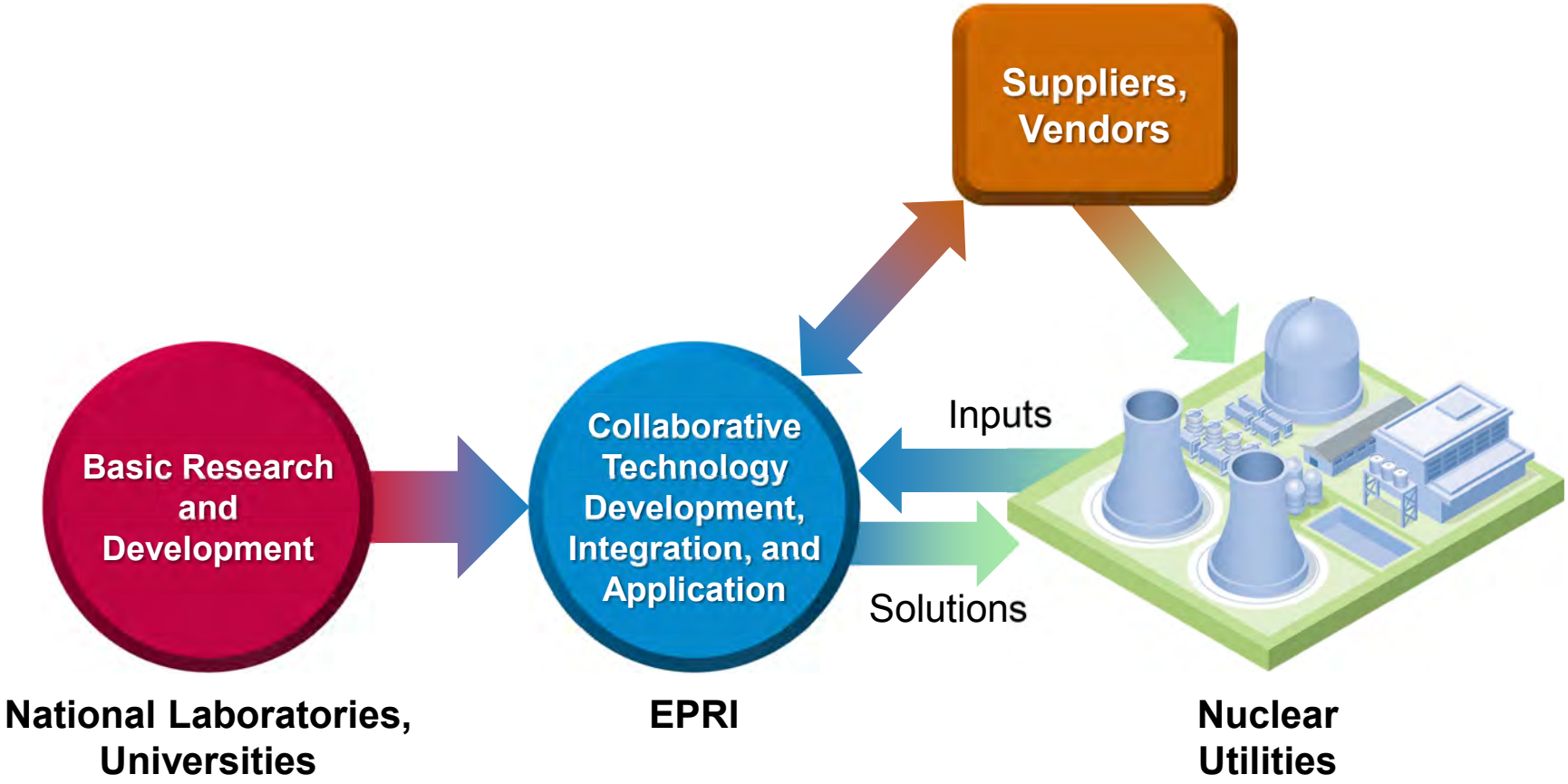
EPRI Collaborative Approach



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Nuclear Collaboration

U.S. Participants	Non-U.S. Participants	Global Breadth and Depth
All U.S. nuclear owners/operators 99 reactors	20 countries >220 reactors	>80% of the world's commercial nuclear units

We collect 1 YEAR of operating experience EVERY DAY!

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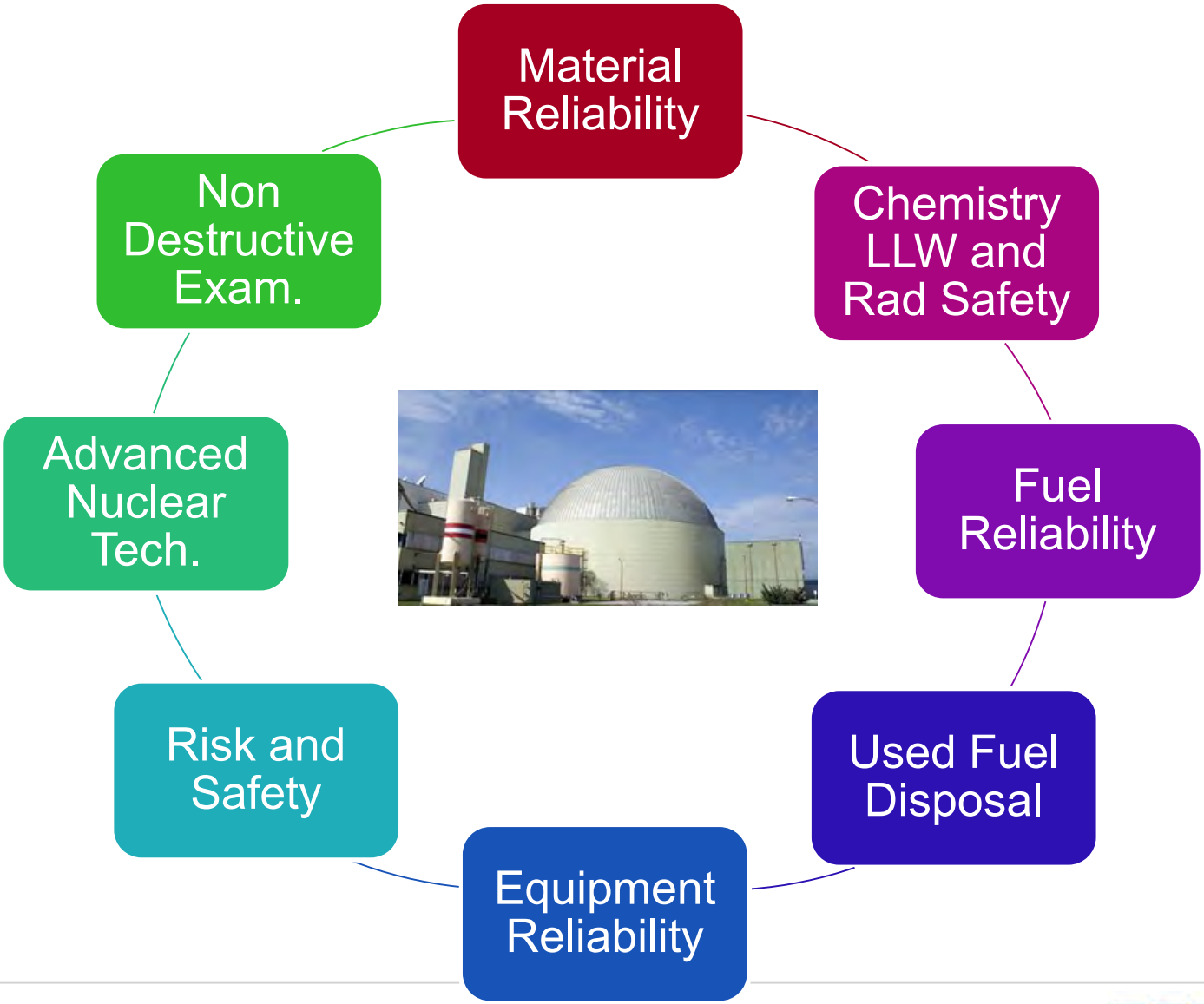
EPRI ELECTRIC POWER RESEARCH INSTITUTE

International R&D Partnerships

Active Collaboration with Global Research Institutes, National Laboratories, Universities and Vendors

Nuclear Sector Technical Areas ($170M/yr.)



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EPRI LTO Program Goals and Objectives

- Technical basis for ***decision*** to operate through extended life time
  - Supports business case for life extension and refurbishments

- Technology to ***manage*** plant assets throughout the lifetime
  - Aging management, asset management and risk management
  - Addresses safety, performance and costs

On track to provide program deliverables in the 2014 through 2019 timeframe.

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EPRI LTO R&D Projects

- Aging Management Research
  - RCS primary system metals
    - Advanced welding for highly irradiated materials –repair strategy
    - Advanced radiation resistant materials - future replacement strategy
  - Reactor pressure vessel embrittlement
  - Concrete and concrete structures
  - Electrical cable systems
- Opportunity for Modernization
  - Centralized on-line monitoring
  - Advanced I&C
  - Enhanced safety and risk analysis
- Enabling Technologies
  - Integrated Life Cycle Management
  - Plant Demonstration Projects

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U.S. DEPARTMENT OF ENERGY

Nuclear Energy

Light Water Reactor Sustainability (LWRS) Program

Program Goal

- Develop fundamental scientific basis to allow continued long-term safe operation of existing LWRs (beyond 60 years) and their long-term economic viability

Developing technologies and other solutions to

- Enable long term operation of the existing nuclear power plants
- Improve reliability
- Sustain safety

Focus areas

- Materials Aging and Degradation
- Advanced Instrumentation and Controls
- Risk-Informed Safety Margin Characterization
- Reactor Safety Technology

Nine Mile Point Courtesy Constellation Energy

U.S. DEPARTMENT OF ENERGY

Nuclear Energy

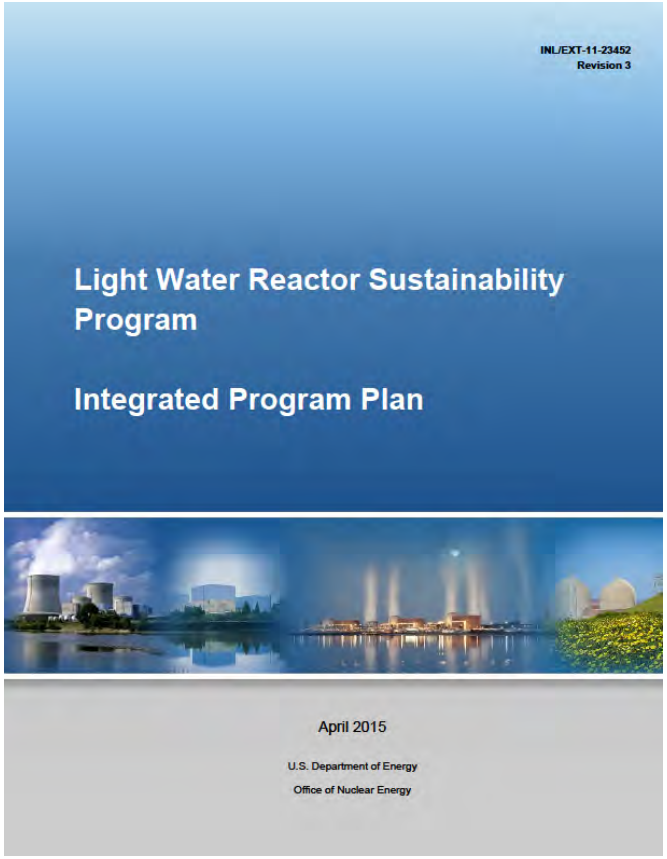
Light Water Reactor Sustainability – Federal Government Role

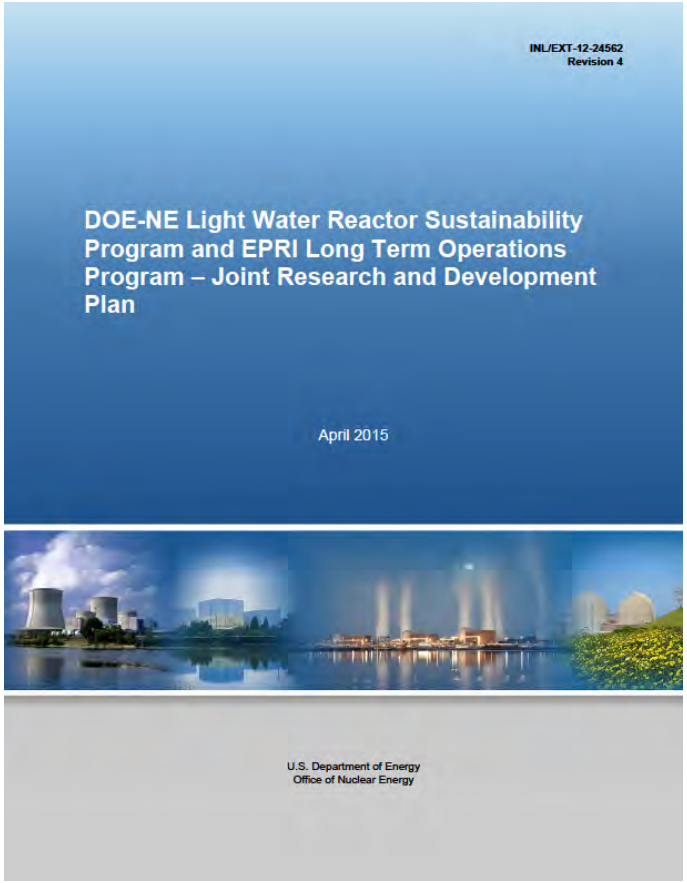
- National strategic interest in the long-term operation of existing plants
  - Supports the Administration's Climate Action Plan and Clean Power Plan
  - Supports energy security
  - Avoids higher cost to ratepayers for new plant replacements
- Addresses fundamental scientific questions where private investment or capabilities are insufficient to make progress on broadly applicable technology issues
- Government (DOE and its national laboratories) holds a large theoretical, computational, and experimental expertise in nuclear R&D
- Benefits will extend to next generation of reactor technologies
- Federal program reduces uncertainty and risk to provide incentives for industry to make long-term investments
- Industry also has an incentive, so cost-sharing is being employed

U.S. DEPARTMENT OF ENERGY

Nuclear Energy

LWRS Program Planning and Coordination/Integration





LWRS Planning Document available at www.inl.gov/lwrs

Materials Aging and Degradation

- **Metals: including Reactor Pressure Vessels, core internals, steam generators, and balance of plant**
  - Irradiation-Assisted Stress Corrosion Cracking
  - High-fluence phase transformations and swelling of core internals
  - High-fluence effects on RPV steel
  - Crack initiation in Nickel based alloys
  - Thermal Aging of Cast Austenitic Stainless Steels
  - Environmentally Assisted Fatigue
- **Concrete: Joint research plan with EPRI focused on radiation effects (supports and biological shield) and monitoring tools**
- **Cables: Joint research plan with EPRI and NRC to better predict and monitor cable aging**
- **Mitigation, repair, and replacement technologies: Weld repair techniques; Post irradiation annealing; Advanced replacement alloys; and Advanced Non-Destructive Examination techniques**

Materials Aging and Degradation

- ***Measurements of degradation:*** Collect high quality data
- ***Mechanisms of degradation***: Understand the underlying mechanisms for better prediction and mitigation
- ***Modeling and simulation***: Use mechanistic models to explore data trends for extended life
- ***Monitoring***: Monitor and validate predicted degradation
- ***Mitigation strategies***: Develop technologies to reduce the rate of degradation, facilitate economic repair, and potentially replace with advanced materials that are less susceptible

R&D Supports Aging Management Program Implementation

- R&D to understand aging degradation
  - Mechanism and failure modes
  - Initiation and growth rates
  - Inspection and Evaluation GLs
- Inspection methods
  - Detection and measurement
  - Non destructive examination and qualification
- Mitigation strategies
  - Chemistry
  - Stress relieving techniques

- Condition Monitoring
  - On-line monitoring
  - In-field detection
- Prediction of Remaining Useful Life
  - Health Monitoring software and algorithms
- Repair & Replacement Decisions
  - Life Cycle Management GLs
  - Integrated Life Cycle Management (ILCM)

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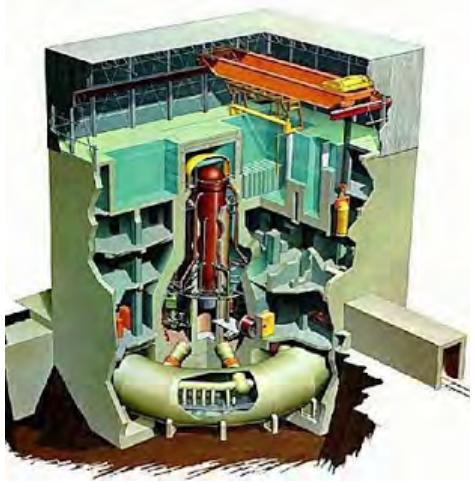
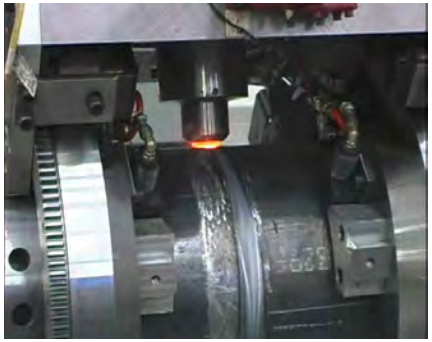
Conclusion from the Research



- ✓ Technical basis is established and in use for aging management
- ✓ Continuous improvements based on research results, inspections and operating experience
- ✓ Coordination and collaboration with global research partners



No show stoppers have been identified based on R&D to date.

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Reactor Pressure Vessel and Reactor Coolant System Research
Joint EPRI-LTO DOE-LWRS Presentation to ACRS
November 17, 2015

Robin Dyle

EPRI Senior Technical Executive

Keith Leonard

LWRS Program Materials Aging and Degradation Pathway Lead

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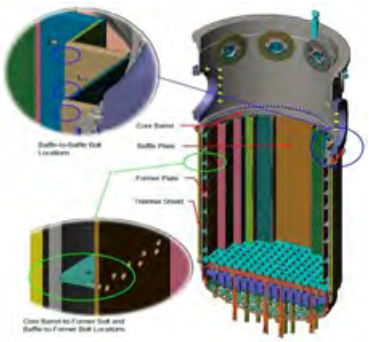
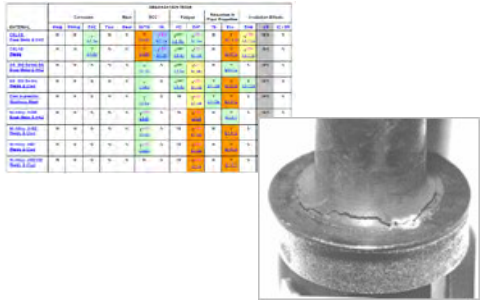
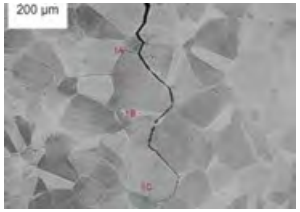
Materials Programs







MRP
PWR Materials Reliability Program

SGMP
Steam Generator Mgmt Program

BWRVIP
BWR Vessel & Internals Program

WRTC
Welding & Repair Technology Center

PSCR
Primary Systems Corrosion Research

Other

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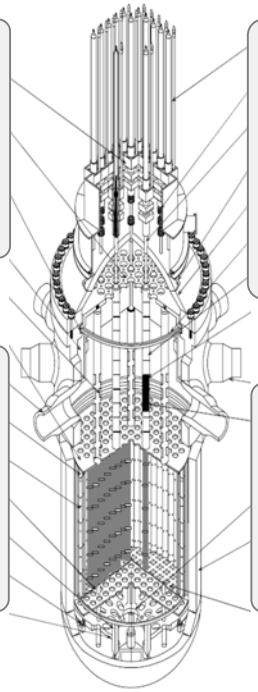
More Than Just Testing…

Assessment –

- What to inspect and when
- Inspection options
- How to dispose any observed degradation

Mitigation and Testing –

- How degradation can be prevented or reduced
- Irradiated material testing
- Non-irradiated material testing



Inspection –

- How to inspect
- What equipment & techniques are available
- What are the associated uncertainties
- What techniques need to be improved

Technical Support –

- Review of inspection results
- Guidelines review and maintenance

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GALL Aging Management Programs (AMPs) addressed by EPRI Materials Programs

- XI.M2 (AMP-103) – Water Chemistry
- XI.M4 (AMP-105) – BWR Vessel ID Attachment Welds
- XI.M5 (AMP-106) – BWR Feedwater Nozzle
- XI.M6 – BWR CRD Return Line Nozzle
- XI.M8 (AMP-108) – BWR Penetrations
- XI.M9 (AMP109) – BWR Vessel Internals
- XI.M11B (AMP-111) – Cracking of Nickel Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in RCPB Components
- XI.M12 (AMP-112) – Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)
- XI.M16A (AMP-113) – PWR Vessel Internals
- XI.M19 (AMP-116) – Steam Generators
- XI.M31 (AMP-118) – Reactor Vessel Surveillance

- The IGALL has additional AMPs that are covered by materials programs but there is not a direct correlation with the GALL AMPs

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Integrated Materials Issues Strategic Plan

- Provides Systematic Approach to Managing Materials Issues
  - Identify vulnerabilities
  - Assess condition (inspect & evaluate)
  - Mitigate degradation initiation and propagation mechanism
  - Repair or replace as required
- Approach Used:
  - Degradation Matrix and Issue Management Tables
    - Degradation Matrix and Issues Management Tables are maintained as living documents
- In 2010 added LTO ‘flags’ to the MDM
  - “late occurring” degradation modes that may not manifest in 40 or 60 years (e.g. void swelling)
  - Declining margins to degradation

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Systematic Approach

Materials Degradation Matrix

3002000628

MDM

Every component, every material, every potential degradation mechanism

- Mapped to 80 years of operation
- Covering BWR, PWR, and CANDU
- VVER in progress

Issue Management Tables

PWR: 3002000634
BWR: 3002000690
CANDU: In Progress

IMTs

Every issue to resolve identified and prioritized

- Covers BWRs and PWRs
- CANDU in progress

All of our plans are publicly available…

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Materials Degradation Matrix

MDM Covers:

- Component
- Material
- Degradation mechanism
- Gap in our understanding

	DEGRADATION MODE													
	Corrosion				Wear	SCC		Fatigue		Reduction in Fract Properties		Irradiation Effects		
MATERIAL	Wstg	Pitting	FAC	Foul	Wear	IG/TG	IA	HC	EAF	Th	Env	Emb	VS	IC / SR
C&LAS: Base Metal & HAZ	N	N	Y b1-3a	N	N	Y b1-6a	?LTO b1-7a	YIMP b1-8a	YLTO b1-9a	N	Y b1-11a	YLTO b1-12a	N/A	N
C&LAS: Welds	N	N	Y b1-3b	N	N	Y b1-6b	?LTO b1-7b	YIMP b1-8b	YLTO b1-9b	N	Y b1-11b	YLTO b1-12b	N/A	N
SS: 300 Series SS Base Metal & HAZ	N	N	N	N	N	Y b1-6c	N	YIMP b1-8c	YLTO b1-9c	N	Y b1-11c	N	N/A	N
SS: 300 Series Welds & Clad	N	N	N	N	N	Y b1-6d	N	YIMP b1-8d	YLTO b1-9d	Y b1-10d	Y b1-11d	Y b1-12d	N/A	N
Cast Austenitic Stainless Steel	N	N	N	N	N	Y b1-6e	N	N	YLTO b1-9e	Y b1-10e	Y b1-11e	N	N/A	N
Ni-Alloy: A600 Base Metal & HAZ	N	N	N	N	N	YLTO b1-6f	N	N	YLTO b1-9f	N	Y b1-11f	N	N/A	N
Ni-Alloy: A182 Welds & Clad	N	N	N	N	N	YLTO b1-6g	N	N	YLTO b1-9g	N	Y b1-11g	N	N/A	N
Ni-Alloy: A82 Welds & Clad	N	N	N	N	N	YLTO b1-6h	N	N	YLTO b1-9h	N	Y b1-11h	N	N/A	N
Ni-Alloy: A52/152 Welds & Clad	N	N	N	N	N	N	N	N	YLTO b1-9i	N	Y b1-11i	N	N/A	N

Global Relevance: 2013 revision included CANDU, 2014 VVER

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Example IMT – BWR Reactor Pressure Vessel

Table A-1 Reactor Pressure Vessel (Continued)

Components, & ID No.	Material	Degradation Mechanisms	Conseq. of Failure	Mitigation	Repair / Replace	I & E Guidance	Gaps
1.3-5 Feedwater Nozzle Safe Ends	Ni-Alloy *(Some safe ends - A600 with A82/ A182 welds)*	SCC: IG/TG Fat: LC-Env RiFP: Env Ext: Pitting	A	**Water Chemistry** BWRVIP-190 BWRVIP-225 **HWC Technology** BWRVIP-62R1 BWRVIP-156 BWRVIP-159 BWRVIP-219 BWRVIP-245 BWRVIP-248 **Oper. Changes** GE-NE-523-A71-0594 *(Low Flow Controller Oper.)* **Stress Improvement** BWRVIP-61 *(Weld Overlay, IHSI, MSIP)*	**ASME Sect. XI** IWA-4000 **Weld Overlay** CC-N-504-2 *(Vendor Controlled)*	**ASME Sect. XI** IWB-2500-1 *(BWRVIP-74-A)* **IGSCC** Generic Letter 88-01 BWRVIP-75-A BWRVIP-222 **LC-Env Fatigue** 1022873 *Break Loc. Postulation* 1022876 *Stress-Based Mon.* 1024995 *EAF Screening* 1025823 *EAF Calc Guidance* NUREG-1801-X.M1 NUREG-6260 NUREG-6909 **HC Thermal Fatigue** NUREG-0619 GE-NE-523-A71-0594	B-DM-03 B-DM-06 B-DM-09 B-AS-07 B-AS-27
1.3-6 Core Spray Nozzles	C&LAS *(SA-508, Cl 2)*	SCC: IG/TG Fat: LC-Env RiFP: Env Ext: Pitting	A	**Water Chemistry** BWRVIP-190 BWRVIP-225	**ASME Sect. XI** IWA-4000	**ASME Sect. XI** IWB-2500-1 *(BWRVIP-74-A)* **LC-Env Fatigue** 1022873 *Break Loc. Postulation* 1022876 *Stress-Based Mon.* 1024995 *EAF Screening* 1025823 *EAF Calc Guidance* NUREG-1801-X.M1 NUREG-6260 NUREG-6909	B-DM-06 B-DM-07 B-AS-07

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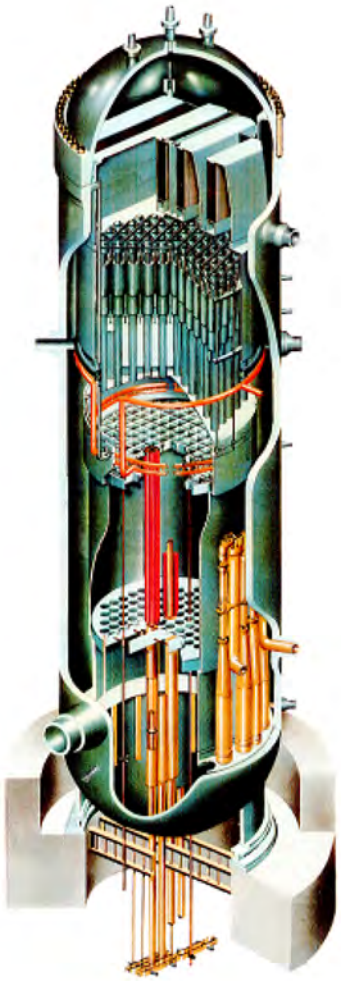
BWR Vessel Internals

- BWR Vessel Internals Program (BWRVIP)
- Issue: Prediction of SCC initiation and growth trends with increased neutron fluence and exposure to the environment
- BWRVIP is a living issue program with utility and OEM (GE) membership
  - Extensive R&D has been completed on modeling and understanding IASCC in the BWR environment
  - The AMP on reactor internals references 32 BWRVIP documents
  - BWRVIP documents are updated based on Operating Experience, inspection results and research



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BWRVIP Guidelines to Manage Degradation

Component	Assessment (I&E) Guidelines	Inspection Guidelines	Repair/Replace Design Criteria	Mitigation Recommendations
Core shroud	BWRVIP-76, R1	BWRVIP-03	BWRVIP-02-A/-04-A	BWRVIP-62, R1/-190
Core spray	BWRVIP-18, R2	BWRVIP-03	BWRVIP-16-A/-19-A/-34	N/A
Shroud support	BWRVIP-38	BWRVIP-03	BWRVIP-52-A	BWRVIP-62, R1/-190
Top Guide	BWRVIP-26-A	BWRVIP-03	BWRVIP-50-A	N/A
Core Plate	BWRVIP-25	BWRVIP-03	BWRVIP-50-A	BWRVIP-62, R1/-190
SLC	BWRVIP-27-A	BWRVIP-03	BWRVIP-53-A	BWRVIP-62, R1/-190
Jet pump assembly	BWRVIP-41	BWRVIP-03	BWRVIP-51-A	BWRVIP-62, R1/-190
CRD guide/stub tube	BWRVIP-47-A	BWRVIP-03	BWRVIP-17/-55-A/-58-A	BWRVIP-62, R1/-190
In-core housing/dry tube	BWRVIP-47-A	BWRVIP-03	BWRVIP-17/-55-A	BWRVIP-62, R1/-190
Instrument penetrations	BWRVIP-49-A	BWRVIP-03	BWRVIP-57-A	BWRVIP-62, R1/-190
LPCI coupling	BWRVIP-42-A	BWRVIP-03	BWRVIP-56-A	N/A
Vessel ID brackets	BWRVIP-48-A	BWRVIP-03	BWRVIP-52-A	BWRVIP-62, R1/-190
Reactor pressure vessel	BWRVIP-74-A	N/A	N/A	N/A
Primary system piping	BWVIP-75-A	N/A	N/A	BWRVIP-62, R1/-190
Steam dryer	BWRVIP-139-A	BWRVIP-03	BWRVIP-181	N/A
Access hole cover	BWRVIP-180	BWRVIP-03	TBD	BWRVIP-62-, R1-190
Top guide grid beam	BWRVIP-183	BWRVIP-03	BWRVIP-50-A	N/A
Bottom head drain line	BWRVIP-205	N/A	BWRVIP-208	N/A

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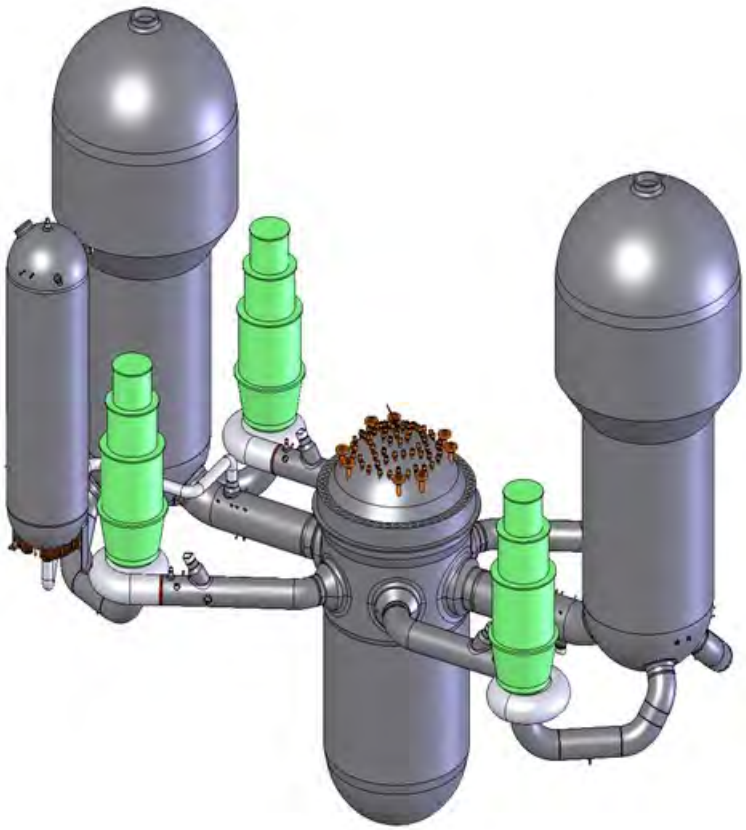
MRP Brief History

- PWR specific materials issues in the late 1990s led to the formation of the EPRI Materials Reliability Program (MRP) within the Nuclear Sector
- The objective of the MRP is to resolve existing and emerging PWR materials issues
- Key legacy deliverables:



Doc Number (EPRI PID)	Document Title
MRP-126 Rev 0 (1009561)	Generic Guidance for an Alloy 600 Management Plan
MRP-227-A (1022863)	Pressurized Water Reactors Internals Inspection and Evaluation Guidelines

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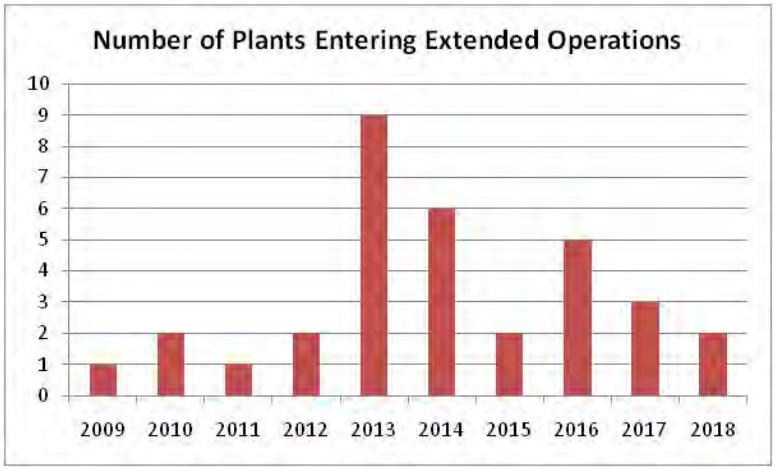
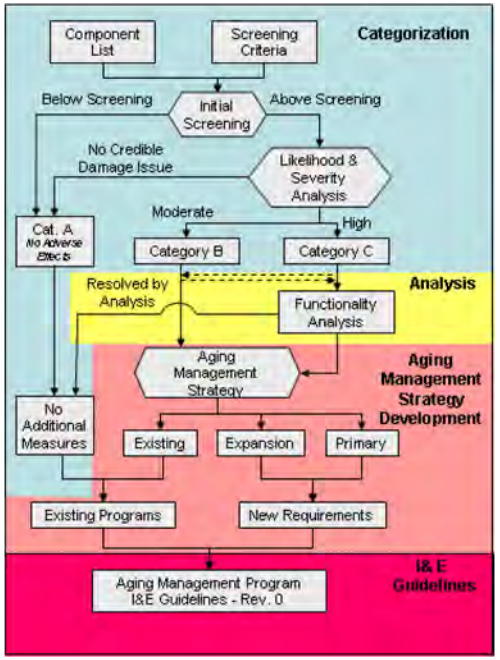
PWR Internals Inspection & Evaluation Guidelines (MRP-227)

- PWR Internals Inspection and Evaluation Guidelines (MRP-227, Revision 0) published Dec. 2008
  - Currently MRP-227-A
  - MRP-227, R1 published Nov. 2015
- Supports
  - License renewal commitments
  - Internals aging management
- Identifies
  - Components to be inspected
  - Timing of inspections





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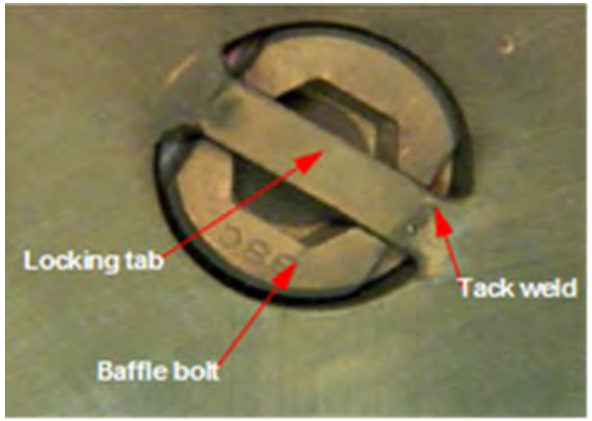
Summary of To Date PWR Internals Inspections

- 15 reactors inspected, others partly complete, as of Dec.2014
- Initial full inspections have not identified any major issues of concern
- Issues of greatest concern not seen
- Observations to date
  - Almost no observed austenitic SS stress corrosion cracking
  - Irradiation Assisted SCC is extremely limited
    - Currently limited to bolts
    - Explained in part by irradiation stress relaxation
  - No macroscopic effects of void swelling
    - No distortions, cracking, or excessive bolting failure
  - High-strength nickel alloy clevis insert bolting failures
    - However, not in safety function load path
  - Guide card wear will require monitoring
    - MRP Interim guidance MRP-2014-006 issued, added to MRP-227-1.



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SGMP Background

- The EPRI Steam Generator program (SGMP) started as a focused effort in 1975 and has continued to present under different arrangements such as owners group and co-funding collaborations

- In the 1970's and 1980's the emphasis was on finding reasons for troubling SG degradation (Leakage events were common)

- Starting in early 90's there has been a shift of emphasis from finding causes of SG degradation to managing SG degradation

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AMP 116 / GALL Chapter XI.M 19 Steam Generator Program

- For license extension, utilities are required by US Technical Specifications to establish a steam generator program
  - All US utility programs are modeled after NEI 97-06 which references the following six EPRI SGMP guideline documents
    - Steam Generator Integrity Assessment Guidelines
    - Steam Generator In-Situ Pressure Test Guidelines
    - Steam Generator Examination Guidelines
    - Steam Generator Primary-to-Secondary Leak Guidelines
    - PWR Primary Water Chemistry Guidelines
    - PWR Secondary Water Chemistry Guidelines
- These guidelines incorporate a balance of prevention, mitigation, inspection, evaluation, repair and leakage monitoring
- These same 6 guidelines are referenced in AMP 116 & GALL Chapter XI.M 19

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Current Status

- A review of the existing Aging Management Programs (AMPs) was performed – EPRI 3002000576, August 2013
- That review identified:
  - AMPs for which no change was needed – 22 AMPs
  - AMPs where additional work could enhance areas well understood – 20 AMPs
  - AMPs where supporting work was needed to address knowledge gaps for 60 to 80 years of operation, to better understand and manage materials performance – 8 AMPs
- Work is underway to enhance the AMPs for longer periods of operation
- However, while a few knowledge gaps were identified, the current AMPs are adequate to assure safe operation for as long as the plants operate in compliance with the existing guidance.

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Reactor Pressure Vessel

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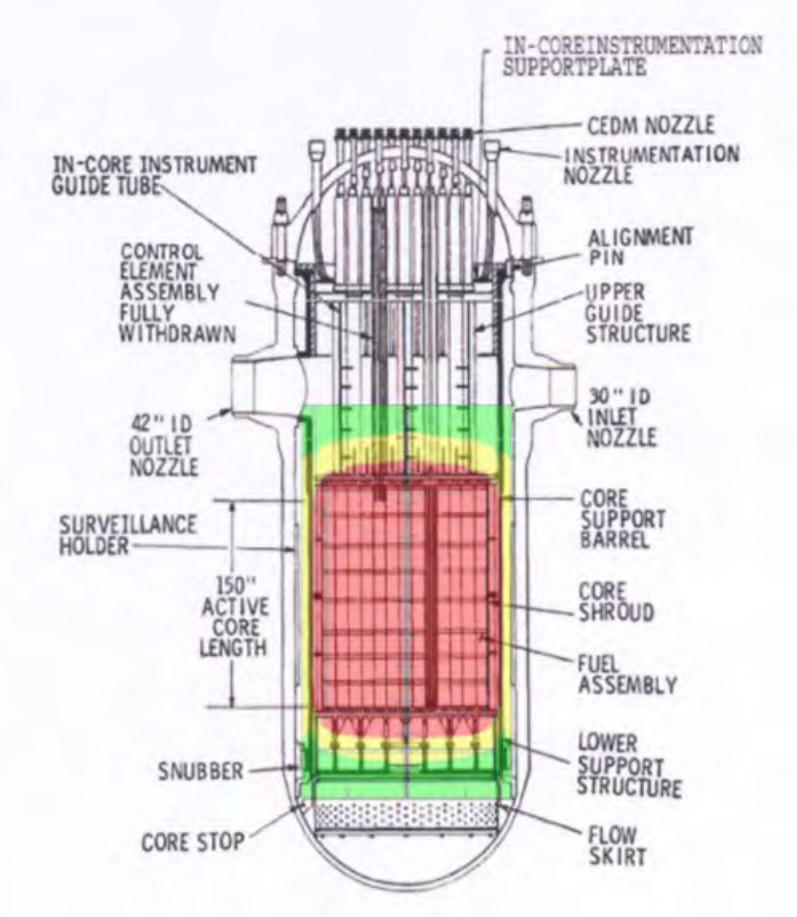
Reactor Vessel Integrity

- Issue: Need to monitor fracture toughness of the RPV and nozzles due to irradiation and other potential degradation mechanisms
- BWRVIP Integrated Surveillance Program (ISP)
- MRP surveillance programs
  - Coordinated RV Surveillance Program (CRVSP)
  - PWR Supplemental Surveillance Program (PSSP)
- MRP reports
  - CRVSP (MRP-326)
  - Low upper shelf energy (USE) equivalent margins analysis (EMA) for LTO (MRP-366)
  - Material selection for PSSP (MRP-364)



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Ongoing/Future Efforts on RPV

- Extension or replacement of the BWRVIP ISP
- PWR Coordinated RV Surveillance Program (CRVSP)
- PWR Supplemental Surveillance Program (PSSP)
- Atomic Probe Tomography (APT) of irradiated samples
- Collaboration with DOE-LWRS/ORNL for testing PWR surveillance materials in ATR-2 capsule
  - Research to develop models to accurately predict TTS for high fluence / low flux RPV conditions using slightly accelerated irradiation
- Support development of ASME Code Case (N-830, R1) to expand use of Master Curve Technology for RPV Integrity Evaluation

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Background: Need for High Fluence Data

- Embrittlement Trend Correlations (ETC) are used to predict transition temperature shift (TTS) of reactor vessel low alloy steels (LAS)
- To support RPV operations to 60 or 80 years, an Embrittlement Trend Correlation (ETC) that is informed by high fluence transition temperature shift (TTS) data is required
- Currently, power reactor surveillance data fully representative of 60 or 80 years of operation are limited
- Additional data are needed to validate or revise ETCs applicable to the high fluence regime before plants reach 60 years of operation

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MRP Programs to Generate High-fluence RPV Data

- Coordinated Reactor Vessel Surveillance Program (CRVSP)
  - Defer withdrawal and testing of scheduled capsules in order to test them at higher fluences
  - CRVSP has been fully implemented
- PWR Supplemental Surveillance Program (PSSP)
  - Design, fabricate, irradiate and test 2 supplemental capsules
  - This program is currently in progress

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Coordinated Reactor Vessel Surveillance Program (CRVSP)

- Objective: Obtain as much high fluence PWR surveillance data (3-10 $x10^{19}$ n/cm^2) as practical by 2025 by deferring scheduled withdrawals and testing at higher fluence
- Development
  - Each U.S. PWR reactor vessel surveillance program (RVSP) was reviewed
  - Capsules were selected for program if, by deferring withdrawal, data at fluence > 3 E+19 n/cm^2 could be achieved by 2025
  - 13 capsules at 13 PWRs were identified; those plants formally changed capsule withdrawal schedule – data available as capsules are tested between now and 2025
- Program has been implemented but 2 capsules lost due to early plant closures
- *Materials Reliability Program: Coordinated PWR Reactor Vessel Surveillance Program (CRVSP) Guidelines (MRP-326)*. EPRI, Palo Alto, CA: 2011. 1022871.

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PWR Supplemental Surveillance Program (PSSP)

- EPRI PSSP will Design/Fabricate/Irradiate 2 supplemental surveillance capsules containing previously-irradiated PWR materials
  - Reconstitute previously-irradiated specimens (per ASTM E1253) before re-irradiation
  - Obtain 25 new high-fluence Charpy transition temperature shift measurements
- Materials were selected for value to inform a future ETC, by filling specific material/fluence gaps in the PWR surveillance database
- Capsules will be inserted into 2 host PWRs (Shearon Harris and Farley 1) in late 2016
  - Irradiate ~10 years, test in 2026-27
  - Flux ~1.2 E+11 n/cm^2/s (~0.35 n/cm^2/year) thus adding ~3.5 E+19 n/cm^2 over 10 years
  - Two irradiation temperatures to more closely match previous irradiation temperature (~ 282°C, 291°C)

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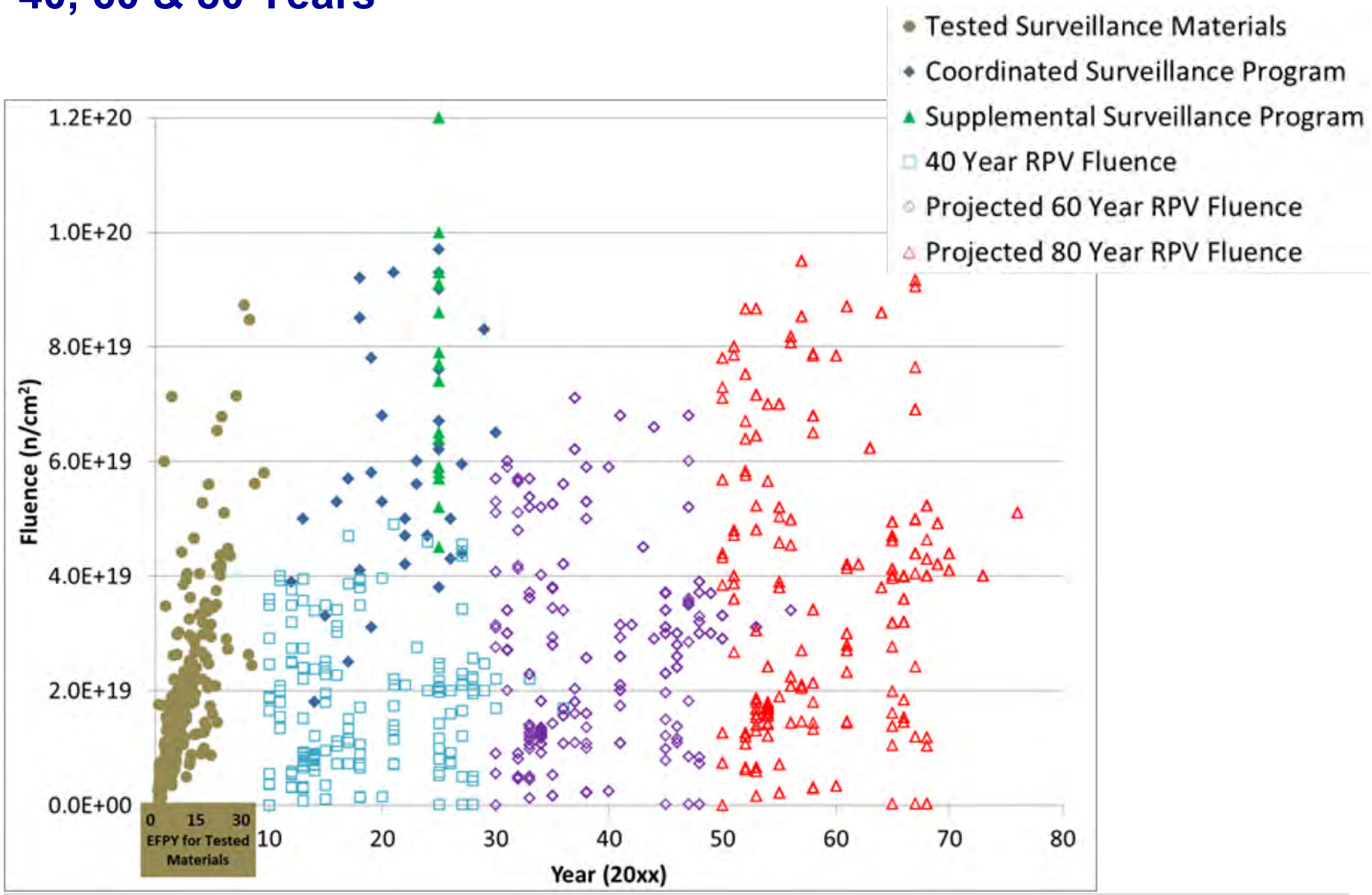
Timeline for PWR Surveillance Data vs. Peak RPV Fluences for 40, 60 & 80 Years



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High Fluence TTS Data from CRVSP and PSSP

Fluence (n/cm^2)	Number of TTS Data Before CRVSP	TTS Data After Implementation of the CRVSP[1]	Maximum Possible Data from the PSSP	Total (CRVSP+PSSP)
$\geq 3.0 \times 10^{19}$	50	58	25	83
$\geq 6.0 \times 10^{19}$	14	24	13	37
$\geq 8.0 \times 10^{19}$	2	10	6	16
$\geq 9.0 \times 10^{19}$	0	4	4	8

[1] Assumes two data points per capsule (one weld and one base metal) from the CRVSP capsules

The CRVSP data consists of weld and plate data from capsules, but only one of the capsule materials may provide unique value to inform an ETC for 80 years. In contrast, each PSSP material was selected for its unique value to inform a robust ETC for extended operations. Therefore, PSSP data – though fewer – will have a high qualitative impact.

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MRP Participation in ATR-2 Test Reactor Irradiation Experiment

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ATR-2 Experiment

- ATR-2 is an irradiation experiment designed (primarily) by University of California Santa Barbara (UCSB) (Dr. Robert Odette)
- Irradiation conducted at the Advanced Test Reactor of Idaho National Laboratory (INL)
- Experiment is a collaboration of the following participants:
  - UCSB, INL, Oak Ridge National Laboratory (ORNL) and the U.S. Department of Energy (DOE) Light Water Sustainability Program (LWRSP), CRIEPI (Japan), Bettis, Rolls Royce (UK), EPRI
- Objective: Research to develop models to accurately predict TTS for high fluence power reactor conditions using slightly accelerated irradiation
- 180 materials, 1661 specimens
- Irradiation started in 2011 and is completed; post-irradiation examination (PIE) started 2015

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PWR Surveillance Materials for ATR-2

- During planning stage in 2010, MRP obtained permission from several PWRs to include 9 archive (unirradiated) PWR surveillance materials in the ATR-2 capsule
- All of these materials have TTS data from the plant surveillance programs that can be compared directly to the test reactor results
  - Quantify flux effect, if any
- Many of same materials are included in CRVSP and PSSP and EPRI-CRIEPI APT programs
- This integration of data for common materials irradiated in test and power reactors at different fluxes will enhance understanding of material behavior in various environments
- In 2015 and 2016 MRP is supporting the testing and evaluation of the PWR surveillance materials from the ATR-2 experiment

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U.S. RPV Surveillance Steels in PSSP and Other Cooperative Research Programs

Material	Heat Number	Cu (wt%)	Ni (wt%)	ATR-2 Irradiation ($\phi t \leq E20$)	MRP PSSP (expected ϕt)	Highest PWR Surveillance (expected ϕt)	CRIEPI-EPRI Atom Probe (Surveillance ϕt)	ORNL AP (Surveillance ϕt)
Shielded Metal Arc Weld	BOLA	0.03	0.9	Yes	1.2E20	8.73E19	8.73E19	--
Linde 124 Weld	4P4784	0.04	0.95	Yes	1.0E20	6.54E19	--	--
Linde 1092 Weld	1P3571	0.22	0.72	Yes	9.1E19	5.62E19	5.62E19	--
Linde 1092 Weld	1P3571	0.36	0.78	Yes	--	6.11E19	6.11E19	--
Linde 0091 Weld	33A277	0.14	0.19	Yes	1.2E20	8.47E19	8.47E19	--
Linde 80 Weld	71249	0.29	0.6	Yes	--	1.29E19	--	--
Linde 80 Weld	61782	0.24	0.52	--	9.3E19	5.8E19	--	1.7 - 5.8 E19
SA533B-1 Plate	C7466-1	0.19	0.6	Yes	1.2E20	8.73E19	8.73E19	--
SA533B-1 Plate	B9004-2	0.05	0.56	Yes	9.1E19	5.6E19/ 8.5E19	--	--
SA508-2 Forging	125P666	0.05	0.69	--	9.3E19	5.8E19	--	1.7 - 5.8 E19
SA508-2 Forging	123X167VA1	0.06	0.75	Yes	--	5.62E19	5.62E19	--

Not all PSSP materials are shown; only those which are also in at least one other program.

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BWR Surveillance for SLR

- All U.S. BWRs participate in the BWRVIP Integrated Surveillance Program (ISP)
  - Integrated surveillance program plan is approved through 60 years: BWRVIP-86, Rev. 1-A
  - 6 capsules to be tested from 2015 to 2018
  - Next capsule in LR period is in 2026
- Beginning 2015 BWRVIP is conducting a project to define options for addressing BWR surveillance needs for 80 years

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Irradiated Stainless Steel Crack Growth Model

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Main Tasks

- Compile crack growth rate (CGR) data on irradiation-assisted stress corrosion cracking (IASCC) of austenitic stainless steels in light water reactor (LWR) environments from EPRI, NRC, Halden and other international programs
- Convene an Expert Panel to review and rank the available CGR data on irradiated materials using appropriate criteria
- Develop crack growth rate models and reference curves using the better-ranked data, for application to BWR and PWR internals
- This work was coordinated between PSCR, BWRVIP and MRP Programs at EPRI

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Initial Data

- Compiled more than 1600 test segments from 6 Labs, on 37 different heats, tested in LWR environments under static and cyclic loading
- Over 800 test segments had CGR under quasi-static loading (constant load, constant K, mild gradients in K, or periodic partial unloading (PPU) with long hold times) and were reviewed for possible use in IASCC models
- An Expert Panel ranked the test segments on a scale from 1 (best) to 5 (unsuitable) after examining detailed experimental data, including crack length vs. time plots
- Data with average ranking < 3 were used for development of IASCC crack growth models for BWR NWC & HWC and PWR primary water environments

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Final Data

- Size of Low-ECP set ranked < 3
  - 187 CGR observations in proposed model
    - 73 in HWC
    - 114 in PWR primary water
  - 22 heats (+ BS03/BR03)
  - 52 specimens
  - Tested in 6 laboratories
- Size of NWC set ranked < 3
  - 282 CGR Observations
  - 22 Heats
  - 56 Specimens
  - Tested in 5 laboratories

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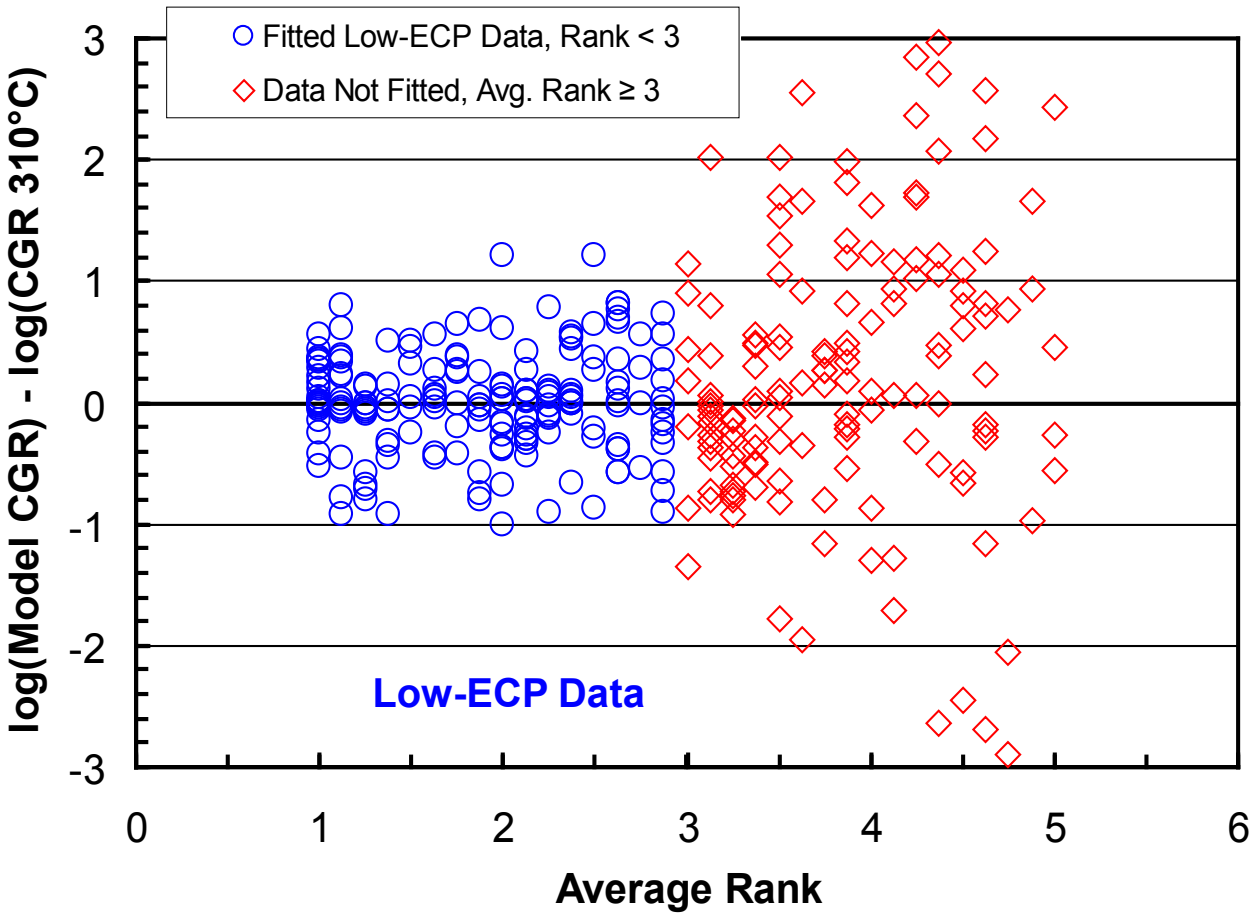
Data for Model Calibration Have Average Ranking < 3



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High and Low ECP Models

- Overall form:

$$\frac{da}{dt} = C_i \cdot f(PPU) \cdot \sigma_{0.2}^{v} \cdot \exp\left[-\frac{Q}{R}\left(\frac{1}{T} - \frac{1}{T_{ref}}\right)\right] \cdot K^{u}$$

  - *da/dt* is crack growth rate (mm/s)
  - C_i is a material & environment coefficient calibrated to each heat
  - *f* (PPU) partial periodic unloading constant = CGR_{PPU}/CGR_{CL}
  - $\sigma_{0.2}$ is the irradiated yield stress at the temperature and dose of the IASCC test specimen (MPa)
  - *Q* is the activation energy (kJ/mol)
  - *R* is the universal gas constant 8.314 J/(mol-K)
  - *T* is the SCC test or model application temperature
  - T_{ref} is a reference temperature at which the model is calibrated
    - T_{ref} = 583.15 kelvin (310°C) for low-ECP data
    - T_{ref} = 561.15 kelvin (288°C) for NWC data
  - *K* is the stress intensity factor, (MPa√m)
  - u and *v* are fitted exponents

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Expert Panel: Crack Growth Models For BWR and PWR Internals

- The models account for effects of irradiated yield stress (dose), stress intensity K, ECP, temperature, type of loading (constant load vs. PPU)
- There was no significant difference in the CGRs between BOR 60 fast reactor and LWR irradiated materials after accounting for the above factors
- Analysis showed that a common model can be used for BWR-HWC and PWR environments with a temperature term to account for higher PWR temperatures
- After accounting for above parameters there is still a large heat-to-heat (and specimen-to-specimen) variability
- A large heat-to-heat variability also has been observed in unirradiated Ni-base alloys (MRP-55 and MRP-115)
- This variability is reflected in heat coefficients in the models

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Project Conclusions

- Stronger technical basis for the new disposition curves for IASCC CGR than BWRVIP-99-A because:
  - they are based on much larger data sets over a wider range of K and dose (especially for low-ECP)
  - they include effect of additional variables T & irradiated $\sigma_{0.2}$ (dose)
  - they allow extension of models to higher fluence and temperatures
- The new mean models are good fits to the data
- The new 75th percentile curves are similar to the older curves if evaluated at the same T & particular values of irradiated $\sigma_{0.2}$ (dose)
- There is large heat-to-heat variability about the final models; further study of causes might improve the models
- The new curves can be used to analyze local variation in dose, such as attenuation in thick components
- The new 75th percentile curves are conservative relative to laboratory data not used for fitting and relative to field inspection data

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Project Status

- Published Report on “Models of Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments: 3002003103, Volumes 1 and 2.
- Presented a paper based on the Volume 2: Disposition Curves Application at the ASME PVP Conference in Boston in July 2015
- Presented a paper based on Volume 1: Disposition Curves Development at the 17the EDEG meeting in Ottawa in August 2015
- These two papers together with a publication in the ASME Journal of PVP Technology in 2016 will provide the technical basis to support the ASME code case
- Initiated the code case at the meeting of the Section XI Task Group on Crack Growth Reference Curves in San Francisco in August 2015
- The ASME code case effort will be jointly supported by BWRVIP and MRP

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Irradiated Materials Testing

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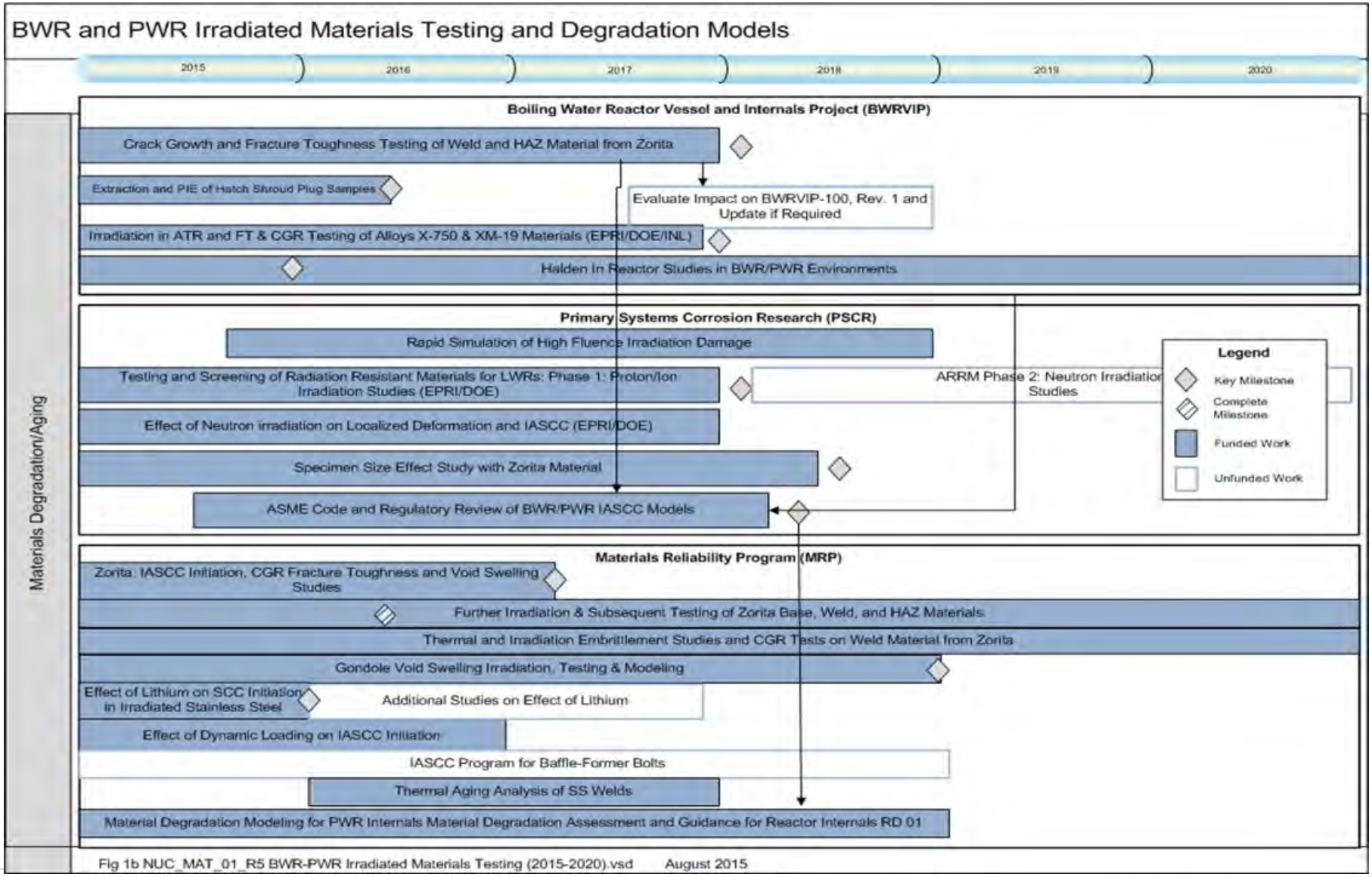
BWR and PWR Irradiated Materials Testing and Degradation Models for Reactor Internals Roadmap



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PWR Testing

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Irradiated Materials Testing
MRP Testing Programs

	Project Title
Zorita Materials Projects	• Zorita Internals Research Project • Thermal and Irradiation Embrittlement and Environmental Effect Testing of Stainless Steel Welds • CGR Testing of Irradiated Stainless Steel Weld and Heat Affected Zone Materials • Determination of IASCC CGR, Initiation Rate and Void Swelling in Zorita Material after Post-Reactor Irradiation • Thermal Aging Analysis of SS Weld Material at High & Low Neutron Irradiation Dose
Collaborative Projects	• Crack Growth Tests in Halden under PWR Environment • GONDOLE Void Swelling Irradiation and Testing
Ongoing Projects	• Effect of Lithium on the SCC Initiation in Irradiated Stainless Steel • Dynamic Strain Effects on IASCC Initiation Rates
Proposed Projects	• IASCC Testing Program for Baffle-Former Bolts

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Irradiated Materials Testing
MRP Testing of Zorita Materials

Advantages of using Zorita materials

- Structural materials after a much greater life/fluence combination than previously investigated (26.4 EFPY)
- In-situ PWR irradiation exposure (as opposed to accelerated test reactor irradiations, or BWR exposure)
- Thick-section material allows proper assessment of plant structural material behavior (including gamma heating effects)
- Sufficient volume of material to develop a detailed, systematic material property database including tensile, fracture toughness, crack initiation and CGR data

MRP is working cooperatively with the U.S. NRC to characterize these materials

- Participants are being sought to ensure a technically-robust testing matrix can be pursued

Weld #1

Weld #2

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Irradiated Materials Testing
MRP Projects Using Zorita Materials

Zorita Internals Research Project

- Increased understanding of irradiation effects on mechanical and microstructural properties of stainless steels
- Tensile testing is complete
- Crack growth rate (CGR) testing complete for one specimen each at 10, 30, and 50 dpa; overall CGRs low
- Crack initiation testing underway; scheduled for 9,000 hours
- Fracture toughness testing of 10 dpa specimens in air complete; testing of 10 dpa in water up next
- Microstructural testing underway at MHI; expected completion 2Q2016

Testing of Zorita Weld and HAZ Materials

- Determine the combined effects of irradiation and exposure to elevated temperature on embrittlement of stainless steel welds and characterization of environmental effect on fracture toughness in irradiated stainless steel welds
- Generate IASCC CGR data in irradiated stainless steel weld and HAZ materials for comparison to existing data for base materials
- Machining of specimens underway
- Fracture toughness testing at room temperature, elevated temperature, and intermediate shutdown temperature to begin in 2015
- CGR testing to be conducted according to ZIRP plate material protocol: 2 stress intensity levels and 3 temperatures

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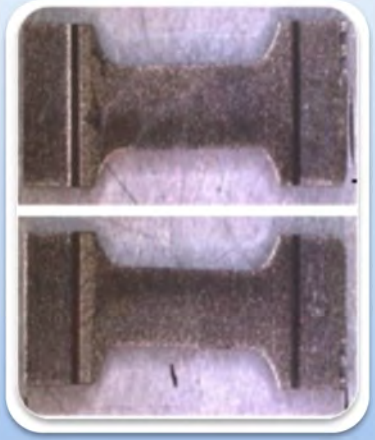
Irradiated Materials Testing
MRP Projects Using Zorita Materials

Determination of IASCC CGR, Initiation Rate, and Void Swelling in Zorita Material after Post-Reactor Irradiation

- Evaluation of IASCC crack initiation and crack growth rates and degree of void swelling in highly-irradiated (near end-of-life conditions) stainless steel base metal and welds
- Testing and additional irradiation of weld and HAZ material to begin in early 2016 as part of Halden Research Program
- Additional irradiation of baffle plate may begin in 2017 pending identification of suitable irradiation source



Thermal Aging Analysis of Stainless Steel Weld Material at High and Low Neutron Irradiation Dose

- Study the combined effect of thermal aging and irradiation on cast and welded stainless (austenite/ferrite) solidification structures
- Project uses undeformed material and deformed material from tensile specimens previously tested in other programs
- Characterize precipitates, phase boundary compositions, and distributions along with strain distributions and localized deformations.
- Program begins in 2016

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Irradiated Materials Testing
MRP Collaborative Projects

Crack Growth Tests in Halden under PWR Environment

- Generation of IASCC CGR data in irradiated stainless steel materials in a variety of PWR conditions (effects of hydrogen, lithium, zinc additions, etc.)
- Continued participation in Halden Research program
- Current program planned for 2015-2017 and includes Zorita baffle plate, weld, and HAZ materials

Gondole Void Swelling Program

- Final irradiation cycle underway (total accumulated dose ~30 dpa)
- TEM analyses to be conducted on irradiated specimen foils and from density specimens
- Preliminary results: only pre-irradiated specimens show swelling greater than 0.5%; several specimens show negative swelling greater than -0.5%

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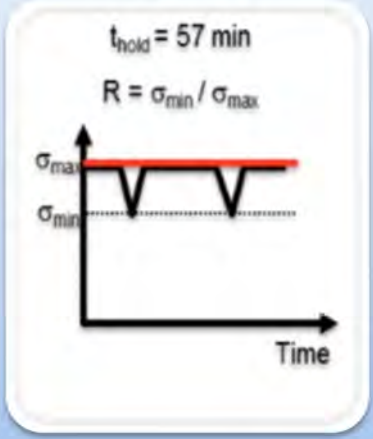
Irradiated Materials Testing
MRP Ongoing Projects

Effect of Lithium on IASCC Initiation

- Determine the effect of lithium (Li) on the rate of IASCC initiation for comparison to recent data generated by EDF suggesting increased Li concentration may enhance IASCC initiation rate
- Testing of O-ring and tensile-type specimens at both 2.0 ppm and 8.0 ppm is complete
- Results suggest an influence of lithium concentration on initiation rate
- Additional work being proposed for late 2016 or 2017



Dynamic Strain Effects on IASCC Initiation Rates

- Compare existing results from static-loaded tests to tests conducted using dynamic loads representative of PWR transients to better understand EDF baffle bolt experience and IASCC test observations
- EPRI report on characterization of transients affecting U.S. PWR fleet issued in 2014
- Testing by CEA and SCK-CEN suggest time to initiation decreases in the presence of transients
- For 2016, EPRI to calculate changes in loads on reactor internals bolting materials resulting from transients

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Irradiated Materials Testing
MRP Proposed Project

IASCC Testing Program for Baffle-Former Bolts

- Determine the stress and dose dependence of IASCC in baffle-former bolt materials and increase understanding of the unanticipated baffle bolt failures at a U.S. PWR
- Optimize aging management strategy (inspection frequency/scope, replacement recommendations) for bolt materials for long-term operation
- Possible tie-in with dynamic strain effects testing program

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Irradiated Materials Testing
Recent Findings

- Effect of Lithium on IASCC Initiation
  - Testing of O-ring and tensile-type specimens at typical research level (2.0 ppm) and elevated level is complete
  - Results suggest an influence of lithium concentration on initiation rate
    - Findings similar to those from EdF which demonstrated initiation occurs at shorter time and/or at lower loads
- GONDOLE Void Swelling Program
  - Density specimens: only pre-irradiated specimens show swelling greater than 0.5%; several specimens show negative swelling greater than -0.5%
  - Gradient specimen: no significant variation in original dimensions

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BWR Testing of Zorita Materials

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Introduction

- From late 2014 to 2018, BWRVIP will conduct crack growth rate and fracture toughness testing of irradiated materials retrieved from the Zorita reactor
  - Testing will cover base metal, HAZ and weld metal from a core barrel weld, as well as base metal from a baffle plate
- This presentation describes:
  - The materials to be tested
  - Fabrication of specimens
  - Testing plans

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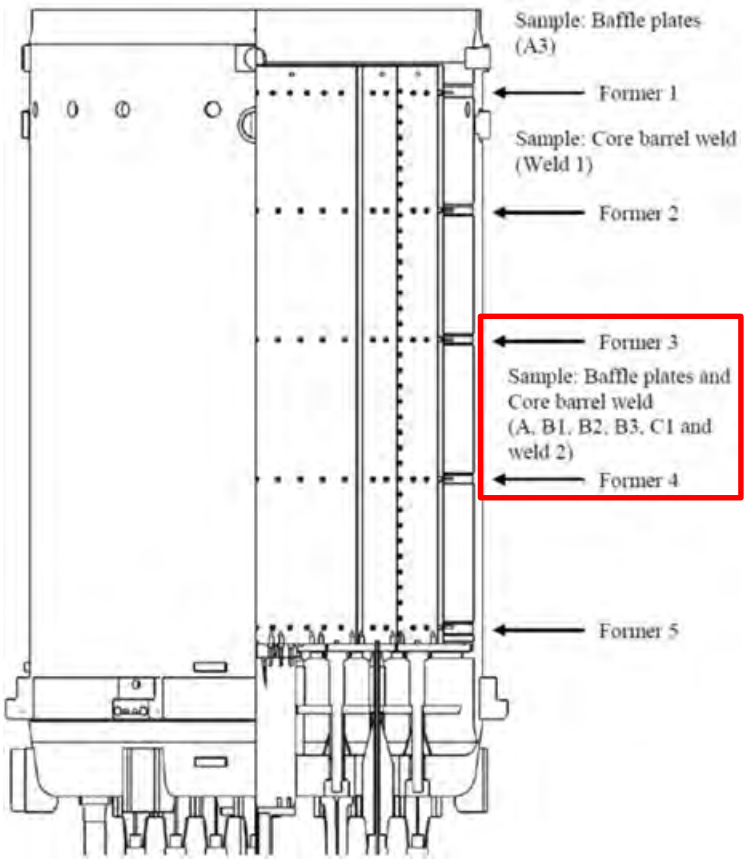
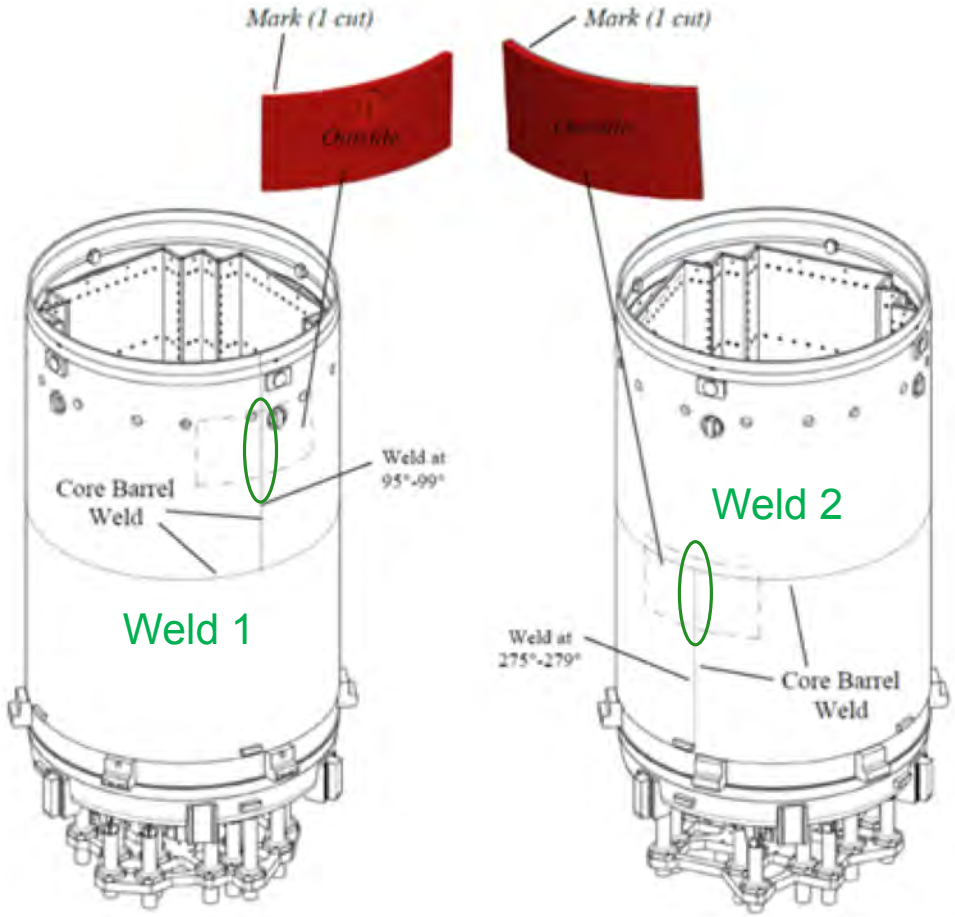
Zorita Materials – Weld 1 and Weld 2





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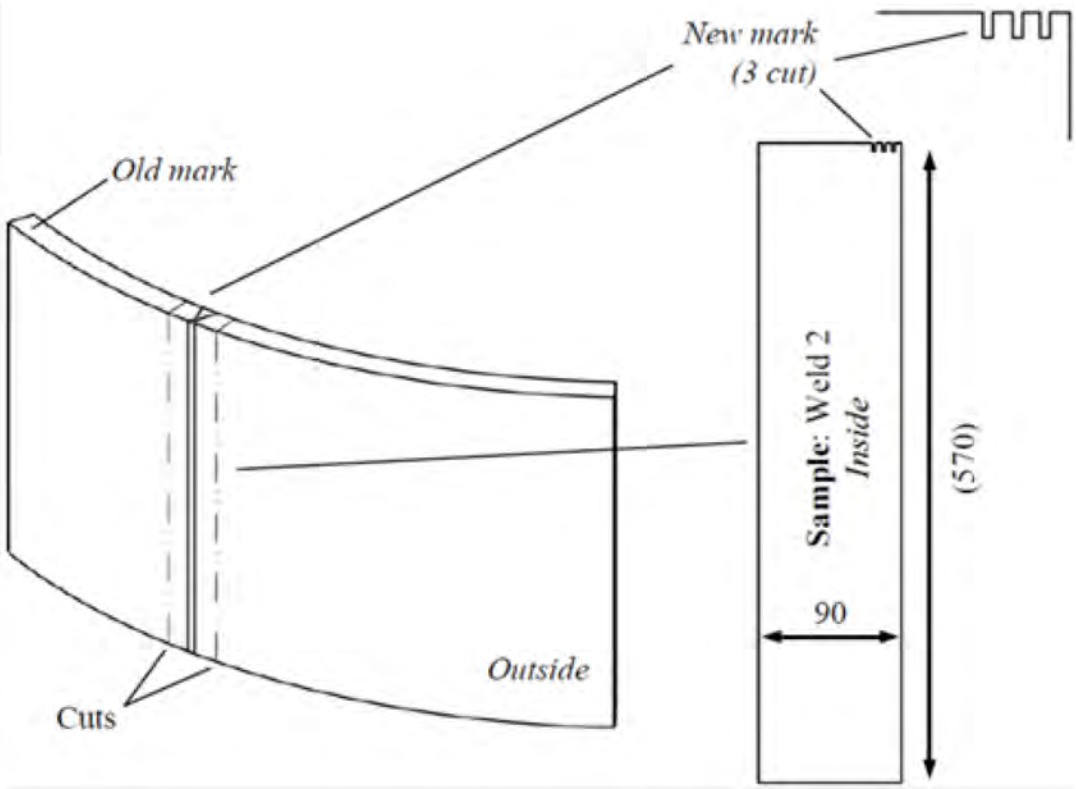
Zorita Material – Weld 2
2 dpa (~1.4E21 n/cm^2)



Inside

Outside

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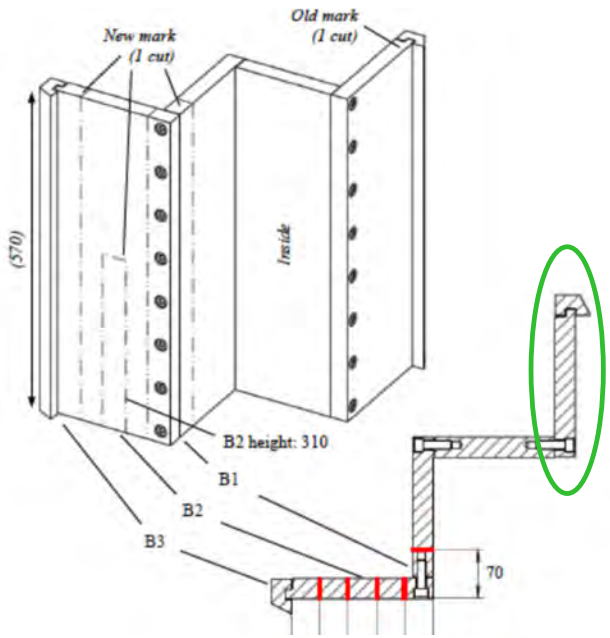
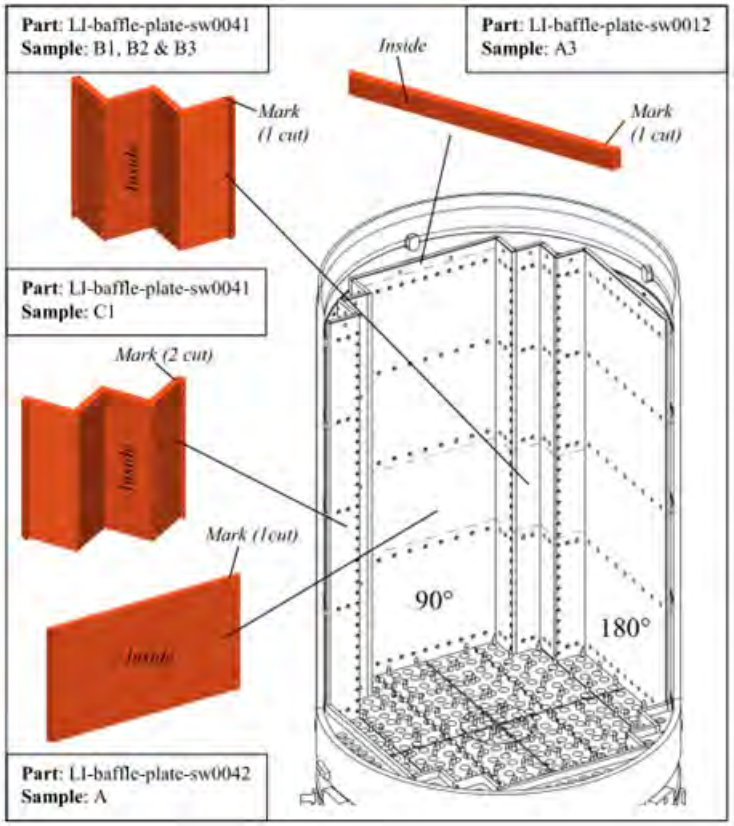
Piece B3 – Baffle Plate
10 dpa (7E21 n/cm^2)





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Fracture Toughness Testing

- Previous fracture toughness testing (published in BWRVIP-154, Rev. 2) mainly covered Types 304, 304L and 316L base metal
  - One 316L weld metal specimen at 6.7 dpa tested
- Material from the Zorita core barrel allows testing of HAZ and weld metal and using larger specimens as compared to existing BWR weld data
- Characterization of Materials
  - Tensile properties
  - Weld geometry to be documented by macro etching
    - Need information to properly orient specimens relative to weld
  - Metallographic cross section to assess rolling direction
    - Previous testing indicate inclusions and their orientation (L-T vs. T-L) is important for the fracture toughness
    - Assess possible variation in mechanical properties by hardness measurements

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Crack Growth Rate Testing

- Previous crack growth rate testing (published in BWRVIP-153, -172, -221, -232 and -265) mainly covered Types 304, 304L and 316L base metal
  - Limited testing on SS HAZ and weld metal at low dose (~1 dpa)
- Material from the Zorita core barrel allows testing of HAZ and weld metal at higher dose than previously tested
- Characterization of Materials
  - Tensile testing at 288°C (duplicate specimens) to determine the mechanical properties of base metal, HAZ and weld metal
  - Weld geometry to be documented by macro etching
    - Need information to properly orient specimens relative to weld
  - Metallographic cross section to assess rolling direction
  - Document microstructure at location of crack path
  - Assess variation in mechanical properties by hardness measurements

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Test Plan and Status

- Three specimens of Weld 2
  - Base metal, HAZ, and weld metal
- Test protocol
  - Three K levels, starting at 11 MPa√m and ending at ~25 MPa√m
  - Testing in NWC and HWC as in previous tests
- Status
  - Preliminary cutting plan completed
  - Number and orientation of specimen defined
  - Machining of specimens is in progress
  - CGR and FT testing well underway
  - Draft report on FT testing to be published in 2015

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BWR Core Shroud Boat Sample

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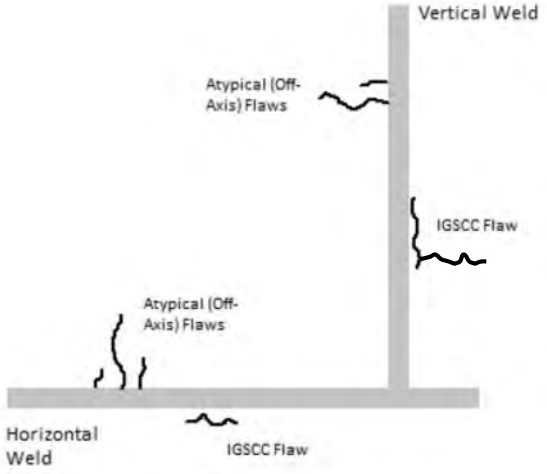
Background:
"Off-Axis" Cracking – what is it?

- IGSCC is a commonly observed phenomenon in stainless steel BWR reactor internals
  - Cracking propagates parallel to the weld and is contained in the weld heat-affected zone
  - Promoted by weld-sensitization, weld residual stresses and cold work
- In contrast - Off-Axis cracking typically propagates perpendicular to the associated weld
  - Cracking extends outside of the weld HAZ
  - In most cases, cracking terminates at the weld toe. However, in a few cases, the cracking appears to propagate into or even across a weld
  - Often, IGSCC branching with a significant off-axis component is also reported as off-axis cracking, especially if the cracking propagates outside the weld HAZ
  - Essentially all off-axis cracks reported to date are located along H4, H5, or along the vertical welds spanning H4 and H5



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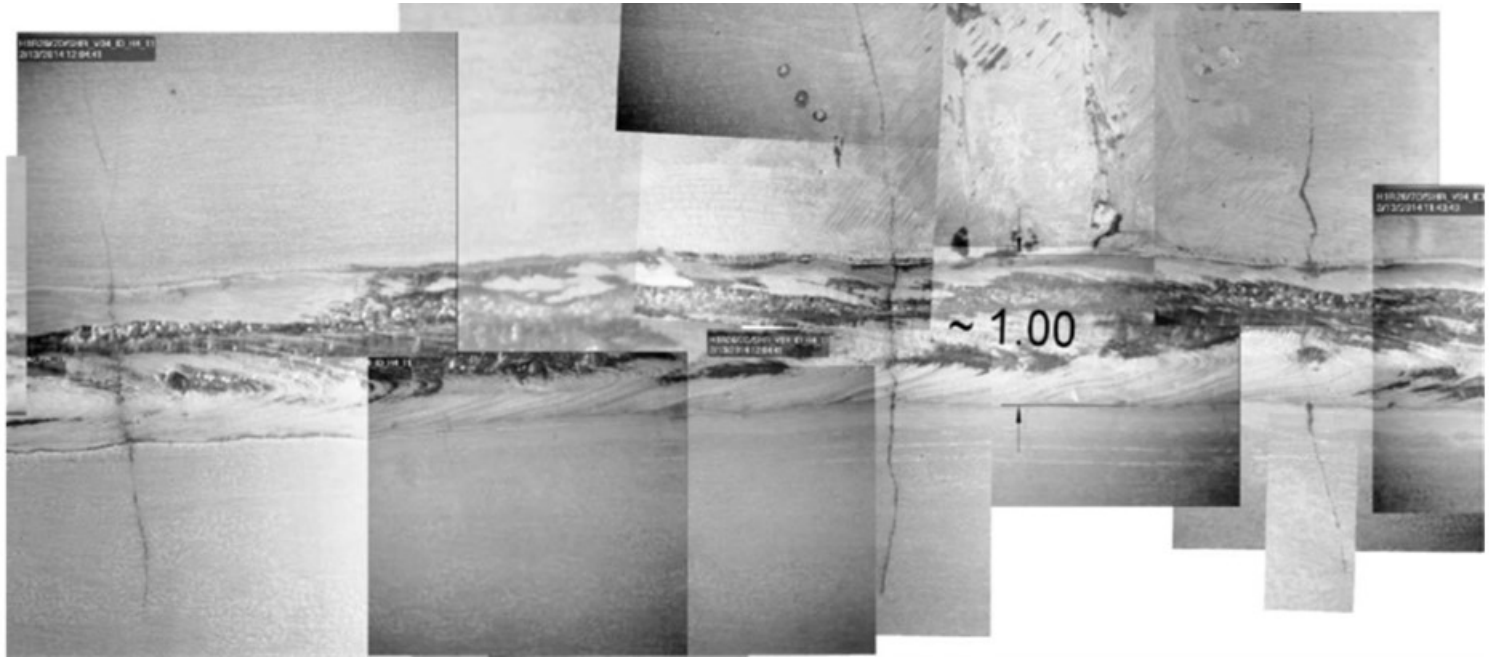
Off-axis Core Shroud Cracking - Example



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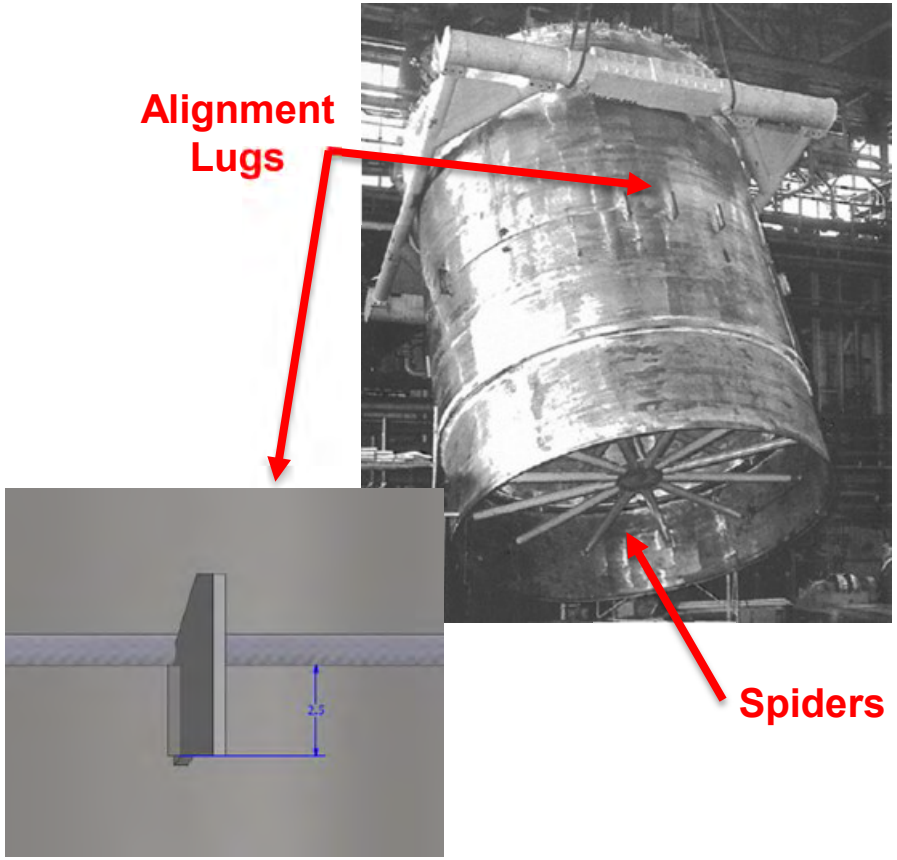
Fabrication Practices – Weld Sensitization/Cold Work

- Weld sensitization likely occurred due to the installation of alignment lugs and spiders
- Image of core shroud shows the use of welded alignment lugs and spiders
- Alignment lugs used to align plate edges prior to welding
- Spiders used to stabilize shroud rings during fabrication and transport



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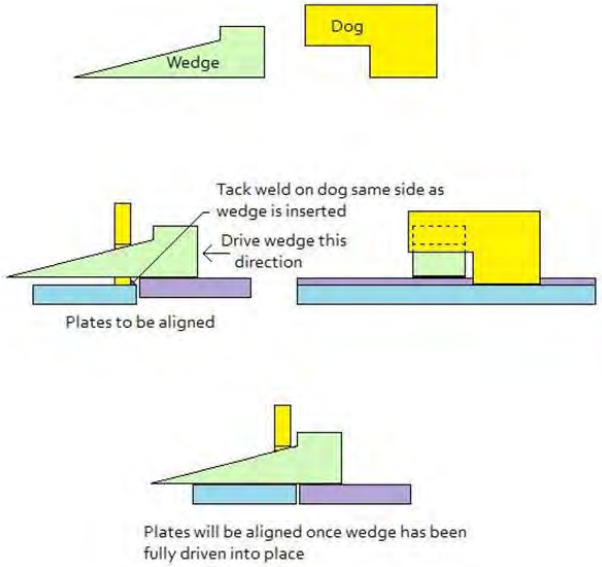
Fabrication Practices – Cold Work / Deformation

- Alignment often attained by local deformation of shroud plate edges
- Illustration at right shows the use of alignment lugs (dogs) and wedges to attain edge alignment
- Alignment lugs used to align edges for welding can result in high stresses and local deformation near weld zones and outside the HAZ
- Subsequent removal of alignment lugs and spiders by grinding introduces surface cold work



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Fabrication Practices

- Fabrication practices described above clearly could have contributed to some instances of observed off-axis cracking
  - Weld sensitization and cold work are known to strongly influence SCC occurrence in stainless steel BWR internals
  - Effected region includes a few inches on either side of welds, similar to the region containing the off-axis cracking
  - Although grinding marks may be seen in visual examinations, the magnitude of the residual stresses induced by the alignment process and evidence of alignment lug welds cannot be detected by examination

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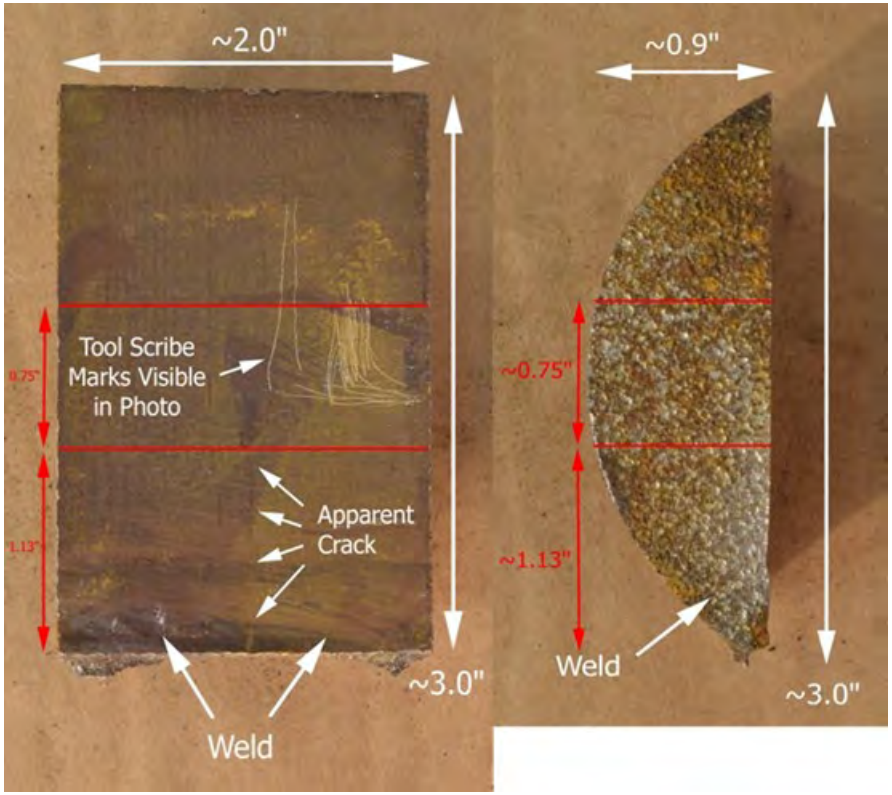
Boat Sample

- Sample taken at a location ~9.5” counter-clockwise from V4 and ~1.5” above H4
- Captured flaw - 0.82” max depth
- Weld, heat-affected zone, and base metal captured
- Utility removed boat sample
- BWRVIP is funding testing and evaluation of the boat sample
- Boat sample evaluation being performed by team of SIA, B&W, and PNNL



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Objectives

- Overall Objective: Determine whether the cracking is due to IASCC or to some other mechanism
- Specific Objectives
  - Determine the chemical composition of the base metal, weld metal and compare to specifications
  - Determine the microstructure and grain boundary chemistry in the regions surrounding the cracks and compare to non-cracked regions
  - Assess Pt deposition within crack
  - Determine the variation in hardness as a function of depth in order to assess the possible role of cold work
  - Measure the fracture toughness and tensile properties in the base metal
  - Estimate the neutron fluence for comparison to code predictions

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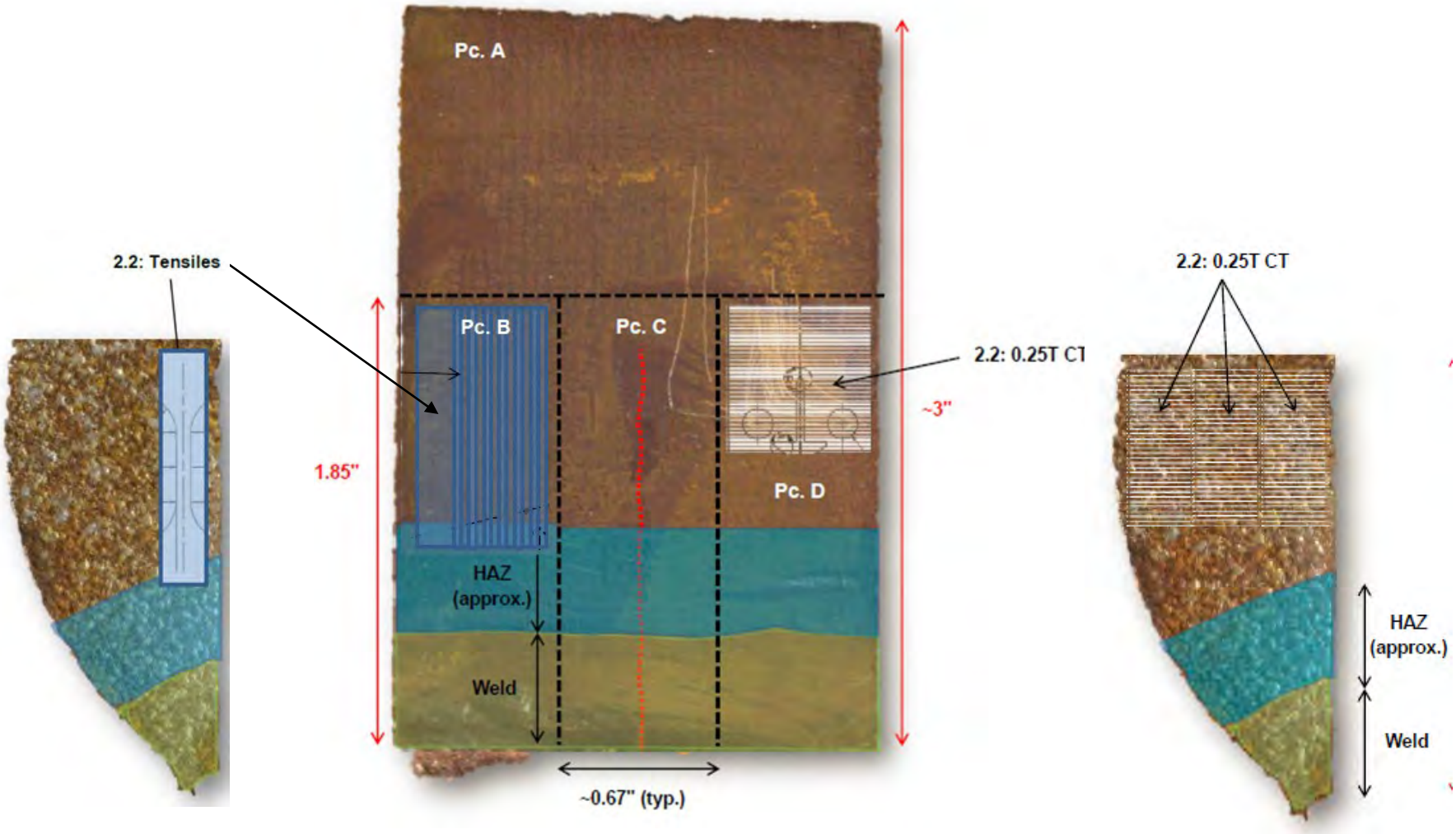
Sectioning Plan – Pieces B and D – Tensile and Fracture Toughness Specimens



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Status

- Testing underway or completed
  - Visual Inspections and Photography
  - Optical Metallography
  - Automated Microhardness
  - SEM/EDS Examinations
  - Tensile Testing
  - Fracture Toughness Testing
  - Dosimetry Analysis
  - Chemical Analysis by ICP/MS
  - FEG-SEM and EBSD Characterization
  - ATEM Characterization of Base and Weld Materials
  - ATEM Characterization of Crack Tips
- Report planned for 2nd quarter 2016

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SCC and Fracture Toughness of Alloy X-750 and XM-19 in a BWR Environment

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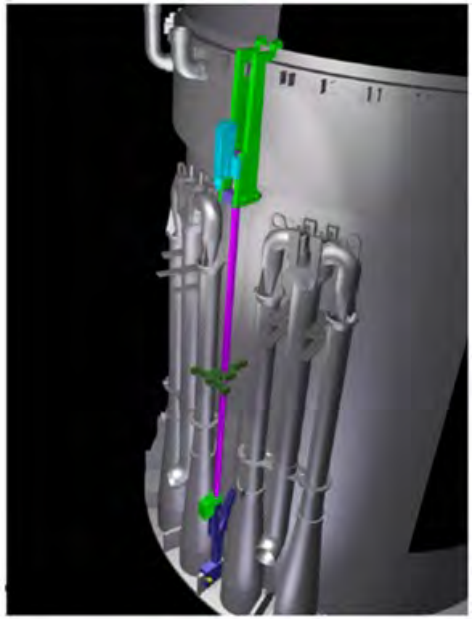
Background

- Alloys X-750 and XM-19 are used in many structural applications in BWRs ranging from original equipment to modifications and repair hardware
  - SCC and fracture toughness data in BWR water chemistry conditions are rather limited, particularly when exposed to neutron irradiation
- A multi-year program has been developed to examine the SCC and fracture toughness behavior of these materials under a variety of BWR conditions



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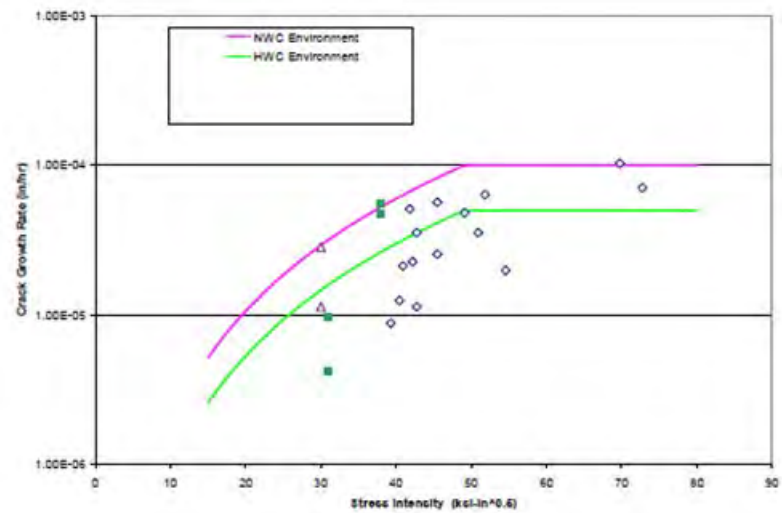
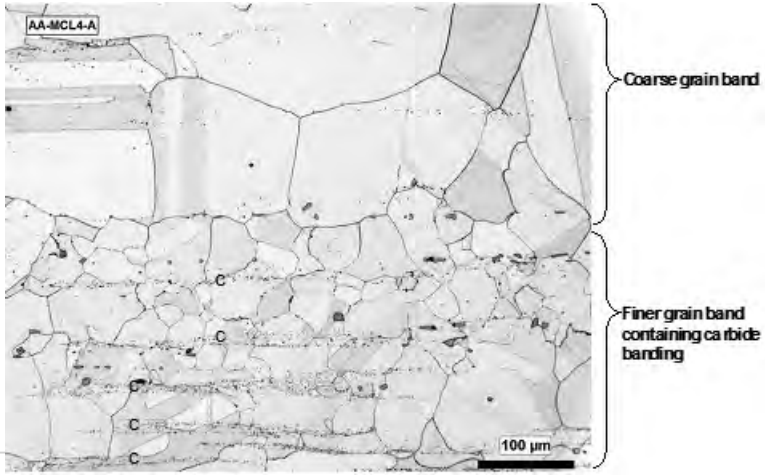
X-750 and XM-19 Data Needs

- SCC growth rates in NWC and HWC
- J-R fracture resistance in air and high temperature water
- Crack initiation
- Effects of neutron fluence
- Effects of microstructure variability





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Project Structure

- GE Global Research
  - Material characterization
    - Micro-macro mapping and SEM analysis
  - Unirradiated testing
    - Tensile, crack growth and J-R tests
    - Crack initiation in constant load Keno test
- Idaho National Laboratory – Advanced Test Reactor
  - Confirmatory unirradiated testing (for comparison to GE-GR)
  - Irradiation at three fluences
  - Post irradiation examination and testing
    - TEM examination, tensile, crack growth and J-R tests

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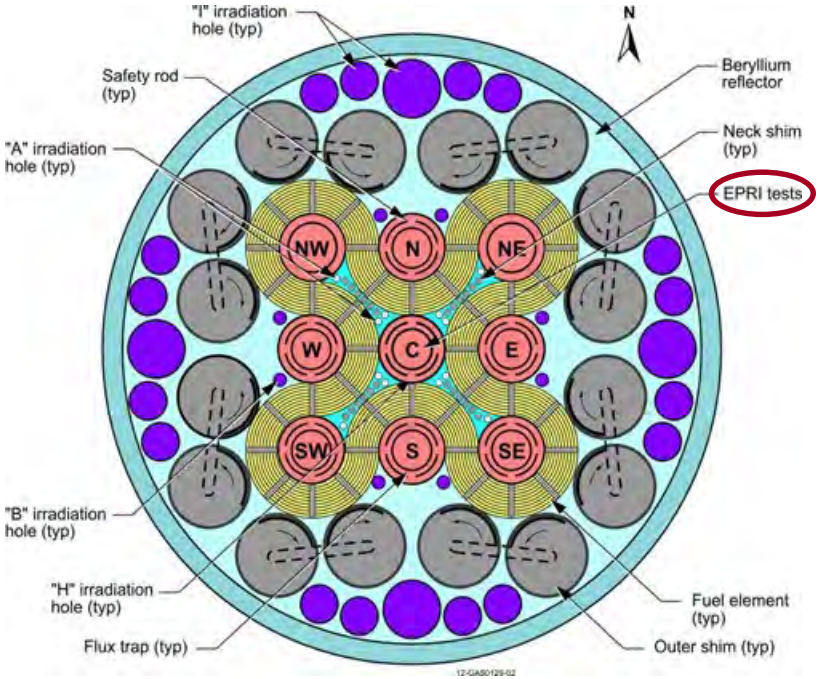
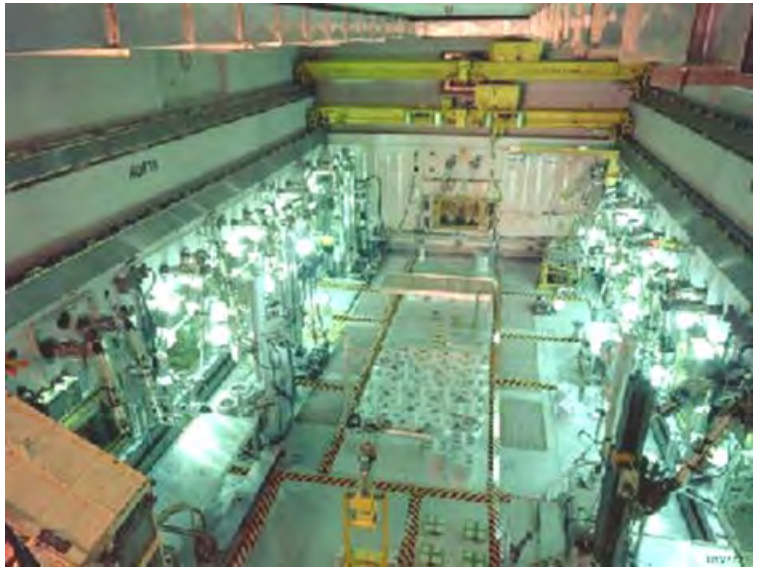
Irradiation of Materials

- EPRI and the Advanced Test Reactor National Scientific User Facility (ATR NSUF) formed an agreement to test representative alloys as an industry pilot program
  - Address BWR issues
  - Establish guidelines for future ATR NSUF research programs

- Experiment primarily aimed at determining the changes in fracture toughness and crack growth rates due to irradiation effects





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Preliminary Conclusions

- SCC CGR tests on Alloy X-750
  - < 2X difference among all microstructures or orientations
    - No evidence of accelerated CGR due to variations in microstructure
  - Strong effect of corrosion potential (HWC) on reductions in CGR
  - INL confirmatory testing compares well with GE-GR
    - *BWRVIP-262NP: BWR Vessel and Internals Project, Baseline Fracture Toughness and Crack Growth Rate Testing of Alloys X-750 and XM-19 (Idaho National Laboratory Phase I). EPRI, Palo Alto, CA: 2012. 1025135.*
- Fracture toughness tests on Alloy X-750
  - Scatter in J_{Ic} and J-Δa is small
  - All specimens exhibit significant ductility

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Primary Systems Corrosion Research

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List of 2015 Projects

Irradiated materials	Development of Radiation Resistant Material (ARRM)	Base
	CT size and orientation effects on IASCC (Zorita)	Base
	IASCC CGR Models Codification	Base
	Modeling of irradiated mechanical properties & Fracture Toughness	Base
	Small-volume mechanical property evaluation for irradiated materials	TI
	Effects of high fluence neutron irradiation on localized deformation and IASCC	TI
	Develop engineering solutions to counter IASCC	TI
	Rapid Simulation of High Fluence -- Ion radiation of LWR irradiated FTT	TI
	Round Robin – APT Data Acquisition and Analysis for Irradiated SS	MAI
SCC	Determine distinct SCC dependence (e.g. effects of microstructures)	Base
	Investigate GB oxidation of Alloy 600/600TT in PWR *(in discussion with PNNL)*	Base
EAF	Mechanistic understanding of loading effects on EAF CGR	Base
	Investigation the effects of irradiation on EAF CGR	Base
	Mechanistic model for environmental fatigue CGR	Base
MDM	Development of MDM-VVER & CANDU - IMT	Base
	Revision of Materials Handbook	Base
	Materials Information Portal (MIP) update	Base

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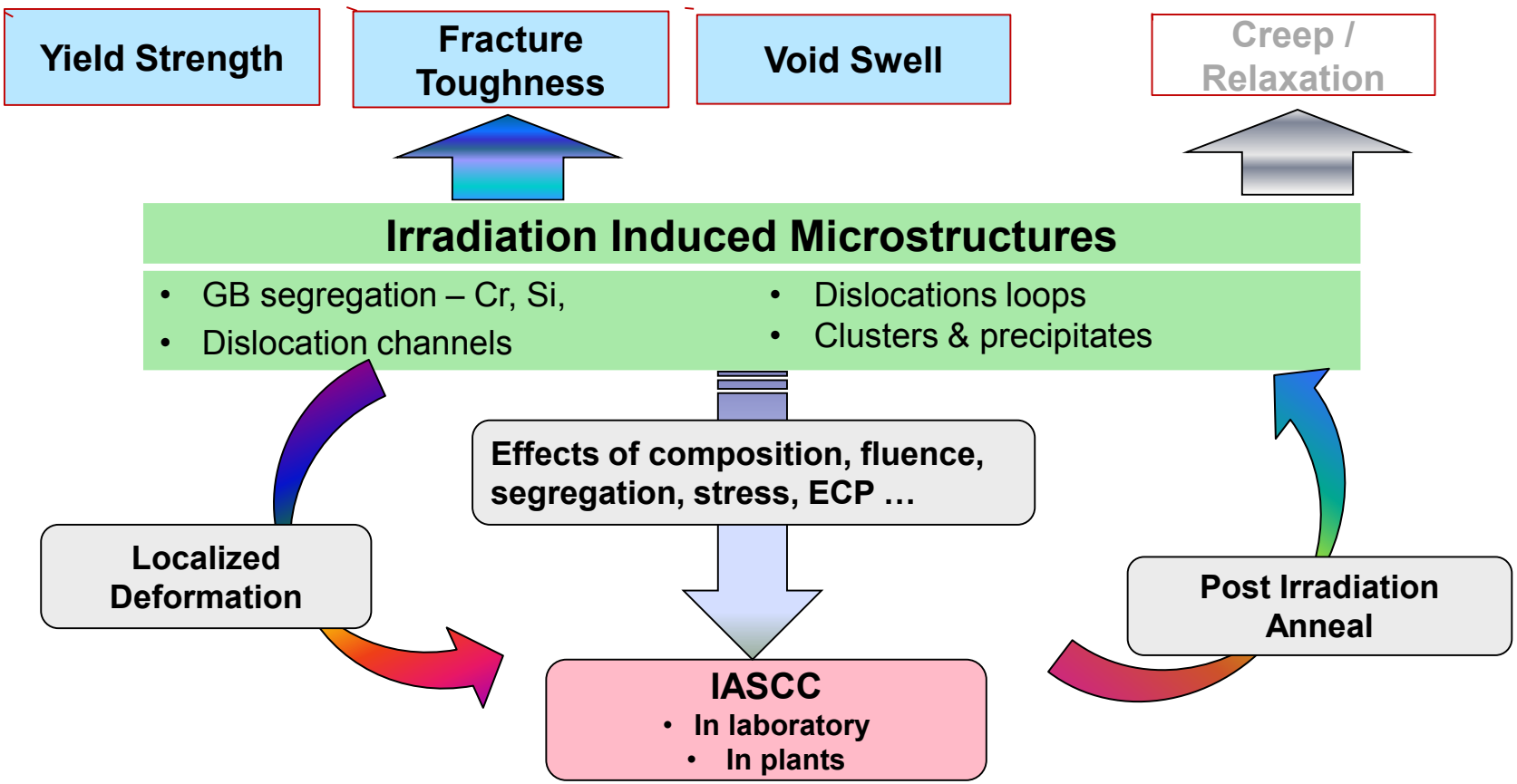
Continuing Focus on Irradiation Damage Mechanisms



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Rapid Simulation of Irradiation Damage in LWR Internals at High Fluence

- Objective: Develop and validate an approach based on heavy ion irradiation (Fe^{2+} or Ni^{2+}) with He/H implantation and fast neutron irradiation for cost effective and rapid simulation of irradiation damage in PWR internals at high fluence

  Background: PWR Internals are expected to experience high fluence exceeding 100 dpa at certain locations during first and second license renewal (60-80 years). There is a need for data and validated models to predict the degree of irradiation damage expected in austenitic stainless steel internals at high fluence.

- Benefit:
  - A validated approach could be used to evaluate the effect of high fluence more rapidly and more cost effectively than using extracted irradiated materials to be re-irradiation in LWRs
  - The project will provide a technical basis for fitness assessment of PWR reactor internal component materials with increased fluence, to ensure the safe and reliable operation of nuclear power plants for extended operation life.

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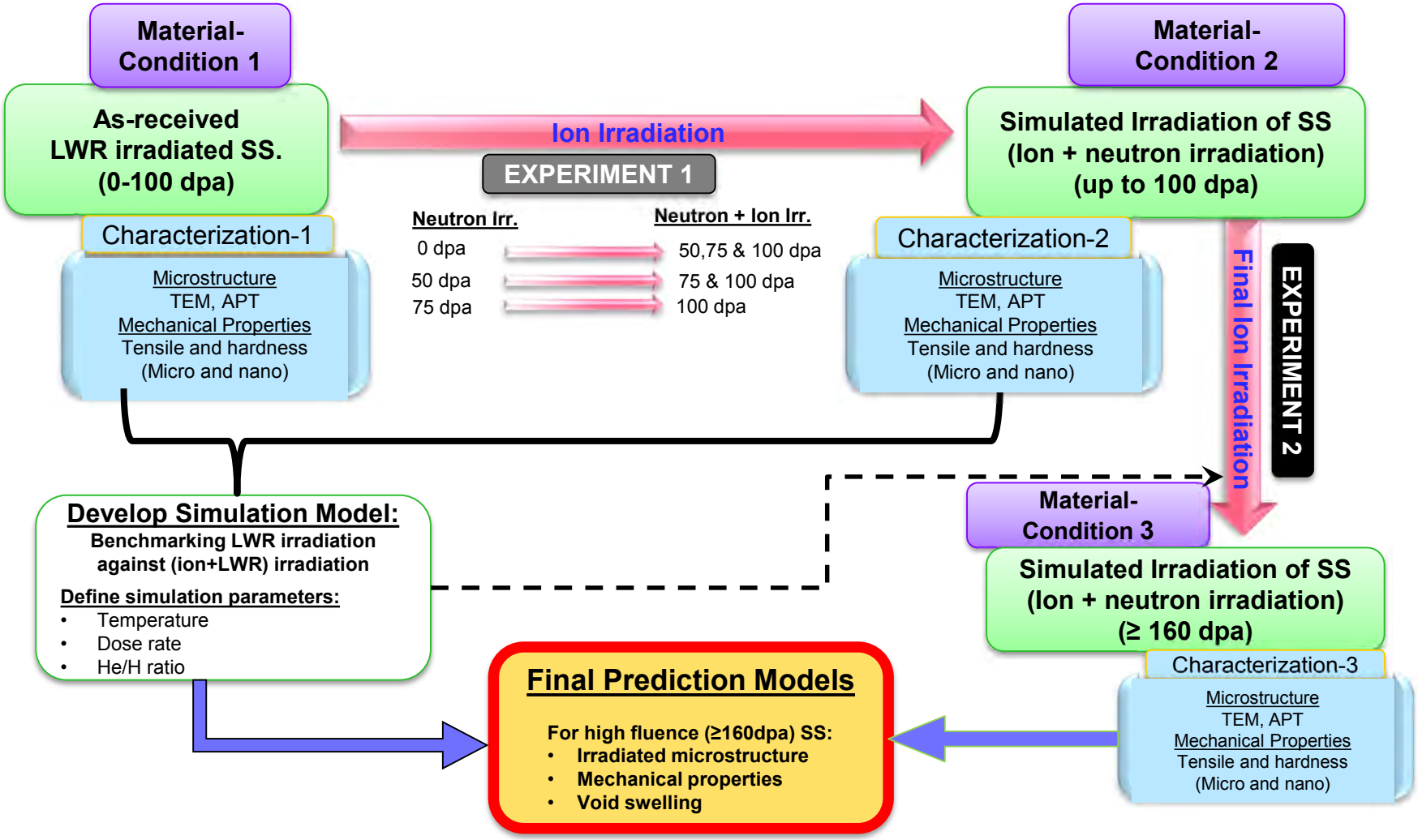

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2015 PSCR Deliverables -- Committed

3002005475	Mechanistic Understanding of Effects of Post Irradiation Annealing on IASCC
3002005476	Crack-tip Strain Rate Model for Environmental Assisted Fatigue
3002005474	Irradiation Assisted Stress Corrosion Cracking (IASCC) Initiation Model for Stainless Steels
3002005478	Summary Report on Recent Oxidation Studies – Literature Review
3002005623	Guidelines for Material Procurement for Use in LWR Environments
3002005470	Materials Handbook for Nuclear Plant Pressure Boundary Applications
3002005473	Integrated Materials Information DVD

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Objectives of ARRM Program

- ARRM is a collaborative research effort to develop the next generation of materials for in-core structural components and fasteners in LWRs
- Sponsored by EPRI, DOE, Bechtel Marine Propulsion
- The two primary research goals of ARRM are:
  - By 2022, to develop and test a degradation-resistant alloy that is within current commercial alloy specifications
  - By 2024, to develop and test a new advanced alloy with superior degradation resistance

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Main Concerns for Long-Term Operation

- Structural Large Section Components (Low Strength)
  - Dimensional stability (void swelling)
  - SCC & IASCC
  - Corrosion fatigue
  - Fracture Toughness
- Fasteners & Springs (High Strength)
  - Radiation creep
  - SCC & IASCC
  - Corrosion fatigue

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Proton and Ion Irradiation of ARRM Material

- The University of Michigan will conduct a program to characterize the response of a set of advanced materials to irradiation and irradiation assisted stress corrosion cracking.
  - Sample fabrication
  - Proton irradiation
  - Self-ion irradiation
  - IASCC (CERT) testing
  - Microstructure analysis
- Promising materials from this study will be selected for further evaluation under neutron irradiations
- Microstructural Analysis
  - Irradiated alloys will be examined for changes in microstructure (dislocation microstructure, precipitation, voids) and radiation induced segregation using TEM, FEG-STEM and APT
  - The same alloys analyzed in the irradiated condition will be examined in the non-irradiated condition to establish a baseline

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Microstructure analysis

- Irradiated alloys will be examined for changes in microstructure (dislocation microstructure, precipitation, voids) and radiation induced segregation using TEM, FEG-STEM and APT
- The same alloys analyzed in the irradiated condition will be examined in the non-irradiated condition to establish a baseline against which comparison of the irradiated condition can be made

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Welding on Irradiated Material

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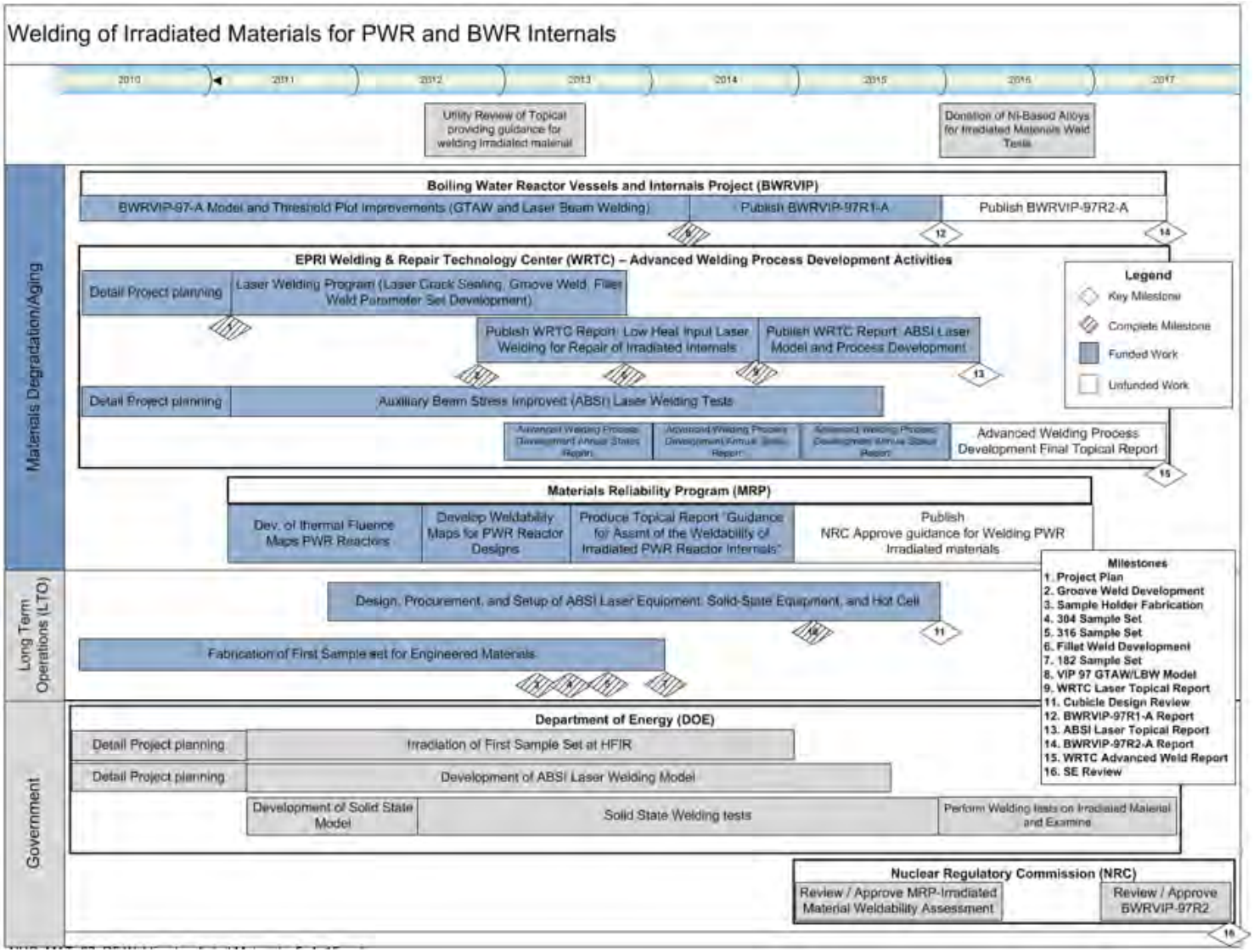
Irradiated Material Weld Development

Coordinated Roadmap



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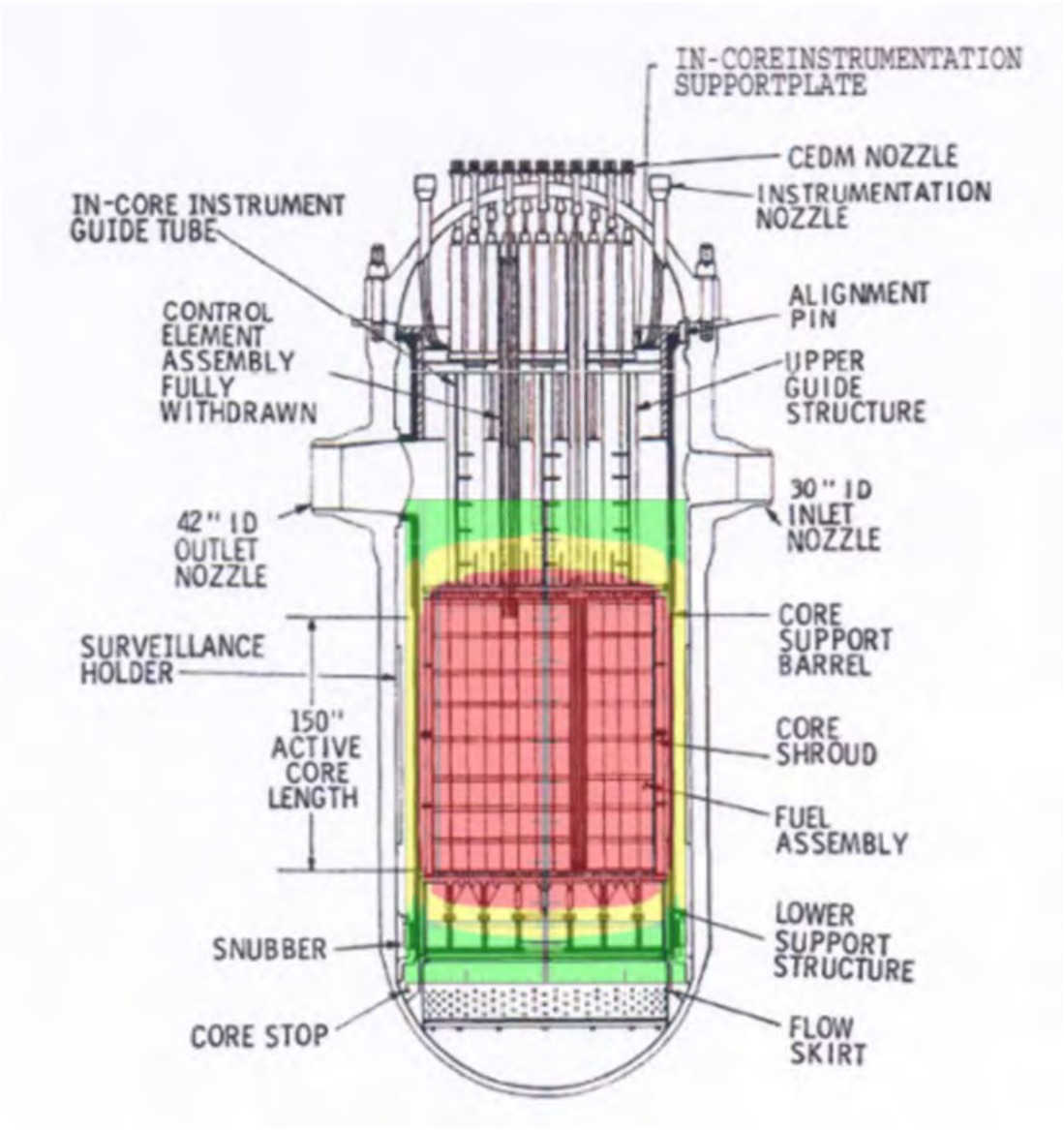
Welding Thresholds Mapped
(example is BWR at 60 EFPY)

Welding Thresholds

(Zone He generation in appm)

- Red >10: Not weldable with current welding processes
- Yellow 0.1 – 10: Weldable with heat input control
- Green <0.1: No special process control is needed



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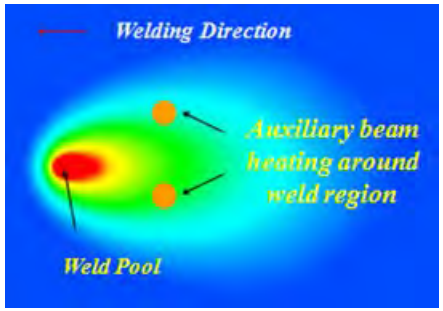
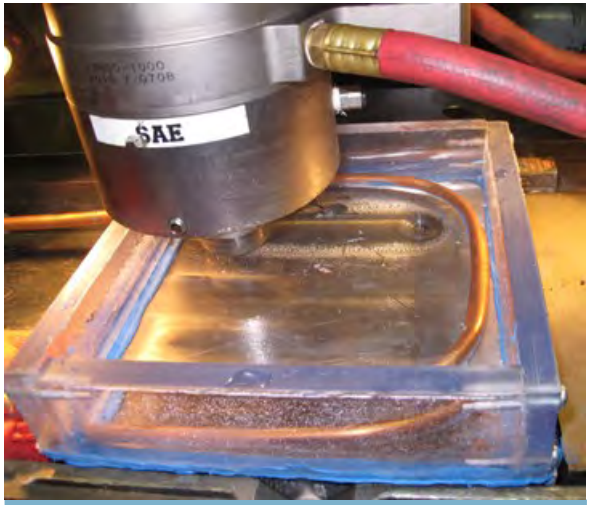
Advanced Welding Process Development

- Goal of this work is to develop a generic welding specification and validation testing of advanced weld repair processes
  - To address repairs on reactor internal components
- Processes under investigation
  - Low force friction stir welding
  - Low force friction stir cladding
  - Auxiliary beam laser welding
  - Low dilution laser beam welding
- Weld process testing and modeling conducted to prepare for hot cell validation tests





FSW process in-air and underwater

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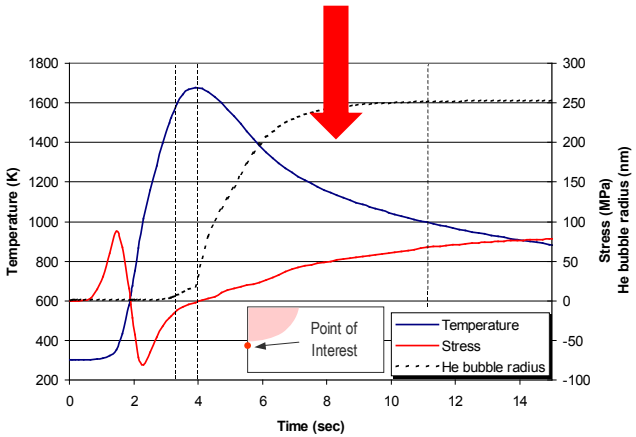
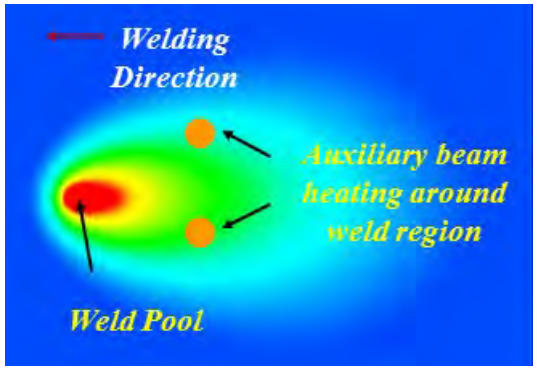
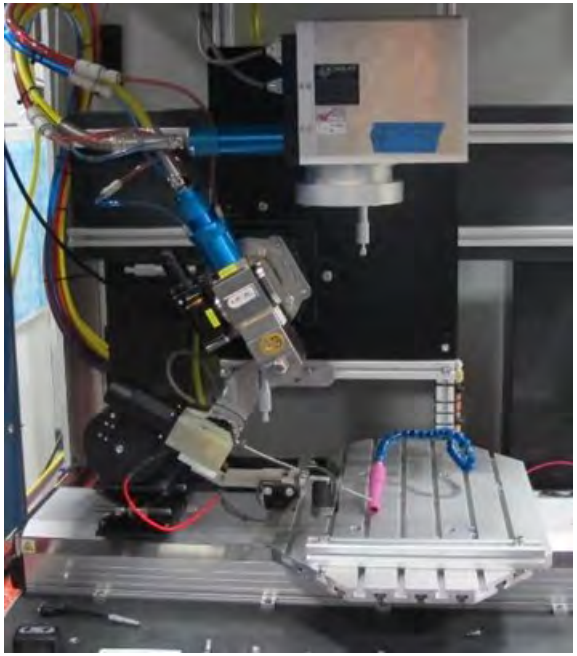
Auxiliary Beam Stress Improvement Laser Welding Process

- Concept:
  - Additional auxiliary heating sources are applied around the weld region to create a **compressive stress field** behind the weld pool (i.e., during the weld cooling cycle).

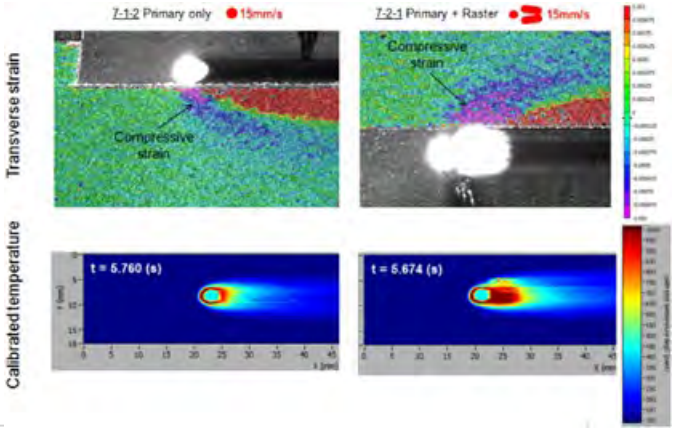






Auxiliary beam laser welding system

- Approach:
  - Advanced computational modeling is used to design and optimize the beam manipulation: beam intensity, spot size, location relative to the weld pool, etc. to effectively change the local stress/strain field around the weld pool



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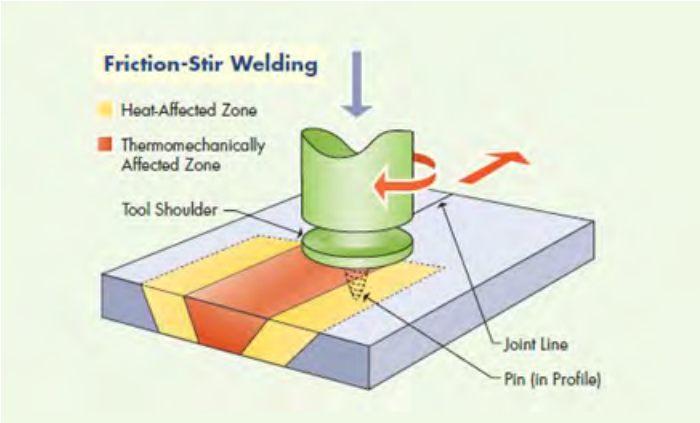
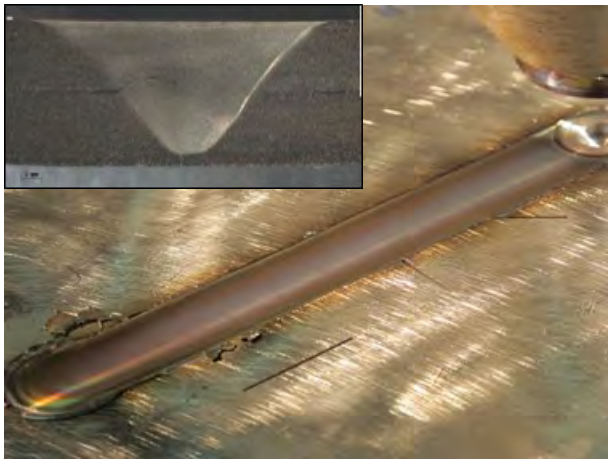
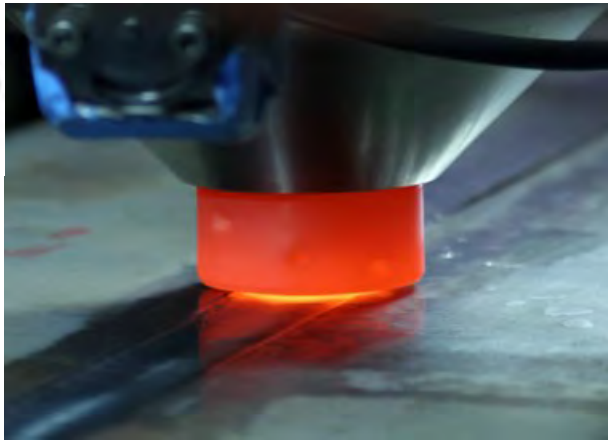
Solid State Friction Stir Welding (FSW)



- No arc, weld performed entirely in solid state
- Temperature controlled below melting temperature
- Homogenous joining occurs by the “stirring” of the plasticized material





EPRI/ORNL are investigating the FSW process for healing shallow cracks in irradiated materials

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Other Projects Supporting LTO

- Environmentally Assisted Fatigue
- Alloy 690 PWSCC
  - CGR testing and modeling
  - Initiation testing
- xLPR
  - Significant degradation modeling
- Peening
  - Upper head penetrations
  - Bottom head penetrations
  - RV Nozzles
- BWRVIP Chemistry Mitigation programs

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Reactor Pressure Vessel and Reactor Coolant System Research
Joint EPRI-LTO DOE-LWRS Presentation to ACRS
November 17, 2015

Robin Dyle

EPRI Senior Technical Executive

Keith Leonard

LWRS Program Materials Aging and Degradation Pathway Lead

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DOE / LWRS Areas of Research

- High fluence effects on RPV steels
- Aging of cast austenitic stainless steel
- Environmentally assisted fatigue
- SCC initiation in Ni-base alloys
- Mechanisms of IASSC
- Modeling Tasks:
  - High fluence irradiation-induced phase transformations
  - Radiation induced segregation
  - High fluence swelling
- Post-service examination of materials
  - Zion RPV
  - R.E. Ginna baffle former bolts
- Advanced replacement alloys
- Weld repair techniques

Measurement

Mechanisms

Modeling

Monitoring

Mitigation

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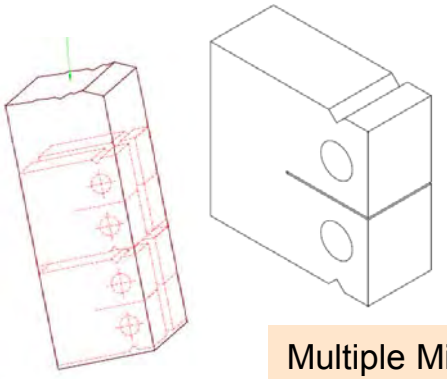
High Fluence Effects on RPV Steels

Tasks and Objectives

- High fluence properties of RPV steels.
- Develop multi-scale modeling of phase development and properties.
- Development of new test methodology: small fracture toughness specimens and use of small angle neutron (SANS) and X-ray scattering techniques.

Methods

- Detailed mechanical and metallurgical examination of specimens form commercial reactor surveillance programs and compare to similar materials irradiated in research reactors.

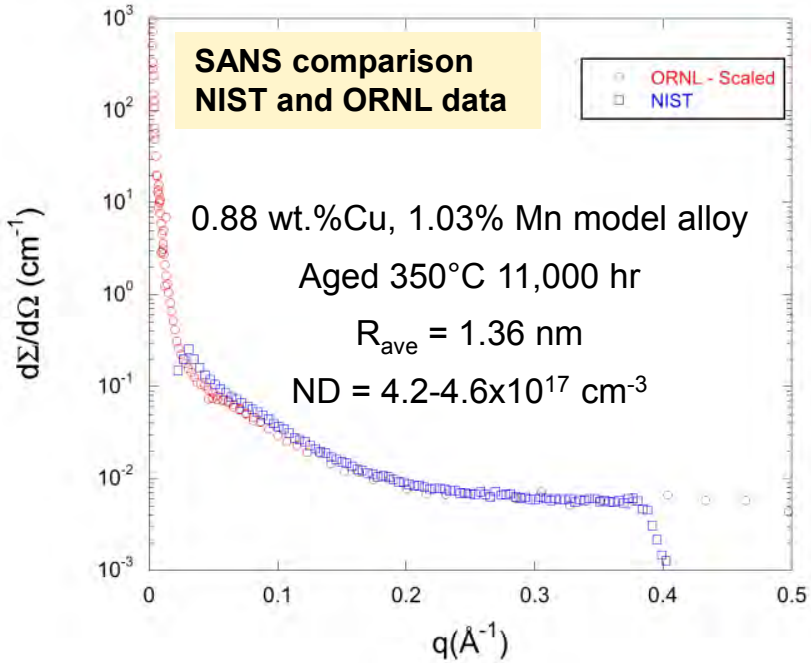


Mini-C(T) specimen

Charpy impact specimen

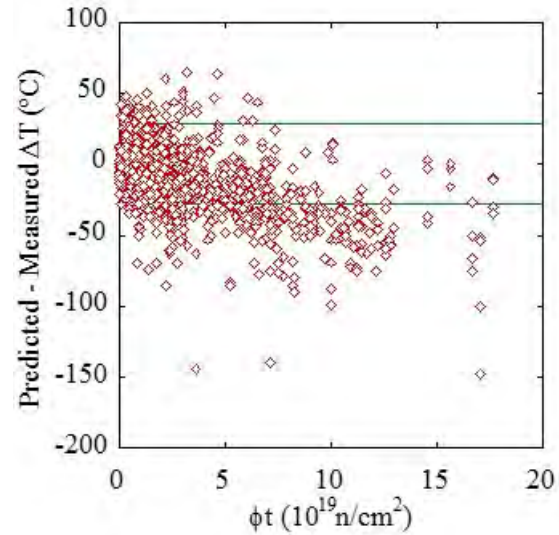
Multiple Mini-CT specimens machined from one half of a broken Charpy specimen

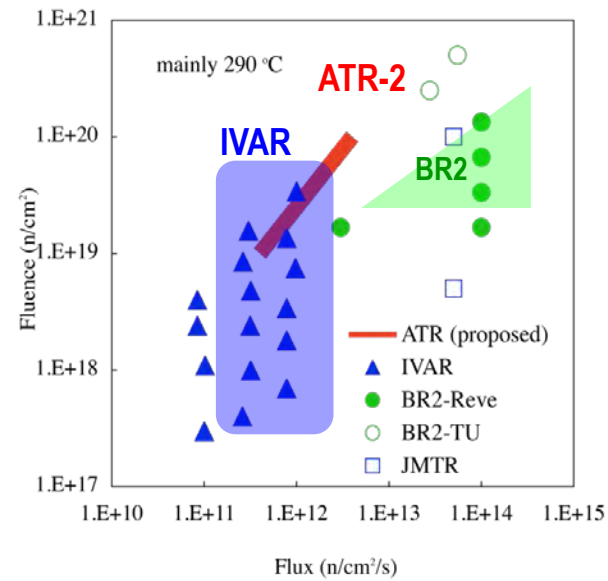
ORNL tests in very good agreement with 1T compact specimen data by Central Research Institute of Electric Power Industry (Japan).



ATR-2 Experiment

- Current RPV embrittlement models can under-predict transition temperature shift data.
- High fluence data is needed as reactor surveillance data is sparse above 3×10^{19} n/cm^2.
- Materials performance to $\phi t \approx 10^{20}$ n/cm^2 needed for extended lifetime evaluation.
- *Total of ≈ 180 RPV steel alloys (~ 1600 specimens) -* including IVAR program (CM, L-series), split-melt alloys, and nine commercial surveillance materials.
- Peak/Average fluences = $1.3/1.0 \times 10^{20}$ n/cm^2 .
- Four Temperature zones: 250, 270, 290 and 310°C.





Modeling Support in RPV Steels

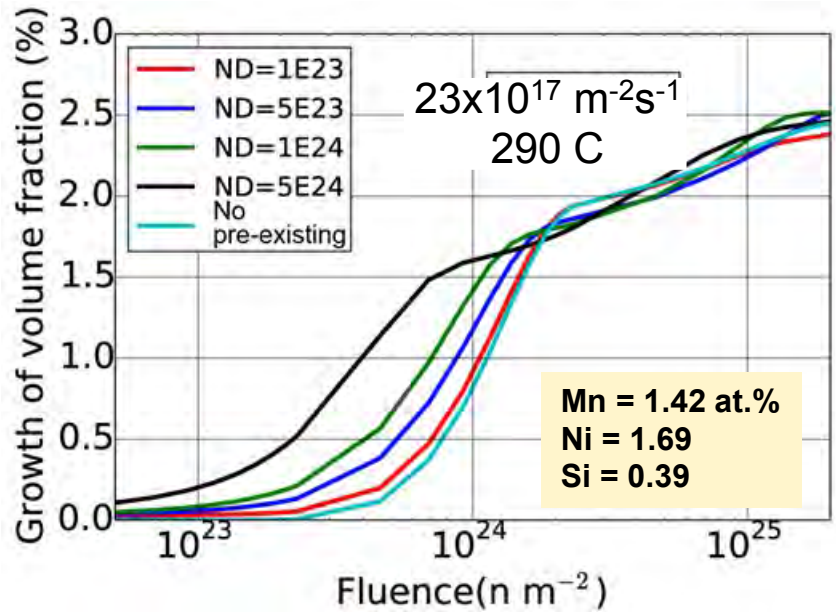
- Develop physically-informed model of Cu-Mn-Ni-Si precipitation in RPV steels to support engineering level models of embrittlement.

✓ Phase 1: Establish parameters for the evolution of Cu-rich precipitates.

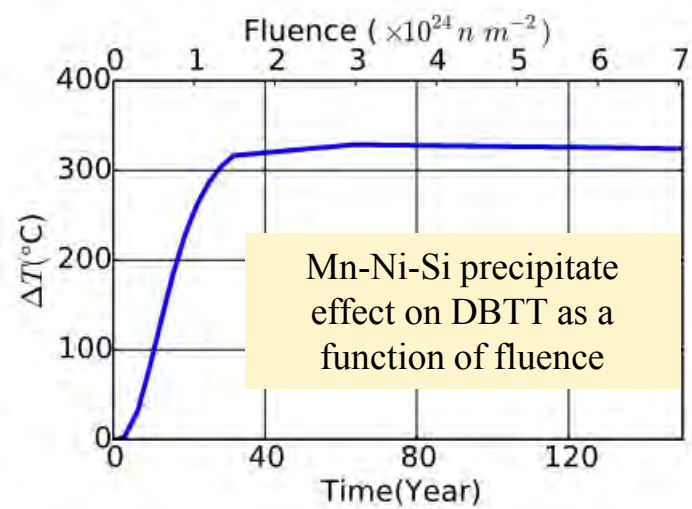
✓ Phase 2: Mn-Ni-Si (MNS) precipitation as a function of alloying content and fixed Cu-rich precipitate concentration.

- Development of cluster dynamics model to calculate evolution of MNS precipitates
- Fitting and validation against both irradiation and thermal annealing experimental data.

➢ Next, phase 3: Integrated coupling the fast Cu and slower MNS precipitate formation.

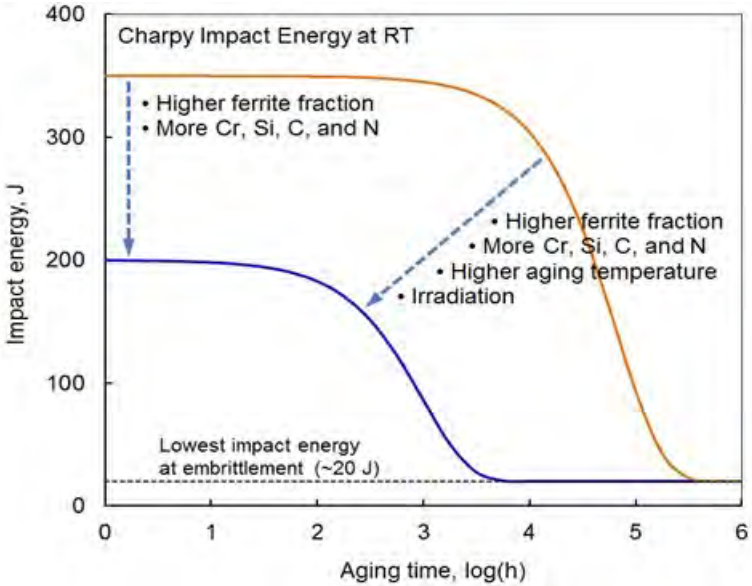


Growth of MNS precipitates for different pre-existing nuclei densities (ND)

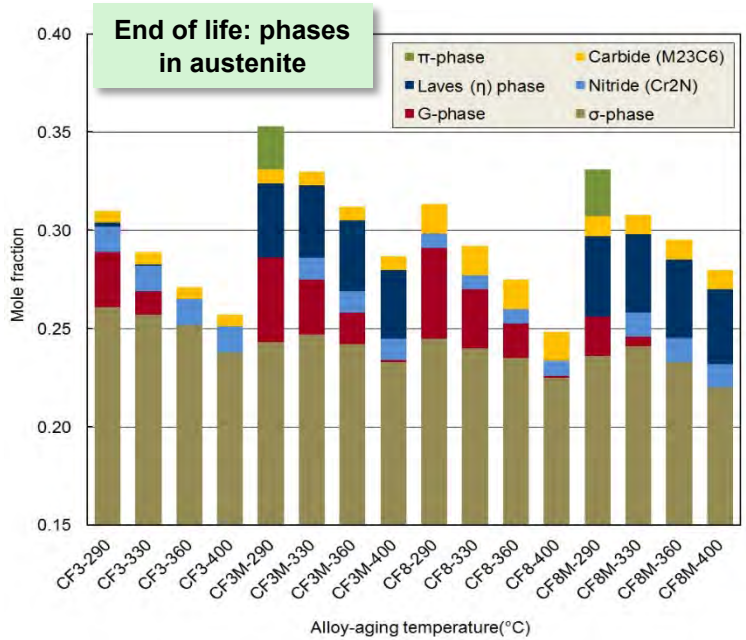


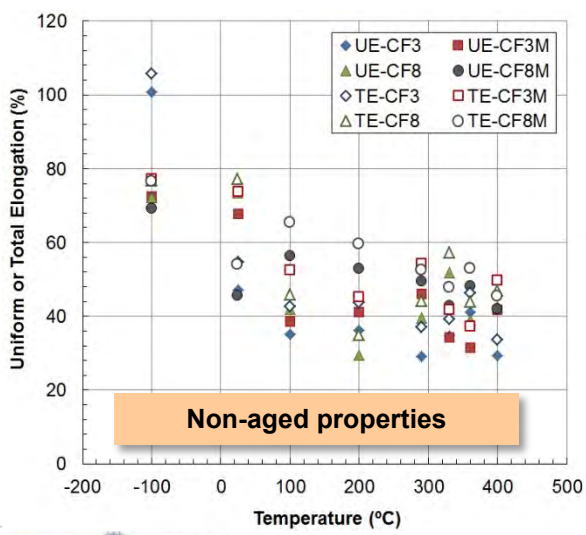
Cast Austenitic Stainless Steel (CASS)

- Identify any potential thermal aging issues of degradation beyond 60 yrs through accelerated aging.
- Address DOE/NRC EMDA and EPRI MDM concerns regarding lack of long term data.
- Aim: to systematically build a scientific knowledge base for the thermal aging behavior within a limited time of five years.
- Materials:
  - Model, static cast: CF3, CF3M, CF8 and CF8M
  - EPRI provided: centrifugal cast CF3 and CF8 vintage alloys, and CF8 elbow block.
  - Comparison to wrought 304L and 316L.



Structure of LWRS CASS Thermal Aging Research





- **Long-term aging treatment**
- **Yearly specimen retrieval**

✓ Existing data collection

✓ Computational studies (phase decomposition, precipitation, segregation)

✓ Non-aged baseline material properties

Aged Material

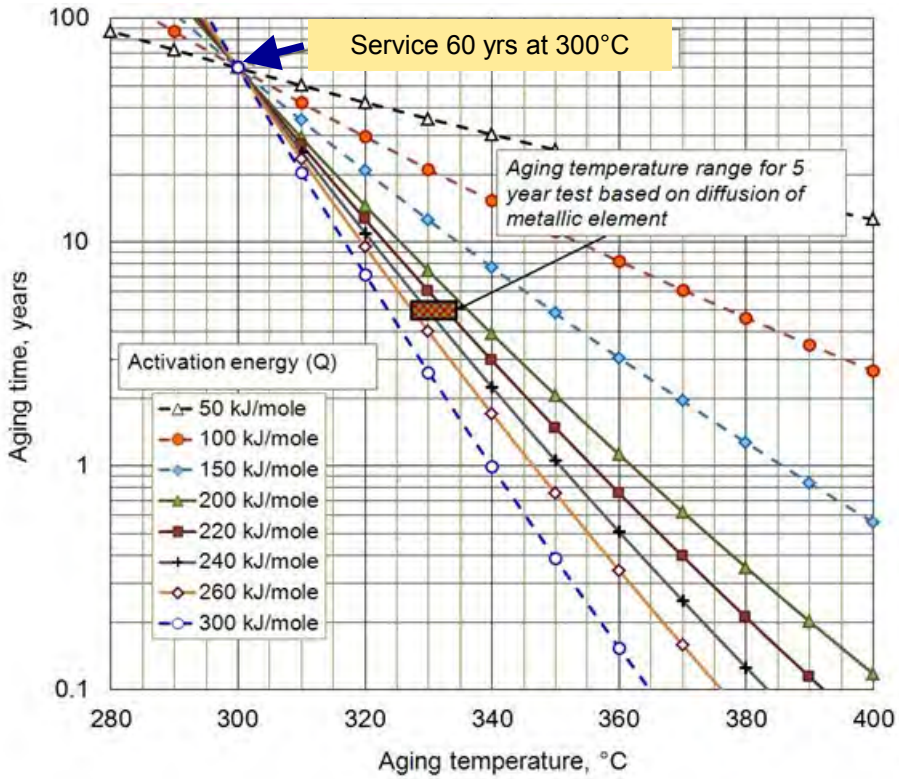
- Microstructure changes: precipitation, phase distribution, solute redistribution
- Mechanical testing: strength, ductility, impact toughness, hardness, etc.

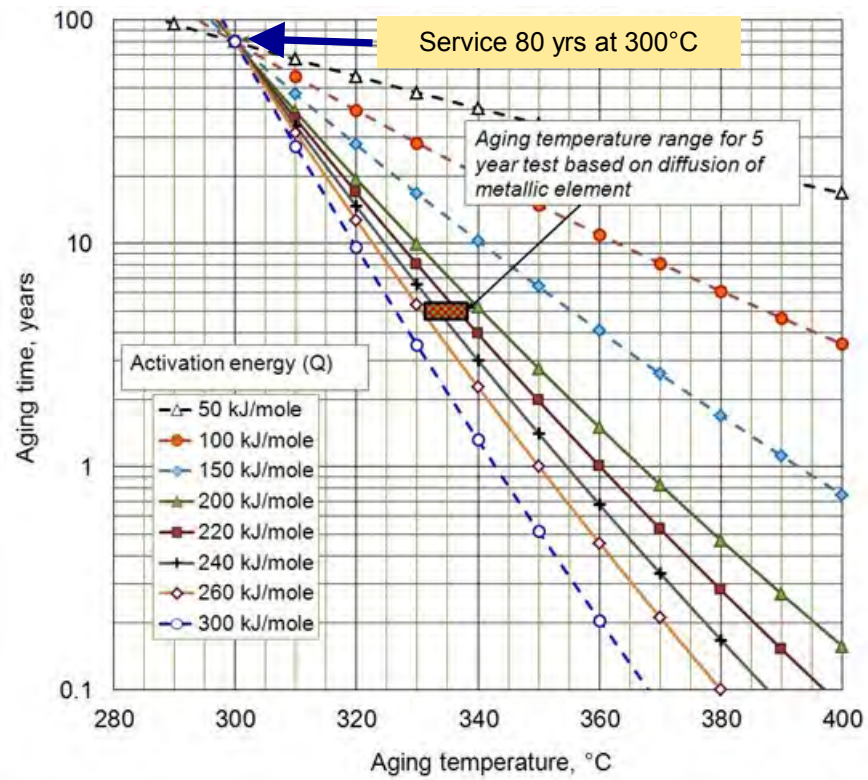
- Integration of data into property prediction models

➢ **Conclusive predictions for aging behavior of CASS during extended services**

➢ **Extended knowledge base (experimental & computational data, mechanisms)**

Determination of Aging Temperatures





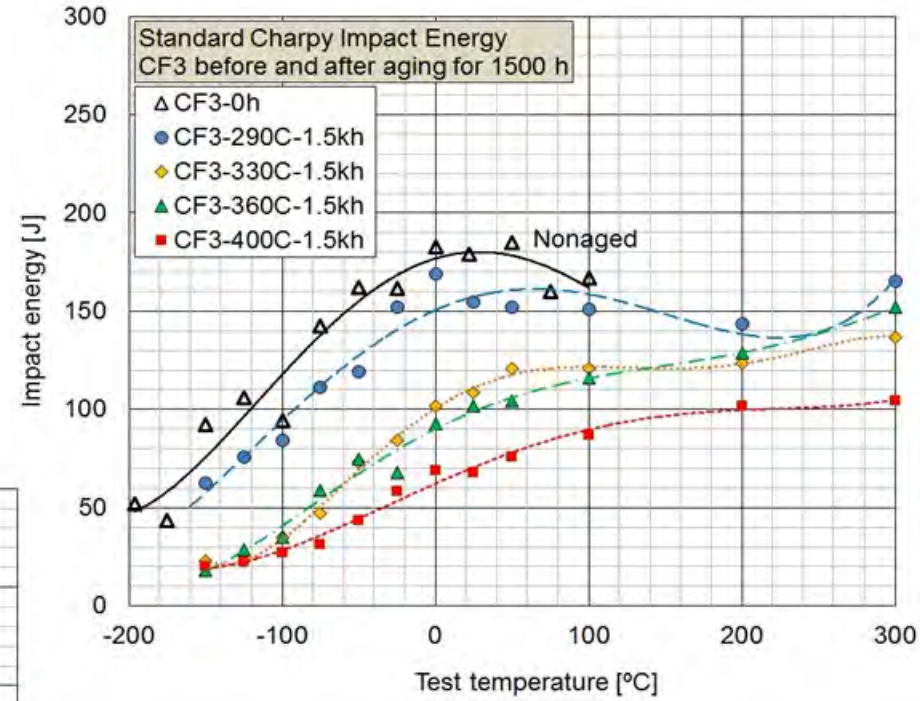
$$t(T) = t_o \times e^{Q/RT} / e^{Q/RT_0}$$

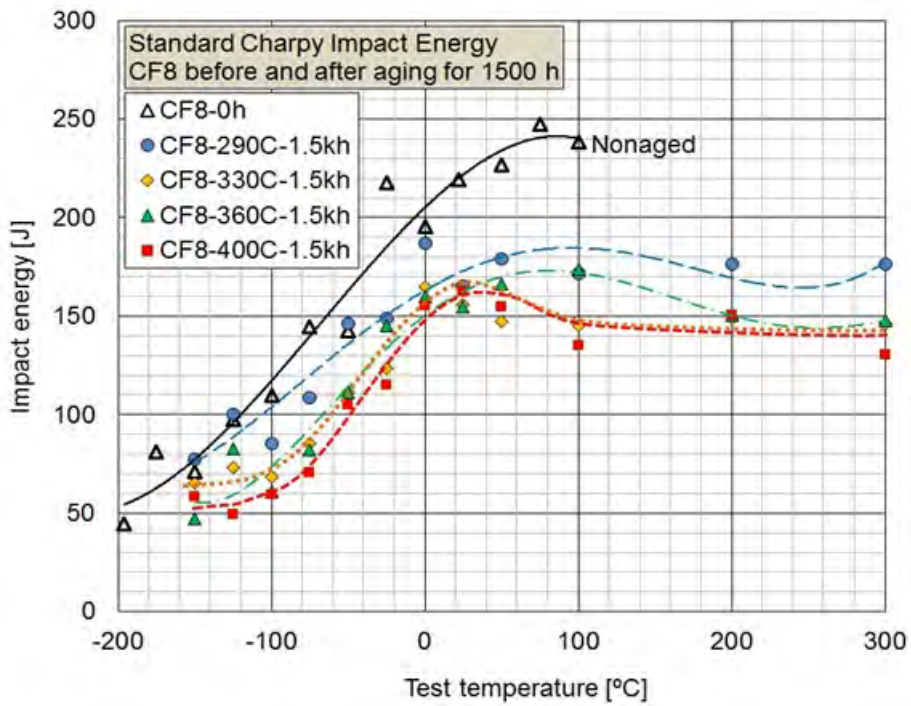
- Aging temperature selection needs to ensure that appropriate mechanisms are active in both aging and in-service conditions.

Preliminary 1500 h Aging Charpy Impact Data

Model CF3 alloy

- 15% ferrite - initial
- Upper shelf energy and transition temperature strongly influenced



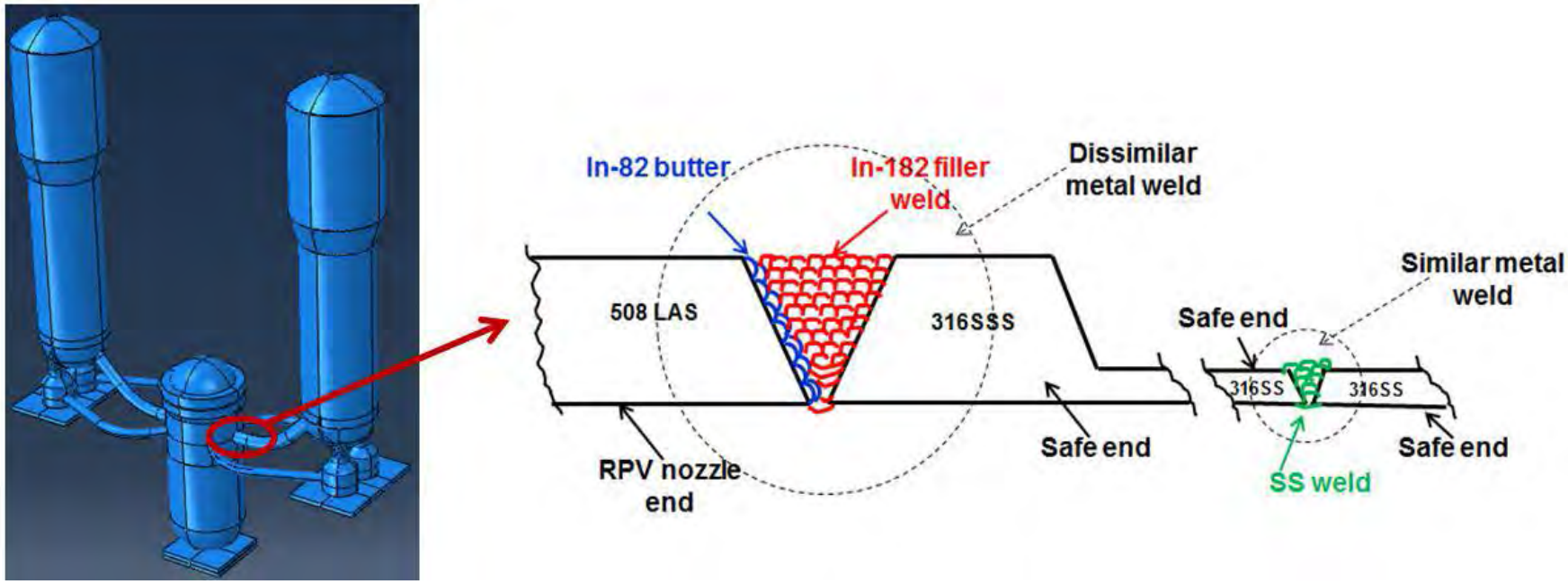


Model CF8 alloy

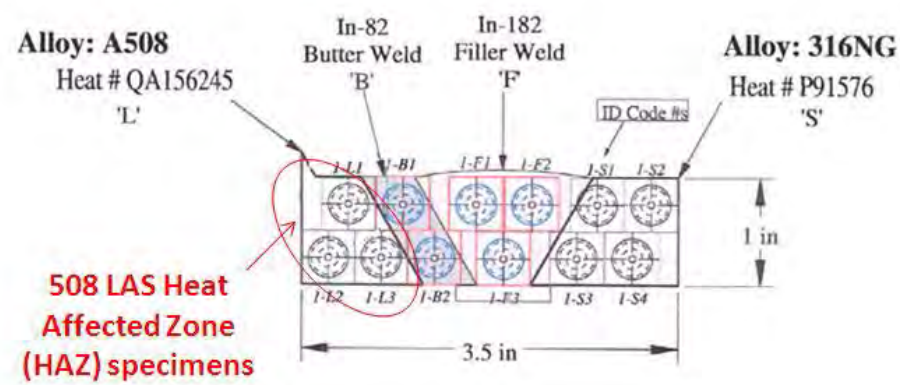
- 5.6% ferrite - initial
- Influence of aging

Aging effects on tensile properties are less significant for both alloys.

Modeling of Environmental Assisted Fatigue (EAF)

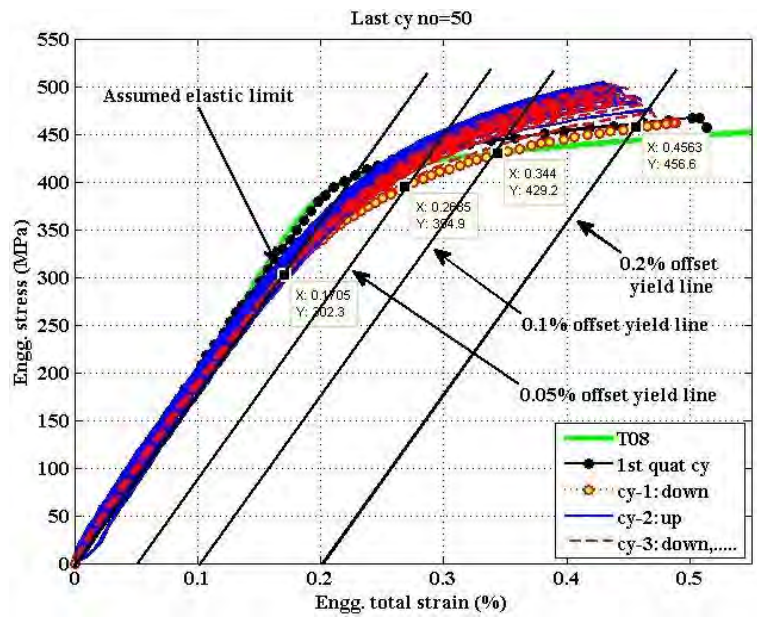


- Model EAF mechanisms through mechanistic based approach supported by experimental studies to identify key variables.
- Model will provide a capability to extrapolate severity of the mode of degradation under realistic reactor environment loading cycles and under multi-axial stress states to extended-life conditions.

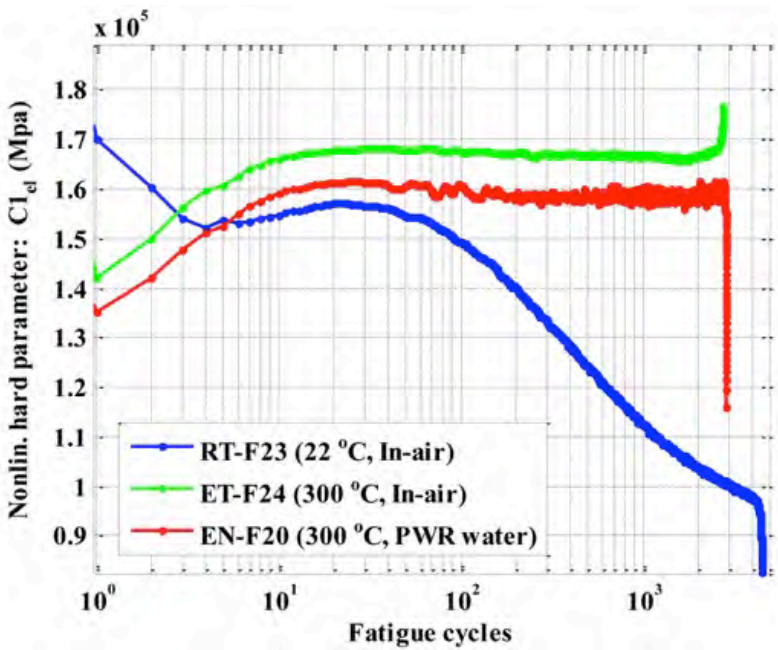


Evolutionary Material Model for Predicting Environmentally Assisted Fatigue

- Material hardens / softens as a function of accumulated plastic strain or time leading to yield surface expansion / contraction and translation in stress space.
- Monotonic stress-strain curve derives the cyclic evolution of the elastic limit.
- Parameters (C1 and γ1) have to be used to model the kinematic hardening stress.



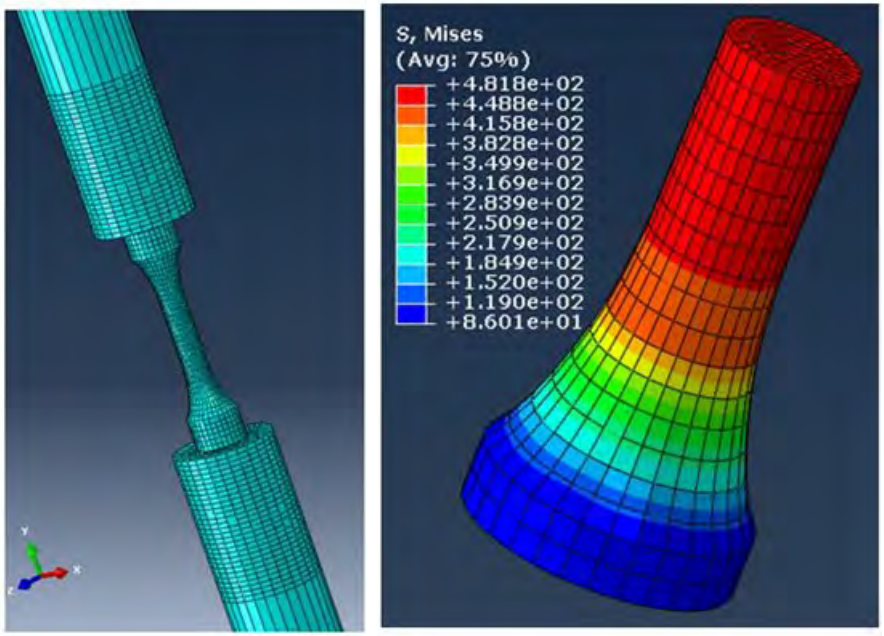
Equivalent monotonic stress-strain curve fatigue test at 300°C, PWR Water, compared 300°C in air tensile test data.



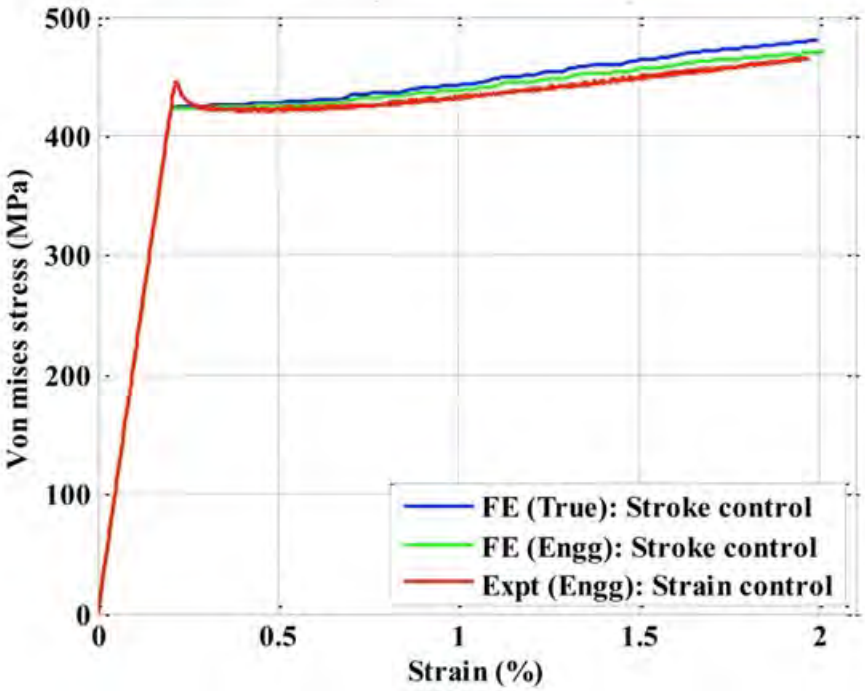
Nonlinear kinematic hardening parameter C1, for 508 LAS base metal specimens fatigue tested under different conditions.

508 LAS Base Metal Fatigue Test and Materials Model

- Finite Element mesh of tensile specimen and pull rods developed.
- Development and refinement of models through experimental data at various conditions.



Von Mises stress distribution at the end of simulation

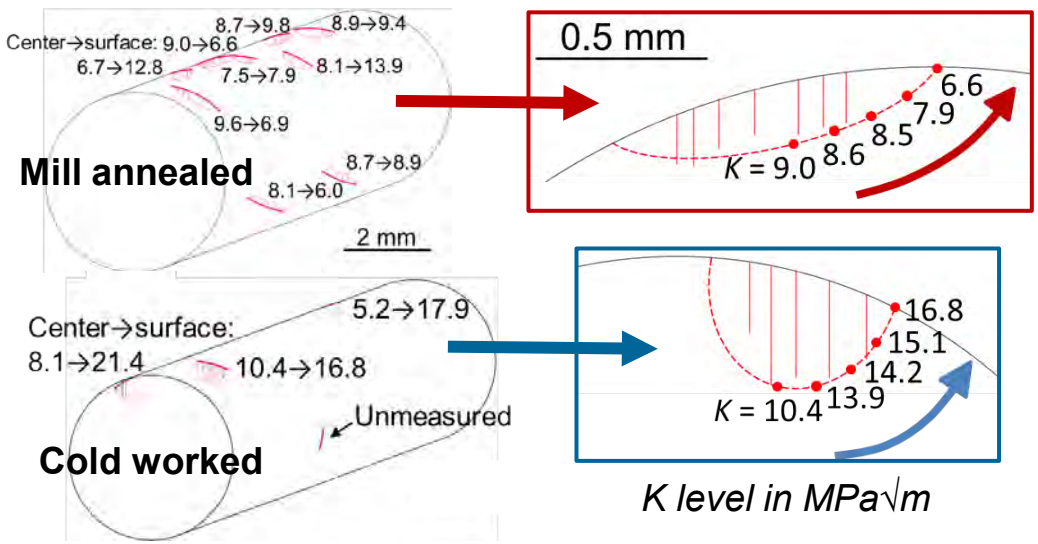


✓ Good correlation between model and experimental data.

➢ 2018 - Model for environmentally assisted fatigue in LWR components

Crack Initiation in Ni-Base Alloys

- Identify the mechanisms and precursor state to predict the causes for crack-initiation leading to more focused inspections, informed regulatory guidelines and aging-management programs.
- Alloy 600 testing:
  - Established grain boundary composition and inter granular attack (IGA) compositions and their susceptibility response for various heats, levels of cold work and surface conditions.
  - Crack morphology and critical stress intensity for stress corrosion crack initiation.



- *High stress corrosion crack susceptibility in cold worked materials.*

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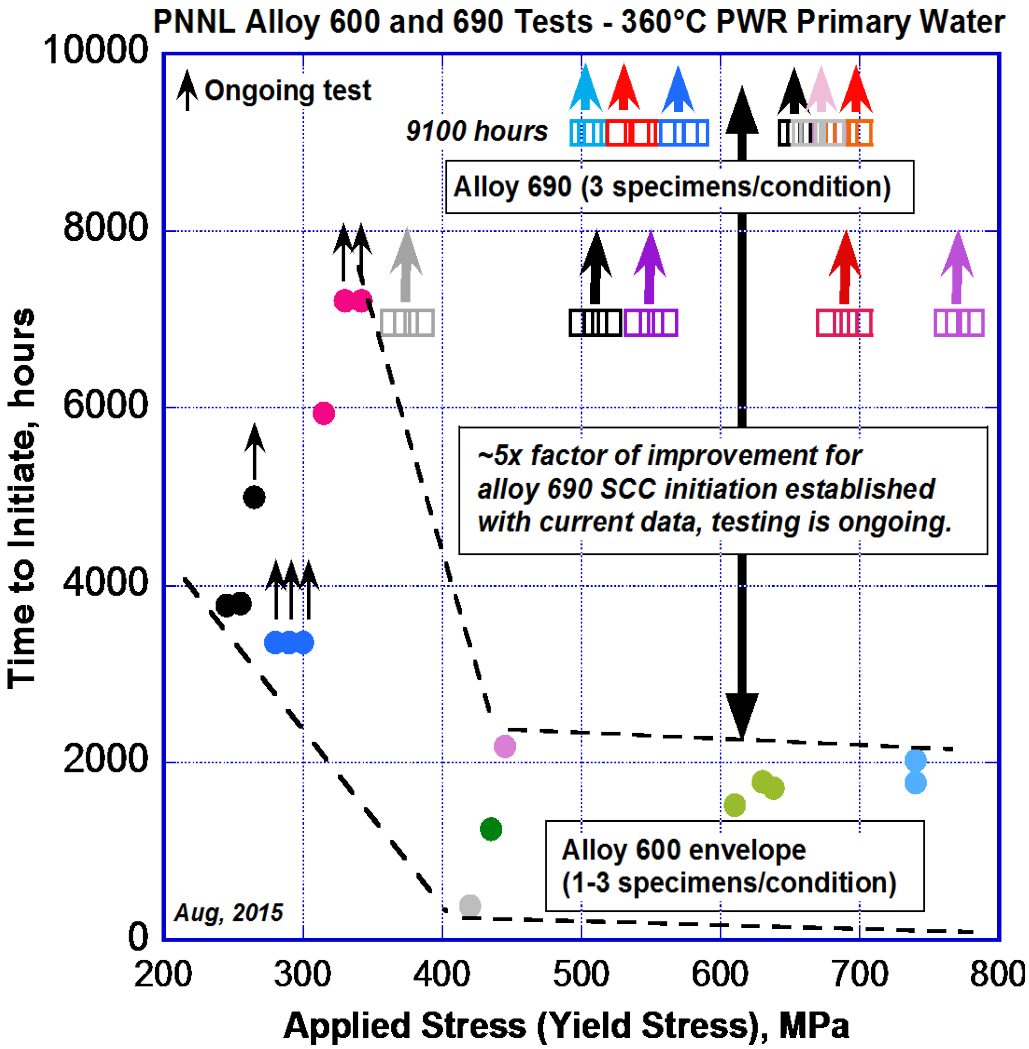
SCC Initiation Testing on Alloy 690 Materials

- Evaluating long-term response of cold-worked alloy 690 materials at constant load in 360°C PWR primary water at high applied stress levels.
- Multi-year test with yearly surface exams for possible precursor damage by SEM and FIB trenching.
- Direct comparisons being made to cold-worked alloy 600 heats.
- Have reached >1 year of exposure without DCPD detection of SCC initiation, indicates ~5x factor of improvement for SCC susceptible alloy 690 over alloy 600 so far.



M. B. Toloczko, Z. Zhai and S.M. Bruemmer, PNNL

Precursor Grain Boundary Damage Leads to IGSCC Nucleation in Alloy 690 Blunt Notch Specimens

- Alloy 690 is ***resistant*** to IGSCC initiation ***under constant*** load conditions due to formation of protective Cr_2O_3 layer at grain boundaries.
- New blunt notch SCC initiation tests on cold-worked alloy 690 reveal that internal grain boundary creep cavitation can form during ***dynamic straining*** and ***promote IGSCC crack nucleation*** in 360°C PWR primary water.

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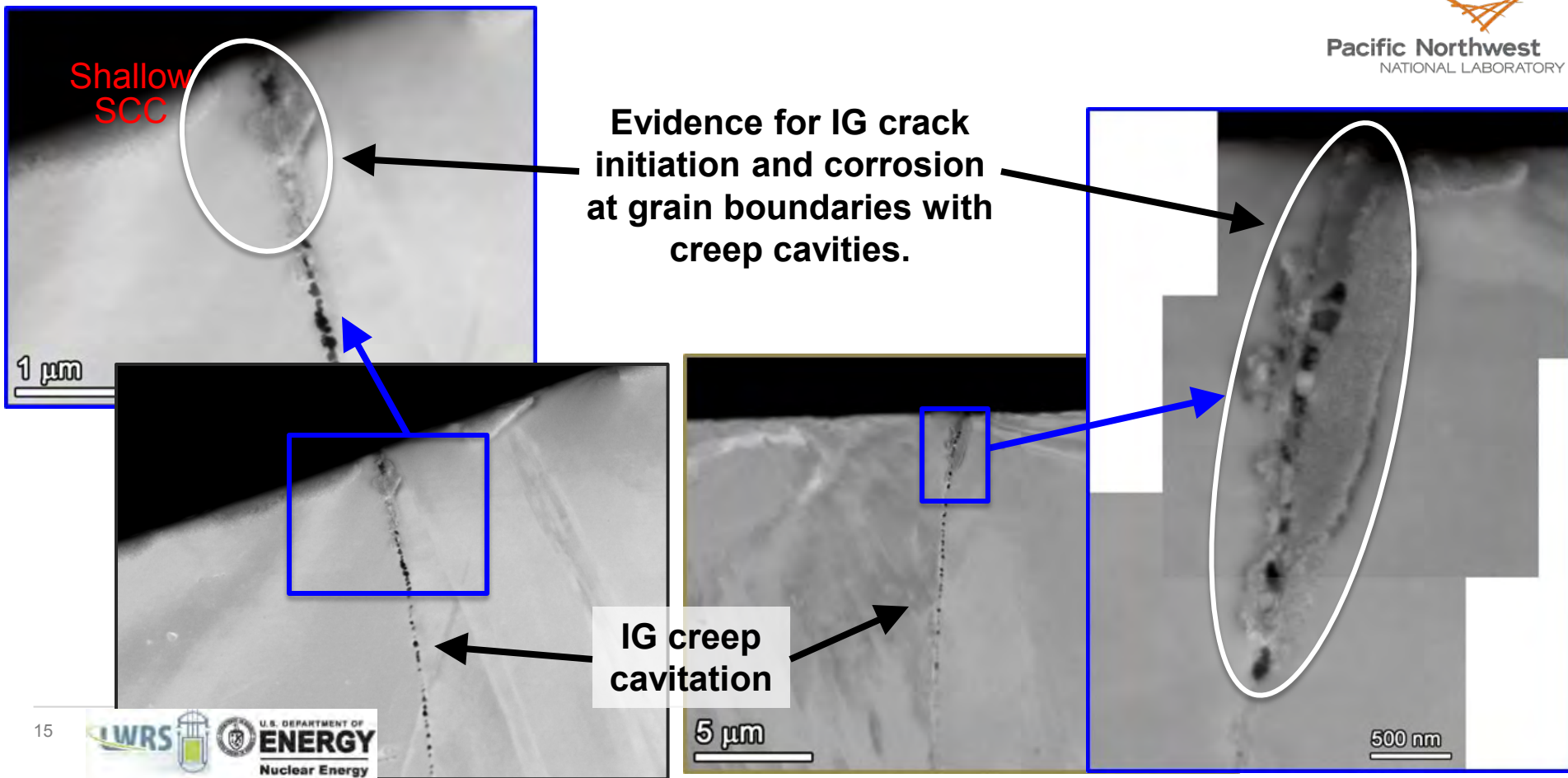

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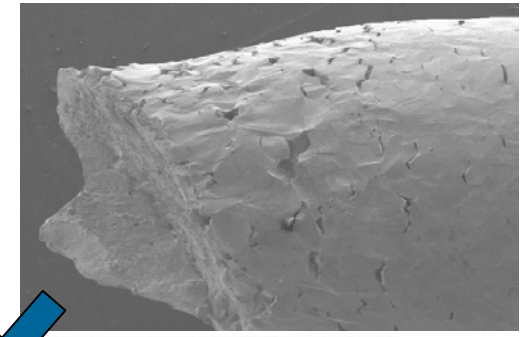
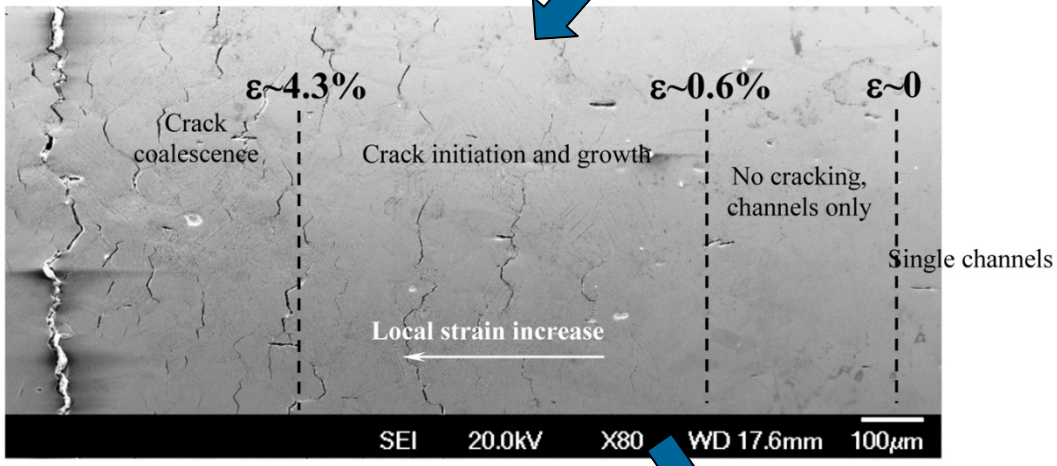
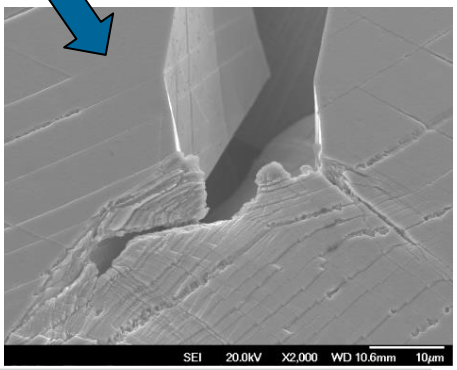
Irradiation Assisted Stress Corrosion Cracking Mechanisms

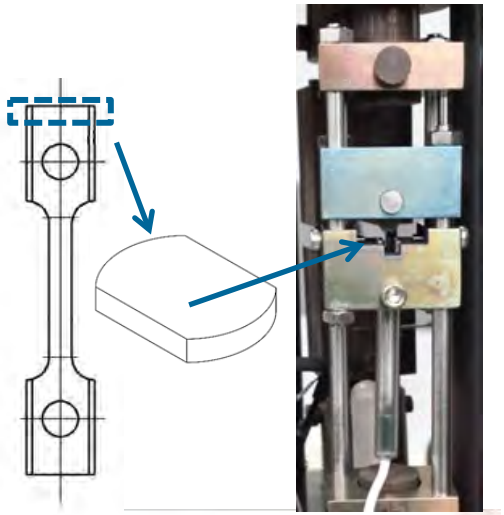
- Analysis of key contributing variables in IASCC to develop a mechanistic understanding from which mitigation or controlled management efforts be applied for long-term operations.
- Coordinated effort between: DOE/LWRS-ORNL, EPRI and U. of Michigan.
- Activities
  - ✓ Materials procurement and IASCC susceptibility evaluation
  - ✓ Mechanistic testing of IASCC conditions
  - ✓ Initiate modeling and theoretical studies to develop predictive capability
  - ➢ 2017: Initiate benchmark testing for IASCC using plant component materials
  - ➢ 2019: Predictive model capability for IASCC susceptibility



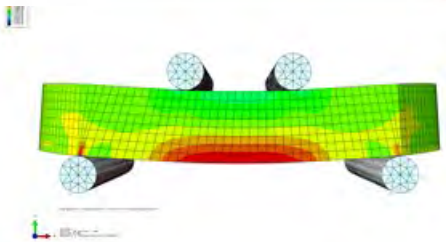




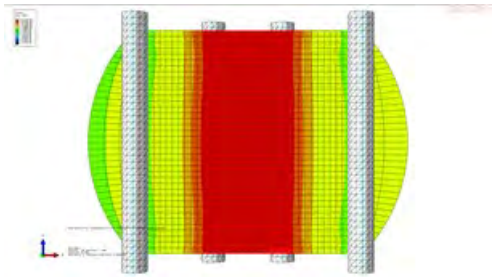
Multi-scale / Multi-tool Approach in IASCC



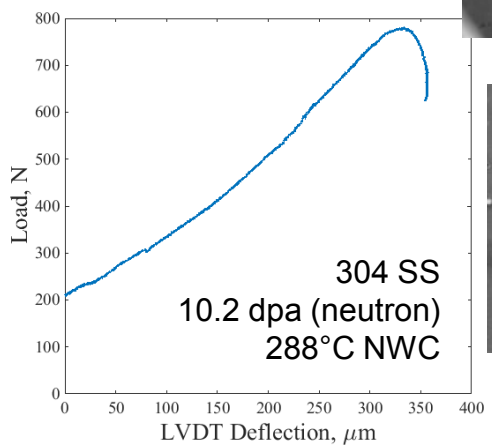
- Characterization of the corrosion processes and localized deformation in irradiated materials subjected to testing in a corrosion environment.
- 4-point bend test developed:
  - Similar conditions for crack initiation as continuous extension rate tensile tests
  - Creates IASCC initiation with minimal crack propagation.

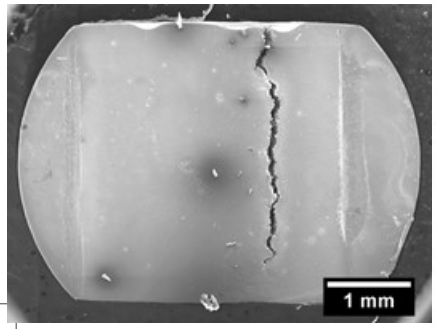


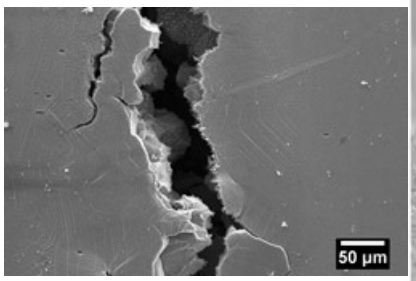
Side view of FEA simulation



Bottom view of FEA simulation

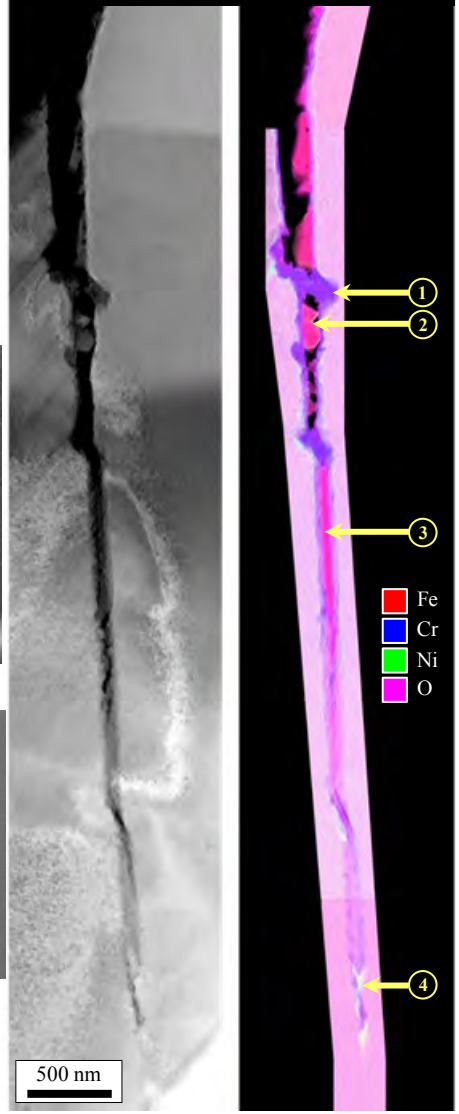






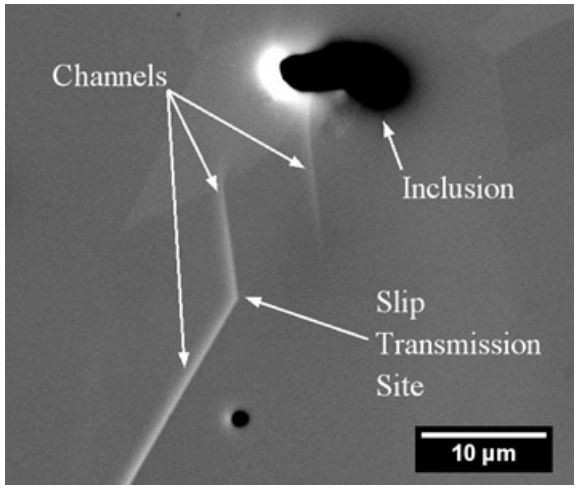
Anatomy of a crack

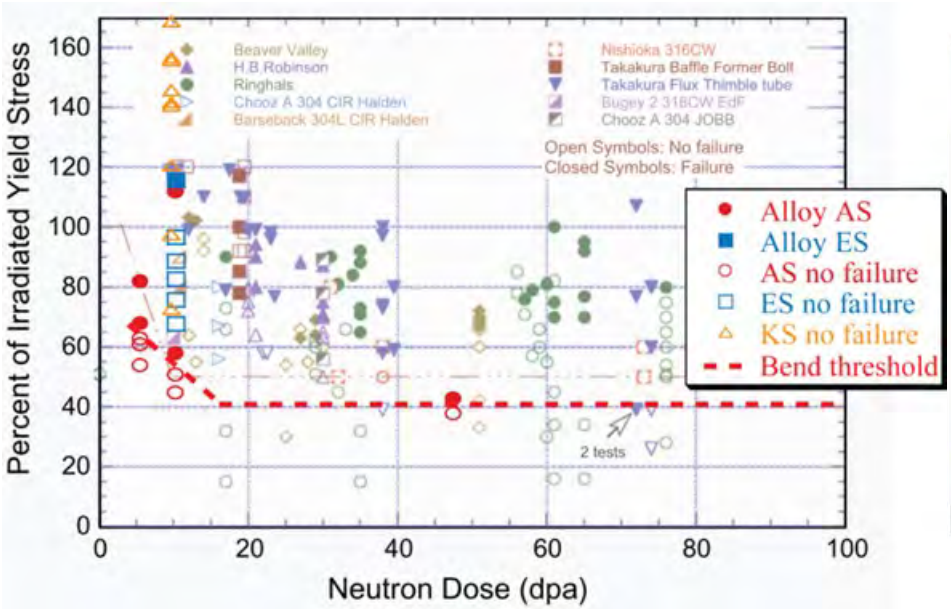
4.4 dpa, PWR-conditions

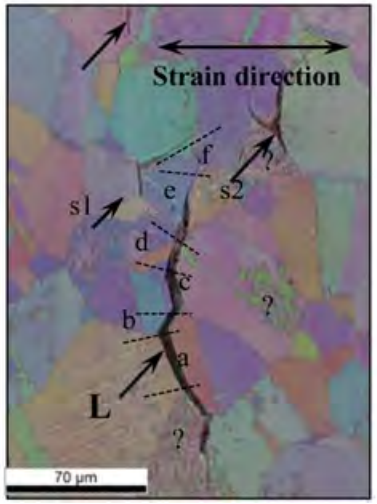


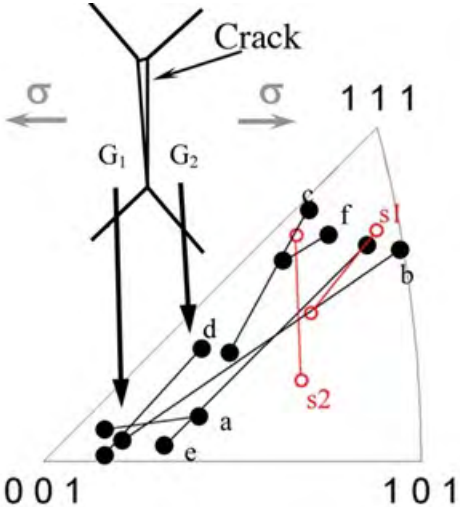
Understanding Mechanisms of IASCC

- Four point bend results reveal a stress threshold for IASCC initiation that decreases with dose in agreement with constant load tests.
- Alloy purity: type and amount of inclusions can have an effect on IASCC susceptibility - commercial vs. high purity 304L.
- Alloy microchemistry at grain boundaries and grain boundary orientation relative to the applied stress direction effects dislocation channel transmission - and susceptibility for crack initiation.









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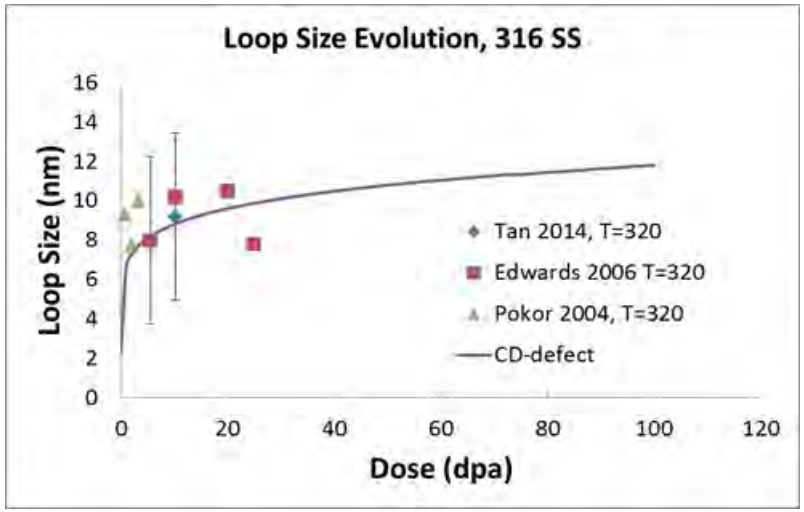
Precipitation Modeling in Irradiated Stainless Steels

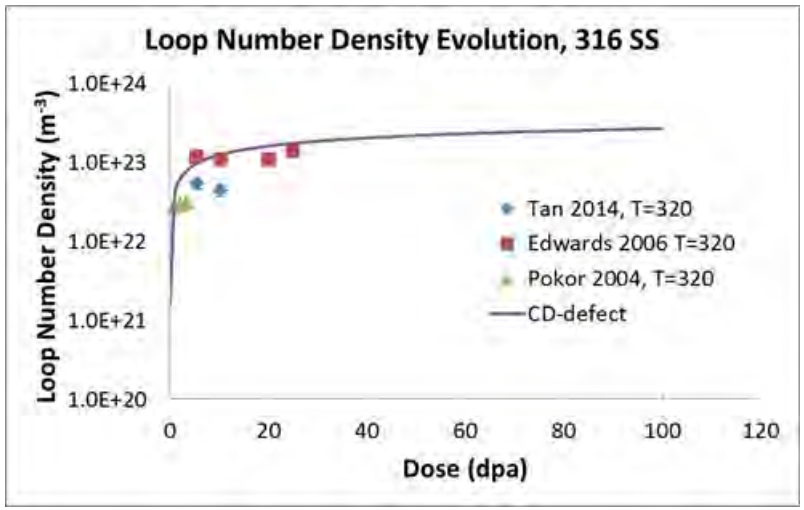
Project goals: Develop model that incorporates radiation-induced defect influences of radiation enhanced precipitation.

Phase 1: Developing a Cluster Dynamics code (CD-defect) and fitting to experiments to estimate sink (loop, void) and point defect evolution under irradiation:

- ✓ Successful model for predicting dislocation evolution and vacancy concentration to input into precipitation model



Loop size evolution of 316SS under irradiation



Loop number density evolution of 316 SS under irradiation

Predicting Phase Development

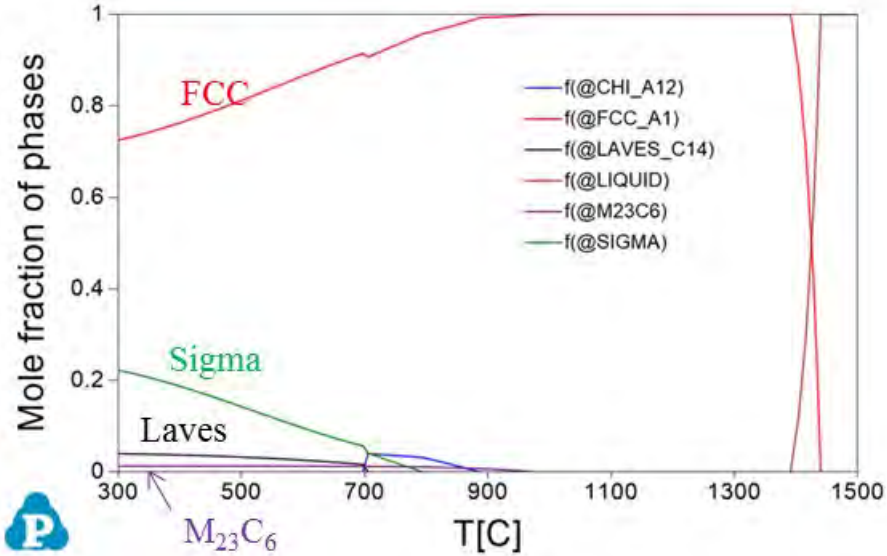
Phase 2: Predicting precipitation kinetics by combining the cluster dynamic defect predictions with classical nucleation theory precipitation models.

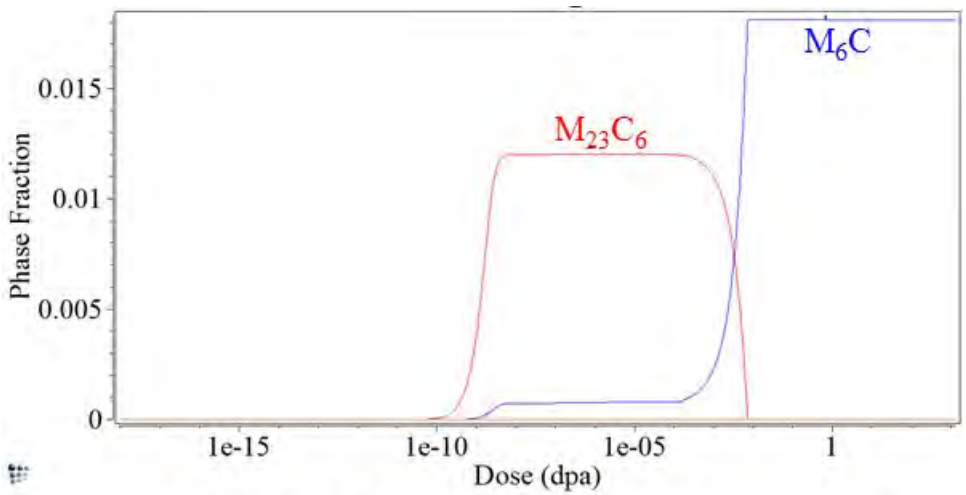
Under LWR Conditions (300°C):

- $M_{23}C_6$, Laves, and Sigma are stable, but the later two are kinetically inhibited.
- $M_{23}C_6$ is the fastest emerging phase but, it dissolve at expense of M_6C.

✓ Precipitation model results qualitatively match experiments

➢ Further validation in progress between low and high dose rate conditions.



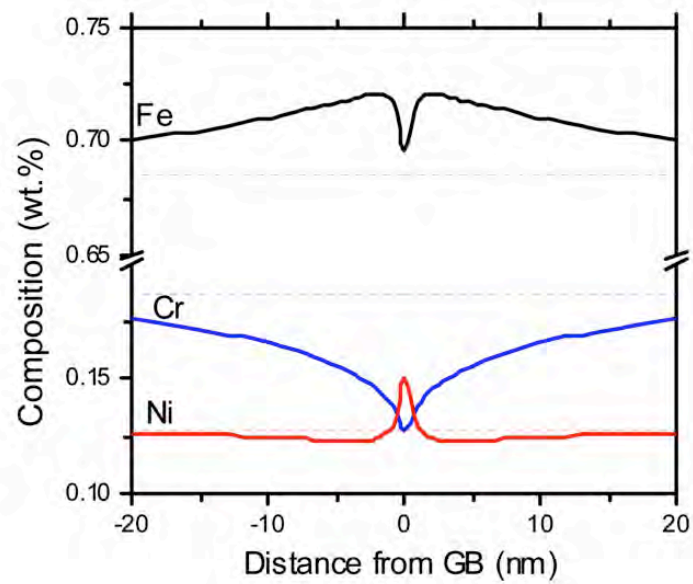


Precipitation in 316 SS with matrix composition.

Modeling Radiation-Induced Changes

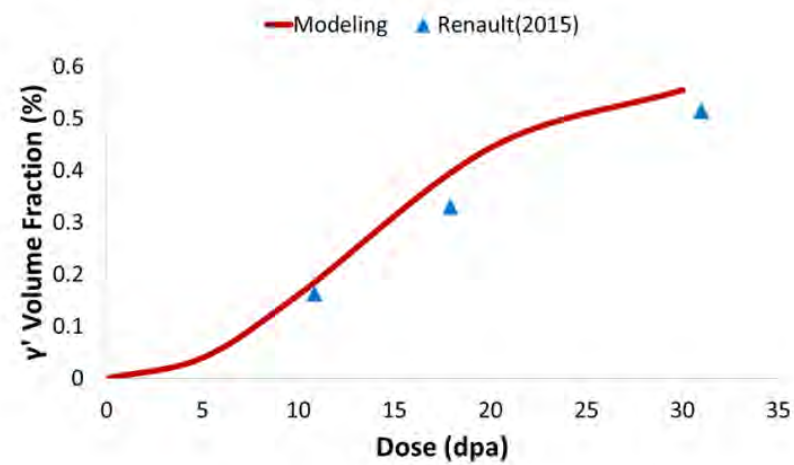
Radiation induced segregation (RIS) models:

Grain boundary interstitial-modified inverse Kirkendall model developed to account for RIS in austenitic alloys as a function of grain boundary character.



Radiation induced precipitation (RIP) models:

Prediction of γ' utilizing the RIS model, loop size/number density data and cluster dynamics model for vacancy concentration.



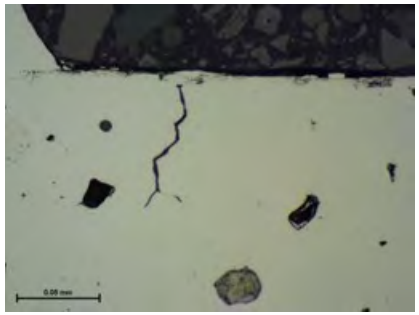
High fluence swelling models:

Further development of a predictive model to provide an evaluation of risk for high-fluence core internals:

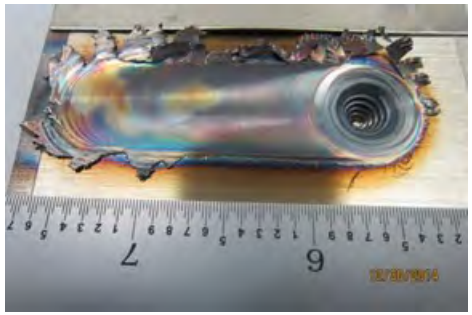
- Revised reaction-diffusion components of the model to account for additional diffusion mechanisms and relevant sink strengths for extended defects.
- Account for the neutron energy spectrum.

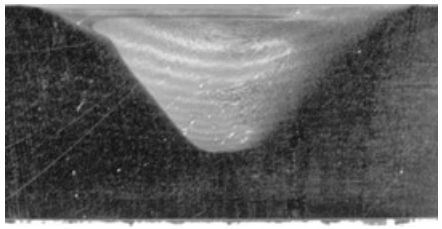
9.4×10^{-7} dpa/s irradiation at 390 °C in cold-worked 316 SS

Advanced Welding R&D May Provide Solutions to Repair and Mitigation Concerns



Huge voids and cracks with fusion welding

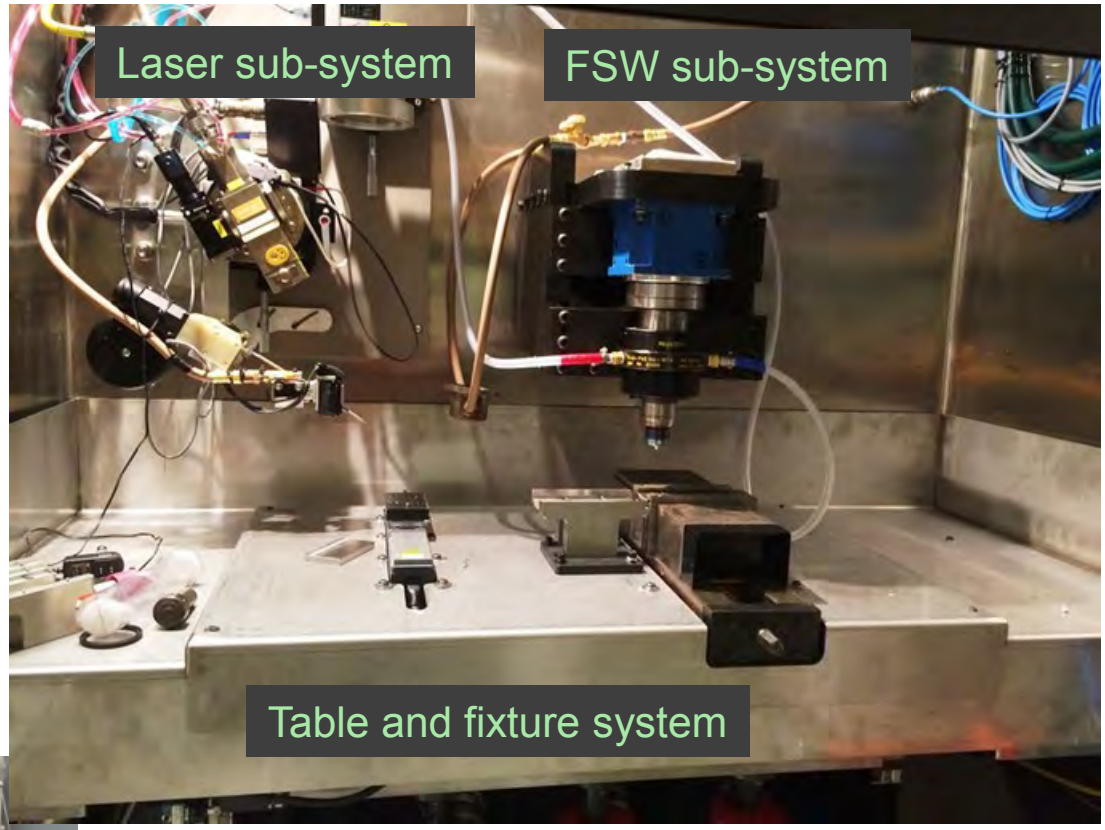




FSW and cross section

- Demonstration of advanced weldments techniques for irradiated materials - cost effective repair and replacement applications.
- Experimental and computational supported work.
- Recent work performed on high helium content stainless steel produced by power metallurgy.
- Friction stir welding (FSW) suppressed voids and cracks due to its solid state low welding temperature.
- Transfer of technology to industry.

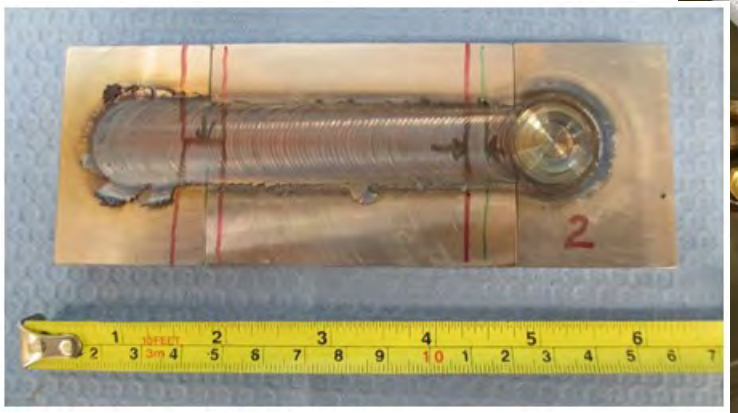
Welding Hot Cell Validation Experiment

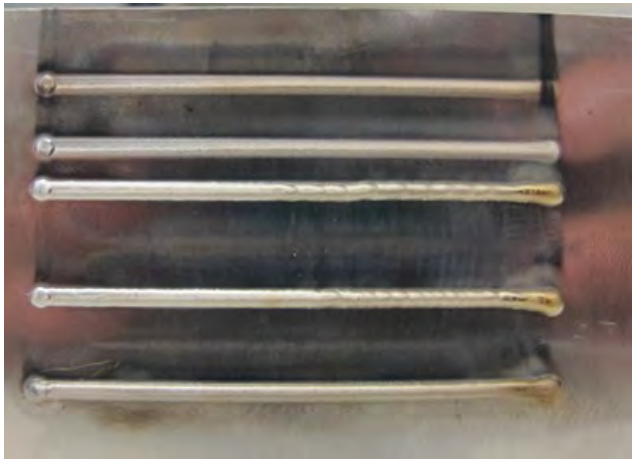


Inside Hot Cell

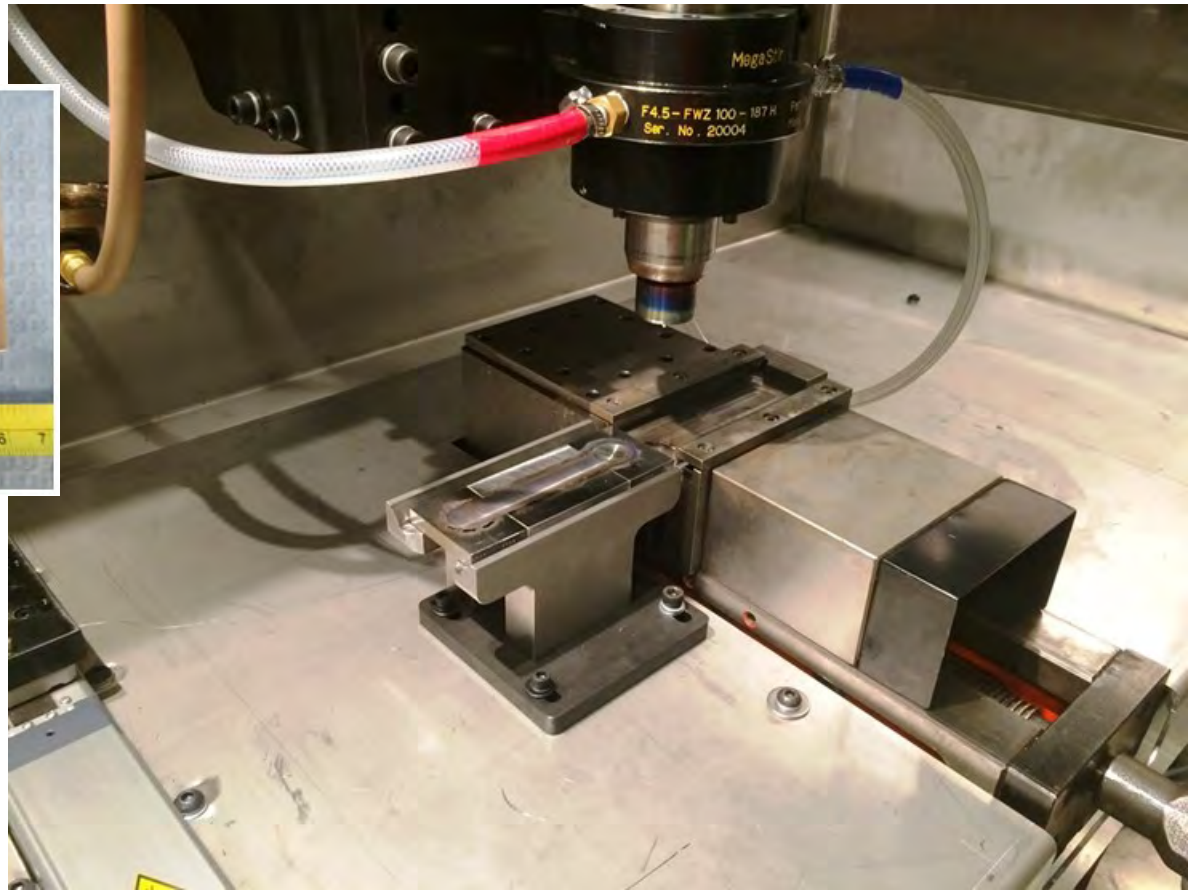
Welding Hot Cell Validation Experiment

Friction Stir Weld (FSW) Block





Laser Weld Test Block

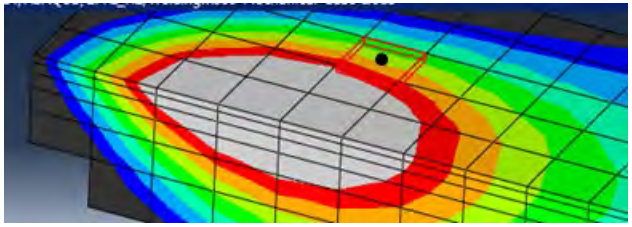


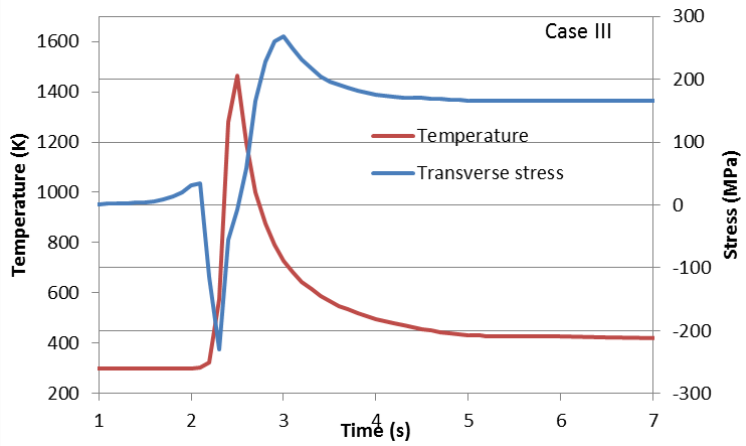
After FSW and transferring sample for pickup

FE Thermal and Mechanical Modeling and Experimental Validation

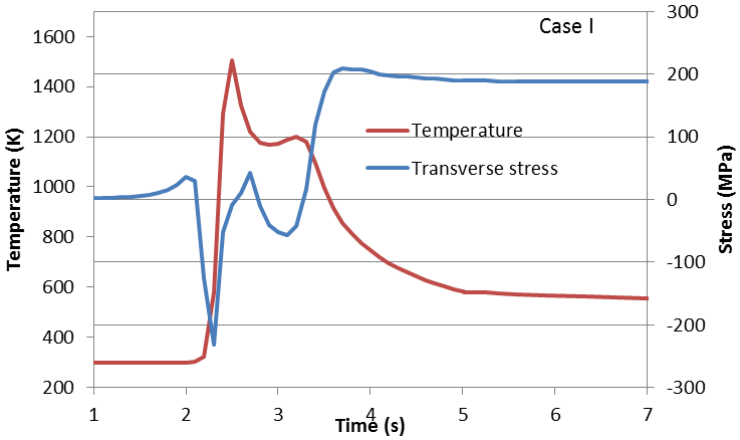
- FE models were developed to proactively manage the thermal input and control the stress distribution during the welding process.
  - With proactive thermal management, the tensile stress adjacent to the weld pool at temperatures above 1000K can be effectively reduced.

Modeling: adjacent to weld pool on top surface





Conventional method

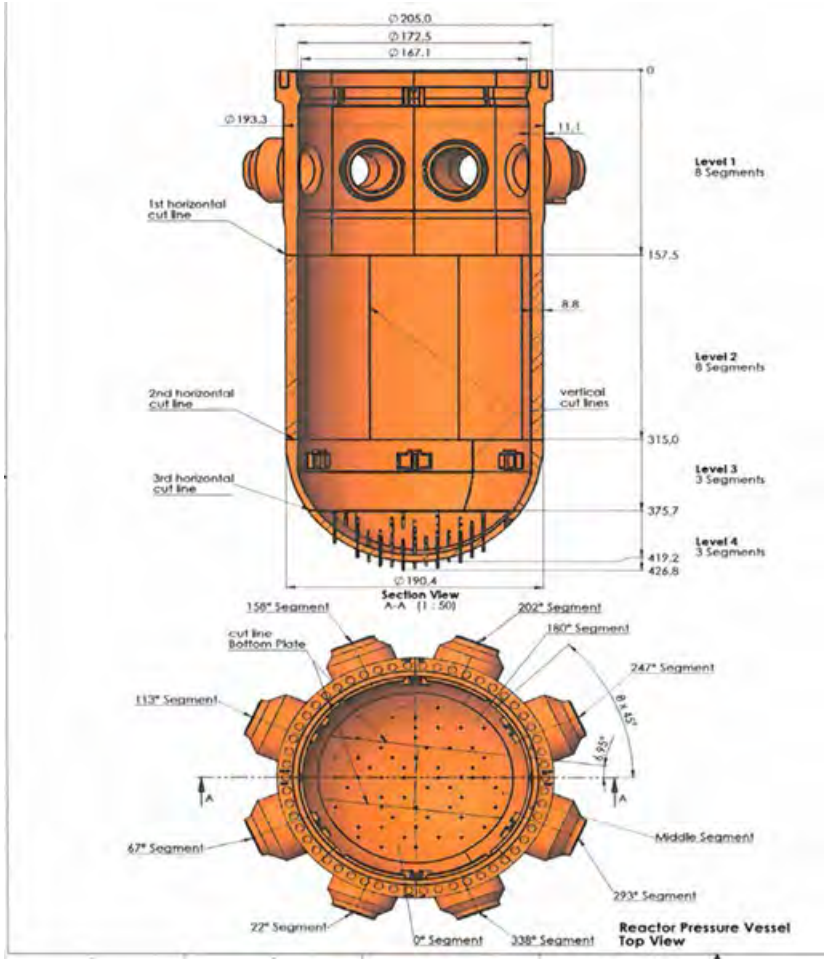


Proactive thermal management

Recent Publication: Chen, J, Yu, X, Miller, RG, and Feng, Z. (2015) *In situ strain and temperature measurement and modeling during arc welding.* Sci Tech Welding Joining, **20**(3):181-188..

Decommissioning of the Zion Unit 1 & 2: Opportunity to Examine Service-Aged Materials Degradation

- Evaluation of service-irradiated RPV material to validate physically-informed transition-temperature-shift models such as EONY model that is based on data from test reactor experiments and analysis of surveillance capsules.
- Provide service exposure data for WF-70 (Linde-80) welds.
- Opportunity evaluate through wall attenuation studies: *chemistry, mechanical (tensile, hardness, fracture toughness), microstructural studies, thermal annealing studies.*



Zion Unit 1 RPV Harvesting Plan

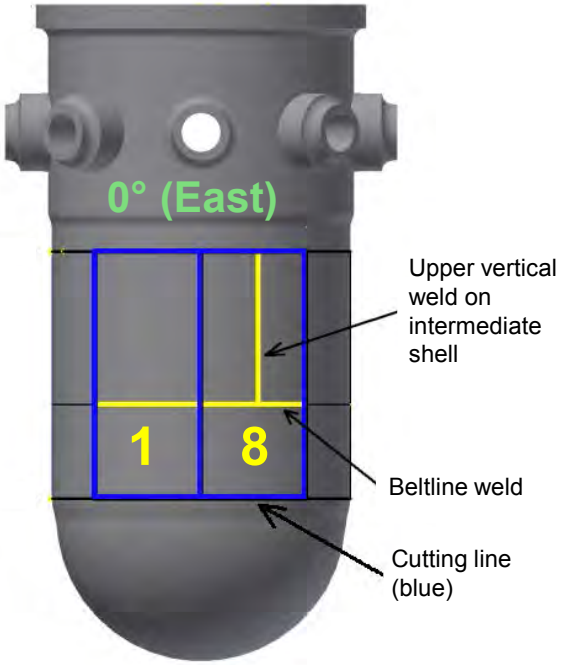
Segment 1: beltline weld only

- Linde 80 WF-70
- Beltline weld nominal composition: 0.27%Cu, 0.46%Ni and 1.6%Mn.
- Contains high dose region of WF-70 beltline weld, peak: $7x10^{18}$ n/cm^2 (>1MeV)
- Anticipated transition temperature shift of 145 °F

Segment 8: beltline and vertical axial welds

- Axial weld of Linde 80 WF-70, peak fluence of $2x10^{18}$ n/cm^2 (E>1MeV)

Level 2 (intermediate and lower shell) harvesting in November (Nozzles cut & harvested in September/October)

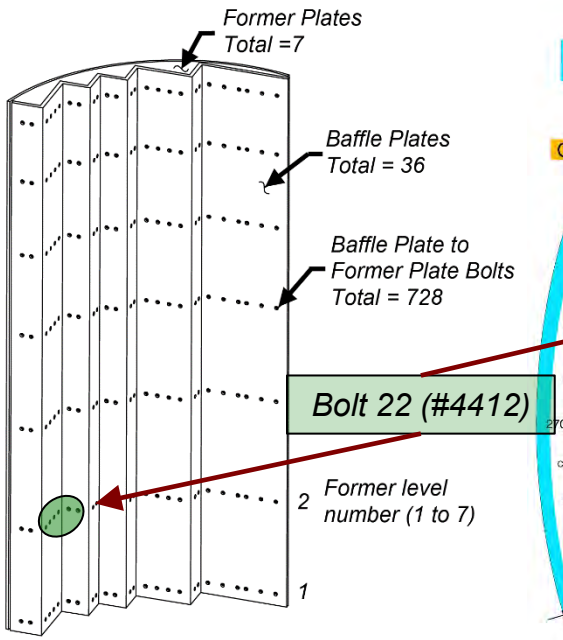


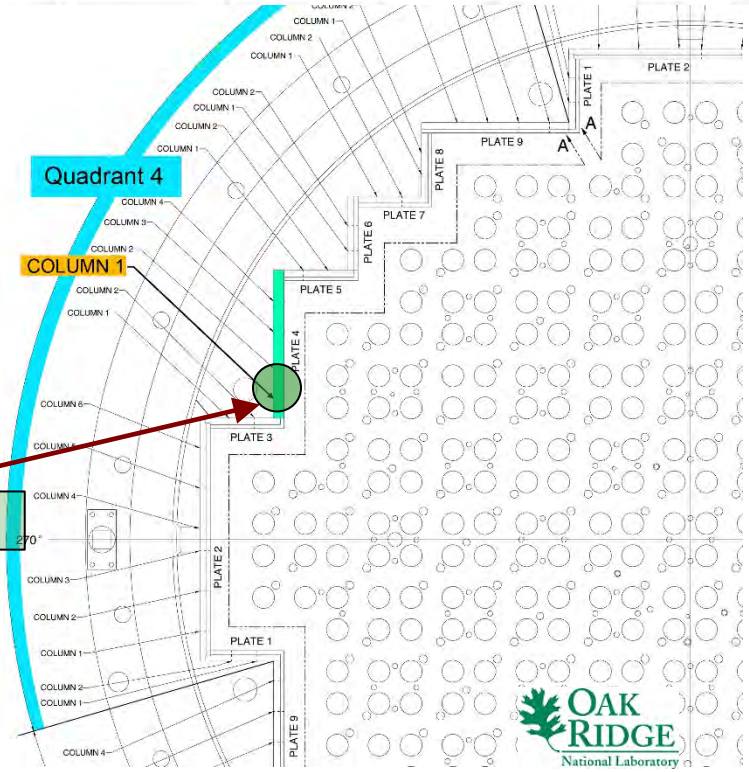
R.E. Ginna Baffle Former Bolts Examination Project

- Value in providing comparative information to research reactor irradiation data and computer model predictions.
- Highest fluence bolts selected: $1x10^{22}$ to $2.78x10^{22}$ n/cm^2 (E>1MeV).
- Inspection of bolts around high stress region: transition between bolt head and shank.
- Mechanical and microstructural characterization along the dose and temperature profile of bolt.

		Fluence (x10 ^{22} n/cm^2) E>1 MeV		
Bolt #	Ginna ID	Head	Mid-Shank	Mid-Thread
22	4412	2.78	2.27	1.46
21	3933	2.21	1.78	1.09
24	4424	2.20	1.77	1.09
23	4416	1.91	1.56	1.00
17	3814	1.84	1.56	1.11
20	3925	1.62	1.30	7.90
25	4435	1.62	1.30	0.79
26	3326	1.34	1.14	0.84
5	1262	1.00	0.80	0.49
15	3212	1.00	0.80	0.49
16	3215	0.99	0.80	0.49

Retrieval of bolts from spent fuel pool February 2016.





Discussion?

LWRS

Light Water Reactor Sustainability

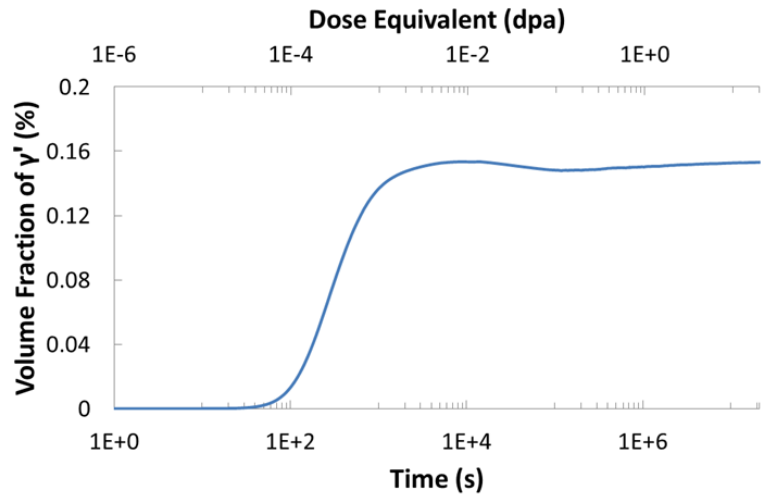
Additional Slides

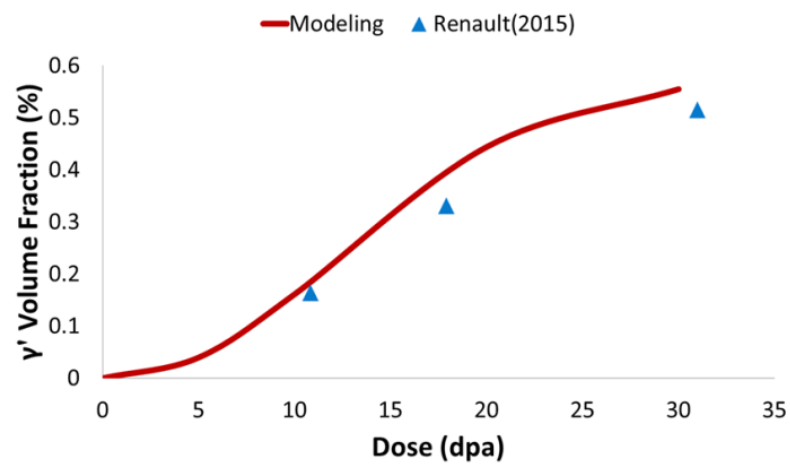
Predicting Radiation Induced γ′ Development

***Phase 2 results of radiation-induced precipitation models**….. Continued*

- Complex model development assuming γ′ forms inside the cylindrical regions around dislocations that act as sinks producing radiation induced segregation.
- Model uses experimental radiation induced segregation and loop size/number density data with Cluster Dynamics model for vacancy concentration giving for the the radiation enhanced diffusion.

✓ The proposed model is successful in predicting the γ′ evolution under irradiation in fast reactors.

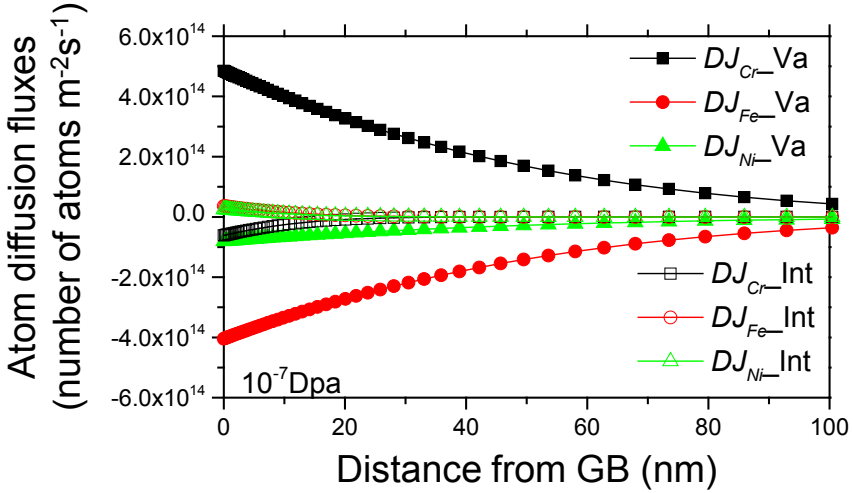




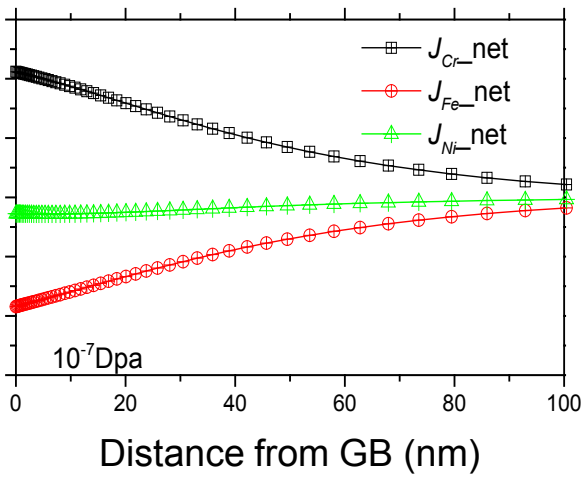
9.4×10^{-7} dpa/s irradiation at 390 °C in cold-worked 316 SS

Development of Computational Tools for Simulating Radiation-Induced Segregation in Steels

Diffusion fluxes through preferential coupling with Vacancy and Interstitial



Net diffusion fluxes

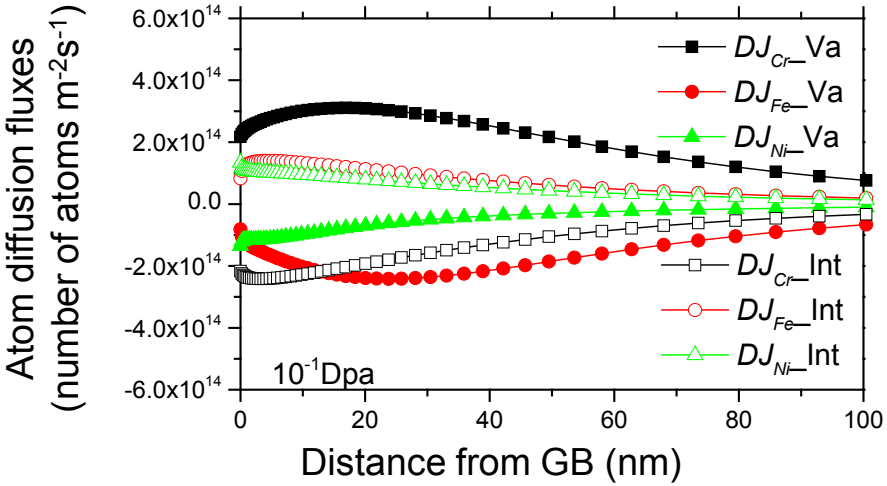


Motivation: Understanding the role of vacancy / interstitial diffusion have in changing grain boundary microchemistry of irradiated Fe-Cr-Ni alloys.

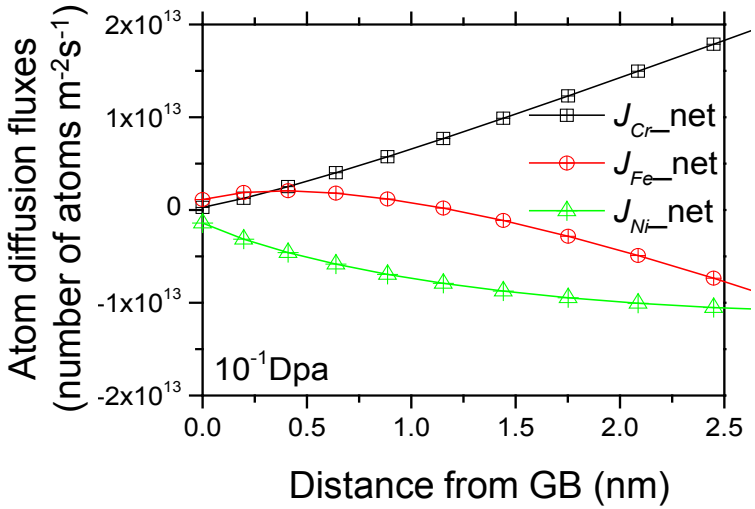
- Shown above: radiation-induced segregation in the transient stage, the major diffusion mechanism of Cr, Fe and Ni is via vacancy - with Ni being not so dominant.
- As a result: Fe and Ni enrichment and Cr depletion occur at this stage.

Modeling the Chemistry Near Grain Boundaries

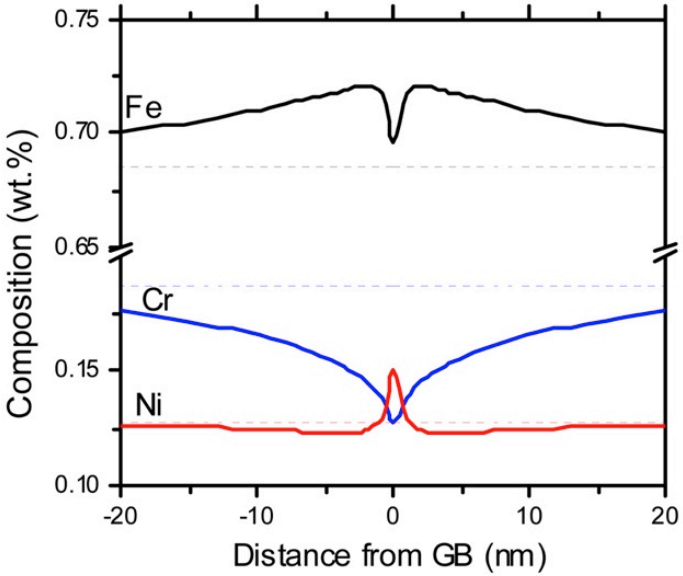
Diffusion fluxes through preferential coupling with Vacancy and Interstitial



Net diffusion fluxes



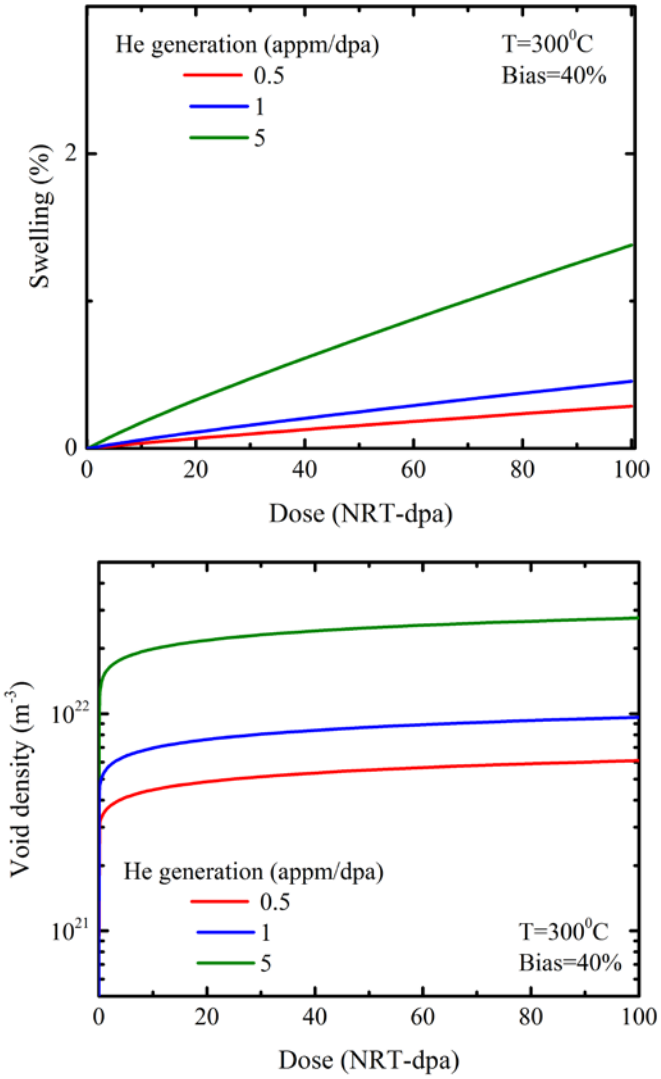
- Chemistry changes at grain boundaries modifies the magnitude and shape of the diffusion flux.
- Cr is still depletion, Ni becomes enriched in the vicinity of the grain boundary but depletes in the outlying region. Fe depletes at GB but enriches in further distance (W-shape profiles).



Continued Development of a Comprehensive Radiation-Induced Swelling Model

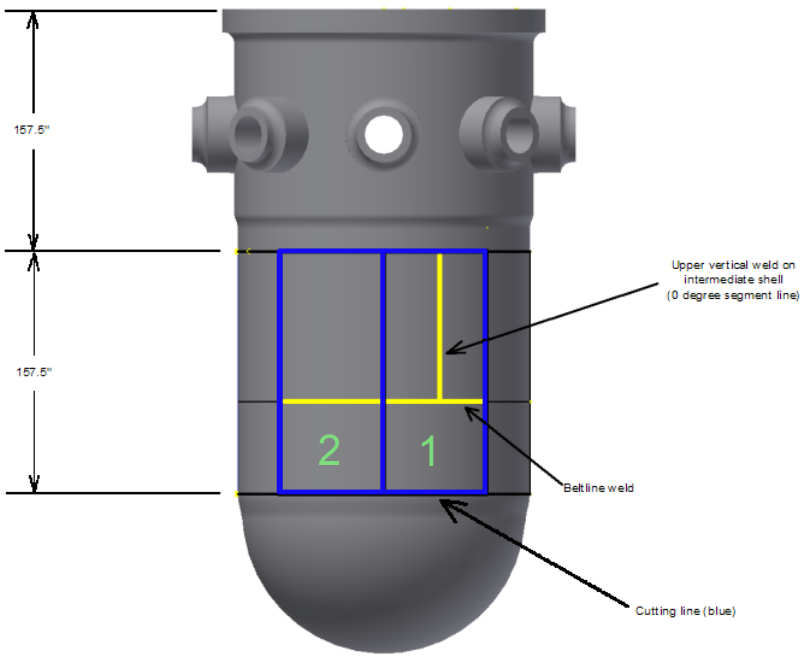
Development of a predictive model to provide an evaluation of risk for high-fluence core internals.

- Modified description of primary radiation damage formation to account for results obtained from atomistic simulations of damage production.
- Improved grouping scheme to discretize defect cluster populations to provide greater fidelity in describing nucleation and growth of size distributions
- Revised reaction-diffusion components of the model to account for additional diffusion mechanisms and relevant sink strengths for extended defects.
- Account for the neutron energy spectrum.



Segmentation of Unit 2

- Segments to include beltline with vertical axial weld locations.
- ***Problems during cutting operations:***
  - Due to the position of overhead bridge the gripper crane could not move the necessary 22.5-degrees offset to center the vertical weld (section 1).
  - Cutting was too close to the axial weld creating a temperature rise.
- Thermal analysis suggested temperature from flame cut < 250 C ~ 0.5 – 1.0” from weld
  - Determination: too close considering low fluence location of axial weld



U.S.NRC

UNITED STATES NUCLEAR REGULATORY COMMISSION

Protecting People and the Environment

Subsequent License Renewal (SLR) Key Technical Issues

Office of Nuclear Reactor Regulation
Division of License Renewal

Office of Nuclear Regulatory Research
Division of Engineering

November 17, 2015

Agenda

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- SLR Schedule
- Department of Energy and Electric Power Research Institute presentation on reactor coolant system metals and reactor pressure vessel
- NRC staff presentations on fluence, reactor vessel neutron embrittlement and internals
- Concrete and containment degradation
- Electrical cable qualification and condition assessment

SLR Schedule

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Protecting People and the Environment

Timeframe	Description
December 2015	• Issuance of Draft Generic Aging Lessons Learned (GALL) Report and Standard Review Plan (SRP) for SLR • Public comment period begins
January 2016	• Public meeting on draft SLR GALL Report and SRP
February 2016	• Public meeting on draft SLR GALL Report and SRP • ACRS Sub-Committee Meeting • Public comment period ends
March 2016	• Regulatory Information Conference Sessions
February 2017	• ACRS Full-Committee Meeting
June 2017	• Issuance of final SLR GALL Report, SRP and Technical Basis NUREGs
2019	• First SLR Application

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Neutron Fluence Evaluations for Subsequent License Renewal (SLR)

Ben Parks and Matthew Hardgrove
Office of Nuclear Reactor Regulation
Division of Safety Systems

Jay Wallace
Office of Nuclear Regulatory Research
Division of Engineering

November 17, 2015

Outline

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Current Regulatory Framework & Technical Aspects
- Technical Issues
- Efforts to Resolve Technical Issues
- Status of Work
- Schedule and Milestones
- Interim Measures

Regulatory Framework

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- Regulatory framework requires addressing effects of neutron irradiation on reactor vessel embrittlement:
  - Neutron irradiation damage must be factored into reactor vessel embrittlement evaluations in accordance with:
    - General Design Criteria (GDC) 14 and 31
    - 10 CFR 50.60 and 10 CFR 50.61/50.61a
    - 10 CFR Part 50, Appendix G
  - Neutronic exposure must also be factored into reactor vessel material surveillance programs in accordance with GDC 32 and 10 CFR Part 50 Appendix H
- NRC provides guidance for determining pressure vessel neutron fluence in Regulatory Guide (RG) 1.190, “Calculational and Dosimetry Methods for Determining Reactor Pressure Vessel Fluence”

Technical Aspects

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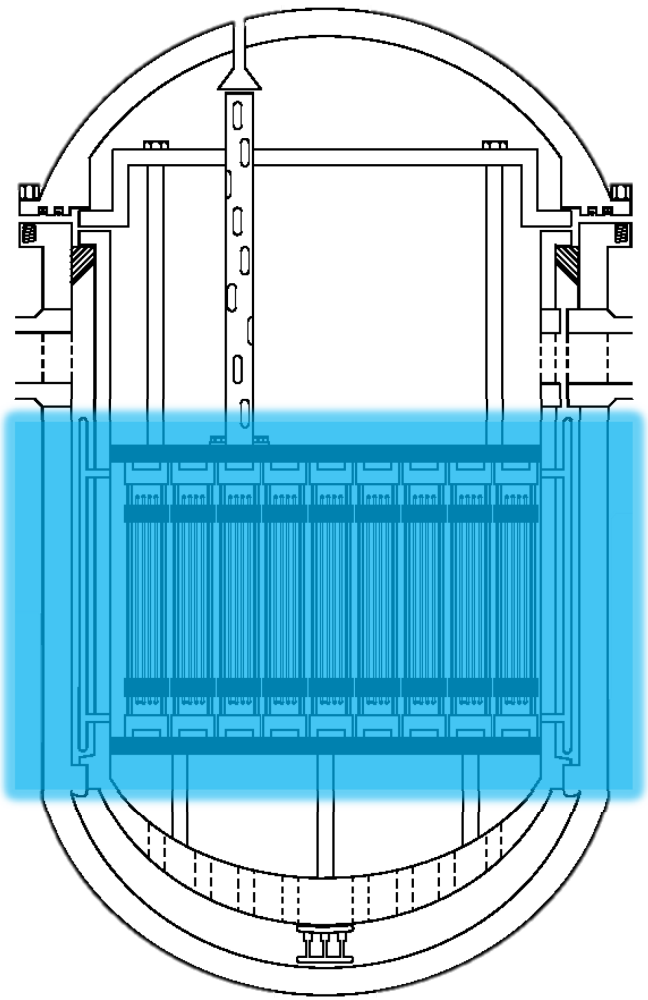
Protecting People and the Environment

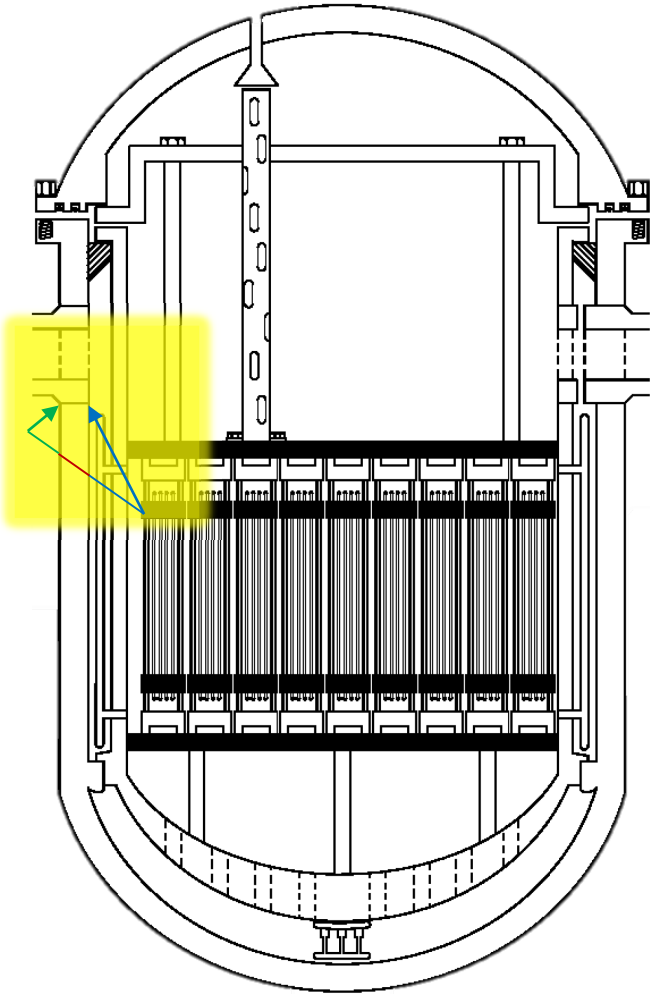
- Most fluence estimates are determined using computational methods, consistent with guidance contained in RG 1.190
- Guidance for calculating pressure vessel fluence:
  - Collapsing nuclear data
  - Modeling the core neutron source
  - Performing discrete ordinates or Monte Carlo transport calculations
- Guidance for qualifying fluence methods for pressure vessel embrittlement analysis:
  - Standard benchmarking experiments
  - Computational benchmark problems
  - Plant-or problem-specific qualification
  - Uncertainty requirements
    - ± 20% acceptable for reactor vessel embrittlement analysis

Technical Issues



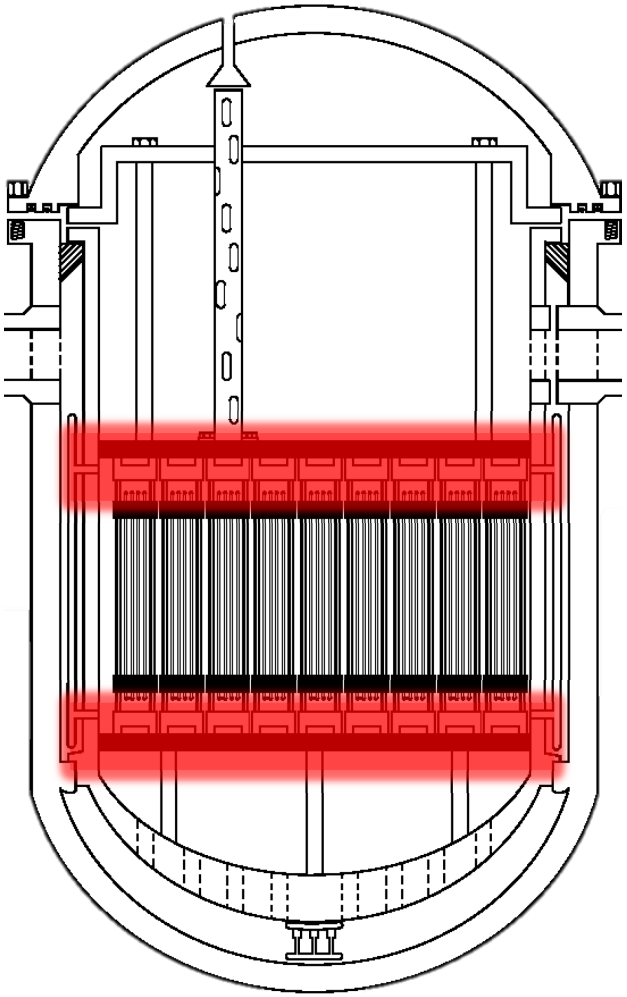
- RG 1.190 is intended for “beltline” region fluence calculations:
  - More detailed modeling is required for peripheral assemblies
  - Angular quadrature requirements are based on distance from core periphery to vessel
- For original, 40-year license, “beltline” region coincided with active core height
- Increased exposure causes the beltline to expand
  - More of the vessel surface area exceeds a fluence of 10^{17} n/cm^2

Technical Issues



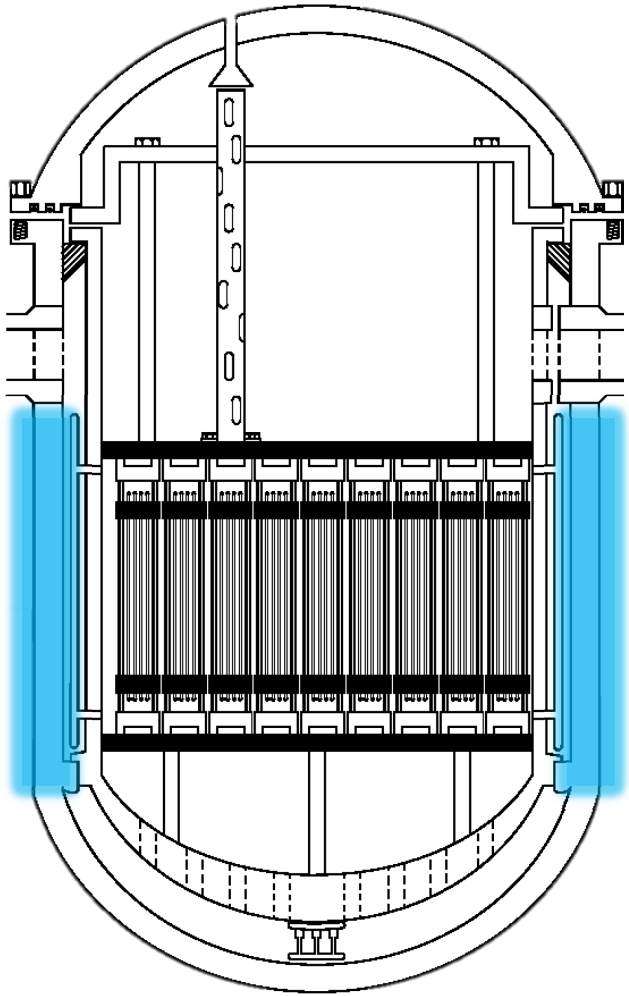
- Additional vessel components are treated as beltline materials
  - Fluence may become important for nozzles and nozzle welds
  - Effects of cavity streaming may become important
  - Neutrons travel greater distances to target

Technical Issues



- Irradiation embrittlement and irradiation-assisted stress corrosion cracking become a concern for reactor vessel internals close to the core
  - Incident neutrons originate from central assemblies, rather than peripheral assemblies
  - Guidance for vessel fluence calculations allows simpler representation of central fuel assemblies

Technical Issues



- Application-specific qualification of neutron fluence methods has been focused on the “beltline” region of the vessel

Summary Technical Issues

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- Guidance for determining fluence is focused on the reactor vessel, in an area surrounding the reactor core
- Broadening the application of RG 1.190-adherent fluence methods requires new guidance:
  - Calculating fluence over greater transport distances (more mean free paths) from the core
  - Calculating fluence in regions other than the core periphery (e.g., vessel nozzles and upper internals)
  - Using methods other than discrete ordinates or Monte Carlo
  - Qualifying methods to calculate fluence in different locations
  - Using fluence methods for other than reactor vessel embrittlement evaluations

Efforts to Resolve Issues

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- NRR issued User Need Request 2015-002
- RES is developing a technical basis to revise RG 1.190 or issue additional guidance

Task	Purpose	Schedule

Study and determine highest-importance phenomena	• Ensure resources are applied to most important phenomena:
	  • For NRC, while performing research
	  • For stakeholders, while developing and qualifying methods and calculating fluence
		July 2016

Determine best-suited types of methods	• Verify that existing methods remain appropriate • Broaden guidance to other types of methods	July 2017

Determine appropriate calculational rigor	• Provide guidance relevant to:
	  • Additional types of methods
	  • Additional problem geometries
		January 2018

Efforts to Resolve Issues

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Task	Purpose	Schedule
Identify available benchmarking data	• Attempt to provide in-situ or experimental benchmarks based on existing data • Minimize need to perform new experiments or gather new data	September 2018
Determine reasonable qualification requirements	• Ensure extending fluence calculations to additional regions produces accurate and reliable results • Avoid unduly burdensome qualification exercises	September 2018

Interim Measures

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- Fluence calculations for locations like reactor coolant nozzles or reactor vessel internals are considered on a case-by-case basis
- Decisions are based on an evaluation of the following:
  - Demonstrated reliability of methods
  - Availability of appropriate qualification data
  - Margins available to applicable limits
  - New neutron fluence aging management program

United States Nuclear Regulatory Commission

Protecting People and the Environment

Reactor Pressure Vessel Embrittlement at High Fluence

Carolyn Fairbanks and Allen Hiser
Office of Nuclear Reactor Regulation
Division of Engineering
Division of License Renewal

Mark Kirk and Robert Tregoning
Office of Nuclear Regulatory Research
Division of Engineering

November 17, 2015

Outline

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- Discussion overview
- Regulatory framework for reactor pressure vessel (RPV) aging management during the period of subsequent license renewal (SLR)
- Ongoing work concerning formulae to predict embrittlement effects
- Generic evaluation criteria used to assess embrittlement effects
- Summary

Overview of Discussion

Three Topics

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SLR Impact

1. Regulatory Framework

Is there a regulatory framework in place adequate to address RPV aging management through 80 years of operation?

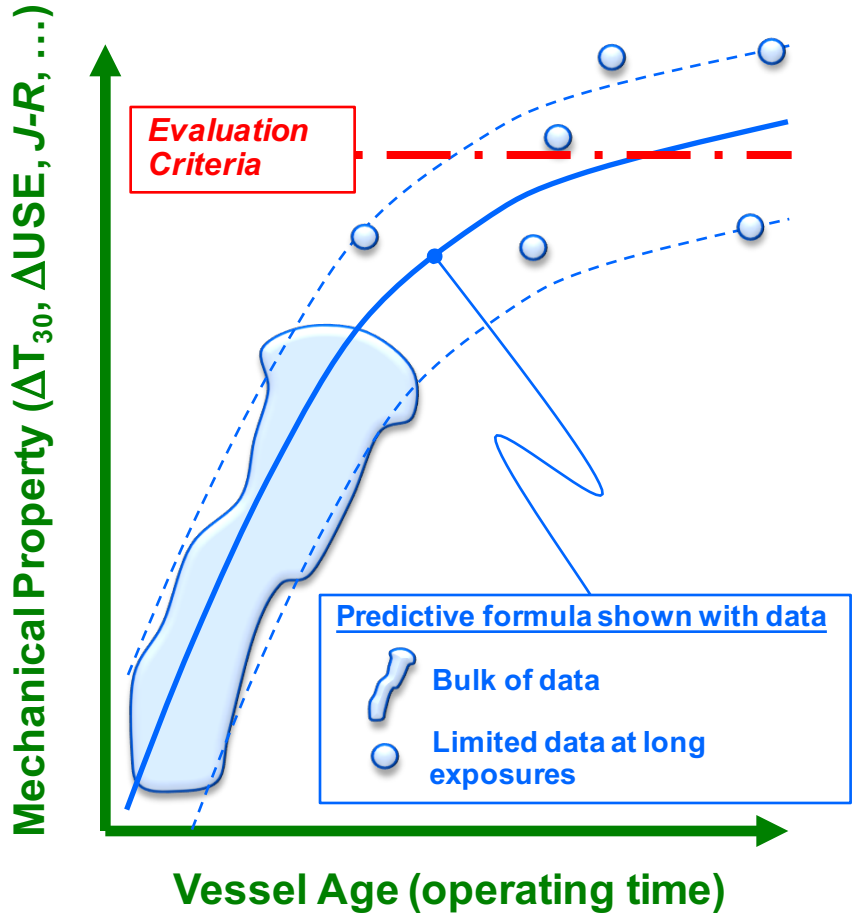
On-Going / Related Work

2. Predictive Formulae

Are the models used to predict/represent data trends validated/confirmed to 80 years of neutron exposure?

3. Generic Evaluation Criteria

Have evaluation criteria been established? Are they likely to be re-evaluated because of licensing through 80 years?



1. Regulatory Framework

Summary of 10 CFR Part 54 & 10 CFR Part 50 Requirements

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10 CFR PART 54: Requirements for Renewal of Operating Licenses for Nuclear Power Plants

§ 54.21 Contents of application, technical information

(c) An evaluation of time-limited aging analyses.

(1) A list of time-limited aging analyses, as defined in § 54.3, must be provided. The applicant shall demonstrate that:

(i) The analyses remain valid for the period of extended operation;

(ii) The analyses have been projected to the end of the period of extended operation; or

(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation

10 CFR Part 50 Requirements & Regulatory Guidance (RG)

- 10 CFR 50 Appendix H
- 10 CFR 50 Appendix G
- 10 CFR 50.61
- 10 CFR 50.61a
- 10 CFR 50.55a
- 10 CFR 50.66

- RG 1.99
- RG1.161
- RG1.162
- Draft RG-1299

1. Regulatory Framework

Assessment

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- 10 CFR Part 54 explicitly requires licensees to manage aging, ensuring component functionality and plant safety
- Methods to demonstrate management/compliance expected to mirror practices established in 10 CFR Part 50
  - 10 CFR Part 50 incorporates the American Society of Mechanical Engineers Code and certain ASTM Standards by reference
- Adequacy of equivalent margins analysis documents need to be demonstrated to 80 years
- **Conclusion**
  - Nothing unique/different needed in 10 CFR Part 54 for SLR, therefore,
  - Existing regulatory framework adequate to 80 years

2. Predictive Formulae

Overview

- Where are we making predictions?
- What are we predicting?
- Documents containing prediction formulae

2. Predictive Formulae

Where are we making predictions?

Photo Credit: https://www.asme.org/about-asme/who-we-are/engineering-history/landmarks/47-shippingport-nuclear-power-station

- Traditionally irradiation effects monitored for materials with fluence > 1×10^{17} n/cm^2)
- In extended operations (40-60, 60-80 years) this region enlarges
  - May encompass nozzle course
    - Unirradiated properties sometimes not available – need generic values
    - Through-wall attenuation near nozzles may differ from central shell course (see neutron fluence presentation)

2. Predictive Formulae

What are we Predicting?

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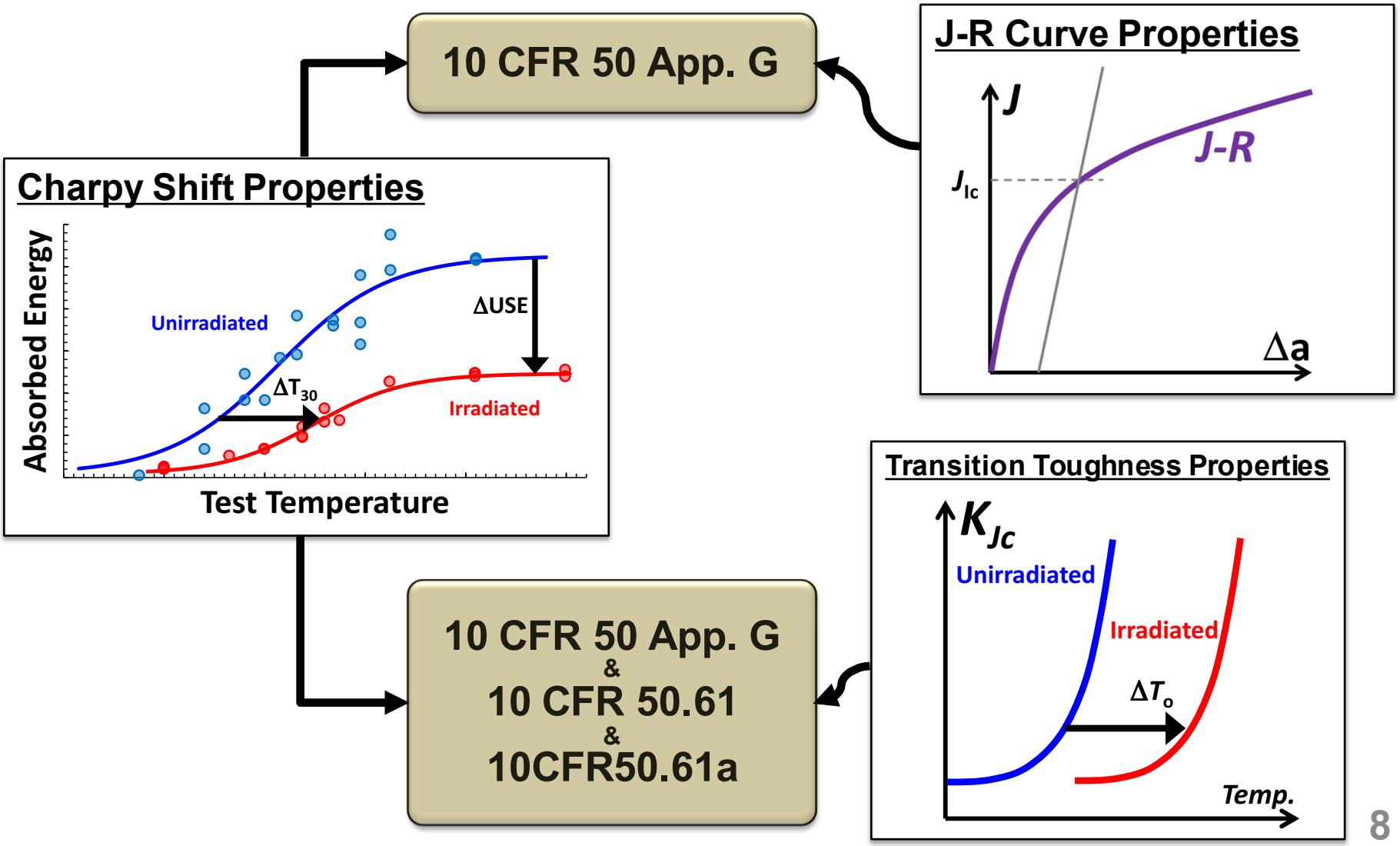

2. Predictive Formulae

Appearing in Regulations & Guidance

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	Documents	Last Revised by NRC	Last Reviewed by NRC
ΔT_{30}	RG 1.99	1988	2014 (full report due in 2016)
	10 CFR 50.61	1996	Mid 2000s
	10 CFR 50.61a	2010	2010
ΔUSE	RG 1.99	1988	2014 (full report due in 2016)
J-R	RG 1.161	1995	2014
Surveillance	10 CFR Part 50 Appendix H	1983	Underway (in Rulemaking)

2. Predictive Formulae

Appearing in consensus Code & Standards

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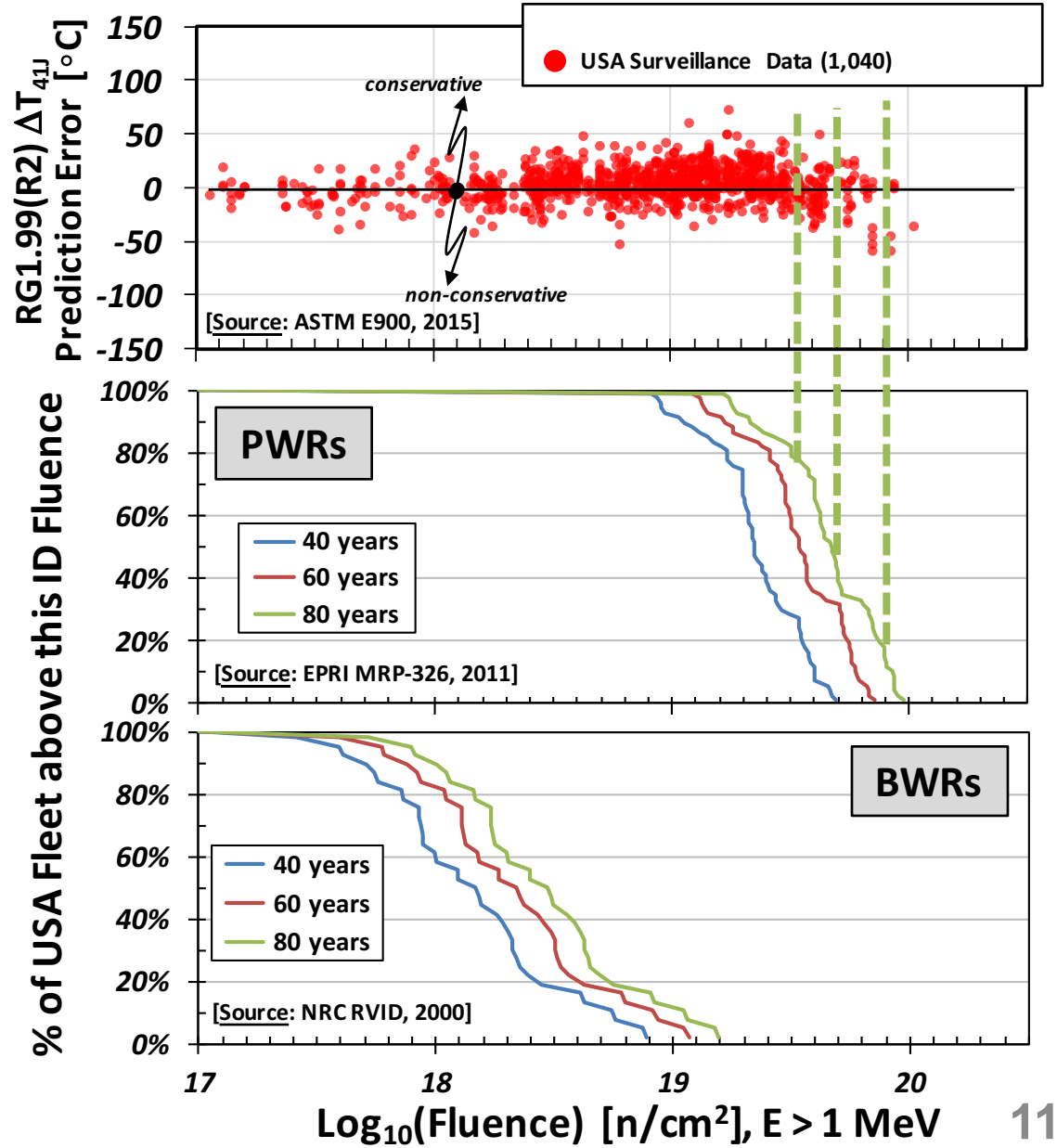
	Documents	Last Reviewed and Revised by Code and Standards Committee	Last Reviewed by NRC
ΔT_{30}	ASTM E900	2015	--
K_{Ic} & K_{Ia}	ASME SC-XI App A & G	2014	Underway
Surveillance	ASTM E185	2015	Underway*
	ASTM E2215	2015	Underway*
Comprehensive Models	ASME CC-N830-R1	2014	--

* NRC review underway as part of 10 CFR Part 50 Appendix H rulemaking

2. Predictive Formulae

Current Status

- **ΔT_{30} & ΔUSE**
  - Evaluation & on-going work
    - Comprehensive ΔT_{30} database compiled by ASTM. Staff has access to database & tech basis docs, but has not yet reviewed.
    - Staff report due in 2016 on RG1.99 (both ΔT_{30} & ΔUSE)
    - 2 industry programs collecting more high fluence data (CVSRP, PSSP)
  - Additional considerations
    - Surveillance capsule lead factors ensure data availability before effects manifest in-vessel
    - Generic trends informed by plant-specific data
- ***J-R*, K_{Ic}, K_{Ia}**
  - Evaluation & on-going work
    - Activity underway within ASME Code (Code Case N-830, Working Group on Flaw Evaluation)
  - Additional considerations
    - Large margins & structural factors on driving force



2. Predictive Formulae

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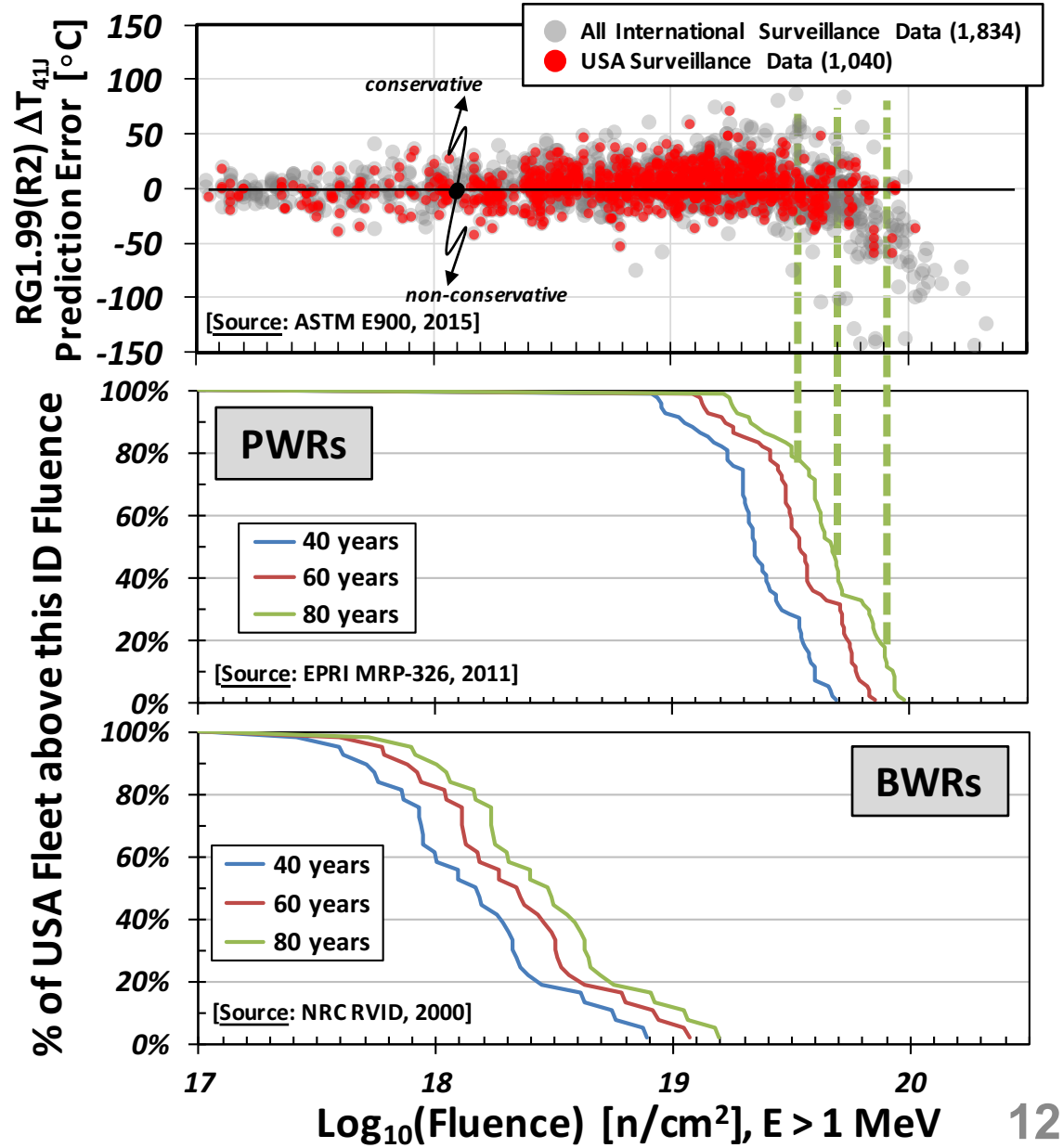

3. Generic Evaluation Criteria

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- NRC assessment of the adequacy of current evaluation methods (including the associated predictive formula) informs the need for potential future modifications
  - Generic revision only considered for issues affecting many plants
- This approach is consistent with past efforts by both NRC and industry
  - NRC Example: 10 CFR 50.61a
  - Industry Examples: Master curve
    - BAW-2301 (Zion, Point Beach, Three Mile Island)
    - Kewaunee
- Some on-going efforts may factor into any needed revisions:
  - NRC: Continuing to evaluate possibility of risk-informed revision of 10 CFR Part 50 Appendix G
  - Consensus Codes & Standards: Alternative fracture toughness-based flaw evaluation criteria (ASME SC-XI Code Case N-830-R1)

Summary

60-80 Years of Operation

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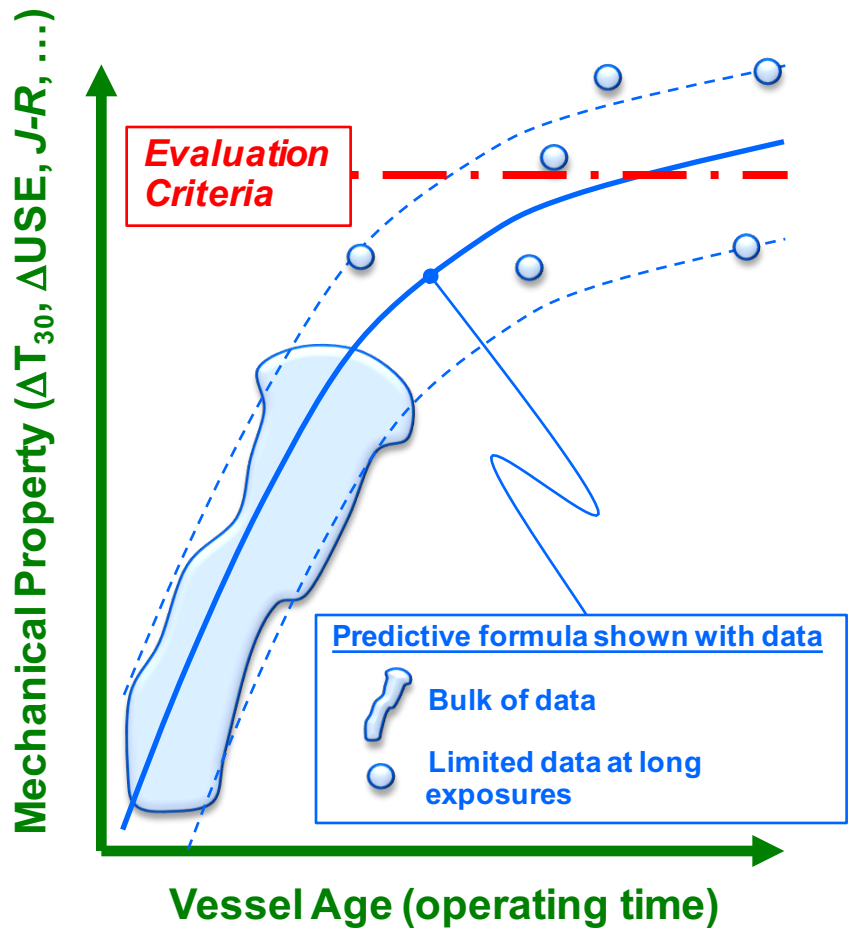
SLR Impact	
	1. Regulatory Framework Adequate to 80 years
On-Going / Related Work	**2. Predictive Formulae** – Adequacy of all formulae now under evaluation – Continuous process … will continue past this GALL update – While full evaluation is underway the adequacy of current approaches remains ensured • Lead factors provide advance data • Plant-specific indexing of generic trends **3. Generic Evaluation Criteria** – May be updated in future, no current plans – Need for updates driven by developing knowledge on embrittlement trends and plant operating needs – Removal of unnecessary conservatism undertaken consistent with Commission policy guidance



Backup Slides

Distribution of High Fluence Data

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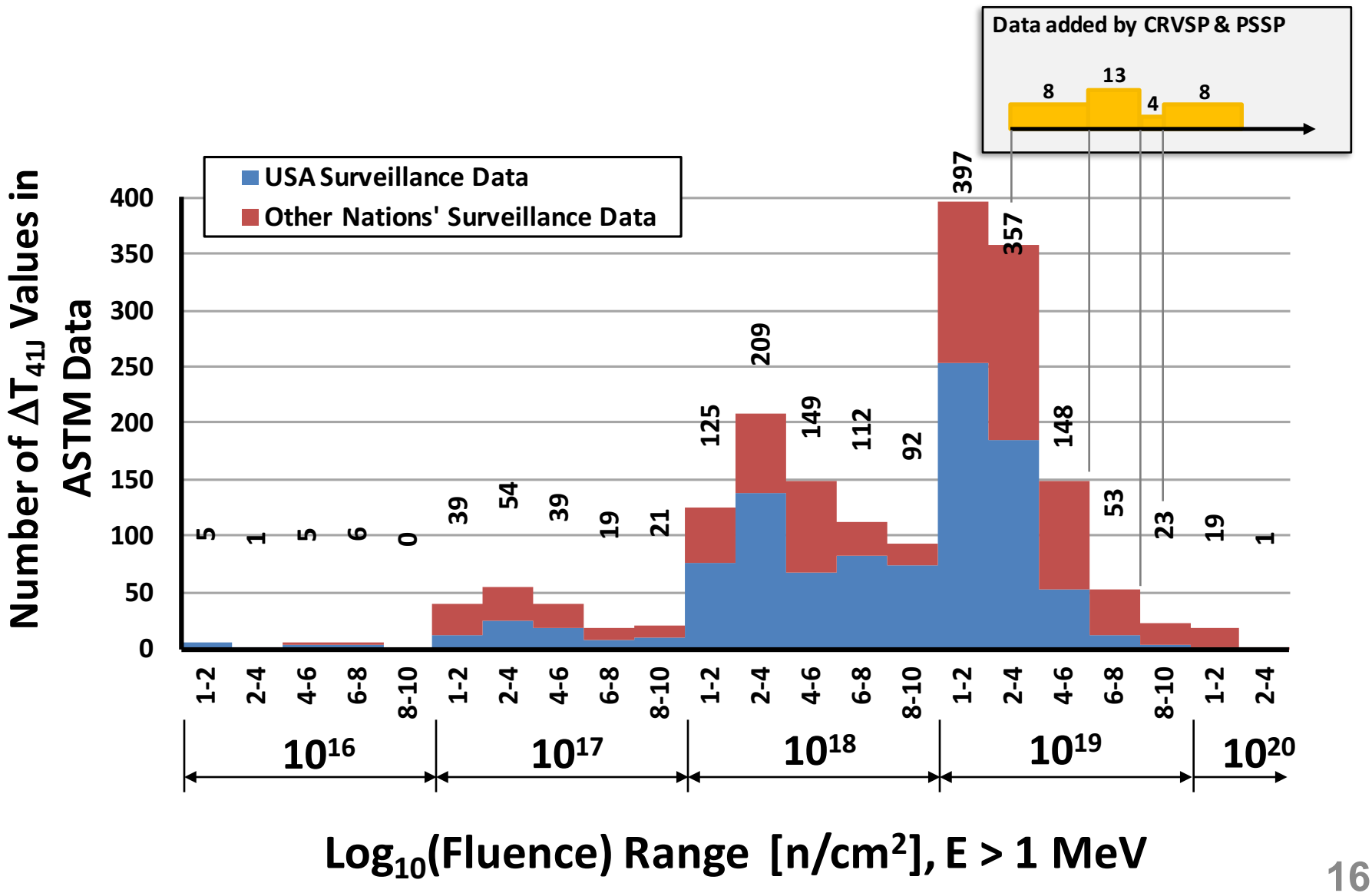

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Protecting People and the Environment

Research Supporting Aging Management of Reactor Vessel Internals for Subsequent License Renewal (SLR)

James Medoff and Seung Min
Office of Nuclear Reactor Regulation
Division of License Renewal

Appajosula Rao, Matthew Hiser,
Robert Tregoning, and Amy Hull
Office of Nuclear Regulatory Research
Division of Engineering

November 17, 2015

Outline

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Regulatory framework for reactor vessel internals (RVIs)
- NRC-sponsored research on RVIs:
  - Irradiation-assisted degradation of stainless steel plate and weld materials
  - Embrittlement of cast austenitic stainless steel

Boiling Water Reactor (BWR) Vessels Internals: Regulatory Framework

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- Regulatory framework for BWR Vessel Internals is based on staff-approved BWR Vessel Internal Project (BWRVIP) documents

- The staff-approved BWRVIP documents provide guidance on inspections, flaw evaluations and repair/replacement activities for BWR vessel internals

- The staff continues to assess operating experience, review and approve new revisions of BWRVIP documents as Topical Reports

Pressurized Water Reactor (PWR) Vessel Internals: Regulatory Framework

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- Regulatory framework for PWR Vessel Internals implements Electric Power Research Institute (EPRI) Materials Reliability Program Inspection and Evaluation Guidelines (MRP-227-A)

- Applies to Westinghouse, Combustion Engineering, and Babcock & Wilcox RVI components

- License renewal application must demonstrate report is applicable and bounding for design of the RVI components at their facilities and address applicable action items

Key Technical Issues

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- Irradiation-assisted degradation (IAD) for stainless steels and high strength bolting at higher fluences associated with extended operation
  - Microstructural changes, including void swelling, may impact mechanical properties, such as fracture toughness and stress corrosion cracking
- Loss of fracture toughness for cast austenitic stainless steels (CASS)
  - Combined effect of thermal and neutron embrittlement
  - Thermal embrittlement from prolonged exposure at operating temperatures

Reference: NUREG/CR 7153, Expanded Materials Degradation Assessment Volume 2: Aging of Core Internals and Piping Systems, Prepared by Expert Panel: P. Andresen, K. Arioka, S. Bruemmer, J. Busby, R. Dyle, P. Ford, K. Gott, A. Hull, R. Staehle. 351pp, Oct. 2014.

Latest Knowledge from NRC Research Supporting SLR on IAD

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- Identified the RVI material chemistry parameters that influence susceptibility to irradiation-assisted stress corrosion cracking (NUREG/CR-6965)
- Established relationship between neutron fluence and neutron embrittlement (NUREG/CR-7027)
- Established the differences between the effects of fast reactor radiation and light water reactor (LWR) irradiation on reactor core internal materials (NUREG/CR-7128)
  - Concluded that fast reactor irradiation is a non-conservative surrogate for void swelling in LWR conditions

Research Activities to Support SLR: Aging Degradation of RVIs

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Research Objectives:

- Assess the effects of IAD and void swelling during the SLR period under representative plant conditions, including irradiation and temperature

- Investigate the combined effect of thermal and irradiation embrittlement of CASS under representative temperature and irradiation conditions

Research Activities to Support SLR: Aging Degradation of RVIs

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Research Projects:

- NRC's IAD research focuses on cooperative research with the EPRI on ex-plant materials from the Zorita reactor in Spain
  - Zorita materials are representative of operating conditions in an LWR with slightly accelerated irradiation
  - Cooperative research allows maximum leveraging of resources for expensive irradiated materials testing
- NRC's CASS research focuses on independent testing of irradiated and thermally aged CASS to understand combined effects

IAD Research Plan

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Fluence (dpa)	Plate	Weld	Heat-Affected Zone
1	Previous research	Welds at Studsvik	
2			
5		Welds at Halden	
8			
10	Zorita Internals Research Project (ZIRP)	Beyond expected fluence at 80 years (Expected fluence at 80 years)	
25			
50			
65	High Fluence Plate		
80			

Testing and characterization includes crack growth rate (CGR), fracture toughness (FT), tensile properties, and microstructure (void swelling).

ZIRP

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- **Scope**:
  - Plate material at 10, 25, 50 displacements per atom (dpa)
- **Approach**:
  - Conduct radiation and temperature analysis
  - Harvest RVI plate materials from Zorita reactor
  - Evaluate mechanical properties and microstructure
- **Evaluation Scope**:
  - Tensile, CGR, FT and crack initiation testing
  - Microstructural characterization/void swelling measurements
- **Collaboration**:
  - EPRI and international regulators and utilities
- **Status**:
  - Harvesting complete; testing is ongoing

B

Welds at Studsvik

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- **Scope**:
  - Weld/heat-affected zone (HAZ) material at ~1 and ~2 dpa
- **Approach**:
  - Harvested weld and HAZ materials from core barrel of Zorita reactor
  - Testing as-harvested materials without further irradiation
- **Testing Scope**:
  - Tensile, CGR, and FT testing
  - Characterization of fracture surfaces
- **Collaboration**:
  - EPRI
- **Status**:
  - Harvesting complete; specimen preparation ongoing

Welds at Halden

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- **Scope:**
  - Weld/HAZ material at 2, 5, 8 dpa
- **Approach:**
  - Further irradiate and test Zorita weld/HAZ materials
- **Testing Scope:**
  - Limited CGR tests as-received at 2 dpa
  - Further irradiation in Halden reactor up to 5 and 8 dpa
    - ~1 dpa per year in LWR neutron energy spectrum
  - CGR, FT, and tensile testing at 5 and 8 dpa
- **Collaboration:**
  - Irradiation and testing perform under Halden Reactor Project Joint Program
- **Status:**
  - Irradiation and 2 dpa as-received testing begins in 2016

High Fluence Plate

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- **Scope:**
  - Plate material at 65, 80 dpa
- **Objective:**
  - Address high-fluence IAD through further irradiation of Zorita plate materials followed by material evaluation
- **Approach:**
  - Irradiation source and testing conditions not finalized
- **Collaboration:**
  - EPRI; also interested participation by DOE and others
- **Status:**
  - NRC and EPRI staff co-authored a paper at the recent Environmental Degradation conference
    - Evaluated potential approaches to assess effects of high fluence IAD during SLR
  - Evaluating options to initiate program

B

Latest Knowledge from NRC Research Supporting SLR on CASS

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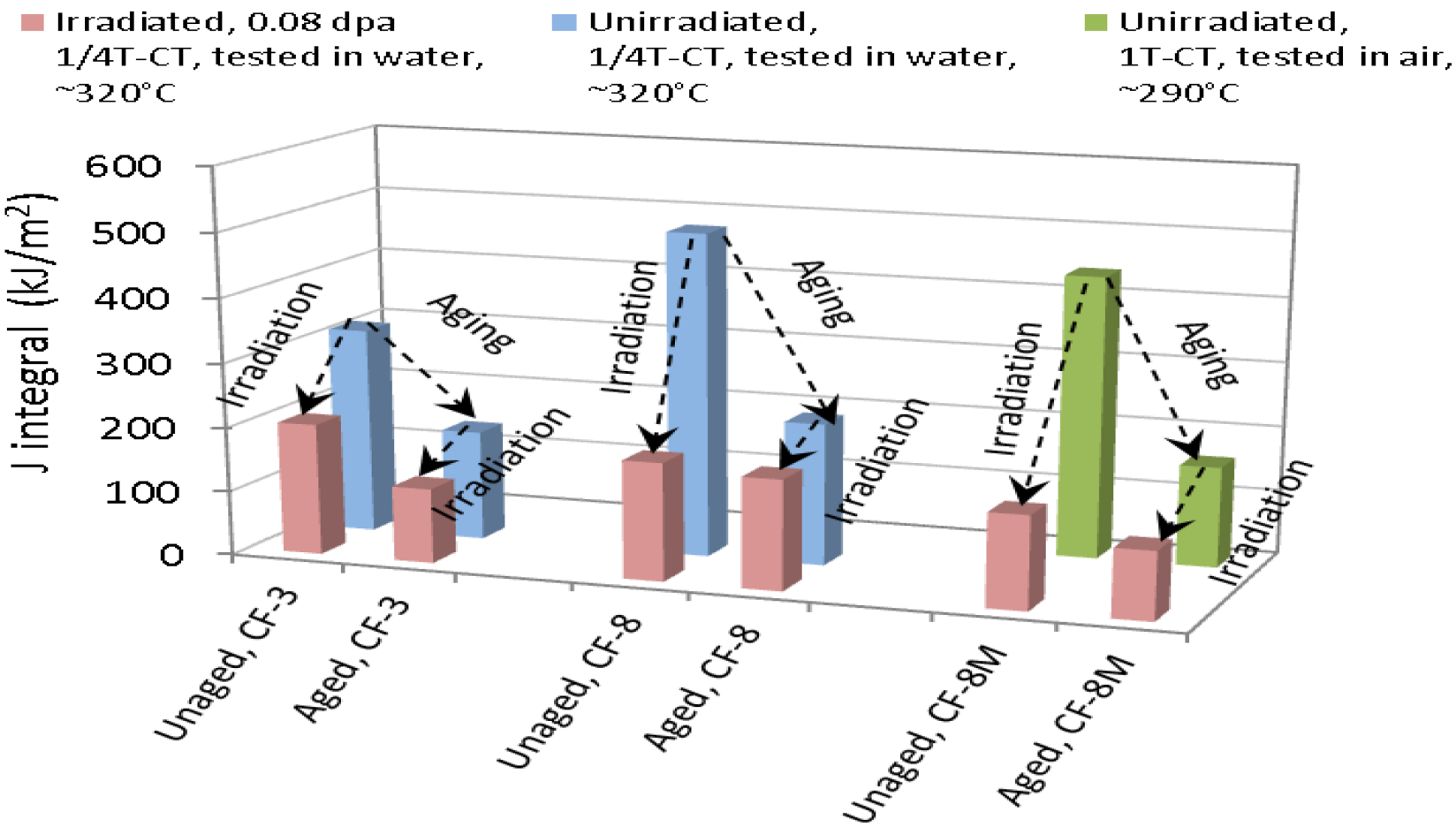
- Evaluated the combined effects of thermal and neutron irradiation on the fracture toughness of CASS materials at very low fluence (NUREG/CR-7184)
- Proposed bounding values of delta ferrite for thermal and neutron irradiation having acceptable fracture toughness; values are applicable for developing CASS screening criteria (NUREG/CR-7184 and NUREG/CR-7185)
- Developed updated methodologies to calculate fracture toughness and established mechanism of thermal embrittlement of CASS, accepted in proposed modification to ASME Code Case N-838 (NUREG/CR-4513, Rev. 1)

CASS Fracture Toughness



Source: NUREG/CR-7184, *Crack Growth Rate and Fracture Toughness Tests on Irradiated Cast Stainless Steels*, July 2015

CASS Research Plan

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Project	Scope	Status
CASS and Welds Gap Analysis	Literature review to update guidance and address neutron and thermal effects	• Expected completion in June 2016 • NUREG/CR-4513, Rev. 2 • Technical letter report on welds
Irradiation of CASS to 3 dpa and Testing	Testing of irradiated CASS materials to observe effects of thermal and neutron embrittlement	• Early stages: expected completion in 2017

- All work being conducted at Argonne National Laboratory

RVI Research Timeline

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Area	Project	2013	2014	2015	2016	2017	2018	2019→
IAD	ZIRP							
	Welds at Studsvik							
	Welds at Halden							
	High Fluence Plate							
CASS	Validation of Thermal Aging (NUREG/CR-7184)							
	0.08 dpa CASS (NUREG/CR-7184)							
	CASS and Welds Gap Analysis							
	3 dpa CASS							

First Subsequent License Renewal Application

Summary

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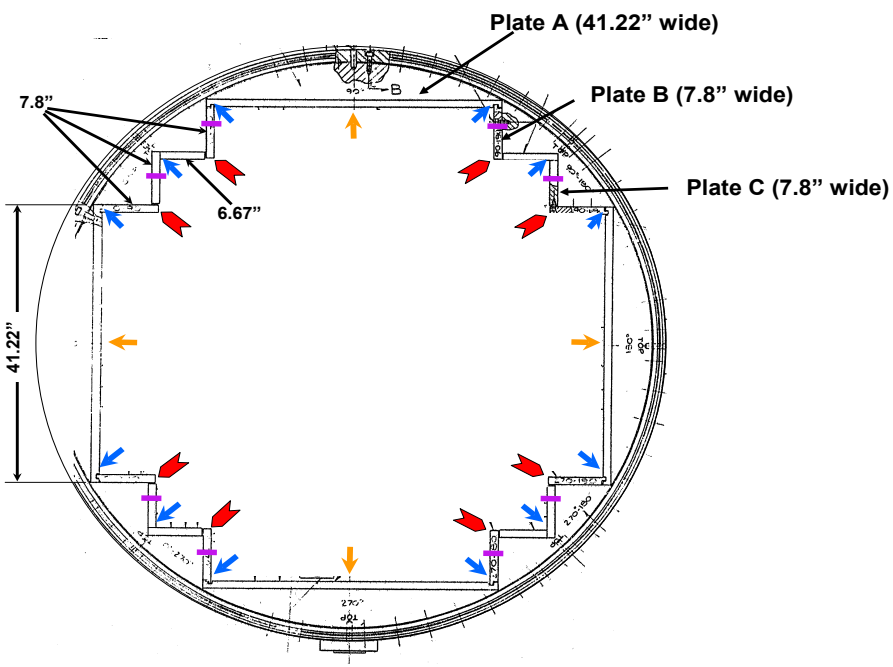
- Described regulatory framework for RVIs and identified important technical issues
- Discussed ongoing research to address degradation mechanisms for RVI during SLR period
  - Effects of IAD on stainless steel plate and weld materials at higher fluences
  - Loss of fracture toughness in CASS materials
- Continue staff collaboration to address and disposition any emerging issues identified through this research

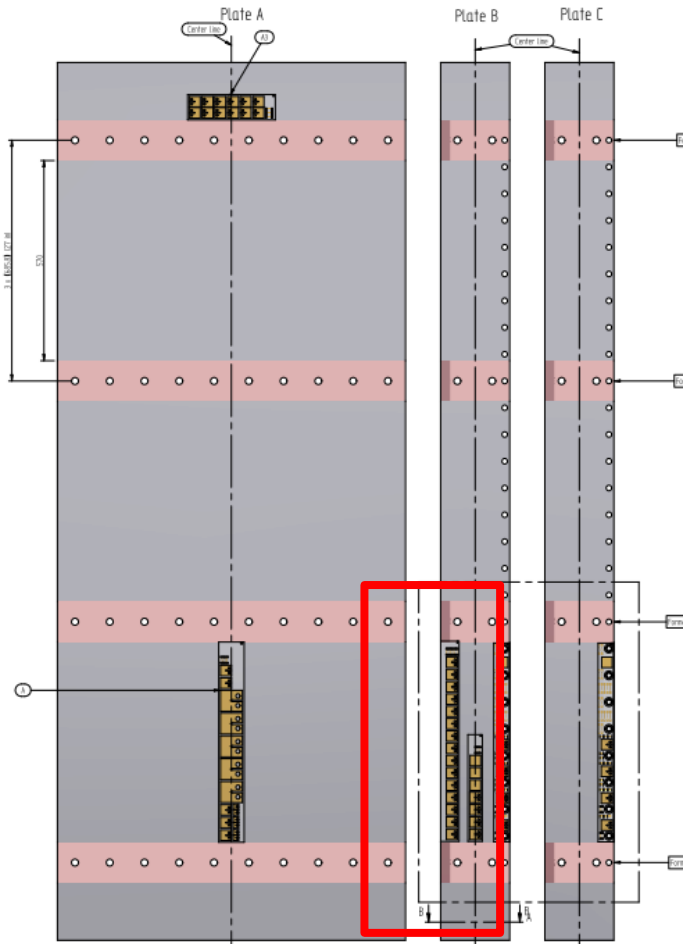
Backup Slides

Zorita Background

- Westinghouse single loop PWR (160 MWe)
  - Operated for ≈26 effective full power years

- Baffle plate material between 3rd and 4th formers
  - Type 304 Stainless Steel
  - Doses ranging from a few dpa to ≈50 dpa





Figures from EPRI presentation to IAEA CRP, Jan. 2015

Approaches to Address High Fluence IAD

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- Five potential approaches were considered to evaluate the effects of IAD to support extended reactor operation beyond 60 years:
  - Existing data and ongoing research programs,
  - Additional data from ion irradiation,
  - Additional data from fast reactor irradiation,
  - Additional data from near-LWR spectrum irradiation, and
  - Additional data from near-LWR spectrum further irradiation of harvested ex-plant materials

High Fluence Research Approaches Pros and Cons

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Approach	Benefits	Drawbacks
Existing Data and Programs	• No cost or schedule for new irradiation program	• Limited data beyond 50 dpa in LWR-representative conditions increases uncertainty at higher fluences
Ion Irradiation with Fresh Material	• Provides information on microstructural changes to relevant dpa range • Lower cost and shorter schedule than other approaches	• Limited damage depth would limit mechanical testing, particularly crack growth rate and fracture toughness. • Significantly different radiation flux, energy spectrum, irradiation type, and irradiation temperature.
Fast Reactor Irradiation with Fresh Material	• Provides information on microstructural changes and mechanical properties to relevant dpa range	• Higher cost and longer schedule • Higher damage production rate and lower gas production rate in fast neutron spectrum may underrepresent extent and significance of void swelling compared to LWR spectrum conditions
LWR Spectrum Irradiation with Fresh Material	• Provides information on microstructural changes and mechanical properties to relevant dpa range • Approach would likely be representative of LWR conditions important for IAD	• Extremely high cost and long schedule make it difficult for this approach to be practical
LWR Spectrum Irradiation with Ex-Plant Material	• Provides information on microstructural changes and mechanical properties to relevant dpa range • All irradiation exposure performed in LWR or near-LWR spectrum conditions	• Higher cost and longer schedule • Some uncertainties exist associated with further irradiation of ex-plant materials

Concrete – Aging Management Research

Joint EPRI-LTO DOE-LWRS Presentation to ACRS

November 17, 2015

Sherry Bernhoft

EPRI LTO Program Manager

Keith Leonard

LWRS Program Materials Aging and Degradation Pathway Lead

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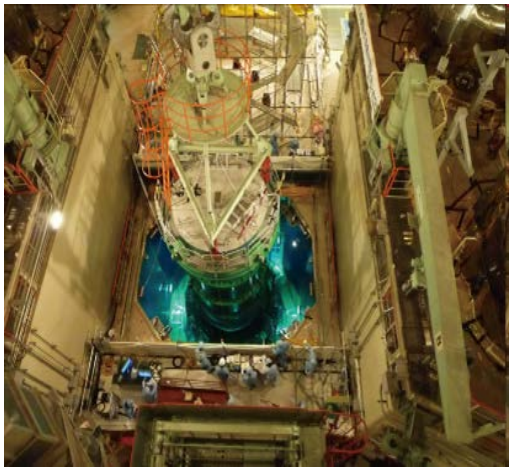
Containment and Concrete Structures

- Monitoring and aging management of concrete structures due to radiation and environmental exposure

- EPRI Technical Reports:
  - ✓ Nuclear Concrete Structures Aging Reference Manual (1023035)
  - ✓ Nondestructive Evaluation Inspection of Concrete Structures (1025627)
  - ✓ Literature Survey of Effects of Radiation on Concrete (1025584)
  - ✓ Expected Conditions of Reactor Cavity Concrete at 80 Years of Radiation Exposure (3002002676)
  - ✓ Boric Acid Attack of Concrete and Rebar in PWR Spent Fuel Pools (1025166)
  - ✓ Nonlinear Ultrasound to Evaluate Integrity of Aged Concrete (1026501)
  - ✓ Pilot Plant Augmented Containment Inspection Results (3002002335)
  - ✓ Literature Survey of Concrete Creep (3002003220)
  - ✓ ASR Risk Screening (3002005389)



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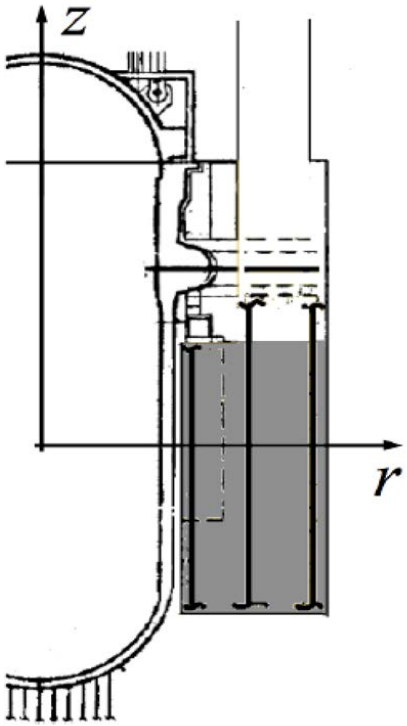
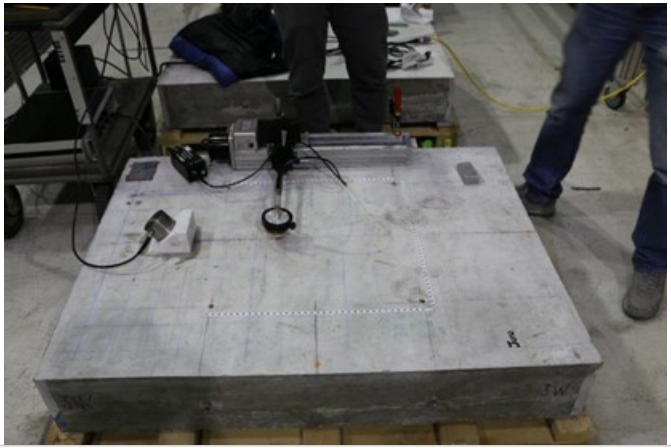
R&D Topics

- Effects of Irradiation on Concrete
- Concrete Degradation due to Alkali-Silica Reactions
- Boric Acid Damage of Spent Fuel Pools
- Creep Impact on Post-tensioned Containment





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Effects of Irradiation on Concrete

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Irradiation

Problem Statement:

Fast neutrons from the reactor core exit the RPV and interact with the concrete in the reactor cavity.

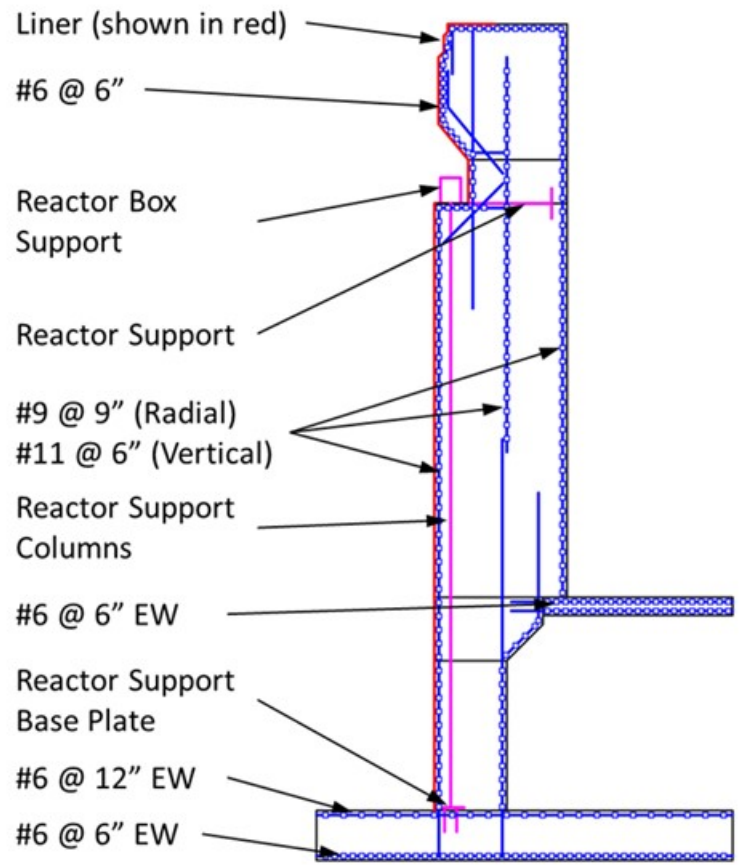
Issue:

Understand the impact on the structural stability of the reactor cavity and vessel supports

Duration: 2012 – 2018

TR 3002002676: Expected Condition of Reactor Cavity Concrete After 80 Years of Radiation Exposure (published 2014).

Technical Report: Structural Effects of Chronic Radiation Exposure in RPV Support Structure.



Schematic of RPV support structure

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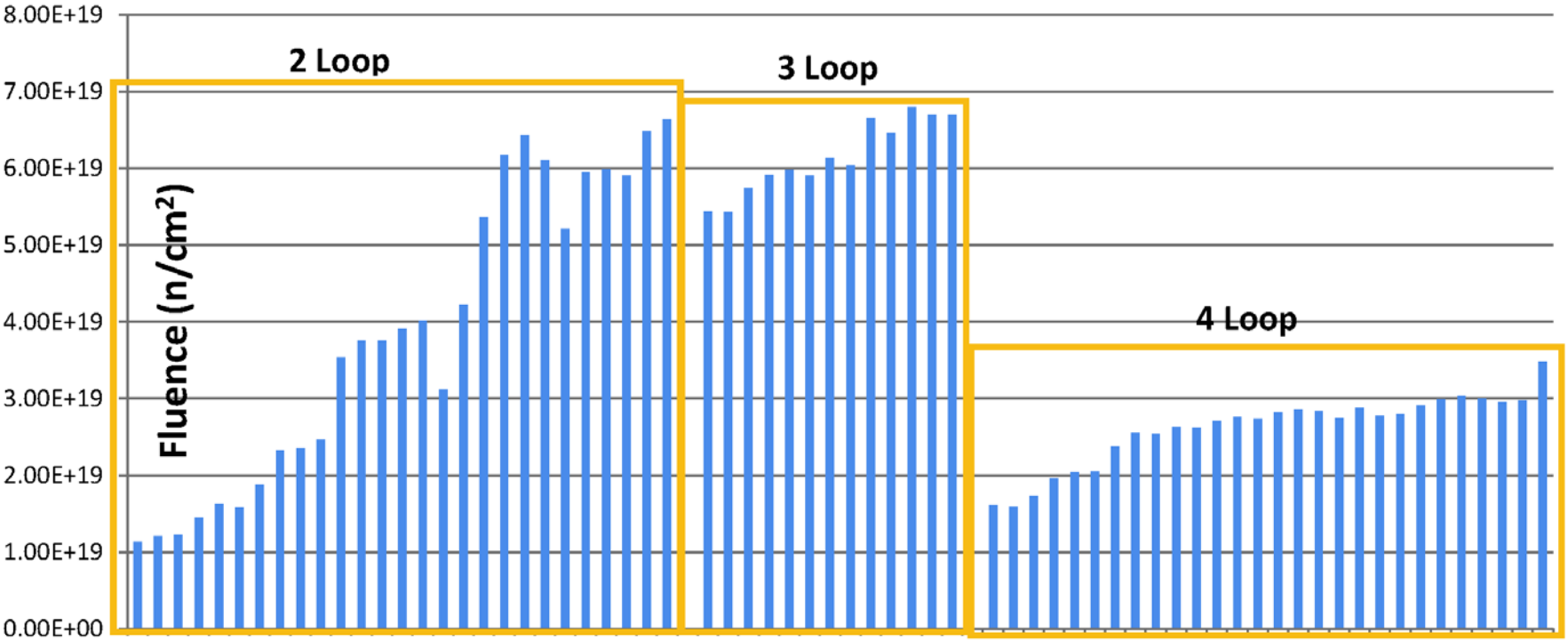
Irradiation Damage of Concrete – Ongoing Research

- EPRI and LWRS have partnered to study the effects of radiation damage on reactor cavity concrete
  - LWRS Tasks at ORNL
    - Fundamentals of radiation damage
    - Modeling of fluence through the biological shield (complete)
    - Neutron and ion irradiation of mineral analogues to characterize swelling
    - Structural significance of radiation damage including swelling due to irradiation
  - EPRI Tasks
    - Estimation of bounding fluence (complete)
    - Structural significance of radiation damage including swelling and changing mechanical properties due to irradiation

LWRS U.S. DEPARTMENT OF ENERGY Nuclear Energy OAK RIDGE National Laboratory EPRI ELECTRIC POWER RESEARCH INSTITUTE

PWR 1T Fluence for 80 y Operation (E > 0.1 MeV)

- Neutron fluence in operating US PWRs at 80 years operation (data extrapolated from ex-vessel fluence reported in ADAMS).
- BWR fluence is ~ an order of magnitude lower than PWR.



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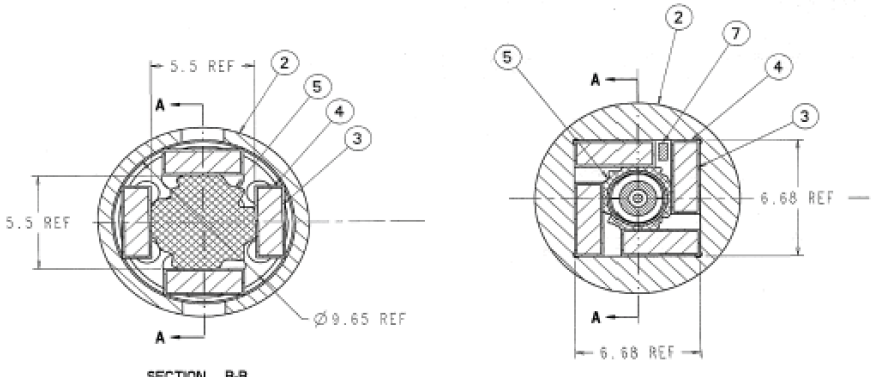
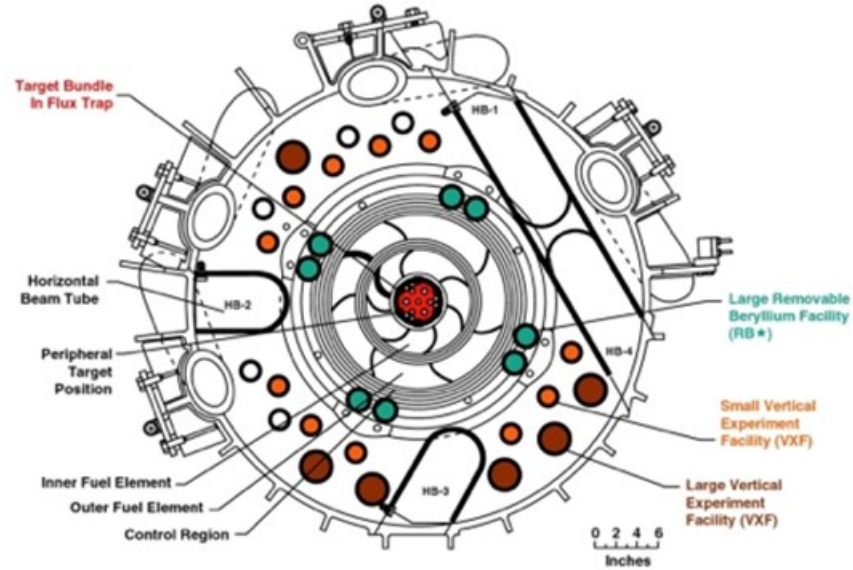
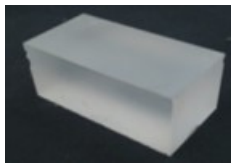
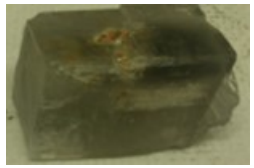
Neutron Irradiation of Monocrystals at HFIR (Test Reactor)

- Neutron fluences (E > 0.1 MeV): 0.5, 4 and 20 x 10^{19} n.cm^{-2}
- Two different irradiation capsule designs to test effects of irradiation temperature:
  - Sealed ~150 ^{o}C
  - Perforated ~50 ^{o}C (hydraulic port fluid)
- Post-irradiation examination has initiated to gauge degree of Radiation Induced Volumetric Expansion (RIVE) susceptibility
  - XRD, optical microscopy, dimensional / density measurements, hardness, modulus.
  - Quartz, SiO_2, exhibits directional (crystal structure) dependent swelling as well as possible temperature dependence.
  - Calcite, $Ca(CO_3)_2$, limited swelling.
  - Dolomite, $CaMg(CO_3)_2$, limited swelling.

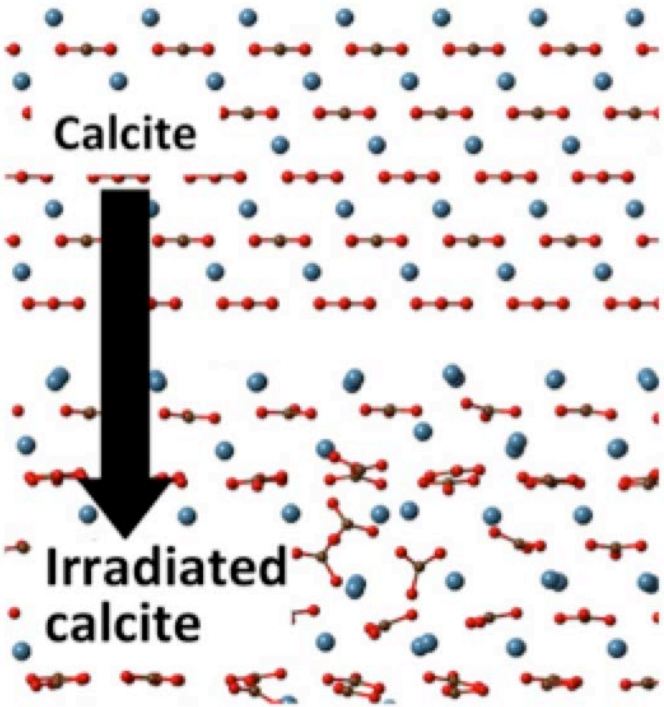


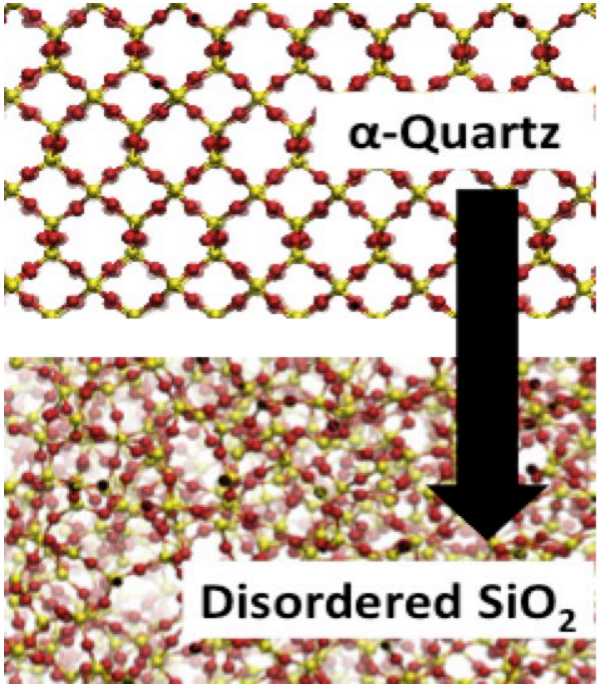






Molecular Dynamics Simulation of Irradiated Minerals



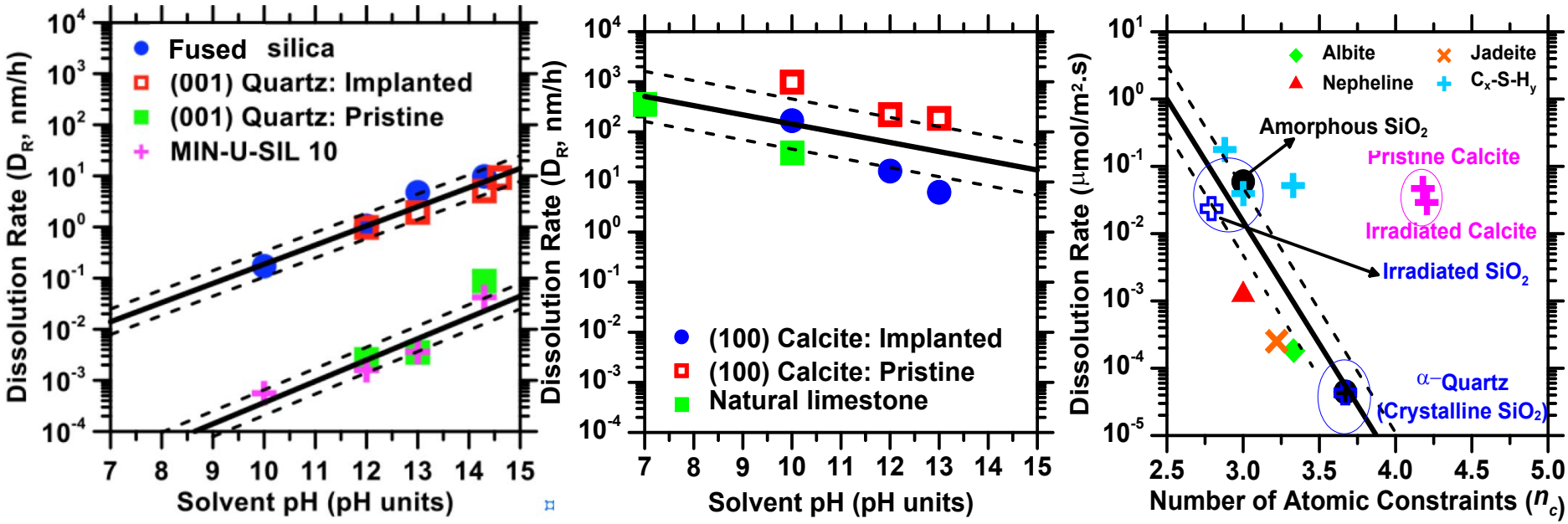


- Carbonate groups experiencing random rotations and distortion in the irradiated structure.
- Blue: Ca atoms, Red: O atoms, Brown: C atoms

- Progressive disordering of -quartz to disordered silica under energetic
- particle exposure. Red: O atoms, Yellow: Si atoms

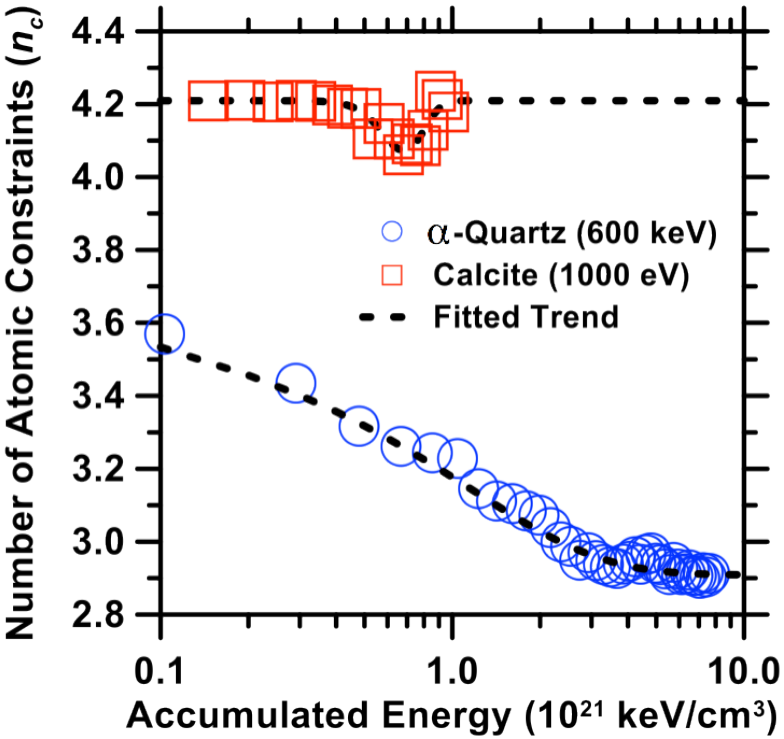
The covalent nature of silica bonds (and thus, siliceous aggregate) explains its susceptibility to neutron irradiation, as opposed to the ionic nature of calcite.

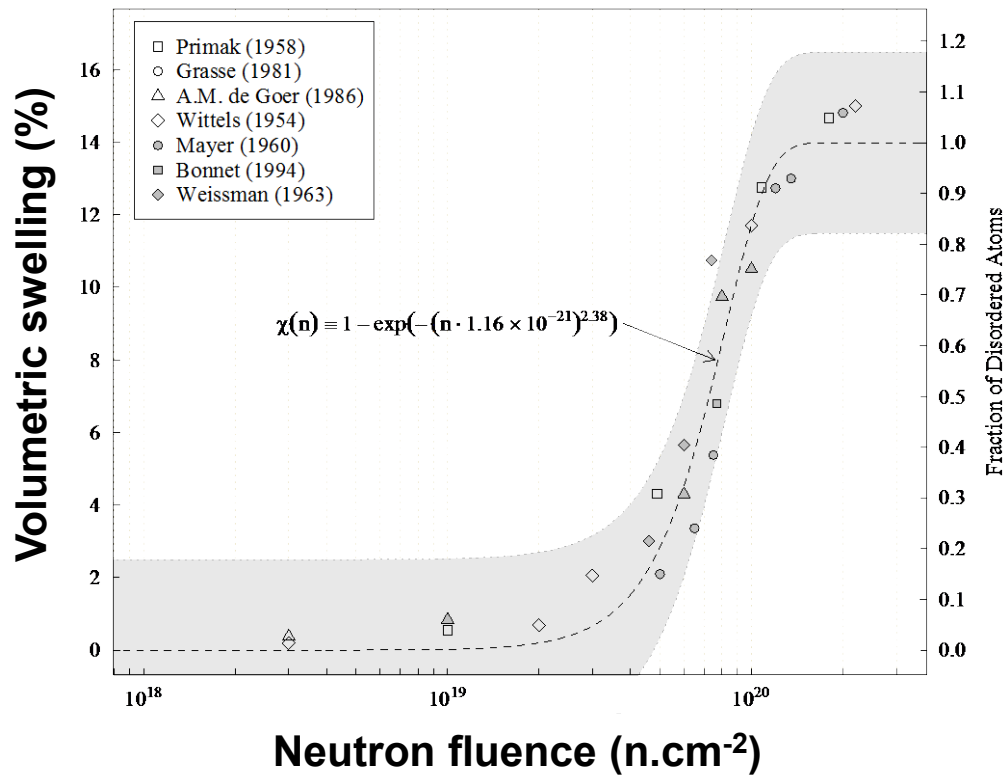
Chemical Consequences of Neutron-Induced Amorphization



- Increase of the dissolution rate of irradiated silica in alkali solution
- Possibility of Irradiated-Assisted Alkali-silica Reaction

Mechanical Consequences of Neutron-Induced Amorphization





- Loss of atomic constraint, i.e., loss of rigidity and Young modulus
- Radiation-Induced Volumetric Expansion (RIVE)

Effects of Aggregate Amorphization on Concrete

RIVE of silicate minerals

↓

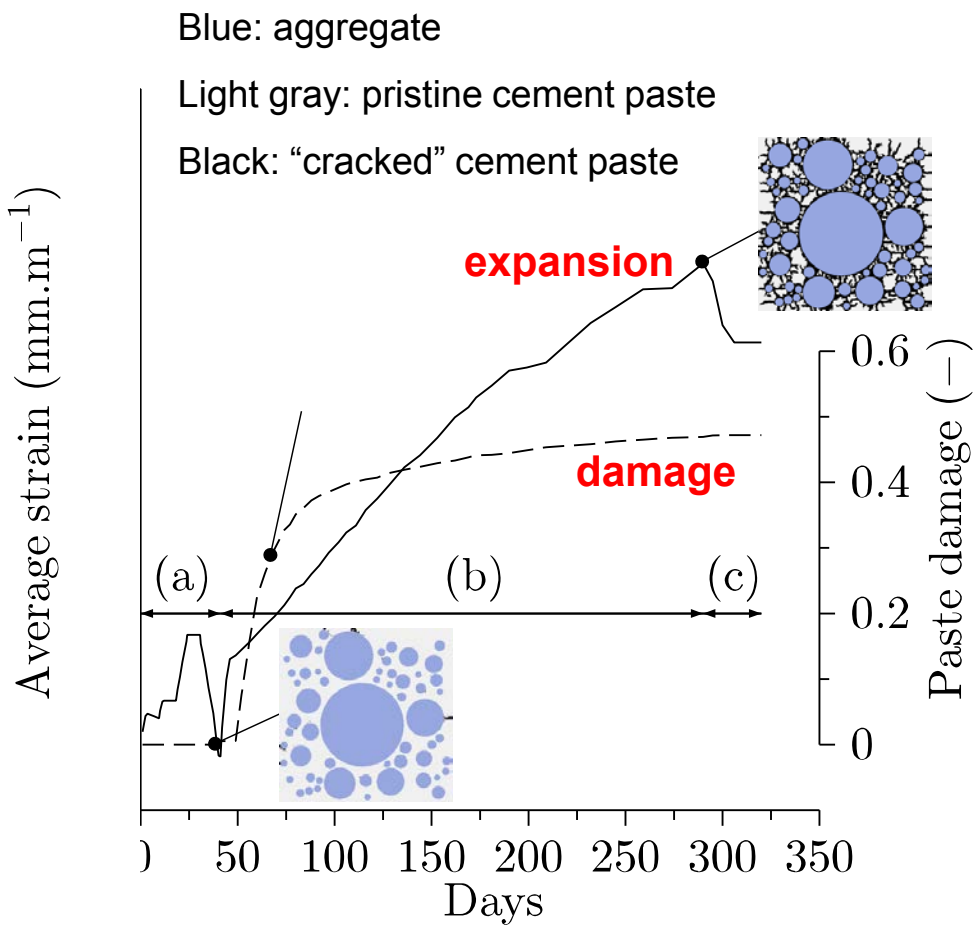
Expansion of aggregate

↓

Damage/cracking of the cement paste

↓

Loss of mechanical properties

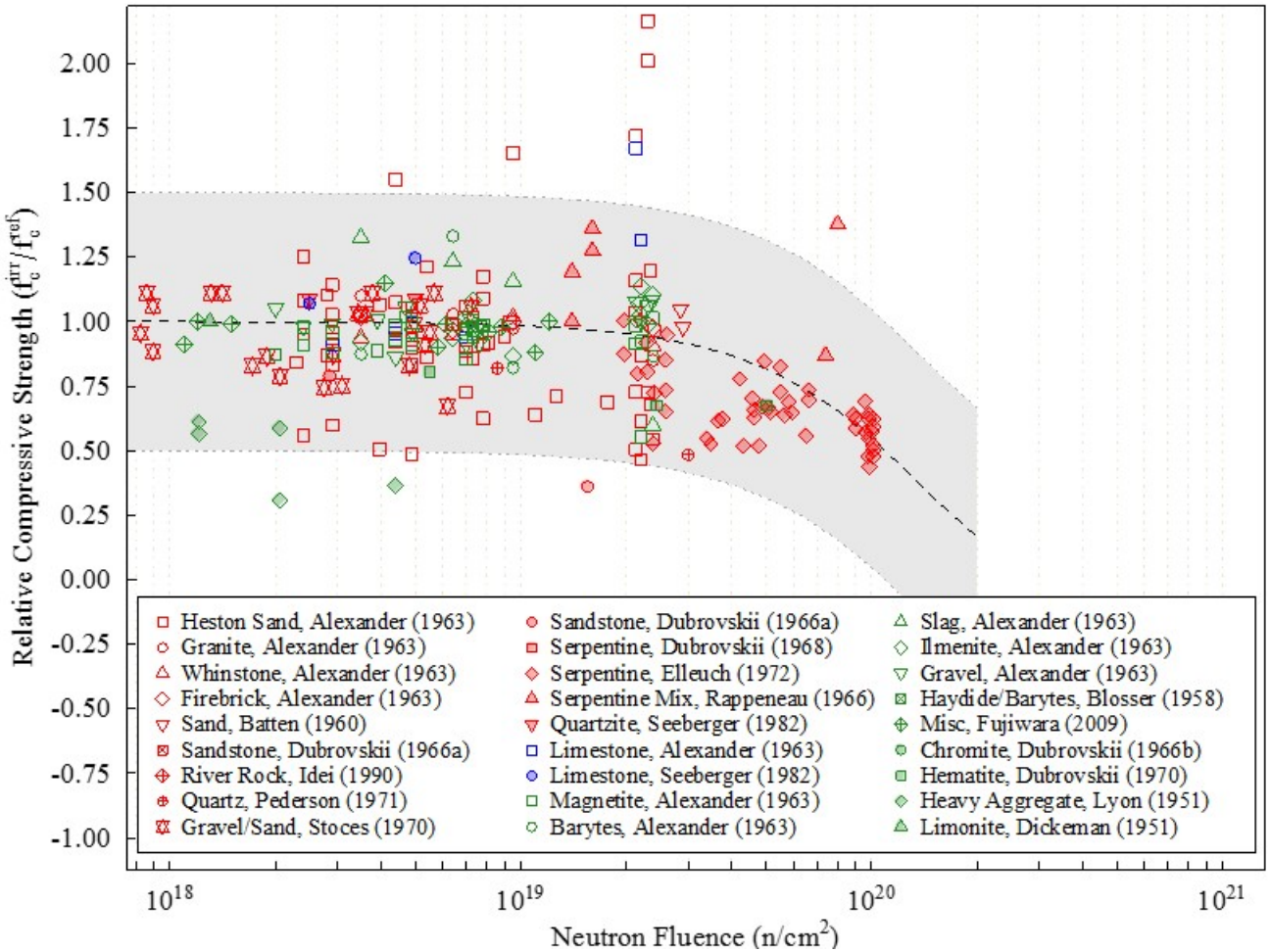


Meso-scale finite element simulation with Automated Mechanics Integrated Environment (AMIE)

Effect of Irradiation on Concrete Properties

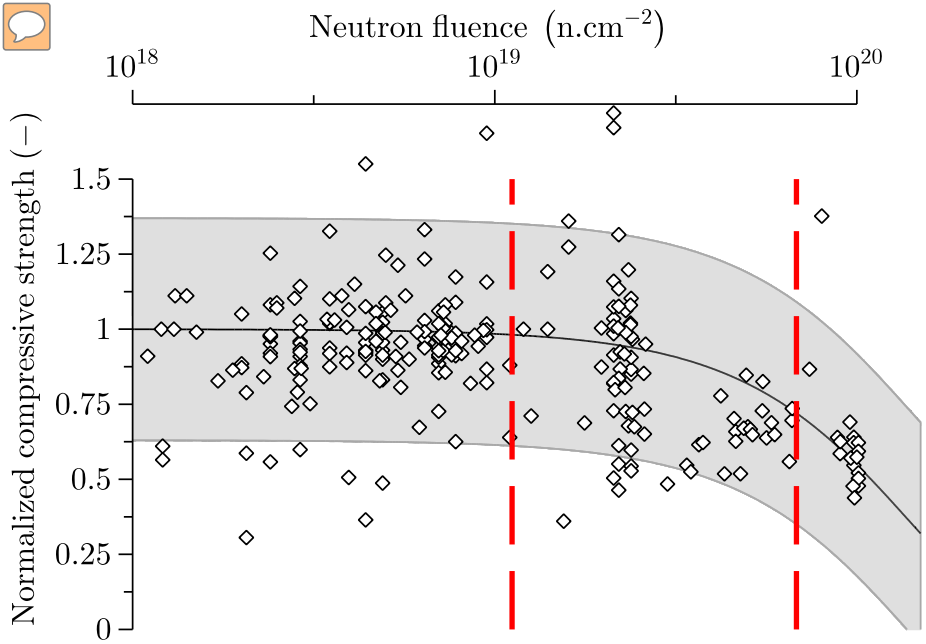
Literature review updates Hilsdorf's curve (1978):

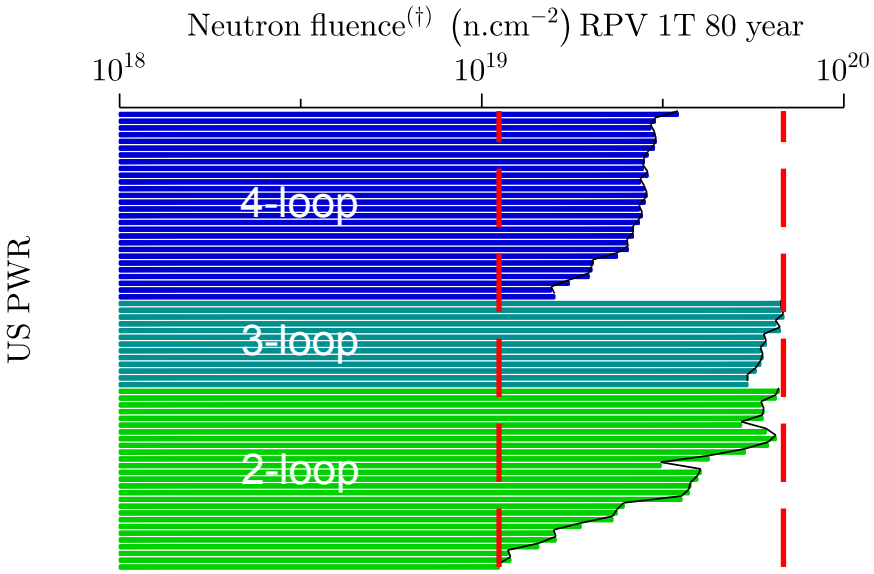
- 307 compression strength data
- 62 tensile strength data
- 138 elastic modulus data
- 114 linear expansion data



Red: silicate
Blue: limestone
Green: miscellaneous and heavy aggregate

Radiation Exposure of Concrete Biological Shield





PWR's:

- Inner diameter fluence 10^{19} to 7.0×10^{19} n/cm^2 (E > 0.1 MeV)
- For 80 year PWR operation (92% capacity)[1], a 4-loop PWR < 4.0 x 10^{19} n/cm^2.

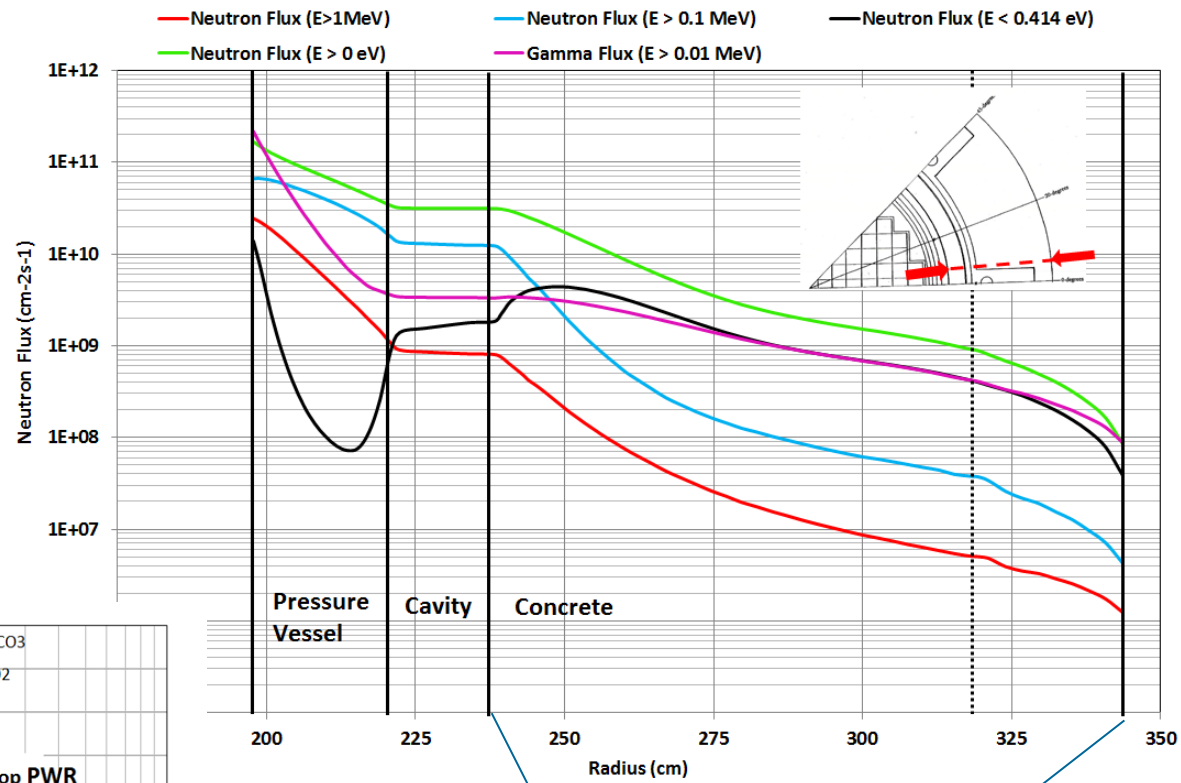
BWR's:

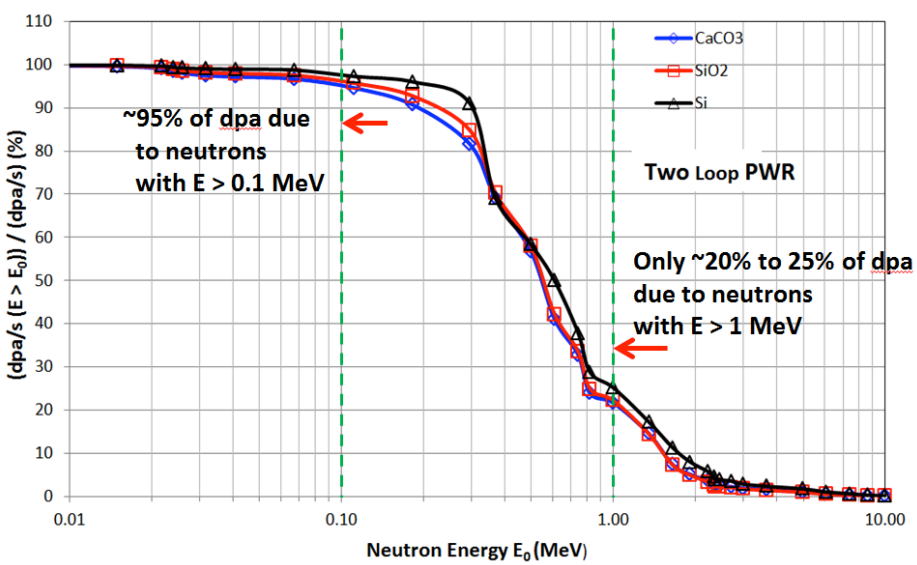
- Max. fluence ~ 10^{19} n/cm^2 at 80 years of operation.

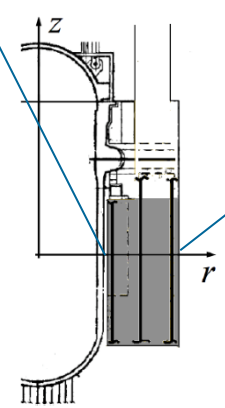
Flux attenuation [2] : one order of magnitude per 12 to 15 cm (4 to 5 in.).

Neutron Attenuation and Energy Cut-Off

- Strong fluence attenuation in the concrete biological shield (~1 order over 12-15 cm)
- 95% of dpa occurs for E> 0.1 MeV



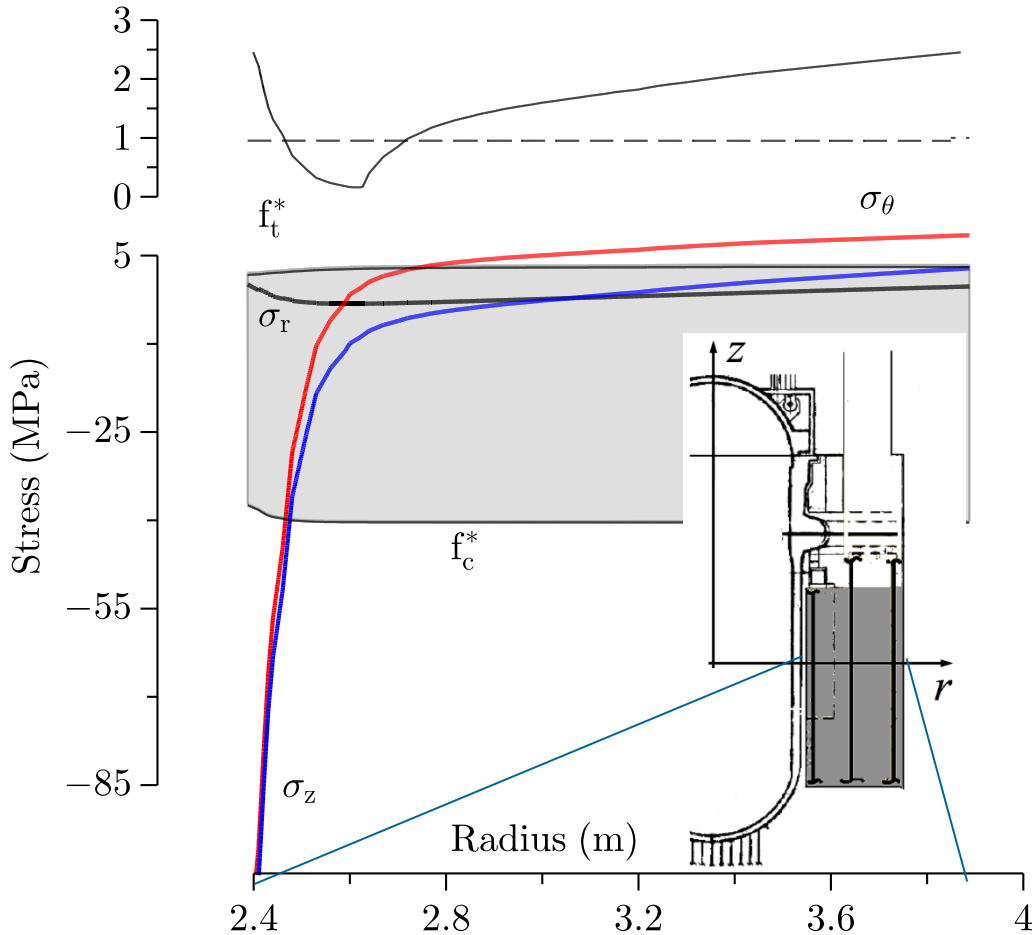




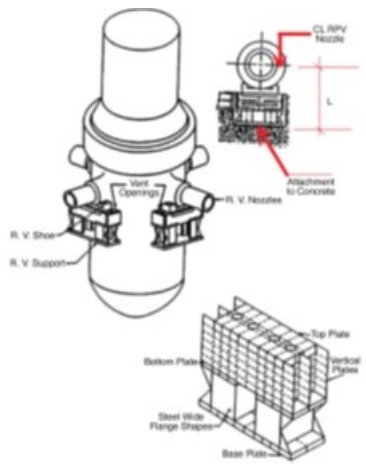
Structural Effects of Irradiation on Concrete Biological Shield (preliminary)

Radiation Induced Volumetric Expansion (RIVE) creating elastic stresses:

- Biaxial compression near the ID
- Circumferential tension toward the back
- Constraint effects need to be incorporated into model.

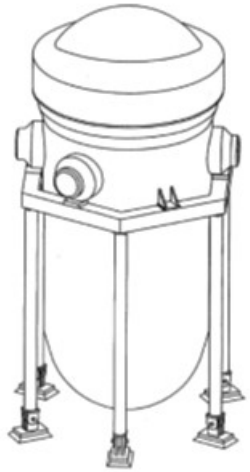


Engineering Structural Evaluation: Supports designs – PWR type 1/2/3/4



Type 1 – Shoe on Bio Shield

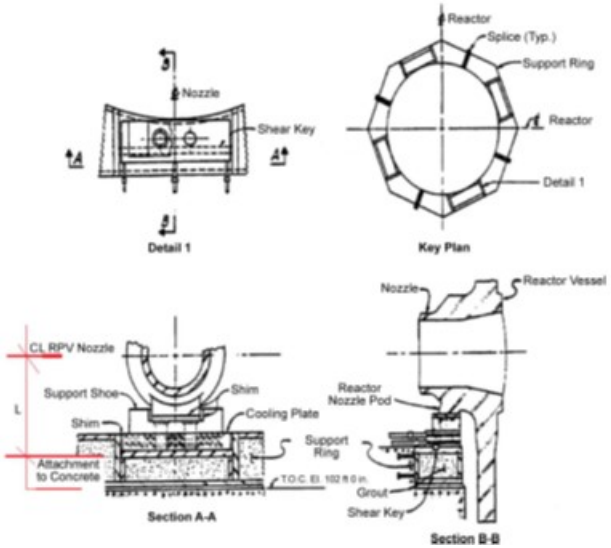
Support of interest



Type 2 – Metal Frame

Not in Scope

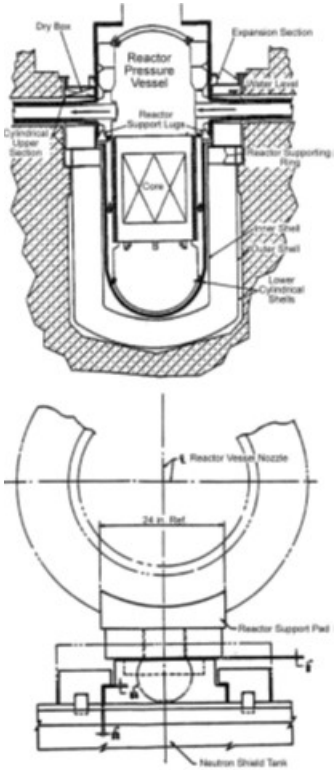
Bio shield not structural



Type 3 – Shoe on Bio Shield

Not in Scope

Used only on W 4-loop units



Type 4 – Shoe on Shield Tank

Not in scope

Neutron shield tank – low fluence

- 11 units are in scope for this study (2 or 3-loop, Type 1 supports)

- Documentation of the choice of Type 1 will be included in the 2016 report.

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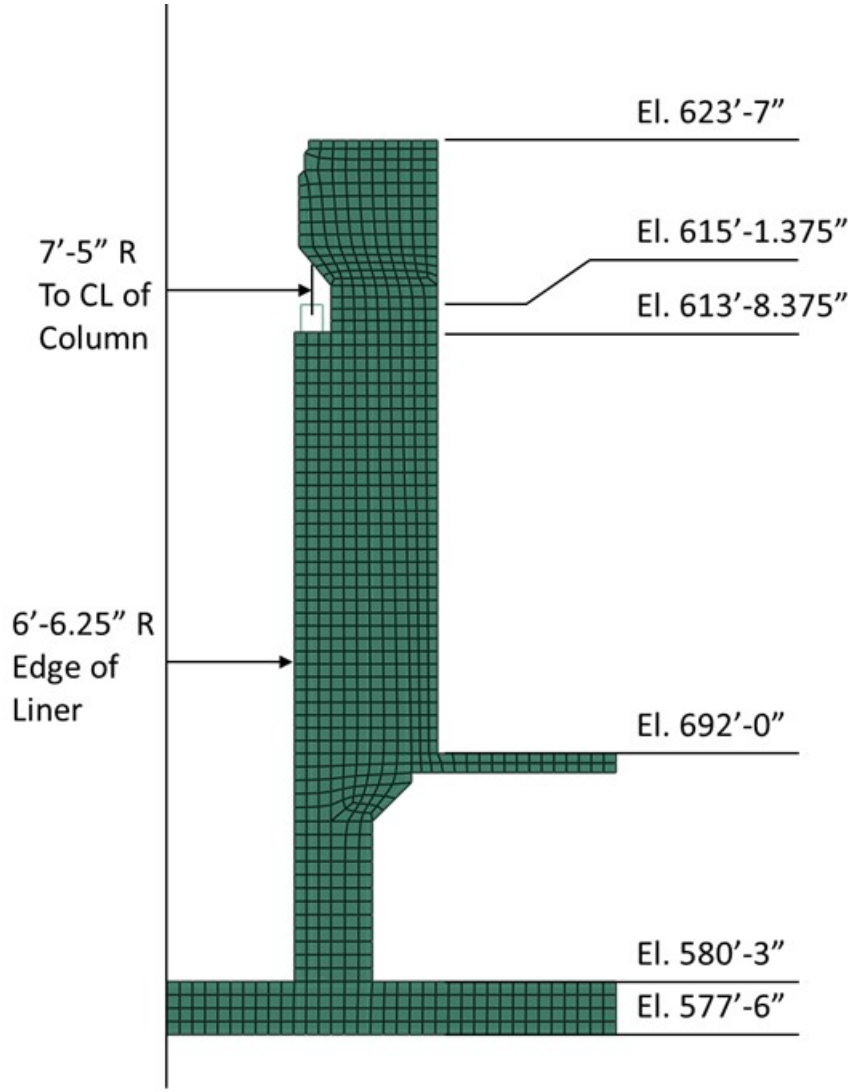
Engineering Structural Evaluation

- EPRI is working with structural vendor to perform a detailed analysis of the type 1 support design
  - Detailed drawings from a 2-loop and 3-loop PWR
  - Modeling the reduction in margin due to neutron irradiation:
    - Swelling of aggregates/concrete
    - Change in mechanical properties
  - Results of the analysis will be published as an EPRI Technical Report in Q4, 2016.



Finite element modeling mesh of Type 1 RPV support

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International Committee on Irradiated Concrete (ICIC)

- International Irradiated Concrete Information Exchange Framework Meetings (Barcelona and Helsinki). First General Meeting, Knoxville, TN, Nov 2-5, 2015
- Provide a framework for exchanging information (charter) and aligning research Chair: T. Rosseel (ORNL, USA), Vice-Chair: I. Maruyama (Nagoya U., Japan), Secretary: M. Ferreira (VTT, Finland)
- Technical Areas:
  - Structural Performance and Mechanistic Understanding of the Effects of Radiation on Concrete (Y. Le Pape ORNL, USA)
  - Harvesting and Characterization of Service Irradiated Concrete (M. Ordonez, ENRESA, Spain)
  - Accelerated Irradiation Studies of Concrete & Components (M. Koleska, RC-Rez, Czech Repub.)
  - Characterization of Irradiated Concrete (C. Andrade, CSIC, Spain)
- Members from the US (DOE, EPRI, NRC), Europe and Japan

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Concrete Degradation due to Alkali-Silica Reactions

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Integrated Efforts for ASR Studies

Mechanisms Understanding

- Prior existing knowledge 1940-2012 Transportation, Hydro-dams. Power Distribution
- Possibility of IAASR In SB/RVP support 2013-2015 See irradiation roadmap
- NIST Transport of Ions in ASR-Affected Structures 2014-2016

Materials Characterization

- ASR Risk Screening Early detection 2014-15
- Characterization of Expansion/Damage of Concrete (UTK) 2016-2017
- RILEM TC ISR WG1: Residual Expansion 2015-2017
- NIST Estimation the Degree of Reaction in ASR-Affected Concrete/Expansion 2014-2016
- NIST Prognosis and prediction of Expansion 2014-2016
- NIST Assessment In-Situ Mechanical Properties Properties 2014-2018

NDE/Monitoring

- NEUP ASR NLUT NDE 2013-2015
- ASR/Freeze-Thaw Mock-Ups for NDE 2013-15
- Enhanced/new NDE 2017-2018
- RILEM WG3: NDE & monitoring 2015-2017
- Guideline on Monitoring ASR-Affected Nuclear Structures 2015-2016

Engineering Validation

- Beam Shear/Anchorage Testing at U. of Texas 2013-2015
- U. Co. Confined ASR-Beam Testing 2015-2017
- Effect of Confinement on ASR Damage and Shear Capacity 2014-2016
- RILEM TC ISR WG2: Numerical Modeling Guidelines 2015-2017
- ASR Development of Confined concrete block at U. of Tennessee 2013-2015
- In-plane Shear Testing at U. Toronto 2014-2016
- Confined ASR block (U. Sherbrooke for Hydro Quebec)

Structural Significance

- Aging Management Toolbox 2013-18
- Modeling of Moisture Transport 2014-2015
- Computational Simulation of ASR-Affected Structures 2014-2015
- NIST Seismic Response Characteristics (In-Plane Shear) 2016-2017
- NIST Assessing Bond/Anchorage in ASR-Affected Structures 2014-2016
- RILEM TC ISR WG4: Large-Scale Testing 2015-2017
- Survey/Lessons Learned on ASR Modeling 2015-2016
- Structural System Modeling and Reliability Analysis

R&D gaps
Prior Research
DOE Actions
EPRI Actions
Third Actions
NRC Actions

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Risk screening for ASR

- Concrete that previously was considered as non-reactive may be classified differently today due to new improved testing methods.
- For those structures that have the potential to develop this degradation, a set of tools for early detection of this pathology will be provided.

Screen 1: Assess Plant Susceptibility to ASR

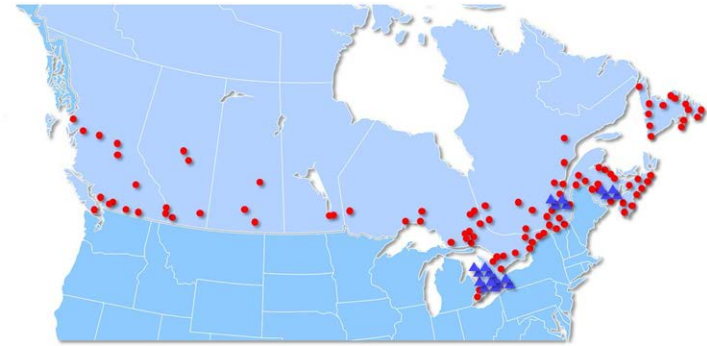
• Review plant information
  ○ Review plant records from original construction (aggregate procurement information, aggregate reactivity test results, concrete mix design information, and historical petrographic examination results).
  ○ Review plant history of misalignment issues.
  ○ Review plant history of water ingress.
• Review studies on ASR susceptibility.
  ○ Appendix A provides information on the reliability of aggregate reactivity testing methods.
  ○ Appendix B outlines how to gain insights on aggregate susceptibility for a given geographic region.

Screen 2: Examine Potential Symptoms of Concrete Degradation due to ASR

• Perform a high-level walk-around of accessible concrete structures to look for conditions that may indicate ASR. Specific indicators to look for include:
  ○ Pattern cracking with darkened crack edges on exposed concrete
  ○ Misalignment of walls, doorways, piping and component supports, piping penetrations, etc.
  ○ Water ingress
• Review reinforcement configuration in areas of interest.
• Perform a detailed examination of areas where ASR is suspected. Clean concrete surfaces enable close examination (e.g., remove paint).

Screen 3: Confirm the Presence of ASR
• Take cores from areas where ASR is suspected and perform petrographic examinations to confirm the presence of ASR. • Perform laboratory analysis of ASR gel. • Use uranyl acetate fluorescence method to detect ASR gel on concrete cores. • Evaluate the use of NDE techniques such as Raman spectroscopy to identify ASR gel in the field.

www.fhwa.gov



www.nrc-cnrc.gc.ca

EPRI Technical Report: 3002005389

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Inspection of ASR affected structures

NEUP ASR NLUT NDE

2013-2015

ASR/Freeze-Thaw Mock-Ups for NDE

2013-15

As of today there are no NDE tests that can reliably
a) identify ASR; b) determine the level of ASR in a structure.

- EPRI has developed 6 large specimens with varying degrees of ASR to be tested by Universities and vendors to evaluate the reliability of NDE techniques

- New inspection techniques are normally developed in small laboratory sized specimens.
- Real structures are larger and behave differently.
- EPRI specimens will be used for testing and validation of new NDE techniques

Deliverables

Reliability of commercially available techniques to detect ASR – 2Q 2016

Results form Non-linear UT for detection of ASR – 4Q 2015 (NEUP project)

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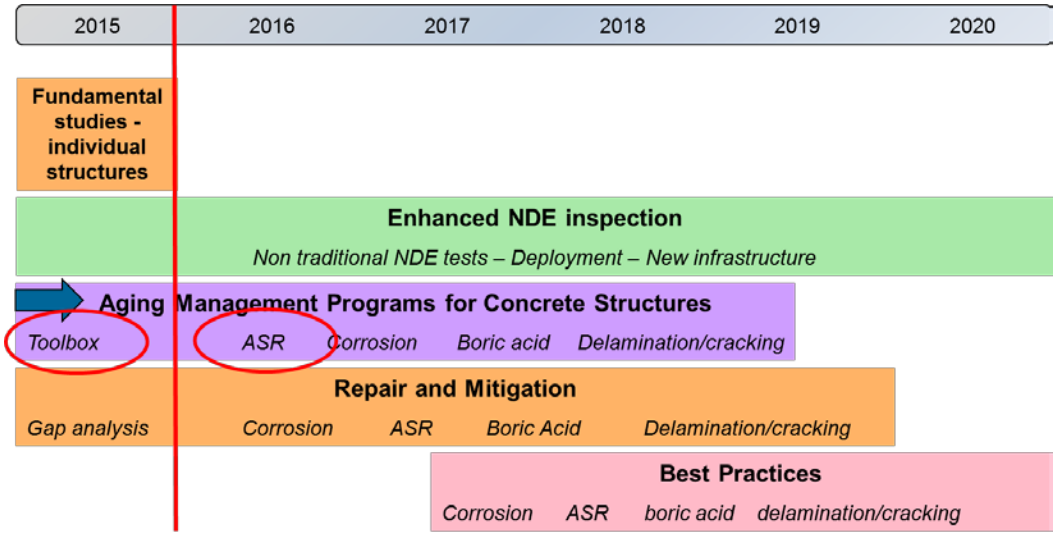
Aging Management Programs for ASR

Aging Management Toolbox
2013-18

- Project on Aging Management Programs for ASR affected structures starts in 4Q 2015



Coordination with EPRI Nuclear Sector Strategic Initiative on Aging Management

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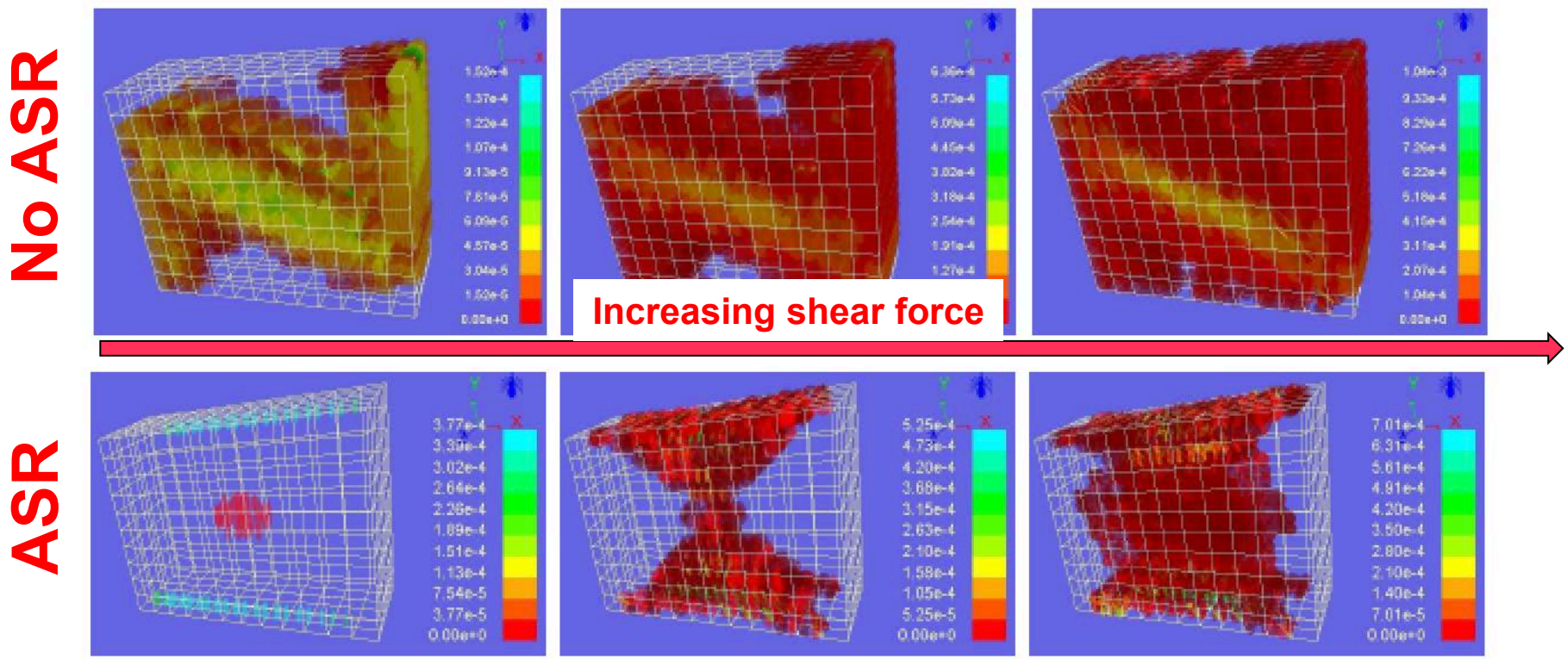
Modeling Shear Resistance of Reinforced Concrete Subjected to Alkali-Silica Reaction

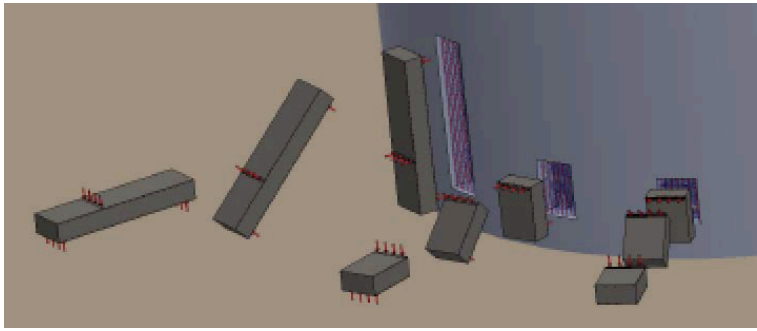
- Cooperation with the University of Colorado, Boulder
- Finite Element Analysis: Code *Merlin* specifically developed to account for ASR effects (Saouma and Perotti, 2006)
- Objective: evaluation of the post-ASR residual bearing capacity



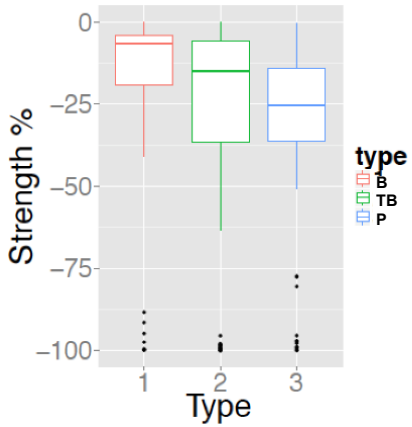
Parametric Study of the Effects of Confinement on the Shear Resistance of ASR-Affected Structural Members.

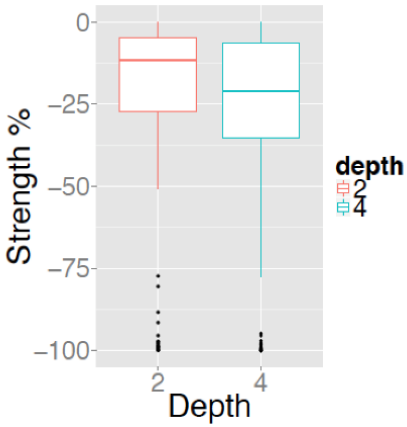
Parameters

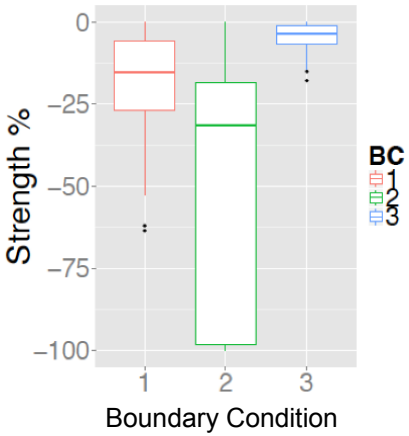
- Geometry Types: Beam (B), Truncated Beam (TB), Panel (P)
- Boundary Conditions: (1) Unrestrained, (2) Restrained, (3) Partially Restrained
- Thickness: 2 ft., 4 ft.
- AAR Expansion 0.1%, 0.2%, 0.3%
- Reinforcement Ratio 0.2%, 0.5%, 1%
- Residual Relative Elastic Modulus 0.7, 0.9
- Residual Relative Tensile Strength 0.7, 0.9



Loss or gain of shear strength results from the competition between material damage (ASR) and structural effects.

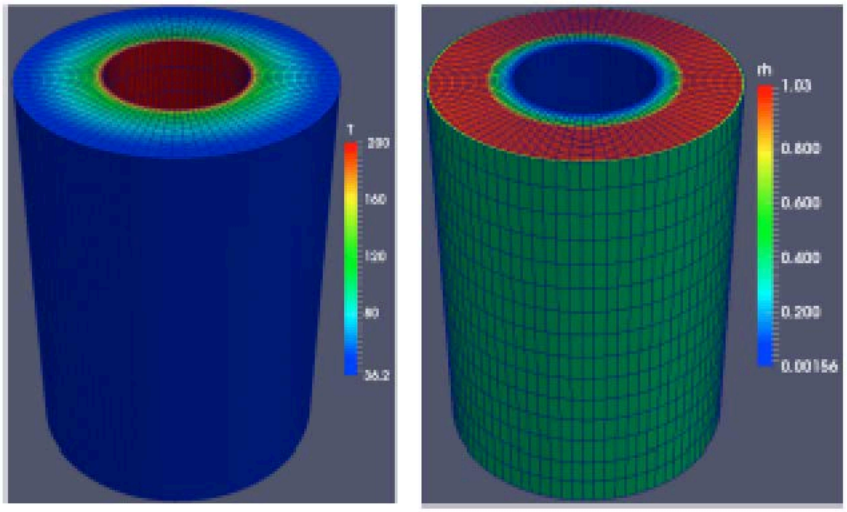


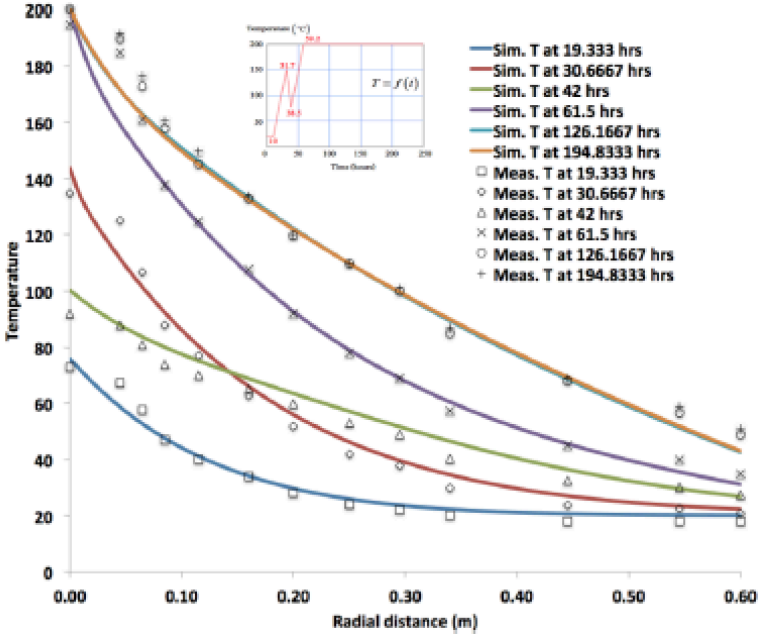




Development of Concrete Modeling Capabilities in GRIZZLY

- **MOOSE** (Multiphysics Object Oriented Simulation Environment)
- **Grizzly:** aging/degradation models
- Concrete related models implemented in 2015:
  - Coupled moisture transport/heat transfer, i.e., Bazant and Thonguthai's model
  - ASR swelling model, i.e., Saouma's model
- Models to be implement in 2016-7:
  - Cracking/damage
  - Radiation-induced volumetric expansion
  - Creep





Status of the Development of the Experimental Mockups at the University of Tennessee

- Objectives: study the effect of structural constraints on the development of ASR and the out-of-plane shear resistance of thick reinforced concrete panel
- Three blocks (1 non reactive, 1 reactive/"free" expansion, 1 reactive/constraint in one plane): size 100" x 80" x 40"
- Monitoring (bulk and surface): expansion, temperature, relative humidity, internal stress, damage
- PI: U. of Tennessee, TPO: ORNL
- Collaborative actions between: U. of TN (Knoxville), U of Alabama, U. of Colorado, U. of South Carolina, ORNL, INL, PNNL and EPRI…
- Kick-of meeting: September 4 2015
- Duration: 4 years

RILEM Technical Committee on Internal Swelling Reactions

- International group of experts
- Activity initiated in June 2014
- Chair V. Saouma, Secretary Y. Le Pape
- State of the Art and Needs for hydro- and nuclear industries (lead. V. Gocevski and F. Amberg)
- WG1. Compilation of test procedures for the Estimation of the Residual Expansion of Concrete in Structures Affected by Alkali Silica Reactions from France, Switzerland and North America. (lead. M. Hassanzadeh and V. Saouma)
- WG2. Survey of existing advanced ASR-models. Benchmark in preparation (lead. A. Sellier)
- WG3. NDE/monitoring. State of the art report (lead. L. Jacobs and P. Rivard)
- WG4. Large-scale testing (lead. Y. Le Pape)

Y. Le Pape, RILEM TC ISR Summer 2015 Activity Report, ORNL/LTR-2015/407, Aug. 2015

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Boric Acid Damage of Concrete in Spent Fuel Pools

Creep Impact on Post-tensioned Containment

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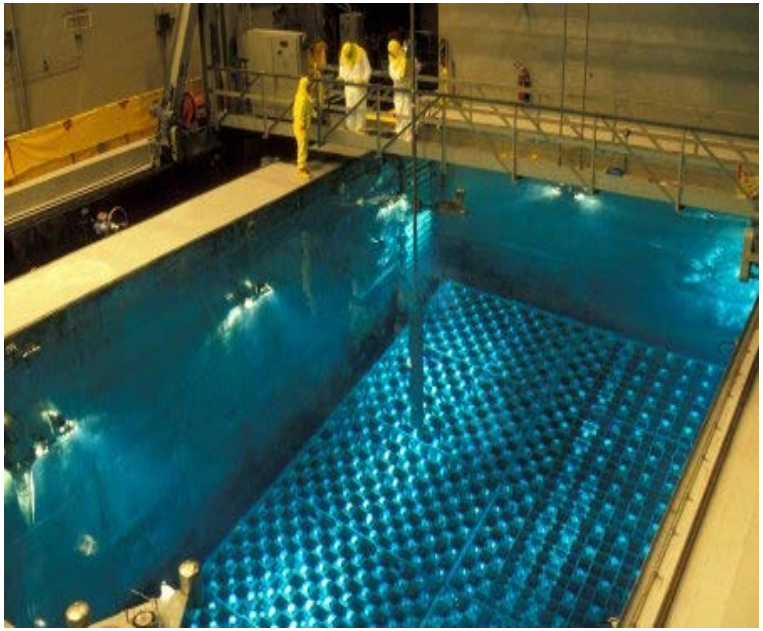
Boric Acid – Leaking PWR SFPs

Problem Statement:

Leakage of PWR spent fuel pools causes aqueous boric acid to come into contact with the concrete substructure.



Issue:

Impacts of accelerated leaching on concrete

Duration: 2013-2016

Deliverables

TR 1025166: Welding and Repair Technology Center: Boric Acid Attack of Concrete and Reinforcing Steel in PWR Fuel Handling Buildings

Technical Report: Framework for Aging Management of SFP Concrete Exposure to Aqueous Boric Acid Q4, 2016

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Boric Acid on Spent Fuel Pools

- 3 year experimental and modeling study conducted at CEA in Saclay, France.
  - Leaching and reactions of aqueous boric acid with cement paste
  - Reactions between aqueous boric acid and aggregates
  - Leaching and reactions between aqueous boric acid and concrete
  - Computational modeling of reactions

- Study will be completed fourth quarter 2015
- The results of the CEA study will be published by the Materials Aging Institute and used as a basis for the 2016 AMP framework report.

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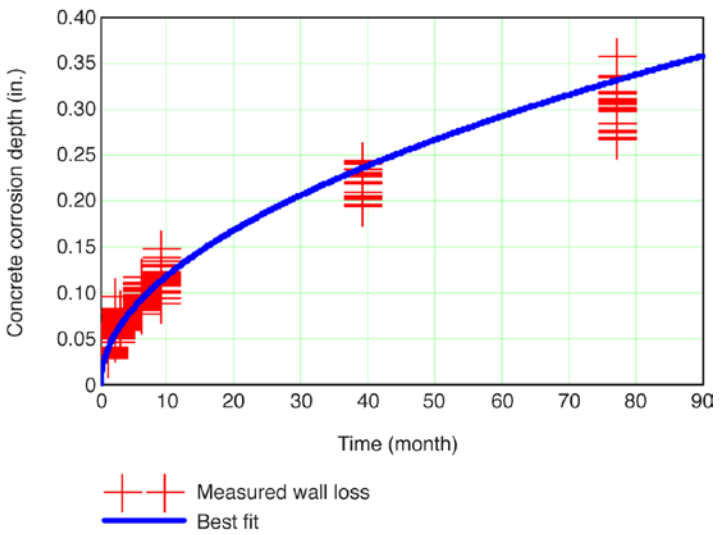
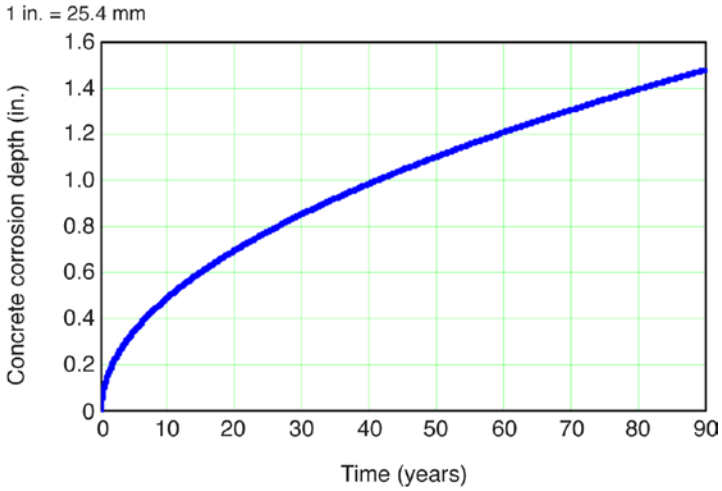
Technical Basis for Aging Management of BA on SFP

- Currently, a framework for aging management of leaking spent fuel pools is being developed
  - Review of CEA, MPR and Sandia National Laboratory studies.
  - Focus on both groundwater protection and potential structural effects.
  - Overall goal is to give utilities guidance for development of site-specific AMPs for leaking spent fuel pools specific to the concrete substructure.
  - 2016 AMP framework EPRI report





Plots taken from EPRI TR - 1025166

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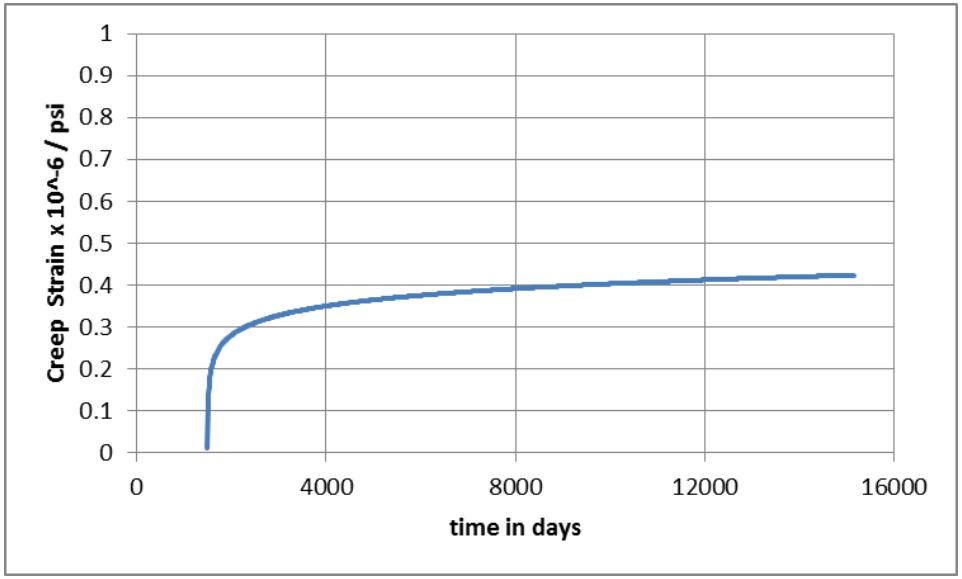
Concrete Creep in Containments

- EPRI published a literature review (3002003220) on creep in concrete containments in 2015
  - Post-tensioned containment structures are designed to allow for creep of the concrete.
  - Several models of creep in concrete are presented.

- Reg Guide 1.35.1 provides guidelines for the prediction of tendon losses – including the effects of long term concrete creep.
  - Creep strain rate decreases as a log function with time.
  - Creep is generally occurring as predicted by models used for containment design.
  - Creep does not appear to be a significant aging effect.



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Nonlinear Ultrasound Characterization of Concrete Creep

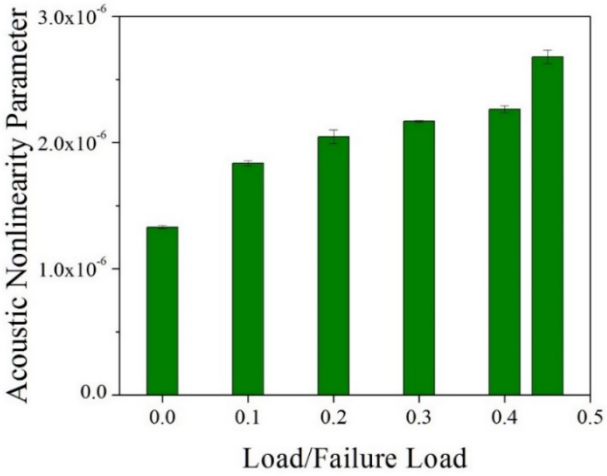
Problem Statement:

Long term creep (deformation) may affect the performance of post-tensioned concrete containment structures and cause deformation of containment liners.

Issue:

Nondestructive evaluation method to characterize creep strain

Duration: 2014-2015



Deliverables

TR 3002003220: Program on Technology Innovation: Concrete Creep of Nuclear Plant Containments

Technical Report – Nonlinear Ultrasound Concrete Creep Feasibility Study: Q4 2015

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  - V. Saouma, M. Hariri-Ardebili and Y. Le Pape, Effect of Alkali-Silica Reaction on Shear Strength of Reinforced Concrete Structural Members, ORNL/TM-2015/588, Sep. 2015
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United States Nuclear Regulatory Commission

Protecting People and the Environment

Aging Management of Concrete for Subsequent License Renewal (SLR)

Angela Buford
Office of Nuclear Reactor Regulation
Division of License Renewal

Madhumita Sircar and Jacob Philip
Office of Nuclear Regulatory Research
Division of Engineering
Division of Risk Analysis

November 17, 2015

Outline

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Regulatory & Technical Framework
- Technical Issues
- Plan to address Technical Issues
- NRC's Research to Address the Issues
- Expected Outcome
- Summary

Regulatory Framework

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- Regulatory requirements for condition assessment of concrete structures & components include:
  - 10 CFR 50.55a, “Codes and Standards” (ASME Code, Section XI, Subsection IWL – Concrete Containment Inservice Inspection Program)
  - 10 CFR Part 50, Appendix J, “Primary Reactor Containment Leakage Testing”
  - 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”
  - 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants”
  - 10 CFR Part 54, “Requirement for the Renewal of Operating Licenses for Nuclear Power Plants”

Technical Guidance

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- Generic Aging Lessons Learned (GALL) Report - recommended guidance for the condition survey and evaluation for concrete structures and components includes:
  - American Concrete Institute (ACI) Standard 201.1R, “Guide for Making a Condition Survey of Concrete in Service”
  - ACI Standard 349.3R, “Evaluation of Existing Nuclear Safety-Related Concrete Structures”
  - ACI Standard 318, “Building Code Requirements for Reinforced Concrete”
  - ANSI/ASCE 11-90, 99, “Guideline for Structural Condition Assessment of Existing Buildings” (American National Standards Institute/American Society of Civil Engineers)

Technical Issues

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- Expanded Materials Degradation Assessment Report (NUREG/CR-7153, Vol. 4) identified four degradation mechanisms with potential knowledge gaps for assessing integrity of concrete structures for operation up to 80 years
  - Alkali-Silica Reaction (ASR)
  - Effects of irradiation on concrete structures
  - Creep and potential for creep-fracture interaction of post-tensioned containment
  - Effects of potential boric acid attack on concrete and steel in pressurized water reactor (PWR) spent fuel pool (SFP)

Technical Issues

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- **ASR:**
  - Significance of ASR degradation
  - Long-term overall impact on structural performance
  - Programmatic aspects for managing this aging effect
- **Irradiation of Concrete:**
  - Characterization of degradation due to radiation
  - Structural significance for long-term operations
  - Programmatic aspects for managing this aging effect

Technical Issues

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- **Creep and Creep-Fracture Potential in Post-Tensioned Concrete Structures:**
  - Gradual loss of prestress
  - Periodic re-tensioning and cyclic activation of primary creep
  - Creep-fracture interaction under loading
- **Effects of potential boric acid attack on concrete and steel in PWR SFP:**
  - Kinetics and the extent of the attack
  - Role of concrete composition

Plan to Address Technical Issues

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Ongoing Domestic and International Interactions:

- NRC is leveraging SLR research with the Electric Power Research Institute (EPRI), Department of Energy (DOE), and international researchers
- OECD/NEA/CSNI/IAGE organized and conducted an international workshop "Assessment of Nuclear Structures Subject to Concrete Degradation" on June 30-July 1, 2015
  - Reports from workshop participants are being compiled and recommendations for the next phase being developed
- NRC is participating in International Committee on Irradiated Concrete
- NRC is in talks with the French Institute for Radioprotection and Nuclear Safety on exchange of research information on concrete degradation in the respective programs

Plan to Address Technical Issues

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- NRC and industry research initiatives underway for ASR and irradiation mechanisms. NRC issued User Need Request from NRR for research on the aging effects of:
  - ASR (May 2012)
  - Irradiation effects (May 2015)
- NRC is monitoring industry research initiatives regarding:
  - Creep and potential for creep-fracture
  - Potential boric acid attack on concrete and steel in the SFP

NRC Research Initiatives: ASR

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- **Research at National Institute of Standards and Technology (NIST)**
  - **Objectives:**
    - Assess the structural performance of ASR-affected concrete structures for design basis static and dynamic loading and load combinations through its service life (including 80 year SLR period)
  - **Outcome:** Technical basis to update regulatory guidance to establish:
    - Current structural capacity to resist static and dynamic loads, and an estimate of future structural capacity
    - Programmatic aspects for aging management (monitoring and managing) of ASR degradation

NRC Research Initiatives: ASR

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- Research at NIST, continued:

Task No.	Description
1	Assess in-situ mechanical properties (in progress)
2	Assess bond and lap-splice lengths of reinforcing bars
3	Assess seismic response characteristics of structural members
4	Estimate the degree of reaction in the concrete and the corresponding expansion (in progress)
5	Predict future and ultimate ASR expansion
6	Assess change of transport properties of the concrete and its potential for causing other degradation mechanisms

Reinforcement for Task 1 Concrete Block Specimens for NIST project

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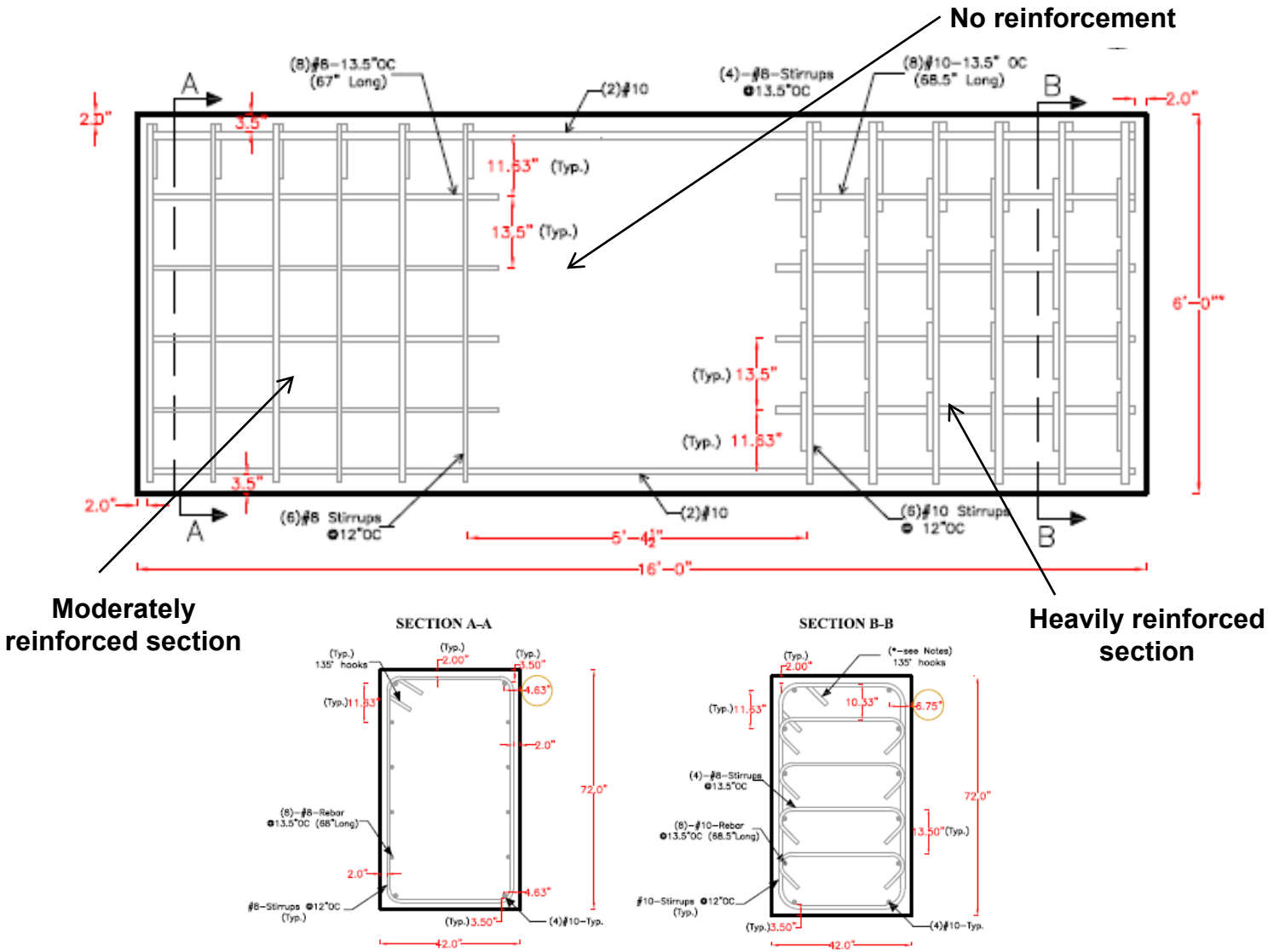

NRC Research Initiatives: ASR

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- **Research at NIST, continued:**
  - **Status:**
    - Two 16x6x3.5 ft skeleton block specimens, fully instrumented, have been built. Work on the third beam and the control specimen is in progress.
    - The design of the experiments for Task 2 has been completed. NRC to discuss the design with NIST before final
    - The mix designs for the concrete pours for Task 1 block specimens have been mostly completed
    - It is anticipated that the concrete pours for the block specimens will take place in early December
  - **Duration:**
    - Fiscal Year: 2014-2018

NRC Research Initiatives: ASR

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Research Grant - Service Lifetime Extension of Nuclear Power Plants: Prediction of Concrete Aging and Deterioration Through Accelerated Tests, Non-Destructive Evaluation and Stochastic Multiscale Computations," Northwestern University (2015-2018)

- **Objective:**
  - Develop a multiscale model for ASR deterioration mechanism
  - Incorporate stochastic representation of concrete material properties
  - Use accelerated tests and non-destructive testing for stochastic parameter identification
  - Formulate a predictive stochastic framework that incorporates mechanics-based models, random field models and laboratory/field data
  - Validate the framework against a reduced-scale case study
- **Outcome:**
  - Formulate and validate a stochastic framework for the prediction of concrete deterioration with specific focus on ASR damage

NRC Research Initiatives: ASR

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Research Grant – "Experimental and Numerical Investigation of Alkali-Silica Reaction in Nuclear Power Plants", University of Colorado

- **Objective:**
  - Assess experimentally the shear strength of ASR-affected reinforced concrete confined beams
  - Develop a methodology to predict future ASR expansion
  - Perform numerical investigation of nuclear power plant (NPP) - type structures (containment and shear walls)
- **Outcome:**
  - Develop a framework for assessing the shear strength degradation and numerical simulation of structural performance for ASR-affected NPP type structures

NRC Research Initiatives: ASR

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- **Research grant: University of Colorado (continued)**
  - **Status:**
    - Refining mix design
    - Making arrangements to cast beams
    - Designing curing facility
    - Preparing experimental setup
    - Started work on numerical simulation
  - **Duration:**
    - Fiscal Year 2015-2018

NRC Research Initiatives: Irradiation Effects

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- **Objectives:**
  - Estimate the level of irradiation on concrete structures close to the reactor pressure vessel for the SLR period
  - Assess the significance of radiation on structural integrity and shielding performance
- **Research Phase 1:**
  - Perform confirmatory review of the technical and scientific bases for the use of screening irradiation thresholds, including their applicability to the various reactors from the point of view of the acceptability of this approach for licensing decisions
  - Review attributes and characteristics of data compiled by DOE and EPRI (neutron fluence and flux and energy, gamma dose, temperature, uncertainties)

NRC Research Initiatives: Irradiation Effects

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- **Research Phase 1 (continued):**
  - Examine sufficiency of the data to account for, or bound (for each reactor type)
    - Concrete variability (e.g., aggregate variability)
    - Structural layout and details and its variability
    - Irradiation environments (e.g., neutron and gamma, related temperature, humidity) and their variability
    - Combined effects (e.g., neutron and gamma, irradiation and temperature, irradiation and irradiation-assisted ASR)

NRC Research Initiatives: Irradiation Effects

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- **Research Phase 1 (continued):**
  - Establish the radiation thresholds above which irradiation (neutron, gamma temperature, stresses) induced degradation may affect properties of concrete and reinforcement (including steel embedment)
  - Perform characterization and screening of concrete degradation due to irradiation under representative conditions of light water reactors that will affect the structural (strength, stiffness, bond, anchorage) and shielding functions
- **Status:** Contracting work in progress
- **Duration:** Fiscal Year 2016-2018

NRC Research Initiatives: Irradiation Effects

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- **Research Phase 2 (Plan):**
  - Conduct confirmatory testing using concrete harvested from decommissioned nuclear power plants and also laboratory-created concrete specimens (possible collaboration DOE-EPRI-NRC; discussion on-going)
  - Perform independent confirmatory evaluation, including numerical simulation and validation of structural safety significance and shielding performance to maintain required functions
  - Evaluate aging management strategies for structures that are in close proximity to reactor
- **Duration:** Fiscal Year: 2016-2020 (plan)

NRC Research Initiatives: Irradiation Effects

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- **Research Phase 2 Expected Outcome:**
  - Technical basis for regulatory guidance that captures staff position on:
    - Significance of radiation effects on structures that are close to reactor: structural and shielding performance to maintain the intended functions
    - Programmatic aspects for aging management (monitoring and managing) for radiation effects on concrete structures

Summary

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- **Resolution of technical issues:**
  - **Near-term: Plant-specific evaluation**
    - GALL Report aging management programs and the Standard Review plan updated for SLR to address the technical issues through recommended “further evaluation” on a plant-specific basis
  - **Long-term: Confirmatory research**
    - Research will form the technical basis for updating regulatory guidance in the future, if necessary

Electrical Cables – Aging Management Research

Joint EPRI-LTO DOE-LWRS Presentation to ACRS November 17, 2015

Keith Leonard

LWRS Program Materials Aging and Degradation Pathway Lead

Sherry Bernhoft

EPRI LTO Program Manager

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EPRI Research in Electrical Cables

- Aging of cable insulation materials due to radiation, temperature and submergence
- EPRI Technical Reports:
  - ✓License Renewal Electrical Handbook
  - ✓Cable Aging Reports
    - ✓Low voltage thermal/radiation
    - ✓Low and Medium voltage wet/submerged failure mechanisms
  - ✓Medium and Low Voltage Aging Management Implementation Guidelines
  - ✓Cable Testing Guidelines
  - ✓In-containment Radiation and Temperature Measurements

Electrical cable systems are performing well and tools exists for aging management.

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Cable Program Summary

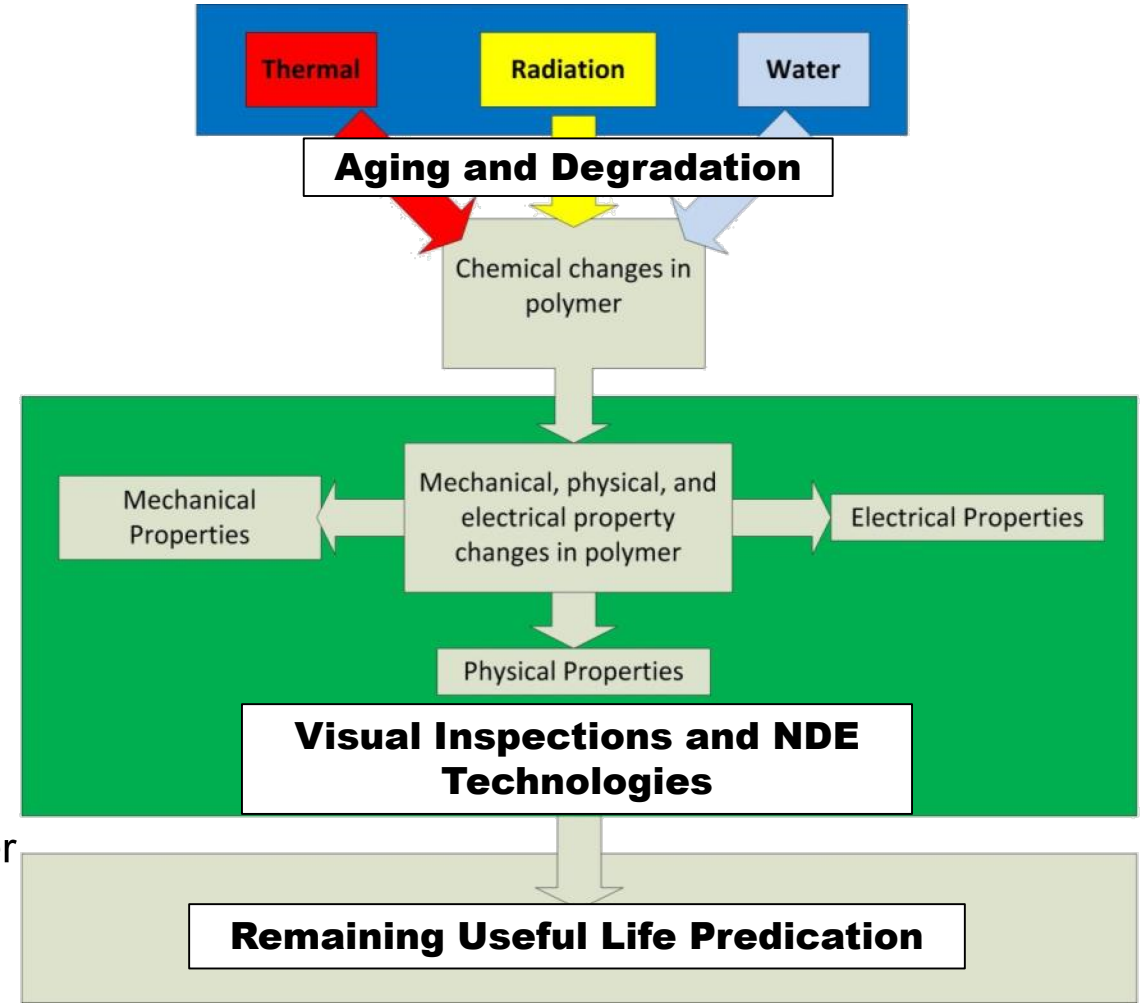
Pacific Northwest NATIONAL LABORATORY

Cable Stressors

Chemical Changes

Changes in Properties

Changes in Performance over Time



Detailed Understanding

Effective Treatments

Key Indicators of Cable Aging

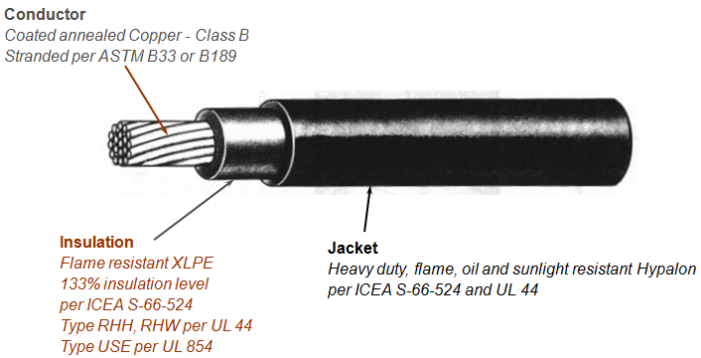
Transformational NDE

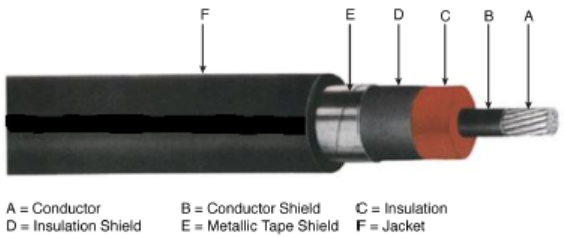
Methods for Life Prediction

**LWRS NDE R&D Roadmap 2012* PNNL-21731

OAK RIDGE National Laboratory
Pacific Northwest NATIONAL LABORATORY
EPRI
U.S.NRC United States Nuclear Regulatory Commission Protecting People and the Environment

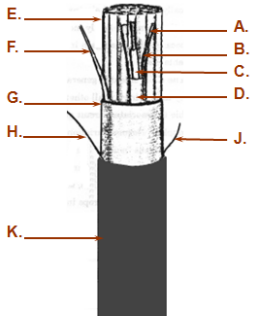
Cables Types





Low-voltage cable <2 kV

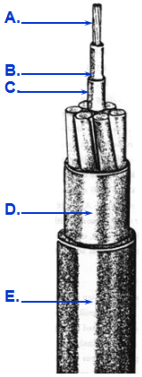
A. Bare stranded copper conductor
B. Insulation
C. Tinned stranded copper group drain wire
D. Aluminum-polyester isolated group shield
E. Twisted, shielded pairs/triads
F. Communication wire
G. Aluminum-polyester cable shield
H. Tinned stranded copper cable drain wire
J. Rip cord
K. Jacket



Instrument Cable

Medium-voltage cable 5-46 kV

A. Coated stranded conductor
B. Insulation
C. Jacket
D. Extruded belt or cable tape and fillers
E. Outer jacket



Control Cable

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Primary Aging Stressors for Cable Type

Low Voltage (LV) Cables (<2kV)

Major aging concerns

- Thermal >50° C
- Radiation >200 kGy (20 Mrad)

Lesser Aging Concerns

- Chemical
- Polymer stability in water

Low voltage cables age more thermally due to thinner insulation layer

Medium Voltage (MV) Cables (< 46 kV)

Major aging concerns

- Electrical stress induced degradation : water treeing, partial discharge in splices or terminators

Lesser Aging Concerns

- Thermal
- Radiation
- Ohmic heating
- Chemical

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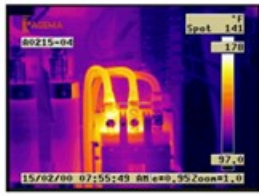
Nuclear Power Plant Cable Aging Management Strategy

- **Evaluate for susceptibility –** focus on rooms/areas with highest temp and highest radiation. Also give special attention to most safety critical components. Select samples for test if signs of aging are seen
- **Visual walk-down** looking for visible indications on jackets.
- **Tan-Delta and Withstand Testing for MV and Insulation Resistance for LV (and unshielded MV) l**ooking for worst case areas of degradation on sample of cables.
- **Local specific NDE** (indenter, capacitance, UT, …) at local area identified with bulk tests.
- **Repair/replace** where indicated. Extend evaluation to other similar areas and the next most severe locations. Take additional actions based on visual/specific evaluations

Heat

Water

Electrical



Mechanical

Corrosion

Radiation

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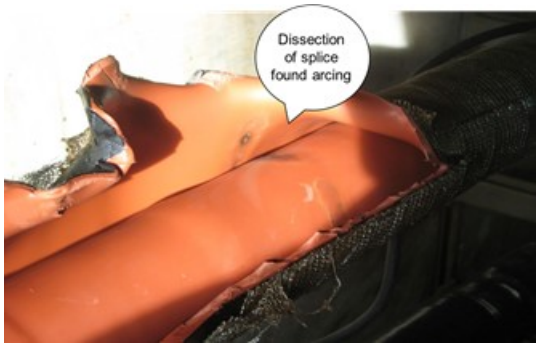
Condition Monitoring: Tan δ Testing Shielded Medium Voltage Cable

- Tan δ testing is used to evaluate shielded medium voltage cable
- Applying EPRI Tan δ acceptance criteria will identify degraded cable insulation due to water treeing or thermal damage in MV cables, cable terminations and splices
- EPRI collects member test data since 2009 to evaluate effectiveness
- Report 1025262-June 2012 validated assessment criteria
- Report 3002005321-September 2015
  - Degraded cables, wet and dry, identified by tan δ
  - Failure mechanism confirmation of tan δ results of cables that tested in severely degraded criteria
  - No failures of cables with good or slightly degraded tan δ



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Radiation and Temperature Assessment for Cables

- Phase 1:
  - Existing radiation and temperature data collected from US NPPs
  - Limited responses, mostly temperature data (14 plants) with limited radiation data
  - While data indicated less severe conditions then qualification lifetime values, there was insufficient data to draw any specific conclusions
- Phase 2:
  - 55 radiation and temperature monitors have been installed in containment and other high thermal/radiation areas
  - Collect data radiation and temperature data for a fuel cycle
  - Quantify cable “operating environment”
  - Evaluate collected data to see if it can inform inverse temperature, synergistic effect research
  - Provide methodology for future studies

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Harvesting of Cables

- Harvesting of aged cables provides valuable research data:
  - Crystal River 3
  - Zion
  - International (Krsko and Ringhals) may have cables available from cables replacement work
  - Evaluating other harvesting opportunities
- Cable Harvesting guide issued (3002002994)

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DOE / LWRS Research in Electrical Cables

- Synergistic and inverse temperature effects due to thermal and radiation degradation
- Diffusion limited oxidation in LOCA tests
- Dose rate effects
- Limitations of activation energy values
- NDE (key indicators, methods)

Addressing Gaps in the Expanded Materials Degradation Assessment

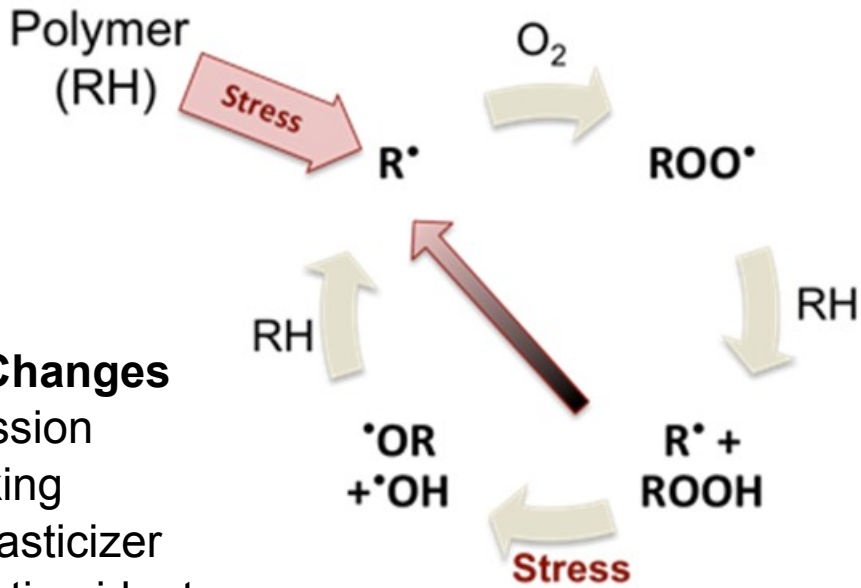
Research and experimental test develops data on key factors and synergies contributing to cable degradation from which predictive models can be developed to evaluate actual cable lifetimes.

LWRS U.S. DEPARTMENT OF ENERGY Nuclear Energy

Multi-variable Environmental Effects on Cable Insulation

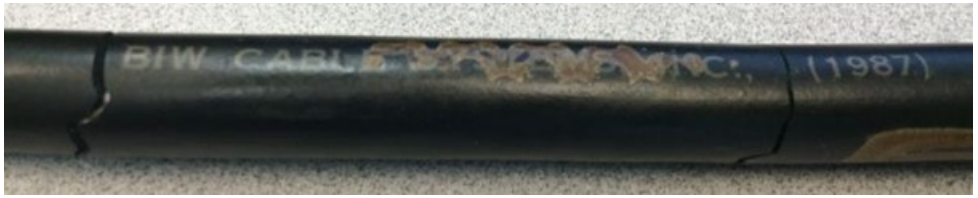
Environmental Stressors

- Radiation (UV, gamma)
- Thermal
- Mechanical (tensile, vibration)
- Moisture



Chemical Changes

- Chain scission
- Cross-linking
- Loss of plasticizer
- Loss of anti-oxidant

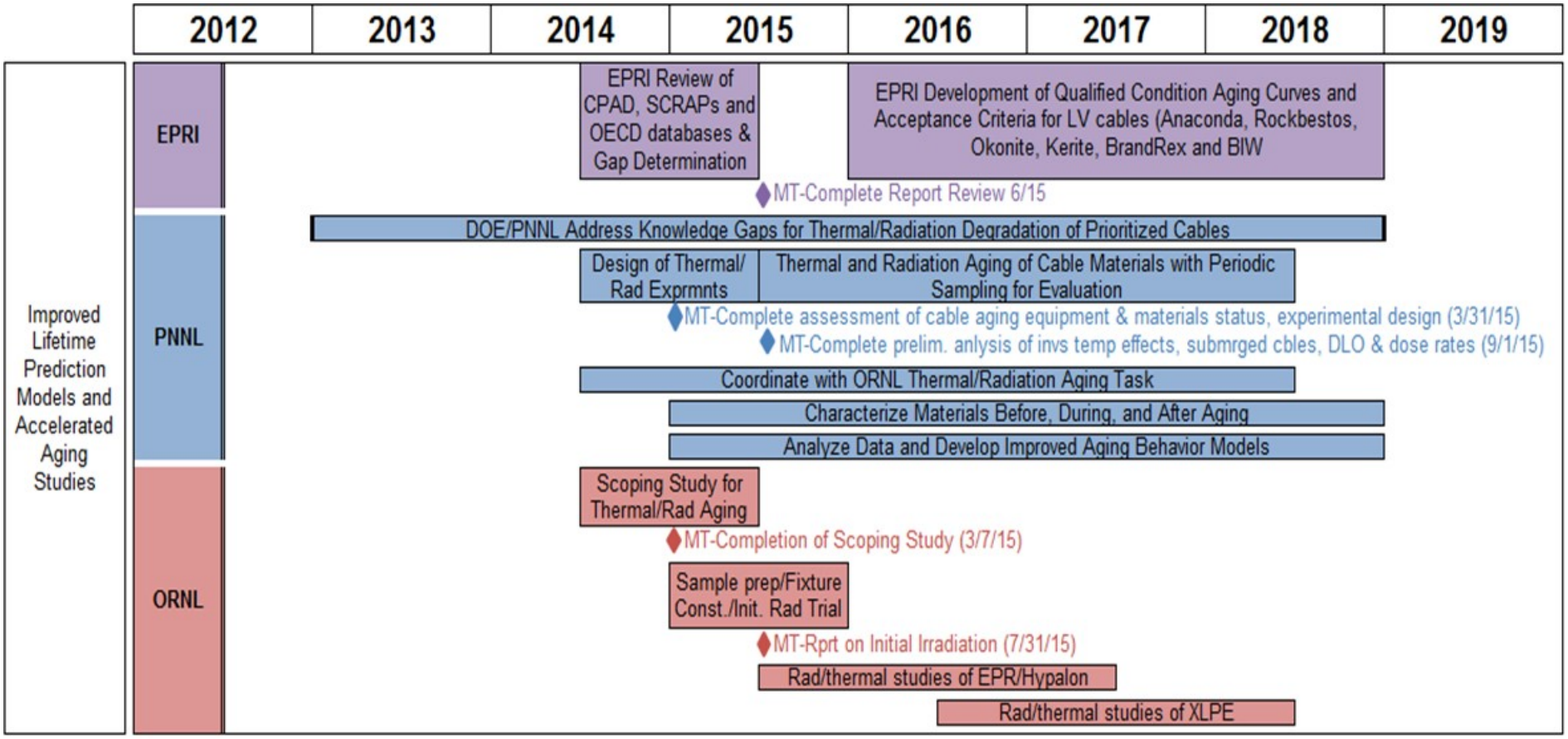


Property Changes

- Mechanical
- Electrical
- Physical

LWRS
U.S. DEPARTMENT OF ENERGY
Nuclear Energy
Pacific Northwest NATIONAL LABORATORY

Improved Lifetime Prediction Models and Accelerated Aging Studies



Materials Focus on Cable Aging Research

Applications

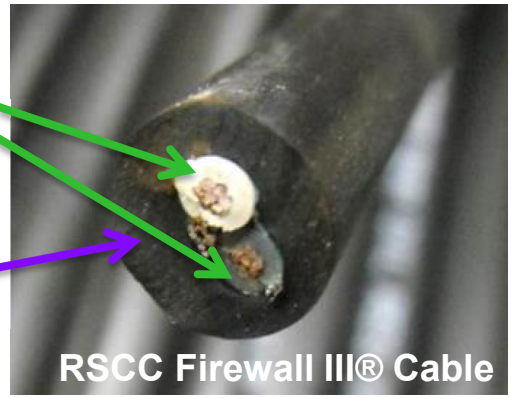
- Instrument and Control (81%)
- Power Cables (15%)
- Communication (5%)

Insulation

- XLPE - cross-linked polyethylene
- EPR - ethylene-propylene rubber
- Silicone Rubber (SiR)

Jacketing

- Hypalon® - Chlorosulfonated polyethylene (CSPE)
- Neoprene - Polychloroprene
- CPE - Chlorinated polyethylene elastomer
- PVC - Poly(vinyl chloride)



Cables in US Plants

36% of cables are XLPE

36% of cables are EPR

5% of cables are SiR

Cables in Containment

90% of units have XLPE

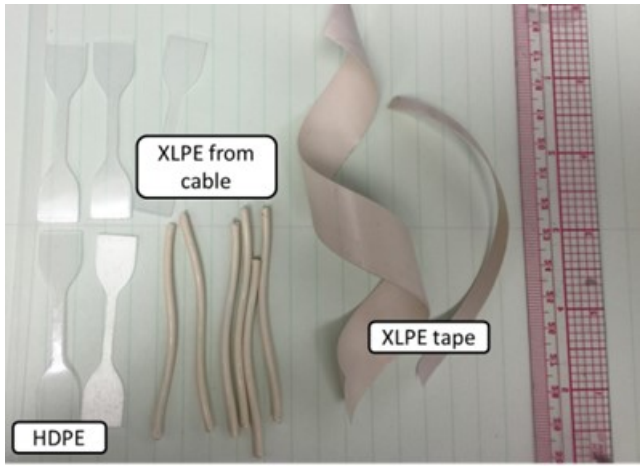
70% of units have EPR

30% of units have SiR

Separate and Synergistic Thermal / Irrad. Aging

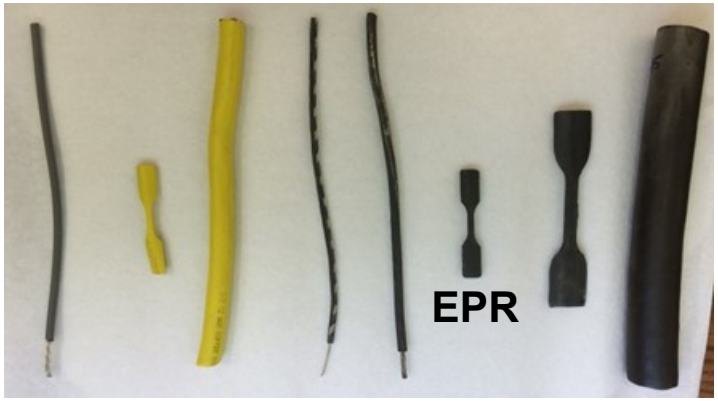
Method /Approach

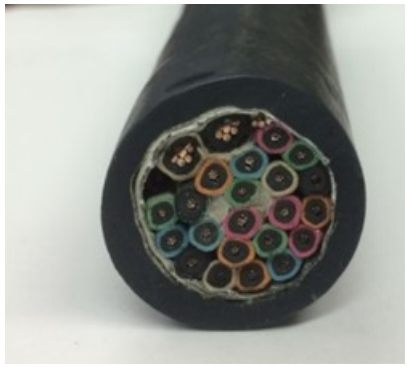
- Coordinated, accelerated aging of harvested cable insulation/jacket materials that are representative of current NPP systems
- Electrical and mechanical characterization to determine key factors in cable aging
- XLPE and EPR materials tested



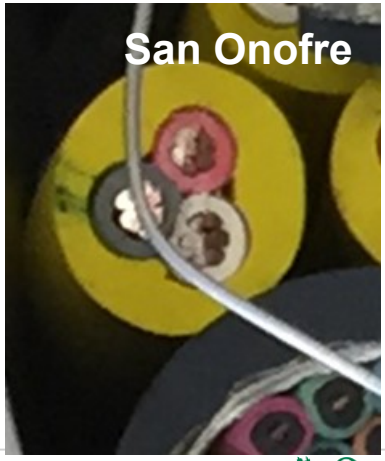
Synergistic Testing Conducted on Harvested Cables

- Insulation and jacket samples produced from harvested cables from decommissioned NPP's
- Remaining useful life determination
- *Callaway control rod cable*
  - Hypalon jacket with EPR insulated wires
  - In vessel operation for approximately 30 yrs
- *San Onofre power cable*
  - Firewall III CSPE jacket with XLPE insulated wires
  - Stored on site since 2006, never placed into operation

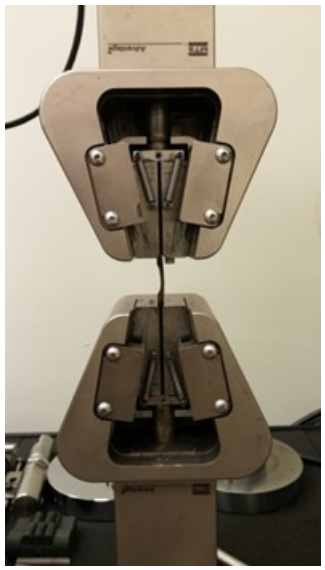




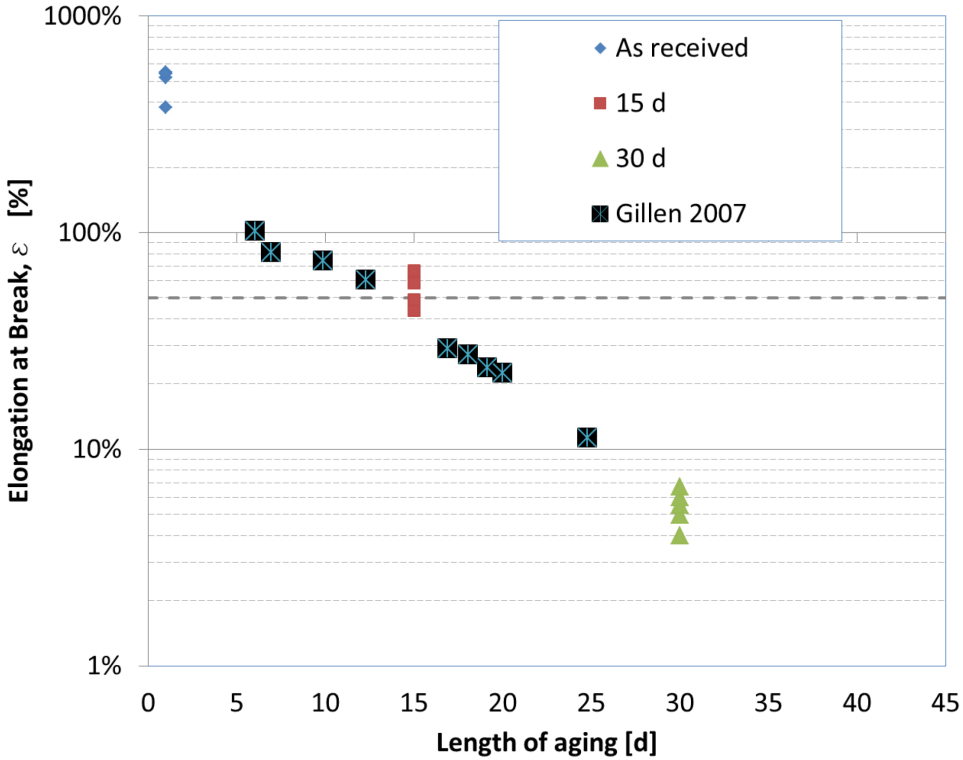
Callaway



Thermal Aging of Callaway Control Rod Cable Jacket



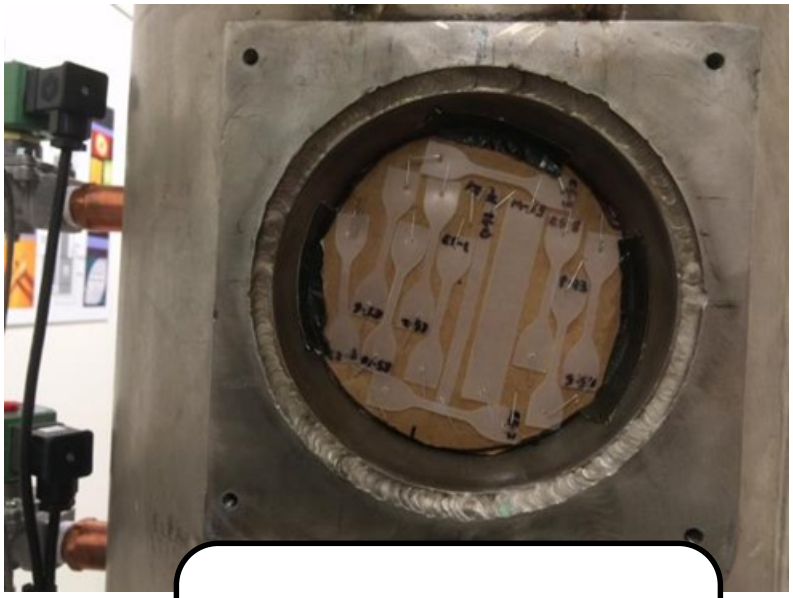
Elongation at break (EAB) measured with Instron 4465 Tensile Tester according to IEC/IEEE 62582-2



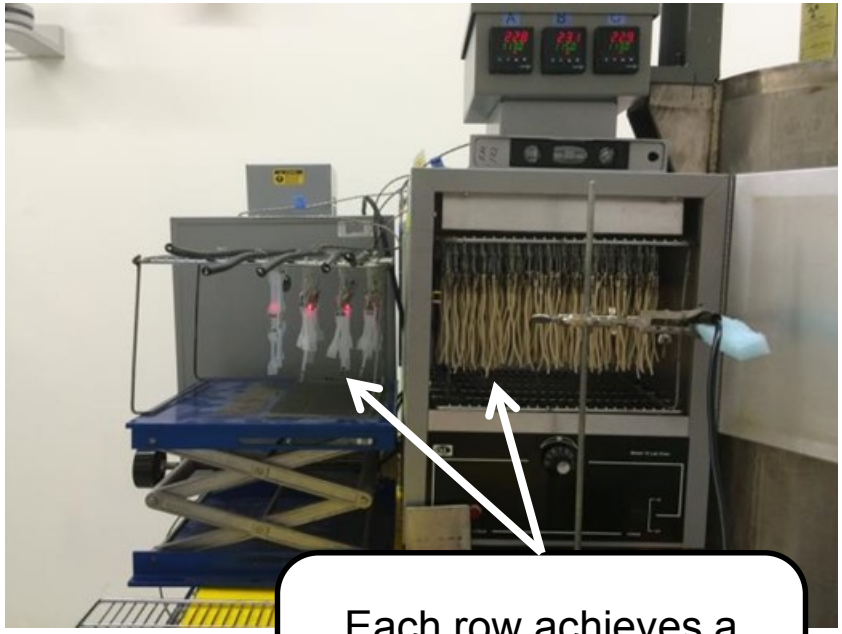
EAB with respect to length of aging in air at 125°C

Good agreement possibly indicates little evidence of aging since 1984 installation: Additional condition verification of Gillen 2007 and Callaway under way

Combined Exposure Experiments at PNNL



PE Samples inside source cone—highest dose (>1kGy/h)



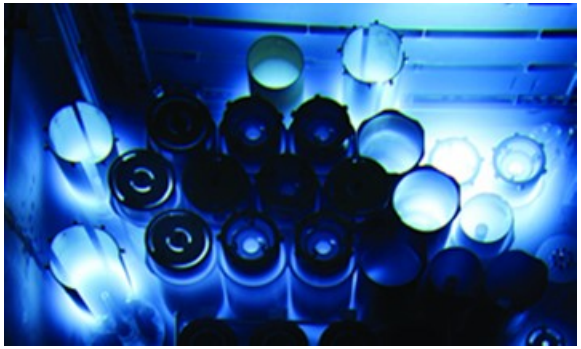
Each row achieves a distinct dose rate at constant temperature

Cable Irradiation Aging at ORNL (Dose Rate and Effects of Oxygen)

- Co-60 irradiator
  - 60 day irradiation
  - Dose rate 140 Gy/hr
  - Accumulated doses of 50, 100, 150 and 200 kGy.

Sample canister and sample arrangement for insertion into Co-60 irradiation

- HFIR - Gamma Irrad. Facility
  - 10 day irradiation cycle
  - Dose rate 410 Gy/hr
  - Accumulated doses of 20, 50, 80 and 100 kGy

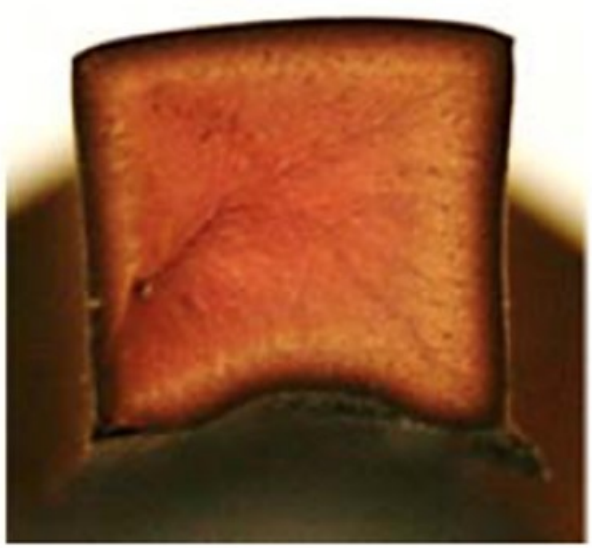


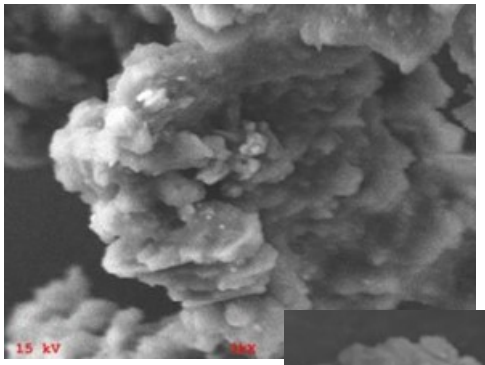
Sample holder for high dose GIF exposures

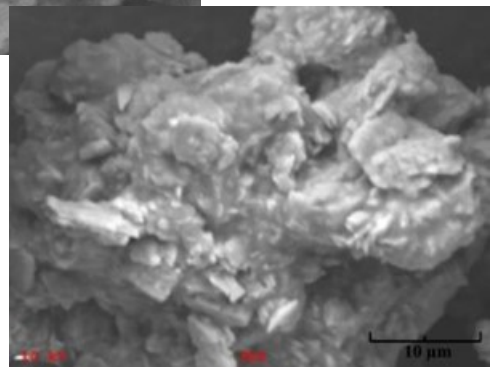
Data at multiple dose rates and accumulated doses provide fundamental information for model development

Inhomogeneous Aging Study
Understanding of Mechanisms

- Diffusion Limited Oxidation
- Nucleation of Degradation
- Effect of Sample Geometry



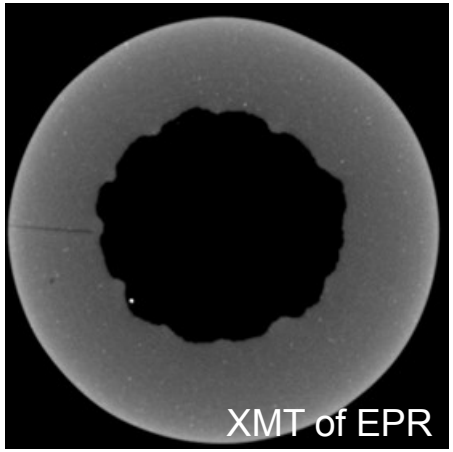


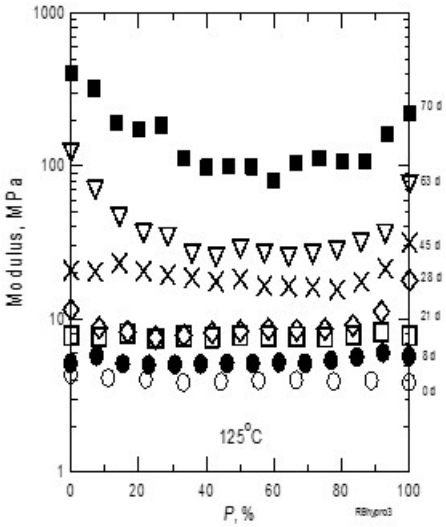


Microstructure Analysis

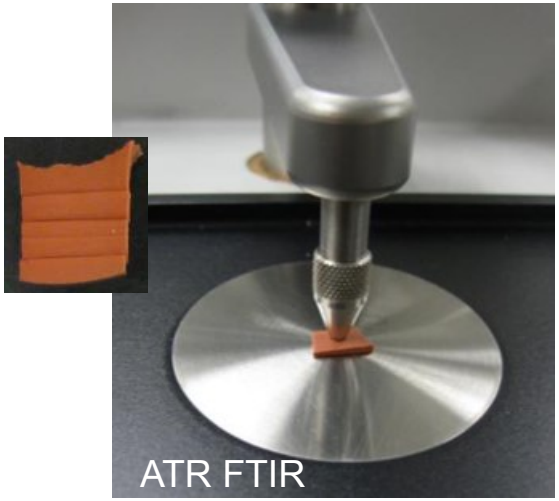
Imaging and Quantifying Degradation

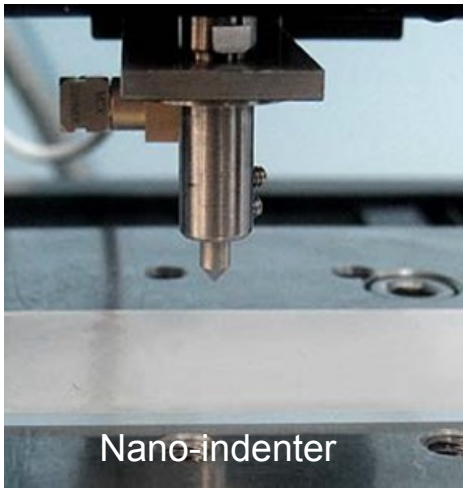
- Defect mapping
  - X-ray microtomography
- Chemical mapping
  - TOF-SIMS/XPS
  - X-ray diffraction
  - FTIR/Raman
- Mechanical mapping
  - Nanoindenter



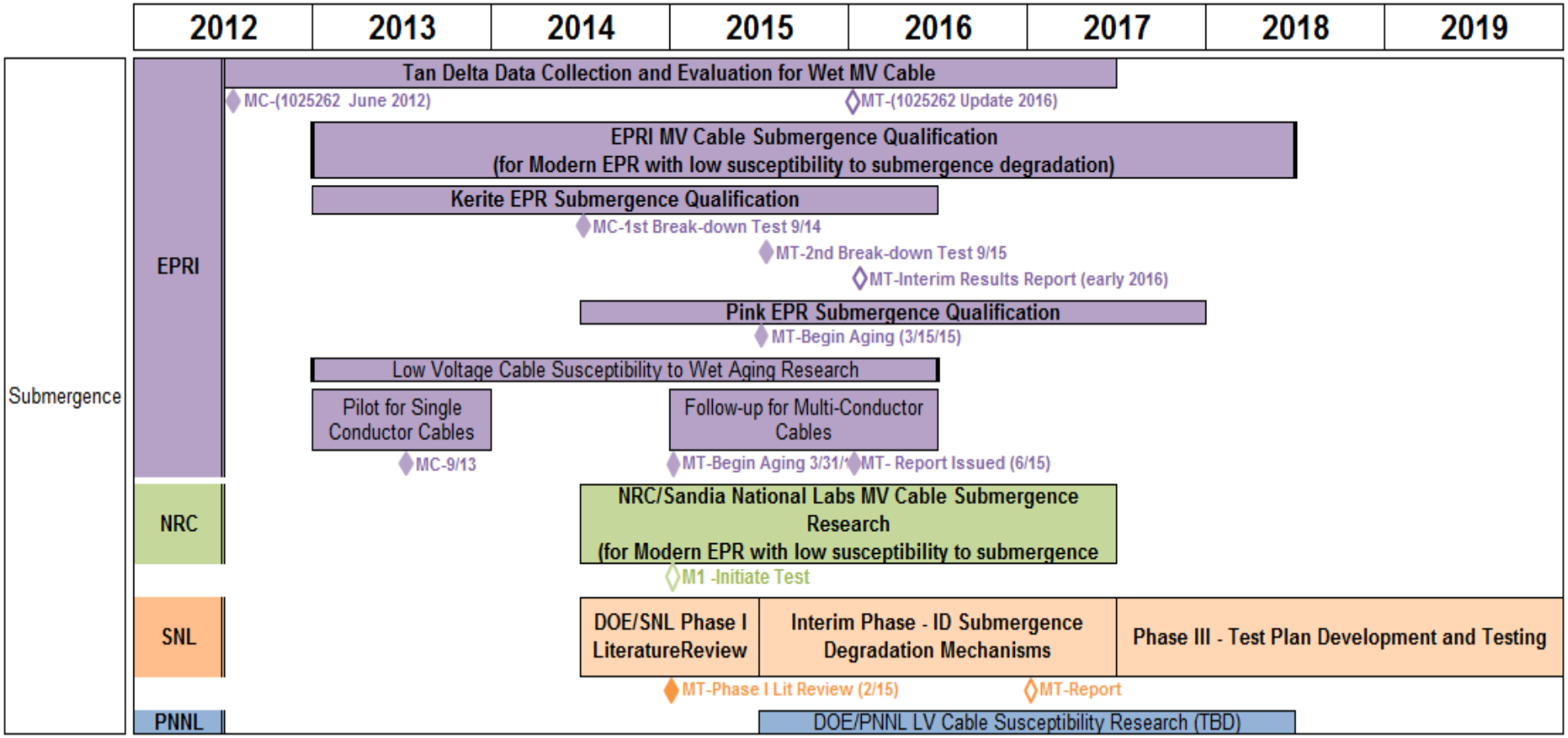


*NUREG/CR-7153, Vol. 5





EPRI and NRC Coordinated Research Effort on Submergence Research



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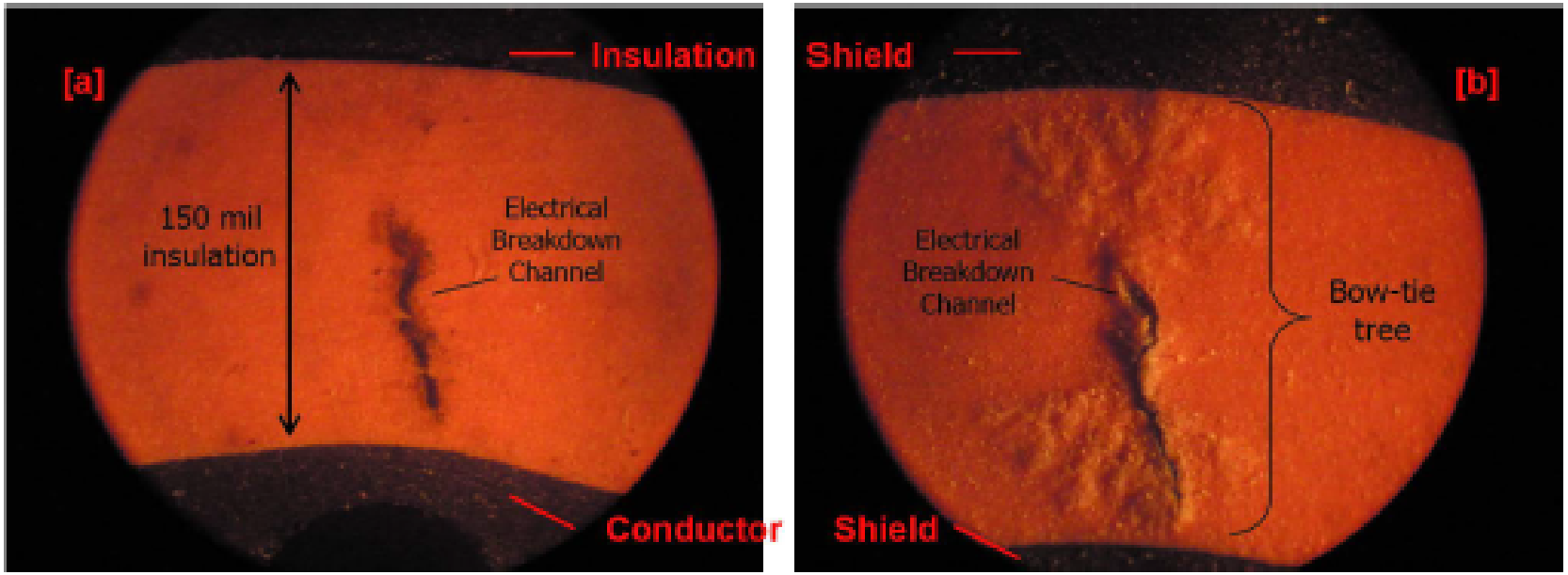
Water Treeing: Cable Degradation in Wet Environments



- Water Treeing, conditions needed:
  - Voltage stress > 0.8 kV/mm
  - Water or high humidity
  - Polymer
- Medium voltage typically operates > 1 kV/mm
- Low voltage typically operates < 0.2 kV/mm

Only Medium Voltage Cables Water Tree

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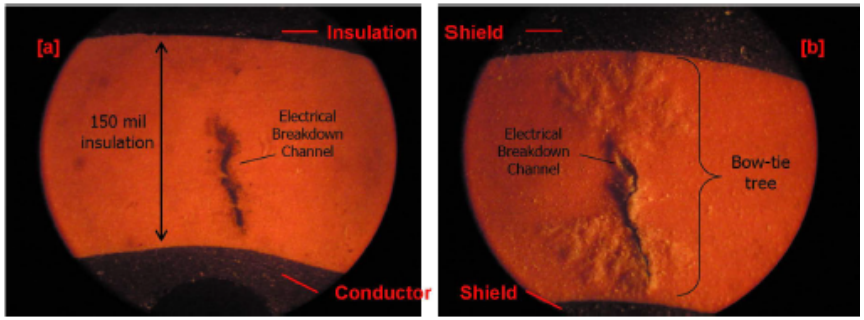
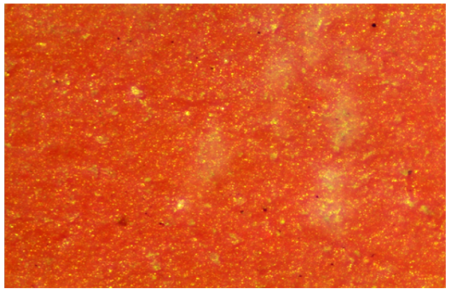
Understanding Water Treeing

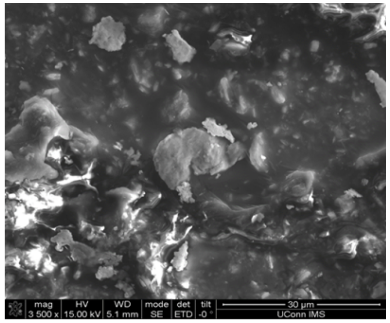
- Failure Mechanism Research Report 3002005323, Update 7 (August 2015)
- Completes wet failure mechanism research
- Confirmation that defects in EPR are identified by tan δ testing



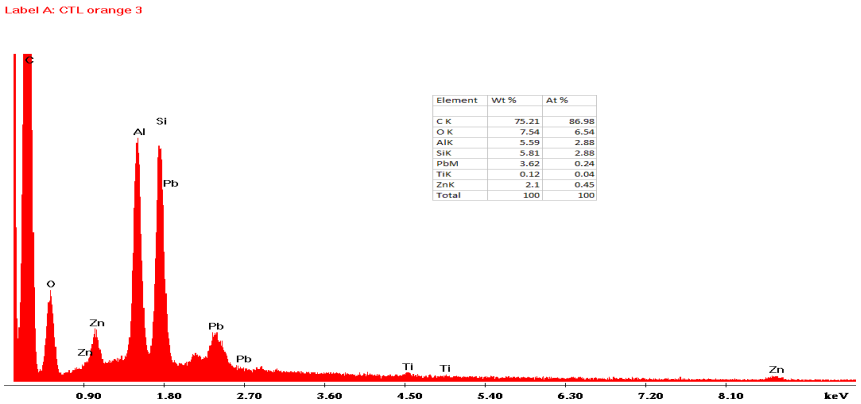
- Pink EPR, 8 kV, manufactured 1979-80
- Water tree root contains metallic elements
- Metallic elements act as catalyst for water tree growth



Optical photomicrograph of CTL-G bow-tie tree



FE-SEM image of CTL-G wafer water tree



Element	Wt %	At %
C K	75.21	86.98
O K	7.54	6.54
AlK	5.59	2.88
SiK	5.81	2.88
PbM	3.62	0.24
TiK	0.12	0.04
ZnK	2.1	0.45
Total	100	100

EDX spectrum of the area in question.

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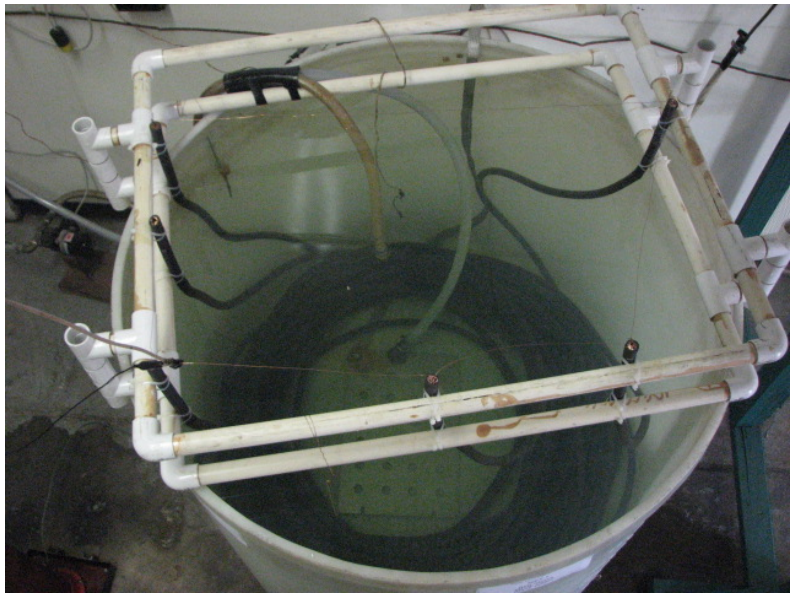
Qualifying MV Cables For Submergence

- **MV Cable Submergence Qualification-Supplemental Project**
  - Attempting to accelerate aging via high frequency and high voltage to obtain "accelerated aging factor" for submergence
  - Brown ethylene propylene rubber cable is 2+ years aging, complete in 2016 or 2017
  - Pink ethylene propylene rubber cable complete in 2018
  - Power law used to determine aging factor

Qualified Life= (Operating Time) + (Accelerated Aging Time X Aging Factor)



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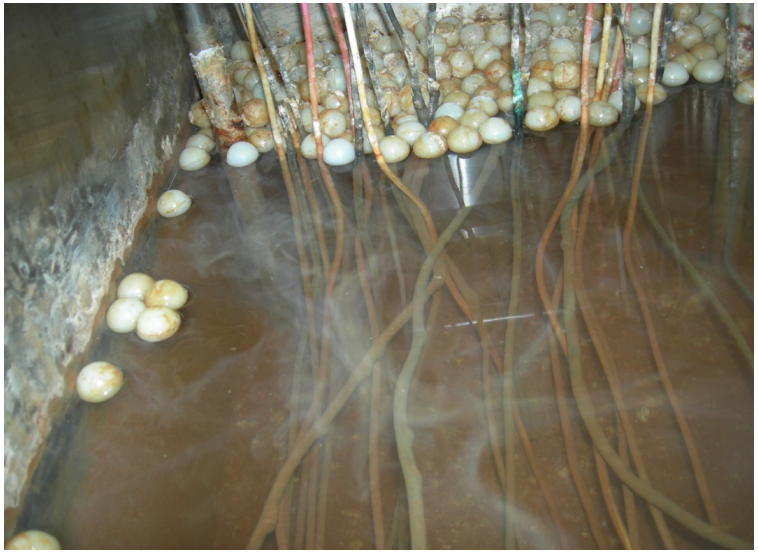
Researching Low Voltage Cable Wet Aging

- Low voltage wet susceptibility study: Do LV cables wet age?
  - Pilot study: No issue with Cross-linked polyethylene
  - Possible issue with ethylene propylene rubber insulations, however
  - Follow-up study will look at jacketed, multi-conductor cables with no thermal or insulation physical damage
  - 1 year aging at 90°C water temp, AC and DC voltages
  - Results of Pilot and follow-up published in late 2016



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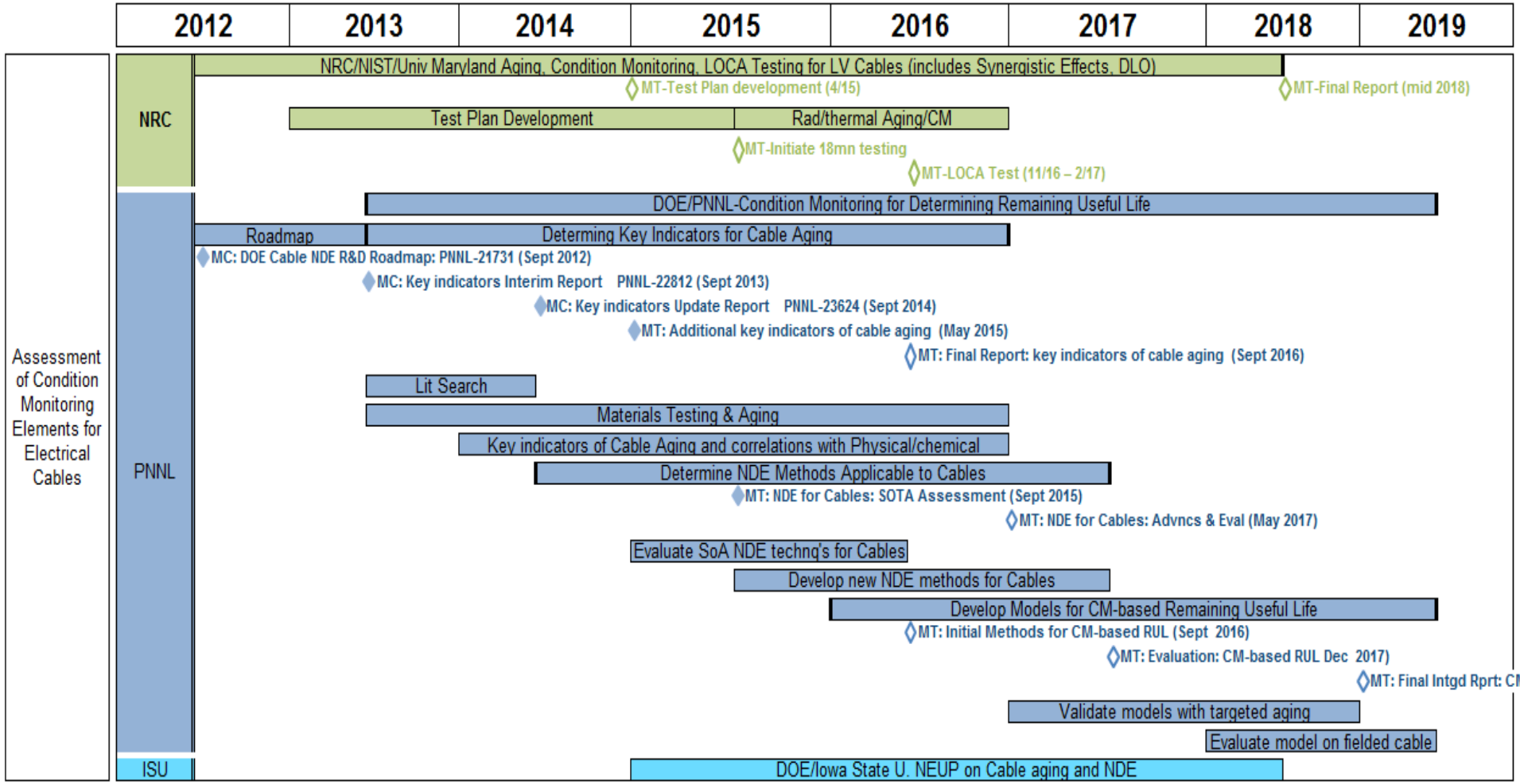
Assessment of Condition Monitoring Elements for Electrical Cables



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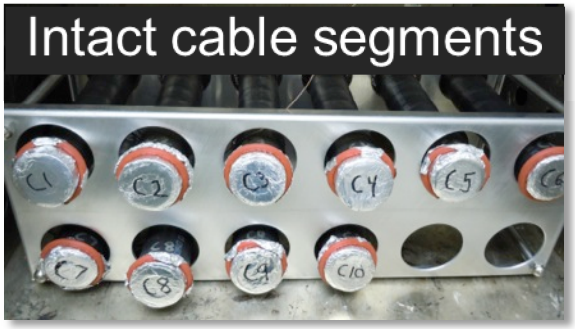
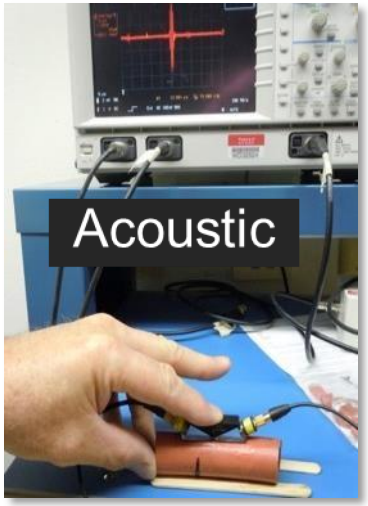
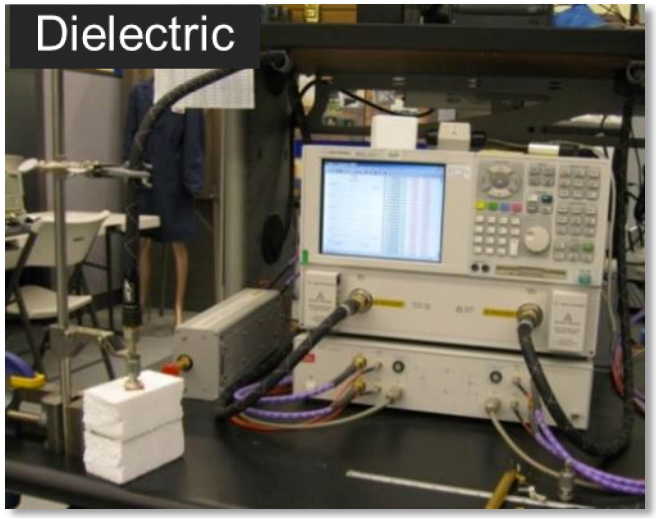
Non-Destructive Evaluation (NDE) of Cable Remaining Useful Life

Pacific Northwest NATIONAL LABORATORY

- Coordination of Aging and NDE
- Sensitivity analysis of key indicators
- Correlation of destructive and non-destructive data
- Assessment of NDE methods







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Condition-Monitoring Techniques for Electric Cables Used in NPPs

(NRC Reg Guide 1.218)

Test	Applicability	Ends	Damage	Comment
DC High Pot/ Step Voltage	Cable – 2/C	Both	Maybe	Not trendable
Very Low Freq. Tan-Delta	Cable – C/S	Both	Maybe	Common for Med Voltage Cable
Visual / Illum. Borescope	Visible exterior	No	No	Not quantitative
Indenter	Local Jacket	No	No	Trendable
Dielectric Loss Dissipation	Cable – 2/C	Yes	No	Not for long/lrge cble
Insulation Resistance	Cable – 2/C	Both	No	Not trendble/uncrtain
Partial Discharge	Cable – 2/C	Both	Yes	Locates weak point
Time Domain Reflectometry	Cable – 2/C	Both	No	Limited val for insul.
Frequency Domain Reflectometry	Cable – 2/C	Maybe	No	Can ID local flaws
IR Thermography	Under load	No	No	Weak signal for insul.

2/C=2 conductors required
C/S=Co-ax conductor & shield

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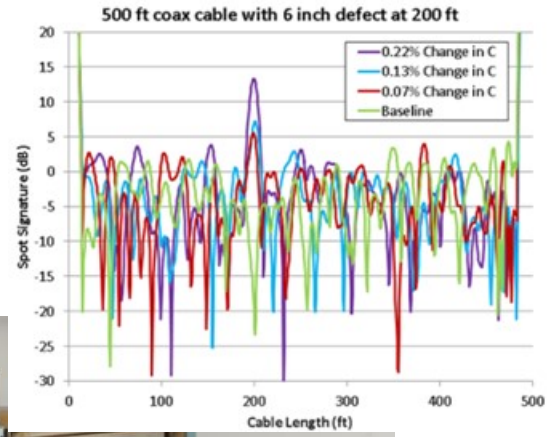
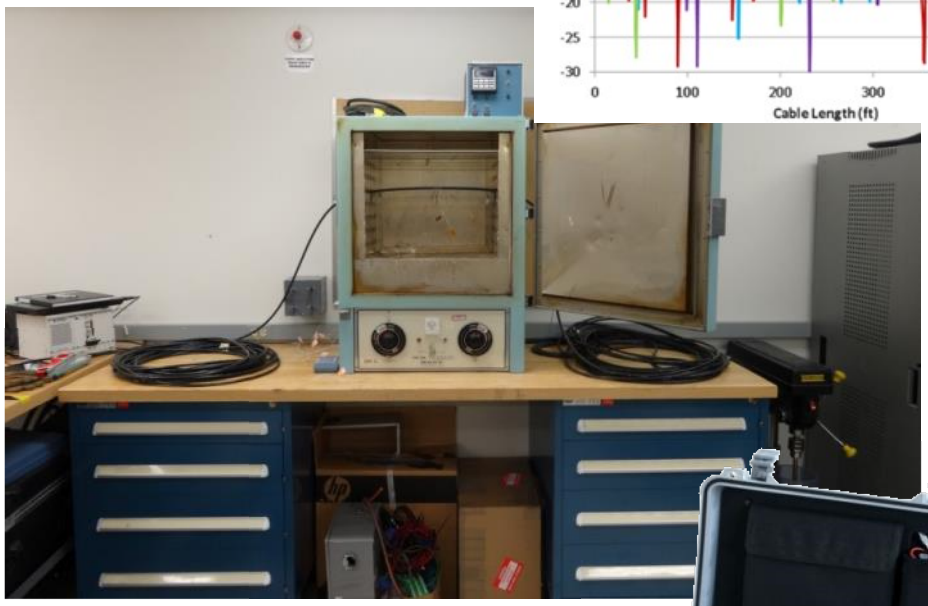
Pacific Northwest NATIONAL LABORATORY

Long Length Cable Measurements

- Frequency Domain Reflectometry
- Dissipation Factor (tan δ)
- High Pot
- Partial Discharge

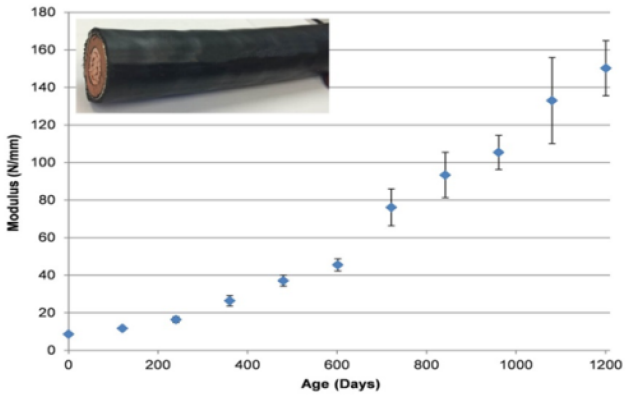


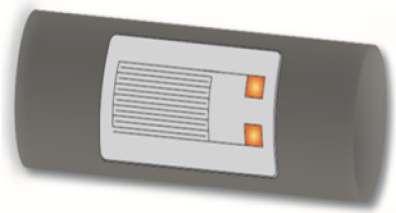


Credit: WireScan

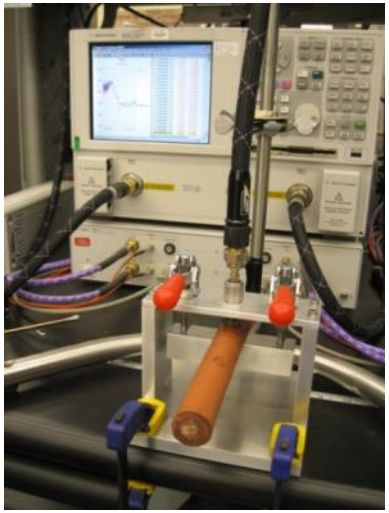
Local Spot Measurements

- Indenter
- Capacitance
- Acoustic
- Dielectric Constant

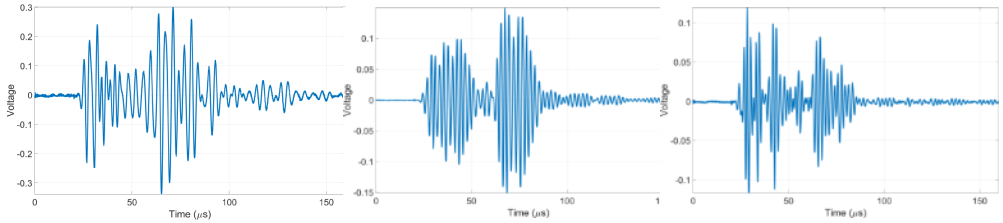


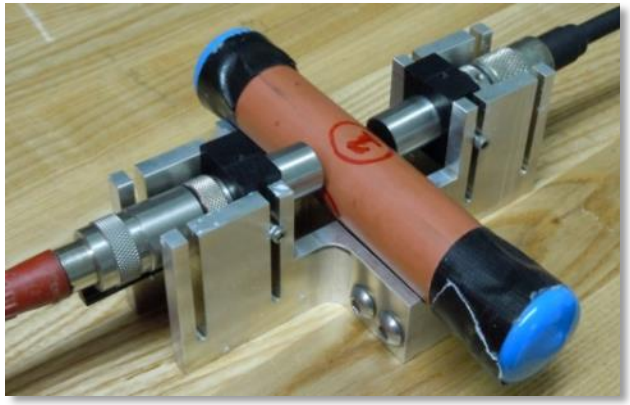


Credit: Bowler Group, Iowa State University



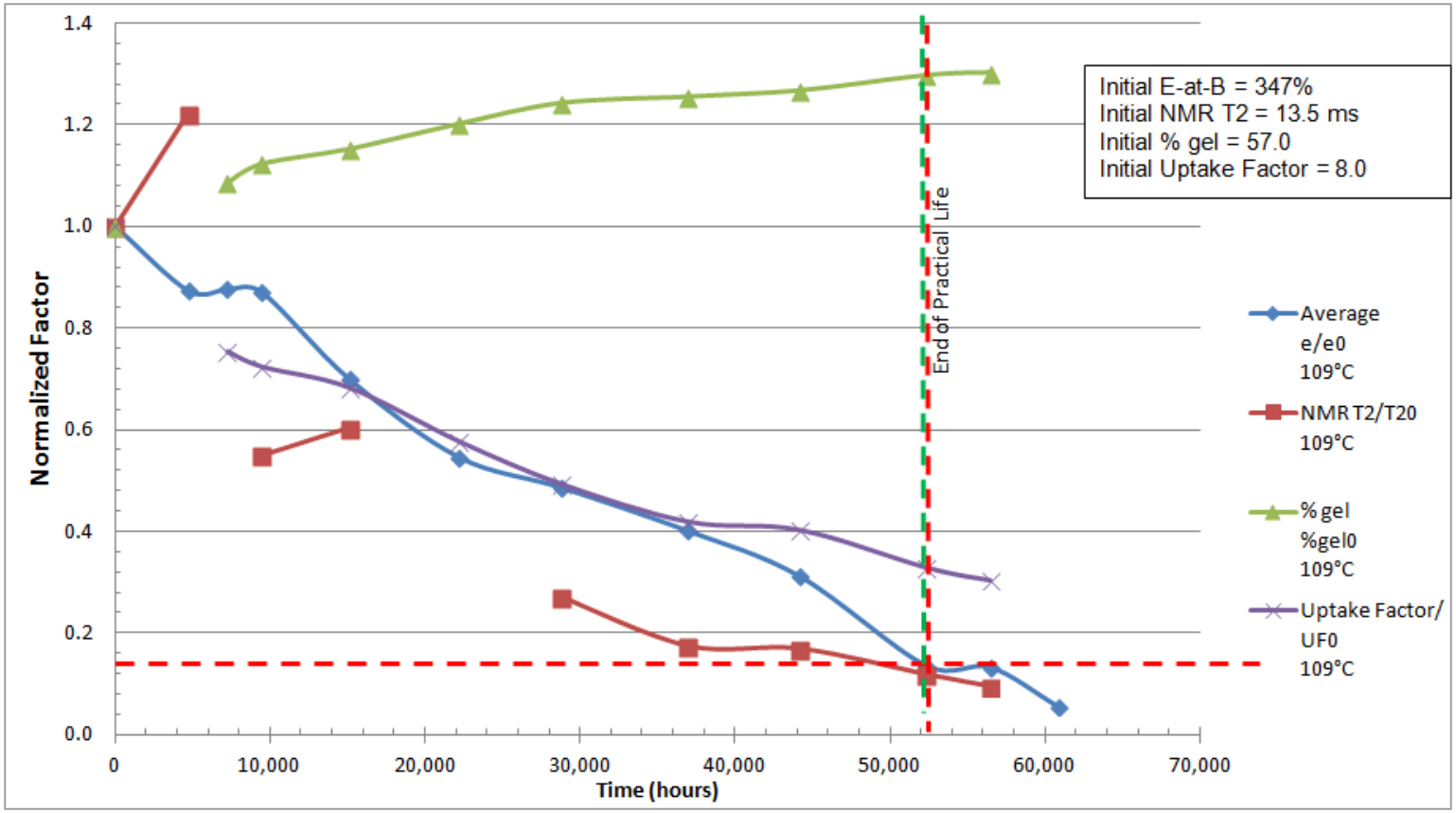
250 kHz 500 kHz 750 kHz





Condition Monitoring: Use of Aging Models to Determining Qualified Condition

- Through correlating cable history with NDE and other examination methods, a better picture of cable aging and remaining useful life begins to emerge.



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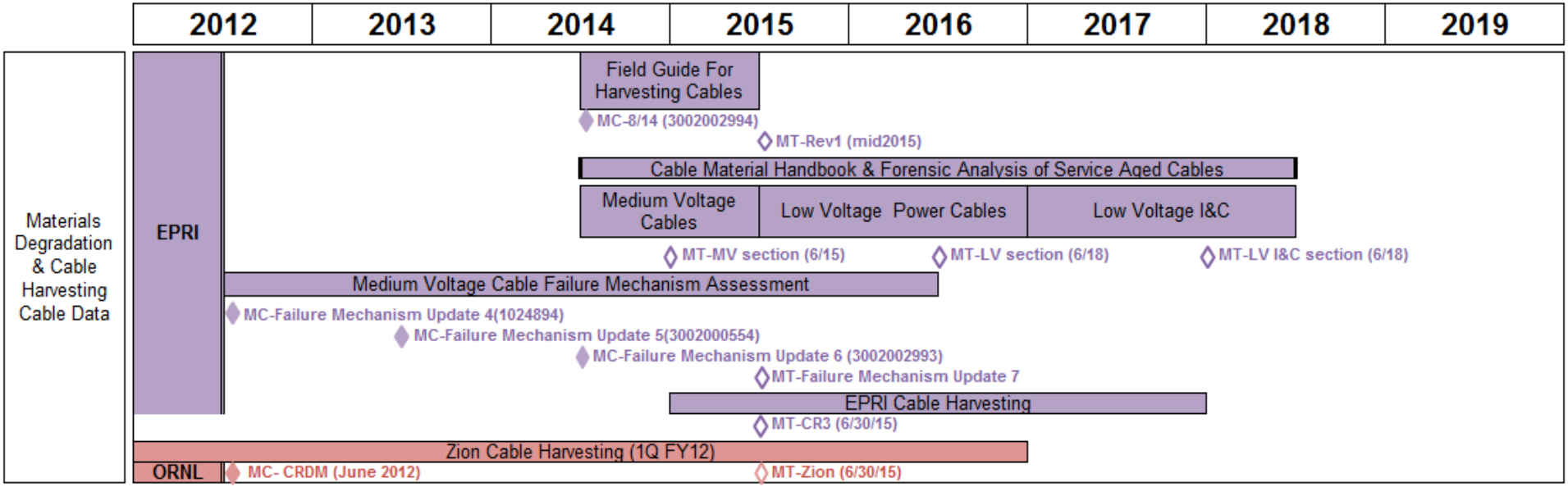
Materials Degradation & Cable Harvesting Cable Data



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Harvesting of Service Aged Cables: Zion Unit 1 and 2

- LWRS and NRC have requested sets of low voltage cables from the Zion Station Units 1 and 2 for harvesting -
- Following EMDA recommendations for service aged cables.
- **Current Work**: Six 25 to 30-ft lengths of Control Rod Drive Mechanism power and position indicator cables, harvested from Unit 1 in 2012 . Testing of the NRC cables has begun at the National Institute of Standards and Technology.
- **Spring 2016**: Six sets of 20 to 60-ft length cables to be harvested from Zion Unit 2.
- Cable exposure conditions of interest:
  - Thermal : steam tunnels
  - Thermal / radiation : in-containment cables
  - Benign environment (cable spreading room) : baseline comparison

Cable Program Summary

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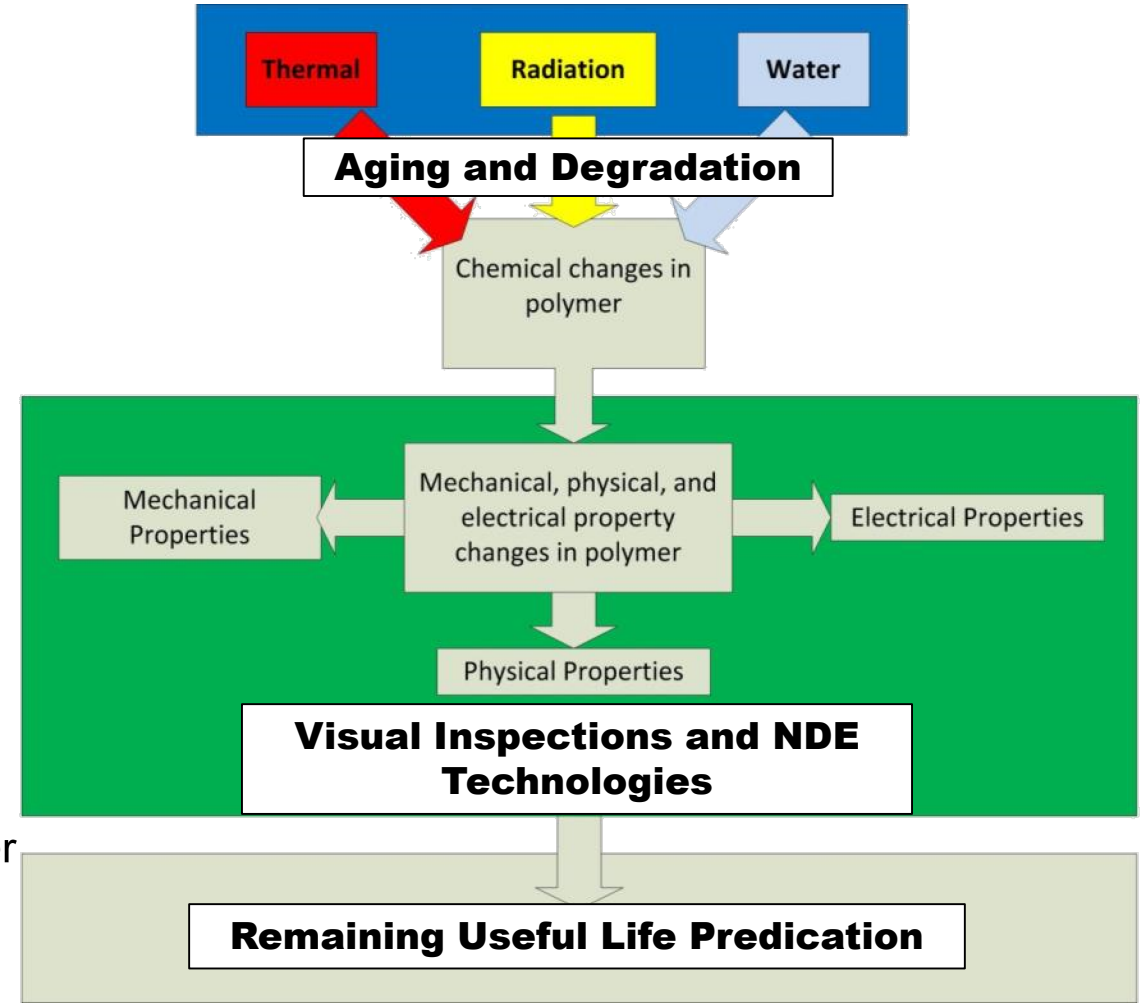
Cable Stressors

Chemical Changes

Changes in Properties

Changes in Performance over Time



Detailed Understanding

Effective Treatments

Key Indicators of Cable Aging

Transformational NDE

Methods for Life Prediction

**LWRS NDE R&D Roadmap 2012* PNNL-21731

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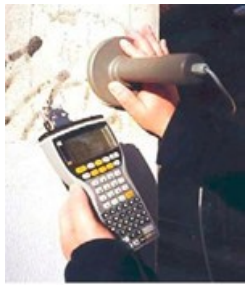
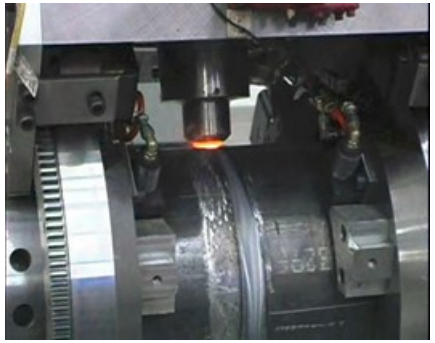
U.S.NRC United States Nuclear Regulatory Commission Protecting People and the Environment

Conclusion from the Research

- ✓ Technical basis is established and in use for aging management
- ✓ Continuous improvements based on research results, inspections and operating experience
- ✓ Coordination and collaboration with global research partners





No show stoppers have been identified based on R&D to date.

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Questions

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Existing Nuclear Sector- Medium Voltage Cable Research

Research Area	Subject	EPRI Report
Cable Aging Management	Aging management program guidance and harvesting guide	3002000557, 102107, 3002002994
Condition Monitoring-	Test applicability matrix: Tan Delta test evaluations:	1022969 1025262, 3002005321
Stressor/Degradation Modes-MV Cable Failure Mechanism Research	Evaluation of wet aging of various MV cable types	1018777,1021069, 1022965, 1024894, 3002000554, 3002002993, 3002005323
Mitigation- Life cycle management	End of life guide for MV cable and accessories Black EPR Rejuvenation	1025259 3002000551
Operating Environments and Improved Condition Monitoring-Cable LTO Research	In-containment radiation and temperature data, Kema On-Line PD measurement	3002000816, 3002003001

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Existing Nuclear Sector- Low Voltage Cable Research

Research Area	Subject	EPRI Report
Cable Aging Management	Aging management program guidance and harvesting guide	1020804, 1021629, 3002002994
Condition Monitoring-Cable Testing	Test applicability matrix: LIRA testing:	1022969 1015209
Stressors and Degradation Modes and Condition Monitoring-LV Cable Failure Mechanism Research	Adverse environment, indenter, visual tactile, acceptance criteria, polymer condition monitoring date (CPAD, NEPO)	TR-109619, TR-104075, 1001391, 1008211, 1011874, 1011873
All-Training	Low voltage training course, CBT on visual tactile, ALEE	Held every July in CLT. 1022979
Operating Environments - Cable LTO Research	In-containment radiation and temperature data	3002000816 (phase I), Phase II in progress

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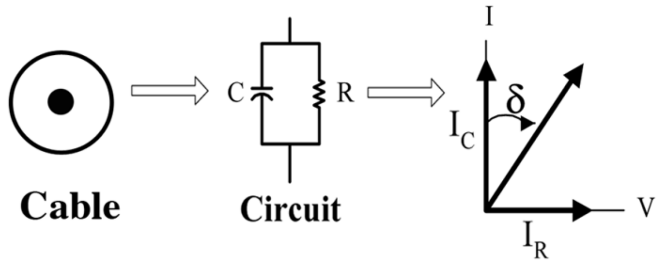
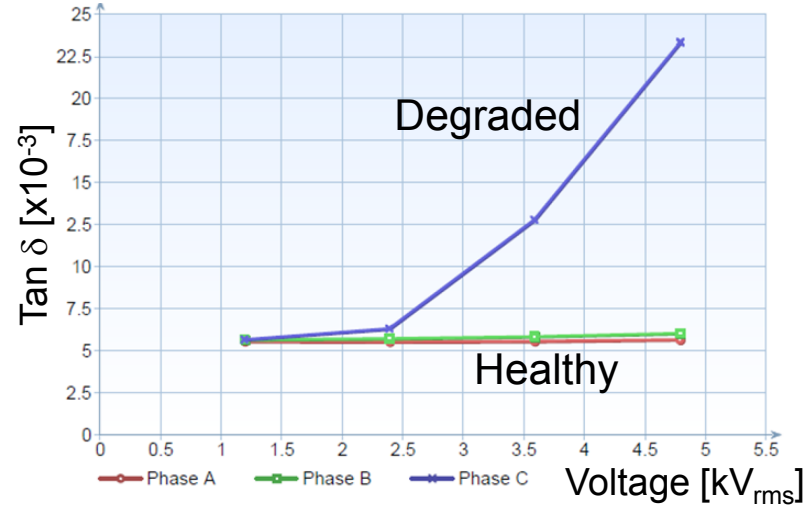
NDE Tools: Effective in Detecting Degradation

Tan δ: loss angle / dissipation factor

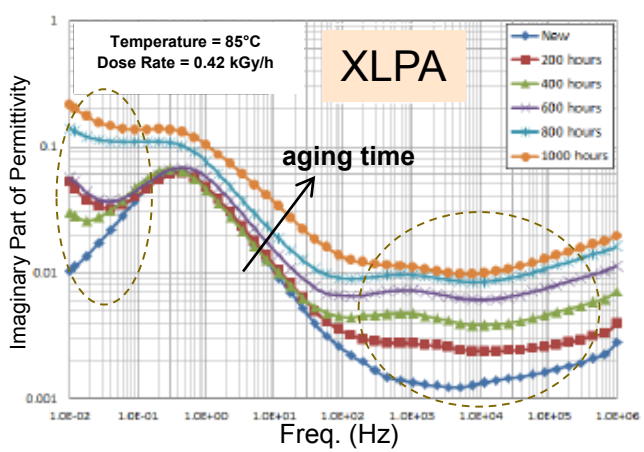


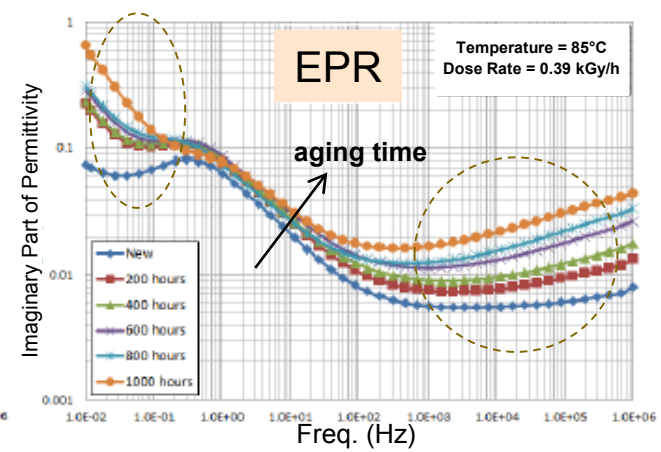


- Healthy cable: I_c is large and Tan δ is small.
- Defects change the capacitive/resistive nature of the insulation resulting in increased Tan δ values.

Low Voltage Dissipation Factor/Spectroscopy

- Advance results indicate potential for aging assessment (thermal, radiation) for any multi-conductor or shielded cable - *adaptable for different insulation materials*.





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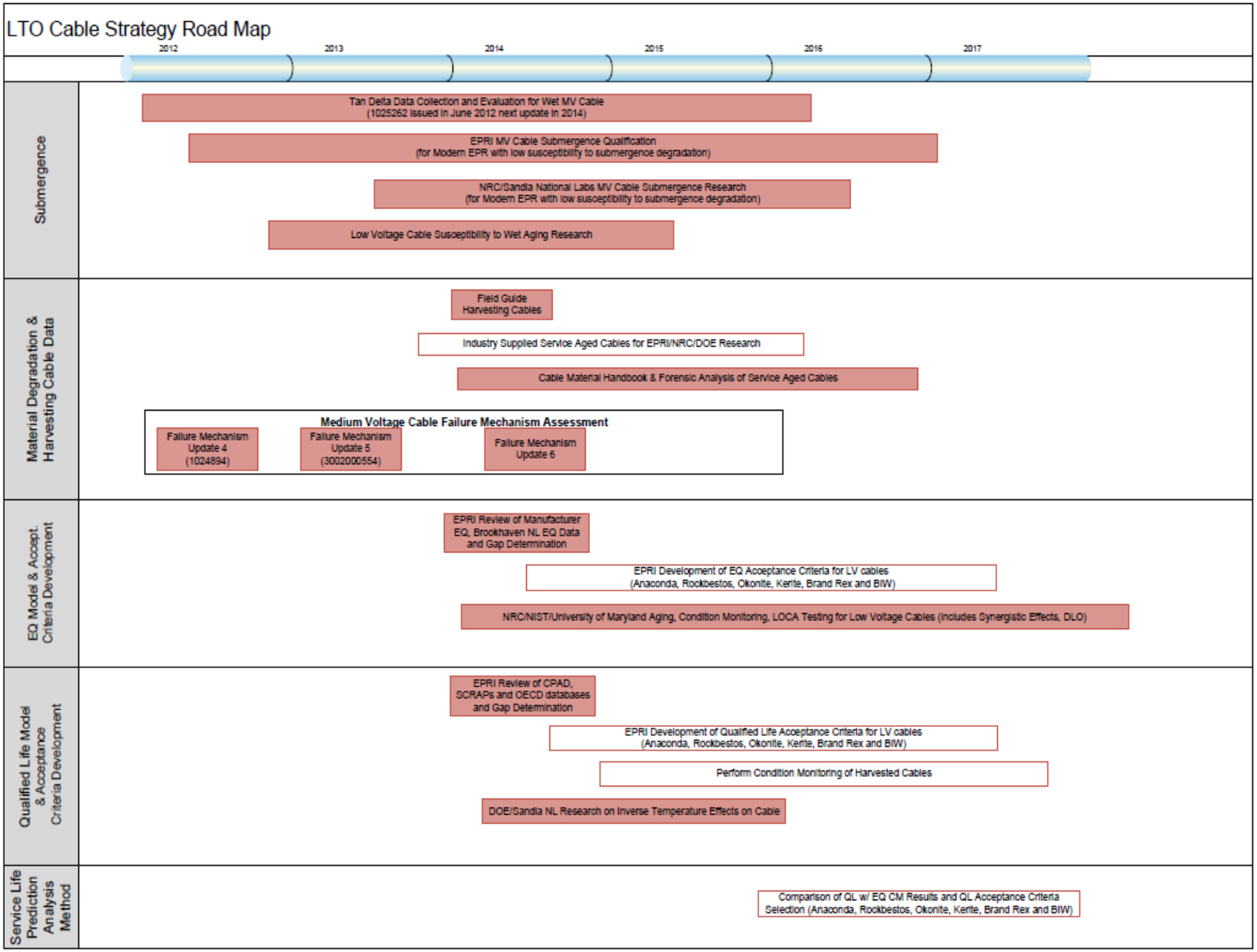
Continued Cooperation on Joint R/D Roadmaps Developed for Key Research Areas



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U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Electrical Cable Qualification and Condition Assessment for Subsequent License Renewal (SLR)

Darrell Murdock
Office of Nuclear Regulatory Research
Division of Engineering

Sheila Ray, Clifford Doutt and
Mohammad Sadollah
Office of Nuclear Reactor Regulation
Division of Engineering
Division of License Renewal

November 17, 2015

Outline

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Protecting People and the Environment

- Regulatory & Technical Framework
- Technical Issues
- Research Efforts to Address Technical Issues
- Status of Work
- Schedule and Milestones

Regulatory Framework

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- **Regulatory requirements for electrical cable condition assessment include:**
  - 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants”
  - 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”
  - 10 CFR Part 50, Appendix A, “General Design Criteria For Nuclear Power Plants”
  - 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants”
  - 10 CFR Part 54, “Requirement for the Renewal of Operating Licenses for Nuclear Power Plants”

Regulatory Framework

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- **Applicable Regulatory Guidance:**
  - Regulatory Guide (RG) 1.89, “Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants”
  - RG 1.211, “Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants”
  - RG 1.218, “Condition Monitoring Techniques For Electric Cables In Nuclear Power Plants”

Technical Issues

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- **Expanded Materials Degradation Assessment Volume 5: Aging of Cables and Cable Systems**
  - Examined cable degradation based on the following criteria:
    - Technical and Operating Experience Knowledge Levels
    - Expected Severity of the Degradation
    - Likelihood of Occurrence
  - Considers technical issues up to 80 years of operation
  - Provides the basis for a number of technical issues to follow

Technical Issues

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- **Cable Environmental Qualification for SLR:**
  - Confirmation or validation (through testing) that electrical cables environmental qualification will be maintained through the SLR period
- **Condition Monitoring:**
  - Development and validation of criteria for cable condition monitoring methods and techniques for the SLR period to perform service life prediction of environmentally qualified and non-environmentally qualified cables

Technical Issues

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- **Diffusion Limited Oxidation:**
  - Takes place when aging simulations employ highly accelerated aging conditions such as very high radiation dose rates and/or very high aging temperatures
  - Dissolved oxygen concentration drops often to zero, significantly reducing or completely eliminating oxidation reactions in the interior parts of materials

Technical Issues

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- **Activation Energy:**
  - Many materials exhibit non-Arrhenius behavior
  - Arrhenius activation energy determined under short-term high temperature aging conditions drops to a lower value as the aging temperature is reduced to conditions closer to the ambient plant conditions
  - Predictions made from extrapolating high temperature activation energy values can significantly over-estimate the lifetime of materials at lower temperatures

Technical Issues

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- **Synergistic Effects between Thermal and Radiation Aging:**
  - In certain polymeric materials age related degradation is most severe in concurrent aging than in sequential aging
  - Qualification testing is conducted under the assumption that thermal-degradation and radiation-degradation are additive and that there are no synergistic effects
  - The existence of synergistic effects can be determined by performing concurrent radiation- and thermal-aging

Technical Issues

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- **Inverse Temperature:**
  - Some EPR and XLPE/XLPO (semi-crystalline) materials experience a higher degradation rate at lower temperatures than they would at higher temperatures when the aging dose rate is constant
  - Inverse temperature effects are observed typically at a temperature range of 50°C (122°F) to 60°C (140°F), depending on the material
- **Dose Rate Effect:**
  - Age related degradation at lower dose rates could be significantly higher than the degradation that occurs for the same total dose at a higher dose rate

Technical Issues

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- **Cable Degradation due to Submergence:**
  - Research is needed to systematically understand the stressors that come into play in submerged environments and their synergies
  - Need to develop consistent end-of-life criteria
  - Advances in condition monitoring are needed to determine a parameter that could provide a better indication of the cable as it degrades in submerged environments

NRC Research

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- NRC issued User Need Request "Assessment of Electrical Cable Condition Monitoring" (NRR-2011-014)
  - **Assessment of Condition Monitoring Methods:**
    - New and naturally aged cable samples will be aged to simulate 80 years of operation
    - Condition monitoring techniques are going to be evaluated during the aging process
    - Aging for this project is going to be conducted synergistically at low dose and low temperature
  - **Evaluation of Cable Degradation in a Submerged Environment:**
    - Perform a literature search of past projects in the realm of submerged cables

NRC Research Status

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- **Assessment of Condition Monitoring Methods:**
  - NRC contracted the National Institute of Standards and Technology (NIST) to carry-out the project
  - NIST will subcontract the radiation exposure portion of the project to a vendor with the required technical expertise and radiation facilities
    - The subcontractor will perform the chambers functionality tests
    - Thermal and radiation exposure should commence once the functionality tests are performed

- **Evaluation of Cable Degradation in a Submerged Environment:**
  - The NRC will focus on evaluating the Electric Power Research Institute criteria for using tan delta to perform aging management

Schedule & Milestones

U.S.NRC
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Assessment of Condition Monitoring Methods	
Task	**Timeframe**
Carry out chambers functionality test	February 2016
Harvest cables from Zion and Crystal River	March 2016
Execute 24 month thermal/radiation exposure	May 2018
Subject samples to a simulated loss-of-coolant-accident test	October 2018
Write NUREG report summarizing the findings • Update applicable regulatory guidance	December 2018

Evaluation of Cable Degradation in a Submerged Environment	
Task	**Timeframe**
Evaluation of EPRI's criteria for submerged cables	April 2016

Conclusion

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- There are a number of challenging technical issues that need to be addressed for the SLR period

- The NRC staff believes that continued engagement between NRC and industry in matters related to research efforts to address the technical challenges associated with SLR is critical to achieve a satisfactory solution

U.S.NRC

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Questions?