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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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694TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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WEDNESDAY

APRIL 6, 2022

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The Advisory Committee met via teleconference, at 8:30 a.m., Joy L. Rempe, Chairman, presiding.

COMMITTEE MEMBERS:

JOY L. REMPE, Chairman

WALTER L. KIRCHNER, Vice Chairman

DAVID A. PETTI, Member-at-Large

RONALD G. BALLINGER, Member

VICKI M. BIER, Member

CHARLES H. BROWN, JR., Member

VESNA B. DIMITRIJEVIC, Member

GREGORY H. HALNON, Member

JOSE A. MARCH-LEUBA, Member

MATTHEW W. SUNSERI, Chairman

DESIGNATED FEDERAL OFFICIAL:

CHRISTOPHER BROWN

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P R O C E E D I N G S

8:30 a.m.

CHAIRMAN REMPE: Good morning. This meeting will come to order. This is the first day of the 694th meeting of the Advisory Committee on Reactor Safeguards. I'm Joy Rempe, the Chair of the ACRS.

Members in attendance are Ron Ballinger, Vicki Bier. Charles Brown will be here soon; he's been a little bit delayed by traffic. Vesna Dimitrijevic, Greg Halnon, Walt Kirchner, Jose March-Leuba, Dave Petti, and Matthew Sunseri. I note we do have a quorum today.

Today, the Committee is primarily meeting in person with some of the ACRS staff, NRC staff, and participants attending virtually.

The ACRS was established by the Atomic Energy Act and is governed by the Federal Advisory Committee Act. The ACRS section of the U.S. NRC public website provides information about the history of this Committee and documents, such as our Charter, Bylaws, Federal Register notices for meetings, Letter Reports, and transcripts of all full and subcommittee meetings, including all slides presented at meetings.

The Committee provides advice on safety matters to the Commission through its publicly-

available Letter Reports.

The Federal Register notice announcing this meeting was published on March 15th, 2022. This announcement provided a meeting agenda, as well as instructions for interested parties to submit written documents or to request opportunities to address the Committee. The Designated Federal Officer for this meeting is Mr. Christopher Brown.

A communications channel has been opened to allow members of the public to monitor the open portions of the meeting. The ACRS is now inviting members of the public to use the MS Teams link to view slides and other discussion materials during these open sessions.

The MS Teams link information was placed in The Federal Register notice and agenda on the ACRS public website. If you are a member of the public who does not yet have the link, please email Lawrence Burkhart at lawrence.burkhart@nrc.gov. Again, that's lawrence.burkhart@nrc.gov.

It's my understanding we have received no written comments of requests to make oral statements from members of the public regarding today's session.

Periodically, the meeting will be open to accept comments from participants listening to our

meetings. The written comments may be forwarded to Mr. Christopher Brown, the Designated Federal Officer.

During today's meeting, the Committee will consider the following topics: Radiation embrittlement of reactor vessel materials; BWRX-300 Topical Report on Containment Evaluation Methodology, and the Point Beach subsequent license renewal application. As stated in the agenda, portions of the BWRX-300 Topical Report may be closed.

A transcript of the open portions of the meeting is being kept, and it's requested that speakers identify themselves and speak with sufficient clarity and volume, so they can be readily heard. Additionally, participants should mute themselves when not speaking.

At this time, I'd like to ask other members if they have any opening remarks.

Hearing none, I'd like to ask Ron Ballinger to lead us through our first topic of today's meeting. Ron?

MEMBER BALLINGER: Thank you, Chairman Rempe. Today, we will have a discussion related to, I guess I would call it the evolution of pressure vessel embrittlement monitoring and prediction, and the staff's work in this area and their proposals for

a rulemaking to update, if you will, the process.

And we'll have presentations from Dave Rudland, who's sitting in the front of us. And if you would like, we will have him put his mask on, so we'll be able to identify him.

(Laughter.)

MEMBER BALLINGER: And Elliot Long from EPRI. So, with that, unless there are comments from the members, Dave, it's all yours.

MR. RUDLAND: Thank you. Thank you.

First of all, I just want to say thanks to the ACRS for allowing me to come here to talk about these issues and the rulemaking plan that we put together to address those.

And before we get started, I just wanted to acknowledge the team that put this work together. This has been an ongoing issue that we've been working on, and it's been a cross-office issue that we've been working on with NRR and Research, as well as NMSS, to solve this problem. So, I wanted to just acknowledge them and thank them for that.

And again, I want to thank you for having this in person. I think is the first time in two years that I've worn a tie. So, I'm glad to be able to do that and to see all your faces in person.

So, let me get started. I'm going to talk about some issues today with reactor pressure vessel embrittlement. We're going to focus a little bit on the embrittlement trend curve that is in Regulatory Guide 1.99, Rev 2, as well as the same trend curve that's in 10 CFR 50.61. I'm also going to talk about some issues with Appendix H surveillance testing. And then I'll move into the staff's thoughts on how to address those issues through rulemaking, as part of a rulemaking plan that was delivered to the Commission in March.

As a bit of a background, embrittlement trend curves provide an estimate of the change in fracture toughness. It's a regulatory requirement that the licensees monitor the reactor pressure vessel toughness. And so, we have embrittlement trend curves that allow the right estimates for change in fracture of toughness as a function of fluence.

In this illustration that's on the bottom left, this red curve represents a schematic of one of those, a prediction using that embrittlement trend curve. And as you can see, the embrittlement increases as a function of time and order of fluence. And an embrittlement trend curve is just a formulation, a curve fit to data that, in this

particular case, this is a function of material chemistry and the fluence level.

In addition, we have requirements for surveillance capsule testing that provides monitoring to ensure that this generic embrittlement trend curve predicts the plant-specific behavior properly, so that the data points on this schematic represent the data capsules, materials that are pulled from these capsules that are inside the plants and tested to make a prediction of fluence. And again, embrittlement is a function of these.

We then add a margin to that term, once we have that. An embrittlement trend curve can be adjusted to try to better represent the plant-specific behavior. And then we add a margin to that, which then gives us what's called the adjusted reference temperature, which is then used in 10 CFR 50, Appendix G, and 50.61, to give us both the pressure temperature limits for a normal heatup and cooldown, as well as screening criteria for PTS.

So, the ideal scenario is to have the embrittlement trend curve, a margin that provides a conservative prediction of embrittlement, and surveillance data that covers all operating periods, that allows us to verify that the plant-specific

behaviors are matching the trends that are provided in the embrittlement trend curve.

However, if we have a trend curve behavior that underpredicts the data, we can have some uncertainty in what the real embrittlement is, and/or if we don't have high fluence data, you can also have an increased amount of uncertainty. And those increased amounts of uncertainty can challenge the confidence the staff has in ensuring that the plants are safe from embrittlement.

CHAIRMAN REMPE: This is Chairman Rempe. And I don't see any other mics in the room on except David's and the ceiling, and I'm not hearing any other noise.

MR. RUDLAND: I will move the microphone a little closer. I don't know if that will help or not. Okay, sure.

So, the staff's current perspective on these issues -- before I go to the issues, I'm going to tell you what our perspective is. It's not as if I'm going to tell you the end before I tell you the story.

The staff has high confidence that that the current operating plant remain safe; that the decisions that we've made in terms of embrittlement

and reactor pressure vessel safety still remain valid. However, there is insufficient embrittlement monitoring and underprediction of that embrittlement that is in the current trend curves and in the current regulations that will eventually impact the staff's confidence in the integrity of the reactor pressure vessel in long-term operation.

And by long term, you know, the first time this may become an issue is if it was within about 10 years for pressurized thermal shock and much later, about 20 years, for pressure temperature limits, due to the expected fluence levels of the current operating fleet. And as I'll show in a minute, both the safety margins to brittle fracture and performance monitoring will be impacted.

At this point again, like I've mentioned, we've written a rule plan to the Commission, but further work is really needed to determine which plants are impacted by this potential issue, and I'll talk about that.

MEMBER HALNON: Dave, this is Greg Halnon. I'll know you'll probably get into this when you say "a story," but the high confidence on the first bullet is based on actual data trending with actual models and other --

MR. RUDLAND: That's correct. And where the plants are right now in terms of (audio interference).

MEMBER HALNON: Okay. So, when you say it's going to be insufficient, that embrittlement monitoring is down the road --

MR. RUDLAND: It's in the future.

MEMBER HALNON: -- it's past a certain fluence level?

MR. RUDLAND: That's correct.

MEMBER HALNON: Okay. Thanks.

MR. RUDLAND: And I'll show some examples of that. Let me start with the embrittlement trend curves. The embrittlement trend curve that's currently in Reg Guide 1.99, Rev 2, was published in 1988. It's a fit to the data that was available at the time, which is about 177 surveillance data points. Again, it's a function of the chemistry of the material and the fluence level. That particular trend curve, then, in 1991, was also included in 10 CFR 50.61 also. Recently, that embrittlement trend curve was reevaluated for continued adequacy in 2014 and more detail in 2019, leading to today's effort.

So, this is an illustration of the issue. In this plot, the vertical axis represents the

difference between the embrittlement predicted by Reg Guide 1.99 or 50.61 and the measured values from surveillance data. So, a zero on the vertical axis represents a perfect prediction. There is scatter in the data. The solid-dashed lines represent the standard deviation that's in the Regulatory Guide. The horizontal axis is the fluence.

As you can see, the predictions are very good until you get to a fluence of about 3 times 10 to the 19th neutrons per centimeter squared. And then it begins to deviate. And by the time you get to about 6 times 10 to the 19th neutrons per centimeter squared, it become statistically significant, that there is an underprediction. You end up with a negative value on this chart, meaning that the measured values are higher than the predicted values. And what this 180 degrees means in terms of impact, we'll talk about that in a few minutes.

The other issue is with surveillance capsules. Appendix H to Part 50 requires periodic monitoring. And it incorporates by reference an ASTM standard that's a 1982 standard that was originally designed for a 40-year life, but is not very specific in what the lifetime needs to be. Typically, these programs were set up with four or five capsules within

the reactor, such that, by the time they pulled the second-to-the-last capsule, they had enough data to give them the embrittlement data for a 40-year life.

So, the last capsule was always allowed to be tested at two times the design fluence. This also allowed the last capsule to be held without testing -- just a byproduct of the 1982 standard.

In 1997, the Commission, through some work that was happening at the Perry Plant, came with a finding that, decided that, if there was going to be a change to the surveillance capsule withdrawal schedule, that if the plant wanted to make a change to the capsule withdrawal schedule, that only a verification check needed to be done verifying that it was in conformance with the ASTM spec, but not based on technical or safety considerations.

And because of those two things, that allowed some plants to repeatedly delay their last capsule to acquire higher and higher fluences, as they applied for license renewal and/or subsequent license renewal. And here's an example of that.

This particular plant, again, on the vertical axis is neutron fluence; on the horizontal axis is the date on which a surveillance capsule was pulled. So, in this case, there was four out of five

capsules were pulled. And you can see the last capsule was pulled around in the early 2000s. The fifth capsule was supposed to be pulled later that decade. However, it was delayed multiple times to try to reach an 80-year fluence, as this particular plant applied for subsequent license renewal. And so, this plant has no data at this particular point in this high fluence area.

Many licensees have done this, and here's some examples here on this plot that show plants that have delayed this last capsule pull, but not all have. Some have been following and periodically pulling the capsules.

So, what's the impact of that? This shows the same data that I just talked about, embrittlement on the vertical axis; fluence on the horizontal axis. The four data points that I talked about here in this plot here; you see four data. If a draw a trend curve from Reg Guide 1.99 using just the chemistry of the material, I would get this particular curve.

Like I mentioned earlier, you can adjust the trend curve to match, to best fit your data. So, the blue represents the best fit through those four data points.

Again, there is no data of high fluence

for this plant. If I assume I have high fluence data that follows the trend on the prior slide, I would have data that's way up here. I could have a 150-degree underprediction of embrittlement of that particular data.

Again, I can fit that data. If I fit all of those data points, I get the yellow curve, but I still have a 75-degree underprediction of embrittlement. And that's because of the functional form of the embrittlement trend curve. If you look at the functional form, again, it was fit to only about 170 data points, to a limited number of fluence, limited level of fluence. When you get beyond that, and you try to extrapolate that trend, it actually reaches a peak, and then decreases, which is causing this behavior that you see.

MEMBER BALLINGER: This is Ron. What you're saying is you can't extrapolate?

MR. RUDLAND: You can't extrapolate.

MEMBER BALLINGER: It's just an empirical fit --

MR. RUDLAND: That's right. That's correct.

MEMBER BALLINGER: If you could put enough dimensions on the fit, you can make it do anything you

want?

MR. RUDLAND: That's right. That's right.

So, again, this particular plot was just for the data that I showed prior where they had no data. This is a recent plant data that we just got surveillance data for. The same plot. The same things. And you can see they pulled their fifth data point and it came way up here. Relative to where they are in terms of life, their 80-year fluence should be here. So, this data point is actually greater than their 80-year fluence. But, again, the point is you can see that, even with trying to fit that data, I still could have up to 60-degree underprediction in embrittlement.

So, what's the safety case on this? The staff did a very robust, risk-informed analysis where we did a series of analyses using probabilistic fracture mechanics, looking at a variety of different scenarios, with the amount of data that we had. Again, we had to try to use whatever we had, and we didn't have enough to do individual plant-specific analyses. So, we tried to do something that we thought was maybe bounding.

We also looked at the other tenets of risk-informed decisionmaking. We looked at safety

margins. We looked at performance monitoring. We looked at difference in depth. So, I wanted to talk about a few of those today.

So, again, the conditional probability of failure that we calculated may increase by several orders of magnitude, and I'll show that, an example of that in a second. But, again, because that's a conditional probability, the expected transient of the -- the transient frequencies that we assumed in the analyses are probably low. It's unknown, but they're probably low.

In some cases, the plant may exceed their pressurized thermal shock screening limit if you correct this underprediction. But, again, the analysis suggests that the risk is low, but a lot of uncertainty exists and a lot of plant-specific details are unknown.

So, the staff also focused on looking at safety margins and performance monitoring. And these predictions, when we looked at safety margins, the underprediction, as well as the increase in uncertainty, really impacts the safety margins. And I'll show an illustration of that in a second also.

And as I mentioned, if you're not testing the material, and then you have performance monitoring

issues delaying these capsules and not testing them at the higher fluences, it does not provide adequate performance monitoring to ensure that the trends are reasonable.

So, some examples of the risk. This particular plot, again, is conditional probability of failure for a variety of different flaw sizes. The horizontal axis is, basically, this underprediction, how much underprediction that there is. If I have a 50-degree underprediction, I could have about two-three orders of magnitude change in the conditional probability. If I have 150 degrees, I can up to six, five-to-six orders of magnitude change in conditional probability.

So, that seems really large, but, again, the transients that are used in this particular case are very low. That drives the overall probabilities, or through all crack frequencies, to be relatively low. But we don't know exactly what those frequencies are. We don't know what the actual plant fluence variations are, and we don't know whether these analyses are really bounding. So, we didn't want to base our case just on the risk of failure or pressure temperature limits.

For pressurized normal shock, the analyses

were very similar. Again, if we correct for the underprediction, it could be that some of these plants may go over their screening criteria, which is 270 for plates, forgings, and welds, axial welds, and 300 for circumferential welds. But we took that into account on our analyses and, again, calculated the through-wall crack frequencies for that. Most of those were relatively low, suggesting that, even if they passed the screening criteria, it's not a huge risk problem.

Now we looked at safety margins. The uncertainties in the risk calculations are high, like I mentioned, and increasing with time. There's a lot of unknowns also. So, we wanted to try to take a look at what the fundamental safety principles were in terms of the basis for design and operation.

And again, the margins that I'll be talking about really provided the reasonable assurance against brittle fracturing, and I think those are most impacted.

I can show that through this simple illustration. This is a pressure temperature plot. Pressure on the vertical axis; temperature on the horizontal axis. The operating window is a typical operating window for a plant that may be cooling down from a higher temperature or higher pressure. They

stay within this window until they get to a low-pressure condition.

There is a structural limit out there that can be calculated or estimated. If the plant cools down faster than it decreases the pressure, it could cross that limit and increase, significantly increase the risk of embrittlement fracture.

So, what we do in the regulations is we have an offset that has a particular margin on it, an adequate margin, so that we regulate, then, this PT limit. So that, when they're cooling down, they need to stay to the right of these PT limits. And those margins are dictated by the amount of uncertainty that's in the analysis to calculate the structural limit. So, more margin is needed for greater uncertainty.

MEMBER BALLINGER: This is Ron. This typical operating -- is that correct? I mean I've operated a plant, and in that gray area down at low temperature, that's a pump curve. And so, you go into that; you can cavitate the pump. So, you're not in the typical operating window here.

MR. RUDLAND: Well, this is the typical -- yes, the typical, this is the typical cooldown. I guess maybe that's a better terminology.

MEMBER BALLINGER: Because you're actually --

MR. RUDLAND: A typical heating cooldown.

MEMBER BALLINGER: You operate actually between the gray area and what now would be the orange line. Because there's a box in the upper right which is the actual operating regime for the plant, right?

MR. RUDLAND: You know, the operating regime is up in this region. The plants have administrative controls --

MEMBER BALLINGER: Yes.

MR. RUDLAND: -- to keep them away from --

MEMBER BALLINGER: Yes.

MR. RUDLAND: -- pressure temperature limits. That's correct.

But there's no regulatory control over those operations. So, what I'm talking about here is just the regulatory control that we have over that.

With this underprediction --

VICE CHAIRMAN KIRCHNER: Isn't it, though, Dave, true that the tech specs, you operate within that, quote-unquote, "operating window," and that provides regulatory control because you approve those tech specs --

MR. RUDLAND: That's right. One of the

tech specs are these pressure temperature limits.

VICE CHAIRMAN KIRCHNER: Yes, exactly. So, there a regulatory basis for stating within it, even if it's not prescribed --

MR. RUDLAND: Right, but --

VICE CHAIRMAN KIRCHNER: -- in 10 CFR --

MR. RUDLAND: Yes, but the pressure-temperature curve is a tech spec limit.

VICE CHAIRMAN KIRCHNER: Yes.

MR. RUDLAND: And the fact is that it's not in the right place because of this underprediction. So, then, that's kind of the point I'm trying to make here, is that, you know, for the proper embrittlement prediction, this orange curve is where PT curve should be, but because of the underprediction, it's actually in this red area. And so, we actually have a reduced margin to structural failure because the plants believe that they're following -- the plants are then instructed to follow this PT limit that's kind of the reddish color here because of that underprediction.

Then they can actually, you know, again, if it wasn't for the administrative controls, they could actually cool down in a region that's between these orange and red curves. And that margin is, of

course, reducing, and continues to reduce as the uncertainty gets larger. So, the margin is going down, but the uncertainty is going up. And so that's why we have these --

MEMBER BALLINGER: This is Ron. I'll say it again. That design, the typical operating window should be moved up and to the left, and should be between the gray or bluish-gray area and the orange --

MR. RUDLAND: Yes.

MEMBER BALLINGER: Wait a minute.

MR. RUDLAND: Really, I guess really what the shaded area is, it's the typical cooldown --

MEMBER BALLINGER: Right, right.

MR. RUDLAND: Yes, you're right. You're right. Are there any other questions?

VICE CHAIRMAN KIRCHNER: What is the largest contributor to uncertainty in your opinion, Dave?

MR. RUDLAND: I think the largest contributor is not having enough data at high fluence at this point, yes. I mean, again, as you look at the amount of data points -- I'll send you back just real quick to this plot here.

As you look to this plot, what I didn't mention is that the red data points are U.S.

surveillance data, and the gray data points are international surveillance data.

So, you can see that there is not a great amount of data in this region where the underprediction occurs to really get a feel for what the embrittlement is. So, I know I have an issue with the embrittlement trend curve. That has some uncertainty with it. I don't have any -- I have very limited data at higher fluence, which also increases that uncertainty.

MEMBER BALLINGER: We have limited data for U.S. plants at high fluence.

MR. RUDLAND: That's right.

MEMBER BALLINGER: But that doesn't invalidate the gray data.

MR. RUDLAND: Exactly right. That's exactly right. That's right.

And again, in terms of any individual plant, the largest uncertainty is the lack of -- you know, if they have data or not at the high fluence time. Because that really, then, is a measure of what their actual embrittlement state is. So, if we don't have the actual embrittlement state of any individual reactor, then that uncertainty on whether that reactor is nearing a limit or causing an issue is unknown; the

uncertainty is large.

Okay. So, in quick summary, the current state of knowledge, the generalized analyses that we did, the probabilistic fracture analyses suggests that the overall risk per fracture is low. But those uncertainties that are associated with that analysis are high and increasing with time, especially due to the lack of surveillance data.

And so, safety margins can be impacted and are decreasing as that uncertainty increases. And delaying these capsules and not testing at high fluence represents a lack of sufficient performance monitoring.

I know Elliot will be talking about a little bit more some of the ongoing industry programs to try to increase the amount of data that we have at high fluence. So, he'll be talking about some of that. But, again, those data aren't expected for years.

And these issues that I'm talking about now really are mainly impacted by plants that are, or will be, at higher fluence, which is, in this particular case, about 6 times 10 to the 19th neutrons per centimeters squared.

MEMBER HALNON: Dave, this is Greg. Do we

understand physically what's happening; why it's not following what 1.99 is --

MR. RUDLAND: Yes, I think so. I think Ron pointed it out. It's an empirical fit.

MEMBER HALNON: But, I mean, materialistic, though, materials-wise, what is occurring that's different than the predicted microscale? I mean, what --

MR. RUDLAND: I don't think anything different is happening. I think it's the same behavior that was happening at lower fluence. I just think our predictions don't extrapolate well because we didn't have the data. So, we empirically fitted to the data that we had. And then, when you try to extrapolate that, it didn't work out very --

MEMBER BALLINGER: It's a simple case of overfitting. In other words, you've got X data points, and you do an n-parameter fit for X data points. You can get a pretty darn good correlation within that database. But when you get outside that database, there's no physical -- there's no physics anywhere.

MR. RUDLAND: Yes, yes.

MEMBER BALLINGER: And so, it just goes nuts. That's exactly what it shows if you plot that

curve.

MR. RUDLAND: Yes.

MEMBER BALLINGER: It just goes weird.

MR. RUDLAND: Yes.

MEMBER HALNON: I get the plot and I get the math piece of it, but what's happening at an atomic level that's causing the embrittlement, and does that make sense in the higher fluences? Now that we know that, should we have seen that back when we first started seeing embrittlement, and the materials --

MEMBER BALLINGER: But the trend is actually consistent at higher --

MR. RUDLAND: Yes, yes.

MEMBER BALLINGER: So, it's not going weird.

MR. RUDLAND: Yes.

MEMBER BALLINGER: The correlation is going weird.

MR. RUDLAND: ASTM has taken all of the worldwide embrittlement data and done another fit, basically, using the same types of inputs, a couple of other inputs, but, basically, the same kind of inputs, and are able to predict all that data at the high fluence.

VICE CHAIRMAN KIRCHNER: But another way of asking Greg's question, though, is from a materials standpoint, first of all, it suggests to me that the current curve, as you say, it's just an empirical fit and there's no physics behind it. In other words, it's fit empirically to data, but it doesn't necessarily have -- well, it's a function of fluence, I suppose, but only in the sense that it fit the data. But the empirical -- the correlation doesn't have any physics. That is my point.

MR. RUDLAND: Yes, there's a relationship with copper and nickel, right, because they know those are the things that impact embrittlement.

VICE CHAIRMAN KIRCHNER: Right. Right. But now, we know enough about materials that, isn't there a way, Ron, to build a model --

MR. RUDLAND: A mechanistic model?

VICE CHAIRMAN KIRCHNER: -- a mechanistic model?

MR. RUDLAND: You know, research has been -- I don't know how many years -- 30 years of research; 40 years research has been going on, and trying to be able to mechanistically model embrittlement and the correlation between temperature and fluence, and the material behavior has been a bit

elusive for the researchers.

MEMBER BALLINGER: Yes, and it's just your PhD against my PhD.

(Laughter.)

MEMBER BALLINGER: Well, it's metallurgy. So, that ought to tell you a lot.

VICE CHAIRMAN KIRCHNER: But, like in CHF correlations, there are physical properties in the correlation that make sense.

MEMBER BALLINGER: Somebody gives you 150 data points. They don't tell you where they come from or anything. You go into Excel; you hit "Fit," and, lo and behold, you'll get a fit.

(Laughter.)

MR. RUDLAND: I think it was a little bit more sophisticated than that, but --

(Laughter.)

MEMBER BALLINGER: But, well, it was pretty close to that. But then be careful going outside that range because it just goes crazy.

MR. RUDLAND: And again, I think the ASTM group that I just talked about spent some time trying to look at really what the drivers are as best they could, based on the physics, as we know it. So, there are other parameters in the newer fits that aren't in

the Reg Guide 1.99.

MEMBER BALLINGER: And by the way, NRC staff participated in that analysis. Am I correct?

MR. RUDLAND: Yes. Oh, yes.

MEMBER BALLINGER: So, the NRC staff has been up to their ears --

MR. RUDLAND: And I just saw a comment come in in the chat, which I can't see right now, from Dan Widrevitz, who I know is on the ASTM committee that is working on that stuff. So, we do have staff on it.

MEMBER MARCH-LEUBA: Please don't share technical information in the chat because it is not recorded in the transcript.

MR. RUDLAND: So, Dan, if you wanted to say something, I guess you have the opportunity to do that.

MR. WIDREVITZ: Sure. I just wanted to mention that there were some mechanistic insights incorporated into the varieties of fits. And so, this isn't a totally pure "let's throw an algorithm at our data methodology," even back for Reg Guide 1.99. It's just the issue is the particular methodology and the data that they had at the time are not extrapolating well now. But there's always been a certain level of

mechanistic insight in all of these trend curves.

MEMBER BALLINGER: But, in the end, as soon as you put an adjusting parameter on it, it becomes an empirical fit.

MR. RUDLAND: Agreed.

MEMBER BIER: I could have probably done it, despite knowing nothing about materials.

(Laughter.)

MEMBER BALLINGER: You would have probably done a better job.

MR. RUDLAND: Okay. So, I want to talk a little bit about who's impacted by this. And again, like I mentioned earlier, yes, the embrittlement underprediction is a function of fluence. And it's for the thing that, the term put this in that it's in the regulations and in our guidance. It becomes an issue when you get to this fluence level about 6 times 10 to the 19th neutrons per centimeter squared.

And so, again, it's mainly a PWR issue, because the BWRs will never get to that, at least in the expected 80 years of life for those plants. So, about 34 percent of PWRs may be impacted.

But, again, it's important to point out that a lot of plant-specific details are really needed to determine which, if any, one individual plant would

be affected by this or not. So, the staff needs to do a little bit more work to determine how to focus these on plants that are impacted.

And, of course, anybody who delays their last capsule would be impacted by this lack of surveillance data. And that's true for license renewal and subsequent license renewal.

VICE CHAIRMAN KIRCHNER: Now, Dave, you picked 8 times 10 to the 19th to display. Could you explain --

MR. RUDLAND: Yes, I just --

VICE CHAIRMAN KIRCHNER: -- how you came up with 8 times 10 to the 19th? Because I would think you would go -- I mean, at 6 times 10 to the 19th, you see a divergence. What's so magical about 8 rather than --

MR. RUDLAND: No, there isn't any --

VICE CHAIRMAN KIRCHNER: -- 10 to the 20th?

MR. RUDLAND: Yes, I showed it just for illustration on how fast it's dropping off in terms of the number of plants that are impacted. I'm not sure off the top of my head what the expected highest fluence is of the plants that are coming in, but it is around, you know, 20 to the 20th, right? So, this is

halfway in between those in 6 and in 8 --

MEMBER BALLINGER: Are you going to say something about where this fluence is measured in relation to where the calculations require?

MR. RUDLAND: Well, and I did briefly at the beginning. So, again, this particular number represents an ID fluence. So, this is the fluence that's at the ID of the vessel. So, when you're doing a PTS calculation, you use the fluence value that's at the ID of the vessel. When you're doing a PT calculation, you use the fluence that's at the 1/4T and 3/4T locations.

MEMBER BALLINGER: So, at those, it's much less than that?

MR. RUDLAND: That's right. Since it attenuates through the wall, it's much less at a 1/4T, which is why at the beginning I said it becomes to be a problem about 10 years for PTS and 20-something years for PT.

MEMBER HALNON: Is there a general feeling what fluence level would become so operationally restrictive that they won't be able to operate these plants?

MR. RUDLAND: I think that, as long as we fix things and they change their PT curves and follow

that, you know, that's a plant-specific issue on when they think it's not enough operating space for them to operate in, where they will need to do something (audio interference), or whatever other mitigative techniques they would have.

MEMBER HALNON: Okay, but you don't have a feel --

MR. RUDLAND: I don't. No, I don't have a feel. Again, that's a plant-specific problem.

All right. So, to solve this, the staff came up with some alternatives that we wrote up in our rulemaking plan to change this. And the first alternative is, basically, to do nothing, to make no changes, but to handle this issue on a plant-specific basis, either by orders or by something within their license renewal applications, or by some generic communications. So, that's the first alternative.

I'm going to talk about the third one first, next. The third alternative is a comprehensive solution where we replace the embrittlement trend curves in the regulations, as well as in the guidance, to a new trend curve that is better suited for the expected periods of operation, as well as to include some additional requirements in Appendix H for additional surveillance testing and long-term

operation.

And then the second alternative is a much more focused alternative, where we're only going to make the changes and requirements effective for those plants that may be impacted. So, it's a much more focused solution.

And how we do that is something that staff still needs to figure out, but we figure that -- yes. So, I'll talk about what we decided to do here in a second.

No matter what we do, it's going to be a backfit. So, if we do it on a plant-specific basis, there will be individual backfits. If we do it in terms of rulemaking, it will be one backfit. But, either way, it's going to have to be a fact that we'll have to go through the proper analysis to justify the backfit.

And again, modification of H will require some plants to test a capsule during SLR period, that they maybe weren't planning to, or sometime other, maybe in license renewal at their (audio interference). And then the modification to the trend curve may cause some plants to modify their PT limits and/or PTS calculations.

But, again, for at least alternative two,

we want to make sure that we only focus on those plants that are highly affected.

MEMBER BALLINGER: Is there a likelihood that using a more sophisticated probabilistic analysis, some of this "hit," if you will, could be avoided?

MR. RUDLAND: It's possible. Again, it's not the analysis sophistication that is needed; it's the inputs, right? So, we need plant-specific inputs. That's really what it comes down to.

So, the staff recommended alternative two, which, again, is a focused solution. It's the one that would be the least resource-intense and affect the plants that are highly impacted without impacting those that at this point don't need it. Like I mentioned earlier, the BWRs don't have an underprediction problem, and they won't for 80 years. And so, it's possible that we can work the guidance and regulations to exclude those from many changes until some point where they get to those higher fluence levels. But how we're going to do that needs to be worked out during the regulatory basis, because it's going to take some investigation into plant-specific details and other things to be able to do it.

I'll quickly talk about the schedule that

we had in the rule plan. Like I mentioned, we have some work to do in the regulatory basis development. So, we actually have about 14 months to do that, once we receive an SRM from the Commission, and after that, developing the proposed rule and final rule are both about the same time, about 15 months each. And that puts us at about 2027. If we get an SRM sometime a little bit later this year, that puts at 2027 before we issue the final rule.

MEMBER HALNON: How much time do we have before the first plant would reach the fluence of concern?

MR. RUDLAND: Like we talked about, for the ID, it's about 10 years.

MEMBER HALNON: So, this has got a lot of margin to it?

MR. RUDLAND: It's got some margin to it.

MEMBER HALNON: Okay.

MR. RUDLAND: That's why we decided to do this now instead of waiting until it was an issue.

MEMBER BALLINGER: But, to be clear, it's really an issue of the effect of uncertainty on the predictions?

MR. RUDLAND: That's correct.

MEMBER BALLINGER: It has nothing to do

with the actual performance of the vessel?

MR. RUDLAND: What it has to do is it has to make sure that they're operating to the right PT curves.

MEMBER BALLINGER: Right.

MR. RUDLAND: Yes.

MEMBER BALLINGER: Yes, there's a mean that's going on there, but then there's this big uncertainty band which you showed.

MR. RUDLAND: Yes.

MEMBER BALLINGER: And that's the thing that's getting us --

MEMBER HALNON: But, I mean, that's coming from the science piece; I'm coming from the operations piece. When does my PT curve change to the point --

MR. RUDLAND: That it becomes a problem.

MEMBER HALNON: -- that it becomes a problem, and I have to change my operating procedures and my simulator exams, and everything else that I do? So, I'm looking at the operations piece.

MEMBER BALLINGER: They have to do a PT curve every startup, right?

MR. RUDLAND: They have to do a PT curve anytime something changes. So, if they change their fluence levels. A lot of times, their PT curves are

out farther in EFPY than they're operating, basically.

MEMBER HALNON: Right, and that's what I'm talking about -- when the PT curve gets so close to the operating curve that I have an issue with my training, and everything else that goes on.

MR. RUDLAND: And to your point, Ron, the scatter that's in the first part of that is handled by margins. We put a margin term on there to handle that. But it's when it becomes non-conservative that it is just more than our margin.

Okay. So, in summary, we have high confidence that right now the plant remains safe and the licensing actions remain valid, because this issue is really a longer-term, time-dependent issue. And like I just mentioned, it's about 10 years before we begin to have an issue for PTS and about 23 years before PT limits.

And then that really just impacts our confidence in the integrity of the vessel for long-term operation. But we need to do further work, and we want to be proactive to ensure that reasonable assurance is still maintained. And we're going to do that through a risk-informed, performance-based solution.

Again, we delivered the rulemaking plan to

the Commission in March. And we want to develop a change to the rule that is focused on only those plants that are impacted.

I think that's all I have.

MEMBER BALLINGER: I'm having trouble finding the clock. Oh, there it is.

Great.

Are there any questions from the members before we switch? I don't think it's late enough to have a break. So, should we just push forward?

VICE CHAIRMAN KIRCHNER: Ron, I have a question, if I may?

MEMBER BALLINGER: Uh-hum.

VICE CHAIRMAN KIRCHNER: Dave, is there any -- you know, this is an area where there's been a lot of work done by the industry and the staff. There's a lot of regulatory history invested in this, or effort I should say, not history.

Is there active research over in RES looking at this matter and the falloff when you get to beyond 6 times 10 to the 19th fluence?

MR. RUDLAND: I think the only thing that's really active that's going on is our involvement into Code standards, and allowing those types of things to be developed by a wide variety of

technical experts. So, it's the ASTM standard, as well as in ASME to developing standards related to the prediction of fluence.

And so, we're relying -- you know, we weren't being involved in those, but we're relying on those experts as well as our experts.

VICE CHAIRMAN KIRCHNER: I didn't phrase my question well. Yes, the standards work is important, and that's the codification, an accepted standard showing this phenomenon informed with more data. Okay, I get that. You update the standard.

But I meant more from a real research standpoint.

MR. RUDLAND: Right, phenomenological, working on a phenomenological problem? No, there's not right now, no.

VICE CHAIRMAN KIRCHNER: Maybe that's something for research engineering, when we have our presentations from RES, we could inquire about.

MEMBER BALLINGER: The problem is it's just -- now this is a personal opinion -- it just isn't needed. I mean, if there was true uncertainty, then that would be one thing. But the general trend plus margin is fine.

And the research, you know, extracting

samples from vessels that were retired or, you know, all of this taking samples is extremely expensive. And those vessels, when they were put in service, there was not a whole lot of knowledge about the vessels themselves. And so, you have to sort of divine what those conditions were in the vessel to start with in order to make sense of the data that you get from doing Charpy tests or fractured toughness tests. So, it becomes very expensive with minimal payoff.

VICE CHAIRMAN KIRCHNER: What about the delayed capsule testing?

MEMBER BALLINGER: Well, they're going to get those anyway.

MR. RUDLAND: Yes. And I think EPRI is going to talk about some of that here.

VICE CHAIRMAN KIRCHNER: Okay. So, there is an effort, though, then, to --

MR. RUDLAND: There's an EPRI effort.

VICE CHAIRMAN KIRCHNER: -- inform, better inform, the space that we're looking at?

MR. RUDLAND: Yes, I think Elliot is going to talk about that.

VICE CHAIRMAN KIRCHNER: Thank you.

MEMBER BALLINGER: If some of these

capsules were to come out and, all of a sudden, there's, you know, something really weird, then they would have to figure out; that may be something different.

MR. RUDLAND: Researchers have been chasing mechanisms that cause an increase in embrittlement for years. I mean, there's been a lot of talk of late-blooming phases that occur that cause the embrittlement. And they've been chasing that for years without much success.

MEMBER BALLINGER: But it put a lot of graduate students through school, especially at Santa Barbara.

(Laughter.)

MEMBER BALLINGER: Okay, anybody else?

(No response.)

MEMBER BALLINGER: Then again, I ask the same question: it's too early for a break?

CHAIRMAN REMPE: I think you have another presentation that's coming up.

MEMBER BALLINGER: Yes.

CHAIRMAN REMPE: And so, I think it's a little early for a break.

MEMBER BALLINGER: Yes.

CHAIRMAN REMPE: Let's go ahead and --

MEMBER BALLINGER: Okay. So, Elliot, are you up and ready? Whoops, here we go. Shoot. Elliot, are you up and ready?

MR. LONG: Elliot Long speaking. Can everyone hear me?

MEMBER BALLINGER: Good. The floor is yours.

MR. LONG: Okay. Chris or Dave, can you share my slides? Or do you want me to share myself?

CHAIRMAN REMPE: Chris, are you doing it?

MR. BROWN: Elliot, if you want to, you can do it. If not, I can do it.

MR. LONG: All right. Let me see if I can pull this off.

MEMBER BALLINGER: We see it now.

CHAIRMAN REMPE: Elliot, yes, if you could switch to presentation, that would be great. Thank you.

MEMBER BALLINGER: Okay, let's go.

(Pause.)

MEMBER BALLINGER: I think somebody's on mute or --

CHAIRMAN REMPE: I don't know if Elliot's on mute. Elliot, we see your hand up and we've lost your slides, and we can't hear you. Oh, it muted you.

MR. BROWN: I can also share it.

(Pause.)

CHAIRMAN REMPE: You can't unmute?

Oh, can you share, Chris? I can lower your hand, but I can't unmute you, from what I can tell.

(Audio interference.)

CHAIRMAN REMPE: Oh, you're coming through a phone, your sound is. I was looking at your presentation.

MR. BROWN: You might have to use star-6 to unmute.

CHAIRMAN REMPE: Yes.

MR. LONG: When I shared my slides, it wouldn't let me unmute. I muted and I couldn't unmute. I apologize.

CHAIRMAN REMPE: Okay. Are you using star-6 on your phone line, if you're coming through a phone line?

MR. LONG: Yes, and that's when I heard the disembodied voice that I wasn't allowed to unmute.

CHAIRMAN REMPE: Oh, okay.

MR. LONG: And that's after I shared my slides. It was good before.

CHAIRMAN REMPE: Chris Brown is going to

share his slides. So, just give us a minute here, and we'll hook you up.

MR. LONG: Okay. Yes, sorry about that technical difficulty.

(Pause.)

MR. BROWN: Elliot, I have your slides here.

MR. LONG: This is good. Everyone ready again?

CHAIRMAN REMPE: Yes, we are.

MR. LONG: Okay. I will start over. Okay. I am Elliot Long with the Electric Power Research Institute. I am the current Technical Leader for Reactor Pressure Vessel Integrity and Low Alloy Steel Research.

I'm going to speak today about our RPV embrittlement monitoring and prediction in long-term operation and some of the current programs that EPRI has today.

Next slide. We have two current, ongoing, long-term programs to generate high fluence capsule data for the domestic fleet. The first is the PWR Coordinated Reactor Vessel Surveillance Program, CRVSP, and the second is the PWR Supplemental Surveillance Program, PSSP. And I'm going to talk

about both of those today.

And then, afterwards, I want to review some of the prior EPRI conclusions from the November 2019 ACRS meeting, and then take a look at the potential impact to PT limit curves.

Next slide, please. So, we'll start off with CRVSP discussion.

Next slide. As I mentioned, the goal of this project is to optimize the current U.S. PWR surveillance capsule withdrawal schedule to increase the amount of high fluence data available to the industry. This data can then be used to inform future embrittlement trend curve correlations with applicable fluence changes for 60 years and beyond.

The original revision of this report was completed in 2011, and it reviewed every reactor vessel surveillance program from the U.S. fleet, and then recommended changes for certain plans to select withdrawal schedules for the capsules, to increase the amount of high fluence data by the selected date of 2025.

Last year in 2021, EPRI MRP updated this report to review how we did; what has happened; what's left to do, and when will we have that data available across the fleet.

This update included a review of all capsules withdrawn since 2011. It looked at future capsule pull schedules. It updates any capsule fluence values, and we also analyzed and addressed any plants that closed or are planned to close in the near term.

Next slide. The results of this program to date are that 16 of the 30 CRVSP capsules are tested or plan to be tested. The remaining 14 capsules -- there are 14 remaining capsules -- half of these will not be tested or are not planned to be tested, mainly due to plant shutdowns, or they will be delayed beyond 2025. So, we'll still get the data; it will just take a little bit longer.

In summary, for high fluence data that is currently available, there are 48 capsules with a test of fluence greater than 3 times 10 to the 19th. Four of these are greater than 8 times 10 to the 19th. By 2025, when the remaining seven planned capsules will be tested, all of them will be over 3 and two of those will be over 8 times 10 to the 19th.

Any questions on the CRVSP program before we move on to the PSSP?

(No response.)

MR. LONG: Okay. Next slide. PWR

Supplemental Surveillance Program, PSSP. This is a similar program to the CRVSP. The goal is to generate additional high fluence surveillance data, again, to inform development of embrittlement trend curves applicable to high fluence or long-term operation.

The overall objections are to fill in gaps in the tested surveillance capsule database and to utilize commercial PWR data to inform future embrittlement trend curve and not to need to rely on test reactor data.

The overall project goal is to irradiate two supplemental surveillance capsules for approximately 10 years before withdrawal testing, evaluation, and publication of the reports.

These two supplemental capsules have an additional 288 Charpy specimens from 27 unique materials across the three main material types -- plates, forgings, and welds.

The data generated from these capsules will yield 24 new transition temperature shift values and, also, three additional upper-shelf energy results. The fluence range on the specimens is 4.5E19 up to 1.2E20. So, well into the future for fluence, very high fluence.

So, this slide documents some of the

history of this program. The program designed and fabricated two supplemental capsules containing previously-irradiated and reconstituted PWR materials. These two surveillance capsules were inserted into Farley Unit 1 in October of 2016, and then Shearon Harris in April of 2018. The fabrication report was published in 2016, at the right.

Let me describe, first, the images. So, the image at the right shows how you make a reconstituted Charpy specimen. The top figure, broken Charpy, is after testing of a Charpy, you end up with two halves, of course. One of the halves is machined into the specimen insert with end tabs, which are then welded -- in the middle image there. The welded and reconstituted specimen is then machined square, and then cut to length at the bottom. And that bottom, the fifth picture there, is then inserted into the new reconstituted capsule.

The current plan is to withdraw Farley Unit 1 in the spring of 2027 and the Shearon Harris capsule in the fall of 2028. Those schedules were revised in the draft as part of the work done for MRP-326, Revision 1.

The testing of the capsules will then commence through 2028 to 2030. The capsule reports

will be delivered approximately 18 months after the withdrawal date in two separate reports. And then, after the data is completed and compiled, we will analyze the impact, 2030 to 2032.

That bottom right picture is the image of one of the capsules. That's the Shearon Harris capsule seated in the capsule holder inside the reactor vessel, and you're looking downward at the top end plug of the surveillance capsule.

Any questions on the PSSP program before we move forward?

(No response.)

MR. LONG: Okay. I now want to review some of the prior EPRI conclusions on the potential revision of Reg Guide 1.99, R2. In November of 2019, EPRI made an original presentation to the ACRS, discussing the potential revision of the Reg Guide. The conclusions from that meeting -- that are on the right -- have not changed.

If, in the future, we were to revise this Reg Guide, the AFPM E900-15 model remains the preferred embrittlement trend curve today.

As Dave mentioned a few times in his presentation, below a fluence target of about 16-19, the current Reg Guide remains adequate. It's above

that fluence threshold where it begins to break down.

Consistent with that idea, no BWRs will reach that threshold. And when we're looking at PT limit terms, we're really focused on the 1/4T limit. So, what fluence will reach the 1/4T fluence of 16E19?

So, on the next slide, I detail how much surface fluence you would have to have before your 1/4T or 3/4T influence is equal to 16E19 threshold. So, to do this, I utilize the attenuation formula from the current Reg Guide.

Again, I want to note that BWR units will not reach a fluence of 16E19. In fact, the highest 1/4T SLR fluence is actually less than 1E19 at 80 years for the U.S. BWR fleet.

But the story is a little different for the PWRs. The table at the top right shows what surface fluence you need to reach 16E19 at 1/4T or 3/4T.

So, let's do an example. For the Westinghouse Four-Loop plant with a B&W vessel, the thicknesses are approximately 8.5 inches and you need a surface fluence of 9.99 on 10 to the 19th. You have a corresponding 1/4T fluence of 6 before the Reg Guide begins to break down.

So, looking at this chart, it is observed

and understood that the Three-Loop Westinghouse design has the highest surface fluence at end of life. We can also understand that the 3/4T fluence values are well beyond anything we would expect at the surface. For example, the larger Westinghouse Four-Loop with the CE vessel needs a surface fluence of 28.3 times 10 to the 20 to have a 3/4T fluence of 16E19.

So, I look next at the surface fluence values for the six plants that have submitted SLR applications. That chart is shown on the bottom right. For five out of six, the surface fluence values is below the value needed to have 1/4T fluence of 16E19. For example, Plant C, the surface fluence is 5.56, and that value is below the value necessary of 9.62 to have an issue for PT limits through 80 years of operation.

Only one plant will ever hit a 1/4T fluence of 16E19 for 80 years. That's predicted to occur well into the SLR operating period, and I believe, consistent with Dave, it will be about 23 years into the future. So, that 23 years is the first time where the PT limit for the first plant, the lead plant, would be impacted by a potential change to the Regulatory Guide.

Any questions on that?

(No response.)

MR. LONG: Thank you. That is the end of my presentation.

(Pause.)

CHAIRMAN REMPE: Okay, try again.

MEMBER BROWN: Yes, Charlie Brown. I think I asked this question last time; I've forgotten the answer. One of them was you emphasized, for the PWRs, that you wanted to get away from test reactor data and get actual plant data. Is that a cost issue or is it a materials issue that you want to be more conforming with the actual materials that are in the plants? That's question one.

MR. LONG: Okay. We would like to be as close to conforming with the actual material in the plants as we can. All of the materials in the PSSP program were originally irradiated in a commercial, domestic U.S. reactor. The Charpys were tested, and then reconstituted into new Charpys, and they'll get a higher fluence data point when they are removed and tested from their host plant in the PSSP program.

MR. RUDLAND: And the data suggest that there's no -- sorry -- the data suggests that there's a flux event.

MEMBER BROWN: So, there is a difference

between test reactor data and in plant --

MR. RUDLAND: Yes.

MEMBER BROWN: Okay.

MR. RUDLAND: It may be able to be corrected, right? But I think in order to be true to that, what Elliot said, it's better to use the actual surveillance data.

MEMBER BROWN: Okay. My second question was on the Charpy specimens. When you break it, and then you cut it, I mean, it's broken, so you have to machine it to reconstitute it. And I think I asked this question. Are they big enough that the machining doesn't detract from your ability to get the data you want?

MR. LONG: Yes. There are ASTM standards that we complied with to meet the reconstitution dimension, and these specimen passed those criteria. So, yes.

MEMBER BROWN: Okay. All right. Thank you. Now that you said that, I remember that answer.

MR. RUDLAND: They put the notch in areas that are far enough away from the cut edges --

MEMBER BROWN: Yes, I knew that was an issue, just from past experience.

VICE CHAIRMAN KIRCHNER: And then do they

reanneal them and everything else to --

MEMBER BROWN: They shouldn't do that.

MR. RUDLAND: No. No, because they want more fluence --

VICE CHAIRMAN KIRCHNER: Okay. So, you're just accumulating fluence on an existing sample? Thank you.

MEMBER BROWN: Thank you very much.

MR. LONG: You're welcome.

MEMBER BALLINGER: Other questions?

(No response.)

MEMBER BALLINGER: Okay. I think on the schedule we have an opportunity for public comments. So, in keeping with that, if there are any members of the public that would like to make a comment, I think, since you have a Teams invitation, it should be easy to do. Please identify yourself and make your comment.

(No response.)

MEMBER BALLINGER: If you're on a phone call, I think you have to do the star-6 thing still, right? Yes.

(No response.)

MEMBER BALLINGER: No? The five-second rule. Hearing no public comments, thank you very

much, both of you, for the presentations and, by the way, for the next years of work that's been going into this.

And I'll turn it over to the Chair.

CHAIRMAN REMPE: Thank you. So, at this point, we're going to go off the record and we are going to take a break. Is 10 minutes enough, folks, or 15 minutes? Okay, so let's do a 15-minute break, and we're going to restart at 9:55.

And Ron has a draft letter.

Oh, you want to do a 20-minute break? I'm looking at the computer clock, okay, Jose?

MEMBER MARCH-LEUBA: Yes.

CHAIRMAN REMPE: Okay, yes. So, at 9:55, we're going to restart, based on the computer clock, not the clock in the room.

And we'll listen to Ron reading his draft letter.

And at that point, we're going to recess, folks.

We'll be coming back on the record at 12:30 p.m. for the BWRX-300 discussion. Thank you.

(Whereupon, the above-entitled matter went off the record at 9:40 a.m. and resumed at 12:30 p.m.)

CHAIRMAN REMPE: Okay, the clock on the computer says it's 12:30 p.m. on the East Coast, and we're back in session.

And at this point, I would like to ask Member March-Leuba to lead us through our second topic today on BWRX-300's Containment Evaluation Topical Report.

MEMBER MARCH-LEUBA: Thank you, Member Rempe.

I assume we're going to be listening to our General Electric Hitachi Topical Report on BWRX-300 Evaluation Method for the Containment.

So, without much more ado, we're going to let the staff make introductory remarks.

Mike Dudek?

MR. DUDEK: Thank you.

Technical Session Chair Mark Dudek, and I'd like to once again thank Chairman Rempe and the rest of the Committee for this opportunity to present the staff's findings associated with the BWRX-300 Topical Report entitled, "Containment Evaluation Methodology."

We presented to the Subcommittee on this Topical Report on March 18th; just outstanding discussion associated with those efforts, a lot of

good questions, a lot of good answers. And the staff's back today to address some of the concerns that were identified by the Committee and to give another high-level overview of the Topical Report. And really, at the end of the day, you know, the staff just wants to present its findings associated with the Topical Report and the methodology on this methodology for containment thermal hydraulic performance.

This was a year-long effort by the staff, and really, it involved the staff taking a look at the containment evaluation method through the TRACG, the Transient Reactor Analysis Code, and then the reactor pressure vessel and the containment through the GOTHIC code. There were several RAIs. There was an audit. And as I said, this was a year-long effort to verify the acceptance criteria anywhere from AOOs, loss of AC power, ATWS, small break LOCAs, and large break LOCAs.

Tangentially to this SER being approved by the staff in early January, we've been working in parallel on just an outstanding effort with the Canadians to do a joint review on this Topical Report. And really, the purpose of that joint review, we had an MOU with Canada to conduct it. We embedded staff with each one of the organizations for continuous learning. Both regulators evaluated how the

performance would be used in each other's regulatory infrastructures, our pre-application engagements, and during our application reviews. We had mutual understanding. We had several workshops to kick off this initiative in which each regulator presented its regulatory infrastructure and how we do business.

So, to use an analogy that I've been using, we learned how to dance. We learned how each other operates. We learned how each other reviews. We learned how to conduct a joint review. It was just an outstanding effort with Canada.

And at the end of the day, we had four mutual learnings. We had four joint conclusions and an observation that we had. The learnings for this report can be found in ML22091A201.

Now what's important, also, about this joint review is that nothing in the report fetters the powers of either CNSC or NRC, and nothing in the report can be construed or interpreted as affecting the jurisdictions or discretion of either regulator. There are no new learnings identified in the joint report, but there is a fulsome description of exactly how the staffs interacted, exactly what they reviewed, and how they did it. Just an incredible effort.

So, I would be open to any discussions or

questions associated with that joint report. And without any further ado, I'll turn it back to Jose March-Leuba and GEH to actually present on the topic at hand, which is the SER for containment evaluation performance.

Thank you.

MEMBER MARCH-LEUBA: Yes, Mike, this is Jose. Just for clarification, this joint report was an excellent activity that the U.S. performed. I'm sure it produced a better review of the Topical Report than without it.

MR. DUDEK: Right.

MEMBER MARCH-LEUBA: But it does not affect the SER from the NRC. When GEH submits a license to build a BWRX-300, they will have to refer to the SER that you wrote, not the joint report, correct?

MR. DUDEK: That is correct. That is correct. The SER is the United States' publication of our safety findings. I think CNSC took their own learnings back and will incorporate their learnings into their own report. But, yes, the SER is the U.S.'s findings on the Topical Report submitted to us by our applicant.

MEMBER MARCH-LEUBA: Yes. Now on this

joint effort, in my personal opinion, I find it very valuable. And we, as a Committee, I think, would like to hear more about the effort itself, something at the 30,000-feet level to oversee what the whole effort is. We don't want to know the details or limitations and conditions, and everything, but at the high level. And our people will be in contact with your people to see if we can establish a time mutually acceptable.

CHAIRMAN REMPE: Right. I'm not sure if you're the one to be asking this, Mike, but I know from the popular press of this MOU, and interactions with other folks here at the agency who have mentioned it. And I know it encompasses more than just thermal hydraulics. It includes fuel reviews and other aspects of licensing applications.

And so, yes, I would like to share what Jose is mentioning about having a general overview of the types of activities and documents that have been produced. And so, our staff will be reaching out to the appropriate person, and we will be following this up.

MR. DUDEK: Absolutely. I've taken that for action. This is the third joint review that we have on the books, and I think we have a few more planned. So, we would be more than open to coming and

talking to the Committee about the initiative.

CHAIRMAN REMPE: Wonderful. Thank you.

MEMBER MARCH-LEUBA: And with that, we'll give GE the floor, George Watkins, and I won't try to butcher your title. You introduce yourself.

MR. WATKINS: Okay, I will.

Good afternoon. My name is George Watkins. I am the Vice President of New Nuclear Plants and Products Licensing at GE Hitachi.

Today, we are going to discuss the LTR NEDC-33922P, the BWRX-300 Containment Evaluation Method Licensing Topical Report. The BWRX-300 design leverages our economically-simplified boiling water reactor design with some additional innovations to improve simplicity of the plant while also enhancing safety, reducing cost in the end for the plant to ensure its commercial viability.

So, one area in the evolution of the design from ESBWR to BWRX-300 is in the containment design. The ESBWR had a dry well/wet well containment with a suppression pool. It used a different methodology for post-accident cooling, a different methodology for reactor pressure control and level control after an accident.

So, we have a much simpler, dry, inerted

containment which has no major subcompartments. So, it's a very simplistic containment design to analyze. So, as a part of that simplification, we looked at using a more simplified approach than TRACG for evaluating the thermal hydraulic response of the containment atmosphere after an accident.

So, therefore, we decided on GOTHIC, the universally-used code in a lot of applications, and we went about developing the evaluation method with GOTHIC, based on the BWRX-300 conceptual design. And so, this was the whole reason for the LTR, was to present this as the evaluation method going forward for future licensing applications.

So, I want to thank the NRC management and staff for their review; also, CNSC for their joint review of this LTR. I agree that I think it greatly improved the review process. We did have a very significant audit process as part of this, which also aided in developing a very good product, in my opinion.

So, with that, I'm going to turn it over to Lisa Schichlein, who is our U.S. Licensing Manager.

MS. SCHICHLEIN: Good afternoon. My name is Lisa Schichlein and I'm the U.S. Licensing Manager for New Power Plants and Products at GE-Hitachi. I

want to thank the ACRS Full Committee for the opportunity to present BWRX-300 Containment Evaluation Method Licensing Topical Report.

In addition to George Watkins we have in attendance from GEH, the U.S. licensing manager for the BWRX-300; and Necdet Kurul, principal engineer in Containment. Supporting us on the call we have the lead licensing engineers on the topical report: Frostie White and Lou Lanese, as well as Charlie Heck, who's a consulting engineer for nuclear applications; Dan Pappone, the chief consulting engineer for plant performance; David Hines, the principal engineer for plant integration; and Rosanne Harrington, who's the manager of LOCA and containment analysis for GE-Hitachi. In addition we have Dr. Tom George on the call, who's a consultant from Zachry Engineering. As questions arise I may ask one or more of those folks to address the question.

GEH is seeking NRC approval to apply an analysis method for evaluating the dry containment thermal hydraulic performance for the BWRX-300 small modular reactor using the TRACG and GOTHIC computer codes. The driver for this new method is the inerted dry containment of the BWRX-300, therefore we're using a different code than we have used in the past for

reactor and containment thermal hydraulic performance. Specifically, GOTHIC is now being used with inputs from TRACG to evaluate thermal hydraulic performance.

The scope of the topical report included method description and qualification, sensitivity studies, and demonstration cases. The analysis method used for the BWRX-300 containment thermal hydraulic performance demonstrates that the containment design complies with 10 CFR 50, Appendix A General Design Criteria 2, 4, 6 -- sorry, 2, 4, 16, 38, 41, 50 and 51.

MEMBER MARCH-LEUBA: Let me -- more clarification. The methodology that we are approving today does not demonstrate anything. Can be used to demonstrate --

MS. SCHICHLEIN: Yes.

MEMBER MARCH-LEUBA: -- once we have a design, correct?

MS. SCHICHLEIN: That is correct.

MEMBER MARCH-LEUBA: Okay. Thank you.

MS. SCHICHLEIN: The specific acceptance criteria for the BWRX-300 containment performance are discussed in Section 4 of the NRC-approved licensing topical report, NEDC-33911P, which is entitled BWRX-300 Containment Performance. And I'd like to

emphasize that GEH is not seeking NRC approval for any exemptions from regulatory requirements.

There are several design features of the BWRX-300 that are relevant to the containment evaluation methodology in the topical report. First, the containment is a dry enclosure inerted with nitrogen at near atmospheric pressure during normal operation. The containment design pressure and temperature are within the experience base of conventional BWRs. And there are no sub-compartments containing large bore high energy lines and these sub-compartments have sufficiently large openings so the boundaries of the sub-compartments do not experience large pressure differentials from pipe breaks outside of the sub-compartments.

The limiting large pipe breaks evaluated in the topical report are the main steam pipe and feedwater pipe with all large pipe breaks assumed to rapidly isolate at the reactor pressure vessel nozzle. For small breaks the limiting small break is an un-isolated instrument line break.

MEMBER MARCH-LEUBA: And I can probably say this in open session: The small breaks are therefore the limiting ones?

MS. SCHICHLEIN: Correct. The containment

evaluation method involves the use of two codes, as we mentioned before: TRACG and GOTHIC. TRACG is used to evaluate the mass and energy release in a manner consistent with its use in evaluating the mass and energy release for the ESBWR. The ESBWR TRACG-LOCA method is applied to the BWRX-300 mass and energy release calculations to evaluate the large and small steam and feedwater pipe break case and conservative cases -- base and conservative cases.

GOTHIC is used to evaluate the containment thermal hydraulic response and uses a new containment model developed for BWRX-300. The GOTHIC code has been benchmarked to separate effect and integral tests and benchmarking to the test data of a similar size containment is included in the topical report.

The containment response evaluation method for BWRX-300 includes both base and conservative cases and for the conservative cases key inputs, assumptions, and modeling parameters are simultaneously conservatively biased, which is the same approach used and approved for the ESBWR.

In conclusion I would like to wrap up this presentation by restating that GEH is seeking NRC approval for application of an analysis method to be used for evaluating the BWRX-300 dry containment

thermal hydraulic performance. GEH is not requesting NRC approval for exemptions from any regulatory requirements. And just to restate, the method is using two computer codes: TRACG and GOTHIC. TRACG uses the applicable parts of the approved topical report EBC33083E-A, Revision 1, for the application of TRACG to ESBWR, which is incorporated in the approved ESBWR design certification. And GOTHIC is the standard code used for thermal hydraulic containment evaluations in the nuclear industry.

With the new methodology individual key inputs, assumptions, and modeling parameters are conservatively biased simultaneously, which is the same approach taken for the ESBWR containment method in the NRC-approved topical report entitled TRACG Application for ESBWR.

That concludes my brief presentation. Are there any questions or comments for GE-Hitachi?

MEMBER MARCH-LEUBA: Yes, this is a methodology report and we still don't have an X-300 design, but in your report you have provided a large number of demonstration analysis on conceptual concepts for your reactor. Do we have a lot of margin on your conceptual X-300 or -- I mean, yes, can we talk about margins in an open session?

MR. KURUL: Currently we do have margin in our calculations. The results first. It is the large break LOCA calculations that are the limiting cases, which is back through containment design limits. Small break cases are non-limiting. They are just there to demonstrate that they are not --

MEMBER MARCH-LEUBA: Can you state your name for the record?

MR. KURUL: It's Necdet Kurul, GE-Hitachi. For the large break cases we do have sufficient margin. And as we develop containment design further, we are watching that margin so that we don't end up coming right up against the design limits as we develop the containment further. And if it so happens that we end up getting into the margins much, then we would have to increase the design limit and the containment design would have to accommodate that increase in the --

MEMBER MARCH-LEUBA: Basically we have to put more concrete on the containment?

MR. KURUL: More concrete actually we find that we have more concrete then. But we put more concrete or put more steel, whatever we need to do to increase design pressure. So any changes in the containment design development goes along with

maintaining margin.

MEMBER MARCH-LEUBA: Yes, and I realize that you are more focused than I am on containment. The small break LOCA is more limiting on conductivity, correct? And that will be a transient analysis, not containment?

MR. KURUL: That's correct.

MEMBER MARCH-LEUBA: Yes.

MEMBER HALNON: Yes, let me step back to the licensing aspect. It doesn't sound like either code is new to the NRC and it's not new to the application of establishing containment performance. Are you asking for this approval as a feel good to make sure that you're confident going forward or is it something that really has not previously been approved for use by the NRC even though it's on this application?

MR. WATKINS: Yes, it's George Watkins, GEH. Yes, the simple answer is we do want to mitigate licensing risk going forward with our applications with the various utilities. The genesis of our licensing topical report strategy has been to mitigate new features that are in the BWRX-300 beyond the ESBWR base design so that when we get to the point of an actual applicant for a construction permit or a COLA,

whichever path we end up going down there, these issues will already have some certain regulatory basis going forward. So that's the real reason that we're asking for these approvals at this time.

The BWRX-300 containment is -- kind of goes back to the root. Some of our early BWRs had dry containments, but everything since then has been the suppression-type containment. So we wanted to make sure that we could demonstrate to the NRC staff that we had the analytical capabilities and understood the method sufficiently to apply to this new design.

MEMBER HALNON: Okay. So its iterative approach is like you said for regulatory certainty? I mean this could easily have been done -- once you get your design -- your containment designed you could have established all this at --

MR. WATKINS: That is correct.

MEMBER HALNON: -- one time.

MR. WATKINS: So it is appropriate to say we're asking approval of the methodology, not its total application. That will come with an actual license application. And the staff will of course talk about their limitations to conditions which address primarily those types of issues that --

MEMBER HALNON: Okay. I was just --

(Simultaneous speaking.)

MR. WATKINS: -- apply to an actual application.

MEMBER HALNON: All right. I was trying to figure out the iterative approach that you took --

MR. WATKINS: Right.

MEMBER HALNON: -- and I understand that. Thank you.

MR. WATKINS: Okay. Thank you.

MEMBER MARCH-LEUBA: Members, any more questions for GE?

VICE CHAIRMAN KIRCHNER: I'd just like to make a comment and I'm pleased to see that you're using the TRACG code. Just for the record, TRAC had its genesis in investments made by research at Los Alamos starting in 1976. I wrote the transfer package for the code. I bet if you showed me your code, I wouldn't recognize it anymore.

(Laughter.)

VICE CHAIRMAN KIRCHNER: But GE picked up the code. I'd estimate that was '78, 1978-1980 timeframe. I don't know if any of your colleagues are still with you that had picked that code up in that time, but I was one of them at that time.

So, but the point of my comment isn't to

draw attention to me; it's to draw attention to the RES investments that were made way back and still paying off for the industry and for the staff because TRACE had some similar genesis. Thank you.

MEMBER MARCH-LEUBA: Any more comments or questions for GE?

Okay. So we're going to have a short transition to the staff presentation. Will you have to set up your computers here?

While we're in transition I want to say that this is the first time we have had visitors for outside in this room in -- since forever. So you're very welcome. Thank you for being here.

CHAIRMAN REMPE: It's really nice to see people and actually see their smiles or frowns at us, as the case may be.

MEMBER MARCH-LEUBA: Yes, and I have asked Ken for tomorrow's presentation, which you'll be hearing, to have name tags so nobody calls George Greg again.

(Pause.)

MEMBER MARCH-LEUBA: Okay. So we're back in session. We have the staff properly set on the table. And please go ahead and start making your presentation.

MR. SHEA: Good afternoon. This is Jim Shea of the staff. Before we start I wanted to -- Becky Patton wanted to make some opening remarks.

MS. PATTON: Yes, this is Becky Patton. I'm the branch chief of the Nuclear Methods Systems and New Reactor Branch in NRR. My colleague Scott Krepel from one of the other reactor systems branches addressed the Subcommittee last time. As me mentioned we had an integrated review team of multiple branches working on this as well as our colleagues in the Office of Research who provided essential support throughout the review. And then there was also coordination with the CNSC on this review.

So there were a lot of people involved. There was very close communication including the year-long audit that took place during the review. And I understand the Subcommittee meeting was very engaging. There were a lot of questions. We really appreciate the Subcommittee's look into these matters and we look forward to the presentation today as well.

The staff did look at the issues they felt were most important, most significant in terms of safety, and that's always our number one thing for all of this. And even though the review was conducted on a more compressed timeframe than would be typical for

this type of review, because we did it in such an integrated manner and with resources put to bear, we believe it was very effective and it was also a benefit to be able to discuss with our colleagues at the CNSC. So I appreciate the attention of the Committee and go forward with the presentation.

MEMBER MARCH-LEUBA: Thank you. Let me interrupt before you go in. Remind everybody, we're in open session. If we get into a closed -- we have a closed line if necessary, but I hope we don't have to use it. Just stay non-proprietary.

MR. SHEA: Good afternoon. I am James Shea of the NRC's Office of Nuclear Reactor Regulation in the Division of New and Renewed Licenses. I am the lead project manager for the pre-application activities for the GEH BWRX-300 design.

I wanted to thank the ACRS Full Committee for this time to allow the staff to present its review of the Licensing Topical Report: Containment Evaluation Method, as reflected in the staff's advanced safety evaluation submitted publicly on March 9th, 2022.

The staff's review focused on the approval of the BWRX-300 containment evaluation methods for evaluation of the design-basis events. As stated in

the safety evaluation report, the NRC will evaluate the compliance of the final design of the BWRX-300 containment performance during future licensing activities in accordance with 10 CFR Part 5, or 10 CFR Part 52.

The staff presented its approval of this topical report to the ACRS BWR Subcommittee on March 18th, 2022 and the staff today will provide a high-level summary of the Subcommittee presentation.

Just to recap, the LTR was submitted to the NRC on September 25th, 2020. Revision 1 supplemented with requests for additional information, RAI, responses was submitted on November 19th, 2021 and Revision 2 was submitted on December 17th, 2021.

I will now turn over the staff presentation to our lead NRR technical staff reviewer for the containment evaluation methods, Syed Haider.

MR. HAIDER: Thank you, Jim.

Good afternoon to everyone. My name is Syed Haider. I am the lead reviewer of the GEH licensing topical report, LTR NEDC-3392P, which is an LTR related to evaluating the containment pressure and temperature response that would result from the limiting design-basis accidents involving mass energy release from the reactor pressure vessel into the

containment for the BWRX-300 SMR.

My Nuclear Systems Performance Branch at NRR, the lead for the review under Branch Chief Scott Krepel. My NRR colleagues Carl Thurston and Shanlai were the reviewers for the LTR part related to the TRACG modeling and mass energy release calculation methodology. I myself mainly reviewed the GOTHIC modeling and containment pressure and temperature response calculation methodology. Chang Li from NRR reviewed combustible gases inside the containment.

Throughout the review the NRC staff was supported by the research staff that included Peter Lien, Joe Staudenmeier, Andrew Ireland, and Shawn Campbell. Research was responsible for the development of the TRACE and MELCOR confirmatory models that were used to analyze and validate the TRACG and GOTHIC model results and trends presented in the LTR.

This slide shows the outline of the NRC staff's presentation today during which we will not discuss any proprietary information about the BWRX LTR. Should there be a need, we can have a closed session to respond to any proprietary questions.

First, the staff will give an overview of the purpose and scope of the BWRX-300 LTR. It's worth

mentioning here that even though the LTR title is BWRX-300 Containment Evaluation Method, it covers the methodology and modeling details for both the BWRX containment as well as the reactor systems.

First, we will explain the acceptance criteria for the BWRX-300 containment response and underlying regulatory requirements as followed in the evaluation methodology.

Then we will highlight the BWRX containment design background that drives the core of the evaluation methodology.

We will review the key components of the BWRX-300 design as they related to the TRACG and GOTHIC code-based reactor systems and containment analysis methodologies with the eventual objective of predicting the containment thermal hydraulic response for design-basis accidents with sufficient conservatism.

Then we will give a summary of the BWRX containment evaluation method demonstration analyses that are presented in the LTR.

After this my NRR colleague Carl Thurston will present a few slides on our TRACG codes-based mass energy release calculation methodology review.

After that I will present a few slides on

our review of the GOTHIC code-based containment response calculation methodology.

Then I will provide an overview of the NRC staff's confirmatory analysis that were conducted to develop additional insights and support our regulatory review findings.

At the end Carl and I will describe the four conditions and limitations on the LTR as they emerged during the staff review of the LTR.

Finally, we will present the staff conclusions for the LTR review.

This slide describes the purpose and scope of the BWRX-300 containment evaluation methodology LTR as reviewed by the staff. Basically GE-Hitachi has submitted this LTR to obtain the NRC's approval of the BWRX-300 containment pressure and temperature analysis methodology. The approved methodology is intended to be used to perform the BWRX-300 containment safety analyses, to support either a licensing application for a construction permit and operating license under CFR 50 or a design certification and combined license application under 10 CFR Part 52.

This slide also highlights the four acceptance criteria used in the methodology for the BWRX-300 containment response. These acceptance

criteria are identified in the LTR and are essentially driven by several NRC regulations.

The first acceptance criterion is a rooted in General Design Criteria for -- or GDCs 16 and 50 of 10 CFR Part 50, Appendix A, and ensures that the peak containment pressure and maximum shell temperature calculated for the limiting mass energy release design-basis accident are bounded by the BWRX-300 design pressure and temperature with sufficient margin.

The second acceptance criterion is driven by GDC 38 for containment heat removal. That is interpreted by the Standard Review Plan as to ensure that the containment pressure is reduced to less than 50 percent of the peak accident pressure for the most limiting LOCA within 24 hours.

GDC 38 also requires that the containment pressure after 24 hours for all LOCAs are maintained below 50 percent of the peak pressure for the most limiting LOCA which constitutes the third acceptance criterion for the LTR.

These three acceptance criteria collectively ensure that the containment structure can accommodate the pressure and temperature condition resulting from any mass energy release from the RPV

without exceeding the design leakage area and with sufficient margin.

The fourth acceptance criterion requires that the containment atmosphere remain sufficiently mixed such that deflagration or detonation does not occur inside the containment.

MEMBER HALNON: Syed, on your most limiting LOCA is it the same most limiting LOCA for each one of those, or it is the most limiting LOCA for that specific design criteria? In other words, it would not be the same and the one that gets you to below 50 percent in 24 hours may not be the same that challenges your 50 percent average. Is it the same or is it different?

MR. HAIDER: It's the same. I think that your question has two part. I heard you're also asking whether the peak containment pressure accident is the same as the peak shell temperature?

MEMBER HALNON: No, no. The most limiting LOCA --

MR. HAIDER: Yes.

MEMBER HALNON: -- where the second bullet there talks about the one that gets you down to -- it's got to be less than 50 percent after 24 hours.

MR. HAIDER: That's right.

MEMBER HALNON: Is that the same most limiting LOCA scenario for the third bullet, which is the one that may challenge the 50 percent heat pressure?

MR. HAIDER: Okay. The third bullet applies to all LOCAs.

MEMBER HALNON: Okay. In other words --

MR. HAIDER: In any LOCA --

(Simultaneous speaking.)

MEMBER HALNON: -- the most limiting LOCA is the one that challenges that the most?

MR. HAIDER: That's right.

MEMBER HALNON: That's not necessary the same scenario all the way through?

MR. HAIDER: That's correct.

MEMBER HALNON: Okay. Thanks.

MEMBER MARCH-LEUBA: But this is related to what I mentioned earlier that there is the most limiting LOCA with respect to containment pressure, core uncovery, long-term cooling and heating. So when you say most limiting LOCA, I assume you refer to the peak pressure --

MR. HAIDER: That's correct.

MEMBER MARCH-LEUBA: -- in this context?

MR. HAIDER: That's correct. In this

context, that's right.

MEMBER MARCH-LEUBA: Yes, because even though we're evaluating a methodology for containment -- to identify what the containment atmosphere does during LOCAs, I know you have also included in the review what happens inside the vessel, especially with no condensables and such. So you keep in mind the whole -- the holistic safety of the reactor, not just the containment when you were doing your review, correct?

MR. HAIDER: That's correct.

So now we will summarize the state of the art of the BWRX containment design and its relation with the staff review. This slide presents the salient design features of the BWRX-300 containment that are relevant to the evaluation methodology for the design-basis accidents.

These features drive several initial and boundary conditions in the containment GOTHIC model. Basically BWRX-300 has a dry containment that is inerted with nitrogen for normal operation. It does not have a suppression pool inside the containment like ESBWR had. This information is relevant to the initial containment pressurization and the post-accident mixing of steam and radiolytic gases inside

the containment.

A key design feature of the BWRX-300 design is early closure of the RPV isolation valve for the line breaks that limits the mass energy release into the containment during large break LOCA which dictates the peak containment pressure. However, the RPV isolation valve does not close for small breaks that remain un-isolated resulting in a continuous break flow throughout the small break LOCA analysis.

The BWRX-300 design employs a passive containment coolant system, or PCCS, to remove heat from the containment to a reactor cavity pool that is located above the containment. PCCS plays an important role in the long-term containment pressure reduction and mitigation. In this regard the LTR demonstration analyses were performed with the specific PCCS design configuration and described in the LTR. The reactor cavity pool that's used -- the PCCS for containment heat removal is located above the containment dome.

So at this stage it's worth emphasizing that the applicant has not submitted an ESAR or DCD yet and the containment design pressure and temperature are not available. Therefore the staff has performed this LTR review with an understanding

that the BWRX-300 design is not final and the analyses presented in the LTR are demonstration cases.

So the objective of the present review is to evaluate the applicability of the methodology of the BWRX-300 design and assess the degree of conservatism built into the methodology. And this has required the staff to focus on the uncertainties in the methodology and to predict trends without getting unduly hooked to exact figures. The staff understands that the inputs and outputs near ready for the final design that will be submitted at the licensing stage.

MEMBER MARCH-LEUBA: I have a comment. I hear from the applicant something I like very much and which I have inferred by reading that the approach is to develop and approve the containment pressure -- overpressure methodology calculation so that they can design the containment properly as opposed to -- what they could have done is guess at what the containment needed to be and then hope their methodology proves that they were right. So the fact that they chose to define and get approved the methodology by which the containment will be judged and then build the containment according to that is smart. I think it's a good approach. Do you agree?

MR. HAIDER: Yes. Yes, I agree because

this too -- the limit of margin is not coming to -- into the play let. We are dealing with uncertainties and conservatisms --

MEMBER MARCH-LEUBA: Yes.

MR. HAIDER: -- in this review.

MEMBER MARCH-LEUBA: It is -- I mean I'm guessing right here, I'm assuming that the margin they're going to have for containment will be sufficient to be comfortable, but won't be over design because that costs money. But by going this approach of having the methodology given approval ahead of time, we can agree on what pressure will be in the containment and what pressure you need to design your containment. So that's -- we haven't seen that for all the applicants that -- while I'll been here in this table. Some of the other applicants review their methodology after they got a design.

MEMBER HALNON: Is it fair to say the demonstration cases are close to the ballpark where they expect the design parameters to be in the final design so that we know that we're at least in this general area? Methodology works fine?

MR. THURSTON: Yes, this is Carl Thurston.

MEMBER HALNON: Thanks.

MR. HAIDER: Yes.

MR. THURSTON: Yes. So yes, that is correct. So we understand that the design of the reactor vessel is pretty much final, near or finalized. So we expect that the conservatism that we deserve for this demonstration cases will be carried over exactly.

MEMBER HALNON: Okay. So the exact values you see coming in are going to be pre-representative of what you've been feeling?

MR. HAIDER: That's true. That's true.

MEMBER HALNON: Okay.

MR. HAIDER: But at least by presenting the demonstration cases they are able to scale the level of conservatism that exist in the good sense of -- or the level of conservatism is built into the methodology. The margin will be on top of. And they may choose to in some --

MEMBER HALNON: Is there an idea of what the uncertainties are? Thanks.

MEMBER MARCH-LEUBA: Carl, for the benefit of the court recorder, can you state your name because he doesn't know your voice.

MR. THURSTON: Yes, Carl Thurston. Thank you. NRR DSS SB.

MR. HAIDER: So for further methodology

evaluation purposes for the present BWRX-300 containment design the LTR presents demonstration analyses and results for the thermal hydraulic performance in order to show that the acceptance criteria are satisfied for the limiting design-basis accident. In this backdrop the LTR presents results for large break loss of coolant accident, and small break loss of coolant accident inside the containment.

The containment design-basis analyses presented in the LTR include both liquid and steam space breaks and the applicant showed that all acceptance criteria were satisfied for the demonstration cases, however the applicant would be required to demonstrate satisfying the acceptance criteria for the BWRX-300 design submitted at the licensing stage while also meeting the four limitations and condition the staff has imposed on the LTR as a result of this review.

The CE methodology uses TRACG code to calculate the mass energy release for the RPV into the containment and GOTHIC code to calculate the resulting containment hydraulic response.

Before we describe the TRACG and GOTHIC parts of the review it's worth explaining that the applicant submitted two stand-alone or decoupled

models that were based on the TRACG code or mass energy release calculation from the RPV and GOTHIC for containment pressure and temperature response calculations. The NRC staff used TRACE and MELCOR codes to develop confirmatory models by using the input specifications that will directly derived from the TRACG and GOTHIC decks as submitted by the applicant. This allowed the staff to ensure a direct comparison between the confirmatory model results and the TRACG/GOTHIC results for both large break LOCA and small break LOCA cases.

The staff confirmatory analyses were mainly performed to accelerate a regulatory review by developing necessary insights to focus the staff's efforts on safety significant issues in the BWRX-300 phenomenal review.

MEMBER MARCH-LEUBA: Let me interrupt you there since you are breaking. I wanted to say this before but I got sidetrack. A key feature of the BWRX-300 is the isolation valves that you assume they work for your analysis, and rightly so for Chapter 15-type analysis. But when we move into the risk analysis at Chapter 19-type analysis, there will have to be an evaluation by the staff of why we have so much confidence that those valves will close on time

and the pipe upstream of it won't break. Just put a marker there that when we do that, let's not forget to review that because it is a key. If those valves don't work, the concept does not work. So we believe it does. It will. But we need to cross our Ts and dot our Is. Okay. Thank you.

MR. HAIDER: Thanks. I really appreciate.

So now I will turn over the presentation to my NRR colleague Carl Thurston to present a summary of the TRACG modeling of the mass energy release for BWRX containment.

MR. THURSTON: My name is Carl Thurston and I am the primary reviewer for the TRACG code calculation of mass and energy release.

TRACG code has a long history of review by the NRC staff due to several previous topical reports submitted for SBWR, for ESBWR, and for the existing fleet of BWR/2 through 6 plants.

The staff focused its review of the code on changes made since the ESBWR timeframe, which is around 2008-2009. The latest TRACG code was used for this analysis and the staff confirmed that there were no significant changes made since ESBWR.

The RPV model and internal components are pretty much scale from the ESBWR design. The

methodology uses what we call a decoupled approach though the containment is allowed to blowdown continuously into atmospheric pressure. So the boundary condition for the TRACG code is fixed at atmospheric pressure and that maximizes the amount of mass and energy release that will go into the GOTHIC code.

The methodology relies heavily upon previous TRACG topical reports and their compatibility with the BWRX-300. This is particularly true for the amount and the extensive qualifications done for the ESBWR design. So the ESBWR qualification was extended to BWRX-300 such that the ESBWR PIRT and the modeling basis are applied for the RPV and its internals. The BWR/2 through 6 methods and topical reports are invoked since some events result in core uncovery. The RCS, the isolation condenser have a significantly more important safety function than for ESBWR and is modeled in considerably more detail. Staff reviewed these modeling features and deemed them to be adequate for M&E release calculations with the limit and conditions applied.

Next slide. The next slide reviews unique features of the BWRX-300 design as compared to ESBWR. And as was mentioned, the large break LOCA isolation

valves are a significant change and are significant in allowing the pressure to be limited because these valves close very quickly once a pipe break occurs.

In addition, said several times, there's no suppression pool and the isolation condensers are the primary method for decay heat removal. So they have to function at rated conditions or better in order for the plant to cool down.

The RPV isolation valve limits break flow and M&E release for large piping, but the small breaks are un-isolated and they can continue to blowdown for the full time of the transient. The methodology assumes that one of the ICS trains are inoperative due to single failure.

The staff reviewed the conditions assumed and found that the inputs were conservative, related to initial power, power history, scram time, tilt flow modeling, atmospheric pressure for the bounding conditions indicated, and the operating conditions assumed for the plant, at generally 102 percent.

MEMBER MARCH-LEUBA: Do you remember what was the target pressure for the -- target peak pressure for containment? Is it like 50 psi?

MR. THURSTON: I don't recall if we had a target pressure.

Syed, you can chime in.

MR. HAIDER: It's -- you mean predicted heat containment pressure? Yes, 61 psi. Yes.

MEMBER MARCH-LEUBA: Yes, approximately 60. So I'm going to say pressure boundary condition is probably not very conservative at the beginning when you're 1000 psi inside, but eventually it will become conservative.

MR. THURSTON: Sure. Yes, until you ignore the pressure in the containment, right, and force as much mass and energy out of the reactor that you can.

MEMBER MARCH-LEUBA: Yes, that's clearly one of those obviously conservative assumptions.

MR. THURSTON: Correct. Yes.

Next slide, Jim.

So next we will review the findings, the limiting conditions that the staff had related to the BWRX-300 topical report methodology.

The first issue was related to radiolytic gas accumulation. The staff determined that the TRACG model did not adequately account for the accumulation of radiolytic gases in the IC tubes and its subsequent effect on heat transfer performance. Staff issued an RAI and GEH responded that a design change would be

implemented to limit build-up of non-condensable gases to negligible amounts.

MEMBER MARCH-LEUBA: For the deflagration margin had you guys given any consideration to the fact that there really is no oxygen there, that if any gas were to enter it would be nitrogen? Would you grace for that or you say however much hydrogen you have assume the oxygen miraculously shows up?

MR. THURSTON: Where they delve in those details Shanlai is -- we can --

MR. HAIDER: It's actually a little bit proprietary information, so we can definitely provide discussion on that if you --

MEMBER MARCH-LEUBA: No, I was just curious.

MR. HAIDER: Yes, because there is a number there.

MEMBER MARCH-LEUBA: Yes, my question is what if GE comes up with an argument, would you be willing to consider it?

MR. LU: Yes. I can provide you that, at least the open session answer. And I think during the normal operation there is a purge line through the top of the isolation condenser, continuous purging the non-condensable gas there. So therefore,

consideration for that one is very minimal in terms of the existing oxygen pressure inside of the isolation condenser. During normal operation it's completely flooded anyway. So I think that's a good question and if you want to talk about a little more detail, I think that's a proprietary session.

MEMBER MARCH-LEUBA: I know you're proprietary answer because I was in the Subcommittee, so I don't need it.

MR. LU: Thank you. Oh, this is Shanlai Lu from the staff.

MARCH-LEUBA: Carl?

(Simultaneous speaking.)

MR. THURSTON: -- any more questions related to the first one, then move to the second issue.

So our next issue was related to reverse flow of steam in the ICs return line. The staff noted that the design does not include a trap to prevent reverse flow of steam and radiolytic gases from the RPV back into the ICs and ICs tubes.

The staff issued an RAI and GEH indicated that in the final design a trap would be included to prevent this type of reverse flow.

MEMBER MARCH-LEUBA: And is their

commitment part of Revision 2, or is still on the RAI?

MR. THURSTON: Huh?

MEMBER MARCH-LEUBA: The commitment is part of Revision 2 or it's still in the RAI, the commitment to include a loop seal?

MR. THURSTON: Yes, so it's included as a limitation, so we expect them to --

MEMBER MARCH-LEUBA: Okay.

MR. THURSTON: -- comply.

MEMBER MARCH-LEUBA: Just part of the SER?

MR. THURSTON: Yes, sir.

MEMBER MARCH-LEUBA: I think the SER says include a loop seal or analyze it properly.

MR. THURSTON: That's correct. It does say or, but we are --

MEMBER MARCH-LEUBA: But we --

MR. THURSTON: Yes, we don't want to depend on something for analysis if we know that design can do -- and minimize any type of extensive analysis.

MEMBER MARCH-LEUBA: I think you hit the nail on the head. When we deal with an operating reactor that is already built, making modifications like this is very difficult, very costly and the operator -- the ignition that have to go there will

get a significant dose. So we tend to rely on calculations, more and more accurate calculations, expensive calculations to show it can happen.

At the design stage I applaud the approach you guys have taken to modify the design. So a simple calculation shows you cannot have backflow.

MR. THURSTON: That's right.

MEMBER MARCH-LEUBA: And at the design time it doesn't cost money. It doesn't cause a lot of considerations. This is the time to do it. Please continue. Continue both with your presentation and with the good work.

MR. THURSTON: Now I'll turn it back over to Syed.

MR. HAIDER: Okay. Thanks, Carl. This is Syed Haider. This slide presents an overview of the GOTHIC code as reviewed by the staff for the BWRX-300 containment response modeling in the LTR application submitted for the NRC approval.

GOTHIC code is a commission-developed computer code that has been widely used in the industry for containment thermal hydraulic response analyses. This is a continuously improved code that is compliant with the pertinent NRC regulation 10 CFR Part 50, Appendix B on quality assurance. The

applicant has used the latest GOTHIC version, 8.3, for the BWRX-300 analyses demonstrated in the LTR. They also clarified that they even follow development NRC process should there be a need to use a different GOTHIC version at the licensing stage. As far as the GOTHIC-based application in the past, our concern GOTHIC code has been previously approved by NRC for containment response analyses.

The staff also reviewed the BWRX-300 containment PIRT phenomenology and found it covered by the available GOTHIC code functionalities.

The staff also reviewed the GOTHIC benchmarking against the CVTR test data provided in the LTR for the BWRX-300 containment methodology qualification and found it acceptable.

The information provided in the LTR and RAI responses demonstrated that GOTHIC is qualified for modeling thermal stratification and multi-dimensional flow. These features are important for evaluating the nodalization-based containment design that employs distributed PCCS units.

Next slide, please? This slide captures the key features of the GOTHIC model development and the containment response calculation methodology in the LTR. The BWRX-300 containment evaluation method

has been developed following the applicable element of Reg Guide 1.203. Reg Guide 1.203 presents a multi-step process for developing an acceptable evaluation method to analyze transient nuclear power plant response for reg-defined figures of merit during the postulated design basis events.

The BWRX-300 containment evaluation methodology is based on a decoupled calculation between the TRACG and GOTHIC codes where TRACG model first calculates the mass energy release from the RPV into the containment assuming it remains at the same initial atmospheric pressure. This way the TRACG mass energy release calculations do not account for the increasing containment back pressure.

Using this decoupled approach the stand-alone GOTHIC containment model independently uses the mass energy release calculated by TRACG as a containment boundary condition for calculating the containment pressure and temperature response. The staff agrees that ignoring back pressure ensures a conservative mass energy release into the containment with respect to containment pressure and temperature response.

As described in the LTR the containment evaluation method is based on a four-component GOTHIC

model that include a nodalized main containment, a nodalized dome, a nodalized PCCS, and a lumped reactor cavity water pool within the space. A conservative diffusion layer model is used and justified for modeling condensation heat transfer on the containment shell and PCCS surface. The DLM condensation modeling option is available as a functionality in the GOTHIC. DLM is a mechanistic model that represents the underlying condensation heat transfer phenomena which is based on heat and mass transfer analogy as opposed to a curve fit of the test data.

The GOTHIC modeling approach accounts for thermal stratification inside the containment, which is important due to the containment and PCCS nodalization used in the BWRX-300 containment evaluation methodology.

This slide summarizes the review NRC staff performed of the BWRX-300 GOTHIC containment response methodology. The staff reviewed the important physical phenomena again defined in the GOTHIC PIRT table for the BWRX-300 design and found them acceptable. The applicant submitted GOTHIC decks for the base and conservative cases for both large break and small break LOCAs. The staff reviewed the GOTHIC input models and used the submitted GOTHIC decks to

gather information to develop the TRACE and MELCOR models for confirmatory analyses.

Key modelings, uncertainties, and conservative biases were reviewed to evaluate the overall GOTHIC model conservatism. We reviewed information about the nodalization sensitivity studies for the containment and PCCS. We also reviewed the GOTHIC code benchmarking as documented in the LTR against the CVTR test data that's applicable to BWRX-300 containment design as well as the justifications for bounding the containment condensation and natural convection heat transfer correlations by using the opening test data.

The staff also made sure that the BWRX-300 containment response analyses for small break and large break LOCA duly account for the appropriate break location and break flow direction sensitivities. In this process we identified a small un-isolated liquid break as the limiting small break LOCA.

The staff also reviewed the modeling of the PCCS that is used to reduce and mitigate the containment pressure in the long term. Containment mixing was also reviewed for potential deflagration and detonation due to combustible gases accumulation.

Now this slide summarizes the key typical

issues that the staff encountered on the GOTHIC containment side of the review. We present only the non-proprietary details about the issues or resolutions here. Any prop information on confirmatory analysis results will require a separate session.

First, the originally-submitted revisions of the LTR had not performed any break location or breakthrough directional study while identifying limiting breaks with respect to containment pressure and temperature. The staff also noticed that the limiting small break LOCA was a steam break while no containment analysis was performed for a small liquid break. So RAIs were issued and their response did show sensitivity to break location and break flow direction which modified the limiting large break location and break flow orientation that ended up increasing the limiting containment pressure. The response also showed that liquid rather than a steam break was the limiting small break LOCA with respect to peak containment pressure. The entire LTR was duly analyzed.

The staff also noticed that even though the applicant had presented a containment nodalization study for large break LOCA with four different

nodalization schemes most of that sensitivity study was performed for the small break. So and I've already showed the resulting small break LOCA nodalization study revealed that even though the peak containment pressure for a small break LOCA was sensitive to nodalization scheme, it was always smaller than the limiting peak containment pressure that occurs during the large break LOCA for the containment design used for the demonstration analyses.

The staff noticed that the sensitivity small break LOCA nodalization diminished in the long term and the methodology bounds the containment pressure by delimiting RPV pressure after the RPV in containment pressure equalize. However, using both the limiting RPV pressure and containment pressure for SB LOCA.

The long-term containment response evaluation raises a distinct possibility of break flow reversal from the containment back into the RPV with no -- with non-condensable gases from the containment entering the RPV and degrading the isolation condensers. This has led to the development of limitation with condition No. 3 that essentially requires that the BWRX-300 design has to be

demonstrated for no safety-significant break flow reversal.

There were soil and water containment heat transfer modeling shows that involve significant differences between GOTHIC-predicted dome and PCCS heat transfer reserves and the confirmatory analyses reserves and the justification that was needed for the flow direction-dependent condensation and natural convection heat transfer biases used in the model.

In order to address these concerns the applicant deep dived into the GOTHIC decks and identified immediate error in the PCCS condensation heat transfer modeling. This was a significant correction to a large number of plots in the LTR that led to the revision of LTR Revision 1 to LTR Revision 2.

The staff also issued an RAI on PCCS modeling and performed a PCCS confirmatory study to validate the PCCS sensitivity study presented in the LTR. In the RAI response the applicant did mention that even though they have presented the demonstration analyses with a specific PCCS configuration they may use a different PCCS configuration for the final design. In this backdrop the staff has introduced Limitation and Condition No. 4 for the applicability

to the final PCCS design for licensing basis.

So far we have described the TRACG and GOTHIC codes and the evaluation methodology the applicant used for the BWRX-300 design. Now this slide shows a schematic of the overall approach by the NRC staff followed for using the TRACE and MELCOR codes to develop the confirmatory models. We evaluated the methodology for calculating the containment pressure and temperature response.

In this application the applicant has used TRACG to model the reactor systems and used the atmosphere as a pressure boundary condition relevant to the containment vessel to calculate the mass energy release through the break. The mass energy release is then used as a flow boundary condition in the stand-alone GOTHIC containment model. Basically it's a decoupled calculation between GOTHIC and TRACG codes.

However, to investigate the detailed response of GOTHIC calculation the staff developed several containment models using TRACE and MELCOR codes using the same TRACG mass energy release as boundary condition. The staff calls these models as stand-alone models. These models include both lumped models such as TRACE-CONTAIN and MELCOR-based models and 3D fine-node models that is based on TRACE also.

So there were three types of stand-alone confirmatory models that we developed.

The staff also developed the TRACE reactor system model balanced to the applicant's TRACG model. Like the applicant's TRACG model the reactor system in the staff's TRACE model interfaces with atmosphere and calculates M&E for a stand-alone containment model. The staff calls this as decoupled model. However, investigate the feedback of containment pressure on M&E during the LOCA transient the staff directly interfaced the TRACE reactor system model with base containment model which we call the decoupled model. So in case of TRACE we have coupled model that we can use to study any feedback that the containment pressure may have for the RPV.

So with these three different approaches; that is, the stand-alone, the decoupled, and the coupled calculation, the staff was able to develop insights and validate important findings that are summarized on the slide.

In summary, the confirmatory analysis showed that the proposed CE methodology based on TRACG and GOTHIC codes is converted and confirmed that the containment response is sensitive to nodalization, break location and orientation. It demonstrated that

the accumulation radiolytic gases in the isolation condensers is possible during LOCA which has led to Limitation and Condition No. 1.

Confirmatory analyses underscored the need to have an isolation condenser return line water trap which led to Limitation and Condition No. 2.

The confirmatory analyses also showed that flow reversal is possible from the containment to RPV if the PCCS heat removal capacity is not sufficient. It led to the development of Limitation and Condition Nos. 3 and 4.

MEMBER HALNON: So, on limitation four, you said earlier it was a significant error. Did that call into question their QA as another thing that might be looking at to have a significant error provided to the staff, and you guys catch it, which is good, but not good on their part. Did you guys look in to make sure that their QA for establishing these calculations in the first place were good, or do we know how the error was made?

MR. HAIDER: We understand what the error was, and how for us to correct. And also the error was a concern with -- they were under stimulating the condensation heat --

MEMBER HALNON: Okay, but an error is

still an error.

MR. HAIDER: Yes, that's right.

MEMBER HALNON: So, it could have easily gone the other way if it was an unknown thing. So, did you ask the question how did this occur, and what kind of corrective actions have you made?

MR. HAIDER: Yes, we did have the discussions --

MEMBER MARCH-LEUBA: Okay Charlie, you can come ahead, and get the microphone. Charlie Heck, say your name.

MR. HECK: Yes, this is Charles Heck at GEH. I want to say that this error was discovered as a result of us continuing to dig into what we found as sort of a discrepancy in the results. So, I think it's mischaracterized to say that it was completely discovered by the staff. We were both questioning during this long part, we were both questioning, and looking at the results.

But the error itself was actually discovered by us as we've been trained to do, questioning adding to what the results were telling us.

MEMBER HALNON: Okay, discovered nevertheless, did you guys enter this into your

corrective action program, and establish how it was an error in the first place, and correct any process, or personal performance that occurred during this timeframe?

MR. HECK: The corrective action program has to do with -- at least has to do with errors that escape our release, which was not in that category.

MEMBER HALNON: Okay, did you still figure out why it happened, and who incorrected, even though you didn't have to put a corrective action program?

MR. HECK: Yes, we did correct -- we had verification processes ongoing, and we corrected those, and we continued to verify. Those results ended up in the Reg 2.

MEMBER HALNON: All right. I'm left wanting, but I'll defer to the chairman.

CHAIRMAN REMPE: What was the root cause for the discrepancy that was an error Charlie?

MR. HECK: I'm going to let Necdet address that, because I think he can do it better than me. So, please Necdet.

MR. KURUL: This is Necdet Kurul, GEH. I can describe the error in detail in a closed session.

MEMBER HALNON: I'm interested in the organizational aspect, not necessarily the specific

detail of it.

MR. KURUL: It was a very subtle feature in the code that we are using, we've got it. It was not obvious which models use what inputs, and so it was overlooked, and the results were looking fine, it was more limited than we expected to see. And so the results were more limiting than we expected to see what they were. And so we took those results as they were.

MEMBER HALNON: So, I'm going to just put my management hat on, did you go back and check, is there other features in the code that could be overlooked because of subtleties, or experience of the users, or any of that situation?

MR. KURUL: Yes, we have been looking at those all along actually. This was just one that was difficult to detect actually, it took us some time to find out.

MEMBER HALNON: Okay, we can leave it there, the message, I hope, has been --

MEMBER MARCH-LEUBA: No, let's follow up. I'm thinking here, the whole industry, us included, are trying to minimize costs, and maximize the sureness of the time to design reactors. But was part of the error the fact that you were rushed into

submitting this LTR? Because what you're telling me is you submitted an LTR, and in power you kept looking. Is there something that you can say in that respect at the high level?

MR. KURUL: Well, no, actually we have taken the time to submit the LTR. Even before we submitted the LTR, we have spent a lot of time developing. The reason why we didn't catch it in the first place was that the results were very conservative. More conservative than what we could calculate by our hand calculations, and such.

MEMBER MARCH-LEUBA: So, you were happy with the results, and didn't look further because you were happy with the results, happy in the sense that they were conservative. The code was calculating something worse than you had expected?

MR. KURUL: That's correct. It's basically the reason why we didn't keep thinking in the first place.

MEMBER HALNON: I would argue the error was not conservative, because it ended up in having a design change. Why would we have a limitation condition if it was --

MR. THURSTON: So -- Carl Thurston. So, that particular issue was not related to design

change.

MEMBER HALNON: Okay, well it ended up in a limitation.

MR. THURSTON: It did end up in a limitation.

MEMBER HALNON: So, the design change aside, I guess the point is when you overlook something, the root cause is either skill, experience, going too fast, not enough reviews, that sort of thing. And I would hope that you've gone back to your process, and figured out which one of those, or combination of those caused this overlooking. And if it was because we were happy with our results, that's not a questioning attitude.

So, at least insufficient. So, those are the types of management things I was hoping to hear as we went back, and looked at it. Instead, I think all we got was we corrected the error. So, I just wanted to sum that message up, we don't need to continue on, thanks.

MEMBER MARCH-LEUBA: And a message for the staff, that even though the applicant is under a PSSP and they have to do the reviews on the peer reviews, and the reviews of the peer reviews, doing something last line of defense, the staff does the final peer

review. So, please keep all this in mind, and keep a questioning attitude.

CHAIRMAN REMPE: I have a question before you leave, it goes along with what Greg said. It sounds like -- correct me if I'm wrong, but this is a certain model you had not used in the code until you were trying to apply it to this design, is that a true statement?

MR. KURUL: Yes, that's a true statement.

CHAIRMAN REMPE: So, it was being unfamiliar with that particular model is, again, probably a root cause.

MR. KURUL: As we mentioned, there are several causes, that might be one of the causes.

CHAIRMAN REMPE: Perhaps, then, there isn't a concern this could -- although you may have other aspects of what you're trying to apply here that you're not familiar with, you don't have to worry about well we may have been incorrectly analyzing our isolation condensers in the operating fleet, this is something limited because of this application to this design?

MR. KURUL: Yes, it is actually limited to PCCS, not even the containment surface, condensation on containment surfaces is not related to that either.

CHAIRMAN REMPE: Okay, thank you.

MR. HECK: This is Charlie Heck, I'd like to also interject that there was a component to this related to how you do the nodalization of the containment in how the PCCS units are placed within that nodalization. So, the error that was discovered manifests itself differently depending on the nodalization. So, that's another subtlety that made it difficult to find.

MEMBER MARCH-LEUBA: Okay, I think we've peeled the topic enough. So, please continue.

MR. HAIDER: Another thing that the staff would like to add is that after they made the correction, they resubmitted the reserves, and they communally brought the predictions, the GOTHIC predictions. So that's something that I would like to bring into notice. That confirmatory analysis did play a role in finalizing this issue, around this issue.

MEMBER MARCH-LEUBA: And not finding it, but confirming the solution was likely correct.

MR. HAIDER: That's right, because it was the confirmatory analysis in the first place that had led the staff to think that there was something that was not right, because the heat transfer and PCCS were

not matching between the confirmatory predictions, and the GOTHIC predictions, but then they came closer.

MEMBER MARCH-LEUBA: So, now I will ask Carl Thurston to describe the first two RPV specific limitation conditions.

MR. THURSTON: Yeah, so this is Carl Thurston again. So, these limitation conditions were added to the SER, and I explained the reasons why we added them. I don't think we need to go through word by word, but the first one again is related to the modeling of radioactive gases. And again, I reiterate that the isolation condensers are the primary, the only method of K heat removal.

So, they have to work, and any degradation of the ICS performance is a significant issue. The second issue is --

MEMBER MARCH-LEUBA: I know we're a little late, but I'm going to make it even later. When you said the ICS is the only method of K heat removal that we take credit for in passive mode. In reality the glow was open and valves and used the main condensers, right? And especially when we look at risk analysis, we'll have all the branches of the variations.

MR. THURSTON: That's correct. But for our analysis because we used the safety systems --

MEMBER MARCH-LEUBA: Chapter 15, you're 100 percent correct. And one concern about the Chapter 19, you have to do everything.

MR. THURSTON: Right. So, like I say, both of these limited conditions are based on potential degradation of heat transfer to the ICS. And so the second one again is related to the piping design for the ICS return line to make sure again, that the gases are not able to escape, and go back to the seam, back into the ICS tubes. So, with that I'll turn it -- go to the next slide, and turn it over to Syed.

MR. HAIDER: Thank you. So, this slide provides the whole containment specific limitation condition number three, and four that the staff had imposed on the application. And the VWR standard containment evaluation methodology as observed on the staff review, limitation condition number three emerged out of the potential degradation of the isolation condenser heat unit for performance should there be a reverse flow with non-condensables from the containment to the RPV.

Such a flow reversal could take place in the PCCS if not properly sized to deal with the most limiting flow reversal conditions. However, at the

licensing stage, the applicant would need to demonstrate that either no reverse flow could occur, or any reverse flow that occurs under the most bounding flow reversal conditions is not safety significant with respect to the methodology's acceptance criteria.

And limitation number four requires that if any item of PCCS, our design configuration and placement is used on the VWRX300 design at the licensing state other than the one presented in this LTR, the applicability of this rapid, and the PCCS modeling approach would have to be reviewed, and approved by the NRC.

MEMBER MARCH-LEUBA: And you don't mean by this that sizing the PCCS a little larger with the same configuration type, that's not what this limitation condition applies. If you go with an air cooled system, completely different, is that correct?

MR. HAIDER: Yes, that's my understanding, or a different geometric design. Next slide, please. In this presentation we have discussed the later design issues, and compare records involved in the staff review of the methodology. And finally we present the staff conclusions for our LTR review. In summary, the proposed VWRX chamber in analytical

approach, and the type G and GOTHIC modeling of the mass and energy released and containment response are acceptable with the appropriate conservative biases used in the LTR.

With the four limitations, and conditions specified in the SER, evaluation methodology presented in the LTR revision two is acceptable for the VWRX standard heat containment, pressure, and temperature analysis, for the design, this is excellent. The NRC staff will be required to evaluate the regulatory compliance of the final VWRX standard containment design using the containment evaluation methodology in future licensing activities.

In accordance with 10 CFR Part 50, or 10 CFR Part 52 as applicable in the application. This concludes the staff presentation, thank you very much for your attention. Now the staff would like to address any additional questions the committee members might still have.

MEMBER MARCH-LEUBA: Yes, I have one question above your pay grade. I like the approach of going with a topical report, so by the time we're done with the topical SERs, all the hard questions have been answered. So, we are going to -- I suspect we're going to have a very accelerated review of the SAR.

Is there any planning -- are we going to have a six month review of an SAR? Or it's certainly not going to be six years.

MR. HAIDER: I would like to defer this question to Becky, or --

MS. PATTON: I was going to defer the question to Mike --

MEMBER MARCH-LEUBA: Becky, state your name.

MS. PATTON: Rebecca Patton. Mike, do you want to take that one, or do you want me to? Okay, all right. So, I will say that there continues to be a working dialogue I think, between -- within the projects branch, and we obviously in the tech employee area have been put into that as well. And it takes into account the applicant's needs. But it's my understanding none of that has been set, or decided on.

And obviously we don't determine a review schedule until an application is actually tendered. So, there's some initial planning based on scope of changes for budgeting, and other considerations. But the actual schedule doesn't get set until after then.

MEMBER MARCH-LEUBA: But some interaction with the projects branch, or section, or whatever they

are, is really worthwhile, because if the technical staff is expecting that they would need to perform confirmatory of the final number crunching, you cannot do a three month review.

MS. PATTON: Yeah, absolutely. Mike Dudek, and I talk very frequently, as well as the other branch chiefs, and --

MEMBER MARCH-LEUBA: Unfortunately you've been on the receiving end of this, that products tells you you've got until October, and I want your SER then. And it'd be best to be productive, if you anticipate something to have a backup, to have planning for the review.

MS. PATTON: Right, we do that even years in advance right, when we first hear it comes down. I'm just letting you know, that does happen on a very frequent basis where we have dialogue.

MEMBER MARCH-LEUBA: Excellent, anymore questions from the members?

MEMBER HALNON: Yeah, just a comment. Notwithstanding my rant earlier, the topical report, and the SER were very well done. I want to say you guys, something that I could understand, which was not easy, that it's written, from my perspective, comprehensively.

MEMBER MARCH-LEUBA: Thank you.

VICE CHAIRMAN KIRCHNER: Just an observation Jose, that when the actual application comes in, one would want to look at a spectrum of break locations, and sizes such that you could ensure that no ingestion of nitrogen from this inert containment degrades the performance of the ICS.

MEMBER MARCH-LEUBA: So, the staff recognize, I think we all recognize that the ingress of non-condensable nitrogen into the vessel would be really bad. So, that's again, what I was talking about, about reserving some time for the review. Because the temptation will be to run only the limiting local that they've already found, where the final design is still doing the whole break spectrum of scenarios.

And it's something that you guys, now that you have the review fresh in your mind, it would be good if you provide yourself a to do note, and tell Mike this is going to take this many months. Anymore comments, questions? While you think about it, I'm going to open up the line for the public, members of the public. Anybody wants to make a comment, not questions, but comments.

Please do so this moment. If you are in

teams, just unmute yourself. If you are on the phone line, you need to use star six. Five second rule, we have no questions. Chairman, the floor is yours.

CHAIRMAN REMPE: Thank you. At this point we're going to go off the records, and we'd like the court reporter to come back for the Point Beach subsequent license renewal application. And how about we take a nine minute break, and come back at 2:15 east coast time to read in your draft letter? Okay, so we're going to recess.

(Whereupon, the above-entitled matter went off the record at 2:07 p.m. and resumed at 3:00 p.m.)

CHAIRMAN REMPE: Okay, it's 3:00 p.m. here on the east coast, and we're back in session, and on the record. I'd like to call on Member Matt Sunseri to lead us through the Point Beach subsequent license renewal discussion today. Matt?

MEMBER SUNSERI: Thank you Chair Rempe, and good afternoon members, and attendees. This session is for the Point Beach subsequent license renewal application review. I'm happy to say this is the first time that we are exercising our one step versus two step review process for subsequent license review applications that don't have any open items, confirmatory items, or other issues requiring a more

in depth inquiry.

The way this will flow, is today we will hear from the applicant NextEra, and staff on their safety review. Following the presentation, we'll ask for any public comments, then the committee will discuss any significant elements that should go into our letter report. At our May full committee meeting, I will present a draft letter report for the final deliberation by the committee.

So, are there any questions about the way this will proceed today? All right, two quick points before I turn the floor over to VNR, excuse me. First off, I want to thank the staff, and the applicant for the thorough, and comprehensive reports, and material that they've prepared for this review, and this meeting. Clearly the experience base with subsequent license renewals is continuing to mature, and the quality of the material is excellent, and I'm really looking forward to today's presentations.

Secondly, due to some outside work that I do with a private company, I will recuse myself from any deliberations on material analysis of the primary system components associated with this review. At this time, I now turn to Mr. Brian Smith, division director for New and Renewed Licenses for the

introductions, and the presentations. Brian, the floor is yours.

MR. SMITH: All right, thank you Member Sunseri. And good afternoon Chairman Rempe, and members of the ACRS. As Member Sunseri said, I am Brian Smith, the director of the Division of New and Renewed Licenses in the Office of Nuclear Reactor Regulation. We appreciate the opportunity today to present to the ACRS the results of the staff's review on the fifth application for subsequent license renewal.

This application was submitted by NextEra Energy for the Point Beach Nuclear Power Plant Units 1 and 2, located in Two Rivers, Wisconsin. By way of the background, Point Beach Units 1 and 2 received approval for their initial license renewal from the NRC on December 22nd, 2005. The NRC review at that time was performed using the guidance contained in the initial issuance of the generic again lessons learned report, or the GALL report.

The NRC guidance for license renewal, and subsequent license renewal over the years has evolved through enhancements, and improvements based on the lessons learned from NRC application reviews, and from consideration of both domestic, and international

industry operating experience. The initial GALL report for license renewal went through two revisions. Additional interim staff guidance issued following revision two.

GALL revision two, along with the ISGs in place at that time was used in the activities to develop the guidance for subsequent license renewal that's contained in the GALL SLR report. In addition to the previous license renewal guidance, the GALL SLR report included additional focus on aging management, and time limiting aging analyses for operation in the 60 to 80 year time period.

The staff recently initiated work to develop the first revision of the GALL SLR report. We will incorporate the current four ISGs that have been issued. We'll include updated new, or clarified staff positions, and we'll add improvements that have been identified during previous reviews. We will interact with the ACRS as we proceed throughout this process.

The NRC project manager for the Point Beach subsequent license renewal application review is Bill Rogers. Bill will introduce the staff who will be presenting, and addressing the questions regarding the statute review. Part of the management team here with me today is Lauren Gibson, chief of the License

Renewal Projects Branch, and branch chiefs for the staff involved in the technical review, including Joe Colaccino, Steve Bloom, and Angie Buford.

Dr. Allen Hiser will also be available to address questions from the committee. In addition, John Bozga, Region III senior reactor inspector will discuss the regional inspection activities. I'd like to note that the staff completed its review with no confirmatory, or open items in the safety evaluation. Finally, we will address any questions you may have on the staff's presentation, and we look forward to a productive discussion today with the ACRS.

At this time I'd like to turn the presentation over to Mr. Bill Maher, NextEra director for nuclear licensing projects to introduce his team, and commence the presentation. Mr. Maher?

MEMBER SUNSERI: You're on mute.

MR. MAHER: Trolling. All right, Steve had a couple of buttons, I was told I'm muted, sorry. All right, I appreciate the time for both ACRS, and staff enable for us to be able to prevent the results of the staff's review on the subsequent license renewal for Point Beach. Next slide. So, we're on slide two. So, we'll go through introductions as we go, once I go through the agenda here, and we'll get

started.

So, next will be a performance philosophy with Mike Strope, who is online also, as the site vice president he'll present that to ACRS. Then Steve Franzone will go through the site overview, and then that'll turn over to Steve Hale, who will do the process that we used for subsequent license renewal evaluation. Also, there'll be specific topics of interest that we'll talk through, and then we'll go through closing remarks.

Again, I'd like to appreciate the staff's time, and the thorough review that they've done for us. What we're going to present to you is the results of their review, as well as what Point Beach is doing to maintain adequate safety margins for the subsequent period of extended operation. So, my name's Bill Maher, I'm the senior director of licensing for nuclear projects for NextEra Point Beach. Mike, would you like to introduce yourself?

MR. STROPE: Good afternoon. I'm Mike Strope, I'm the site vice president at Point Beach.

MR. FRANZONE: And I'm Steve Franzone. I'm the licensing manager for subsequent license renewal for Point Beach Units 1 and 2. And then Steve Hale?

MR. HALE: Hi, my name is Steve Hale, I'm one of the technical leads for ENERCON for the license renewal process. I had the same role for the Turkey Point SLR. Prior to that I was at NextEra for over 46 years, had several roles including the engineering director at Turkey Point, and I was directly involved with Turkey Point, and St. Lucie original license renewal, as well as the Point Beach ECU.

MR. FRANZONE: Thank you. Maribel Valdez?

MS. VALDEZ: Yes, good afternoon, my name is Maribel Valdez, and I am the reactor vessel internals program owner.

MR. FRANZONE: Thank you. And Anees Udyawar?

MR. UDYAWAR: Good afternoon, my name is Anees Udyawar, I work at Resting House on fraction mechanics, and I'll make a brief presentation on the RDO structural steel supports, thank you.

MR. FRANZONE: Okay, and back to you Bill.

MR. MAHER: Thank you. And I'd like to introduce Mike Strope to give some opening remarks, and we'll go to slide three.

MR. STROPE: Good afternoon everyone, appreciate the opportunity to speak to the committee. Point Beach is one facility of the NextEra fleet we

operate for our nuclear excellence model. We have the desire to be a team that delivers consistent, and excellent performance on a sustainable basis. We execute on our values based on our core principles, and we do so in four strategic focus areas of operational excellence, organizational effectiveness.

Generation reliability, and effective business, and financial performance. At Point Beach specifically, we use the slogan that Point Beach is a team, it's not a place. And we execute on those strategic focus areas through three site focus areas, those being people, behaviors, and never satisfied. We want to engage our people, and build the best team that we possibly can.

We're very focus on behaviors, both meeting our standards, and also as well as being never satisfied on what we do, and knowing that we can always do it better. Next slide, please. The information here I will not read to you, but our plant performance, based on our plant capacity factor, has been sound. The areas with lower percentages are related to when we have had planned refueling outages.

Our last reactor plant trip was associated last summer. Other than that we've had breaker to breaker runs for the last two outages on both units.

Our regulatory status is as listed, and noted in previous conversations when we're introducing the presentation. Thank you.

MR. FRANZONE: Yes, good afternoon again, this is Steve Franzone. Again, I'm the licensing manager for subsequent license renewal. Just to note, please stop us at any time during the presentation if you have questions, and as a reminder for our team, please introduce yourselves when you speak. This slide, as well as the next few slides provides some facts, figures, and photos which will provide a brief overview of the site.

Turn to slide six now. Point Beach Units 1 and 2 are located on the shores of Lake Michigan, near the city of Two Rivers, Wisconsin. We are one of the largest single sources of electricity in Wisconsin. For example, we generated about 14 percent of all the electricity created in Wisconsin in 2019, and powered 950000 homes, and businesses. The star in the center of the circle marks the location of the site.

It is located approximately 29 miles southeast of Green Bay, Wisconsin. Point Beach has two Westinghouse two loop NSSS systems, as well as Westinghouse electric turbine generators. The cooling

water is pulled from Lake Michigan. The black dashed circle represents the 50 mile radius from the plant, and finally, as you can see on the figure, that the decommissioned Kewaunee Power Station is located due north of the site.

Please turn to slide seven. This photo provides a view of the entire site. North is to the top of the figure, Lake Michigan is on the right hand side of the photo.

A little history about the site. Site construction started in 1967 and 1968, Units 1 and 2, respectfully. Unit 1 received its operating license in October of 1970, followed by Unit 2 in March of 1973. The Unit 1 steam generators were replaced in 1983; likewise, the Unit 2 steam generators were replaced in 1996.

This figure is taken from the environmental report. The red line identifies the exclusion area boundary, which coincides with the site boundary. With the dotted line in the inside circle marks the protected area boundary.

The site buildings are located in the light red, the Point Beach independent spent fuel storage installation is located northwest of the units, as you can see. I'll turn to slide eight.

Point Beach initial license renewal was approved in December of 2005, and was based on GALL Rev Zero. One of the lessons learned from our experience at Turkey Point was to conduct the NEI aging management effective review early.

In fact, the site completed the effective review in 2018, which helped our project team complete our feasibility study in 2019. At the risk of stealing Steve Hale's thunder, on slide 18 the review concluded that all AMPS continue to be effective with no failed elements. He will also be discussing results of the phase four inspection on the same slide.

One challenge we will be facing is, given that the current license for Unit 1 expires in 2030, Unit 1 will already be in a few years into the 10-year pre-window of the subsequent period of extended operation, if we obtain approval of the subsequent license renewal application in the next couple of years.

I'll turn to slide nine. Slide nine and ten provide a listing of modifications and upgrades for the plant since the first license renewal. Although not a comprehensive list, we identify some of the major modifications to the unit. As you can see

by these two slides, we have made significant investment in various plant systems, and components to allow the safe, and efficient operation of the units into the future. For example, we added two new motor driven auxiliary feed water pumps, but turned the existing auxiliary feed water motor driven pumps into additional non-safety backups.

Thereby improving system reliability, they are now referred to as the standby feed generator pumps. I'll pause a moment for questions before we go to the next slide.

MEMBER HALNON: Steve, this is Greg Halnon, you mentioned the AMPS continue to be effective. I was struck by how many revisions to the AMPS were made in the table. And many of those revisions look like they already should have been into a decent, good program. Can you help me with that perception, is that incorrect, or? Maybe you can answer the more direct question, why were there so many revisions required?

MR. FRANZONE: I'm not sure I quite understand the question, are you saying --

MEMBER SUNSERI: It's exceptions is what you're talking about Greg, right?

MEMBER HALNON: Right.

MEMBER SUNSERI: There's a lot on here with exceptions.

MEMBER HALNON: And just the table, I guess in the SER.

MR. FRANZONE: We have a slide, I can go to that slide, or if you want we can hold until we get to that slide, and we can follow up.

MEMBER HALNON: We can hold, that's my lingering question. We can hold, if it seems appropriate I'll ask it later.

MR. FRANZONE: Yeah, when we get to that slide we can talk to that.

MR. HALE: Yeah, I will address that as well.

MEMBER SUNSERI: So, Steve, one of the things that was mentioned earlier is you replaced the steam generators, can you go over those dates again, when the steam generators were replaced?

MR. FRANZONE: Yes. The steam generators were replaced -- Unit 1 was replaced in 1983, while Unit 2 was replaced in 1996. The Unit 2, actually, at the time in 1983, they did a sleeving, I guess, versus a replacement at the time. And that allowed them to continue operating until 1996. And overall, the steam generators have very few -- two plugs for each unit.

I want to say it's like 14 total on Unit 1, and I think five or six total on Unit 2, total.

MEMBER SUNSERI: Thank you.

MR. FRANZONE: Yeah, five total on Unit 2, I'm looking at the numbers now. And we'll go to slide 11 now. Yeah, slide 11. The extended power up rate was a multi-year process which culminated with an approximately 17 percent thermal power up rate in 2011 for both units. As you can see, we had quite an extensive list of modifications.

In fact Steve Hale, one of the ENERCON technical leads, was actually our NextEra lead for the Point Beach extended power upgrade, and is familiar with many of the issues associated with such a large effort. And we'll go to slide 11. Preparation for the application started in 2019 with a detailed feasibility study, and we submitted the application in November of 2020.

The project team has many years of license renewal, and Point Beach experience. This is essentially the same team which successfully completed the Turkey Point subsequent license renewal. Additionally, our team had our third application for St. Lucie Plant accepted for review last year, and we have just finished our break out session for that

project.

The multi discipline team consists of ENERCON as the lead preparer for the submittal, as well as Westinghouse, Framatome, and Structural Integrity Associates in supporting roles. Also the project team was supported by fleet, and site program owners, as well as various subject matter experts, such as Scott Boggs, who is the owner's group representative.

Or Maribel Valdez, who is our reactor vessel internals program owner. Every aging management program for SLR was actually assigned a program owner to support a portion of the application preparation in NRC review. The project team generated over 100 reports which supported the application. These reports not only provide the next level of detail for various aging management programs, and other parts of the application.

They will also provide a way to ensure the knowledge will be passed on to personnel who will need it in the future. As part of our implementation process, these reports will be incorporated into the plant's licensing basis as control documents. The technical leads for our project partner, ENERCON, are both former NextEra employees with almost 80 years of

combined experience with NextEra.

Both developed the original license renewal applications for Turkey Point and St. Lucie. In addition, we have to establish a subsequent license renewal lane of opposition now in order to ensure we will successfully transition to the subsequent period of extended operation. I will turn the presentation over to Steve Hale, who is one of the technical leads I just talked about. We'll go to slide 12.

MR. HALE: Thanks Steve, this is Steve Hale again here for ENERCON. For the Point Beach subsequent license renewal application, we followed the guidance of NEI 17-01, which was developed specifically for subsequent license renewal. We also reviewed REIs, and responses from first three SLR reviews, which included Turkey Point, Surry, and Peach Bottom.

Other activities we conducted to ensure quality, subsequent license renewal application included extensive interviews with that AMP owners both on site, and at the fleet level, and in several pre-application meetings with the NRC. As Steve indicated, the input from the program owners is very important to ensuring we have an informed review of the operating experience at the site, as well as

ongoing efforts with regards to AMP implementation.

Our approach going in was to comply with NUREG 2191, and 2192 to the greatest extent possible, and we believe we have been able to accomplish that. We, along with the NRC, worked diligently to ensure we met the SLR goal of an 18 month safety review schedule, and this was accomplished with issue of the SER without any open items in February. Go to slide 13 Steve.

Having been involved with original, and now subsequent license renewal, I have a unique perspective as to what is involved with the integrated plan assessment, or IPA for both efforts. We thought the best way to present our methodology for Point Beach was really to discuss the differences between original license renewal, and subsequent license renewal.

For scoping, and screening, there were minimal challenges, because the criteria in 10 CFR Part 54 really hasn't changed a lot. We do have to address the modifications, as you saw in the slide Steve presented were quite extensive, and we also needed to do some updates relative to what we call the A2 scoping, and screening criteria that has to do with non-safety, which can affect safety.

And this has to do with the fact that when Point Beach originally got their license renewal approval, there was an absence of specific guidance documents related to alpha two scoping, and screening. As we moved into the AMRs, and AMPs, you really see where the differences between license renewal, and subsequent license renewal reside. Point Beach, as Steve mentioned, was a GALL Rev Zero plant.

And with the issue of GALL Rev One, Gall Rev Two, subsequent ISGs, and GALL SLR, there are a number of aging effects we needed to address beyond what was addressed originally. The most significant differences however are in the aging management programs. Point Beach originally had 27 aging management programs, and for subsequent license renewal, we're up to a total of 48.

If you go to slide 14 Steve, you'll provide some of the specifics there. With regard to our consistency with GALL, we were over 98 percent consistent with the over 2500 line items in the GALL AMR tables. And of the 48 aging management programs for SLR, 9 are new, and 39 are existing, and all were evaluated against the GALL AMP requirements on an element by element basis.

Differences were addressed either through

proposing enhancements that were captured as commitments, or exceptions if necessary. There are ten AMPs with exceptions to GALL, all of which have been reviewed, and accepted by the NRC. Most have to do with specific design features, or characteristics at Point Beach, which require taking an exception, and I'll get into that in the next couple of slides.

There were no plant specific AMPs. One of the things we did, as I mentioned earlier, was addressing RAIs. In each of our AMP basis documents, which are the technical documents that support each AMP, we have a separate section specifically addressing the RAIs for Turkey Point, Peach Bottom, and Surry. And those documents were available to staff to assist them in their technical review.

We'll go to slide 15 Steve.

MEMBER SUNSERI: Let me ask a question here. I think Greg, is this where you had your question earlier?

MEMBER HALNON: Yeah, I guess in the 27, if you go back one slide, 27 programs consistent with enhancement, it's not told whether that number is the significant number of enhancements required on some of the programs. And some of them just struck me as something that should have already been there. For

example, the fire protection AMP, one of them is enhanced plant procedures that indicate that personnel performing fire protection inspections would be qualified to do so.

That just seems rudimentary, that it should have already been there. Another one is, you specify that penetration seals will be inspected for indications of increased shrinkage, and lesser strain, it seems like some of those should still just be there. And there's some cross talk between this one, and the external services. It just struck me that there was -- I guess my perception was it felt like the previous AMPs were not sufficient on some of the most rudimentary type issues.

And that was my perception, I'm not saying that I did enough research to obviously come to a conclusion, but I was hoping that you could help me with that perception.

MR. HALE: Yeah, and what you find that not all programs have all the prescriptive requirements stipulated in GALL. When we do our reviews on an element by element basis, we delve into what the GALL is telling us we need to do, and review it against the specific implementing procedures to make sure those particular aspects are captured. In

some cases, if it's not rigorous, or not specifically called out for, we flag that as an enhancement to ensure that's captured as a commitment for subsequent license renewal.

So, the programs, as they are currently at the site, existing programs, of course were never lined up against the specific GALL requirements. In some cases we may be doing that, but it's not specifically detailed enough, we felt, and the program owners felt in the program commitments. We wanted to make sure that they were done, and they're good for the rest of the plant operation life.

MEMBER HALNON: So, repeating back in short hand, the change from GALL zero, to subsequent license renewal GALL, plus the ISGs, plus just doing a lot more rigorous job on it.

MR. HALE: That is correct, and it really provides a benefit to us, because we have the AMP guideline that we need to address ourselves to. And in a lot of cases, some of these aging management programs we call existing may not have had a GALL program description beforehand. And so that really gives us the guideline that we need to follow, and to make sure that the commitments we capture are good from here on out.

MEMBER HALNON: Okay, that's fair, thank you.

MR. HALE: Okay, next slide Steve, I believe that's 15. I just wanted to show these, these are what we are calling the new AMPs, because these specific AMPs were not around when Point Beach went through license renewal originally. And you'll notice there are, a majority of them electrical. Because when we went into license renewal originally, there was very little in the electrical area.

So, this has highlighted a number of activities, and new AMPs that we'll be implementing for the SBEO. Now to slide, I believe it is Steve, slide 16. Now, to address your comments with regards to exceptions to GALL, we've highlighted some of the specifics here. For example reactor head closer sub bolting, there is a preventative action in GALL that says the bolting should not be high strength from a concern over stress corrosion cracking.

Because the records are not specifically clear as to what the ultimate strength of the studs were, we just defaulted to the conservative, and assumed they were all high strength. And these bolts are looked at under section 11, every outage regardless. However, because the GALL calls out that

preventative action, we took the exception. Another one that may be interesting to you, if you look under water chemistry.

Because we have steam heating around the plant, and that's fed by a boiler to ensure that chemistry is maintained appropriately, and this has to do with a two over one concern, and there is no specific requirements regarding boiler standard heating chemistry. We took an exception to the specific requirements in the EPRI document to secondary steam quality, and ensured that we were following the appropriate boiler standards.

This was also consistent with something that Peach Bottom did. Any other questions here? This is five of the AMPs with exceptions? If there are none, the next slide has the remaining exceptions. One of these is fuel oil chemistry, ENERCON has had quite a bit of extensive experience with implementations at the various sites. And one of the challenges with fuel oil chemistry is some of the smaller tanks.

Like the skid tanks on the diesel cannot be specifically drained fully, and to do a complete internal inspection. So, based on our experience there, we've requested an exception to that. However,

we are doing the fullest extent possible. And then, under the GALL requirements for reactor vessel material surveillance, if you have to change your capsule removal schedule, that's an automatic exception to the reactor vessel material surveillance aging management program.

Now, I hope I've addressed our questions regarding the AMPs exceptions. Typically it has to do with the certain design features at the plant. And when you line up yourselves with the GALL requirements, it's just not possible to comply with them. Any other questions there?

MEMBER SUNSERI: Members, any?

MR. HALE: Okay. Moving on to slide 17 -- I'm sorry slide 18, right? When you look at the override, and Steve said he was stealing my thunder with this, we did conduct an effectiveness review at the site of all the AMPs in May of 2018. This was the lessons learned at Turkey Point, that before you embark on a subsequent license renewal effort, you really need to do the effectiveness review.

It gives you -- I'll call it a benchmark as to how well you're doing. And if there are some issues, you can identify those before you go ahead with a subsequent license renewal project. There is

guidance out there from NEI, it's 14-12. All the AMPs were reviewed regarding those requirements, and all were determined to be effective, with none of the ten elements under each AMP -- none of the elements were considered to be failed.

Also beneficial is the NRC performs a phase four inspection, that's typically done after five years of operation into the period of extended operation. The NRC came in, and they looked at seven of the specific AMPs at Point Beach, and no findings were identified with that inspection. Slide 19 Steve? I'm sorry, slide 20. No, 19, I'm sorry. Okay, for commitments.

We have 51 total for Point Beach. These will be maintained separate from current license renewal, and that's really for clarity, and to avoid confusion. There are two license conditions specifically related to Point Beach in the SER. Those are to incorporate a supplement into the UFSAR, that's already prepared, and ready to go in once we get the subsequent renewed licenses.

And to implement all of the programs, and complete activities including pre-FPO inspections prior to the subsequent period of extended operation. There will be a new chapter, 16, to the UFSAR, which

as I've indicated previously, will be maintained separately. And included in that new chapter is a complete table of all the commitments which will be maintained until they're all implemented.

The project team, both ENERCON, and NextEra have a lot of experience with commitment management, and expectation, so there's a high level of expectation that these will all be accomplished within the required time period. Slide 20. Now, that's all I was going to discuss with aging management programs. Are there any questions?

MEMBER SUNSERI: Members, any questions with the aging management? No, please go ahead.

MR. HALE: Okay, I'll now move into the time limited aging analyses. We did have the dispositions of I, II, and III, some of them did change for Point Beach. And this primarily has to do with the fact that they've issued several GALL AMPs to address specifically some of the TLAAs, like fatigue. And as a result of moving that way some of your analytical dispositions have moved to more of an aging management disposition.

One of the challenging efforts we have is addressing environmentally assisted fatigue. There were a number of vendors involved with that, and the

updates there were primarily due to the fact that there's a number of the guidance documents that have been issued since original license renewal, and we had to adjust ourselves to those updated requirements. We followed the same process of searching the current licensing basis to establish the time limited aging analyses for Point Beach.

We did not identify any new ones for subsequent license renewal. Can you move to slide 21 Steve? The next two slides present all of the TLAAs for Point Beach for SLR. Rather than going into them specifically, we thought that we would leave it up to the committee to pose any questions they might have, and please recognize as you can see, there were a number of vendors involved.

And some of their specific reports are considered proprietary information. So, if we delve into a high level of detail into some of these, we might require a special session for the proprietary information.

MEMBER SUNSERI: I think we're good with this. You have a selected set of topics of interest that I think are going to be most interesting to the committee. So, if we can kind of speed up a little bit, and get to those topics, I think it would be

beneficial.

MR. HALE: Okay. So, with that I guess I'll turn it back over to you Steve.

MR. FRANZONE: Thank you Steve. So, now we have four topics that we selected, reactor vessel internal, and then irradiated concrete, and supports are kind of like one, and the same topic. And then the last is the epoxy resin based grout. And so at this point I'll turn it over to Maribel Valdez, who will talk about reactor vessel internals. Maribel?

MS. VALDEZ: Thank you Steve. Good afternoon again, my name is Maribel Valdez, and in this slide I'll discuss the proposed analysis regarding the gap analysis for the reactor vessel internals aging management program. The site RVI program was recently implemented to implement MRP-227 Revision 1-A, which is the latest NRC approved revision of the guideline.

The GALL SLR allows the use of the existing 60 year RVI AMP if supplemented by a 60 to 80 year gap analysis with MRP-227A as the starting point. When Point Beach was revised, the SLRA was revised to use the NRC interim staff guidance. The ISG that's listed on this slide, on the third bullet, and the fifth bullet. And that allows us to use MRP-227

Revision 1-A as the starting point, as opposed to MRP-227-A.

Also showing on the slide, you'll see the two industry guideline documents that we use to complete our analysis, and the gap analysis. And these two documents, MRP-191, and MRP-2018-022, there pertain to screening, categorization, and ranking of the reactor vessel internals components for a period of 80 years of operation. In the latter part of the slide, you'll also see that we are going to continue to use industry operating experience to inform the program.

And NextEra Energy will continue to actively participate in the joint industry issue programs, and update the RVI aging management program as needed. So, I'll pause here to see if there are any questions. With that, I would like to turn it over back to Steve, thank you.

MR. FRANZONE: We had to move the microphone over. The design configuration of the reactor cavity area at Point Beach consists of a three foot two inch thick biological shield wall, which I'll refer to as the BSW, and it surrounds the active fuel region. This BSW is integral with a six foot six inch primary shield wall, which I'll refer to as the PSW.

The BSW provides radiation shielding for the PSW, but it does not perform a license renewal structural intended function. All of the concrete in this area for Point Beach is 4000 PSI concrete strength. Go to slide 26. This is a diagram of the reactor cavity area at Point Beach. What I've highlighted in beige here is the BSW, or biological shield wall.

And in blue, is the primary shield wall, or the PSW. The reactor is shown here in phantom. Above the beige area, along the blue wall is where the horizontal supports attach to the PSW, and at elevation ten foot is where the column supports are anchored into the concrete. Next slide, slide 27 Steve. Based on the calculations, which were performed by Westinghouse for 72 effective full power years.

Which equates to the 80 year life of the plant for SLR, the neutron fluence, and gamma dose exposures on the biological shield wall at the reactor vessel belt line were both above the NURER-2191, and 2192 thresholds. As a result of that, you are essentially required to go back, and take a look at what the impacts are on the concrete strain.

Based on the methodology that the industry

has been using provided by EPRI, the effects of those exposures on the biological shield wall are as follows. And these assumptions were made as part of our structural evaluation. It's assumed that there's zero remaining strength for the first 3.9 inches horizontally into the biological shield wall, and we apply that to the entire active fuel length.

Even though you do get some drop off, and it's more like a curve, which drops off at the upper, and lower ends of the active fuel region. And this accounts for the neutron fluence effect, including radiation induced volumetric expansion more commonly referred to as RIVE. For an additional 20 inches horizontally into the biological shield wall, we assumed there was 80 percent remaining strength into the BSW for the whole active fuel length.

And again, that's a conservative assumption, since you're applying that reduction in strength both horizontally, and vertically with that full loss. The evaluation confirmed that the BSW will maintain its structural integrity, and not impact the PSW under design basis loading conditions. I would like to point out that as I identified where the supports were actually attached to the concrete, those areas are both left in the NUREG-2091, and NUREG-2192

thresholds.

Those are conclusions based on these evaluations that we performed, that there were no further actions beyond what is currently done as part of the structure's monitoring program. And with that, I'll open it up for any questions you have for the concrete evaluations.

MEMBER HALNON: So, what does the failure look like, is it when it loses structural integrity, does it just crumble, or fall, what is that?

MR. FRANZONE: One design feature I'd like to point out is all of the surfaces of both the BSW, and PSW have a quarter inch liner plate that's anchored into the concrete. So, what we do is we actually look at the strain within the concrete, and when -- this is using the conservative methodology, so the concrete, although it's assumed to have zero strength, the fact that we have a liner plate there would contain any crumbling, or cracking you may have underneath it.

MEMBER HALNON: Okay, so back to the question, what does failure look like? Is it even a consideration that we have to worry about?

MR. FRANZONE: No, because if you'll see, the buttresses, and the steel reinforcing that goes

into the BSW, you will not get a failure. You'll have a reduction in strength of the evaluation we conducted confirm that you would not have a failure.

MEMBER HALNON: Okay, you got it, thank you.

MR. FRANZONE: Yeah.

MEMBER SUNSERI: So, before you go on, I'd just like to remind everybody listening in, if you're not actively speaking, there's some tapping background noise coming from some place, so appreciate that, thanks.

MR. FRANZONE: Okay, so I believe now it's going back to Anees Udyawar, and he will talk about the radiation evaluation done by Westinghouse of the fuel reactor vessel for us.

MR. UDYAWAR: Thank you Steve, and probably that tapping was when I turned on the mic, so my apologies. My name is Anees Udyawar, I'm from Westinghouse, and I work on various fraction mechanics, and ASME Section 11 small evaluation projects. The next set of slides will be on the Point Beach steel support assessments. And the next two slides will provide the basic configuration, and geometry of the supports.

But let me first give a brief background

on the assessment. As part of the SLR program, the Point Beach structural steels were analyzed based on a fraction mechanics evaluation to determine whether the complements are structurally safe with consideration of decreased fracture toughness duly written for 80 years.

The objective was to demonstrate that there is a sufficient level of flow tolerance to justify continued, or current tenure ASME Section 11 in service examinations into the future. The analysis is consistent with the methodology previously accepted by the industry, and the NRC in NUREG-0933, 1509, and NUREG-5220, which had all previously determined that the supports were in general, structurally safe for 40 years, and for 60 years of radiation.

So, on this slide, we see the Point Beach steel structure, which has a six sided structural steel ring, girder, supported at the top by a 19 foot long steel column, which extends down into the interior concrete structure that is below the RPV. The columns are about 12 inches diameter steel cylinders, and those columns support the box we're anchoring.

Let's go to slide 29, thank you. The center of each of the segments of the ring girders

provides lateral, and rotational restraints by the I-beam structural members, which are embedded in the surrounding primary shield wall concrete, and that's another figure on the left here. The six columns of the support structure are bolted at the top of the anchor, and pinned at the bottom to the floor anchors, as you can see in the figure on the right there.

The RPV has six support pads, one of each of the four nozzles has a support pad, and then there are two additional gusset-braced support pads, which are welded directly to the vessel. These vessel supports bears on the support shoe, which is fastened to the support structure. The support shoe is designed to restrain the vertical, lateral, and rotational movements of the RPV, but allow for thermal grill by permitting radial sliding on the barrier plates at each of the supports.

Let's go to slide 30. Now, this slide shows the column supports, these are the vertical columns that go from the ring girder to the bottom of the RPV. This particular support is surrounded by concrete, and this model is coming from the neutron transport analysis, which was used to generate the iron DPA. The industrial guidance requires us to use iron DPA neutron energies above .1 MMEs.

But the evaluation period considered iron DPAs both above, and below .1 MME for a very comprehensive analysis. Shown here is the maximum DPA, 5.4 times 10 to the -4 in the column. But we also calculated other iron DPAs along the entire length of the column, and the ring girder, and to the bottom of the columns to be a very comprehensive analysis.

Both the Point Beach units were similarly evaluated to determine the DPAs. Next slide, slide 31. Some of the columns are not surrounded by concrete, so the DPAs were calculated as well for these column points since the DPA were slightly higher than the columns which were encased in concrete. Based on the iron DPA, the shift, and build up to the temperature was calculated based on regulations provided in NUREG-1509.

And these were previously also used in the familiar investigation. Next slide, slide 32. Lastly, following the guidance of NUREG-1509, the critical source size for each of the ten RPV support complements were determined by setting the applied stress intensity factor equal to the fracture toughness of the material. All the loading conditions were considered, such as normal, upset, emergency, and

faulted.

The critical flaw sizes were then compared with the ASME Section 11 allowable flaw sizes, which is the recommendation for NUREG-1509. In many cases the critical flaw sizes were larger than section 11 allowables. For other cases, we could compare the critical flaw sizes to the initial fabrication requirements of the design specification.

Since there is no significant transient, or thermal cycling at the supports, there will be no significant crack growth over time. Thus the initial fabrication flaw sizes on the order of the critical flaw sizes would have been repaired, or the concrete, and lead would be replaced. Thus the Point Beach supports were evaluated to be structurally safe, and stable, and the flawed tolerance constraint 80 years of radiation embrittlement was considered.

It should be noted that per NUREG-1509, physical examination of the complement is also essential to assess the overall condition of the support structure. Thus, in accordance with ASME Section 11 IWF requirements, the Point Beach Unit 1 supports were examined 2005, 2007, 2016. And the Unit 2 supports were inspected in 2006, 2008, and 2015. And based on all these latest examinations, there are

no relevant conditions identified that were applicable for the fraction mechanics assessment.

So, in conclusion no further action is recommended beyond doing the current ASME Section 11 IWF inspection requirements. This concludes the discussion on the structural steel supports, and if there's any questions, I can field them. Thank you, and I can pass this back to Steve Franzone for discussions on epoxy residue grout. Thank you.

MR. FRANZONE: Yes, thank you Anees. All right, so the next couple of slides, we want to cover several questions from --

MEMBER BROWN: This is Charlie Brown, can you go backwards for a minute? I'm not a metaller, so I'll ask a question. You say in most cases the critical flaw sizes are larger than the allowable flaw sizes by a large margin. And then you said but they're okay out to 80 years, going from 60, to 80 years. Is there a basis for that? Like the stresses are so low that they're not going to increase?

Or the temperature variations are so minor, that you don't get any structural stressing from that standpoint?

MR. UDYAWAR: Yeah that's -- sorry to interrupt. Yeah, that's a good question. Certain

components, the stresses were very low. Some of them were less than six KSI, or even lower, or some of them were compressive. So, in those cases, we were not getting any crack growth on any kind of critical flaw sizes.

In other parts of the complement, if they're away from the belt line region, the fluence embrittlement was also very, very slow. So, also for those considerations, the critical flaw sizes were much lighter. So, you're correct, it's either small stresses, or minimum level of iron DPA for certain locations.

MEMBER BROWN: What's a DPA?

MR. UDYAWAR: Displacement per atoms, it's --

MEMBER BROWN: Okay, thank you, I had never heard that one before. Thank you, appreciate it.

MR. UDYAWAR: Thanks.

MR. FRANZONE: All right, are we ready to go on? All right. Again, so the next couple slides we hope to cover several questions for epoxy resin based grout. What is it? How, and where is it used at Point Beach? Why do we care? And what did we do? So, let's dive right into it. Epoxy resin based grout

consists of a resin, a hardener, and an aggregate. It is designed for a high dynamic load capacity, and improved load transfer, which improves stability, and alignment, and minimizes vibration.

Here at Point Beach, the use of epoxy resin based grout is controlled by the engineering change process. Our engineering change process controls how various engineering products, and activities are developed, and control changes to the plant's configuration. For example, it provides engineering with a vehicle to communicate to the craft, the necessary details to control for instance the surface preparation, and mixing procedures.

And it includes hold points for engineering a quality control verification. Full scale mock ups were used to validate the installation procedures, and demonstrate the use of epoxy resin based grout in these specific applications. The engineering change also provides the proper vehicle to evaluate the change from concrete, and mortar to the epoxy resin based grout.

The use of epoxy resin based grout is not permitted in areas where normal temperatures exceed 120 degrees Fahrenheit, and is not used in areas of high radiation exposure. The process was used for

both the service water -- or this process was used for both the service water, and cooling water pumps. And just by the way, the last component cooling water pump is being replaced this outage.

If you'll turn to slide 34. Almost 40 years ago Information Notice 83-40 was sent out by the NRC staff to their licensees to potential issues with the use of epoxy grout. Two issues were discussed, and I quote, the compounds had shown significant loss of strength at temperatures above 120 degrees Fahrenheit, and relatively low creep strength of epoxy.

The science in this industry has come a long way since those days, and epoxy resin based grouts have proven to be very effective in certain uses. Epoxy resin based grout used as an aggregate, which significantly reduces the effective creep. It also acts as heat sink, and stabilizer. It also increases ultimate strength, and reduces shrinkage, and cracking.

One would have to ask is it better than sliced bread? Current aging management practices for the structured monitoring program, a visual inspection addresses the aging effect discussed on the slide with the exception of creep. To address this aging effect,

we added requirements to conduct checks for tightness. That it initial, and periodic torque checks of all the anchors within the scope of license renewal embedded with epoxy resin based grout.

The issues associated with pump alignment, and vibrations have virtually disappeared. And with that, does anybody have any questions on this issue? Okay, thanks for the opportunity. I will now turn the presentation back to Bill Maher for closing remarks.

MR. MAHER: Thanks Steve. So, even though Point Beach was a GALL Rev Zero plant in the first round of license renewal, we have adopted the SLR Gall with minimal exceptions. Like Mike stated in his remarks earlier, in keeping with the sustainability that he spoke about, the goal now is to focus on building, and maintaining the margin that we currently have in order to get to 80 years of operation.

We have, as part of that, to oversee that maintaining of margins, we have a dedicated individual on site to oversee the implementation of these aging management programs as we transition to the subsequent period of extended operation. Any questions?

MEMBER SUNSERI: Members, any questions for the applicant, NextEra? What about the transition? Okay, so thank you very much for your

presentations, good job.

MR. MAHER: And thank you to the staff.

MEMBER SUNSERI: All right, now we will transition back to the DNRL staff. Brian, your team, are you ready?

MR. ROGERS: Yeah, good afternoon, we're just switching slides.

CHAIRMAN REMPE: So, there appears to be two microphones open, and we're getting a bit of an echo. So, whoever is speaking should have their mic open, and others please mute themselves.

MEMBER SUNSERI: So, Bill it's on your end, if you're talking through your phone you've got to mute your phone, and the computer.

MR. ROGERS: Okay, can you hear me now?

MEMBER SUNSERI: Perfect, thank you.

MR. ROGERS: All right, excellent, thank you very much. Okay, good afternoon Chairman Rempe, and members. Can you see my slides?

MEMBER SUNSERI: Yes.

CHAIRMAN REMPE: Yes we can.

MR. ROGERS: Okay, thank you. Good afternoon Chairman Rempe, and Sunseri, and members of the ACRS. My name is Bill Rogers, and I'm one of the senior license renewal project managers in the Office

Nuclear Reactor Regulation, and we are here today to discuss the staff's safety review of the Point Beach Nuclear Plant subsequent license renewal application, or SLRA as documented in the safety evaluation of NRC.

And joining me at my table today, I have Angie Buford, chief of the Vessels and Internals Branch, who is assisting me with the slides, and presentation. We have additional technical branch chiefs, and staff also on the line, and Dr. Hiser, previously the license renewal SL, and now we've hired a new one since in DNRL that's with me. And also Lauren Gibson, chief of the Projects Branch.

Next slide, please. So, we'll begin today's presentation with an overview of the Point Beach licensing history before moving onto the Point Beach aging management programs. We've already discussed selective technical errors that we believe are of interest to ACRS, and hear from the Region III inspections, and plant material conditions before assuring the conclusion of the staff's safety review.

Next slide, please. The Point Beach SLRA described -- excuse me, one moment. The Point Beach Units 1 and 2 were initially licensed on October 5th, 1970, and March 8th, 1973, respectively. In February of 2004, the applicant submitted the initial license

renewal application. The initial renewed licenses were issued December 2005, extending the expiration dates to October 2030 and to March 2030 for Units 1 and 2, respectively.

On October 16th, 2020, NextEra submitted an SLRA for Point Beach Units 1 and 2. The application was accepted for review on January 8th, 2021, and the safety evaluation was issued on February 23rd, 2022 with no open or confirmatory items.

Next slide, please. The Point Beach SLRA described a total of 48 AMPs, 39 existing programs, and nine new programs.

This slide identifies the applicant's original disposition, and these AMPS as updated in the SLRA in the left column, and the final position is documented in the SE in the right. All of the AMPs were evaluated for consistency with the generic aging lessons learned, subsequent license renewal, our SLR report, and ultimately all of the AMPs were found to be consistent with the broad SLR, including those with enhancements, or exceptions.

The applicant did not introduce any plant specific AMPs. So, I'd like to add a bit about the work we did perform to do the review of the aging management programs, and the other technical

information in the application. We had extensive audit activities, including a review of documents that were placed in a portal -- a real voluminous collection of documents.

We had three weeks of breakout sessions that addressed most all of the aging management programs. We did an extensive OE review, also through the portal. And in addition, the applicant was unable to provide us with some current visual representation of items of interest on the plant and in the plant. And the reason we chose that approach was due to the COVID pandemic that affected our ability to travel during the review.

We'd like to thank the applicant for providing this, an alternative method for us to see things of interest. We had 14 RAI sets, 62 RAIs. I'll note that we only had four second round RAIs from this particular review. The applicant made five SLRA supplements with an annual update. We had six public meetings. The public meeting topics included the irradiated reactor vessel supports.

We had two for reactor vessel internals. We had a third on reactor vessel internals. We had a fourth meeting on epoxy grouted anchors, fatigue amounts and cycles, selective leeching, and cathodic

protection. We had a second meeting, the fifth meeting also included cathodic protection, and a sixth meeting on the fire protection response.

So, based on our review, the results of the audits, and additional information provided by the applicant, the staff concluded that the applicant's aging management program activities end results were consistent with the criteria of the standard review plan, the requirements of 10 CFR Part 54. Any questions on the information I presented so far?

MEMBER SUNSERI: No questions, please go ahead.

MR. ROGERS: Okay, next slide, please. So, this slide represents certain to specific areas of the SLRA review, these are technical issues that we spent some time on. The first bullet refers to staff's evaluation of epoxy resin grout for the embedded fasteners, which had been used to replace the more common cementitious grouted anchors.

During review of the application, and the NRC staff audit reviews, the applicant indicated that epoxy resin grout was used for the embedded anchors in the service water pump and the component cooling water pump base plates. The staff questioned the qualification of epoxy grouted fasteners for these

applications to better understand the potential aging mechanisms.

The applicant provided additional information that indicated that epoxy resin based grout materials have been purchased as safety related from the supplier, and they qualify for use in safety related applications. This data is consistent with the requirements of 10 CFR Part 50 Appendix B. In addition the plant application was subject to an engineering change process review for use in these particular applications like the service water pump, and CCW pump base plate.

As previously mentioned, it was noted in the NRC Information Notice 83-40 identified the need to consider the creep strength of epoxy, and recommended that were anchor bolts are embedded into the epoxy grout, and tensioned to any appreciable preload, it might be important to periodically verify that the preload has not been lost by creep in the grout.

The applicant enhanced the aging management program, establishing the structure monitoring program to address the potential aging effects of creep associated with epoxy grout anchors, and to do so by periodically checking the anchor

bolts, and different epoxy resin based grout, verify that preload had not been lost due to creep.

The remaining four bullets are those areas referred to in the staff requirements SECY-14-0016, ongoing staff activities to assess regulatory considerations for a power reactor with subsequent license renewal. Those four items are reactor pressure vessel neutron embrittlement, reactor vessel internals, radiation assisted stress corrosion cracking, irradiated concrete containment, and electrical cable qualification, and condition assessment.

For each of these four areas, the applicant provided quality information as contained in the SLRA as provided during the core of the staff's review. The staff was able to disposition the applicant's information using the GALL SLR report, the SRP SLR, and the memo on staff guidance, which allowed the staff to obtain reasonable assurance, and made its conclusion through the normal review process with the expected level of engagement.

So, I know we've discussed this a bit previously, but if there are any questions, or further information needed in this area, we have the technical staff currently available, and prepared to respond.

MEMBER SUNSERI: Members, any questions? Doesn't sound like any, go ahead.

MR. ROGERS: Okay, thank you. Well, at this time we're going to turn it over to John Bozga, Region III senior inspector. And he's going to present some slides on these colonies. But to do so, we're going to turn the presentation over to this camera. So, we'll need to pause a moment while John takes possession of the presentation. John, are you ready?

MR. BOZGA: Yes, I am. All right, Mike, thank you.

MR. ZIOLKOWSKI: Yeah, this is Mike Zilokowski, Region III, let me know when you can see the PowerPoint. I will present while John speaks.

MEMBER SUNSERI: All right, we have it up, you might want to put it in presentation mode.

MR. ZIOLKOWSKI: Okay, thank you.

MEMBER SUNSERI: Very good, there it is.

MR. BOZGA: All right, thank you. Good afternoon everyone, I'm John Bozga, senior reactor inspector in the Region Three Division of Reactor Safety Engineering Branch One. One of my inspection duties is to perform license renewal inspections. Also joining me on behalf of the region is my branch

chief Mike Zilokowski, as well as the senior resident inspector for Point Beach, Tom Hartman.

I'm here to provide Region Three's review, and assessment of the implementation of aging management programs, material condition, and overall regulatory assessment of Point Beach Units one, and two. The license renewal inspection program, and the reactor oversight baseline inspection program are both used to inspect aging management activities at Point Beach.

I'll start with the activities performed under the license renewal inspection program, then discuss the LOP inspections, and follow up with material condition of the plant. So, in order to assess the adequacy of the license renewal program of the initial period of extended operation, inspection procedure 71003 recommends using a four phased approach to license renewal inspection. This current slide you see lists the specific license renewal inspections that have been performed at Point Beach.

The phase one, and phase two inspections were performed for both units on the dates listed with no findings identified. Phase three inspections were not required, because no findings were identified in phase one, or two. Finally, the phase four

inspection, which typically occurs five to ten years into the period of extended operation has been performed in September of 2019 for both Units 1 and 2 with no findings identified.

Next slide, please. In addition to the inspections mandated by the license renewal inspection program, inspectors used several ROP baseline inspection procedures to evaluate the implementation of aging management activities. One example is the baseline inspection of the in service inspection program. This inspection is performed each refueling outage, and provides the inspectors the opportunity to review, and assess inspections credited for aging management.

The second example is the heat sink inspection, which provides the inspectors an opportunity to review the service water system, as well as the ultimate heat sink. All of these activities are within the scope of license renewal, additionally the design basis assurance inspection includes a review of aging management activities for safety related structure systems, and components that are selected during this inspection.

At Point Beach, the regional inspectors have found no violations, or findings of greater than

green significance as a result of these inspections performed using these aforementioned procedures. Also of note, the triennial fire protection procedure has been recently updated to review aging management of the fire protection equipment.

Next slide. Currently, Point Beach Unit 1 and 2 are in the licensee response column, and have all green findings in performance indicators. This indicates that the licensee has been able to effectively identify conditions adverse to quality, and correct them in a timely manner. We have reviewed all inspection findings over the last ten years to gain insights related to aging components. We did not identify any findings with an aging management aspect related specifically to license renewal. Next slide, please.

I will now speak to the material condition of Point Beach from a regional inspector viewpoint. We have no concerns with the overall material condition of the plant that needs to be addressed outside of the baseline reactor oversight process. The licensee has been successful at completing large capital improvement projects that maintain, or improve the material condition of its structure, systems, and components.

The license renewal program inspections did not identify any substantial weaknesses in the station's performance in managing the effects of aging at the site. The inspectors will continue to inspect, and assess the licensee's ability to manage the effects of aging through the baseline inspections. Lastly, are there any questions?

MEMBER HALNON: This is Greg. Do you have any feel for where they are on the cross cutting aspects? I understand that there's no better than green findings, which doesn't really give you a full story. Were you seeing any things, or anything producing out of that program?

MR. BOZGA: I'm not aware of anything, I'll let Tom, and, or Mike speak to that as well.

MR. HARTMAN: Hi, I'm Tom Hartman, I'm the senior resident here at Point Beach. I've been here for just about five years now, and we do not have any themes, or patterns, even though they have not triggered any of the cross cutting thresholds, but we do not have any transient themes for any of the areas at this time.

MEMBER HALNON: Okay. Because what I was kind of poking at was back to my earlier conversation about the large amount of enhancements. And I

understand that it changes standards over the years, and it's more rigorous, but there was a lot of procedural issues in there. And I was just curious if there were any procedural themes showing up in the cross cutting aspect.

So, it sounds like these are truly enhancements that aren't affecting the material condition of the plant at this point.

MEMBER BIER: I have another quick question, this is Vicki Bier, and this may be just over interpreting the wording, but it was said that there were no aging management issues specifically related to license renewal. Were there any aging management issues not related to licensing renewal, or am I being overly persnickety?

MR. BOZGA: I'm not exactly sure what we're getting at. We're looking for aging management issues, are we talking about service life issues, or?

MEMBER BIER: I'm just wondering why that choice of wording, and maybe somebody just threw in some extra words that were not needed.

MR. BOZGA: No, the wording I would not say is thrown in there, but sometimes there is confusion, either on the NRC, or even on the industry side with the difference between service life, and an

aging effect. Say a vendor recommends to replace a component every ten years, and the stations decide to go further than that, and the component fails due to some aging degradation mechanism.

I would consider that different than a license renewal aging management issue, which may not have to do anything with service life.

MEMBER BIER: Okay, thanks for the explanation.

MR. BOZGA: You're welcome.

MEMBER SUNSERI: so, this is Matt, and I'll ask the question this way. I really do appreciate, and value what the resident inspectors bring to this process. And so from your individual observations at Point Beach, are there things below what I'll call the regulatory threshold that we might be interested in? Because those kind of things can sometimes be a precursor to challenging a regulatory threshold.

So, let me give you an example, does the plant have a large amount of leaks, whether they're steam, oil, water, gas? Do they use extensive amount of temporary leak repairs, or are their maintenance practices holding the plant together well? Any kind of insights like that that you can share with us, and

we recognize that your obligation is to the regulatory threshold.

But we don't get to see these plants, and so anything you could kind of offer us would be insightful.

MR. HARTMAN: Sure. This is the fourth plant that I've been an inspector at in my tenure with the agency. I actually -- for being as old as it is, in terms of being licensed back in the early 70s, the material condition is actually quite good. The station does a good job of maintaining the situation. There's not a lot of leaks, and material condition issues.

There have been in the past, issues with underground cables that the station was very good to address, and we've documented those in our reports. But since I've been here in the last five years, there's not been any significant concerns that I have about any -- even the low level things like continually leaking -- leak management programs, or stuff like that, having to repair, or do any type of ASME repairs to fuel wall leaks, or any of that kind of stuff.

I've not seen -- as with every plant, they occur, but nothing that I would call astounding,

especially for a plant that's already 50 years in.

MEMBER SUNSERI: All right good. And then from a cultural standpoint, kind of following up with Greg's question about cross cutting issues, how is the staff as far as questioning attitude reporting problems, do you see anything there?

MR. HARTMAN: Again, it meets our standards. I have not seen any issues with -- I do talk to the individuals out in the station, do pulsings of people. I've not seen any issue with safety conscious work environment, or safety culture, they're all willing to bring up safety issues. I'm hoping that's in line with your question. I don't have any concern about the cross cutters.

Again, every station has challenges, communications, procedure use adherence, there's little things here, and there, but nothing that I would call alarming, or trend worthy.

MEMBER SUNSERI: That's good, I did notice in one of the inspection reports there was a commitment that had been missed, but they were identified by the utility that they missed it, and they presented to you that they missed it, the corrective action plan when it was going to be corrected, and when the inspection was going to be

done. So, I mean I thought that was a healthy -- like you said, plants are complex.

They have a lot of requirements, and people make mistakes every once in a while. It's how you deal with those mistakes, and how you react to them. So, it seemed to me from that one data point, that they're pretty responsive. But I just wanted to kind of test you in a more generic sense, you had a larger sample than we did. That was good, you answered my question.

Members, anything else? Okay then, I guess are you about concluded with your presentation for the staff side?

MR. BOZGA: I'm going to turn it over to Bill, so thank you.

MEMBER SUNSERI: Thank you. I'm sorry, we may have lost you if you're speaking, I don't hear anything.

MR. SMITH: Just one moment, we're reclaiming the business.

MEMBER SUNSERI: Got it, thanks.

MR. SMITH: Well, actually I can do this without the slides, that's okay. Can everyone hear me?

MEMBER SUNSERI: Yes.

MR. SMITH: Okay. What I wanted to say for the final slide was thank you. And say in conclusion for the SLRA safety review, the staff finds the requirements of 10 CFR 5429A have been met, and the subsequent license renewal for Point Beach Nuclear Plant Units 1 and 2. And we'll now be happy to answer any additional questions that you might have. Thank you.

MEMBER SUNSERI: All right, well thank you, and thank you to staff. So, members, this concludes the two presentations, one from the applicant, and one from the staff on their safety review. Any questions you have, anything from the staff?

MEMBER HALNON: I don't see anybody nodding their head.

MEMBER SUNSERI: All right, good. Yeah, that's the downside here, I can't see you, you guys can see me, but I can't see you.

All right, then, I suppose at this point we can turn to public comments, and see if we have any comments. So, let's open up the phone line. Members of the public listening in, this is the opportunity to make a comment, so just unmute yourself, state your name, and provide your comment.

Let's open the lines up now. Okay, and not hearing any comments, I'll give it one more second, or two here. All right, and so that's it for the public comments. I will provide the same courtesy to the members in the room there, if there's anybody in the room that at this point cares to make a comment, please do so. All right, well I guess that wraps up our comment period then.

So, members I will now ask Chairman Rempe how you would like to proceed. But at this point in time we are complete with the presentation. If you want to entertain a brief committee discussion on any topics that they would prefer to see in the draft report before I take it out to the subcommittee to prepare that draft report for the full committee meeting in a month, I would appreciate any comments.

MEMBER HALNON: Matt, this is Greg. I think a short explanation of the GALL zero, one, two ISG question I had earlier would help make sure that we understand the large number of commitments that were made, and enhancements to the existing programs. That really stuck out to me, that there was a tremendous amount of it.

MEMBER SUNSERI: Yeah Greg, so I mean we're kind of spoiled I think a little bit by the last

two of these things that we were doing, which had very few enhancements, and exceptions. We have some other reviews where there's similar numbers, but yeah, we can definitely include that in the discussion.

MEMBER HALNON: Yeah, it's this timing thing. I mean this is an older plant license renewal, the initial license happened a long time ago, and it's not a negative at all, I think it actually could be a positive if the --

MEMBER SUNSERI: Yeah, right, got it. It's normally not that the plants aren't doing the things, they just aren't committed to doing them. So, like having qualified people do the work, right?

MEMBER HALNON: Right, there's no indication that unqualified people were doing it, clearly. So, it kind of cements some of the more stronger standards that we see today than we did in the early 2000s.

MEMBER SUNSERI: Perfect, yeah, that's good, I got it.

VICE CHAIRMAN KIRCHNER: Matt, this is Walt, what's your take on the AMP with exception regarding the capsules for the pressure vessel radiation?

MEMBER SUNSERI: Well, so as it was

presented, any exception to the surveillance capsule schedule is an automatic exception. And I think based on our previous topic this day, I imagine they are revising their surveillance schedule to kind of get in sync with those new PWR requests. So, we can explore that a little bit more, but that was my take away on that. So, it's just simply an exception because they're changing their schedule to address the new procedures.

MEMBER BALLINGER: Point Beach has one extension, so it's only had one extension.

VICE CHAIRMAN KIRCHNER: One extension of what?

MEMBER BALLINGER: Extension of the time to take the capsule out.

VICE CHAIRMAN KIRCHNER: Okay, so when is their capsule coming --

MEMBER BALLINGER: I don't know what the date is --

VICE CHAIRMAN KIRCHNER: Because this is an older plant, I think it would be an interesting data point.

MEMBER BALLINGER: Yeah. In my head I seem to remember that Point Beach, and Dave Rudland's gone, but it is the plant that they were plotting the

data points in his presentation. I should go back, and --

CHAIRMAN REMPE: It looks like the licensee has --

VICE CHAIRMAN KIRCHNER: Yeah, Steve Hale, could you answer that?

MEMBER SUNSERI: Steve Hale is here, you're muted Steve, we can't hear you.

MR. HALE: We had one capsule we were going to remove at a specific timeframe to account for 60 years, and we essentially just delayed removal of that capsule so we could get a capsule that had 80 years of exposure. So, we were going to remove one, but in order to ensure that we captured 80 years of fluence, we just delayed the time we would remove it. But the way the GALL AMP is written for the reactor vessel surveillance program, if you have to make a change to your existing capsule removal schedule, that is an automatic exception.

But as part of the application, we submitted all the requirements that you need to follow to change your surveillance capsule removal. We have the specific timeframe, I believe.

MR. FRANZONE: Yeah, it's 51 effective full power years, which right now is estimated to be

in the spring of 2035.

MR. HALE: That was strictly just a delay in the removal of the final capsule to capture 80 years of fluence.

MR. FRANZONE: And that's for Unit 2, that's the capsule in Unit 2.

MEMBER BALLINGER: If I recall, based on the EPRI presentation this morning, 2035, by that time they will have a lot more data from other -- from their program if I recall.

MR. FRANZONE: Correct, it's a living program, so as we move forward, we'll have a lot more knowledge, so things may change again.

MEMBER SUNSERI: Right, no doubt. So, yeah, thanks for that explanation. Walt, does that address your point?

VICE CHAIRMAN KIRCHNER: Yeah, thank you.

MR. MAHER: This is Bill Maher with NextEra Point Beach, we did want to address some of the exception questions --

MR. FRANZONE: Yeah, the enhancement questions.

MR. MAHER: The enhancement portion, so Steve has some discussion on that.

MR. FRANZONE: Yeah. For our purposes, we

don't write the actual implementation procedures until after we obtain approval -- or we actually don't implement them, our implementation procedures are not approved until actually we get the license. And so although we have the enhancements, I think other utilities kind of do it, they actually write their procedures, get the enhancements in place, so they actually have something to show the reviewers.

We don't do it that way, we list the enhancements in our basis documents, which support the application. And I think that may be one reason for say the other sites, like North Anna, or Surry where they actually have gone through, and actually have written all those, and have them place, where we don't do that. So, it may look like we have. There may be a big difference, but it may not really be a difference, it may be just a tiny difference in that.

So, I just wanted to throw that out there. But we do like to give clear directions, and I think we're hard, we get to pass this on, people have to implement this maybe ten years from now. And so we want to make sure it's very clear to the future out there, the people who are actually implementing the feature. And so I think for us, for our benefit, we're very clear, and we're kind of -- I'll say maybe

overly conservative in our directions to make sure we capture it for future use.

MR. HALE: Yeah, one thing I wanted to add Steve also, is our threshold for identifying enhancements is a lot lower than what it was for original license renewal. So, we've taken all the way down to specific procedural enhancements in some cases. And having it there as a commitment, it's out in front of everybody, and it's very difficult to step away from that. You have to follow a formal regulatory process to change that commitment.

MEMBER SUNSERI: Yeah, I think as it was mentioned here, we're not saying that we see this as an issue, or anything. But because people read our reports, and pay attention to them, and we usually replicate this table, or the statistics in our reports, it would be -- I think prudent for us to at least have a statement about why this is occurring here.

So, that if anybody looks, doesn't just take our report on face value, and try to make something negative because of a previous report. That's all I think we're trying to do.

MEMBER HALNON: Yeah, let me just ask one question to clarify. So, in the fire protection AMP,

there is a commitment to make sure that semiannual dry chemical inspections -- dry chemical and fire extinguisher inspections are completed. So, from a face value just looking at this, you could make the conclusion that right now you don't do semiannual dry chemica, and fire extinguisher inspections.

I have a hard time believing that's the case, because that's required by the -- that was what got my interest up. So, I'm assuming, and you can correct me if I'm wrong, that if you see a deficiency based on this review, you would fix the deficiency, not wait ten years to do it, is that correct?

MR. FRANZONE: So, probably the rev zero program was probably different than what's in the GALL SLR. I think that's where you see the difference. The level of detail now that you see with GALL SLR is much more, we found that at Turkey Point and Point Beach, that the level of detail required by GALL SLR is much higher. And so you'll see in the fire protection program, and I think even for St. Lucie, we've included a lot more detail in there to capture all those requirements, and enhancements.

Whereas before the program wasn't required, we may be doing those under the fire protection program, but not under the license renewal

fire protection program.

MR. HALE: And I'd like to also mention that the GALL fire protection guidance actually goes beyond what's required currently under NFPA. We have requirements in GALL that requires you to do things beyond say what you're currently doing under the fire protection program. And with regards to ongoing inspections of dry paint systems, and things of that sort that are currently not part of the NFPA, for example 25 program that you're doing.

So, when you line yourself up, you take your existing program, and you line yourself up to GALL, you'll identify those areas, and in some cases we say hey, we need to strengthen that requirement, because we don't think it's as tight as it needs to be. And that's why you see those in those enhancements.

MEMBER HALNON: So, you wait ten years to tighten it up?

MR. HALE: No, that is only the requirements for implementation. As we're going through at Turkey Point right now, once we receive the license, we start looking at the commitments under the original license renewal, as well as subsequent license renewal. In some cases it makes a lot of

sense to pull that up, and do it early.

MEMBER HALNON: Is that going to be -- I was going to say is that part of the implementation plan for license renewal, to do that, or is that --

MR. HALE: Yes.

MEMBER HALNON: So, you're not relying on the good will of your program managers to say hey, this sounds like a good idea.

MR. HALE: No, and in fact, as you see with subsequent license renewal, and I'm speaking from experience here, at say later sites for original license renewal, there are a lot of inspections that are required pre SPEO that weren't there for original license renewal. And it's quite an activity, and it's quite a detailed implementation plan if you look at everything that needs to be accomplished, say between now, and once you enter the SPEO just from a component inspection standpoint.

MR. MAHER: This is Bill Maher again. So, that is why for subsequent license renewal we have a dedicated individual that is purposefully meant to make sure that those AMPs, and the procedures are put in place, and the commitment for pre SPEO inspections, and programs are all put in place, that's why that person is dedicated there.

MEMBER HALNON: Okay, and again I'm going to put words in your mouth. When you went through the initial analysis of the AMPs, if you found a deficiency, I assume that it was fixed. I mean, because anytime you put these programs under a microscope like you do when you are developing the license application, you're going to find stuff.

MR. FRANZONE: Right, and so what we did, we identified to the site, and we put it in the plant's corrective action program. And you're right, we did find some deficiencies, and we'd enter it into the plant's corrective action program, and then it would just go through its normal process.

MEMBER HALNON: Okay, that makes sense, good, thank you.

MEMBER SUNSERI: Members, any other feedback? All right then, we will conclude this discussion, and I will just remind everyone that the next time we take up this topic will be at our May full committee meeting, which will be May 4th. This is likely to be the first item on the agenda. So, if you want to see the -- observe the letter writing report, and be able to make any factual corrections if we get something wrong during that process, you're allowed to bring up factual corrections.

So, that will begin at 8:30 Eastern Daylight Time on May 4th. Having no other business on this topic, I return back to Chair Rempe.

CHAIRMAN REMPE: Thank you, Matt. And I'd also like to express my appreciation for the staff to include the folks in the region, as well as the folks in headquarters. I appreciated the discussions that we had with them. At this time I'd like us to go off the record, and the court reporter won't be required to come back until 1:00 p.m. tomorrow, when we take up another VWRX300 topical report.

And why don't we take a 19=minute break, and come back at 5:00 p.m., and we'll continue letter writing on the containment evaluation method topical report for VWRX300.

(Whereupon, the above-entitled matter went off the record at 4:41 p.m.)

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Reactor Pressure Vessel Embrittlement Monitoring and Prediction in Long-Term Operation

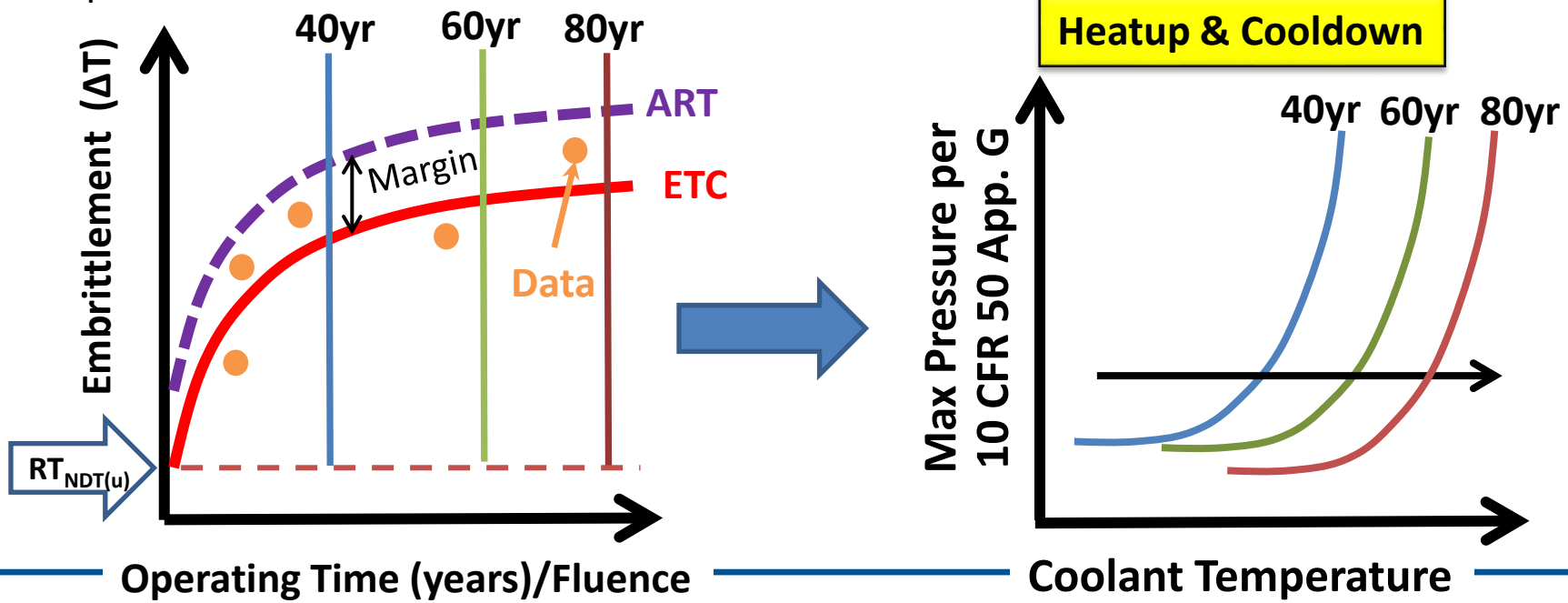
694th ACRS Full Committee
April 6, 2022

Meeting Purpose

- Discussion of Issues
  - Regulatory Guide 1.99 Rev 2 (RG 1.99) and 10 CFR 50.61 embrittlement trend curve
  - Appendix H surveillance testing
- Discussion of RPV Embrittlement rulemaking plan

Background
Monitoring and Prediction of Embrittlement

- Embrittlement Trend Curve (ETC) provides estimates of change in fracture toughness (ΔT or ΔRT_{NDT}) as a function of fluence
- Surveillance capsule testing provides monitoring to ensure ETC predicts plant specific behavior properly
- Together they are used to determine pressure-temperature (PT) limits for normal operation



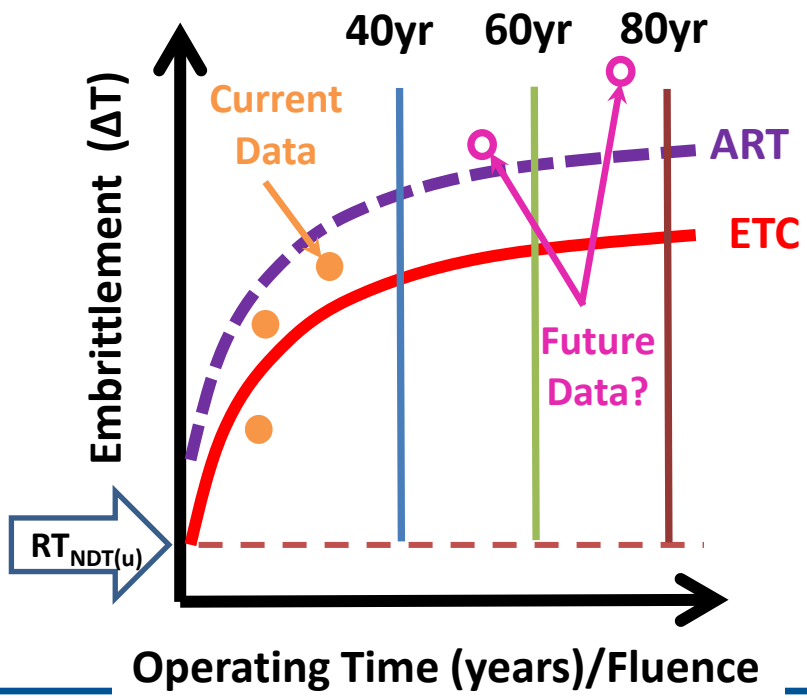
ART = Adjusted Reference Temperature

Ideal Scenario

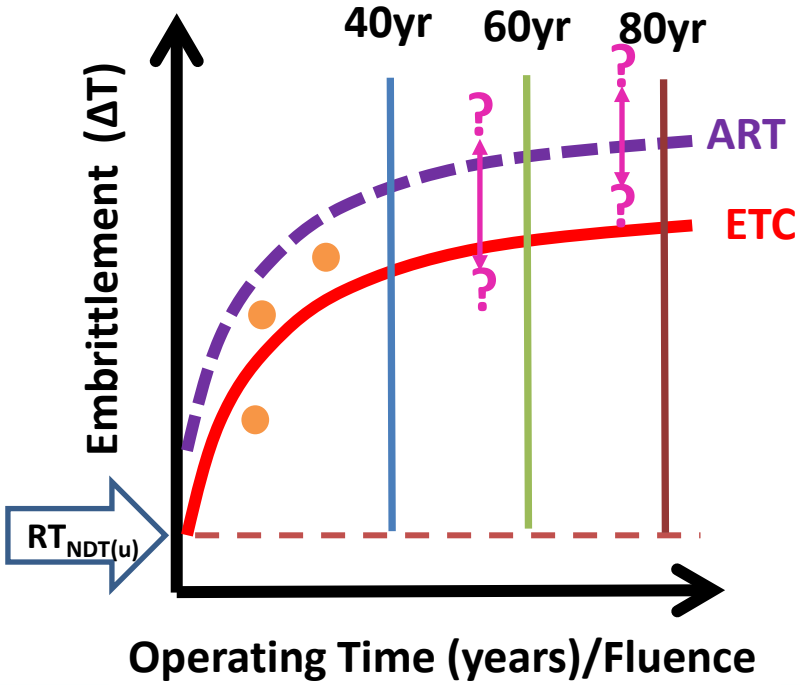
- ETC provides conservative predictions of embrittlement
- Surveillance data covers all operating periods

Potential Uncertainty Sources

IF **ETC under-predicts measurements**



IF **Limited Surveillance Data is Available**



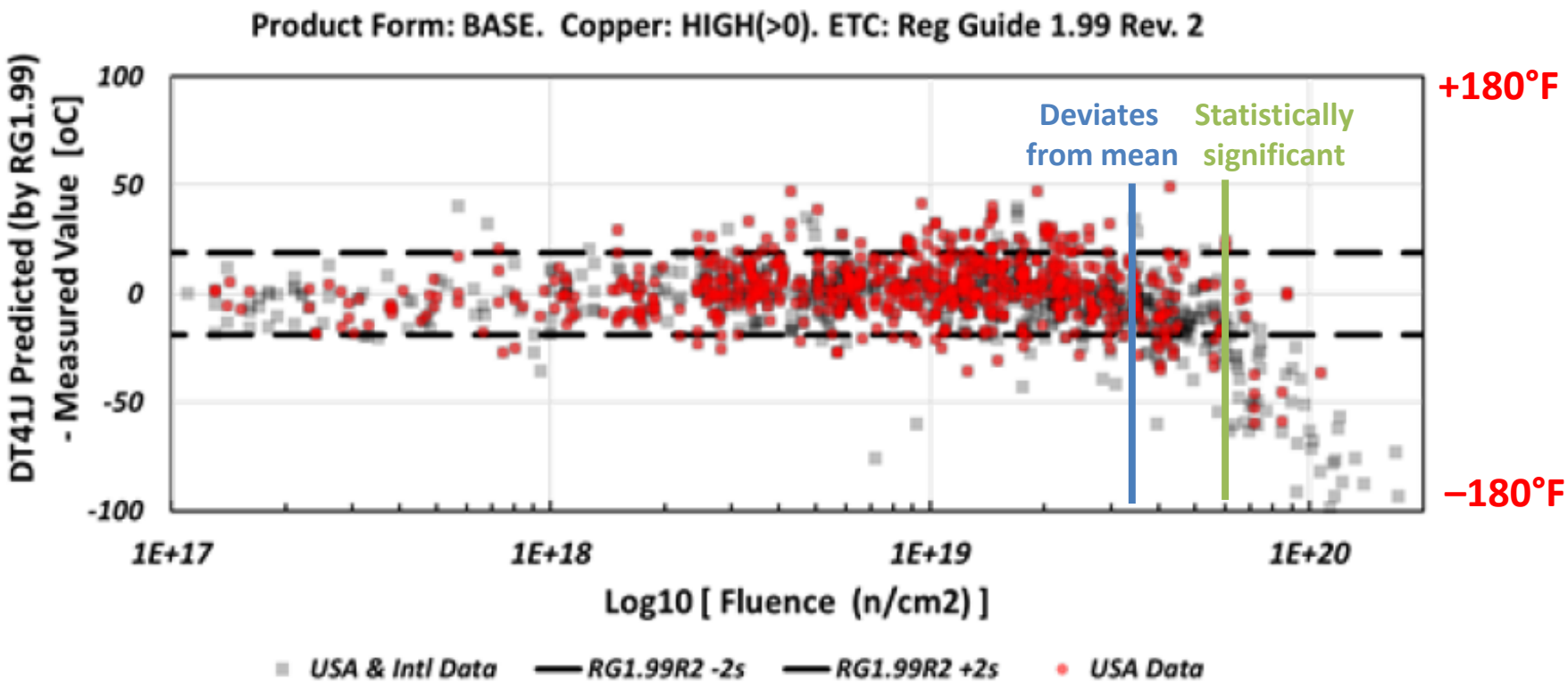
Current Perspective of Potential Issue

- High confidence that currently operating plants remain safe
- Recent licensing actions remain valid
- Insufficient embrittlement monitoring and under predictions of reactor vessel embrittlement will eventually (after about 10 years for PTS; after about 23 years for P-T limits and upper shelf energy) impact the staff's confidence in the integrity of the reactor pressure vessel in long-term operation, i.e., both safety margins and performance monitoring may be impacted
- Further work is needed to determine which plants are impacted by this potential issue

Embrittlement Trend Curve

- May 1988, NRC published RG 1.99, which contained an improved embrittlement trend curve (ETC)
  - Fit based on 177 datapoints
- June 1991, NRC updated 10 CFR 50.61 to include the ETC from RG 1.99
  - Addressed lower than measured predictions (up to 60°F) of embrittlement in some vessels
- This ETC was re-evaluated for continued adequacy in 2014 (ML13346A003) and in more detail in 2019 (ML19203A089)

Issue – ETC



DT41J = ΔT_{41J} is a measurement of embrittlement representing the shift in transition temperature from brittle to ductile fracture at an impact toughness of 41J

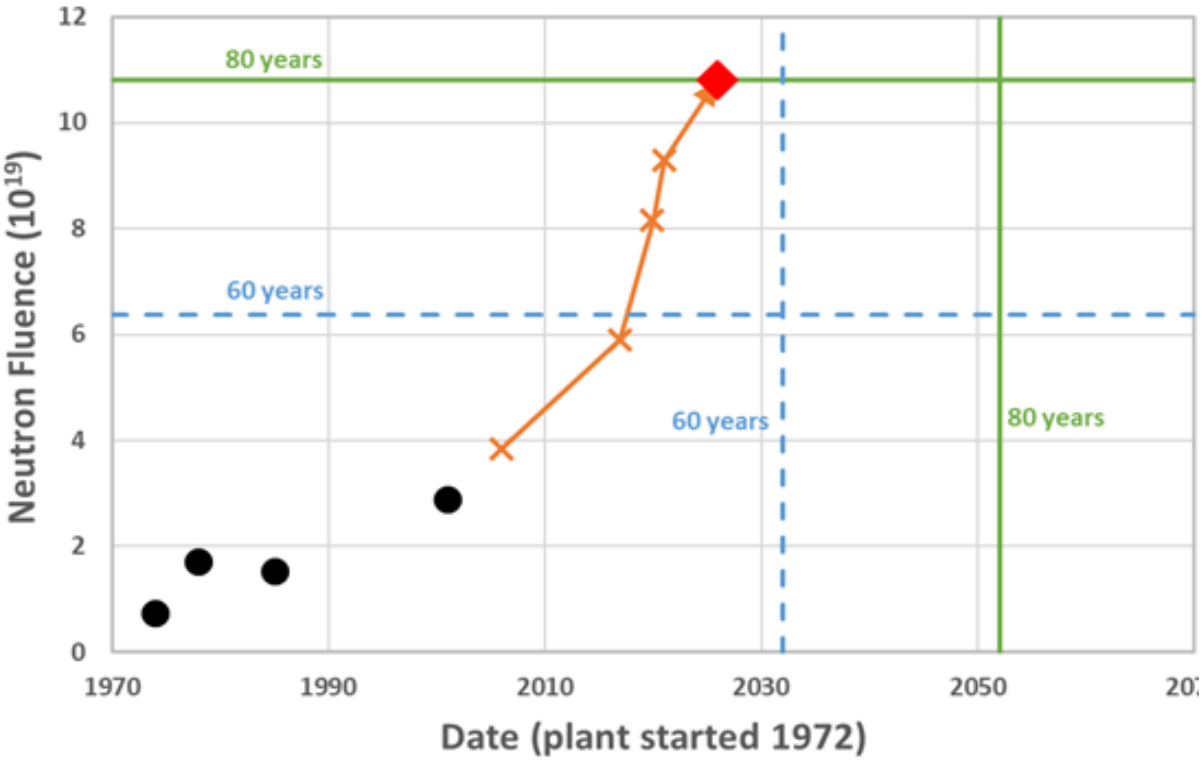
Surveillance Capsule Delays

- Appendix H to 10 CFR Part 50 requires periodic monitoring of changes in fracture toughness caused by neutron embrittlement
  - ASTM standard (E185-82) allows final capsule fluence to be 2X RPV "design" fluence – plants change (intended 40-year) design fluence to current license length (e.g., 60 or 80 years)
  - ASTM standard (for 40 years) permits holding last capsule without testing

- Commission finding ("Perry decision," NRC Administrative Letter 97-04) that staff review of requests to change capsule withdrawal schedules is limited to verification of conformance with the ASTM standard (i.e., not based on technical or safety considerations)
  - Capsule withdrawal and testing repeatedly delayed in some cases to achieve higher fluence

Issue – Appendix H

Performance Monitoring



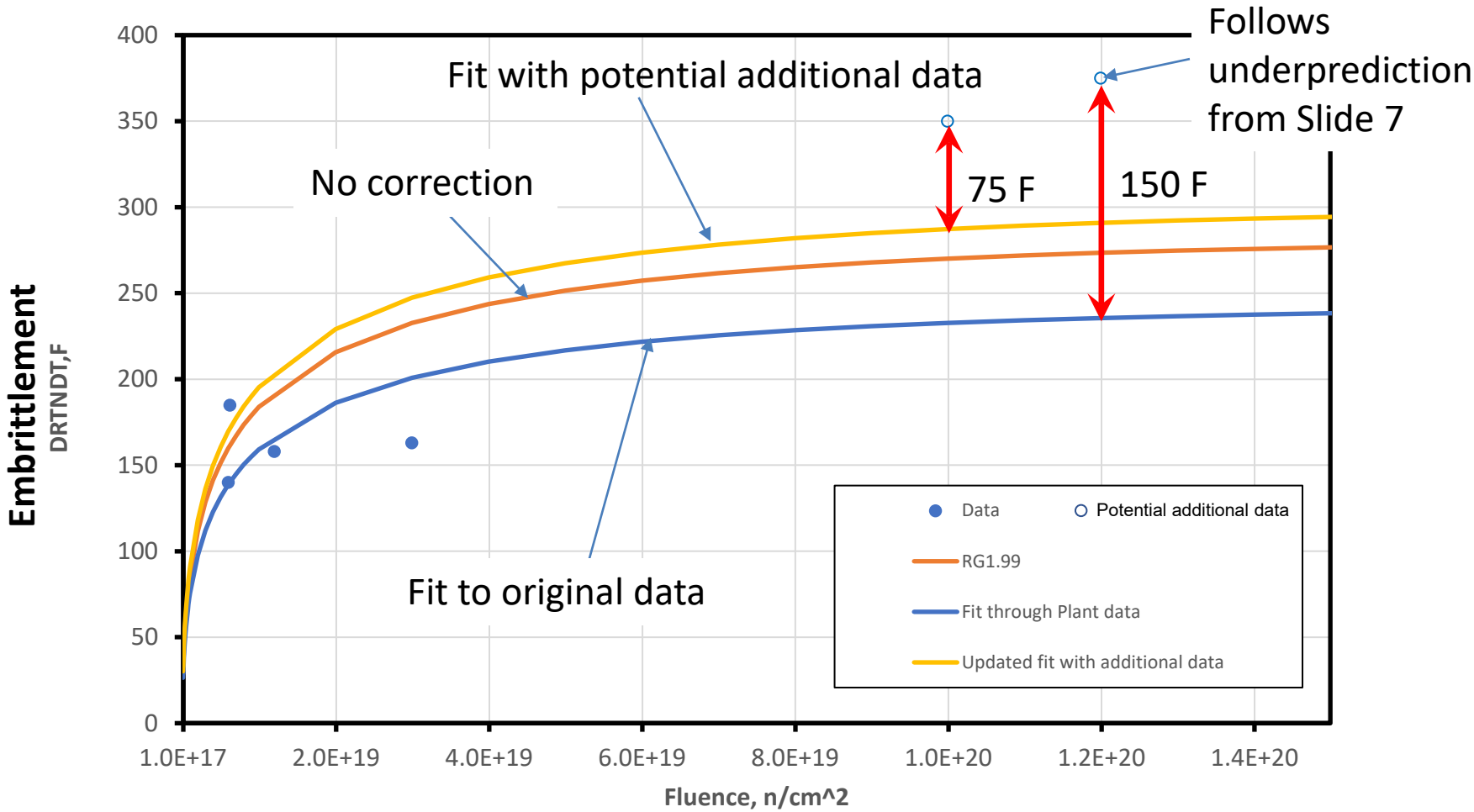
Capsule withdrawal schedule changes include delays in both time and/or fluence

Many licensees have delayed capsules (time and/or fluence), some recent examples:

Plant	Capsule #	# of times delayed
Turkey Point	5	4
Robinson	5	2
Surry U1	5	2
Surry U2	5	2
North Anna U1	4	2
North Anna U2	4	2
St. Lucie U2	4	1
Point Beach	5	1

Not all plants have delayed withdrawal of capsules

Potential Impact of Issue



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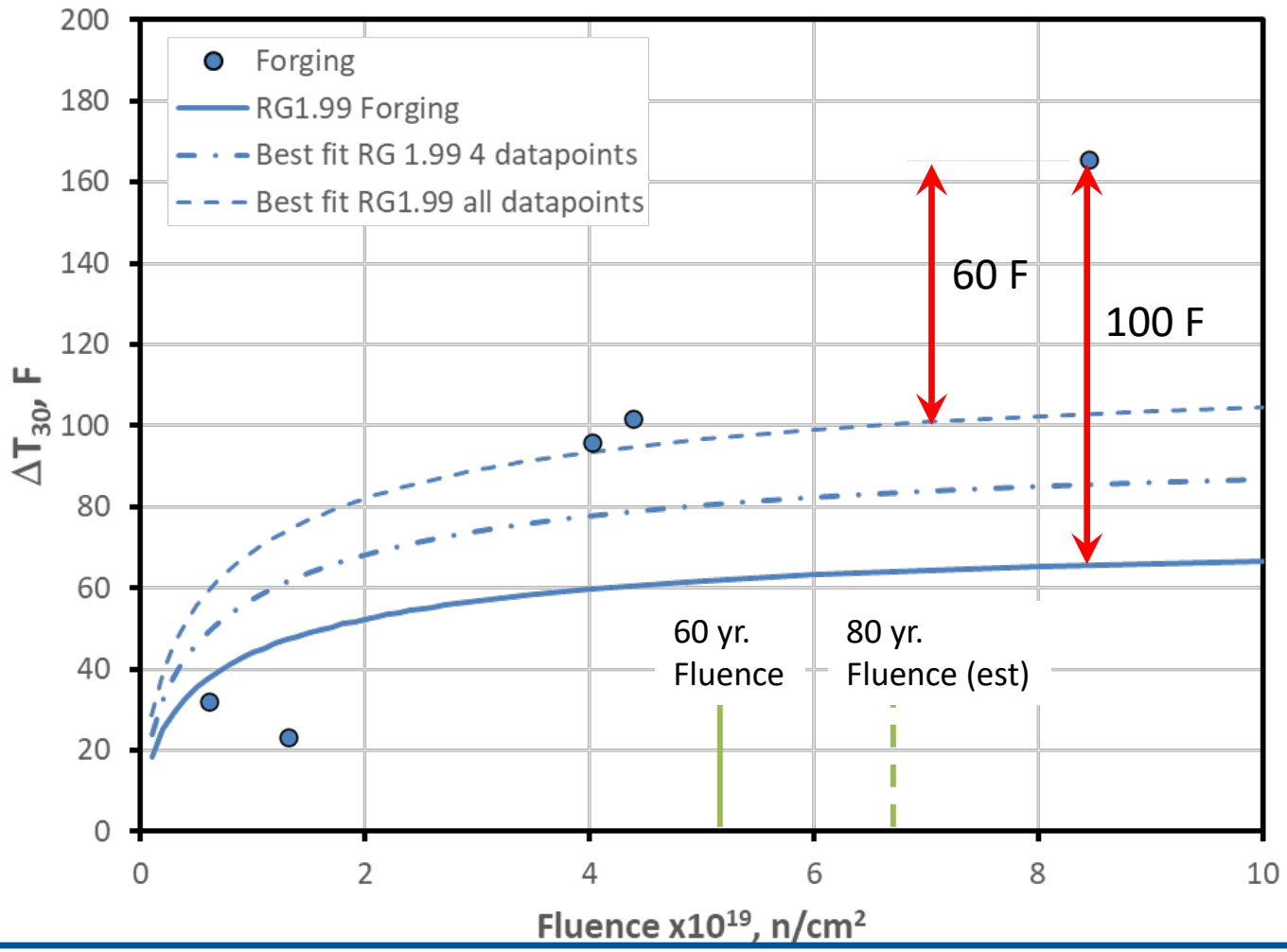
Recent Plant Surveillance Data



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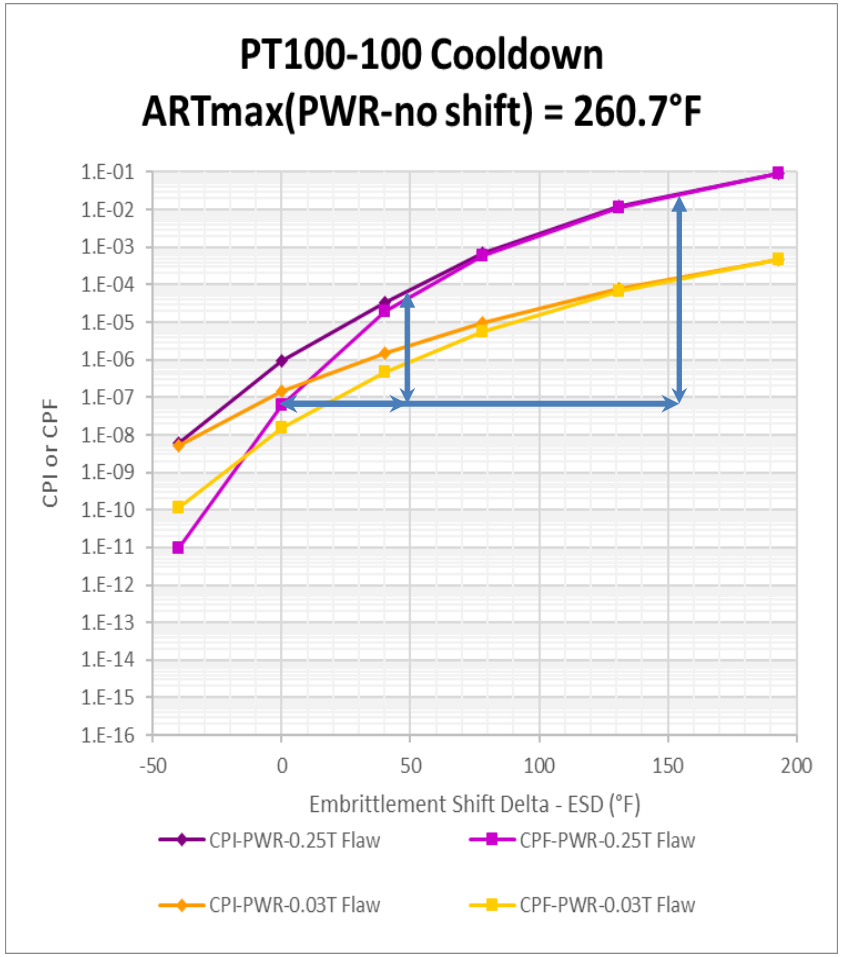
Safety Case

- **Risk of Failure**
  - Conditional probability of failure during normal operation may increase several orders of magnitude (e.g., 3 orders of magnitude at 50F), but expected transient frequency makes overall risk low.
  - Some plants may exceed Pressurized Thermal Shock screening limit in 10 CFR 50.61, but analyses suggest risk is low
  - Large uncertainty exists – many plant specific details unknown

- **Safety Margins**
  - Inaccurate embrittlement prediction and increasing uncertainty due to lack of surveillance decreases safety margins to failure – plant specific

- **Performance Monitoring**
  - Delaying capsules does not provide adequate performance monitoring to ensure embrittlement trends are reasonable

Risk of Failure



ESD represents the underprediction of ΔRT_{NDT}

Large Uncertainties:

- Unknown frequency of transient
- Actual plant fluence variations
- Are these analyses bounding?
  - Unknown plant-specific considerations
- How much protection do administrative and other operational limits provide against violating the PT limit?

"RG 1.99 Revision 2 Update FAVOR Scoping Study," May 6, 2021, TLR RES/DE/CIB-2020-09, Rev. 1, ML21126A326

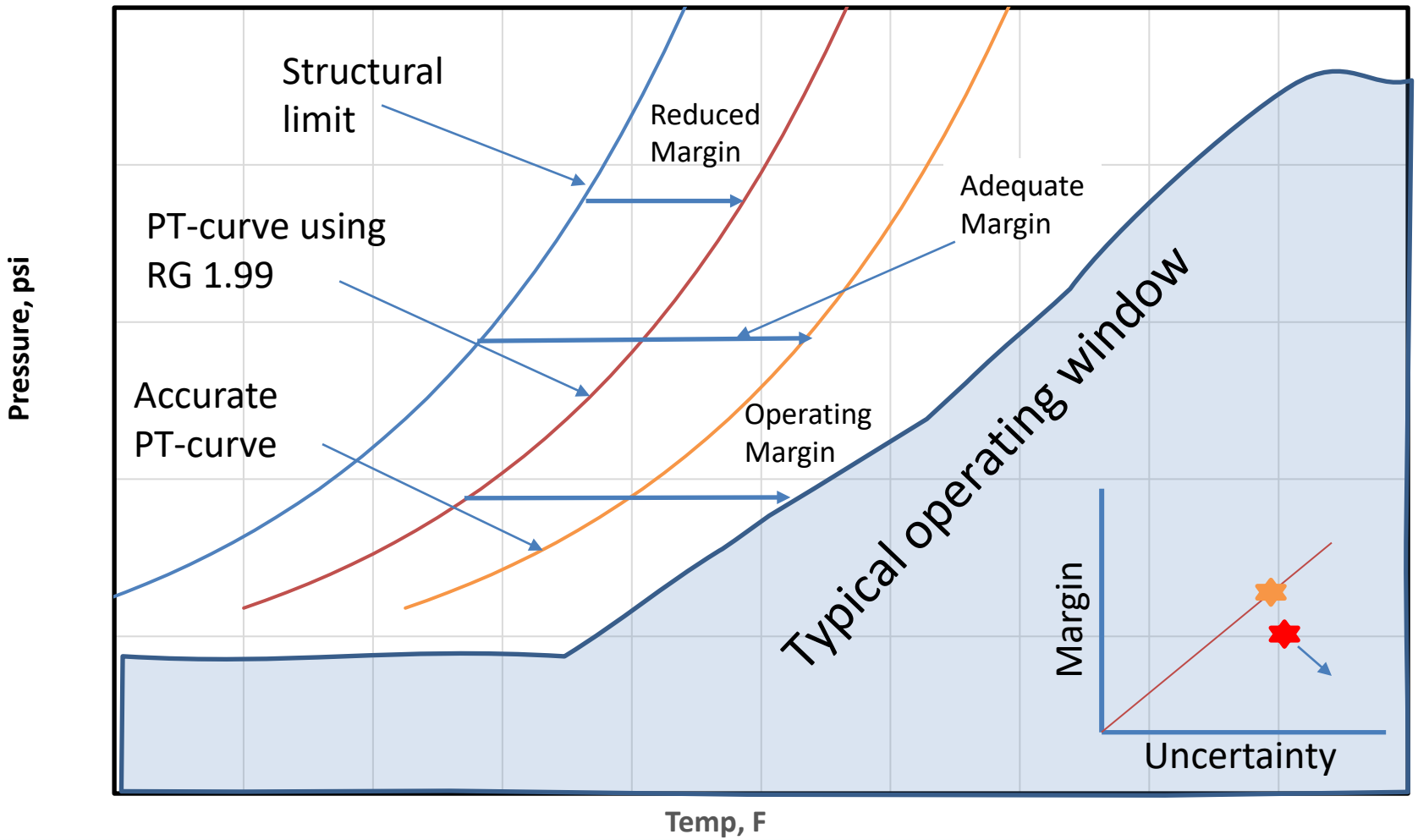
Pressurized Thermal Shock Considerations

- 10 CFR 50.61 uses ETC from RG 1.99
- RT_{PTS} from 10 CFR 50.61 might be impacted
  - Limits of 270 °F for plates, forgings, and axial weld materials, and 300 °F for circumferential weld materials
- However, through-wall crack frequency calculated with corrected embrittlement less than $1x10^{-6}$ for all cases investigated

Safety Margins

- Uncertainties in risk calculations are high and increasing with time
- Even though the risk appears low, resolving these issues will help maintain the fundamental safety principles that are the basis of plant design and operation
- Safety margins, as provided by regulations and current license bases, provide reasonable assurance against brittle fracture

Safety Margins Illustration



Uncertainties increasing due to lack of surveillance, but margin is less due to embrittlement underprediction

Analysis Summary

- With the current state of knowledge, a generalized analysis suggests the overall risk of brittle fracture is low
- The uncertainty in these results is high and increases with time
  - Plant specific details not considered
- Under certain conditions, safety margins are impacted and are decreasing as uncertainty increases
- Delaying capsules at high fluence represents a lack of sufficient performance monitoring
- Issues are plants with fluences > $6x10^{19}$ n/cm^2

Who is Impacted?

- Embrittlement Underprediction

Percentage of Fleet Surpassing Fluence Levels			Percentage of PWRs Surpassing Fluence Levels	
Year\Fluence	6×10^{19} n/cm^2	8×10^{19} n/cm^2	6×10^{19} n/cm^2	8×10^{19} n/cm^2
60 years	6%	0%	9%	0%
80 years	22%	10%	34%	15%

  - Plant specific details (e.g., limiting material, etc.) may contribute to which plants are impacted
  - More work is needed to determine which plants are impacted

- Lack of Surveillance Data
  - Any plant renewing license that chooses to delay last capsule

Staff Alternatives

- Alternative 1 – **Status Quo:** Make no changes to Appendix H to 10 CFR Part 50, 10 CFR 50.61, or RG 1.99. Handle issues through plant-specific action and generic communications.
- Alternative 2 – **Focused Solution**: Revise Appendix H to 10 CFR Part 50 to include additional surveillance testing requirements for long-term operation, revise fluence function fit for only impacted RPV materials.
- Alternative 3 – **Comprehensive Solution**: Revise Appendix H to 10 CFR Part 50 to include additional surveillance testing requirements for long-term operation, update the applicable regulations (e.g., 10 CFR 50.61) to require all licensees to use an NRC-approved ETC that properly accounts for radiation effects, update RG 1.99 to contain an ETC with one that appropriately accounts for radiation effects, and update implementing guidance.

Backfit/Forward Fit Considerations

- Alternative 1 (Plant-specific actions) – backfitting or forward fitting considered for each plant-specific case
- Alternatives 2 and 3 (Rulemaking)
  - Modification of App H will require some plants to test capsule during SLR period
  - Modification of embrittlement trend curve may require some plants to modify P-T limits and/or PTS evaluation

Staff Recommendation

- Alternative #2 – Focused Solution
  - Address issues in a focused and risk-informed manner
    - Target those plants with materials that are impacted by the underprediction issue
    - Modify current surveillance testing requirements to ensure periodic performance monitoring
  - Details of implementation to be worked out during regulatory basis effort

Rulemaking Schedule

- Alternative #2 Schedule
  - Deliver regulatory basis to the Commission—14 months after receipt of the Commission staff requirements memorandum (SRM) approving rulemaking.
  - Deliver proposed rule to the Commission—15 months after the regulatory basis is issued for public comment.
  - Deliver final rule to the Commission—15 months after the proposed rule is issued for public comment.

Summary

- High confidence that currently operating plants remain safe, and recent licensing actions remain valid
- Issue will eventually (after about 10 years for PTS and 23 years for P-T limits) impact the staff confidence in the integrity of the reactor pressure vessel in long-term operation, i.e., both safety margins and performance monitoring may be impacted
- Further work is needed to determine which plants are impacted by this issue
- Proactively ensure continued reasonable assurance though a risk-informed, performance-based solution
  - Staff delivered rulemaking plan to Commission – desires focused solution to only those conditions adversely impacted by this issue

Thank You

This presentation offers the professional opinions of EPRI staff only and does not represent an Industry position of the U.S. utilities.

RPV Embrittlement Monitoring and Prediction in Long-Term Operation

April 2022 EPRI MRP Technical Brief

Elliot J. Long
Principal Technical Leader

ACRS Meeting
April 6, 2022

Presentation Outline

- **Future Sources of High Fluence Capsule Data**
  - PWR Coordinated Reactor Vessel Surveillance Program (CRVSP)
    - MRP-326, Revision 1
  - PWR Supplemental Surveillance Program (PSSP)
    - MRP-412
- **Review of Prior EPRI MRP Conclusions from the November 2019 ACRS Meeting**
  - Potential to impact plant Pressure-Temperature (P-T) limit curves

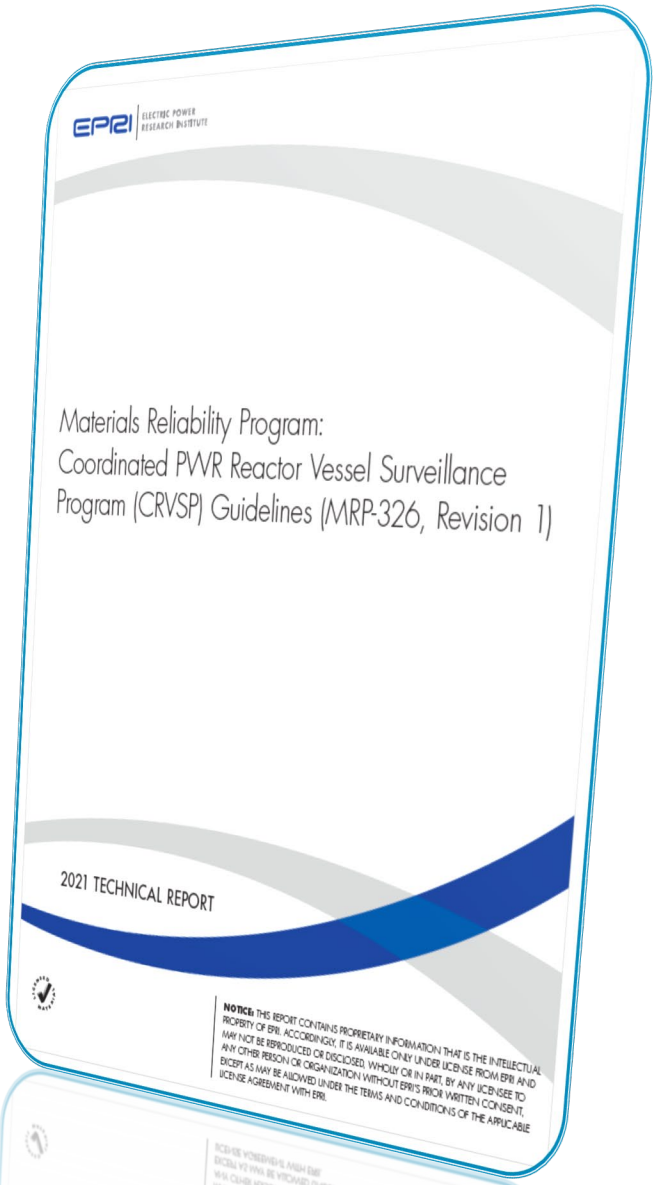
Update to the CRVSP, MRP-326, Revision 1

Coordinated PWR Reactor Vessel Surveillance Program (CRVSP)

- Materials Issue Being Addressed:
  - Optimize the U.S. PWR surveillance capsule withdrawal schedules to increase the amount of high-fluence ($f > 3.0 \times 10^{19}$ n/cm^2) surveillance data which can be used to inform development of embrittlement trend correlations (ETCs) applicable for RPV operation to high fluence (60+ years)
- Objectives of the Project
  - Revision 0 (2011): Review the reactor vessel surveillance programs (RVSPs) of the operating U.S. PWR fleet and recommend changes to selected RVSP withdrawal schedules in order to increase the amount of high fluence surveillance data by 2025
  - Revision 1 (2021): Review of how we did, what has occurred, what's left to do, and when it is most likely to happen across the US fleet
- Updates to the evaluation include
  - Evaluated capsules withdrawn since 2011
  - Future capsule pull schedules
  - Capsule fluence values
  - Analysis of closed (or to be closed) plants

Update to the CRVSP, MRP-326, Revision 1

- Current high fluence capsule withdrawal results
  - 16 out of 30 CRVSP Capsules are tested or planned to be tested
  - There are 14 remaining CRVSP Capsules
    - Half of these are not planned to be tested (i.e., due to plant shutdown) or will be delayed beyond 2025
- Summary of available high fluence data
  - 48 U.S. capsules have been tested at f > 3.0 x 10^{19} n/cm^2
    - 4 of these are f > 8.0 x 10^{19} n/cm^2
  - By 2025, the remaining 7 planned CRVSP capsules will be tested at f > 3.0 x 10^{19} n/cm^2
    - 2 of these are predicted to be f > 8.0 x 10^{19} n/cm^2



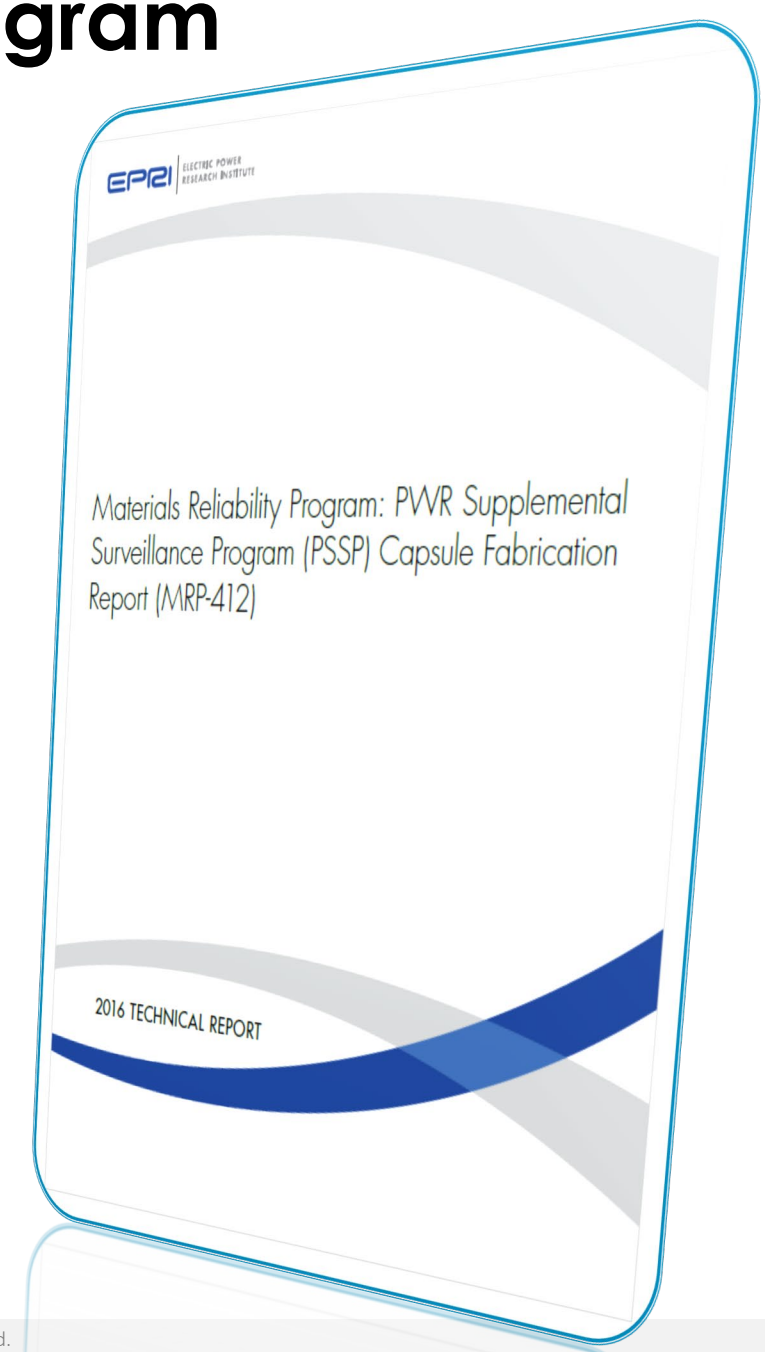
PWR Supplemental Surveillance Program (PSSP)

PWR Supplemental Surveillance Program (PSSP)

- Materials Issue Being Addressed:
  - Additional high-fluence (f > 5.0 x 10^{19} n/cm^2) surveillance data is needed to inform development of embrittlement trend correlations (ETCs) applicable for RPV operation to high fluence (60+ years)
- Objectives of the Project:
  - Fill projected gaps in the tested surveillance capsule database
  - Inform future ETCs using actual RPV surveillance materials from commercial PWRs (not test reactor data)
- Project Goal: Irradiate two supplemental surveillance capsules for ~10 total years before withdrawal, testing, evaluation and publication of capsule test reports
  - These two surveillance capsules have 288 Charpy Specimens from 27 unique plates, forgings and welds
  - The data generated from these capsules will ultimately yield 24 new transition temperature shift results and 3 additional upper shelf energy results
    - Fluence levels of the to be evaluated specimens will range from ~4.5E+19 up to ~1.2E+20 (n/cm^2)

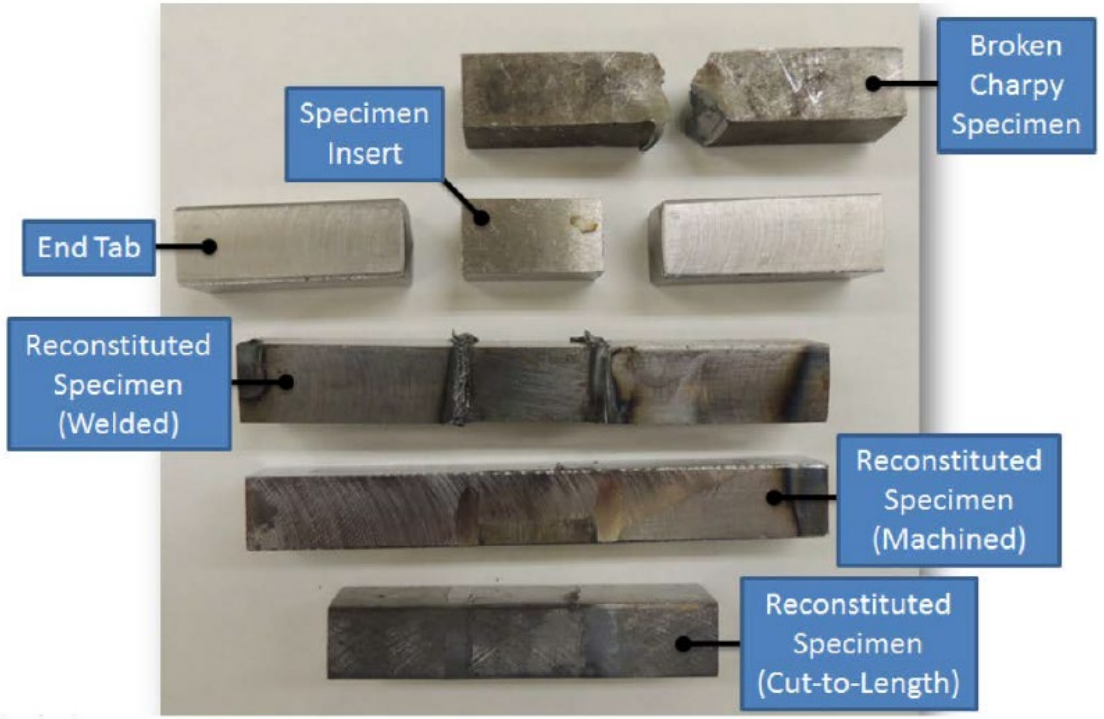
PWR Supplemental Surveillance Program

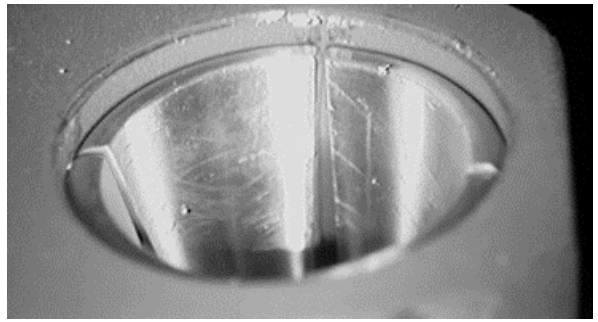
- Project History
  - Program designed and fabricated 2 supplemental surveillance capsules containing previously-irradiated, reconstituted PWR materials
  - EPRI MRP sponsored the fabrication of these 2 surveillance capsules:
    - ALA-P; 14 materials (Host: Farley 1), inserted October 2016
    - CQL-P; 13 materials (Host: Shearon Harris), inserted April 2018
  - MRP-412 (PSSP Capsule Fabrication report) was published in 2016



PWR Supplemental Surveillance Program

- Current Project Status and Timeline
  - Farley 1 Capsule P to be withdrawn in Spring 2027; Shearon Harris Capsule P in Fall 2028 per MRP-326,R1
  - Testing of surveillance capsules and data evaluation in 2028-2030
  - Anticipated Project Deliverable Date:
    - Capsule report within ~18 months of each capsules' withdrawal date (2 reports total)
  - Data evaluation and impact on future ETCs in 2030-2032





PSSP Capsule seated in its holder

Prior EPRI Conclusions on the Potential Revision of RG1.99R2

Conclusions from the EPRI Presentation to the ACRS in November 2019

- EPRI MRP previously presented on the potential revision of RG1.99R2 to the ACRS in Nov. 2019
- The conclusions from that meeting (shown at right) have not changed
- **<u>If</u>** a future revision to RG.199R2 is implemented, ASTM E900-15 remains the preferred ETC model (today)
- It is understood that for fluence values below 6E+19 n/cm^2, RG1.99R2 remains adequate for predicting RPV embrittlement
- The next slide details when certain plant designs will see that fluence level at 1/4T

Conclusions and Recommendations

- Changing the USE prediction model in RG1.99 would result in negligible safety benefit but would cause a significant reanalysis burden on the fleet
- ASTM E900-15 is the preferred alternative ΔT_{41J} prediction formula
- Because RG1.99 is used for embrittlement predictions performed to show compliance with 10CFR50, Appendix G, the appropriate metric for assessing the need for plants to adopt a new ΔT_{41J} prediction model is the RPV 1/4T fluence, since that is the fluence upon which operating limits are based per ASME XI Appendix G
  - BWRs do not reach the threshold of concern through 80 years of operation and can be exempted from the burden of adopting a new ΔT_{41J} shift prediction model
  - PWR adoption of a new shift prediction model would appropriately be based on a 1/4T fluence metric
- Because embrittlement prediction models have significant impact on the RPV operating envelop, it will be helpful to plants considering SLR for the regulator to identify the shift model that will be adopted in RG1.99R3, and guidance for consideration of surveillance data

What *surface* fluence = 6E+19 @ 1/4T and 3/4T?

- The chart at right was developed using the current fluence attenuation formula from RG1.99R2*
- It is understood that the 3-Loop Westinghouse design has the highest surface fluence at end-of-life
- SLR plants' most limiting surface fluence values are also summarized herein
- Only Plant 'A' will hit a 1/4T fluence of 6E+19 n/cm^2 for 80-years
  - This is currently predicted to occur well into the SLR operating period

Design	RPV Maker	Vessel T (in)	Surface Fluence Needed to Reach 6E+19 @	
			1/4T (E+19)	3/4T (E+19)
WEC 2-Loop	B&W/CE	6.5	8.86	19.3
WEC 3-Loop/Smaller CE	CE	7.875	9.62	24.8
B&W NSSS	B&W	8.44	9.96	27.4
WEC 4-Loop	RDM	8.45	9.96	27.5
WEC 4-Loop	B&W	8.5	9.99	27.7
WEC 4-Loop/Larger CE	CE	8.625	10.1	28.3
CE - Special	CE	8.79	10.2	29.2
CE - Sys80	CE	11.2	11.7	45

Plant	Design	EFPY	Surface Fluence (E+19)	Potentially Impacted?
A	WEC 3-Loop	72	10.8	at ~65 EFPY
B	WEC 3-Loop	68	7.26	No
C	Smaller CE	72	6.56	No
D	WEC 3-Loop	72	7.34	No
E	WEC 2-Loop	72	7.80	No
F	B&W	72	2.02	No

*Note that BWR Units have an NRC approved Integrated Surveillance Program (ISP) for up to 60 years for the U.S. BWR Fleet (BWRVIP-86, Rev. 1-A). The implementation plan for Subsequent License Renewal (SLR) has also been accepted by NRC in report BWRVIP-321-A. The highest 1/4T SLR fluence value *is less than* 1 x 10^{19} n/cm^2 (E > 1.0 MeV) for the U.S. BWR fleet.

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NRC – CNSC Joint Report

- The Canadian Nuclear Safety Commission (CNSC) and the NRC have issued a joint report on theBWRX-300 Containment Evaluation Method as part of their ongoing cooperation aimed at enhancing technical reviews of advanced reactor and small modular reactor technologies.
- The report is based on the joint review of the licensing topical report GEH submitted for NRC review approval.
- Both regulators evaluated how the information would be used in each country's regulatory infrastructure and confirmed items that could be documented for mutual understanding.
- Nothing in the joint report fetters the powers, duties or discretion of CNSC or NRC designated officers, CNSC or NRC inspectors or the respective Commissions regarding making regulatory decisions or taking regulatory action.

ENCLOSURE 1

M220049

ACRS Full Committee Presentation Slides for NEDC-33922P,
BWRX-300 Containment Evaluation Method Licensing Topical Report

Non-Proprietary Information

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ACRS Full Committee Presentation

GE-Hitachi (GEH)
Licensing Topical Report (LTR) NEDC-33922P
BWRX-300 Containment Evaluation Method

(Open Session)

April 6, 2022

LTR Purpose and Scope

- GEH is seeking NRC approval for application of an analysis method to be used for evaluating the BWRX-300 dry containment thermal hydraulic performance.
- The LTR scope includes
  - Method description and qualification
  - Sensitivity studies
  - Demonstration cases
- The analysis method used for the BWRX-300 containment thermal hydraulics performance demonstrates that the containment design complies with 10 CFR 50, Appendix A, General Design Criteria 2, 4, 16, 38, 41, 50, and 51.
- Details of the containment performance acceptance criteria are listed in Section 4.0 of NRC-approved LTR NEDC-33911P, BWRX-300 Containment Performance
- GEH is not requesting NRC approval for exemptions from any regulatory requirements.

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BWRX-300 Containment Design Features

The following containment design features are relevant to the purposes of the LTR:

- Dry enclosure, near atmospheric pressure during normal operation
- Inerted with nitrogen during normal operation
- Design pressure and temperature are within the experience base of conventional BWRs
- No subcompartments containing large bore high energy lines
- The subcompartments have sufficiently large openings such that the boundaries of the subcompartments do not experience large pressure differentials resulting from pipe breaks outside the subcompartments

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Limiting Pipe Breaks

- The limiting large breaks are:
  - Main steam pipe
  - Feedwater pipe
- All design basis large breaks are rapidly isolated at the RPV nozzle.

- The limiting small breaks are unisolated instrument line breaks, either in the steam or liquid space.

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Overview of the Evaluation Model

TRACG used to evaluate the mass and energy release

- Applies the ESBWR TRACG-LOCA method
- Performed both base and conservative demonstration cases:
  - Main Steam and Feedwater Large Breaks
  - Small Steam and Liquid Pipe Break Cases

GOTHIC used to evaluate containment response

- New containment model developed for BWRX-300
- Code has been benchmarked to separate effect and integral tests. Benchmarking to the test data of a similar size containment is included in the LTR.
- Performed both base and conservative cases
- Same approach that was taken for the ESBWR containment method

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Conclusion

In summary…

- GEH is seeking NRC approval for application of an analysis method to be used for evaluating the BWRX-300 dry containment thermal hydraulic performance.
- GEH is not requesting NRC approval for exemptions from any regulatory requirements.
- TRACG utilizes the applicable parts of the TRACG Application for ESBWR approved LTR, which is incorporated in the approved ESBWR Design Certification
- Utilizes GOTHIC, a standard code used for evaluating thermal-hydraulic containment response in the nuclear industry
- Individual key inputs, assumptions and modeling parameters conservatively biased simultaneously in the conservative cases (same approach taken for the ESBWR containment method)

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Questions or Comments

Regulatory Review of GEH Topical Report "BWRX-300 Containment Evaluation Method"

NEDC-33922P, Revision 2

NRC Staff Presentation

BWRX-300 Small Modular Reactor ACRS Full Committee Meeting

April 6, 2022

NRC Staff Review Team

- NRR Nuclear Systems Performance Branch (SNSB)
  - Syed Haider
- NRR New Reactor Licensing Branch (NRLB)
  - James Shea
- NRR Nuclear Methods, Systems & New Reactors Branch (SNRB)
  - Carl Thurston, Shanlai Lu
- NRR Containment and Plant Systems Branch (SCPB)
  - Chang Li
- RES Code and Reactor Analysis Branch (CRAB)
  - Peter Lien, Joe Staudenmeier, Andrew Ireland
- RES Fuel & Source Term Code Development Branch (FSTCB)
  - Shawn Campbell

Presentation Outline

- Overview of BWRX-300 Containment Evaluation Method (CEM) LTR NEDC-33922P, Revision 2
- BWRX-300 acceptance criteria for containment response
- BWRX-300 containment design background
- BWRX-300 containment evaluation method demonstration analyses
- TRACG mass and energy release calculation methodology review
- GOTHIC containment response calculation methodology review
- Confirmatory analysis approach and results
- Resulting four limitations and conditions
- Conclusions

NRC Staff Review of the LTR

- The purpose of GEH LTR NEDC-33922P, Revision 2, is to obtain NRC staff approval of the BWRX-300 containment evaluation method (CEM) for peak containment pressure and temperature analysis

- The approved methodology will be used to design the BWRX-300 containment, and support a license application for a CP and OL under 10 CFR 50, or a DCA and COL under 10 CFR 52

- BWRX-300 LTR Acceptance Criteria
  - Accident pressure and temperature are less than design pressure and temperature with appropriate margin
  - Containment pressure is reduced to less than 50% of the peak accident pressure for the most limiting LOCA within 24 hours
  - Containment pressure responses after 24 hours for LOCAs that do not produce the peak accident pressure are maintained below 50% of the peak pressure for the most limiting LOCA
  - Containment atmosphere remains sufficiently mixed such that deflagration or detonation does not occur inside containment

BWRX-300 Containment Design Background

- BWRX-300 has a nitrogen-inerted, dry containment
- No suppression pool inside the containment
- RPV isolation valve closure limits M&E release in LBLOCA
- RPV remains unisolated for SBLOCA with continuous break flow
- Passive Containment Cooling System (PCCS)
  - Long-term containment SBLOCA pressure reduction/mitigation
  - Demonstration analyses with specific PCCS units described in the LTR
- Reactor cavity pool for containment heat removal
- Containment dome interfacing with the reactor cavity pool

BWRX-300 Containment Evaluation Method Demonstration Analyses

- Containment analysis method for BWRX-300 thermal-hydraulic performance is used to demonstrate that the containment design satisfies the acceptance criteria for:
  - Large-Break Loss-of-Coolant Accident (LBLOCA)
  - Small-Break Loss-of-Coolant Accident (SBLOCA)
- Analyzed containment DBAs include liquid and steam breaks
- Acceptance criteria were satisfied for the LTR demonstration cases
- Applicant used TRACG code to calculate the mass and energy release, and GOTHIC code to calculate the containment response
- NRC Staff used TRACE and MELCOR to develop models to perform confirmatory analyses

TRACG Code - Overview for BWRX-300

- Overview of TRACG code
  - Latest TRACG versions used in analysis, no significant changes since ESBWR
  - RPV model and internal components scaled from ESBWR
  - De-coupled method assumes containment remains at atmospheric pressure

- Past TRACG approval and relevance to BWRX-300
  - ESBWR qualification extended to BWRX-300, such that ESBWR PIRT and model biases applied for RPV and internals
  - BWR/2–6 methods evoked since some events result in core uncovery
  - IC's safety function changed and modeled in considerably more detail
  - Modeling deemed adequate for M&E release calculations (w/ L&Cs applied)

TRACG Code – Mass and Energy Release Calculation Methodology

- BWRX-300 unique design features in comparison with ESBWR
  - LBLOCA isolation (Previous Approved LTR)
  - No suppression pool
  - ICs are the primary decay heat removal path
- RPV isolation valves limits break flow and M&E release for large piping but small breaks are un-isolated and continue blowdown for 72 hours
- One ICS train inoperative (due to limiting single failure)
- Conservative inputs for initial power level, power history, scram time, choke flow model, atmospheric pressure break boundary condition and bounding operating conditions

TRACG Code – Mass and Energy Release Calculation Methodology

Significant Issues and Resolution

RAI – Radiolytic gas accumulation and removal in the ICs

L&C 1: total volumetric fraction of radiolytic gases in the IC lower drum limited to a sufficiently low level such that condensation heat transfer in the ICs is not adversely affected and the hydrogen deflagration margin is maintained

RAI – ICs return line steam trap

L&C 2: IC return line layout must include a loop seal, or water trap, that prevents reverse flow from RPV back into the IC return line

GOTHIC Code Overview for BWRX-300

- Overview of GOTHIC code
  - An established industry code widely used for containment response analysis
  - 10 CFR Part 50, Appendix B compliant code
  - Latest GOTHIC version 8.3 used in the BWRX-300 analysis

- Past GOTHIC approval and relevance to BWRX-300
  - GOTHIC previously approved for containment response analysis
  - BWRX-300 containment PIRT consistent with GOTHIC functionalities
  - BWRX-300 relevant GOTHIC benchmarking against CVTR test data reviewed
  - GOTHIC is qualified for the thermal stratification and 3D effects

BWRX-300 GOTHIC Containment Response Methodology

- Based on Reg Guide 1.203, "Transient and Accident Analysis Methods”
- Decoupled M&E release from the TRACG RPV model with no backpressure as a containment BC for the stand-alone GOTHIC containment model
- 4-component GOTHIC model
  - Containment (nodalized)
  - Dome (nodalized)
  - PCCS (nodalized)
  - Reactor Cavity Pool (lumped)
- Conservative Diffusion Layer Model (DLM) used for condensation
- Thermal stratification inside the containment

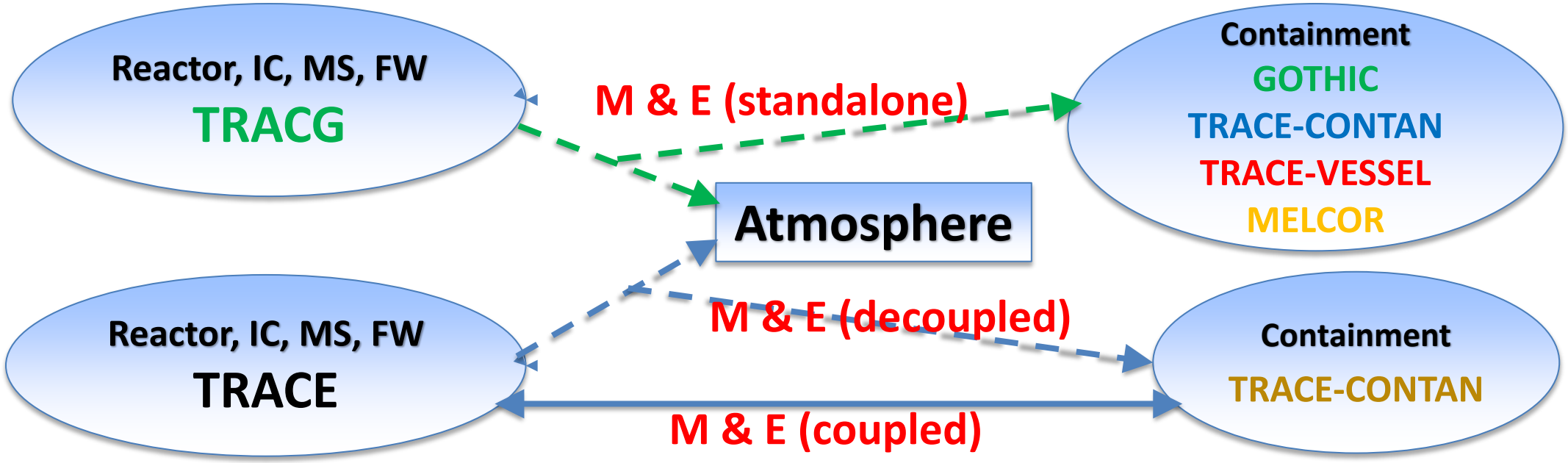
Staff Review of the BWRX-300 GOTHIC Containment Response Methodology

- Physical phenomena (GOTHIC PIRT)
- GOTHIC input model (Nominal inputs, assumptions, and correlations)
- Key modeling uncertainties and conservative biases -- Overall GOTHIC model conservatism
- Nodalization sensitivity studies for the containment and PCCS
- GOTHIC benchmarking of the test data
- BWRX-300 containment sensitivity analyses for large/small breaks
- PCCS capacity to mitigate the containment pressure in the long-term
- Containment mixing for combustible gases

Significant Containment-specific Issues & Resolutions

- Break location & and break flow orientation sensitivities
  - Limiting PCP LBLOCA location and orientation modified
  - A liquid, and not steam, SBLOCA is limiting
- Sensitivity to containment nodalization
  - Potential for reverse flow and non-condensable gas return to RPV
  - L&C #3 –No break flow reversal or reversal not safety significant
- Containment heat transfer modeling
  - Differences b/w confirmatory & GOTHIC Dome & PCCS heat transfers
  - Justification for the condensation/natural convection heat transfer correlations
  - An error identified & corrected in GOTHIC PCCS condensation modeling
- PCCS modeling and nodalization sensitivity study
  - Confirmatory sensitivity study performed
  - L&C #4 – Applicability to the final PCCS design for licensing basis

Confirmatory Analysis Approach and Results



- The proposed methodology using TRACG/GOTHIC codes is conservative
- Containment response is sensitive to nodalization, break location/orientation
- The accumulation of radiolytic gases during LOCA is possible (L&C#1)
- The ICs return line water trap is needed (L&C#2)
- Flow reversal from containment to RPV is possible (L&C#3) with insufficient PCCS heat removal capacity (L&C#4)

RPV-specific Limitations and Conditions

- L&C #1

  The use of this CEM is limited to a BWRX-300 design that limits the total volumetric fraction of radiolytic gases in the IC lower drum to a sufficiently low level throughout a 72-hour period following the event such that condensation heat transfer in the ICs is not adversely affected and the hydrogen deflagration margin is maintained.

- L&C #2

  The use of this CEM is limited to a BWRX-300 design that a proper isolation condenser return line layout is chosen, such as a loop seal or a water trap, to prevent reverse flow from RPV into the IC return line throughout a 72-hour period following the event or where an applicant or licensee referencing this report demonstrates that the TRACG code is capable of conservatively modeling the overall ICs heat removal capacity when reverse flow occurs in the IC discharge lines.

Containment-specific Limitations and Conditions

- L&C #3.

  The use of this CEM is limited to a BWRX-300 design in which the PCCS is sized sufficiently large such that a reverse flow from containment back to RPV does not occur during the first 72-hours into the event. The applicant or licensee referencing this report needs to demonstrate that no reverse flow could occur, or any reverse flow that occurs under the most bounding flow reversal conditions resulting in the degradation of IC heat transfer is not safety-significant with respect to the acceptance criteria for the BWRX-300 CEM.

- L&C #4.

  The use of this CEM was demonstrated for a BWRX-300 design with the PCCS described in this LTR. For any alternate PCCS design configuration and placement, the applicability of this method and the PCCS modeling approach must be reviewed and found to be acceptable by the NRC for BWRX-300 licensing-basis analyses.

Conclusions

- The proposed BWRX-300 analytical approach and TRACG/GOTHIC modeling described in the LTR for M&E release and containment response are acceptable, with the appropriate conservative biases and modeling inputs to address the model uncertainties.

- With the four NRC L&Cs specified in the staff SER Section 7.0, the CEM presented in GEH LTR NEDC-33922P, Revision 2, is acceptable for BWRX 300 peak containment pressure and temperature analysis of the containment DBAs.

- The NRC staff will evaluate the regulatory compliance of the final BWRX-300 containment design using the CEM during the future licensing activities, in accordance with 10 CFR Part 50 or 10 CFR Part 52, as applicable.