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UNITED STATES NUCLEAR REGULATORY COMMISSION'S
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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718TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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WEDNESDAY

SEPTEMBER 4, 2024

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The Advisory Committee met via Video-Teleconference, at 8:30 a.m. EDT, Walter L. Kirchner, Chairman, presiding.

COMMITTEE MEMBERS:

WALTER L. KIRCHNER, Chairman

GREGORY H. HALNON, Vice Chairman

DAVID A. PETTI, Member-at-Large

RONALD G. BALLINGER, Member

VICKI M. BIER, Member

VESNA B. DIMITRIJEVIC, Member

CRAIG D. HARRINGTON, Member

ROBERT P. MARTIN, Member

SCOTT P. PALMTAG, Member

THOMAS E. ROBERTS, Member

MATTHEW W. SUNSERI, Member

ACRS CONSULTANT:

DENNIS BLEY

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIALS:

KENT HOWARD

QUYNH NGUYEN

DEREK WIDMAYER

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

(8:31 a.m.)

CHAIR KIRCHNER: Okay, good morning.

The meeting will now come to order.

This is the first day of the 718th meeting of the Advisory Committee on Reactor Safeguards, ACRS.

I'm Walt Kirchner, Chair of the ACRS.

ACRS Members in attendance today are Ron Ballinger, Vickie Bier, Greg Halnon, Craig Harrington, Robert Martin, Scott Palmtag, Dave Petti, Thomas Roberts, and Matt Sunseri is there, there you are.

I was looking for you over here.

MEMBER SUNSERI: Oh, yes.

CHAIR KIRCHNER: And also joining us virtually is Vesna Dimitrijevic.

Consultants with us today are Dennis Bley, Myron Hecht, and Stephen Schultz.

If I've missed anyone, please speak up.

Quynh Nguyen and Derek Widmayer are the ACRS staff Designated Federal Officers for this morning's full committee meeting.

Member Sunseri has recused himself from this afternoon's discussions due to a potential conflict of interest.

I note that we have a quorum for today's meeting.

The ACRS was established by statute and is governed under Federal Advisory Committee Act, or FACA.

I think we have a live mic out there somewhere. So, if you would silence your mic.

Thank you.

The ACRS was established by statue and is governed by the Federal Advisory Committee Act, or FACA.

The NRC implements FACA in accordance with its regulations found in Title 10, Part 7 of the Code of Federal Regulations.

In these regulations and the committee's bylaws, the ACRS speaks only through its publish letter reports.

Member comments should be, therefore, regarded as only the individual opinion of that member, not a committee position.

All relevant information related to ACRS activities such as letters, rules for meeting participation, and transcripts are located on the NRC public website and can be easily found by typing About Us ACRS in the search field on the NRC's home page.

The ACRS, consistent with the agency, is valued of public transparency and regulation of nuclear facilities, provides opportunity for public input and comment during our proceedings.

We have received no written statements or requests to make an oral statement from the public, although later this afternoon, we will have C-10, a public interest group, providing us an informational presentation.

We have also set aside time at the end of this meeting for public comments.

The ACRS will gather information, analyze relevant issues and facts, and formulate proposed conclusions and recommendations as appropriate for deliberation by the full committee.

A transcript of the meeting is being kept and will be posted on our website.

When addressing the committee, the participants should first identify themselves and speak with sufficient clarity and volume so that they may be readily heard.

If you are not speaking, please mute your computer microphones, if you're using Teams or your phone, by pressing star six.

Please do not use the Teams chat feature to conduct sidebar discussions related to presentations, rather limit the use of the meeting chat function to report IT problems.

For everyone in the room, please put all your electronic devices in silent mode and mute your laptop microphone and speakers.

In addition, as previously said, please keep sidebar discussions in the room to a minimum since the ceiling microphones are live.

With that, I'll just remind everyone that the table microphones are unidirectional and you'll need to speak directly in front of the microphones to be heard on the line.

Finally, if you have any feedback for the ACRS about today's meeting, we encourage you fill out the public meeting feedback from on the NRC's website.

During today's meeting, the committee will consider two topics.

In this morning session, we are going to discuss the X-energy principle design criteria topical report.

And then, this afternoon at 1:00, we will discuss the Seabrook Station alkali-silica reaction informational updates.

During this morning session, we will deliberate on the topical report and the staff's draft safety evaluation for the X-energy's principle design criteria for the Xe-100 small modular reactor.

Regarding principle design criteria, we'll refer to these as PDC.

And we expect to focus today's discussion on X-energy's process in developing these criteria for their high temperature gas cooled reactor design.

And with that, I am going to turn to our lead for the X-energy design center subcommittee, Bob Martin.

You have the floor.

MEMBER MARTIN: All right, thanks.

So, this is a little bit different kind of meeting of the full committee.

A few weeks ago, we had, of course, X-energy here to present their PDCs. Of course, the staff was here, too, to present their SE.

We came out of that meeting basically recommending that a letter was not necessary. And we were fairly content, satisfied with two presentations of which, being pretty thorough, X-energy had abided by the Reg Guide 1232, which is the guidance on developing principle design criteria for non-light water reactors.

And I think it's relevant to note that, in the afternoon, we also had Westinghouse also presenting on PDCs.

So, we had a very interesting day of looking at PDC presentations from two different perspectives.

In fact, this is the fourth time we have seen a PDC topical. We've also seen from Atomics last summer and we saw TerraPower not too long -- a few months ago. And I think having seen four different topical reports on this topic, we get to see kind of an interesting variety and interesting contrast between how the different design centers have approached, you know, the different criteria.

Now, that's not to say that they're all people. Right? So, subsequent to our meeting two weeks ago where there was a kind of a small dialogue, you know, between mostly, you know, Tom Roberts and I and Walt and some of the staff addressing two PDCs in particular.

This is sort of related to PDC 16, 26, 615 being emulsion containment and the 26 being reactivity is the reactivity control systems. What I thought I would do was kind of read in my summary.

So, that was the recommendation, that we would just have a summary report. And if it's okay with everyone, it's a page long, so you're going to hear me ramble a little bit, I thought I'd also bring it up so y'all can follow along. I think I can do this.

It's big enough to read and there we go, perfect. All right, I'd just like to read this in for the record.

Our X-energy Subcommittee met on August 21st, 2024 to discuss X-energy's Xe-100 topical report, Principle Design Criteria Revision 3, and the associated NRC staff safety evaluation.

During these meetings, we had kind of a discussion what staff, representatives of X-energy as well as documents cited in X-energy's topical report and the staff SE.

X-energy plans to pursue the risk informed performance based licensing approach described in NEI 1804, NEI 2107. These industry documents have been endorsed by the NRC and Regulatory Guides 1.233 and 1.253.

The NEI Reg Guide documents emphasize the risk informed approach to the selection of licensing basis events, classification of systems, structures and components and the evaluation of defense in depth.

They also include expectations for the

preparation of PDC that begin with the application of Reg Guide 1232, Guidance of Developing Principle Design Criteria for Non-light Water Reactors.

Reg Guide 1.232 provides reactor designers the guidance on how to establish PDC that ensure safety while accommodating the innovative features and advance reactor designs. Relevant to the Xe-100, this includes generic modular high temperature gas cool reactor design criteria, MHTGR-DC.

X-energy developed the Xe-100 PDC primarily by adapting these MHTGR-DC with changes reflecting design specific details and the licensing process described in Reg Guide 1.233 and Reg Guide 1.253.

Most of these changes were vernacular in nature. For example, per NEI 1804, safety related and non-safety related with special treatment functions and structure systems components are referred to as safety significant. As such, wherever important to safety appeared in the MHTGR-DC was replaced with safety significant.

Another artifact applying to Reg Guide 1.253 is that X-energy presented PDCH required functional design criteria and complimentary design criteria.

RDC and CDC are distinguished as addressing the functional functions provided by safety related and non-safety related with special treatment perspectives.

I also, just kind of a side note, say that X-energy is the first design center to implement this through this Reg Guide 1.253.

So, this concept of required functional design criteria and design is, at this moment, kind of unique to their PDC topical report.

Okay, and to finish the paragraph, in addition, they introduce owner-controlled design criteria, or non-safety-related with no special treatment and functions, NSSEs.

Another frequent departure from Reg Guide 1.232 was deletion of single failure in favor of liability criteria, an option mentioned in Reg Guide 1.233, I say, I wonder if I meant 232 with that one, but I'll go back to that.

A concern raised regarding this change was that the applicant may find reliability characteristics, particularly of concern is difficulty to quantify for SSEs for limited industrial application.

As such, application of single failure

criteria might be an acceptable alternative.

Reg Guide 232 specifies that the MHTGR-DC applied when certain design characteristics were met. These are the use of TRISO fuel graphite as primary structural material for the core and reflector and the same gas as primary coolant, all of which are credited in the Xe-100.

In addition, Reg Guide 1.232 states that designs applying the MHTGR-DC should have operational characteristics addressing anticipated operational occurrences, AOOs, and design basis events, DPEs, applied to MHTGRs.

While specific details regarding operational characteristics and the Xe-100's response to AOOs and DPEs will be an outcome of X-energy's limitation of NEI 1804, NRC staff acknowledged the preliminary information provided by X-energy has been sufficient to support X-energy's claim in the context of their review.

X-energy's PDC 16 functional containment design departed from Reg Guide 1.232 language by replacing, quote, multiple barriers with a required functional design criteria specific to fuel particle and pebble barriers and a complimentary design criteria specific to the helium pressure boundary.

The necessity of the word multiple was debated at our meeting, considering it appears in the definition of functional containment used in 8/28/18 ACRS letter on functional containment and in SECY 18-296.

However, it is not present, as one example, in definition for functional containment appearing in Reg Guide 1.232 rationale for MHTGR-DC.

But regardless, between X-energy's RDC and CDC 16, multiple barriers are identified.

The identified set of barriers does not mention an external barrier such as are in containment.

Rather, prior agency's deliberation on HTGR licensings have deferred to require for external barriers to be determined by specified acceptable radiological release design limits, SARRDLs in RFDC CDC 16 for Xe-100, X-energy substituted the mention of SARRDL with functional containment design limits which was understood to include SARRDLs as well as additional criteria such as fuel temperature.

Related to PCD 16, X-energy plans to use a proprietary TRISO partial fuel referred to as TRISO-X from prior engagement between X-energy and the staff on X-energy's fuel qualification methodology, X-energy will evaluate any discrepancies between their fuel and the TRISO fuel specification's performance data from the U.S. Department of Energy's advanced gas reactor DGR program as presented in DPR -- or excuse me, EPRI-AR-1 topical report. I think that was published in 2020, 2021, it might have been a Rev or something like that, anyway, to confirm its capability as a functional containment.

The other one, other PDC that got some attention was the PDC 26, reactivity control system segregated the PDC into RFDC, DCN, and under control design criteria, the concern was expressed that the language suggested a departure of the defense in depth philosophy that safety will not be wholly dependent on any single element of the design construction, maintenance, or operation of a different facility.

X-energy provided assurances that the set of RFDCs, CDC, and other OCDC could collectively serve to meet that standard.

The staff's limitations and conditions provide final closure for X-energy -- excuse me -- Xe-100 PDCs.

However, these limitations and conditions incorporate language expectant that subsequent development of the Xe-100 safety case engagement with the staff will result in their removal potentially requiring a revision to Xe-100 PDC TR.

The regulatory precedent of a clean set of design criteria established by general design criteria for LWRs should be a strong motivator for the subject of such well-articulated PDCs provide the best assurance of the building and efficiency of staff design for units.

The opinion of the committee is that the staff's SE report is sufficiently complete to recommend that it be issued.

It is also recommended that this write up serve as a record of the subcommittee meeting and an ACRS letter report will not be prepared.

Again, recommendation only and, obviously, we can proceed specific or general and any manner that would be otherwise.

The main issue that came out of the meeting, as I mentioned before, was related to the functional containment.

And while, of course, this is a meeting on X-energy, not necessarily the four, obviously, we should kind of focus on, you know, exactly that.

At the same time, maybe more actions that come out, we can talk about maybe alternative letter, maybe on our observations on functional containment PDCs in general.

And maybe -- well, I don't know how much interest in, you know, in that particular one, there might be, but certainly, PDC 16 caught our attention.

And given that we've seen the four design centers now, we may want to write something.

But I wanted to open it up at this point.

My personal impression was this, I felt like the application of Reg Guide 1.253 which brought in the NEI 2107 framework which is just a kind of attached on to the NEI 1803, you know, kind of provided the kind of the anchor to beating the expectations in Reg Guide 232 which is related to the multiple barriers.

I thought they were very specific. You know, they mentioned the fuel, you know, the particles and the fuel, the pressure boundary specifically, I think, in the tradition of MHTGRs.

They kind of covered the bases.

They did depart from the word multiple, but in doing so, I felt their -- the specificity of identifying the barriers was adequate.

You know, this staff in their SE, I could read that, too, also acknowledged it and maybe even whether they were sensitive or not, kind of acknowledge the fact that multiple was missing, but that they were satisfied that, indeed, it met the multiple barrier standard.

So, with that, I'll open it up and maybe Tom wants to chime in since Tom, Walt, and I were members, at least initially, on this little email and --

CHAIR KIRCHNER: I'd also ask Dave if he would -- because Dave went through the NGMP experience which was really the basis for the SARRDLs concept as well as functional containment as was eventually embodied in the Reg Guides.

Maybe we could turn to Dave first --

MEMBER MARTIN: Okay.

CHAIR KIRCHNER: -- for the background.

MEMBER PETTI: So, the multiple thing doesn't bother me so much because I see multiple values here.

Saying the particle is somewhat short hand. From the NGMP days, we just say the particle had at least three barriers, depending on how you count them.

CHAIR KIRCHNER: Right.

MEMBER PETTI: Right?

Each of the coating layers and the kernel itself.

The kernel itself is a ceramic, is a pretty darn good barrier for some of the fission products.

And then, you have the graphite. X-energy says the pebble. It's a little different than structural graphite.

But there's diffusion coefficients for both of them, so you can model it. It's not, you know, that big a deal.

And then, they have the coolant system pressure boundary.

What they don't have is the building.

But NGMP was a 600 megawatt, a large reactor.

These are small reactors.

And so, yes, do you need that to take credit for that?

Probably not.

As you shrink the size, you know, this whole functional containment study sits around with the source term is at the site boundary.

And as you shrink the size, you should -- the main new cladding, and that gets you some, you know, you can maybe get rid of the one barrier.

But the most that it was ever credited for, it's not credited heavily.

And in NGMP, in fact, there was lots of concern about how much to credit it. It had, you know, was declared.

Furthermore, when you look at Xe-100, which we haven't looked at in detail, but given its size, it probably has peak temperatures and accidents that are not as severe as an MHTGR.

So, that, you know, there's also the details in the accident space that will influence what that source term is.

So, you know, I don't, you know, they felt they wanted to clarify, I mean, what multiple meant, I guess.

VICE CHAIR HALNON: Dave, just one quick question.

When I compare what you just said about three barriers in the TRISO fuel and compare that to what we're traditionally used to, which is the clad RCS and containment building.

I have the question, is -- and what people are used to seeing is three very distinct, separate different manufacturing processes. You've got the fuel plant in one area of the country, you have an RCS, different code, different material, then you had a containment, same thing, different code, different material.

Is the TRISO fuel manufacturing process such that you can't get a pebble or defects such that all three of those barriers could have a single defect problem with that that couldn't be -- may not be detected?

MEMBER PETTI: Oh, no, I mean it is detected. I mean, that's --

Yes, we have -- they have limits on the defects, silicon carbide defect being probably the most important.

But there are other defects and they all characterize -- they all have limits and they're down in the, you know, usually the 10 to the minus 5 rage.

VICE CHAIR HALNON: So, in your opinion, is those three barriers an equivalent level of containment, if you will, as the three barriers that we're traditionally used to seeing because on the different diversity that we have there?

MEMBER PETTI: It's at least two of them to the -- compared to the traditional.

VICE CHAIR HALNON: I'd hate to make it

simplistic --

MEMBER PETTI: Yes.

VICE CHAIR HALNON: -- case out of it.

MEMBER PETTI: Yes.

VICE CHAIR HALNON: But that means that if you reduce one barrier, now you're down to just one barrier --

MEMBER PETTI: Right, right.

VICE CHAIR HALNON: -- for the containment.

MEMBER PETTI: And that's not -- I mean, it's -- there's multiple barriers.

And this, you know, because there's a functional containment, and I believe there will be a mechanistic source term.

The hold up for the different isotopes is different.

VICE CHAIR HALNON: You have to take into consideration the different technology --

MEMBER PETTI: Right.

And then, you know, silicon carbide is, you know, critical for almost all of the radionuclides.

And graphite is very important for cesium and stratum which are strong dose contributors.

Right?

But iodine, which is the other big -- that graphite until they just go through.

So, it's not a barrier there, so you get all these caveats which, when the study gets put together, they should be okay.

If we turn to Kairos, right, they had the fuel as one barrier and then, they had their coolant.

Their coolant is an amazing barrier.

So, what they did was they went and was -- were what gas reactor guys would describe as somewhat cavalier, allowed for a tremendous amount of defects and failures of their fuel in the reactor because they have the strength of the second barrier.

And these things trade off in design space.

And so, it's not surprising that we'll see differences, you know, as they come in.

But the real question is, you know, it's more complex than just that. It has to do a lot with the source term which has to do with the size and the site boundary and all of that stuff happening together.

MEMBER BURKHART: Excuse me for interrupting, the court reporter just has a message.

I think he's new and if you could state your name before you speak, at least for a little while.

And please, get closer to the microphone. It's -- when you sit back, it's difficult for him to hear.

VICE CHAIR HALNON: Yes, this was Greg Halnon was the questioning and Dave Petti was the responder.

MEMBER MARTIN: And I will add, you know, you start pulling the thread now, and this probably comes as no surprise, but you know, as I was kind of preparing for the meeting a couple weeks ago, and you know, pulling the thread on the history, yes, it goes back to the '80s. Right?

And the language that's even used in the Reg Guides that were not so old go back, you know, that these conversations have been done time and again since the mid, late '80s, maybe farther back with per se brain, you know.

But it's not a new topic, you know, containment versus confinement debate, you know, within --

(Simultaneous speaking.)

MEMBER PETTI: -- did have a containment.

And the fuel they had 10 to the minus 3 fuel quality, so a tenth of a percent of the fuel had defects, you know, but that's not the fuel --

MEMBER MARTIN: Was that bi-cell, though? Was that --

MEMBER PETTI: No, that was tri-cell.

MEMBER MARTIN: It was an early version of tri-sell?

MEMBER PETTI: It was carbide fuel. It was highly enriched, but also it's a different means and not made necessarily the same way.

So, the other thing, though, that's important is that we always said in NGMP, GA would always say, you cannot take credit just for the powder cladding.

For at least a 600 megawatt thermal image TGF, you will never meet the pads at 400 meters if you only account for the particles, that even at what we think is incredibly high quality, the dose is too high and it's all driven by iodine.

So, you needed to take credit somewhere else.

Again, 600 megawatts as you make them smaller, the story will change and that's why it's going to be really important to understand what the source term is as you move forward.

But similarly, any of the technologies, whether it be, you know, TerraPower, I mean, there's certain isotopes that you know that you've got to make sure that they understood -- that's all understood.

MEMBER ROBERTS: This is Tom Roberts.

To some degree, the question on X-energy is more philosophic than technical because what they've done is, they've taken a principle with is currently in the Reg Guide that you shall have multiple barriers for functional containment and define the approach they will take to meet that requirement.

So, instead of stating the principle, they're stating an approach that will meet the principle which seems fine, except you wonder, now that the principle is gone, whether there's some, you know, downstream consequence of either somebody leveraging this document, some other, you know, applicant or X-energy themselves say, well, the requirement was to have this fuel, not to meet the multiple barriers criteria.

Like Bob said, we've seen four of these PDC topical reports, two of them did not change the language for the multiple barriers.

The one we're talking about now, they changed the language, but they require, essentially, what the underlying basis was for the multiple barriers.

Then, we'll talk about it tomorrow which is, eventually, they took the language out to say, yes, we're going to do whatever the LNP process tells that we need to do which is now even more general in terms of not talking about any sort of deterministic barriers, but whatever the probabilistic approach tells you you need.

And they will, in effect, have the same architecture as very similar as X-energy with their fuel choices.

But it's just what's stated in the PDC is the question of whether it should be stated as a matter of principle or as a matter of this is exactly what we're going to do and not to state the principle.

So, that's the question I'd leave out there.

We -- this committee wrote a letter in 2018, as Bob alluded to in his comments, that had a recommendation to always have some degree of, you know, redundancy, independence, whatever with the containment barriers, that you should never be down to just counting on one thing as your containment barrier which is a deterministic type of criteria.

And whether you would allow a probabilistic assessment to convince you that that's not necessary.

I mean, it is a different question.

And that's a question that is not currently on the table with the approaches intended by these applicants, but it is, you know, on the path to get there without stating the requirement for redundance or multiple barriers and the PSC.

So, that's the issue I'd like to discuss, whether that applies better for this one or tomorrow's discussion we talked about, but it's something that's what we're pursuing as a committee, whether the statement of principle, you know, we think should remain in the PDC regardless of the implementation.

MEMBER MARTIN: But maybe someone -- I have had a question on this for some time, but the use of general versus principle design criteria, what was the intent of these two words?

I mean, I can see back in Appendix A, in the '60s it provided for maybe a lot of design types, you know, under the umbrella of light water reactors.

And the, here, for advanced reactors, we have a different word here.

And I wonder maybe related to the question of where, you know, how functional the criteria are expressed, whether there is a nuance between the use of the word general versus principle where we intended to be more specific.

MEMBER PETTI: No, I just always saw it as sort of a synonym, without -- because GDCs mean something in an LWR context.

The advanced reactor guys wanted a different terminology.

I may mean the same thing at the deck point, right? But it looks different and it flows different because they're very into top down.

That's how you make it technology neutral and you go with the functional code.

So, that I never thought that there was a difference there.

VICE CHAIR HALNON: I think they needed to distinguish between the old way and new way of doing things.

But there's also another new way which comes back to the philosophical issue, they're seeing the designs --

In the past, you had a very distinct set of general design criteria. You designed your plant to those to make sure that you had the redundancy and single failure proofs and everything else you needed.

Today, what we're seeing is sort of an upside down approach where the PDCs are being modified to meet the design rather than the design being met to meet the PDC.

And that's --

MEMBER PETTI: or at least iterated.

VICE CHAIR HALNON: Iterated.

And it's -- I don't think that's wrong. I think that's kind of the way it's been the construct is driving the folks to do this.

Paradigm in our mind, we've got to get straight that that's not wrong, but we have to make sure that it's effectively meets the need for the safety of the plant now.

In this case, and I think since Larry just walked out, Rob, could you guys take PDC 16 and 26 for the four design criterias? Put them next to each other for us and show us the deviations and see -- let us just take a look at those to see all together whether or not we see a philosophical trend or a physical trend that we need to continue to talk about.

I think that would be useful for us.

MEMBER MARTIN: I mean, I can pull it up. I have it on my scree here.

VICE CHAIR HALNON: All four design centers?

MEMBER MARTIN: You're not --

VICE CHAIR HALNON: Oh, not all four, no just -- no, I'd like to see all four next to each other just so that we can see how these are trending one way or the other, fewer barriers, changing language.

Just, you said, Tom, there's a couple of them had not changed.

So, I'd like to see it in a chronological way, which one was first, second, third, as we're iterating through these things.

Because I think what we'll see is, you know, it's the world of positive plagiarism, you know, what's been accepted before is something where you start with and then, you work from there on these designs.

So, that would be useful for us to see that and I think as we go forward, keeping that up to date.

And we may expand it to more of the PDC, but those two are right now the ones we're talking about.

MEMBER MARTIN: Well, I would say what X-energy did on 16, it's different.

Like I mean, in the sense, you know, I think their main goal was to adapt the Reg Guide, you know, the recent 253 Reg Guide, which the others have not done.

And so, in doing so, they got a little more specificity than the -- so, I throw it up there real quick.

So, you can see the specificity in their statement, which I think is unique among the four that we have seen.

And to the point of uniqueness, it's the combination of the required functional design criteria and the complementary which they said to my question as a really in practice, any difference with how you approach demonstrating these criteria is no, not really, they're segregated a little bit by the events that they'll use to basically demonstrate the criteria.

MEMBER ROBERTS: I want to take a shot at answering Bob's question to see if I got it right.

My understanding is that the PDCs have always existed and the requirement for light water reactors to start with the GDCs.

And you had to meet the GDCs and incorporate them into your PDCs for your design. That may be a super set of them and may be identical, it depends on the details of the reactor.

And then, when the advanced reactors came about, the NRC, I wrote that Reg Guide 1.232, which was the rough equivalent of the GDCs, but there was still an expectation that the individual applicants would take the ARDCs, the SFRDCs and the CRGRDCs were the three kind of pick lists of substitute for general design criteria.

And then, tailor them to their plan to be the PDCs.

I don't think anything's materially changed, you know, over the years, I've always had to start with some set of criteria and show how you met them and add them, add whatever you needed to to fully cover your design.

I got that right? I see some heads nodding.

MEMBER MARTIN: Well, I was hoping, you know, a response kind of like that because there's, you know, they're not just, you know, GDC or PDC, there's always a tier of criteria at the top levels is what the conversation is between the design center and the staff.

And then, when you get into the weeds on other topicals, these other criteria that support the GDC or, in this case, the PDC, you know, are more specific and, you know, and collectively.

Then they rise up to address the more general or principle design criteria.

We will, you know, we will see criteria or requirements in other topical reports that ultimately percolate up to these high level ones.

I think what we might be seeing with the new reactors is that maybe a step down because they know, you know, they know what their design is and they can be more specific.

And so, they're highest level design criteria is going to be more specific than we're used to seeing with, say, a GDC or maybe even some of the other designs, you know, that came prior to.

But this Reg Guide, the Reg Guide 1.253.

MEMBER PETTI: Some of the reorganization in the X-energy stuff sort of made sense.

You know, when there was inspection stuff in five different places, they pulled it all into one.

I mean, some of that stuff, you can

imagine, down the road from an efficiency perspective, demonstrating compliance with it, it'll just make the story easier than, you know, having to pick at five different GDCs.

So, you know, but when I went through that, I went -- I mean, it made me wonder why that wasn't kind of the Reg Guide at that stage.

You know, I mean, there was a lot of time to develop.

MEMBER MARTIN: They put a lot into that.

MEMBER PETRI: Yes, they did and, yet --

MEMBER SUNSERI: This is Matt Sunseri.

I guess, interesting discussion we're having. I've lost focus on what we're trying to achieve, though, here.

What are -- what's our purpose?

MEMBER MARTIN: Well, I think our purpose is to decide whether we, you know, kind of accept, basically, the recommendation from the subcommittee that the summary was sufficient or do we consider further action, I think, on the subject of functional containment.

And, you know, maybe with the tone of informing the staff that maybe we -- there should be more specificity or maybe certain guardrails that need to be highlighted.

I don't know, but I think that's up to the committee to decide whether we look at functional containment in a very specific sense.

Obviously, in doing so, I think we have to look at four designs centers that we've seen and kind of, you know, pull lessons learned from the four.

I think would be -- being a little careful because this is X-energy, right, fundamentally and that we should stay focused a bit on that, but bringing insights from the other experiences is worthwhile.

MEMBER SUNSERI: So, I guess I would -- that I thought our whole objective of the organization or the way the licensing process works is to be able to tailor this to your specific technology and be inclusive of everything.

And we had a good discussion at the subcommittee meeting and I thought we were satisfied that this applicant had met the criteria at that point.

So, I thought we were ready to move on despite the fact that there may be some language complexities here, technically, what you're doing is right.

MEMBER MARTIN: Yes.

MEMBER SUNSERI: Is that an actual statement?

MEMBER MARTIN: Yes, I guess, to kind of move this slightly in a different topic, but still be on functional containment, one issue I kind of had was the deviation from SARRDL to something more kind of, you know, not defined so well, and that was just the functional containment design limits.

I thought it had ambiguity that you needed -- need to be rectified.

I just didn't like, you know, and SARRDLs, I guess now, you know, I think we all kind of understand what that's trying to do.

And when you kind of use a new term there without, you know, I think good definition, it doesn't help the conversation or it leaves the kind of an open item, you know, to be resolved and, you know, other engagement.

And maybe at this state, it's fine. But SARRDLs pops up, you know, time and again in the Reg Guide and it just wasn't in this PDC as such.

You know, except, of course, we talked about this, well, okay, because could SARRDL, but it might include other design limits, which is fine, but ultimately, SARRDLs kind of go back to all those SAFDLs or whatever you want to consider.

MEMBER PETTI: So, this is Dave.

Did they not commit to SARRDLs, thinking it is a lower level issue?

Or did you push the SARRDLs in the subcommittee?

CHAIR KIRCHNER: Can you show the --

MEMBER MARTIN: So, they used, okay, not limits --

CHAIR KIRCHNER: Could you go to I think it's PDC Number 10 of the SARRDLs?

MEMBER MARTIN: Like I said, the point I was making here is that, you know, I mean, this goes back to the Reg Guides.

And they used limits instead of conditions, so I'm not sure if there's any, but the subtlety is there.

I don't know why it does that, I can't seem to --

PARTICIPANT: We can't see it now.

MEMBER MARTIN: Yes, exactly.

There we go, that's a little better just so you can see the words.

So, yes, SARRDLs are there, but I just,

you know, the conversation where on -- and maybe this is just me, but on 16, you know, we don't have SARRDLs, we have just something kind of --

And I'm not sure what the subtlety of conditions versus limits are.

It seems in practice and I have little experience on practice.

It's the SARRDL that we focus on for functional temperature and functional containment and criteria related to meeting SARRDLs goes back to also some say traditional SAFDLs.

CHAIR KIRCHNER: So, Dave, when you -- this is Walt Kirchner.

Dave, when you -- when NGMP first was developing these, then I presume you mentioned, you know, mechanistic source term.

So, behind this then, the test really was the dose at the boundary was within the 10 CFR 50 or 52 limits.

MEMBER PETTI: Well, usually that was easily met. It was usually the pads, the EPA pads on rem at the -- I mean, we certainly had flat as the most stringent dose requirement.

But here, I mean, SARRDLs are -- SARRDLs have less -- have to do with knowing that your functional containment is intact.

CHAIR KIRCHNER: Intact, right.

MEMBER PETTI: Right?

You're measuring radionuclides in the helium, and that tells you how good the fuel is, whether something has failed or not above what you assumed in your SARRDL.

CHAIR KIRCHNER: But then, in implementation, then there was a tech spec limit --

MEMBER PETTI: Correct, it will be tech specs, absolutely.

CHAIR KIRCHNER: -- that you worked against and then --

MEMBER PETTI: Right.

And in --

CHAIR KIRCHNER: -- if you passed that threshold, then you --

MEMBER PETTI: Right.

CHAIR KIRCHNER: -- effectively had to some remedial action --

MEMBER PETTI: Right.

CHAIR KIRCHNER: -- on some documents.

MEMBER ROBERTS: Yes, Bob, let me try this.

This confused me, too.

The SARRDLs are tied to ALOs and normal operation and the PDCs.

Containment is not there for ALOs or normal operations, they're for accident conditions.

But what Dave said earlier is, the key parameter is dose at the whatever boundary you're calculating dose at.

So, I think there's an either unstated or implied limit in PDC 16 which is the dose of the boundary, but the SARRDL concept wasn't intended to apply to containment type of assessments.

Is that right, Dave?

MEMBER PETTI: Yes, I think you're right.

MEMBER ROBERTS: So, I think that's where the SARRDL was there to come up with a way to estimate ALO consequences for a fuel type that doesn't really have, you know, fuel limits that are challenged by ALOs.

And so, it gave you something meaningful to calculate for ALOs, at least that's my understanding.

MEMBER MARTIN: So, you feel that the functional containment design conditions or limits are kind of more restrictive, covering a broader set of events, not necessarily beyond design basis as the focus, the tradition of GDCs kind of stopped at design.

CHAIR KIRCHNER: If I could -- this is Walt Kirchner.

Just enter in here to Matt, at a higher level, I think one of the important things as we look at these submittals, these TRs on NMPDC, is to remember the important role that --

Let me go back to GDCs, and if you think through how the agency -- the staff goes through reaching a reasonable assurance conclusion, it's by and large, it's not limited just to the GDCs, but one of the important steps in making that finding is to test the design against the GDCs to determine adequacy of the design.

So, the ensemble of the GDCs is an important aspect in reaching the reasonable assurance that there's no undue risk to the health and safety of the public.

So, but it's the entire set of GDCs and, hence, you will -- what we see, we take it for granted, but it actually provides structure for the staff's review and they systematically go and make an assessment.

In 0800, the standard review plan or a, I think it's a tailored review plan, essentially, systematically used the GDCs as the basis for making a determination of adequacy of the safety systems, et cetera.

So, in the course of reviewing these, what I have been looking at is, are they -- this is perhaps not the best word -- but are -- when you come down from, as Tom said earlier, from the higher set of criteria to tailoring to the specific design, do you still fully meet the intent that was there in the Reg Guide to being with for that particular safety function?

And that's the -- so, I'm looking at these and you say what's in a word?

If you strike the plural from barriers, so you -- if there aren't multiple barriers for a functional containment, that would be of concern because the whole idea is to provide a robust system -- a containment system that isn't reliant on just one single barrier to accomplish the function.

And we were pretty explicit about this in our letter report back in 2018.

And if I -- if you'll indulge me, I'll just read this.

Item Number Four, that a functional

containment should include multiple barriers as a defense in depth feature that -- and that they should be minimally dependent on each other and diverse in nature.

And I think the latter clause was you didn't want to have consequential failure of the barriers.

So, there -- that introduces some diversity and the concept of multiple barriers.

MEMBER MARTIN: Right.

And I think -- and maybe I'm not remembering this correctly, but the whole concept of the TRISO fuel enables the concept of a functional containment because of the multiple inherent barriers in the design of those particles.

And then, when you combine that with energy that's available to distribute the radionuclides much different than a PWR at 1,000 pounds pressure and 500-something degrees.

Then, you know, these are reasonable expectations.

And I thought we had come, you know, back to this specific example how we could come to the conclusion that this design met that criteria.

That's all I was saying. So, I'm not

disagreeing with your little optical umbrella that you painted here.

I think it is our responsibility to make sure that we've done this review.

But it seems to me that we have done that review, that's all I'm saying.

MEMBER PETTI: I think the issue gets much foggier as we move away from the concepts for which functional containment were developed which is gas reactor, right, eVinci, there's some real uniqueness there.

Let's talk about a dissolved fuel system.

We know the multiple barriers, you know, I mean, then things get much, much cloudier on the spectrum.

MEMBER MARTIN: I think everyone would agree that this should be the easy one. The four we've seen, this should be an easy one.

MEMBER ROBERTS: Yes, Tom Roberts.

I don't know if we'll get to the comparison of the four topical reports, that cracking, that's about a while ago. We can talk about that tomorrow.

But I think it's more of a general question of looking at any of the trend and the types of language and how it's potentially setting precedent for the next one that may be, like Dave said, is considerably different from it was when it was evaluated before, but out of the precedent is well, you just go, use LMP and LMP tells you, you know, what's good enough in terms of containment, which is certainly a part of it.

But that doesn't -- it doesn't seem like all of it.

But I agree with you, Matt, that the X-energy, they've simply described how they're going to meet the requirement in place of the requirement.

It's, from a technical perspective, kind of hard to argue, but the adequacy because whether we say have multiple barriers to the exact language of, you know, MHTGR 16 or what they've written, they'll get to the same place.

So, it's really kind of setting up the discussion for your expectation in general.

The other technical question was 26.

You know, you want to go to that one, Bob?

I wasn't sure I understood the issue in 26.

My recollection from the subcommittee meeting was that there was a staff slide that implied that the natural temperature feedback was the backup means. And I think staff agreed that that slide was wrong.

(Simultaneous speaking.)

MEMBER ROBERTS: And you see in the slide --

MEMBER MARTIN: So, let's just recall that some of this was in closed session, right? So, just --

MEMBER ROBERTS: Okay.

If you assume the slide is wrong and it seems like the text of the -- of their recommended PDC is just fine.

So, I'm wondering if there's any -- if you're seeing any remaining issue.

MEMBER MARTIN: Oh, absolutely not.

You know, we got the clarification of there's specific design and strategy in the closed session.

And I think that that might have been where we got the acknowledgment that maybe that the slide was incorrect, the staff slide or not specific enough.

MEMBER PETTI: Well, being that, you may not be able to answer the question.

The second diverse means of shutdown is not the inherent temperature coefficient?

MEMBER MARTIN: Yes, that's correct.

MEMBER PETTI: Yes, that would be additional defense in depth.

MEMBER MARTIN: Right.

And that was, you know, the inherent really what addresses is 11 -- PDC 11 I believe that's --

MEMBER PETTI: That's a big -- I mean, it's been an issue with gas reactors for a very long time, whether they could do that and not need a second system.

MHTGR always had a second system.

MR. MOORE: To the members, this is Scott Moore, the Executive Director.

We can prepare the side by side comparison of the four PDCs for tomorrow morning, I think.

We're not going to have that during today's discussion.

CHAIR KIRCHNER: Thank you, Scott.

MEMBER PETTI: And again, I would just, you know, say again that this is all iterative, right?

I mean, if you look at NEI 1804, you look at LMP, you go through and you may have to come back, that once you've done your PLA and you have to adjust things and change things.

I mean, designs go through iterations, we just tend not to see them.

MEMBER MARTIN: Right, right.

MEMBER PETTI: Right here at ACRS, that's all in the background.

While here, all this stuff is looped with it so --

VICE CHAIR HALNON: And I think that -- this is Greg -- that's part of the issue is the design's iterative, but the PDC should not be because they set the design criteria.

MEMBER PETTI: But it's hard to see what the right propagate --

(Simultaneous speaking.)

VICE CHAIR HALNON: And that was my point earlier, is that as we come through the designs, that might have a better way of doing it to change this language in the PDC, which means I have to go back through it.

So, that whole aspect is we have to be nimble enough to be able to accept that when it makes sense and it's good for safety and not fall back on our old ways of thinking on some of these.

But so, on the table right now is a question. Do we stay with the subcommittee recommendation to enter -- and we can play with the words if you want to, with the summary that Bob read or do you think we need to write a letter that encompasses the bigger question?

I would offer up, there's another option which is to accept the subcommittee's recommendation at this point and open up a topic based on the action to continue to watch these critical PDCs that are changing in time and having a potential meeting down the road where we have determined that there is now a bigger question that is erupting that may cause us a safety concern.

At this point, I haven't heard a safety concern. I've heard a lot of questions and a lot of concerns from the standpoint of trends and whatnot.

But until we get the data in front of us, we don't really know we see maybe a couple more PDCs, we may not see a trend.

I think it's a fruitful discussion to have philosophically, but, at this point, my recommendation, and I'll put the motion, you know, I guess Walt has to put the motion, but I would say, make a motion that we accept the subcommittee write-up and trend and track -- and maybe, Bob, you and Tom, since you've taken an interest in this, kind of take the lead on looking at these two PDCs and how they are trending down the road. And then, also make -- decide whether or not we need to expand it to other things as well.

That would be my motion to accept it and then open up the top kind of a small task with you two guys watching the trend in the PDCs.

CHAIR KIRCHNER: So, we have a motion on the table.

Do we have a second?

MEMBER SUNSERI: Yes, I would second that.

CHAIR KIRCHNER: Someone seconds?

MEMBER PETTI: Matt did.

MEMBER SUNSERI: Matt Sunseri, second.

CHAIR KIRCHNER: So, discussion?

Vicki, you raised your hand?

MEMBER BIER: Yes, I agree with Greg's recommendation.

It seems to me that there are not a lot of serious issues with the PDCs that we just reviewed in subcommittee, but that the general trend of whether we are on a slippery slope of accepting things that might, you know, reduce safety margin, you know, deserves some awareness going forward.

So, I think that's a sensible manner in which to handle this.

CHAIR KIRCHNER: Any other --

MEMBER SUNSERI: Yes, this is Matt again. I would -- just following up with Greg's point, I agree with, you know, trending the lessons we learned from the various applications we're reviewing.

And I just want to be cautious not to undo all the good work that was done to create the PDCs for the advanced designs because the idea was to move away from a reactor specific set of GDCs like the PWR ones.

And if we try to force the language back to make them all look the same, then we will have the same problem as we had with the light water reactors, is that you have to have all the, you know, take all these exceptions to the GDCs when the new things come around.

So, I just want to be mindful that, you know, we don't inadvertently through wanting to retain institutional knowledge or our decisions that we, you know, create a bad situation going forward.

That's all.

CHAIR KIRCHNER: Well, one thing that I think we should be looking at, though, is the perception that there was a lot of work put into developing those appendices to that Reg Guide.

And there was a lot of thought and there was a lot of experience behind those, particularly for the MHTGR, but also for the sodium fast reactors that codified a lot of experience that the agency had in reviewing earlier designs.

And so, when the applicant has -- it's just guidance that's understood -- but if there is a deviation to accommodate the specifics of a design, one should be asking some hard questions in terms of justification for why there is this variance from, let's just pick Appendix A of 1.232 to start with, which was meant to be just what you said, Matt, technology kind of neutral or inclusive.

And by codify -- but experience had, you know, have been accumulated over time looking at the development of surrogate general design criteria for advanced reactors.

And what I've seen is that, yes, the changes appear to be justified.

Of course, we're in the early part, we haven't reviewed the details of the design. So, we're kind of a little bit under informed, I would say.

So, we're looking at these, so, I'm

thinking, does the ensemble, the totality of these PDCs provide a level of safety, if you will, that is comparable to what is required of the current generation of LWRs?

The advanced reactor policy statement is pretty explicit on some of these things.

It said, regarding advanced reactors, the Commission expects, as a minimum, at least the same degree of protection of the environment and public health and safety and the common defense and security that is required at the current generation light water reactors now.

So, from my perspective, the PDCs are an important first step and demonstrating that the concept is going to provide and comparable or, what is the worry again that I just -- the same degree of protection as is provided with the LWR fleet.

And that's primarily derivative of -- it's not just the general design criteria, it's a lot of other things.

But its' very important first step.

So, that's my concern at a high level that in the tailoring, Matt, that there's not a dilution of the fundamental objection of that particular requirement.

You know, and 26 is -- and there are a few key ones, 10, 16, and 26 because they get at the basic, the three basic safety functions, essentially.

When you think about it, it's, you know, we've got 26, is controlled reactivity.

And you know, the fuel 10 and 16 bear on the fission product barriers.

And then, the others, and one way or other, of course, need to be tailored, but it's control of decay heat.

MEMBER MARTIN: I mean, I know we're in theory in discussion from the motion, maybe I should ask this as a question rather than an observation.

So, with light water reactors, you, of course, you have the general design criteria, they have a hole, you know, regulatory framework that would just establish for LWRs and ultimately, you know, people fall back to NUREG 0800 as, if we follow NUREG 0800, we meet all the GDCs.

But you don't have, at least in my experience as someone just explicitly tracing work, you know, from, all right, I do this analysis. It's addressing this particular GDC and you see it map all, you know, you see how it maps back to, you know, Appendix A.

There's a lot of similarity and there's a lot of differences between the new reactors and light water reactors.

What I wonder may be missing in regard to PDCs is some, you know, and maybe in other Reg Guides where it says, you really need to have that traceability in all the documentation so it doesn't get lost because we don't have a NUREG going on. We don't have settled law on everything that is X-energy or TerraPower or, you know, et cetera, et cetera.

Some effort needs to be made by the applicant to say, I do this work to meet these criteria. And you can trace it all our documents.

VICE CHAIR HALNON: Just real quick, Bob, and all due respect, the PSARs in each of the sections have this satisfies PDC, empty spot, empty spot, empty spot.

So, if you -- in each of the sections of the PSAR does map.

Now, it may not be directly as direct as what you're saying, but it is -- there is a conscious effort to show PDC and this what meets it.

MEMBER MARTIN: For LWRs.

VICE CHAIR HALNON: No, for the -- for what we've been looking at.

MEMBER MARTIN: You mean like in the Regs? So, Reg Guide -- the 233 that --

VICE CHAIR HALNON: Well, I may go back and forth, didn't Kairos, you won't have this, does this satisfy PDC something.

MEMBER MARTIN: So, they were sensitive, you know, but are they doing it because they know that's how you -- that's the best way to present the information or are they doing it because, you know, somewhere they -- a seasoned licensing manager will know to do these things. Right?

So, there's a difference between, you know, being good at your job and, you know, following instructions. Right?

And that's what I'm saying is that, do you need to have the specificity in the Reg Guides to say that you, you know, we expect that this kind of thing because before it was kind of understood if you're following NUREG, you know, 0800, that you were meeting, you know, all the GDCs.

That's my point is that it's not in black and white.

Someone that's done this over and over again I snot going to know to do that.

New reactors, you're going to have a

different -- maybe different path to get -- to show that traceability.

MEMBER ROBERTS: Okay, if I could comment on the motion?

I agree with Greg. I agree with Matt.

I think both of them are expressed, you know, important concerns that shouldn't be tied to the language that's in the current Reg Guide DCs of the Appendix A for technology that doesn't apply to it.

But it all starts with the principle and then, deriving from that principle and whatever the language that derives from the principle ought to be clear to, you know, trace to the principle is, which I think we get there with this one.

So, I'm not very concerned with the E-energy PDC document.

So, you know, I'm going to agree with the motion.

The only caveat is, I think there's some parts of the summary report that probably ought to be tweaked. I don't know what your right is in terms of making comments to Bob and then coming up with some mutually agreed upon version to present tomorrow or what --

CHAIR KIRCHNER: Yes, yes, can you --

MEMBER ROBERTS: But I would, I'd like the chance to, you know, have some input into the write up.

CHAIR KIRCHNER: Sure, can you --

MEMBER MARTIN: We'll do that tomorrow anyway.

CHAIR KIRCHNER: Yes.

Can you distribute that to everyone?

If there's no --

Are there any -- we've already showed that so there's no proprietary restrictions. Okay.

So, all right, we have a motion on the table and we've had discussion.

All those in favor?

Vesna, are you with us?

MEMBER DIMITRIJEVIC: Yes, and I am in favor.

CHAIR KIRCHNER: Thank you very much.

Okay, with that, I think we are finished with the morning session.

Any other comments?

(No response.)

CHAIR KIRCHNER: We will be engaged at 1:00 and we will have the Seabrook informational presentations at that time.

Okay, thank you, everyone.

For the court reporter, we will sign off now and come back on at 1:00 Eastern Time.

(Whereupon, the above-entitled matter went off the record at 9:40 a.m. and resumed at 1:00 p.m.)

CHAIR KIRCHNER: This is the afternoon session of the first day of the 718th Meeting of the Advisory Committee on Reactor Safeguards. I'm Walt Kirchner, Chair of the ACRS.

The ACRS members in attendance in person are Ron Ballinger, Vicki Bier, Greg Halnon, Craig Harrington, Robert Martin, Scott Palmtag, Dave Petti, and Thomas Roberts. Matthew Sunseri has recused himself from these discussions due to a potential conflict of interest.

We also have Vesna Dimitrijevic online with us.

Good afternoon, Vesna.

Consultants attending, I believe --

MEMBER DIMITRIJEVIC: Good afternoon.

CHAIR KIRCHNER: -- thank you -- are Dennis Bley and Stephen Schultz.

If I've missed anyone, please speak up now.

Kent Howard of the ACRS staff is the

Designated Federal Officer for the full Committee meeting this afternoon. And I note that we have a quorum for our meeting.

During this second session today, the Committee will receive an informational briefing and update from Region I staff on the status of the NRC's oversight of the alkali-silica reaction, commonly referred to as ASR, at the Seabrook Station.

And now I'll just go on to make the following note that the ACRS was established by statute and is governed by the Federal Advisory Committee Act, FACA. The NRC implements FACA in accordance with its regulations found in Title 10 Part 7 of the Code of Federal Regulations.

Per these regulations and the Committee's Bylaws, the ACRS only speaks through its published letter reports. Therefore, note that all member comments should be regarded as only the individual opinion of that member and not a Committee position.

All relevant information related to ACRS activities, such as letters, rules for meeting participation, and transcripts, are located on the NRC public website and can be easily found by typing About Us ACRS in the search field on NRC's homepage.

The ACRS, consistent with the Agency's

value of public transparency in regulation of nuclear facilities, provides opportunity for public input and comment during our proceedings.

I note today we have C-10, a public interest group, with us to make a presentation later in the session. Other than C-10, we have received no written statements, comments, or requests to make an oral statement from the public, but we will set aside time at the end of this meeting for that purpose.

The ACRS will gather information, analyze relevant issues and facts, and formulate proposed conclusions and recommendations, as appropriate. A transcript of the meeting is being kept and will be posted on our website.

When addressing the Committee, participants should first identify themselves and speak with sufficient clarity and volume so that they may be readily heard. If you're not speaking, please mute your computer on Teams or your phone by pressing star-6.

Please do not use the Teams chat feature to conduct sidebar discussions related to the presentations. That is limited to the use of reporting IT problems.

I go on to ask everyone to mute any

electronic devices, put them in silent mode, and point out for our presenters today that these microphones that we're using are rather unidirectional. So you'll need to speak directly into the front of the microphone to be heard online and as well by our court reporter.

With that, I am now going to turn the Committee's deliberations over to our Plant Operations Subcommittee Chair, Greg Halnon, and Co-Chair Ron Ballinger.

Greg, go ahead.

VICE CHAIR HALNON: Thank you, Chair Kirchner. The ACRS has been involved with this topic, which is unique to this nuclear plant, since 2018.

After being briefed on multiple occasions by members of the NRC staff and the Seabrook licensee staff, all with ASR expertise, the ACRS issued letters to the Commission regarding the ASR-related license amendment and issuance of a renewed license for Seabrook with two separate letters.

Most recently, the ACRS last met with the staff on April 27, 2022, when they provided an update on the status of ASR at Seabrook and on the Seabrook licensee's implementation of its ASR Monitoring Program since that program's establishment by the license amendment in 2019.

As an independent advisory committee, the ACRS continues to engage the NRC staff and others with ASR expertise on this issue as appropriate, with the goal of advising the Commission accordingly.

Today the Committee will hear from the NRC staff, including Region I staff and inspectors, to discuss the most recent staff evaluation of the progression and impacts of the ASR phenomenon at Seabrook.

We will also hear, as Chair Kirchner mentioned, from C-10, a public interest group that has been following this issue very closely for many years. The committee are focused on the overall safety in Seabrook Station and how the ASR situation is being managed.

At this time, the Committee will hear from the staff and others, and it will discuss how to proceed in the future.

This could include the Committee writing a letter report on this topic with any relevant recommendations to the Commission. There is not yet any schedule for such a letter report. The Committee will discuss next steps at the conclusion of this meeting.

I'll ask Member Ballinger, do you have any other opening remarks?

MEMBER BALLINGER: No, that's fine. Thank you.

VICE CHAIR HALNON: With that, I would like to turn over the presentation to the NRC staff, specifically Mel Gray, Branch Chief, Engineering Branch 1, Region I.

Mel?

MR. GRAY: Thank you.

Thank you, Chair Kirchner and members of ACRS for this opportunity to be here today on behalf of the folks who have traveled from Region I Office of the NRC with me, and our partners in the NRR Office that are here today also.

My name is Mel Gray. I'm Branch Chief in the NRC Region I Office in the Division of Operating Reactor Safety. I have oversight of the region-based engineering inspectors, the specialists who go periodically to Seabrook Station and perform ASR-related inspections.

Also here today is my branch chief counterpart, Matt Young, who oversees the resident inspectors. We work very closely together on oversight of specifically the Seabrook Station, which the licensee and owner is NextEra Energy Corporation. Our slides will simply refer to them as NextEra, for your information.

Our presentation will pick up where we left off with ACRS from April 2022. We are looking to be responsive to your requests. And so to that end, we intend to move sufficiently through our oversight, some history of ASR at the Seabrook Station and what the NRC has concluded there.

And then we will move on to our inspection results with a focus on the containment internal structure, trying to be responsive to your lines of questioning and interest at our last meeting.

Next slide, please.

Here with me are the actual folks who do the inspections day to day. Nik Floyd is to my left over here. He joined the NRC in 2010 as a reactor engineer in our Region I Office as part of the Nuclear Safety Professional Development Program right out of school.

Upon completion of that program, he was a reactor inspector in Region I Division of Reactor Safety, conducting a full range of inspections with a focus on in-service inspection and design reviews. In 2018, he was promoted to a senior reactor inspector.

He's also served the Agency at Indian Point in 2020 and 2021, in its last year of operation, as a senior resident inspector. He holds a bachelor's and master's degree in materials science and engineering for Virginia Polytech Institute and State University.

I have next to me Travis Daun. He's the Senior Resident Inspector at Seabrook, there every day. He's been there since June 2023, every workday. Prior to his current role, he served as a resident inspector at Seabrook Station since June 2019, and prior to that at Susquehanna Station in Pennsylvania.

Travis joined the NRC in 2012 as a reactor engineer in our Region III office. He previously served for 11 years in the Navy as a submarine officer.

He has a bachelor's degree in electrical engineering from Marquette University, a master's degree in cybersecurity from Syracuse University, and a master's in business management from American InterContinental University. It's good to have staff who are smarter than you.

Finally, George Thomas is with us. He has been with us on inspections from NRR office on-site about every six months. George Thomas is a senior civil structural engineer and a reactor technical reviewer in the Division of Engineering and External Hazards, Office of Nuclear Reactor Regulation.

He has been with the NRC since 2006, and worked in the US nuclear power industry with Bechtel for over ten years prior to that. He was one of the technical reviewers for the Seabrook License Amendment Request and License Renewal Programs related to ASR and building deformation.

He has also supported Region I, as I've said, since 2011 in our inspections. He holds a PhD degree in civil engineering from Texas Tech University and is a Registered Professional engineer in the State of Maryland.

Next slide, please.

I'll go over our agenda briefly. Considering that maybe some stakeholders here may not be as familiar with ASR and the Seabrook plant and the phenomena, we intend to go through just sprightly quickly the background, and then move through what our inspection and assessment of ASR is.

Travis Daun will provide the background and describe the requirements we've placed on the facility. Then Nick Floyd will pick up with our inspection and assessment results. Finally, he'll focus on the containment internal structure. And I'll close there with thoughts and next steps.

So with that, I will turn it over. Next slide, please, and turn it over to Travis. Thank you.

MR. DAUN: All right. Thanks, Mel.

As Mel mentioned, I am Travis Daun. I'm the Senior Resident Inspector at Seabrook. I'm going to just cover a brief background on ASR, the history of ASR at Seabrook, and then how Seabrook monitors ASR progression and building deformation, as well as the licensing actions taken to date.

First and foremost, resident inspectors must be cognizant of the ASR degradation mechanisms present at the site and how the effects manifest throughout the structures. This gets factored into our daily and quarterly inspection planning for ASR-specific samples or integrated into normal quarterly samples with an ASR flavor.

So a background on ASR. If anyone has additional questions, please feel free to stop me while we're going through them. I'm going to hit the highlights at a fairly high level, but if you need more detail.

First of all, ASR is a slow, expansive chemical reaction in hardened concrete, which occurs in the presence of water. That chemical reaction is between the alkaline cement and the reactive silica found in some aggregates. The expansion can cause various material impacts, which we'll look at on the next slide.

Next slide, please.

Okay. Here are some of the visual indications of ASR that the residents see throughout the plant. The gel expansion starts off as micro-cracking and can later be observed as macro-cracking on the surface.

The visual appearance is typically in the form of what is known as pattern cracking, which you can see on the image to the right, where you see the deposits in the cracks and gel staining around the cracks.

I just want to make a note, ASR is not new. While Seabrook may be the only known nuclear power plant in the United States where ASR is impacting safety-related Category I structures, there is a lot of experience with ASR in other industries, such as Department of Transportation.

The Federal Highway and Transportation Authority has extensive experience with ASR in the transportation infrastructure, but the design is not a one-for-one. Bridges are not designed the same way that nuclear power plants are, and there are significant differences.

The expansion and cracking of concrete from ASR can potentially impact both structural capacities, so the load carrying capacity of the structure, and the demand, i.e. the load put on that structure. So it can affect both sides of that demand-to-capacity ratio on that structure.

VICE CHAIR HALNON: Travis?

MR. DAUN: Go ahead.

VICE CHAIR HALNON: This is Greg Halnon. Since it's not new and it is elsewhere, how does it fail? What does the failure look like when we say, it's done, it's failing?

MR. DAUN: Do you want to take that, Nik?

MR. FLOYD: Yes, I can. Can you hear me okay?

VICE CHAIR HALNON: Yes.

MR. FLOYD: So based on our discussions with our Office of NR Research, there aren't really a lot of data points on what failure looks like because structures are either remediated or taken out of service prior to failure.

A lot of the failures that have been seen are in unreinforced concrete. Essentially, you get expansion, you get the microcracks, and it will fall apart.

It's a lot different when you throw in the reinforcement. Once it's reinforced, it behaves significantly different. Right now, this is why Seabrook went through the test program, just to see, hey, what is the capacity that the concrete can handle?

As far as that ultimate limit state, what we consider unacceptable is, you shall remain under the demand-to-capacity ratio of 1. Beyond that, it's unanalyzed, and that's unacceptable. So what failure actually looks like, we have not seen it.

VICE CHAIR HALNON: Okay. So we haven't seen in these other applications a catastrophic failure, even in unreinforced, because it's either remediated in time because it's visible and noticeable so we know we have to remediate it, or it's reinforced and hasn't gotten to that failure point; is that correct?

MR. FLOYD: That's our understanding. Correct.

VICE CHAIR HALNON: Okay. Thank you.

MR. DAUN: Okay. Next slide, please.

Okay. So here's a brief history of ASR at Seabrook. In 2009, in preparation for NextEra's submission of their license renewal application, it was identified that the aggressiveness of the groundwater chemistry on Seabrook concrete structures that were in contact with groundwater needed to be determined.

Therefore, testing was performed. In August 2010, NextEra confirmed through a petrographic analysis the presence of ASR in concrete and below-grade walls of several Category I structures.

NextEra then initiated a prompt operability determination to assess the safety significance of the issue and the basis for continued operation. NextEra identified several causes for the ASR issue and several reasons for why it was not identified until the license renewal review.

One cause was that concrete mix for initial construction unknowingly utilized an ASR-susceptible aggregate. ASTM standard screening tests at the time have been determined to have limited ability to screen very slow-reactive aggregates for ASR.

Because of this, NextEra mistakenly assumed that the original cement and aggregate selection would preclude ASR development, and therefore did not initially consider that the observed cracking could be caused from ASR. So initially, NextEra characterized the cracking as shrinkage cracking.

Just a note, new ASTM test standards have since been developed -- this is since the early to mid-80s -- that now better detect this type of slow-reacting aggregate.

And then the NRC did issue an information notice in 2011 -- it's 2011-20 -- on the operating experience identified at Seabrook with ASR identification and ASR confirmation.

Next slide, please.

Okay. NextEra performed an extent-of-condition review and prompt operability determination, as I said on the previous slide, and concluded that from a regulatory standpoint, the affected structures were operable, but degraded and non-conforming, because ASR was not initially taken into account in the current licensing basis.

The prompt operability determinations were based on the material properties and the margins that were seen. Regional inspectors and Headquarters experts -- Mel has mentioned a lot of those --

reviewed the operability determinations and concluded that ASR-affected structures remained capable of performing their safety functions.

In 2012, the NRC increased oversight to ensure structures remained functional while NextEra developed their corrective actions.

Next slide, please.

Okay. To resolve the open operability determinations related to ASR, NextEra chose to perform a large-scale test program at Ferguson Structural Engineering Laboratory at the University of Texas at Austin to better understand ASR's impact on the structural performance of ASR-affected structures.

The concrete test specimens were designed to replicate the reinforced concrete walls at Seabrook, and then the ASR growth was accelerated to allow testing in a reasonable time frame.

Basically, the test was designed to determine how much ASR could occur in a concrete structure before impacting its structural strength, and determine the best way to measure and track ASR progression in a structure similar to the walls at Seabrook.

During the test program, the NRC conducted several audits and inspections with NRR and Region I staff at the test facility, which were focused really on ensuring quality test standards were followed during the testing. This was especially important knowing that the test results would be used to update Seabrook's current licensing basis as part of future license amendments.

Next slide, please.

Okay, the large-scale testing program results. They showed no reduction in structural capacity at the expansion levels tested, and the code equations can be used to the tested levels.

The test results were used to develop expansion limits and monitoring techniques which were incorporated into Seabrook's current licensing basis via a license amendment.

As long as Seabrook stays between the identified expansion limits, they can continue to use the original design equations and material properties to determine the capacity of the impacted concrete.

If they go outside these limits, it does not mean that they have a firm safety limit, but they have gone outside the boundaries of the test program and would need to reanalyze the structures and demonstrate operability.

So basically, the Texas test program, as we refer to it, put guardrails on -- this is how much ASR expansion you can see and confirm the structural capacity of the structures.

Based on the licensee's ASR expansion rate evaluations for September 2022 and March 2023, several of the extensometer locations were projected to exceed the licensed through-thickness expansion limit before the expiration of a renewed Seabrook operating license in 2050. Based on the rates at the time, the projected time to exceed was greater than ten years.

The NRC has tools available to ensure appropriate oversight as they continue that monitoring program to ensure that they're still within those guardrails that we discussed.

VICE CHAIR HALNON: Travis?

MR. DAUN: Yes.

VICE CHAIR HALNON: This is Greg again. Relative to the guardrails of the testing, there is obviously a margin on the upper side from what would be completely unacceptable even by an analysis.

Is there a magnitude that you can articulate about how much margin? Is it a lot, a little? Is it tight? Have they had trouble evaluating cracks a little bit bigger than the guardrails, or is there still a lot of margin there?

MR. DAUN: The guardrails are really set for ASR expansion limits.

So I'll probably either kick that -- Nik, do you want to take it, or George, just on the margin from the Texas testing program that's there?

MR. FLOYD: One of the things we look for when we go on-site is always looking at the monitoring data. We want to see what are the current expansion levels.

It's not consistent across all structures. Some structures have what I'll call moderate expansion. Some have little to no expansion. If you looked at a graph, it would look almost flat. So it's hard to put an exact margin number that's universal to the site. It's really dependent on each structure.

There is still substantial margin for these expansion rates or these expansion limits established for the Texas test program. We looked at it in detail back during our November -- well, not November, but our 2023 inspections.

And what we did see is that the expansion trends, if they stayed constant -- you can only draw a line through so many points, and it's seasonal fluctuations. Depending on humidity and temperature, crack sizes open and close just slightly, and so you get a little bit of scatter on the curve.

The best fit was at, I think, four or five locations. They were projected to exceed that upper limit by, basically, ten years at that point, so 2034-2035.

VICE CHAIR HALNON: Okay. And when they exceed that, that's not necessarily saying they're going to fail in 2034 or 2035; it's just that's where your calculation is pushing it to this point?

MR. FLOYD: Correct. Their testing program has analyzed up to that expansion limit. Now, there is certainly margin past that, but to put a number on it, you can't do it without supporting test programs.

VICE CHAIR HALNON: I was looking for more magnitude of --

MR. FLOYD: Yes.

VICE CHAIR HALNON: -- okay or not.

MR. FLOYD: The one positive was through the test program, there was no reduction in capacity up to those limits tested. That tells you there is still more margin and expansion in load demand that that structure can withstand. So that's the positive piece, but the magnitude beyond that, it's hard to say.

VICE CHAIR HALNON: One last follow-up question. Do those calculations or equations take into effect other mechanisms, such as the Davis-Besse's ice expansion?

They would get ice on their containment structure. They had microcracks in the surface and moisture driven into it and, of course, a similar environment during the wintertime, which is just really, really cold.

Does that take into consideration, too, the potential for accelerated other mechanisms? Just having ASR doesn't immune you to other stuff.

MR. FLOYD: Right. I will have to defer to one of our senior civil structural engineers. I'll let George Thomas chime in on this.

I know there's a lot that goes into the load demand side of that equation, but as far as the capacity and what other degradation mechanisms are considered, I'm not sure. We'll have to ask George.

MR. THOMAS: This is George Thomas, a senior civil engineer. So the other mechanisms, they are managed by inspections. You have the maintenance load inspection.

MR. BURKHART: George, this is Larry. Can you pull the microphone closer, please?

MR. THOMAS: So the inspections look at any other mechanisms. If any degradation related to other mechanisms are found, then it's evaluated to make sure that it's still within the current licensing basis.

VICE CHAIR HALNON: So George, can they distinguish between freeze-thaw cracking and ASR cracking? Are they additive or are they complementary? How do you deal with two, three mechanisms when you're doing a operability evaluation that says I'm good out to 2034 if that's only taking into effect ASR and not other mechanisms?

MR. THOMAS: Yes. In that respect, the primary monitoring mechanism is crack indexing at specific locations. There may be cracks due to other mechanisms, too, but they do consider it all due to ASR.

VICE CHAIR HALNON: So you're assuming it's all ASR --

MR. THOMAS: Yes.

VICE CHAIR HALNON: -- which is aggressive? Well, it's not aggressive. It's slow, but comparable to other mechanisms; is that what you're saying?

MR. THOMAS: It's correlated. And if

there's something out of the ordinary, then they will further evaluate the mechanisms.

VICE CHAIR HALNON: Okay. Thank you.

MR. FLOYD: And just to add to that real quick, some of the locations in question that do have these higher expansion rates, the ones specific to this trend, they're in below-grade structures.

So there really aren't a lot of other damage -- freeze-thaw wouldn't be applicable because you're in indoor, underground structures. The driving mechanism there is ASR.

As far as other damage mechanisms, the visual inspections would support that as far as determining is there something else going on. If you have groundwater coming through, is that leaching, is that just groundwater, or is that just ASR deposits?

So there are ways they would have to disposition that, and we would verify, but yes. That is correct, but just to that.

VICE CHAIR HALNON: Thank you.

MR. GRAY: I guess I can't help but add here. Your first question is meeting the limits might not mean the structures are unsafe. You're correct, but as inspectors, we're given a standard to inspect to.

This is the license conditions and the program that we've approved, so we're keenly focused on the limits of that and to engage in a timely manner to make sure they stay within that. If they see they're not, they're taking action. It's their plan to handle that appropriately.

VICE CHAIR HALNON: Thank you, Mel.

MR. GRAY: Yes.

VICE CHAIR HALNON: I appreciate you bringing it back to the reality of the fact that you're not inspecting up to a design-basis limit. You're inspecting so that you can raise a flag and start a process.

MR. GRAY: Right.

VICE CHAIR HALNON: So I get that. I appreciate that. Thank you.

MR. GRAY: And the second piece, as I understand, maintenance rule is we're looking for if the licensee is responsible to do exams to understand the degradation mechanisms, whether they be freeze-thaw, ASR, or something else.

We do look at that regularly. And Nik is all over that. He's right. It's not freeze-thaw down there. Thank you.

MEMBER PALMTAG: This is Scott Palmtag.

I just have a quick question. I'm new to this. Can you tell me a little bit about the Texas tests? Did you accelerate ASR, and then you put it under load-bearing until it failed? Is that how it went? You said there was no load-bearing cracks from ASR?

MR. FLOYD: Right. As far as the test program went, they sourced aggregate that was highly reactive, so both the sand and the coarse aggregate.

And then after they placed the beams, they put them into -- essentially, you could call them large greenhouses. You want to have higher temperatures and a moist environment to help accelerate the ASR.

They just did the limits because they were looking for results. They did that in an accelerated time frame. You could let it go indefinitely and let ASR continue, but they needed results. So they stopped it just shy of the two-and-a-half-year mark so that they could compile the test report.

The testing that was done, they did a combination of, basically, a bending moment, a shear beam test -- well, not shear. And they also looked at shear test. George could probably again add to this since he was the technical reviewer.

As far as the loading, there was code

equations and load limits that they took it up to, and there was no failures observed. As far as did they take it to failure, I'll have to ask George if they actually went beyond their established load values.

MR. GRAY: This is George Thomas. The specimens that were used, they were subject to accelerated expansion. And at some point, they were subject to load tests to failure.

The load tests to failure were also compared. There were also control specimens that they used, which they tested with no ASR.

So the load capacity between control specimens with no ASR and the load tests to failure, there was no reduction in load carrying capacity. It was fine up to the level of expansion that was achieved. The expansions are measured in three directions, two in-plane and through-thickness.

MEMBER PALMTAG: This is a question. This might be more for NextEra instead of NRC, but are there any plans to do more testing?

MR. GRAY: That is a question for NextEra, but we remain cognizant of what they share with us.

Our footnote in red on this slide was our attempt to let the ACRS be aware that we're closely monitoring that situation. They are doing some exploratory aggregate testing and thinking about what would be needed to support continued plant operation if they choose to do that.

So they are laying the groundwork for that, but they're very early and haven't made a decision on that. Whatever they do, we'll follow closely. We would look at any regulatory-related document or process that they identify.

They would certainly need to apply the Code of Federal Regulations' 10 CFR 5059 process. We would look at that closely to determine whether that warranted prior NRC approval. And if that's the case, we would engage. If it was otherwise, we still would look at that closely and decide what to do.

I'm answering from our purview and our thought, what we're thinking about. We would use the flexibilities in both the inspection product line and the oversight product line to do what's needed for safety.

MEMBER PALMTAG: Currently, the way I understand it, there's guardrails in place. And it's possible they could expand those guardrails, but as of right now, we don't know.

MR. GRAY: That's correct. They would need to provide the technical basis for that, and we would scrutinize that.

VICE CHAIR HALNON: All right. Thank you.

MEMBER BALLINGER: This is Ron Ballinger. It's later in your slides, but they've already taken mitigated measures in reinforcing things as part of the process.

Where there was expansion that was going to violate rules, they've buttressed things up and provided reinforcement already. So there is a solution to that part.

VICE CHAIR HALNON: Thank you. That's helpful.

MR. DAUN: Okay. Next slide, please.

So that was the ASR expansion portion. Now, effects of ASR. We'll move into the building deformation.

The previous two slides covered how Seabrook addressed ASR's impact on material properties of concrete or ASR's impact on the capacity of the concrete. The next two discuss building deformation due to ASR and how it impacts the demand or loads on the structures.

There's a few pictures here I'll talk about. During routine walk-downs in 2014 and 2015, the NRC resident inspectors observed degraded seismic and fire seals that appeared to have been caused by differential movement between adjoining concrete buildings.

It was determined that ASR had caused this additional aging effect through cumulative expansion in ASR-affected structures and from expanding structures pushing on adjacent structures.

Many structures at Seabrook are surrounded by concrete backfill, which may also expand and apply additional loads to structures, even if the structures themselves are not experiencing ASR. These effects were not anticipated and were identified as a different consequence of ASR.

The large-scale test program did not address how building deformation would affect the ability for the structures to perform their intended functions. Therefore, NextEra subsequently developed an additional program called the Building Deformation Monitoring Program to manage this effect.

This program includes a methodology for evaluating ASR-affected structures. The methodology was incorporated into NextEra's license amendment request.

Bulk building deformation adds additional loads to impacted structures. You can see in these photos attached equipment. This is equipment spanning two different Category I structures and the different movement between those two structures.

The top right one is flexible conduit couplings that had moved at different rates, or one moved and the other didn't. And then the bottom photo is lines that became crimped because of the different movement between the two structures.

As resident inspectors, these are the types of things that we're out in the field looking for all the time. Any equipment impacts from building deformation, we're keen to looking for those things, knowing where the structures are, looking at how the structures are impacting each other.

Next slide, please.

Okay. NextEra developed a three-stage analysis methodology to address ASR loads and associated building deformation along with the original design loads. The methodology uses field measurements to estimate the ASR loads both in the structure itself and in surrounding concrete backfill, and then applies the load to the structure as if it were a design load.

The measure becomes more detailed in order to better capture the ASR load as the analysis stages progress. And I'll talk about the three stages here in a second.

The analysis identifies quantitative acceptance criteria in the form of monitoring parameters and limits that demonstrate the capacity is greater than or equal to the demand, including the ASR loads. These parameters are monitored against the corresponding limits going forward.

If a structure approaches or exceeds the acceptance criteria, it is entered into the Corrective Action Program and may be re-evaluated with a higher stage analysis, or a structural modification may be implemented to add additional margin.

So the three stages is, Stage 1, you use the original code equations, plus you add an ASR load factor based on those monitoring results.

A Stage 2 evaluation, you do some finite element modeling and then add those original code equations with ASR load.

And then a Stage 3, which is the most accurate, there you do a full finite element model. And then you take additional field measurements to try to better quantify that ASR impact.

Next slide, please.

Here's a methodology life cycle overview for the Structures Monitoring Program or the Building Deformation Program. We already talked about the three stages.

The first thing they do, they determine which stage that structure falls within. Is it a Stage 1, Stage 2, Stage 3 structure? They always start at Stage 1 and do original calculations, and then they'll move on to Stage 2 and Stage 3 as appropriate.

Then a structural analysis and evaluation of the structure is performed. From these analyses, critical areas based on available margin are identified for additional monitoring.

After analyzing the structure, you apply a threshold factor for each structure, which quantifies the remaining margin between the factored loads, which includes the ASR loads, and the capacity of the design acceptance limit.

A set of monitoring parameters with corresponding threshold limits are also determined for each structure. Any deviations are then entered into the Corrective Action Program for disposition and resolution. Then the life cycle starts over again.

This will continue for the life of the structure. And structures can move from a lower stage to a higher stage, but they continue within -- even after remediation, they'll stay within this life cycle and continue to be monitored and evaluated.

Next slide, please.

VICE CHAIR HALNON: Travis, the three phases, I assume you enter into the phase 1 because of an observation or inspection that looks like there could be an effect.

MR. DAUN: All structures enter as Stage 1. And Nik may --

VICE CHAIR HALNON: Okay. So all the structures --

MR. FLOYD: Just a quick correction there. As far as when you kick into this, this is all part of the Building Deformation because this is when you start recalculating the building load demands to include ASR. You don't enter this process really until you visually see ASR or some other indirect observations of ASR in the structure.

I can tell you every single structure at the site, seismic Category I structure, has been evaluated using this process, except for one service water access vault where they determined it does not have ASR.

Every other structure, they've gone

through this process and determined there's visual indications of ASR, or there's cracking that suggests there's backfill expanding and they've had to go through this process.

VICE CHAIR HALNON: Over time, all the structures of concern that have been put in here, how many evaluations do they do? I don't know if you can give a ballpark. In a quarter or a year or whatever, how many evaluations are actually ongoing?

MR. FLOYD: They've completed all the initial evaluations. At this point, the only evaluations they're doing are when they have to do a reanalysis for structures that they've determined were outside the revised licensing basis.

So those are ones that are still in prompt operability determination space. I'm going to go over that in more detail in a subsequent slide.

VICE CHAIR HALNON: Okay. I'm interested in how many are progressing through this. One of the things that hit my mind was we see the headlines.

We see a finding, a non-cited violation, or a minor violation in inspection reports, but we don't know how many is a sample. If it's 100 percent of what they do, then that's a problem. If it's one of 10,000, then that's not a problem. We only see the peaks of it.

So I'm interested in just how many evaluations are done and whether or not the Corrective Action Program is serving where it's supposed to be serving, which is to maintain continuous monitoring and evaluation.

MR. FLOYD: Just to give you a ballpark, in the program itself, it's either 28 or 29 structures that they evaluated as a structure. Some of those are grouped together.

There's certain groupings for manholes. Rather than just say each manhole is an individual structure, they say, hey, there's a set of manholes here, and here's another set of manholes there. So that's one structure, and that's a second structure.

The Diesel Generator Building, albeit two separate structures that share a wall, they evaluate it as a single structure. I've got it on a subsequent slide. It's either 28 or 29. They've completed all those initial evaluations, though.

And so now, what we've been looking at is when they're doing monitoring and they approach their established threshold limits -- those are some of those areas they identified to have low margin before they hit that demand-to-capacity ratio.

Once they hit that, they have to reanalyze and say, okay, are you exceeding it? And if you exceed it, put in new limits. And now you have to reduce some conservatism.

So maybe reduce a load factor here or take some additional things, like crack section properties, load redistribution, but then you're in operability determination space. You're no longer in conformance with this revised licensing basis. And there's six total structures out of that 28 or 29 that we're monitoring currently.

VICE CHAIR HALNON: That are in prompt operability space?

MR. FLOYD: Correct.

VICE CHAIR HALNON: And corrective actions are in place to --

MR. FLOYD: Correct.

VICE CHAIR HALNON: -- get those out eventually?

MR. FLOYD: Yes.

VICE CHAIR HALNON: Thanks.

MR. FLOYD: And the only other correction there is not every structure enters through a Stage 1. The Stage 1 is the most basic. You just take the original design load calculations, and then you just add in an ASR load.

So for some structures, for example, the one we're going to discuss today, the containment internal structure, they skipped past Stage 1. They went straight to a Stage 2.

That's because they wanted to develop the finite element models to more accurately represent and calculate the ASR loads. They weren't sure what the ASR loads were, so they had to go through that progression and process. So they just skipped right past Stage 1.

VICE CHAIR HALNON: Okay. Thanks.

MEMBER BALLINGER: This is Ron Ballinger again. This was a two-unit site at one time. And a lot of that concrete is in Unit 2. It's been partially constructed.

Did anybody take a look and see what's going on over there, just for grins and chuckles, I guess, for information?

MR. GRAY: Having been here the longest, maybe George would know, but I recollect that that was discussed by the licensee, whether they would take sections of material out of the canceled Unit 2.

MEMBER BALLINGER: Yes.

MR. GRAY: It was the weather that was

uncontrolled for that, exposed continuously. And I think there were industrial concerns with access. So I'm just sharing with you.

MEMBER BALLINGER: Okay.

MR. GRAY: It was thought about and not acted on for their part.

MEMBER BALLINGER: Okay.

MR. DAUN: Okay. Next slide, please.

This slide just summarizes licensing actions associated with ASR at Seabrook. As stated earlier, ASR degradation was not previously addressed in the Seabrook licensing basis because irrelevant design codes of record did not consider loads resulting from the effects of ASR.

Therefore, NextEra submitted a license amendment request to incorporate the expansion limits and to get approval of a methodology to analyze structures affected by ASR.

NextEra also updated their license renewal application to include activities to manage the effects of ASR-related aging on structures. Our revisions were based on test results and ASR methodology provided in the license amendment request.

Following an extensive review, the NRC approved the license amendment request in March of

2019. This review included peer reviews from other offices in the NRC, use of expert contractors from a national lab, as well as reviews by the ACRS, and adjudicated by the Atomic Licensing and Safety Board.

Next slide.

Here are several examples of various monitoring parameters used at Seabrook and the tools used to measure them. The picture on the left is an example of a pin -- the top left is a pin-to-pin measurement location, where a licensee tracks in-plane expansion.

The picture on the right shows a seismic gap that could be monitored based on the output of the analysis so they know where the seismic gap initially was and how that changes.

The bottom left is a typical example of an extensometer. I'll note this is just the cover of the extensometer. The extensometer is anchored into the wall. Core bore, anchor the extensometer, and there you can measure through-plane expansion or through-wall expansion.

And then the bottom right is a crack gauge used to monitor a specific crack location.

VICE CHAIR HALNON: If you have a question, Dennis, go ahead.

DR. BLEY: Yes, Dennis Bley. I hadn't thought about the seismic gaps. Have any of those narrowed to the point that they invalidate the existing calculations for seismic and you had to somehow mitigate that?

MR. FLOYD: Yes. There was one location in the CEVA. That's the Containment Enclosure Ventilation Area annulus region.

So between the Containment Enclosure Building and the Primary Containment Building, they have missile shield blocks. They were installed to shield equipment and other things located underneath them, pretty large. They're essentially slabs connected to the CEB.

That gap, because of deformation -- bulk building deformation of the CEB actually resulted in contact in some areas and reduced gaps in others of those missile shield blocks. And the licensee, NextEra, went through corrective actions to restore those gaps, which is essentially chipping away -- in their case, they used a hole saw to help restore that gap.

DR. BLEY: Okay. So it's always been opening up the gaps again rather than putting in some kind of cushioning material between them?

MR. FLOYD: No, not necessarily. It just depends on where you're at in the structure. In some cases, the gaps widen. In the case of the CEB and the Containment Building, the gap had closed. So it just depends on the relative movement of the two structures.

In all cases, those seismic gaps are established as threshold monitoring limits in their Structures Monitoring Program. And they are monitored on a periodic basis. I have looked at that data many times.

DR. BLEY: Okay. Thank you.

MR. FLOYD: Yes.

MR. DAUN: Okay. Next slide, please.

So here is a summary of ASR license conditions. I'm not going to go through each one, but the NRC inspectors verify licensee performance to these conditions. So this is what was added into their operating license.

Several conditions ensure that test program results remain applicable to Seabrook, including the licensee to verify that the actual measured expansion at Seabrook aligns with the predicted expansion based on the developed correlation.

Seabrook will complete an initial study no later than 2025 and a follow-up study ten years later to validate that the data that came out of their Texas test program is still valid.

I'll note NextEra will be taking cores as part of License Condition B, and those will be subject to petrographic examination. I know that they've already started taking those core bores. It's a large number of core bores. They started that a few months ago.

Next slide, please.

Okay, and then NRC inspection and assessment of ASR. NRC resident inspectors perform daily on-site oversight of the plant status. That's reviewing corrective action reports, walk-downs. We go through all the accessible areas of the plant, looking at those building deformation signs and ASR signs which are not as obvious as some of the building deformation.

Inspectors select risk-informed and performance-based samples, including Maintenance Rule operability determinations, modifications when they do retrofits, and then focused Problem Identification and Resolution samples.

Regional specialists at NRR and Research technical staff have performed focused inspections and continue to perform focused inspections. The results have been documented in their inspection report.

Even if there's no finding, we tend to put a lot more information into those inspection reports just for continuity. And then results and oversight plans are discussed with Region I senior managers during the end-of-cycle reviews.

Resident inspectors do provide real-time feedback on potential ASR issues, such as when we're going through condition reports. It's very easy for me to reach out to Nik, get his thoughts on that. And he can send those condition reports all the way up to George at NRR, if needed.

Next slide, please. And I will turn it over to Nik. You're done hearing from me.

MR. FLOYD: Thank you, Travis.

So since we last met with you all in April of 2022, we have conducted five weeks of on-site inspections. These are inspections with a team of NRC inspectors, including myself. Oftentimes -- actually, every time, I also rely on George Thomas at NRR.

We also utilize other technical expertise from other offices. One in particular was from the Office of Research. For those five weeks, we performed 11 total inspection samples. Again, these are inspection samples dedicated specific to ASR.

And while these numbers seem -- maybe they seem low or maybe they seem high, but they don't reflect the total number of hours spent by the resident inspectors at the site. They're really out there looking at this stuff on a daily basis.

And then the same thing with the Corrective Action Program when they're reviewing condition reports. If they see anything that jumps out, they're sharing that information with us.

So even though we're only getting five weeks -- it's approximately every six months, so it's actually a pretty good touch point -- we're still getting direct feedback as far as the licensee's activities at the site.

While I do spend my time at Seabrook reviewing ASR almost exclusively, there are other team inspections that visit the site, two in particular here. One was the Biennial Problem Identification and Resolution Inspection Team. A second one was one of our focused engineering inspections, the Age-Related Degradation.

They're aware of ASR at the site, sometimes through their sampling. In this case, in their sampling during the two team inspections, they did review samples related to ASR. And in one of those cases, they identified a corrective action finding, which I'll discuss in a subsequent slide.

Next slide, please.

So I just want to highlight two things for you. We continue to conduct inspections focused on ASR. I just discussed that on the last slide. Based on those inspections, we have determined that the Seabrook structures remain capable of performing their intended safety functions.

What I mean by that is NextEra, they're monitoring the ASR in the structures, and they're updating those structural calculations as necessary to ensure adequate capacity remains in those structures.

And the second item here is out of those inspections, we did identify three inspection findings. We documented those in our inspection reports. They were all related to ASR.

We did find them to be of very low safety significance. So when you go through the NRC's Significance Determination Process, that would screen out as a green inspection finding.

In each case, NextEra has addressed it in their Corrective Action Program. And I am going to go through those three findings in a subsequent slide.

Two other things I wanted to highlight here is there's two photos down below. The photo at the bottom left, that's our inspection team from March of 2024. No, it does not look that interesting, but sometimes you get out in the site and sometimes you have to review the detailed design calcs.

The other thing I wanted to highlight is there are two staff in there that are there for knowledge transfer purposes. One of the questions that the NRC has received is, how are you going to maintain oversight of ASR at Seabrook? What happens when you move on to your next employment opportunity, whether it's a promotion or otherwise?

Well, this is how we do it. They just joined that Civil and Structural Engineering Branch with George. So we brought them at the site to knowledge transfer, knowledge management, and bring them up to speed on ASR.

The picture at the bottom right, that was from later in the month that year. And that was a Commissioner Crowell visit to the site.

So we do get a series of NRC management that visits the site, including commissioners, and we do tour them around the site. We show them the ASR areas that are affected just to highlight that, yes, this is a damage and degradation mechanism, it's real, and here's what it looks like.

Next slide.

So to follow up on one of the questions from earlier -- I think this came from you, Greg. You were asking about the total number of structures in the program. Here is just a summary of that.

There are 28 total structures. And again, that number can be slightly deceiving because sometimes there are, some could say, two or three structures within one of those numbers. And that's just how the licensee chose to evaluate that.

Out of the 28, though, there are six structures that are currently outside the licensing basis. What I mean by that is that when we look at the licensing basis, that's the original concrete design code equations that have been revised in accordance with that license amendment to include ASR loads. And that also includes all the load factors, plus a margin for additional ASR expansion.

So you can't put a planned expansion of one for ASR. You have to include ten percent, 20 percent, 30 percent. And you have to include margin for that structure to continue to expand because we know ASR is not static.

We have reviewed all six of those prompt operability determinations. Several of those structures, yes, several of them we have re-reviewed.

So they established operability limits. Those are new limits that they increase their margin just slightly. They don't take all the margin away.

And maybe, hey, you can tolerate ten percent more expansion, but they still remained well below the demand-to-capacity ratio of 1 even with that margin reduction, and they reanalyzed because they had continued expansion.

Well, we wanted to go back and see, how are you managing that? Are you actually approaching your ultimate limit, or do you still have substantial margin for expansion? So that's one of the things we've done when we've looked at some of those reanalyses.

We did determine in each one of those cases that they did perform detailed evaluations to confirm that they're able to perform their design functions.

And one of the things I wanted to point out here is specifically in our Third Quarter 2023 Inspection Report, we tried -- well, we didn't try --

we did document the technical approaches used by the NextEra staff for how they demonstrate that continued functionality.

So we wanted to highlight, hey, here are those, because again, this was a question we were asked. Now you're outside your code and design equations, what are those technical approaches?

Well, many of them are consistent with existing engineering consensus standards. Going back to ACI or the American Society of Civil Engineers, yes, they are consistent.

And the last piece of this is when you're in prompt operability determination, how do you continue to monitor that? Well, in many cases, there's supplemental monitoring that's established in conjunction with the existing monitoring.

And they're done at an increased frequency. In this case and many of them, they're every two months. In some cases, it's every one month.

And then the last piece here is we've looked at the evaluation, but what are they going to do with the structure now? Now we're into the corrective action space. So we look at how the licensee is managing that in their Corrective Action

Program to bring them back into conformance with the licensing basis.

And there's really two options there. You either physically modify the structure to strengthen it or you reanalyze it. In some cases, reanalysis is not an option. You have to do a physical modification.

Two structures in particular that have ongoing modifications are the Control and Diesel Generator Building and then the Emergency Feedwater Pumphouse. Those two are planned for physical modification. The other four structures are still up for potential reanalysis.

Next slide, please.

CHAIR KIRCHNER: For the record, could you just enumerate what the other four structures are?

MR. FLOYD: Yes, I can. The other four structures are the Primary Auxiliary Building, the Service Water Cooling Tower, the Containment Enclosure Building, and the Residual Heat Removal Vault.

Next slide.

VICE CHAIR HALNON: We've got a question from one of our consultants.

Steve, do you want to go on?

DR. SCHULTZ: Yes. Thank you.

The question is, you mentioned four of the structures are in consideration for analysis, improvement, and evaluation. What's the time scale of what's being proposed and what will be done there?

In other words, if we meet a year from now, are there still going to be six or will there be -- what will happen going forward? Can you provide some detail about what that means in terms of analysis time frame?

MR. FLOYD: Yes. One of the items of consideration that we've had from our oversight is timeliness of corrective actions.

Defined in our process, or even if you use the licensee's procedure, usually they try to do corrective actions commensurate with safety significance, which is usually on the order of a single operating cycle.

For Seabrook, that would be 18 months. I put long-term here because these are not one-year projects. These are on the order of years.

The last time that was communicated to us when we reviewed Seabrook's projections for the two structures that are undergoing modifications right now, they were projecting out to 2026 for planned completion of the physical modification with the supporting analysis.

Once you complete the modification, you have to ensure that you meet those as-built drawings and it meets your reanalyzed condition for that additional capacity. So for those two structures, again, that was the projected time line. It was 2026.

For the other four, when we look at the expansion data -- we're always trying to risk-inform our inspection process. The licensee is trying to do the same thing. It was one of the items when we were saying, what do your expansion trends look like? Can you support operability or functionality for the duration of your corrective actions?

In the case of the other four structures, the answer to that is yes. I don't have a specific time line for those structures. They're constantly prioritizing what's the next one to come up. They're using a structural contractor for this, so they have to send that out to the contractor. The contractor is also supporting the modifications.

It would be difficult for them to do all six structures at once, so they have to prioritize which structures to do first. So they have the two in progress. They'll probably have a third one coming up.

Essentially, in the future -- we're going to have a meeting soon. And we're going to have to discuss their next time lines because we are moving forward, I would say, the focus from evaluations. That's where we initially had it. And now we're moving forward to looking at the modifications themselves; or if they reanalyze, we'll look at that.

So I don't have a specific time line. That was a very long-winded response to your question, but just kind of wanted to cover it from the start to end there.

DR. SCHULTZ: When you say you're going to get together soon to have another meeting to discuss time lines and so forth, does that tie into the six-month evaluation that is in the license conditions?

MR. GRAY: Let me take that.

MR. FLOYD: I think Mel wants to comment.

MR. GRAY: It does not. This is our internal plans for inspecting at Seabrook, and we are there about every six months by our own choice. We think that's appropriate for the situation right now. There is a license condition, but it wasn't driven by that.

I would add that the focus of our

inspections, as Nik has stated, is that the licensee maintains margin for these six structures that informs the pace of corrective actions. And we expect them to have a technical basis for the timing of their corrective actions.

The two structures that you mentioned have a rather firm 2026 date. The information we've seen on-site is not that much longer than that. It's not many years. It's a few.

So we are looking for the licensee's performance to move all of these structures within their licensing basis, and then just demonstrate continued monitoring and corrective actions prior to getting in POD space. That's where we are looking to transition towards. Our oversight would then beyond the safe modifications that restore and increase margin going forward.

So I think that was some of the underlying thought processes behind the questions. I'll go back to Nik or open it back to ACRS.

DR. SCHULTZ: Thank you. Both the responses are very helpful. Thank you.

VICE CHAIR HALNON: One last follow-up. So there's two ways of fixing things: you modify it to get back within the licensing basis or you change the licensing basis.

If that's the case, will these require license amendments if the reanalysis shows that I don't want to modify, but I can safely expand my limits?

MR. FLOYD: No. There's been no indication to go outside of the current licensing basis.

Reanalysis in some cases -- just to give an example, for the Service Water Cooling Tower, there was a couple areas where it was difficult to get access to monitor the ASR. We're talking very far down. In some cases, it's below water level.

And so they made a conservative assumption on the ASR expansion level. Well, because of that, that drives up the load demand in the structure. If and when they're able to actually take that data, the consideration is you can reduce that load demand and then bring it back into conformance. That's one example.

It's just getting additional monitoring data to support their underlying assumptions in the original load calcs, but in no case have they indicated to go outside or do something different than that modified code equation that includes ASR.

So there's no process in place right now, or at least no actions in place, by the licensee to go outside of that and submit a license amendment action.

VICE CHAIR HALNON: Okay, but it would be subject to your approval if they do go outside?

MR. FLOYD: If they deviated from the methodology and declared victory and said, this structure is operable and in conformance, we would differ there. And that would be a full engagement.

VICE CHAIR HALNON: Thank you.

MR. FLOYD: Yes.

MEMBER BALLINGER: I have a clarifying question. We're talking about margin, load demand, and the like.

These structures are all designed in accordance with a code of some kind, ACI or something like that. I know the ASME Code that if it's designed in accordance with the code, there's already a factor of 3 margin there.

I'm not sure about what it's like for the concrete codes. Is there a built-in margin when you use a code which is not accounted for here, but is present?

MR. FLOYD: Yes, that's correct. For each load, there is a load factor applied. So inherently, there is margin when you add those up.

George, I don't know if you want to add anything to this specific to Seabrook.

MR. THOMAS: Yes. There are design margins incorporated into the coding in terms of load factors, best estimate loads for expansion and capacity reduction factors.

MEMBER BALLINGER: I didn't hear everything, but you're saying that there's code-built-in margin --

MR. THOMAS: Yes.

MEMBER BALLINGER: -- which is always there, which doesn't factor into any of this? It's just there because you've designed in accordance with the code?

MR. THOMAS: Yes.

MEMBER BALLINGER: Thank you.

MR. DAUN: One thing that was brought up a little bit earlier was we were talking about when they would exceed their ASR expansion limits. And I think --

MR. BURKHART: Just for people out there, please do not mute Sandra Walker's mic as that is the mic for this room.

CHAIR KIRCHNER: For the Teams mic.

MR. DAUN: Nik, I don't know if you want to add in more.

Once they hit that expansion limit, they're outside the guardrails. Even if you do modifications to address the building defamation and load-to-capacity ratios, you're still outside the structural bounds of the expansion limits. That was brought up a little bit earlier. I wanted to find a good time to bring it up.

MR. FLOYD: This is a perfect segue to the next slide because my next two are going to cover some of these corrective actions, which are really the physical modifications.

As we've been describing, there's two options. The physical modification -- and I want to correct something that Travis said. It's not remediation. You can't remediate ASR.

Remediate would insinuate that you're removing the ASR and that process for expansion stops. You can't do that unless you completely remove an entire wall or entire floor. You'd have to remove the old concrete and put new concrete in its place.

So really, what's happening is it's a retrofit. You're adding a physical modification to the existing structure. So that's option 1.

I added these three figures here just to kind of show you a schematic representation of what that looks like, what's currently been used at Seabrook Station. And then on the following slide, I'll show you a couple of pictures.

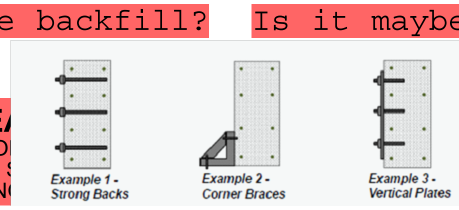
The first one, it's called strong backs. What this is is through-wall reinforcement. You have a through-wall anchor that goes to the back wall of the existing reinforcement, and then you have a tie on the outside. You bolt it down, and that provides your shear or through-wall reinforcement.

The second example is corner braces. It's as simple as it looks. It is a brace that braces the wall to the floor element.

And then the third one is vertical plates. And you can use the vertical plates with the existing strong backs -- that's the easiest way -- or you can do partial through-wall reinforcements to anchor it, but the idea here is you install thick metal plates to resist vertical expansion in the structure.

Why would one choose one method over the other? It just depends on what is the challenging demand on the structure that you're trying to resist.

Is it a bending moment? Is it vertical expansion due to maybe backfill? Is it maybe shear



reinforcement that you need at the bottom of the wall, again, due to the bending moment in the wall? So it just depends on the application.

And then as far as the reanalysis, I gave one example. Maybe it's to verify an assumption. Another case would be, let's say, the structure starts off at Stage 1. It's the simplest way to do it. You take your original design code equations, and you just add ASR load.

Well, what's the next step? Stage 2 or 3, you develop a finite element model and then calculate the ASR loads. And then in Stage 3, you're redefining some of those original design loads.

So now you're actually using in-situ comparisons to formulate those. That takes a lot more effort and a lot more time. And again, that's why you have to prioritize which structure to go after first.

So there are those other options that reanalysis will provide for the licensee. And it is all laid out in the NRC-approved methodology, so it's nothing new that they're doing outside the licensing basis. It's all within that approved methodology.

Next slide.

So real quick, I just wanted to show a couple pictures because these are completed now within the plant. The first one are an example of the vertical plates. That is the Fuel Storage Building.

Now, this was not a building that was in prompt operability determination space. This was a proactive reinforcement. And you might ask, why be proactive? Well, two reasons.

One, it was for -- this is according to the licensee -- this was for proficiency of their maintenance technicians to perform that work. And it's easy to access. This is right on the outside of the building. You can walk right up to it and not have to be around existing operating plant equipment.

You can also see here -- I just wanted to illustrate this. You can see the existing monitoring equipment still in the photo. There's an extensometer just outside the plates and then one right in the bottom left of the plates.

So again, they're not remediating ASR. They're retrofitting it. You still have to continue to monitor for ASR expansion because that's what the program requires.

Just to the right of that is the ASR grid. There's little slots. It's kind of hard to see in this photo, but again, they've retained the ability to continue to monitor the structure.

The figure on the --

MEMBER HARRINGTON: This is Craig Harrington.

MR. FLOYD: Yes.

MEMBER HARRINGTON: Before you move to that, can you explain why the top of those plates is bolted and the bottom seems to be just free?

MR. FLOYD: Yes. I apologize. I should have went into the background on this a little bit. It's actually excavated down about five or six feet. I don't have the exact dimensions.

So where you see the black there, that's been backfilled and then tarmacked on top. They're bolted below-grade into the wall. You're only seeing the top, essentially, the top two-thirds.

MEMBER HARRINGTON: Thanks.

MR. FLOYD: Yes, you're welcome.

And then the picture on the right here -- actually, that's me in the photo with the scrubs and the hard hat -- this is an example of the corner braces. And then you also see the strong backs. So this was, I want to say, reactive, the opposite of proactive.

This was a modification done on the Mechanical Penetration Area to bring that structure back into conformance. So there was a couple of load demands that they needed to resist, and these were the two retrofit options that were chosen by the licensee to be installed there.

That structure, last time we spoke in 2022, was in POD space. It has now been pulled out because it has been restored to conformance in accordance with their licensing basis.

VICE CHAIR HALNON: Are those strong backs in addition to --

MR. FLOYD: Say that again?

VICE CHAIR HALNON: Are those strong backs on the wall in addition to a buttress?

MR. FLOYD: Yes. They are separate.

VICE CHAIR HALNON: Okay, so we shouldn't take those as isolated. These are obviously working together to bring it back into --

MR. FLOYD: They are working in unison. And as you can see -- this is an example -- it's not going to be one or two of the strong backs. There's going to be a number of those as part of that through-wall reinforcement.

MR. DAUN: And then on the opposite wall, you've got even other things. We just didn't include it all.

VICE CHAIR HALNON: You're squeezing it.

MR. FLOYD: Next slide, please.

Based on a combination of inspections -- so this is our ASR-focused inspections, and in one case which I'll point out, it was from our Biennial PI&R Team inspection -- the NRC has identified three findings, all of which were of very low safety significance.

In each one of these cases, they were unique performance issues. They were all related to ASR, but for different reasons for why we documented the findings. This does highlight the fact that continued focus is warranted, continued focus by the licensee on implementing their programs as well as continued focus via oversight by the NRC.

We did review the three findings at the time and made sure that they were entered in the NextEra Corrective Action Program. I'm going to cover the three of these in more details on the next slide, but just wanted to provide you the summary really quick.

Next slide.

I'll start with the picture on the right. This was from our Second Quarter 2022 inspection. This was a failure of NextEra to install extensometers in seven ASR Tier III locations.

What I mean by Tier III -- we briefly covered it in the ASR background, but they developed three tiers for ASR. The third tier is when you reach one millimeter per meter in-plane expansion. At that point, an extensometer is required to be installed to monitor through-wall expansion.

Why that's important is through the test program, what was identified is that in-plane will essentially plateau. And then the remainder of the expansion, because it's only two-way reinforced, will be expansion in the through-wall direction. So it's essential for multiple license conditions to have these extensometers installed in the plant.

What we identified going through the program is that there were seven Tier III locations that were identified by the licensee, and they had not installed extensometers. And it had been that way for a number of years.

When you review the license conditions, in one instance, they require the through-wall expansion to be monitored on a six-month basis. Well, if you go years, not every six months, that's not timely, hence why we wrote the violation.

And I can tell you, to date, all seven

have been installed. We followed up on that on a subsequent ASR inspection, so that finding has been corrected.

Any questions on that finding before I move to the next one? Okay.

DR. SCHULTZ: Sorry, this is Steve Schultz. What was the root cause associated with that?

MR. FLOYD: That was -- and I would be speaking on the licensee's behalf for this, but based on discussions and based on the inspection, they had work orders in place for when areas reached the Tier III threshold to install extensometers. At some of the seven locations, they had a work order in place. Somehow they got lost in translation. I don't know.

That would be a good question for NextEra. I never did get a good answer on how they got left out. Not so much that they got left out, but the fact that they have a proper tracking mechanism in place.

One of the observations I did provide to them at the time is in the Structures Monitoring Program, they have tables which track all their monitoring locations. Well, they had separate tables for the extensometers, and then a second table for their Tier II and Tier III locations.

I provided the observation that it would be useful to have, basically, a cross-reference for the Tier III and extensometer install locations. So they actually did implement that as a corrective action. I feel certain that will help going into the future, but it doesn't answer your question on why this happened in the past.

DR. SCHULTZ: It somewhat identifies the root cause. And what you're saying is that programmatically, they've corrected their approaches so that reoccurrence is not likely?

MR. FLOYD: I would feel confident in yes to that response.

DR. SCHULTZ: Thank you.

VICE CHAIR HALNON: Just a quick time check. It's ten minutes until the end, but we do have some discretion towards the end. A prompt for members to maybe hold your questions to the end unless there's something burning that you need to ask.

MR. FLOYD: Okay. I can speak pretty quick. So if I go too fast, just slow me down.

VICE CHAIR HALNON: Put on your running shoes and go for it.

MR. FLOYD: In the top right, this was a finding identified this year by the Problem

Identification and Resolution Inspection Team. In fact, they contacted me when they identified this. I provided them technical assistance.

What you see here is, this is in their Emergency Feedwater Pumphouse. Circled in blue is the support for the instrumentation tubing. You can see the tube right above the blue circle.

Well, due to fault deformation -- again, here we're getting to why that Deformation Program is important. That's the CEB. That's the Containment Enclosure Building.

That building had shifted or deformed towards the Emergency Feedwater Pumphouse and resulted in contact with that tube support. Well, you can't have two structures contacting one another. You have to retain a seismic gap. So what we did is we -- the team identified that.

The second piece is circled in red. Based on that impact, there are resulting stresses imparted on that square tube support that's welded to the column. You actually see a crack in that tube support piece. It partially cracked down the length there, hence, again, why it's important to monitor these items.

When we did a backwards look at this, we saw in the Structures Monitoring Program they had previously -- they being the licensee -- previously identified it and had not taken corrective action in a timely manner. And this was over the order of years. This was what I would consider a relatively easy action to take to restore conformance.

So we wrote the corrective action violation. It has since been restored. And the inspection team did look at other supports in that room, just seeing was there any other supports contacting the structure.

This is something that does get looked at by the licensee. They actually had documented it in their inspections of the structure as part of the Structures Monitoring Program. They just failed to take corrective actions.

And in the bottom right, that was a finding from our First Quarter 2024 Report. This is specific to the containment internal structure. I'm going to save that discussion for a subsequent slide because I have more details on it, but we will circle back to that.

Next slide.

So here's just a brief time line of the containment internal structure issue. This initially started going back to 2021 during Seabrook's fall refueling outage. The NRC inspectors became aware of cracking and spalling in the reactor cavity pit area of the containment internal structure.

Note this is normally an inaccessible area during operation. It's a high-rad area. Actually, it's a locked high-rad area. It's even higher than that. Simply, it just provides an access point for personnel and serves as a support for some reactor ventilation duct work, as well as it provides a pathway for the in-core instrumentation that goes to the vessel.

I am going to cover some pictures of those conditions that we observed, and that will be on a subsequent slide. I just wanted to highlight here that, yes, we did document a finding on that. That's actually how we got to where we're sitting right now.

The licensee had visual indications of ASR that we considered as possible or likely. We did not see your typical ASR in the concrete itself, but what we did see is that there was potential that what was on the other side of the wall -- again, be it backfill or the containment foundation mat -- something was externally acting or potentially externally acting on the structure that potentially resulted in that cracking.

As a result of that, the licensee should have entered that into their program and evaluated it for ASR. Therefore, we documented the finding.

In response to that, the licensee documented a prompt operability determination. Again, this is in 2021. We reviewed that.

And this was a margin argument where they applied reasonable ASR loads and said, yes, the structure is still able to withhold ASR. That's not the Stage 2 evaluation, though, so it was good for the interim.

Fast forward, licensee completed root cause studies -- I'm going to cover those in a lot more detail in the next slides -- performed root cause studies to best estimate the amount of ASR in the structure. Since you didn't have your typical ASR in the concrete, it was difficult for them to approximate it, so they had to develop finite element models.

They also took a substantial amount of measurements in that area and inputted it into the model to best estimate and understand why they were seeing these structural distresses. They completed that in April 2023, used the results of that root cause study in their official structural evaluations, and completed those in November of 2023.

And then in March of this year, we were able to go on-site and review those root cause studies as well as the completed structural evaluations. As a result, we did identify one finding, which I will discuss.

I know there's going to be questions, but let's just cover a quick overview of the CIS. And then we'll go through each step of that process.

So here's the containment internal structures layout. I realize now it's kind of difficult to read on the screen. I apologize for that.

In blue at the bottom is the reactor pit area. That is the area that's below the negative-30 feet. There's also a four-inch-thick concrete fill mat that begins at elevation 26.

And then the superstructure is above that. That is the slab walls, columns, including the primary and secondary shield walls; basically, everything you see in this picture. What is not part of the containment internal structure is the concrete Containment Building foundation.

That's that 12-foot thick -- it's ten feet thick. It ranges down to six feet underneath the reactor cavity pit area. That's that pink-reddish crosshatch. It does not include that, and it does not include the cylinder or the containment dome. So it's everything else inside of containment.

Next slide.

NextEra approached this issue with multiple steps. They wanted to understand why they were seeing the distresses that they were seeing in the containment internal structures.

It wasn't just in that reactor cavity pit area. What they did see was spalling, cracking in the reactor cavity pit, but there were distresses observed in other areas of the containment internal structure.

If you ever walk around a containment, typically on the fill mat, sometimes you'll see cracks in the fill mat slab. Sometimes you'll see cracks in walls. They wanted to understand what was the cause of all the cracking observed throughout the structure.

So the first step was to collect data through field measurements. That was locating all the cracks, crack locations, collecting temperature, going back and looking at history of the spalling, really just collecting the full data that's going to inform their model.

The second piece of that is they developed finite element models. One was specific for heat transfer because the original design calc just had an assumed design temperature. They wanted to better understand the in-situ loads from operating temperatures. And then the second finite element model was an actual stress model.

As a result of the models, they performed parametric studies. So what I mean by that -- I'll have a graphical picture as well as a summary table -- essentially, they were varying the amount of differential temperature loads throughout the structure, trying to understand.

Could it be 100 percent temperature; could it be no temperature? Could it be 100 percent ASR; could it be no ASR? And then varying the amounts of the two in combination to understand, okay, here's the inputs; what's the results?

And then compare that to the field measurements that they obtained from their walk-downs. Did it make sense that this amount of thermal and ASR load resulted in this cracking or this spalling?

Once they were completed with the parametric studies, they documented the results in root cause reports. There were three specifically. And then they utilized those results as the input to the Stage 2 structural evaluations.

So now, let's go through some of the details here. This is going back to the reactor cavity pit area that caught our attention in 2021.

This is step one, the field observations. This goes through a review of the past monitoring program walk-downs. They also did additional walk-downs in 2021, 2023, and then used field observations for the remainder of the CIS.

Some of the conditions -- and I'll show a couple more in the next slides -- there was cracking, horizontal cracking along the walls. There was spalling on the pit slab, which is the reddened area. And there was also cracking and bulking of some of the other structures. The next slides will be a little bit more helpful.

So again, this goes to 2021, the area that caught our attention. If you look, the left picture is just a schematic. Then there's arrows that point to what was actually observed.

So in 2021, we saw that this area had spalled. Licensee discovered and captured it in the CAP. That's what brought our attention.

This area was previously repaired due to spalling in 2012. That seems pretty unusual for any concrete structure at a nuclear power plant, that repeat spalling, which was just one of the indicators that caught our attention.

The second item here is the steel grating. That was actually buckled -- and you can see it in the picture on the right -- due to what is some type of deformation. There's something going on specific to this area. So those are some of the observations we were seeing.

I do want to note here, this area that spalled, the licensee has installed -- well, one, they repaired the spalled area. They chipped away the old concrete and put new concrete back, but they've installed an encapsulation device on this area, this perforated steel panel that encapsulates it. So if it does respall again, it will catch any of that spalled concrete.

One of the items -- and I'm kind of jumping ahead here -- that was identified in the root cause in addition to thermal loads is that this is a high-stress concentration area. To the right there is a cutaway for the cooling ductwork.

Well, you have a one-inch-thick slab in between two large concrete walls. It produced a very high-stress concentration in this area. So that was one of the items that they identified in the root cause study, hence why they installed the encapsulation device, because there is a chance in the future it could spall again. And that's not good.

Next slide.

Other observations observed was this horizontal cracking along the reactor cavity pit walls. This is typical not just in the area highlighted to the left in green, but throughout the bulk of the structure.

One of the items that we wanted to know is, is ASR actually a contributor or is it stagnant? What's happening? Normally, ASR would result in continued crack changes.

The reason it's important, there have been no apparent changes to date going back to 2017, '21, or '23. Similarly, there was a one-eighth-inch crack in the keyway opening that we documented in our inspection report. That crack has also remained the same, going back to the same time frame.

So we're not seeing changes in those crack profiles, which is good for ASR, but it still needed to be explained on why those cracks were occurring.

Next slide.

So after the compilation of the field

walk-downs and the crack data, the licensee developed finite element models. In particular -- I had mentioned this before -- there was two models developed.

One was for heat transfer, and this was determined in thermal loads. If you have differential thermal, that's going to create a load on the structure, and then that's computed as a stress.

For the global stress model, that utilized all in-situ and design loads, including ASR and thermal. The attempt here was to input different ASR and thermal loads to see what the effects were on the structure.

And not just loads, but the licensee also considered in some localized modeling what are the effects of cycling fatigue, particularly for that slab. As you heat up and cool down, what does that look like for concrete fatigue, the stress concentration in the slab due to the cutout, as well as concrete creep?

So they approached this from multiple angles to really understand what could be affecting and causing those distresses. And then when they ran the model, they compared it back to what they saw.

One of the things I just wanted to bring up was demands were due -- this was a Stage 2 evaluation, so the bulk of the demands came from their original design loads. They were obtained from the original design calculations.

Some demands were unavailable, so they had to calculate those with this model consistent with the original design calcs. Those demands were from ASR expansion, and then for the reactor cavity pit area, they were for thermal. Again, that was the purpose of these two models.

Specific for this review, and this is for the entire containment internal structure, it wasn't just Region I that performed this review. We had assistance from our technical experts from our Office of NRR. That would be George Thomas.

And we also had assistance from Jose Perez, who's a senior technical advisor out of our Office of Research. And then as I pointed out in that picture, we had two observers for knowledge transfer purposes.

So once the finite element models were done, then the licensee transitioned into what they discussed as a parametric study.

Next slide.

And what this parametric study is, this is just a single example. They did parametric studies for each of the root causes that they performed. Above is thermal, below is ASR, and then there's various scenarios for each. I'll quickly walk through this.

The temperature in the reactor cavity pit, as well as the rest of the containment internal structures, that was determined from a steady-state heat transfer analysis.

If you look in scenario 1, that's the design-basis temperature input. That's assuming 120 degrees design temperature. This came straight from the original design temperature calc.

Scenario 2, look at no thermal load. What is the impact on the structure if no thermal loads are considered?

And then scenario 3 is the more accurate representation. This is the operating temperature loads to best estimate what the structure was actually seeing.

There are some assumptions made for this scenario. The licensee has an understanding of what they believe to be the temperature in the reactor annulus, the reactor cavity pit, the walls, and the slab.

This is one of the things that is considered an assumption right now that they're going to have to verify based on installed temperature monitoring equipment that they put in last outage. So during this upcoming fall refueling outage, they will collect the temperature data via data loggers that they installed and verify those assumptions.

And then for ASR, again, there was multiple scenarios performed here. In scenario 1, this is ASR considered just in the Containment Building, so the Containment Building foundation mat.

Scenario 2 is a combination of the building foundation, the reactor pit area, the pit slab. And then 3, it's a variation of just the foundation and the reactor pit.

What's not shown here are the relative amounts of ASR variation. It's looked at anywhere from zero ASR up to high amounts of ASR. And again, the idea here is really just to understand what was the impact on the structure.

Next slide.

So here's a table that summarizes some of the scenarios, the thermal variations, and then whether or not it actually compared to what was observed during the walk-downs. And this is really considered a cause-and-effect table. Does this explain what was observed? I will note this is specific just to the reactor cavity pit area and slab at that elevation negative-44.

In red, this is the comparable representative scenarios for the CIS that were determined to be most applicable. That's scenario 3 with thermal variation echo.

This assumed 0.2 millimeters per meter ASR expansion in the Containment Building foundation. This provided the best estimate and explanation of the spalling that was observed in the slab, as well as the cracking that was observed in the various other areas of the reactor pit walls.

Licensee did run a variation of this where they assumed ASR in the Containment Building foundation as well as the reactor pit area walls, but it didn't correlate as well as ASR in just the Containment Building foundation.

They did, as part of the conclusion of this and the root cause, determine that the distress in that pit area was primarily attributed to thermal loading and cycling fatigue, as well as stress concentrations.

The conclusion was a possible contribution of combined ASR expansion in the foundation of up to 0.2 millimeters per meter in each direction. If you add that up volumetrically, that would be 0.6 millimeters per meter. And this was the upper bound estimate for ASR expansion that they utilized in the Stage 2 structural evaluation.

Now, you might ask, why not higher levels of ASR? Well, they did consider that in their studies. It was one of the questions we had.

It did show, when they used ASR expansions beyond that estimate, it would have produced more severe cracking in those reactor pit walls. Considering the in-situ temperature conditions in the CIS, it was determined to be unlikely.

Next slide.

In addition to the reactor pit area, there were two other parametric studies performed. These were of other distress areas in the CIS.

One of the areas was the fill mat and sump. This was a combination of circumferential and radial surface cracking in the fill mat slab and cracking in the adjacent walls.

They also looked at some indications of the distress adjacent to the personnel elevator. This is specific to the enclosure plate, some slight buckling and cracking of the grout for the supporting structure.

So like the reactor cavity pit area, they did analyze these areas. They performed and utilized the finite element models. In some cases, they utilized localized ASR expansion to study whether or not it was ASR throughout the structure or just ASR occurring locally, and then they compared it to the distress in the plant.

The conclusions were, based on the results of this study, that ASR was likely not the cause. It did not explain the conditions that they were observing.

Particularly for the fill mat slab at elevation 26, they determined the likely cause of that was concrete shrinkage. And then for the personnel elevator, they determined that it was due to a combination of effects potentially, which could have been inefficient anchor bolts, grout shrinkage, vibration, or even imbalanced loads during ASR operation.

In order for ASR -- similar to the reactor cavity pit walls -- to be the cause, you'd have to have very high amounts, which would present prominent signs of ASR cracking in the fill mat slab. There would have been visual indications typical throughout the plant that were just not observed.

Now, the ASR so far determined has been concluded, as best the licensee can estimate, to be in the Containment Building foundation mat and not in the fill mat slab itself, but they have implemented monitoring throughout the containment structure in the fill mat to confirm that conclusion of those areas. I'll cover this on the next slide.

Those areas are outside the secondary shield wall, otherwise called the bioshield wall. They are accessed quarterly while the reactor is operating. And so they are getting quarterly data to monitor that the effects of ASR are not occurring in that structure, specifically the fill mat slab.

One of the things -- and I briefly touched on this -- is that they previously installed data loggers via thermocouples in the reactor pit area to confirm their assumption for the in-situ temperature conditions of that area.

They will collect that data during the upcoming refueling outage and confirm those temperature profiles. And this is also to compare against the root cause analysis to determine their judgment that thermal really was the most significant load demand.

Next slide.

During the NRC review, we did verify that NextEra completed their structural evaluations using the Stage 2 methodology via the NRC-approved methodology document. They developed the finite element models to estimate those ASR loads and recalculated the structural demands using the original design inputs.

We did determine that the CIS meets the evaluation criteria for all factored load combinations with ASR loads. And this does include an ASR threshold of 1.3, so that's 30 percent additional expansion in the Containment Building foundation mat.

I will caution, even though we said they meet all factored load combinations, the one area -- and this was identified in our violation -- was that they didn't include or evaluate the effects for the reactor pit slabs at that elevation in the revised evaluation.

It was excluded from their analysis. It was not taken for any credit in the analysis, but it was excluded from the analysis. And I will go into why, but it's safe to say the remainder of the structure did meet their Stage 2 analysis.

We did review the root cause studies in detail and found they were of appropriate technical details for the licensee to develop those insights as far as what was causing the distresses. It also provided sufficient detail to provide for that upper bound estimate for ASR.

And we did verify that they have plans in place -- and this is what we'll verify in subsequent inspections -- to monitor ASR in that structure to verify whether or not they are experiencing any ASR effects.

Next slide, please.

So circling back, this is the violation that was identified for that reactor pit slab. It is shown in red in the diagram on the right.

What we found is that this slab, it is credited for the attached ductwork. And located directly underneath this slab is the in-core instrumentation that goes to the reactor vessel.

The licensee failed to verify the adequacy of this pit slab in their revised structural calculations. Although not taking credit for it, they still need to evaluate the effects and loads impacted on that structure because if it were to fail, it could impact the equipment located underneath of it.

Right now, it is just a one-foot-thick slab providing a landing access for personnel. Nevertheless, any failure would result in impact to safety equipment.

NextEra, when this was brought up during the inspection, they did perform a prompt operability determination of that reactor pit slab. This was really just to confirm the integrity of the slab during the unusual load combination of a safe-shutdown earthquake to ensure that if an earthquake were to occur, would the slab stay in place. And the answer was yes.

As far as corrective actions going further, they do have an engineering change in progress to reclassify this slab as non-seismic Category I, which will include the necessary analysis to look at the seismic Category II-over-I impacts. Basically, that just means for something that's non-seismic, what failure or what failures could occur to impact safety-related equipment.

That change is in progress. There are some additional items as far as measurements and runs that they want to do, including verification of the temperature in the area, that they need to have prior to completing that engineering change. So this is an action that will be completed during or post this upcoming refueling outage.

And then the last piece here is the slab, right now, if you move it to non-seismic category, you've got to look at the results of any failures. We're talking potential spalling or otherwise impacts of degradation of that structure.

Licensee has an analysis out right now where they're analyzing the potential impacts of the failure of this slab on the underlying equipment. Pending the results of that analysis, if it's no good, they will move forward with a preventative modification, which will consist conceptually of plates installed above the in-core instrumentation to protect it from falling debris. They have not chosen either option to date.

CHAIR KIRCHNER: Is there a time estimate for when that would be effected, the corrective action?

MR. FLOYD: The analysis for the reclassification to non-seismic, that will be performed shortly after the refueling outage.

The analysis for the impact to the equipment from falling debris, I do not have a time frame. I would assume it would go in concert with the non-seismic Category I analysis because it's part of the steps that they have to do.

There are plans to obtain measurements during this refueling outage for the future retrofit. If they were to install that modification, it would not occur during this outage. It would occur in the subsequent outage 18 months from now. Well, when's the outage? October 1st, so 19 months from now.

I would not anticipate any repeat spalling right now. There's been no spalling from this structure. Going back to their calculation, when you look at fatigue, there is substantial margin there, but you still need to protect it in the event. So that's why this is important.

Are there any questions on the containment internal structure? This is my last slide before I wrap it up and turn it over to Mel.

MEMBER BIER: This is really a comment, not a question. I'll keep it brief because I'm not really disagreeing with any of the conclusions, but I just think it makes sense to be careful about using the word cause.

The analysis that was done may be consistent with the hypothesis that such and such was the cause, but hasn't really demonstrated that. There may be interactions between multiple phenomena or whatever that haven't been modeled accurately or completely.

It's a good working hypothesis. It doesn't mean we know it. That was my only observation. Thanks.

MR. FLOYD: Thank you.

MR. GRAY: All right. Thank you. I'll close succinctly.

Our focus has been the capability of structures, that the licensee maintains those. We've done independent inspections which lead us to our conclusions here that the structures are sufficient to perform their safety functions under the most limiting conditions.

We did identify three findings of very low safety significance. And really, the first feeds to the second. They are very low safety significance.

I think one of the questions was asked in getting at, is that a lot in context for what's required in this situation? I guess I would say that this particular plan has a unique challenge that takes a lot of resources and ongoing attention, where they've had to graft into the normal operating practices and procedures structural activity.

And three, in my view, very discrete items doesn't make us change our oversight or think we need to focus more on this licensee than what they're doing right now. That's the best I can say as the regulator.

And that's a good place to be, actually. They're in column 1 of the action matrix. For this issue, they've maintained their position in that column. So that's a good place to be.

Next slide.

Next steps; we're going to continue to use the baseline to focus on ASR-related activities at NextEra and their performance.

And really, the first bullet, I'm looking for them to bring their structures into the licensing basis in the next several years and move on. Monitor your ASR for ASR expansion on the one side and building deformation just to have a handle on that, and take corrective actions in a time frame to keep all of your structures within the licensing basis.

Going back to what was brought up, their conclusions regarding containment internal structures are only a parametric study. They are open to further validation. And we're attuned to that to see what the data shows, to affirm those further or modify them.

Finally, we're focused on licensee performance to show they meet the license conditions. And that feeds into giving us insights on what they may choose to do with their plant that we need to be aware of so we can form our own inspection plans and oversight, whether they choose to perform large-scale testing in the future.

So I would just leave that closed and give it back to the ACRS. I'm here for any questions.

CHAIR KIRCHNER: Thank you, Mel, Travis, Niklas, George. I appreciate the information. We are well behind schedule, but I thought it was important to make sure we had the entire story that you all wanted to tell.

We will take a ten-minute break or thereabouts. We will reconvene at five minutes after 3:00.

We'll give the C-10 folks time to get their computers up and presentation up, and we'll give them their full time. After that, we'll take public comments and then have some time to talk as a Committee.

So at this point, we'll reconvene at five minutes after 3:00 Eastern Time. Thank you.

(Whereupon, the above-entitled matter went off the record at 2:53 p.m. and resumed at 3:04 p.m.)

CHAIR KIRCHNER: Okay, we're online.

VICE CHAIR HALNON: Okay, we're just about at 3:05. If we can get back into session, I'm going to turn the microphones over to Sarah Abramson and Dr. Saouma from C-10.

Sarah, if you're ready to go, you have the floor.

MS. ABRAMSON: Thank you. And what I will do is I'll take about eight or nine minutes to present the first four slides, and then I will stop sharing, and Victor will pick up sharing and present the remaining slides.

VICE CHAIR HALNON: Okay, good. We intend to give you your full time, so please don't worry about the schedule at this point, but certainly take your 30 minutes.

MS. ABRAMSON: Okay, great. Can you hear me okay and see the presentation okay?

VICE CHAIR HALNON: Yes.

MS. ABRAMSON: Excellent. Firstly, I want to thank the NRC and the ACRS for inviting us to present today. It's a really great opportunity to weigh in as the public stakeholder living near the plant and to let you know what our concerns are.

C-10 Research and Education Foundation stands for citizens within the 10-mile radius, which is the evacuation pathway emergency planning zone of the Seabrook Nuclear Plant. We are a citizen-led organization that is apolitical and totally agnostic on energy policy.

We run three programs that support our communities, and that includes real-time radiological monitoring under contract with the Massachusetts State Department of Public Health.

We offer public outreach and education on radiological emergency preparedness, and then our third program is research and advocacy work, which is under this category, in pursuit of public and environmental safety relative to Seabrook Station's ongoing operations, so not dissimilar to the NRC's mission.

I'm Sarah Abramson. I'm Executive Director of C-10. I have an academic background in environmental science and policy. I've worked with federal and state regulators in aviation and transportation in previous professional roles.

I live about eight miles from the plant with my family, and I'm one of about 180,000 people who live inside that 10-mile radius. This ASR issue is of particular concern to C-10, and we have been very lucky to understand this issue better with help from Dr. Victor Saouma, who is an expert in ASR specifically.

We come to this meeting with two overarching questions for the NRC in general and for the ACRS to consider when they look at the ASR issue. Firstly, we're interested in what does the NRC and NextEra plan to do to both expand and preserve their technical understanding of the ASR issue.

Understanding that NextEra's initial analyses of ASR were overly simplistic, as determined by the Atomic Safety Licensing Board in its ruling, we are remiss to see continued examples of the licensee oversimplifying their analyses, such as their failure to properly analyze the reactor cavity pit force lab, and such shortcomings lead to an underestimation of the risks, and thus a public health and safety concern.

We also see just a small group of very smart, but very few NRC employees who are fluent in the ASR issue. They're basically the three gentlemen you just heard from, but realistically they could lead the NRC tomorrow, and we're left wondering what additional internal and external resources can the NRC incorporate into its oversight of the ASR issue to ensure adequate oversight.

Secondly, we have examples from many other inspection reports, both pre-dating and since the last 2022 ACRS meeting on ASR, that show a pattern of violations on the part of NextEra relative to ASR license amendments and license conditions.

That repetition of noncompliance and a pattern of dragging out corrective actions leaves us concerned that the regulatory instruments being used to enforce these ASR license conditions are not compelling enough to prevent this ongoing mismanagement of ASR.

Thus, our question is, what can be done to improve the effectiveness of ASR-related regulation and enforcement? We understand and respect the role of the ACRS as an advisory board, it's the A in the name, and primarily informing the commissioners on what you as scientific experts see as relevant information to regulatory decisions. ASR does not have any specific regulation that is applicable to all commercial nuclear plants.

C-10 did petition for rulemaking for ASR in 2014, but it was denied in 2019. Instead, we're left with only the specific license conditions at

Seabrook Station that are static and relatively unchangeable because they're packaged with their license renewal. This is not dynamic enough to allow the NRC to respond to changes in what becomes known about ASR at the plant, and we believe that more robust regulatory instruments are necessary.

Of some history with C-10 and our interactions with the ACRS. We did attend the last meeting. I didn't, but other C-10 members did. And in that meeting, we learned a lot from the NRC inspector's presentation, and we were really appreciative of ACRS's probing questions.

You zeroed in on a lot of the same things that we had been concerned about.

After reading the transcript, in addition to attending, we formulated a list of pressing questions that we thought were really pertinent to public health and safety, and those questions were submitted by C-10 in June of 2022.

And I've had a lot of interaction with a lot of different people at the NRC trying to get answers to those questions with little success, but I have had the benefit of meeting with NRC inspectors almost quarterly, in addition to a lot of sporadic phone calls with them as questions come up.

So the NRC inspectors have helped fill our knowledge gaps, but it does remain to be seen what the specific responses were from NextEra that the NRC inspectors collected, I believe, to answer those technical questions that were asked by ACRS and C-10 last time around in 2022.

One example on this slide is the seismic considerations and how ASR may impact the probabilistic risk assessment related to seismic activity. We're left wondering where is the visibility for the public about what questions did the NRC inspector, Mr. Newport, then bring back to NextEra? What was NextEra's response? How was that delivered to ACRS? So we're just hoping for a little more transparency on that.

Hopefully it wasn't just an IOU for this next meeting, because two years has passed and there's lots of different faces on the ACRS. Hopefully the information flow is more frequent. We also view our role today to offer robust outside subject matter expertise on ASR.

We are confident in the ACRS and that you take your role seriously to review this information and advise NRC leadership on how to better regulate the ASR problem. Quite frankly, to ensure that while

Seabrook Station is in operation, that someone like me who lives near the plant can feel safe.

With that, I will pass it off to Dr. Victor Saouma, who comes to this topic with a lot of extensive experience with ASR and structural engineering. There's a saying that they wrote the book on it, and in this case, he wrote not just one, but two books on ASR.

So we feel very grateful that Dr. Saouma continues to provide consultation to C-10 on this issue. So I'm going to pass it off to Victor by, I'll stop my sharing, and then hopefully that will enable Victor to pick up his sharing.

VICE CHAIR HALNON: We do see your slides, but you're muted, so if you're speaking, we can't hear you.

DR. SAOUMA: Okay, can you hear me?

VICE CHAIR HALNON: Yes, we hear you now.

DR. SAOUMA: Yeah, my apologies. So my name is Victor Saouma, and I'm a C-10 consultant. It's my pleasure to share with you my thoughts regarding Seabrook. I'd like to walk you through the chronological evolution of ASR's impact on Seabrook's safety.

This exploration will cover some landmark decisions and allow me to highlight key concerns raised by C-10. The stakes are incredibly high, and while my suggestion may sometime differ from those of NextEra, it is because in this case, and for the sake of public safety, science must take precedence over convoluted engineering. It all began with the unexpected and accidental discovery of ASR at Tunnel Bravo in 2009.

This discovery caught everyone at the NRC off guard, as they had not anticipated such a problem occurring in a containment building, despite the existence of hundreds of ASR cases in concrete dams, much more so than in bridges, I may add.

Fast forward to 2016, when NextEra filed a license amendment request, along with a request to modify the timetable for the 10-year mandated air leakage test. My critical review of this request has been published, but today I want to focus on some of the underlying assumptions and justifications put forth by NextEra, which are at best questionable.

Therefore, I will address two critical aspects; the operational basis earthquake OBE and the safe shutdown earthquake SSE. These are not merely technical details, they are foundational to the safety of the entire facility. NextEra's assertion that this parameter did not require modification, despite the presence of ASR, is not only wrong, but potentially dangerous. Such claims demand scrutiny and must be supported by irrefutable evidence.

As a structural engineer, I look at safety in terms of demand and capacity. For a structure to be deemed safe, its capacity must unquestionably exceed the demand. The numerical modeling, specifically the finite element analysis employed by NextEra, is grossly inadequate and sophomoric.

This analysis is supposed to be the cornerstone of structural safety evaluation, yet what has been used lacks credibility to anyone with even a basic understanding of modern finite element analysis.

Keep in mind that we are not bound to 1930 technology for this. I will add emphatically that the absence of review by an independent panel of external experts further erodes confidence in this finding.

NextEra's second filing also contains an argument that the potential for leakage should not be of concern. This position is quite frankly baffling. Micro cracks are known consequences of ASR, and these cracks compromise the integrity of the structure, allowing for leakage that could have serious implications.

The idea that reinforcement alone, given that it is three-dimensional only around the base, could mitigate this risk is dangerously optimistic. It is essential to understand that ASR is not just a surface-level issue. It is primarily internal where the moisture is highest.

For ASR to progress, we have to have moisture, and moisture is highest inside the walls. By the time we observe surface cracks, we are simply looking at the tip of the iceberg. And let me repeat it. By the time we see surface cracks, we are simply looking at the tip of the iceberg. My stance on NextEra's understanding of ASR has been clear and was largely validated by the Atomic Safety Licensing Board 2019 ruling.

It's worth noting that 50 percent of the license conditions imposed by the ASLB were the direct result of C-10 intervention. So when I presented the case on behalf of C-10, it was evident that NextEra's grasp of ASR implications was at best superficial. The board concern mirrored my own, highlighting significant gap in NextEra's approach.

This ruling is a critical acknowledgement of the complexities involved and the need for a more rigorous approach to ASR management.

Chronologically speaking, the NRC understanding and appreciation of ASR complexity have significantly improved since 2016. Indeed, the NRC has funded three major projects at Northwestern, my university, University of Colorado in Boulder, and at NIST. Yet, indications are that the NRC has ignored some of the major findings that two of them determined.

That is the negative impact of ASR on the shear strength of concrete, which is opposite to what was determined in the Texas test. I will develop this in a future report. This progression is crucial, yet it coincides with the troubling spread of ASR beyond the tunnel when it all started. Now to the containment enclosure building itself.

A development I personally observed during my visit. It was clear by then when I visited Seabrook, we had crack all around the base of the CEB. By now, as here, we hear that the projected expansion is likely to exceed the limit set for the license. And yet, we continue to hear that the expansion is slow, a term that does not mean anything unless it is substantiated and based on physical tests.

So, C-10 strongly advocates also for the reassessment of the OBE and SSE parameters. This critical factor that determines the facility's ability to withstand seismic events.

In my professional opinion, the mathematical modeling of ASR, whether in static or dynamic analysis, is fundamentally flawed. Obsolete models are being used for dynamic analysis. And the modeling of ASR is not just flawed, but dangerously so.

Furthermore, the computer programs used have not been, to the rest of my knowledge, validated independently or in compliance with Regulatory Guide 1.168, which is Verification, Validation, Review, and Audits for Digital Computer Software used in safety systems of nuclear power plants.

It is also worth noting that to the best of my understanding, these critical components for the safety assessment have not been subjected to independent peer review by external experts. The assumption that there is no reduction in shear strength due to ASR is not just a misconception, it is a potentially catastrophic error.

Multiple researchers, including two funded by the NRC, have debunked this myth, providing ample evidence that ASR does indeed reduce shear strength. Let's not forget that shear strength is what we need to provide the resistance to seismic load, lateral seismic load. Not only was the wrong large-scale test conducted in Texas, but the specimen had cracked and was damaged even before the shear tests were conducted.

Ignoring this evidence puts the entire structure at risk. So we must insist on a comprehensive re-evaluation of all the safety parameters considering this reality. The safety of Seabrook facilities depends on it.

Revisiting the capacity and demand paradigm, this figure, before you synthesize the case for thorough revision of the parameter, we can follow the sequence of events. First, there is a deterioration of the concrete with time. This has been observed in Seabrook. This would result in a decrease in the concrete capacity.

Moving to the next curve, now going to the vulnerability curve, which is this one, and given that the capacity and demand must be at least equal, we will have a decrease in the earthquake magnitude that can be sustained.

So if you have a decrease in the magnitude of the earthquake which can be sustained, for illustrative purposes, let's say from 0.7 to 0.55 G, then going to the hazard curve, it means that such a decrease in earthquake magnitude will result in an increase in the return period. So all of the above says unequivocally that Seabrook is more vulnerable to earthquake than we are led to believe.

I will turn now my attention to the air leakage test issue. I have previously mentioned the prevalence of internal and undetected crack as mentioned by the FHWA.

I have also mentioned that above the base, about five feet, we no longer have shear reinforcement. Those are present only close to the surface. So once again, with shear reinforcement only close to the base, three-dimensional reinforcement, but as we move above at a certain elevation, we only have skin reinforcement.

Furthermore, the Atomic Safety Licensing Board has implicitly criticized NextEra for its poor and questionable understanding of ASR's internal microcracking. Hence, considering this consideration, C-10 recommends shortening the air tightness test schedule from the current 15-year interval to a performance-based schedule.

This recommendation is not made lightly. It is based on the recognition that ASR is an evolving threat that cannot be adequately monitored under the existing schedule. Now the key question here is how performance is defined and who gets to define it. This must be a transparent process involving independent experts.

The definition of performance must prioritize safety above all else. I now turn to a related but equally important concern. C-10 advocates here for public rights to know. The safety of the public is not just a matter of engineering. It's a matter of democratic transparency.

In an open society, there should be no confidential or proprietary data when it comes to public safety. All relevant information about ASR and its implications should be made public. This is not just a moral imperative. It's a practical necessity.

Hence, we assert that through measurement data should be treated as public domain. Yet, we recognize that NextEra modeling should remain proprietary. The comparison may not be perfect, but it is similar to a public utility sharing water quality measurement results with the public.

I was specifically referring to crack indices recorded by NextEra which have not been adequately disclosed to the public. These indices are critical for understanding the extent of ASR damage. Yet, they remain shrouded in secrecy. This lack of transparency is unacceptable.

C-10 is specifically asking that the NRC makes public the date, tier level, crack index value, location, and the reference value of the core compressive strength closer to the location cast to reconstruction. This last indicator is particularly crucial because it is essential to determine the out-of-plane ASR expansion using the method advocated by NextEra once the crack index reaches critical values.

Without this data, any assessment of ASR's impact is incomplete and potentially misleading. The public has the right to access this information, and the NRC has the duty to provide it. But having access to data is only the first step.

We must also be able to visualize this data in a meaningful way that allows us to quantify its impact on the overall health of the structure. Raw data is not enough. It must be analyzed and presented in a way that reveals the true extent of the risk.

This is not just about number, it's about understanding what those numbers mean for the safety of the people within 10 miles radius of Seabrook. C-10 is already fully prepared to collaborate with the NRC in this endeavor. We found that NextEra tool, unless it has changed since then, the one on the upper left, is inadequate, misleading, and confusing, and in the end practically useless.

So it's not enough to criticize, we must be proactive. And proactive we have been at C-10 by developing advanced software that can display the spatial and temporal variation of ASR or any other measurement in a containment building, such as shown in here.

In this figure, it's easy to visualize the extent of ASR evolution in structure, and the temporal variation and this are shown in the lower two slides. Of course this is for illustrative purposes. The one on the left is at 21. The one at times on the right is at time 22.

Finally, one can hover the mouse over any point and get the relevant data, or clip out all the region with an ASR expected to be below a certain threshold, or simply slice the structure to look into the internal expansion. But as I said, this is hard to visualize, we have to be able to have some sort of an analysis approach, that is to take to determine the health of the whole structure.

So at any given time, based on this data, we can take a snapshot and plot histogram of ASR expansion. These are the number of locations of the surface, which has an expansion below a certain value, and again each one of those bars corresponds to a certain range. So the total area under this histogram serves as a global measurement of the structural health, S times T. This is step 3 and index is 11.3.

Finally, we can plot the time evolution of this index, providing decision makers, such as the NRC, with the data they need to make informed choices. This approach also reassures the public about the structural integrity of the facility that provides the electricity.

So these last two slides are the references substantiating my assertion, along with hyperlink to key documents. C-10 will always welcome questions and provide further scientific evidence of our assertion.

I would like to conclude by saying that this is a particularly complex, very complex problem, unprecedented and not covered by any existing codes. When I heard mention about ACI code and ASCE, none of this code addresses ASR. Once again, none of the existing code covers ASR. We have tweaked this code to, forgive me for the term, pretend that we can model ASR through a simplistic approach.

Therefore, we must first rely on sound, science-based principles. I repeat, science-based principles, and then and only then develop engineering solutions that are directly informed by the scientific foundation.

Thank you for attention, and of course, I'd be happy to answer any question.

VICE CHAIR HALNON: Thank you, Dr. Saouma. I appreciate that. And I'll look to the members and our consultants if there's any questions for either Sarah or Dr. Saouma.

MEMBER MARTIN: Dr. Saouma, this is Bob Martin. And I'm new to this topic, I must admit. But I have to believe that there is some, not necessarily nuclear industry experience, but in structures like bridges and overpasses where this has come up.

And that maybe, in fact, they would lead as far as understanding not only the phenomena, but how to address it in a public safety sense. Can you provide a little insight on what others do on this subject?

DR. SAOUMA: Yes. ASR was first

discovered in the 1940s in the context of dams, concrete dams. We have had hundreds of dams worldwide which have been suffering from ASR. And their safety has been addressed by different means. And by the way, some of them have been decommissioned because it was too dangerous.

I recently was involved in a major project with the Bureau of Reclamation to assess the safety of seminal dam which suffered ASR in Wyoming. So, again, there is what I would call the traditional approach and the more modern approach which has been espoused by few, mostly in Canada and in Europe, but very few in the U.S.

So, for instance, we have to perform accelerated expansion test which NextEra has absolutely refused to undertake. Accelerated expansion test is when we take a core, we put it in a certain condition, heat it, and monitor the expansion and try to determine how much is going to be the future expansion. And that's an approach that, to the best of my understanding, NextEra refused to do.

We can have also to perform certain specific microscopic tests, and that doesn't mean a regular petrographic test that is commonly done, but for instance what is done in Japan to determine what was the past expansion. So, in the area of concrete bridges and dams, the modern approach is to use a numerical modeling where, first of all, the computer program has to be validated.

It has to be validated and to prove that it can capture some of the unique features of ASR; that is, three-dimensional expansion, expansion under constraint condition, expansion which depends on temperature, and on moisture. I would say that in Canada, for instance, Hydro-Quebec has done a marvelous study for Gentilly-2 on addressing a modeling ASR.

So, I don't know if I was able to properly answer your question here, but the idea of modeling ASR as a temperature load and as a volumetric strain is completely wrong, and erroneous and debunked.

MEMBER MARTIN: It's a pretty general statement, not to say it's wrong or right. So, if I'm interpreting what you said correctly, you're saying that generally the maturity of modeling that would be otherwise used to evaluate structures is just immature across the board? Are you saying that there are proven methods that, you know, --

DR. SAOUMA: It is mature.

MEMBER MARTIN: -- are preferable that are there today that have been validated?

DR. SAOUMA: Sorry to interrupt you. It is mature. It has been validated, and some people who have been using it. I give the example of -- for the reactor, I give the example of Gentilly-2 in Canada. I give Hydro-Quebec, which is also for other dams, have been using this modern method.

I can give multiple examples in Europe where the approaches are essentially variation on the team. You have to have a finite element model which can capture the unique feature of ASR, and this has to be validated. No doubt about that.

MEMBER MARTIN: Appreciate that. Thanks.

VICE CHAIR HALNON: Others? I just had one clarification. This is Greg. I got confused on the air test issue. It was my understanding from my previous experience that the air test was for a vacancy in the containment vessel. Can you explain?

Maybe I'm just got a different model in my head about what this containment looks like. Is the pressure vessel metal, and then the concrete that you're concerned about, is a shield building around that metal, or is it the actual pressure vessel?

DR. SAOUMA: We know that there's a liner.

We don't know whether at some point the anchorage has been affected by ASR, and we might lose tightness. We know that ASR is going to cause micro-cracks.

We know that the ASR is going to manifest primarily inside the wall because this is where the moisture is highest. In many cases, we do not see yet manifestation on the surface, yet we have internal micro-cracks. Those micro-cracks can coalesce and eventually cause potential macro-cracks, which would cause leakage.

VICE CHAIR HALNON: Through the vessel itself through the metal liner?

DR. SAOUMA: Of course.

The metal liner, if we assume that it's 100 percent perfect, no, there won't be leakage. But do we know what is the impact of ASR on the anchorage of the liner, where we have hundreds of those? I don't know.

VICE CHAIR HALNON: I understand now. You're talking about the anchorage from the vessel to the concrete itself.

DR. SAOUMA: For instance, related to that question about micro-cracks, when I see the slide number 19 by NRC, where they placed a plate to mitigate the effect of ASR, well, I don't know if it's time for me to make a comment on that, but let's not forget that what they are doing is mitigating on the surface, and then simply throwing up the expansion to be through the wall, because ASR is a volumetric expansion.

You constrain it in one plane, it's going to go back and expand in the out-of-plane. Expand out of the out-of-plane. You already have micro-cracks. What happens? You have internal delamination. And internal delamination is a brittle failure, and you'll get exactly what happened at Crystal River, all of a sudden.

VICE CHAIR HALNON: I understand about that.

Another question. Dave Lockbaum, are you representing C-10? Are you part of that organization? Is that why you raised your hand?

MR. LOCHBAUM: I'm an advisor for C-10, but my question was as a member of the public.

VICE CHAIR HALNON: Okay, we haven't opened up for public comment yet, so hang in there.

MR. LOCHBAUM: Okay, I will. Thank you.

VICE CHAIR HALNON: Any other questions from members or consultants?

Okay, then at this point, I will open it up for public comments. I think Mr. Lockbaum probably has the first hand up, and others, if you raise your hand and or get my attention somehow by unmuting, I will acknowledge it. Please state your name and affiliation as appropriate. Dave, go ahead.

MR. LOCHBAUM: Yeah, this is Dave Lockbaum. I appreciate that. On the issue of the ASR effect on containment integrity and the need for air tests, the slide that the NRC showed of ASR expansion causing a metal grid to buckle, leads me to question whether containment penetrations would be equally vulnerable to ASR expansion causing the penetrate, because it's not a leak right there.

So my concern would be about containment penetrations failing due to ASR expansion in that direction, and the less frequent air tests not picking up on those containment penetration failures. That's all. Thank you.

VICE CHAIR HALNON: Yeah, I understand. Thanks, Dave.

Any other members of the public would like to make a comment? Okay, not hearing any, we will close the public comment session, and now we're into our committee discussion. Any members or consultants have a comment or a question or anything you'd like to say during the deliberations?

DR. SAOUMA: Well, if I may ask a question?

VICE CHAIR HALNON: Yes.

DR. SAOUMA: Victor Saouma.

VICE CHAIR HALNON: Yeah, go ahead, please.

DR. SAOUMA: Yes, again, my question goes back to the slide I've seen, number 19, I believe, or 20 -- 21, sorry. And I'm very concerned by what I saw, because what was presented was an external plate reinforcement aimed at mitigating ASR expansion.

In my opinion, and so to use a term, it's nothing else than a band-aid approach, conceptually dangerous, because if you manage to limit the in-plane expansion, ASR being a volumetric phenomenon, it's going to be redirected out of plane and through the thickness, and this can eventually lead to internal and sudden brittle delamination crack. So, I would urge the NRC to look very carefully at this remedy.

VICE CHAIR HALNON: I see that they're acknowledging you and they're writing it down, so, yes, thank you.

Sarah, I see you raised your hand. I want to give you time, go ahead.

MS. ABRAMSON: Thank you. I had a question that came up during the NRC's presentation. It was around slide nine, and a member of the ACRS asked Nik Floyd, or made a comment that the structural modifications are the solution to those five structures that are going to be outside the unacceptable zone, and I was just hoping that maybe Nik could clarify on that.

Are there structures that, even with physical modification, are on track to be outside the acceptable zone in that 10-year-ish time frame?

VICE CHAIR HALNON: Nik, did you want to clarify?

MR. FLOYD: This is Nik Floyd. I don't have the structures as far as the expansion data in front of me. I can't tell you the specific ones. I know at least one of the areas, to the best of my mind, was in the controller diesel generator building. That is one of the structures on track to be modified currently.

As far as the other structures that are in POD space, I don't have that data. I'd have to get back to you on that, Sarah.

MS. ABRAMSON: Thanks, Nik. And I guess

the core of my question is, just to clarify my own understanding, when you talked about that 10-year time frame, and next you're considering doing another large-scale testing program, what would be the trigger for that? Is that when structures that are already modified are outside the zone?

VICE CHAIR HALNON: Talking about the licensee's retesting program? Is that what you're talking about?

MS. ABRAMSON: Yes.

MR. GRAY: Sarah, Mel Gray. I think your first question was regarding retrofits. They are intended to bring these structures into conformance with the licensing basis in those 10 years. So, that's their intended purpose. Is that responsive? What was the second part of that?

MS. ABRAMSON: Yeah, I might have to get some additional clarification offline from --

MR. GRAY: Any additional large-scale testing by the licensee would be in their estimate. Their current licensing basis only allows for -- it shows that the large-scale testing they did is only valid to a certain level of volumetric expansion.

They would need to elect to do more testing if they felt that that expansion was occurring that would require action on their part for further operation of the plant. That's their decision to make.

MS. ABRAMSON: Thank you. I think that was the clarity I was looking for because I thought I heard someone jump in and assert that the structural modifications were sort of the cure to that expansion being beyond what the original testing program showed. So, I think I understand better now. Thank you.

VICE CHAIR HALNON: Thank you, Sarah.

Dr. Saouma:?

DR. SAOUMA: Yes, I'm curious about the course extracted from the Seabrook.

PARTICIPANT: I mistakenly muted Victor. I'm sorry about that. I meant to lower his hand.

So, Victor, can you unmute yourself and restart?

DR. SAOUMA: Sorry, I forgot. Can you hear me now?

VICE CHAIR HALNON: Yes.

DR. SAOUMA: Yes, I'm curious about those locations where NextEra has installed the through-thickness extensometer. So, presumably, of course, core had to be removed. And according to the protocol that it has advocated, their first step was to determine the past expansion by determining the Young's modulus of that core and compare it with the Young's modulus of concrete cast in the 70s.

How many of those correlations have you made? How many attempts to determine the past expansion have been made so far?

MR. FLOYD: Nik Floyd. Victor, I don't have an exact number for you. There has been a numerous amount of cores extracted from Seabrook. For each extensometer that has been removed for the program, they had to perform testing on that core to establish the through-thickness expansion to date and any cores going into the future.

Specifically, for those seven tier three areas I mentioned earlier where they need to install extensometers, they removed cores for those as well. So, there will be additional cores removed.

DR. SAOUMA: I'm sorry, you're not answering my question. Once you remove the core, you are supposed to measure the Young's modulus of that core and compare it with the Young's modulus of the concrete when it was cast. The difference in the elastic modulus would give you the ASR expansion according to the curve which was determined by the University of Texas.

Have you performed any of those determinations? This is in the license agreement, the new one, that you extract the core, you measure the compressive, you measure the elastic modulus, you compare it with the elastic modulus of the concrete when it was cast.

The decrease in elastic modulus is on the x-axis. On the y-axis, you get the past ASR expansion. Again, that's in the license amendment request. Have you done any of that?

VICE CHAIR HALNON: Ron, would you take this and go back and look at your data and provide some information that may be able to be publicly released at that point.

Thank you, Dr. Saouma. They're going to work on that to make sure you get the data. I just didn't want to get an incomplete answer on the record.

DR. SAOUMA: Thank you.

VICE CHAIR HALNON: So, okay, I'm going to close. No more questions. We're in committee deliberation, but I wanted to since additional information was put out, make sure that there was still no public comment.

So, I'm just going to quickly open up for other public comments. Okay, given none, in fact, the committee is now in deliberation, so everyone will stay muted other than the committee members and consultants. Is there any other comments?

Yes, Scott.

MEMBER PALMTAG: Do you want the court reporter to continue?

VICE CHAIR HALNON: No, I guess that's it. Thank you.

Eric, you're released for the rest of the day. I think for the rest of the meeting, isn't it? Well, did you want to make that statement?

CHAIR KIRCHNER: Yes, thank you, Eric. I will not need your services tomorrow for our planning and procedures meeting so I think we're done with your recording for this meeting. Thank you very much.

(Whereupon, the above-entitled matter went off the record at 3:48 p.m.)

C E R T I F I C A T E

This is to certify that the foregoing transcript

In the matter of: 718th ACRS Meeting

Before: NRC

Date: 09-04-24

Place: Rockville, Maryland

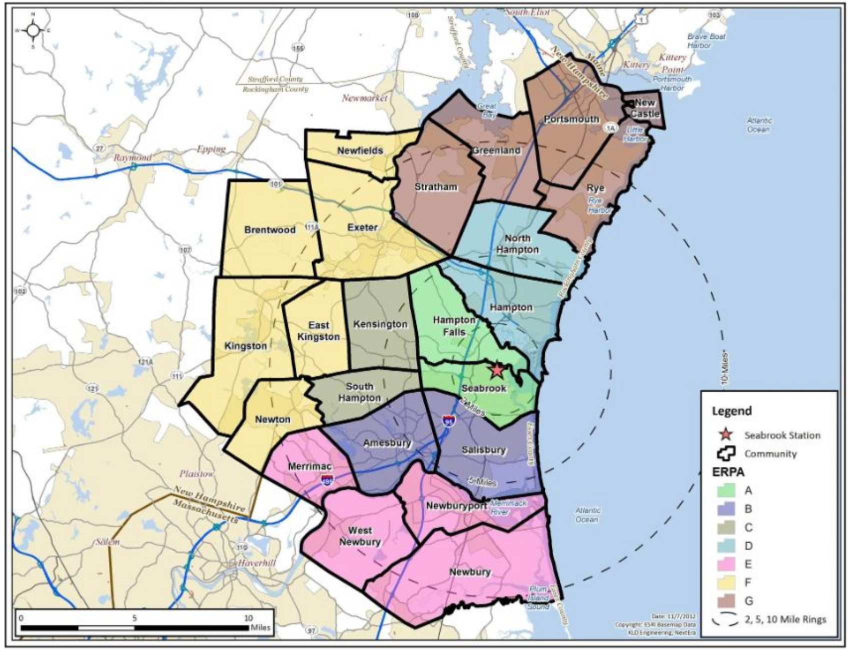
was duly recorded and accurately transcribed under my direction; further, that said transcript is a true and accurate complete record of the proceedings.

Neal R Gross

Court Reporter

C-10 RESEARCH & EDUCATION FOUNDATION

Protecting people and the environment across the communities surrounding Seabrook Station



Presentation to:
US Nuclear Regulatory Commission (NRC)
Advisory Committee on Reactor Safeguards (ACRS)

Presented by:

- Sarah Abramson: C-10 Executive Director
- Victor E. Saouma: Professor Emeritus, Department of Civil Engineering, University of Colorado-Boulder

Representing the 180,000 Citizens within the 10-mile Radius of the Seabrook Nuclear Power Plant.

We Have Two Overarching Public Safety Questions

Learn from the past, prepare for the future, live in the present.
- Thomas S. Monson

1. What does NRC & NextEra plan to do to improve their understanding of the ASR issue?

2. Will NRC continue to find acceptable NextEra's late and incomplete analyses of ASR-impacted areas, especially those in the containment structure?

C-10 Views ACRS Role as Critical in Addressing NextEra's Mismanagement of ASR

- **April 27, 2022:** Last ACRS meeting on ASR
- **June 29, 2022:** C-10 reviews meeting transcript by ACRS members (e.g. seismic PRA), C-10 submits own list of questions to ACRS
- **As of Today, Sept. 4, 2024:** No public record of NRC responses to ACRS or C-10 questions from 2022 meeting
- **C-10 Goals Today:** Offer robust subject-matter expertise to inform on the ASR issue; confident in ACRS to review & advise NRC Commissioners and staff on how to better regulate the ASR problem at Seabrook Station

MEMBER BLEY: Just for reference, the design earthquake is never a big contributor to risk, because we design that out. It's the earthquakes that go beyond the design earthquakes that lead to significant damage. And it's kind of disappointing, do we know for sure that Seabrook has not included that in their Seismic PRA?

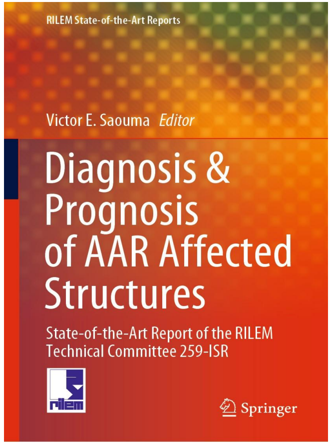
MR. NEWPORT: Let me -- we can verify that, and we can get that back to you for sure.

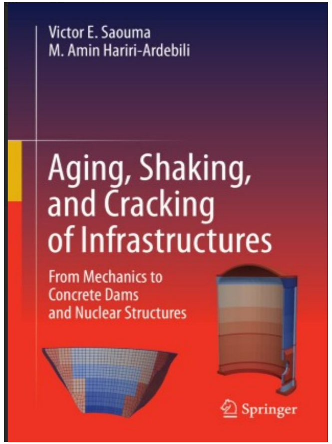
MEMBER BLEY: Okay, it's kind of disappointing that they haven't looked at that,

Excerpt from 2022, April 27 meeting transcript ML22136A319). NRC Advisory Committee on Reactor Safeguards Fuels, Materials, and Structures and Plant Operations, Radiation Protection, and Fire Protection Joint Subcommittees.

Victor E. Saouma, Professor Emeritus, Dept of Civil Engineering, University of Colorado/Boulder

- Conducted theoretical, experimental, numerical, engineering research on ASR (11 major funded projects, two books, 9 major reports, 9 short courses, 15 peer-reviewed publications.
- Past President and Fellow of the International Association of Fracture Mechanics for Concrete and Concrete Structures.
- Past Chair of an International Committee focusing on the diagnosis and prognosis of structures affected by ASR
- Research on ASR funded by the ORNL, NRC, Burau of Reclamation, Tokyo Electric Power Company.
- Past member of the Materials Aging and Degradation (MAaD) External Review Committee (ORNL, Light Water Reactor Sustainability R&D Program) and of the Expanded Proactive Materials Degradation Analysis Expert Panel (PMDA) for concrete in nuclear reactors; Nuclear Regulatory Commission.
- Published extensively: https://ceae.colorado.edu/~saouma/index.php/alkali-aggregate-reactions/





1. Chronology

2. Q1: Revisit Operating Basis Earthquake (OBE) & Safe Shutdown Earthquake (SSE); Correct Analysis Model

3. Q2: Air Leakage Test; Revisit Testing Frequency

4. Q3: Crack Indices (CI), Public Right to Know

Chronology

2009 ASR discovered in Tunnel (Bravo-1) at Seabrook.
2010 Seabrook placed under special NRC oversight.
2012 Nuclear Energy Institute suggests an (up to) 15 years intervals (in lieu of 10) for type A performance leakage rate tests of CBE.
2016 NextEra files a License Amendment Request (LAR)16-03. Regarding seismic analysis, we note the following:

Earthquake levels: No change of OBE & SSE

[W]hen ASR loads are amplified by a threshold factor of 1.2 to account for future ASR expansion[, t]he as deformed condition does not significantly impact the dynamic properties of the structure, and therefore the maximum seismic acceleration profiles for OBE and SSE excitation used in original design of the CEB remain valid.

Seabrook, License Amendment Request 16-03 - Revise Current Licensing Basis to Adopt a Methodology for the Analysis of Seismic Category I Structures with Concrete Affected by Alkali-Silica Reaction.

NextEra-ML16216A240 (2016)

Chronology

Oversimplified Analysis

Seismic loads are applied using a static equivalent method utilizing the design-basis maximum acceleration profiles, which were computed during original design from response spectra analysis. Amplify ASR loads by a threshold factor to account for potential future ASR expansion.

...

Response spectra analysis was performed using a simplified “stick” model.

Evaluation and Design Confirmation of As-Deformed CEB, 150252CA-02,” Revision 0, July 2016 (Seabrook FP#100985)

Simpson Gumpertz & Heger-ML16279A049 (2016)

Comments Below

2016 NextEra files a Request to Extend to 15 years leakage test of CBE.
It alleges that

Chronology

NextEra's justification

- Containment's three directional steel reinforcement arrangements, which inhibits ASR expansion,
- The very limited localized areas of ASR detected on the containment surface, and
- Previous UT inspections of the containment liner local to areas of ASR in which no anomalies or corrosion were identified

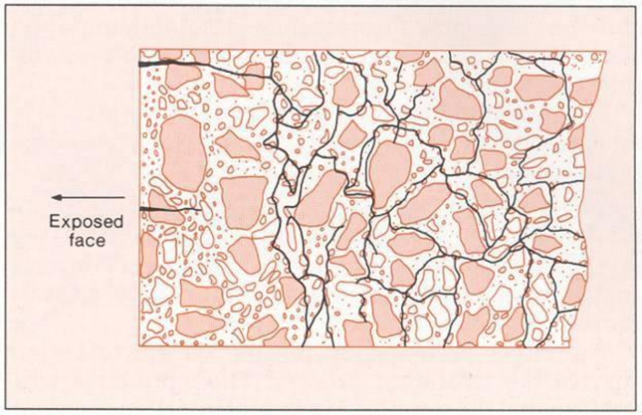
Supplement to License Amendment Request 16-01 Request to Extend Containment Leakage Test Frequency

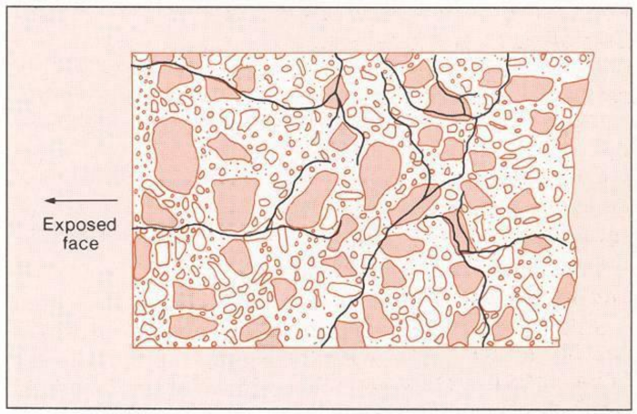
NextEra Energy (2016)

Chronology

C-10 Comments

- ASR causes cracks and microcracks, which not always visible. These may coalesce and create a continuous pathway for gas release.
- There is three-directional reinforcement only around the base, while “skin” reinforcement is applied only on the intrados and extrados.





[Report on the Diagnostis, Prognosis, and Mitigation of Alkali-Silica Reaction (ASR) in Transportation Structures](#) FHWA ([2010](#))

Chronology

2019 Professor Saouma visits Seabrook.
2019 Consolidated documents filed by Dr. Victor Saouma
2019 Proposed Findings of Fact and Conclusions of Law: C-10, NRC, and NextEra.

2020 Atomic Safety Licensing Board (ASLB) Ruling (includes):

... NextEra has not persuaded us that it is properly accounting for the possibility of delamination.

The Board finds that NextEra does not have an adequate screening procedure to detect internal cracking and delamination in Seabrook's concrete."(pg 184)

...[t]he Board is concerned about the potential for sudden significant, localized damage due to shear failure, given that all parties agreed that there may be localized excursions of Seabrook Unit 1 into the nonlinear structure plastifica- tion regime."(pg 184)

Thus, the Board finds that NextEra has not shown, by a preponderance of the evidence, that there is reasonable assurance that the continued operation of Seabrook Unit 1 will not endanger the health and safety of the public with regard to this particular issue of delamination." (pg 185)

In the Matter of NEXTERA ENERGY SEABROOK, LLC (Seabrook Station, Unit 1); Initial Decision

Atomic Safety License Board (2020)

Observations

- In 2010, knowledge about ASR was insufficient, but significant advancements were made by 2020.
- Some premises that initially supported the license renewal were later shown to be incorrect.
- The ASLB comments partially validated this assertion.
- Given the stakes, it is urgent to reconsider two key issues.
  - OBE & OSE
  - Air leakage test

Explanation follows

Q1:Revisit Operating Basis Earthquake (OBE) & Safe Shutdown Earthquake (SSE); Correct Analysis Model

- We have reviewed the dynamic analysis procedure performed by SGH and found it dangerously simplistic.
- The term *significantly* is too vague given the potential impact on the safety of the CEB.
- The reliance on the stick model, a method from the 1970s, is not only outdated but also inadequate; In the 21st century the NRC must demand the adoption of a more accurate model.
- The ASR modeling blatantly disregards well-established principles, directly conflicting with what is universally accepted in the field of modeling.
- Program to simulate ASR not validated.
- The **Capacity** is grossly miscalculated.

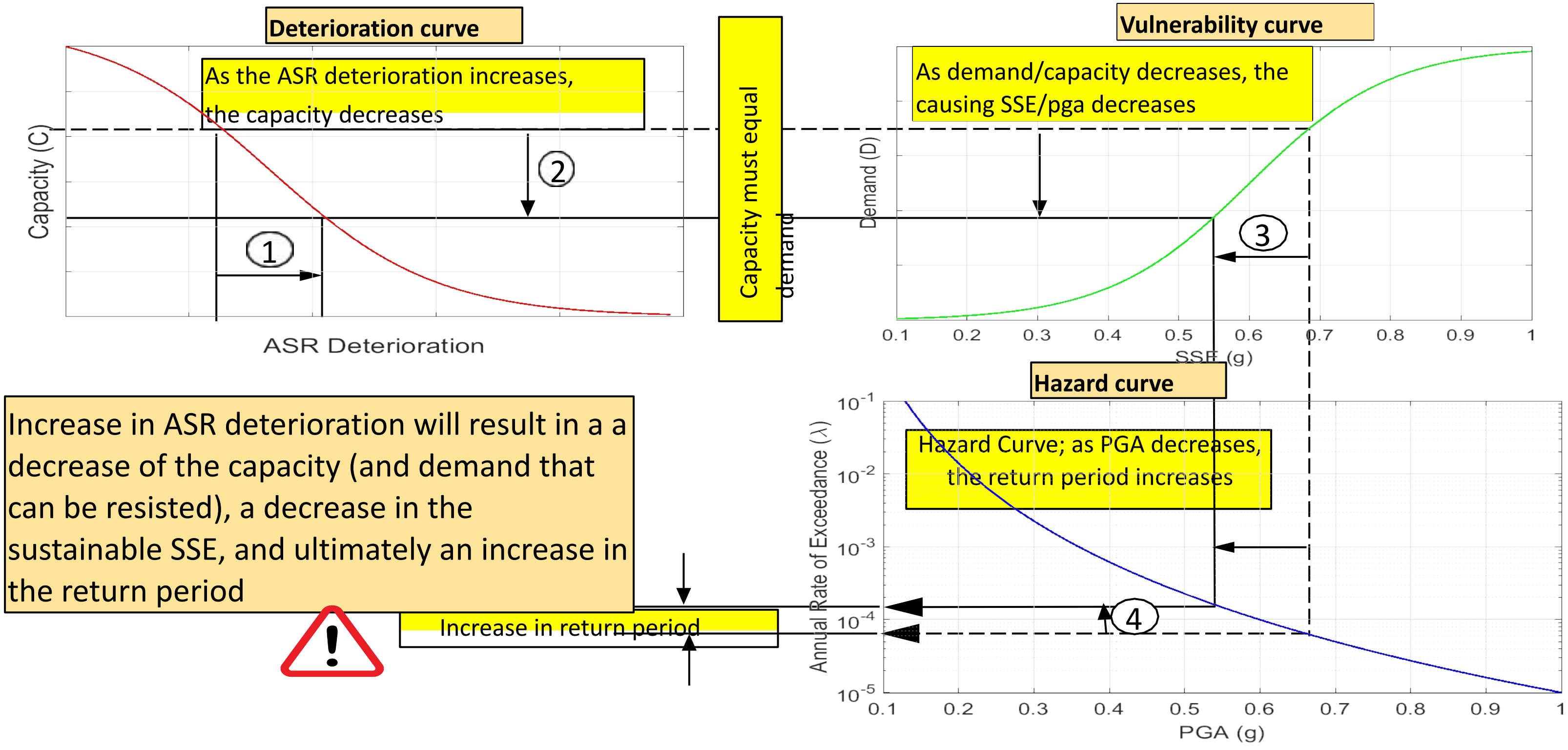
The Myth of No Shear Strength Loss from AAR

Results from the Shear Test Program indicate that there is no reduction of shear capacity in ASR-affected concrete with through-thickness expansion levels up to [redacted]% or volumetric expansion levels [redacted]%, which are the maximum expansion levels exhibited by the test specimens.

Seabrook Station - Approach for Determining Through-Thickness Expansion from Alkali-Silica Reaction
NextEra-ML18141A785 (2016)

- The persistent myth that there is no shear reduction due to AAR in the CEB has been conclusively disproven by findings from two separate NRC-funded research programs.
- AAR will lead to significant shear reduction, critically undermining resistance to earthquake excitation.
- The **Demand** is grossly underestimated

Why the SSE should be updated



Increase in ASR deterioration will result in a a decrease of the capacity (and demand that can be resisted), a decrease in the sustainable SSE, and ultimately an increase in the return period

Q2: Air Leakage Test; Revisit Testing Frequency

Air Leakage Test

- The ASLB firmly asserts that NextEra has no reliable control over where and when cracking will occur.
- This directly undermines NextEra's claim that 15 years cycles for leakage testing are sufficient.
- By 2020, ASR has not only been identified as a critical threat to Seabrook, with hidden cracks often going undetected, but NextEra has also demonstrated a consistently poor record in managing ASR.
- As the years pass areas known to have ASR,and countless unknown areas are experiencing ASR degradation.
- Consequently, we strongly recommend that the full air tightness test schedule be drastically shortened from the current 15 years to a performance-based schedule.

Q3:Crack Indices (CI), Public Right to Know

Public Right to Know Some Data

- The public has a fundamental right to access information that affects their safety and well-being.
- Transparency in sharing data helps build trust between the reactor operator, regulatory agencies, and the public. When data is openly available, it demonstrates that the operator is committed to safety and is accountable for maintaining the highest standards.
- Public access to safety data enables independent experts, researchers, and advocacy groups to analyze the information, potentially identifying issues that may be overlooked by the operator or regulators.
- We understand that NextEra may consider the data confidential; however, we assert that we assert that raw measurement data should be treated as public domain, however we recognize that NextEra's modeling is proprietary.

What Data?
Acceptance criteria for CI measurements

Tier	Structures monitoring program	Recommendation for individual concrete components	Criteria
3	Unacceptable (requires further evaluation)	• Structural evaluation • Implement enhanced ASR monitoring such as through-wall expansion monitoring using Extensometers	1.0 mm/m (0.1%) or greater strain measurement (CCI or pin-pin)
2	Acceptable with deficiencies	Quantitative monitoring and trending	• 0.5 mm/m (0.05%) or greater strain measurement (CCI or pin-pin) • CI or pin-pin measurement of greater than 0.5 mm/m (0.05%) in the vertical and horizontal direction
		Qualitative monitoring	Any area with visual presence of ASR (as defined in [930] accompanied by a CI of less than 0.5 mm/m (0.05%) in the vertical and horizontal directions
1	Acceptable	Routine inspection as prescribed by the Structural Monitoring Program	Area has no indication of pattern cracking or water ingress; No visual symptoms of ASR

Seabrook Station - Approach for Determining Through-Thickness Expansion from Alkali-Silica Reaction

NextEra-ML18141A785 (2016)

Need to Communicate with the Public

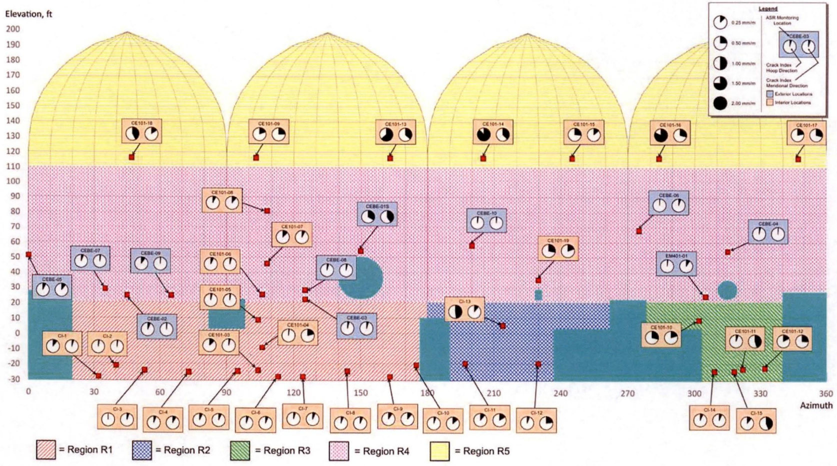
- We ask the NRC to obtain from NextEra and share with the public all measurements related to the crack index

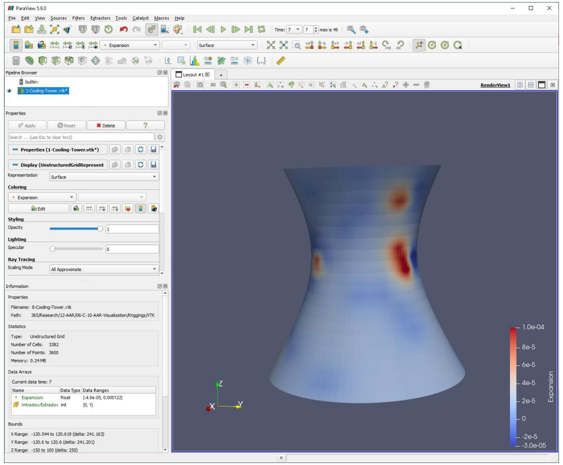
Date	Tier	CI	Location	Ref Values

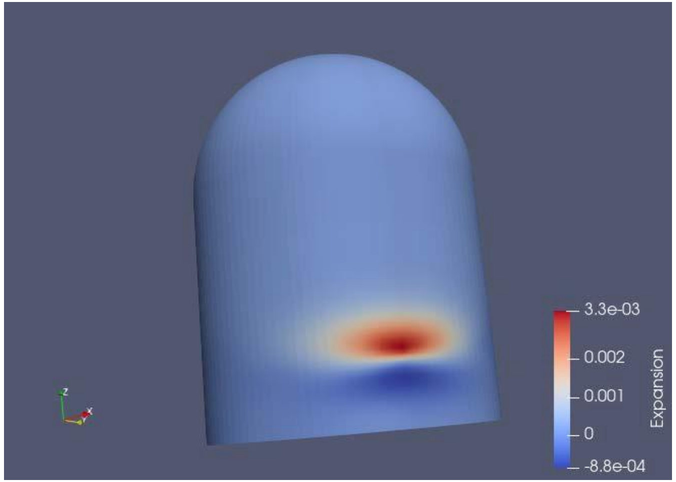
Ref Value: Location of the closest concrete sample cast during construction with known compressive strength f_c'

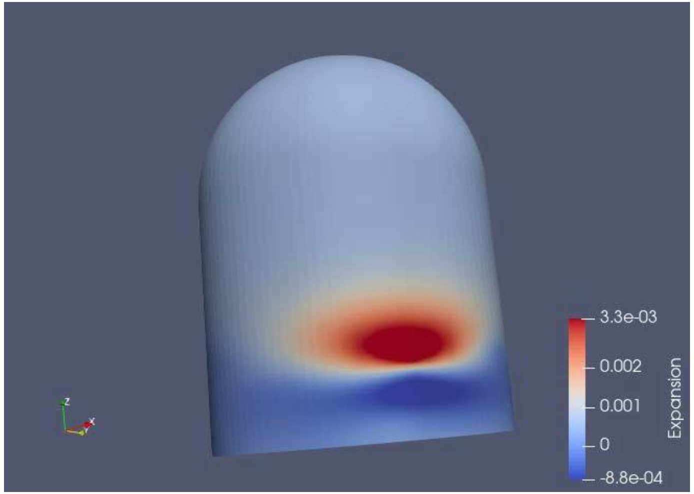
- C-10 has built a data visualizer that we would like to share with the NRC. It is based on the premises that
- ASR has both a spatial and temporal variation.
- It is measured pointwise, yet its impact is spread over volumes
- We only have limited point measurements at discrete times.
- Need to
  - Map
  - Visualize
  - Analyze (and Predict)

Visualize


Elevation, ft
Azimuth
= Region R1
= Region R2
= Region R3
= Region R4
= Region R5
Legend


ParaView 5.9.0
Expansion
1.0e-04
8e-5
6e-5
4e-5
2e-5
0
-2e-5
-3.0e-05


3.3e-03
0.002
0.001
0
-8.8e-04
Expansion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0
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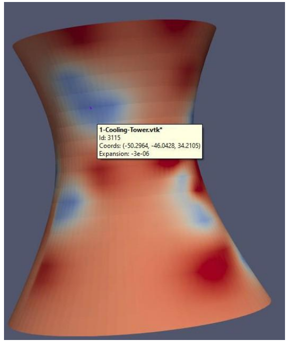
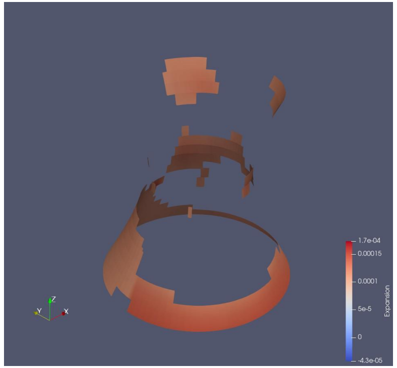
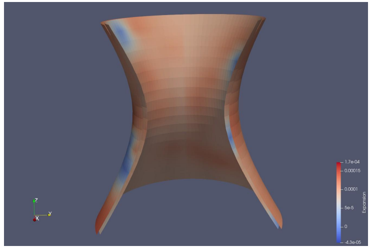
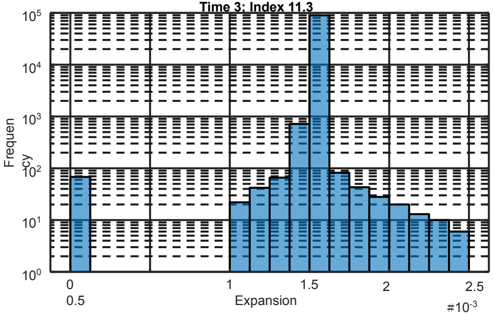
Expansion

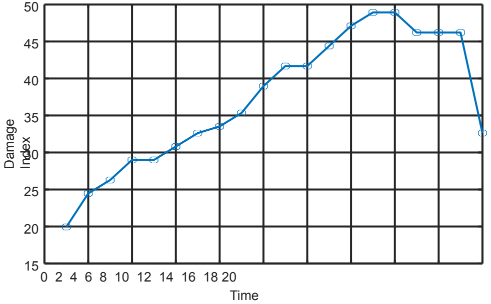
Analyze











Can Quantify structural damage over time.

- Atomic Safety License Board (2020). *In the Matter of NEXTERA ENERGY SEABROOK, LLC (Seabrook Station, Unit 1); Initial Decision*. Docket No. 50-443-LA-2, ASLB No 17-953-02-LA-BD01.
- FHWA (2010). *Report on the Diagnostis, Prognosis, and Mitigation of Alkali-Silica Reaction (ASR) in Transportation Structures*. Tech. rep. FHWA-HIF-09-004. Federal Highway Administration.
- NextEra Energy (May 2016). *Supplement to License Amendment Request 16-01 Request to Extend Containment Leakage Test Frequency*.
- NextEra-ML16216A240 (2016). *Seabrook, License Amendment Request 16-03 - Revise Current Licensing Basis to Adopt a Methodology for the Analysis of Seismic Category I Structures with Concrete Affected by Alkali-Silica Reaction.* ADAMS access number ML16216A250.

NextEra-ML18141A785 (2016). *Seabrook Station - Approach for Determining Through-Thickness Expansion from Alkali-Silica Reaction*. Redacted Document.

Simpson Gumpertz & Heger-ML16279A049 (2016). *Evaluation and Design Confirmation of As-Deformed CEB, 150252CA-02," Revision 0, July 2016 (Seabrook FP#100985)*. Online; accessed 2024-07-16.

Advisory Committee on Reactor Safeguards Full Committee Meeting

Seabrook Alkali Silica Reaction (ASR) Update / Information Briefing

Division of Operating Reactor Safety

U.S. NRC Region I

September 4, 2024

NRC Staff Presenters

Niklas Floyd

Senior Reactor Inspector

Region I

Travis Daun

Senior Resident Inspector, Seabrook

Region I

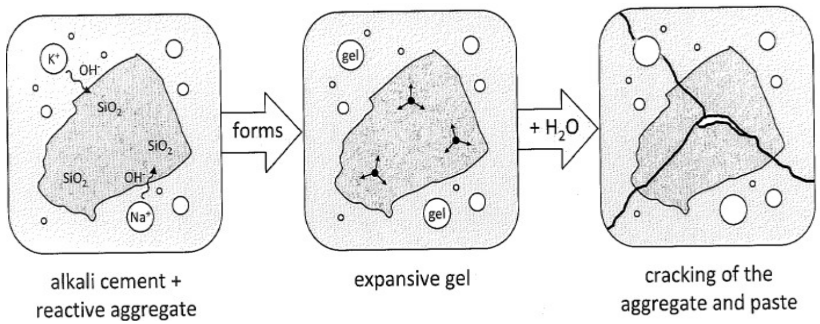
Agenda

- Alkali-Silica Reaction (ASR) Background
- NRC Inspection and Assessment of ASR
- Review of Containment Internal Structures
- Summary

ASR Background

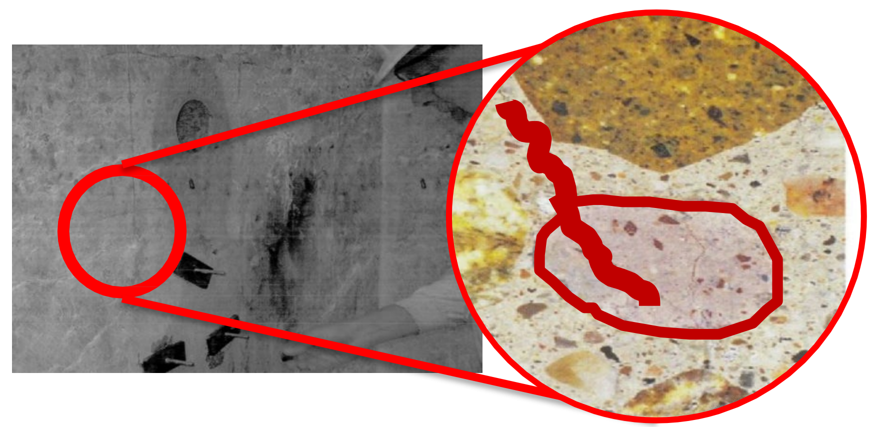
- ASR is a slow, expansive chemical reaction in hardened concrete which occurs in the presence of water, between the alkaline cement and reactive silica found in some aggregates

Reaction:

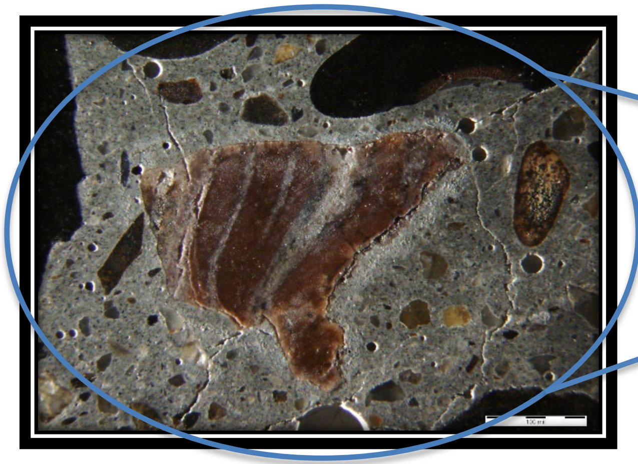


- The expansion can cause various material impacts

- ASR is a very slow-moving phenomenon



ASR Background



Micro-cracking in the aggregate

Visual signs of cracking on the surface of the concrete

ASR Background

- 2009 – 2010: Degradation in Seabrook concrete exposed to groundwater identified during license renewal audit walkdowns
  - Testing confirmed the presence of ASR
  - Codes of record in the Seabrook current licensing basis did not account for ASR
  - NextEra initiated prompt operability determination (POD) and extent-of-condition (early 2011)

- Why ASR occurred and was not identified earlier:
  - Seabrook unknowingly used a slow-reactive aggregate in the concrete
  - Ineffective ASTM Standards at the time of construction
  - ASR development was not expected
  - Inspections were not looking for ASR

ASR Background

- NextEra concluded that ASR-affected structures were operable but degraded and non-conforming
- NRC regional inspectors and headquarters staff reviewed operability determinations and concluded that ASR-affected structures remained capable of performing their safety functions
- 2012: NRC increased oversight to ensure structures remained functional while NextEra developed corrective actions

ASR Background

<u>Large-Scale Testing Program:</u>

- NextEra test program at The University of Texas at Austin

- 2013 – 2016: NRC conducted inspections at the test facility to ensure NRC requirements for quality test standards were met

ASR Background

Large-Scale Testing Program Results:

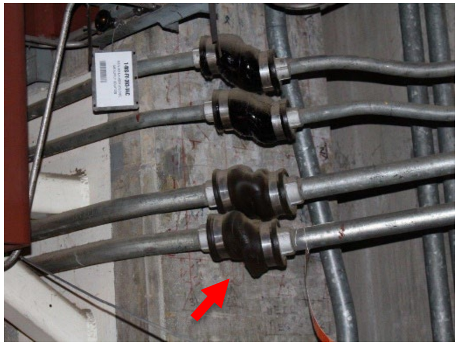
- Showed no reduction in structural capacity up to the expansion levels tested, and code equations can be used up to the tested limits
- Determined through-wall expansion was best way to track ASR progression after in-plane expansion plateaus
- Data was used to develop a correlation between measured modulus-of-elasticity and through-wall expansion; used to estimate expansion until extensometer installation
- Expansion levels* from the testing were added to Seabrook's CLB as expansion limits for capacity limit states

**During 1Q2023, NextEra communicated to inspectors that expansion trends were projected to exceed the limits prior to license expiration and would likely require additional large-scale testing to support increased expansion levels (ML23129A193)*

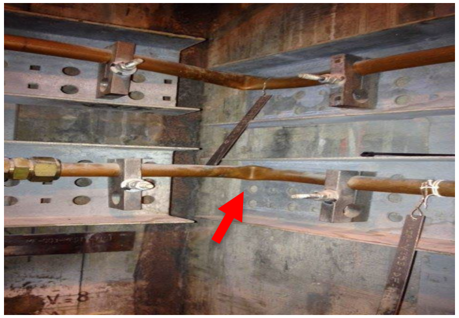
ASR Background

<u>Building Deformation Program</u>:

- 2014 – 2015: NRC identified bulk structural deformation in the following Seismic Category 1 structures on site:
  - Containment Enclosure Building
  - Residual Heat Removal vaults
  - Spent Fuel Building



- Bulk deformation results in additional loading and can impact equipment



- Building deformation was incorporated into NextEra's license amendment request

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Protecting People and the Environment

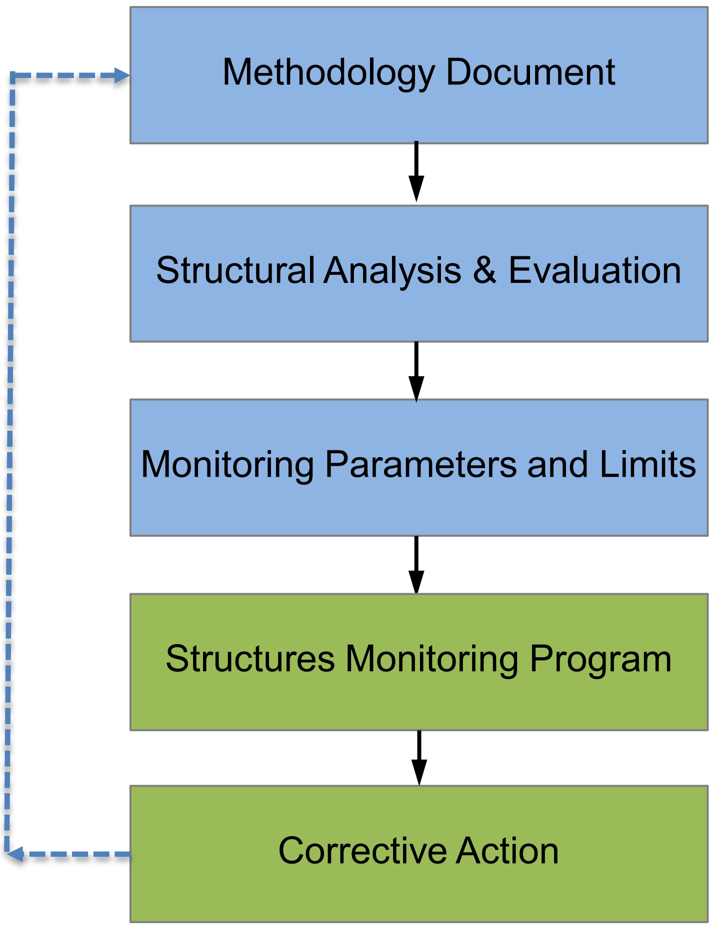
ASR Background

<u>Building Deformation Program</u>:

- 3-stage analysis methodology to address ASR loads
  - More detailed and accurate as stages progress from 1 to 3
- Estimates ASR loads based on field measurements
  - Also accounts for future ASR progression
- Demonstrates Capacity ≥ Demand (including ASR)
- Identifies quantitative acceptance criteria (threshold monitoring parameters, limits) for each structure based on analysis; triggers corrective action when approached or exceeded
- Allows for structural modifications or further evaluation
- Methodology described in the license amendment and incorporated into Seabrook's Structures Monitoring Program

ASR Background

Methodology Overview:



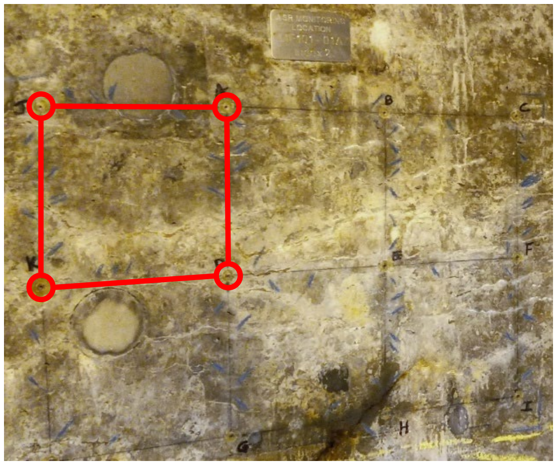
- Stage 1, 2, and 3 analyses
- Load demand calculations
- Design margin for future ASR expansion
- Threshold limits
- Monitoring parameters and frequency
- Trending
- Action if approaching threshold limits
- Further analysis and/or monitoring
- Structural modification

ASR Background

- License Amendment Request:
  - Incorporated test program expansion limits for capacity
  - Detailed methodology for evaluating the effects of ASR on Seabrook structures (incorporates ASR load/demand, acceptance criteria)
  - Monitoring provides for timely corrective action
- License Renewal Application:
  - Aging management programs (AMPs) identify and manage future effects of aging
  - License renewal application supplemented to include ASR monitoring and evaluation programs as AMPs
- NRC approved and issued both the license amendment and renewed license in March 2019.
  - ACRS meetings and ASLB hearing completed

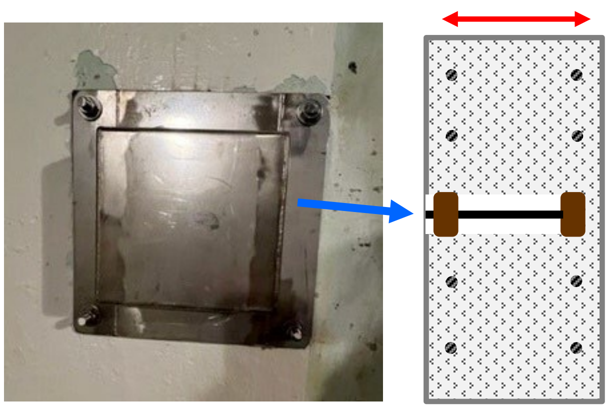
ASR Background

Examples of Monitoring ASR at Seabrook:

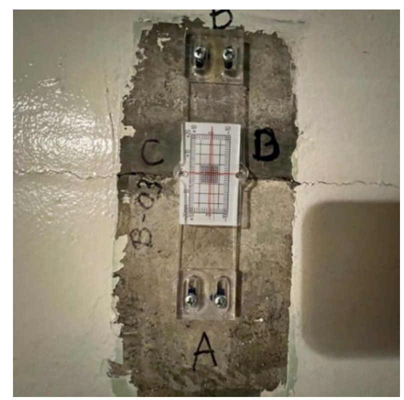


Combined crack indexing (CCI) and pin-to-pin distance

Seismic gap widths between buildings



Extensometers



Crack gauges

ASR Background

Summary of ASR License Conditions:

a) Assess expansion behavior to confirm it is comparable to the large-scale test program and check margin for future expansion.

b) Corroborate, using Seabrook field data, the concrete modulus to expansion correlation used to calculate pre-instrument through-thickness expansion.

c) Conduct a volumetric expansion check for control extensometers every six months.

d) Develop a monitoring program to ensure that rebar failure or yielding does not occur, or is detected if it has already occurred, if the structural evaluations indicate rebar stress may exceed yield.

e) If the ASR expansion rate significantly exceeds 0.2 mm/m (0.02%) through-thickness expansion per year, NextEra will perform an engineering evaluation focused on the continued suitability of the six-month monitoring interval.

f) Each core extracted from Seabrook Unit 1 will be subjected to a petrographic analysis to detect internal microcracking and delamination.

NRC Inspection and Assessment of ASR

- NRC resident inspectors performed daily onsite oversight via plant status (corrective action report review and walk downs)
- Inspectors selected risk-informed and performance-based samples, including maintenance rule, operability determinations, modifications, and focused PI&R samples
- Regional specialists and NRR / RES technical staff performed focused inspections
- Results documented in inspection reports
- Results and oversight plans discussed with Region I senior managers and during end-of-cycle reviews

NRC Inspection and Assessment of ASR

- Oversight of ASR since last ACRS update in April 2022:
  - 5 weeks of on-site inspections by team of regional inspectors and NRR/RES technical staff
  - 11 total inspection samples focused specifically on ASR related activities

- Two other team inspections reviewed ASR-related items during baseline ROP inspections. These included the Biennial Problem Identification and Resolution and the Age-Related Degradation team inspections.

NRC Inspection and Assessment of ASR

- NRC inspections determined Seabrook structures remained capable of performing their intended safety functions

- NRC inspectors identified three findings involving ASR activities since 2022. The violations were found to be of very low safety significance and have been addressed in NextEra's corrective action program

NRC Inspection and Assessment of ASR

- Determined NextEra reviewed Seismic Category I structures using approved Methodology Document
  - **28** total structures in program
  - **6** structures remained outside of the licensing basis

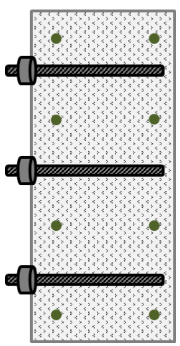
- Determined functionality of six structures is documented in Prompt Operability Determination with supporting analyses
  - Additional monitoring/trending (typically every 2 months)
  - Long-term corrective actions via modification and/or reanalysis

NRC Inspection and Assessment of ASR

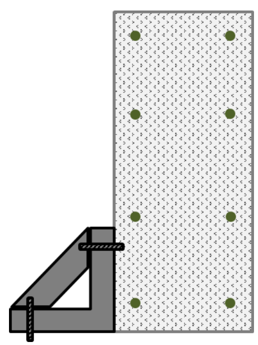
<u>Corrective Actions Planned for Structures Approaching or Outside of Licensing Basis Limits:</u>

- Physical Modification and/or Re-analysis

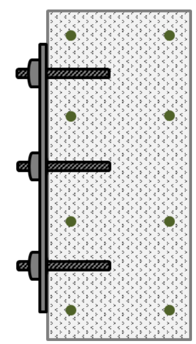
- Physical modification process described in the NRC-approved methodology to evaluate and manage the effects of ASR



Example 1 - Strong Backs



Example 2 - Corner Braces



Example 3 - Vertical Plates

NRC Inspection and Assessment of ASR

Examples of Modifications:

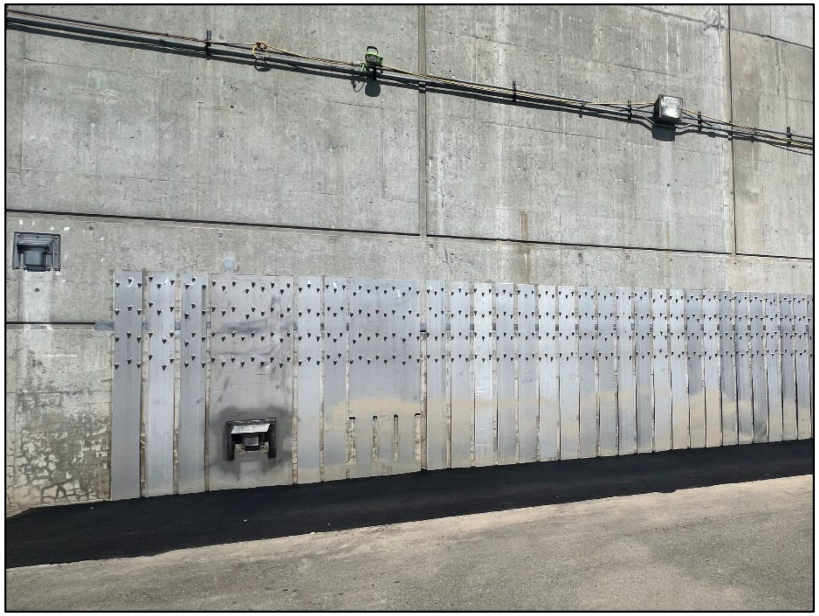


Figure 1. Vertical plates

Figure 2. Corner braces and strong backs

NRC Inspection and Assessment of ASR

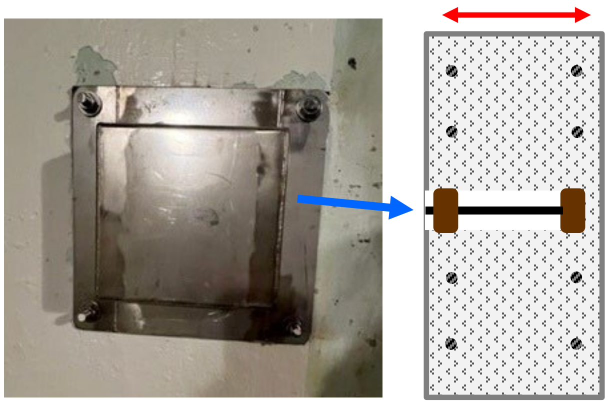
<u>Inspection Findings:</u>

- NRC inspectors identified 3 findings of very low safety significance involving NextEra's performance and ASR
  - *NCV 2022002-01* Did not install extensometers in 7 ASR Tier 3 locations
  - *NCV 2024010-02* Did not correct an adverse condition associated with a tubing support from ASR deformation
  - *NCV 2024001-02* Did not analyze the reactor cavity pit slab in the revised CIS structural evaluation

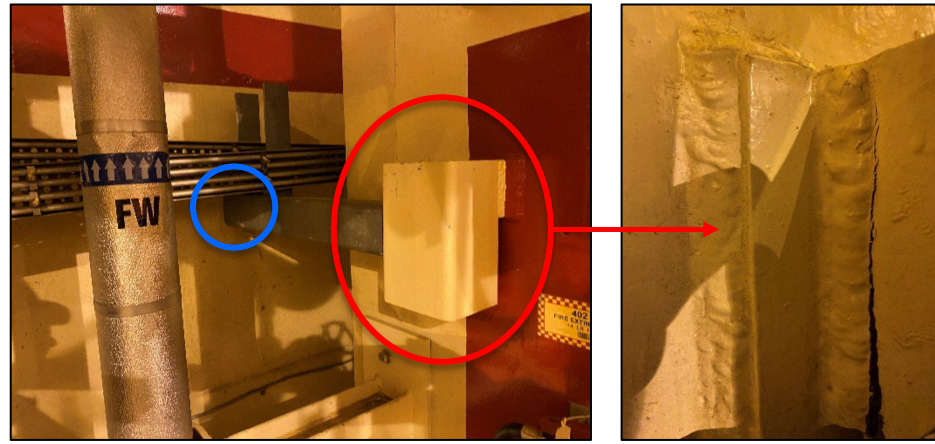
- Determined the 3 findings were addressed in NextEra's corrective action program

NRC Inspection and Assessment of ASR

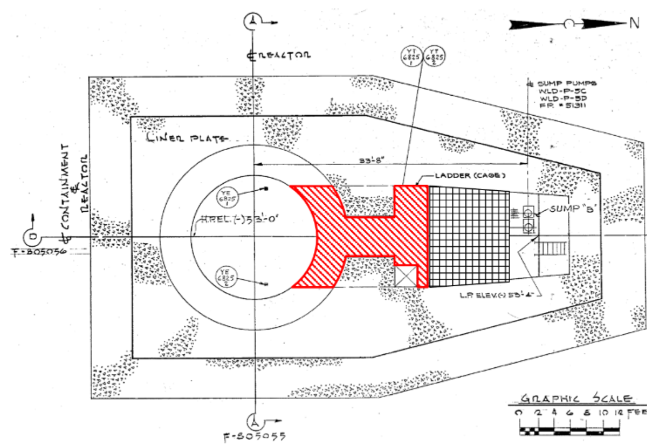
Inspection Findings:



NCV 2022002-01



NCV 2024010-02



NCV 2024001-02

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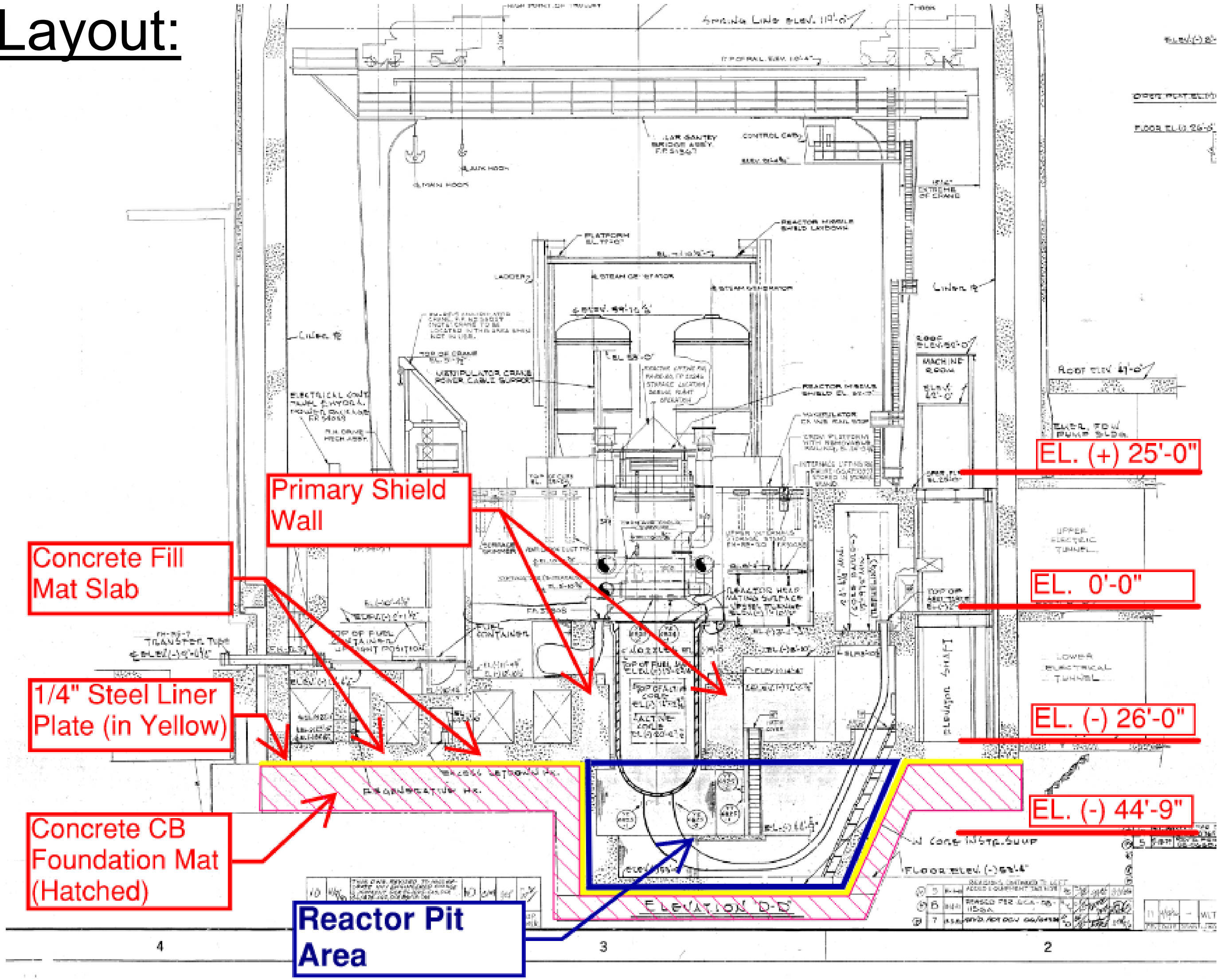
Review of Containment Internal Structures

- **Nov 2021**: NRC identification of possible ASR cracking in reactor cavity pit of the containment internal structures (CIS) documented as Green finding in 4Q2021.
- **April 2023**: NextEra completed root cause studies to best estimate the amount and location of ASR based on observed distress (cracks and spalling).
- **Nov 2023**: NextEra completed the "stage 2" structural evaluations in accordance with NRC-approved methodology.
- **Mar 2024**: NRC reviewed the CIS root causes and structural evaluations. NRC identified one finding because NextEra staff did not analyze a particular element in the reactor pit slab in their CIS structural evaluation.

Review of Containment Internal Structures

CIS Layout:



Review of Containment Internal Structures

<u>NextEra’s Approach for Analysis:</u>

- Collected data from field measurements (crack sizes, crack locations, temperatures, spalling) inside the CIS

- Developed finite element models for heat transfer and stress from "in situ" loads

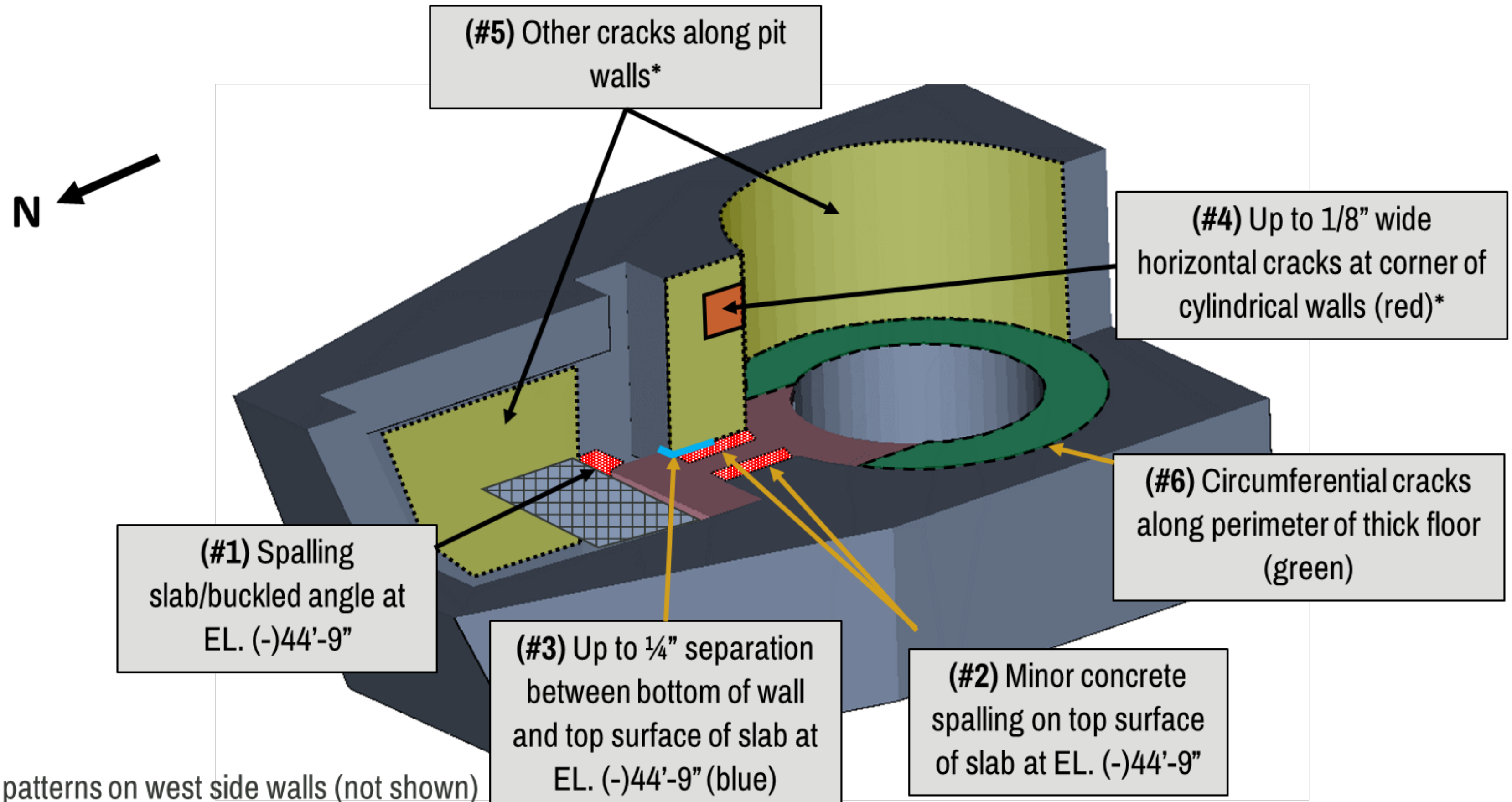
- Performed parametric studies varying the amount of ASR expansion and thermal loads to understand the observed distresses

- Documented the results in root cause reports

- Utilized the results as an input into the CIS stage 2 structural evaluations:
  - Reactor cavity pit area
  - Superstructure (areas above the reactor cavity pit, including the fill mat)

Review of Containment Internal Structures

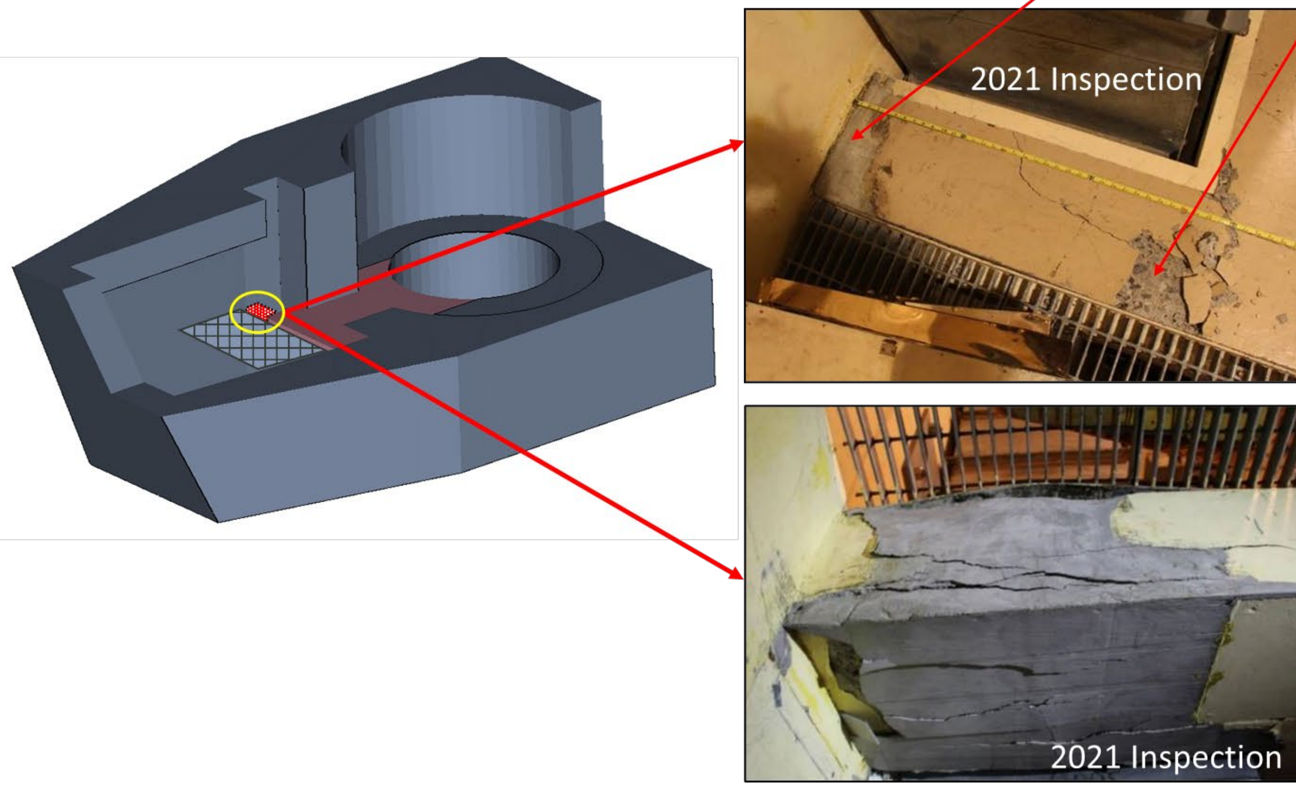
CIS Field Observations:



*Cracking patterns on west side walls (not shown) are similar to those on east side walls (#5)

Review of Containment Internal Structures

CIS Field Observations:

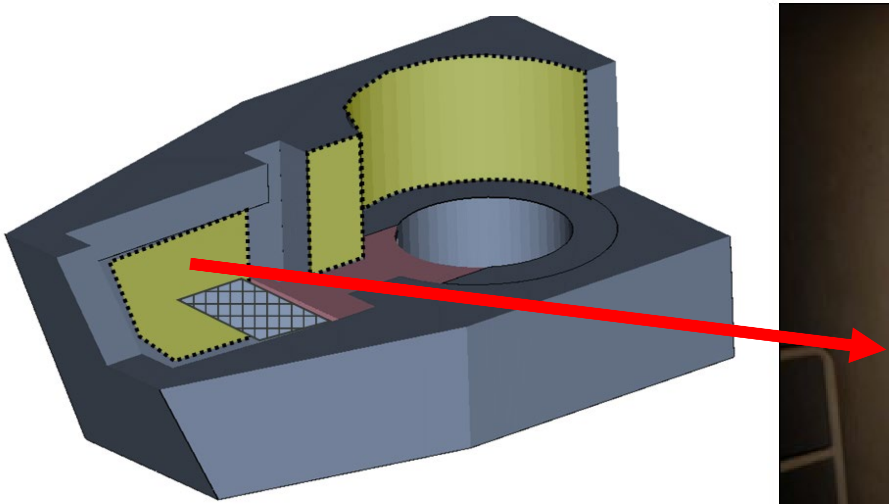


- In 2021, failure of 2012 repair was observed, and second repair was performed
- In 2023 outage, an encapsulation was installed for spalling concrete

Review of Containment Internal Structures

CIS Field Observations:

- No apparent changes in 2017, 2021, or 2023 inspections



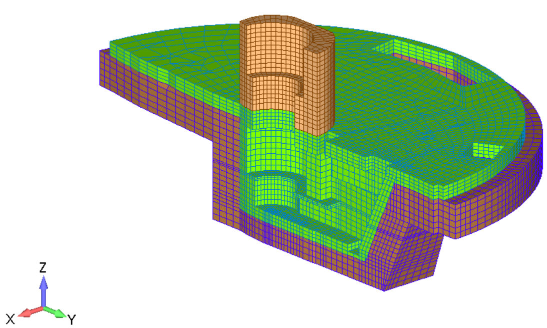
Typical Cracking Patterns Along Pit Walls

Review of Containment Internal Structures

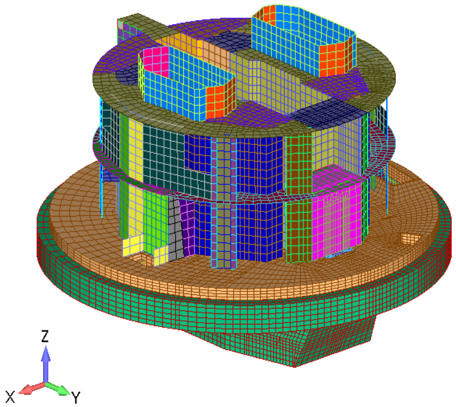
<u>Finite Element Models:</u>

- NextEra developed models and completed parametric studies to identify the likely causes of observed structural conditions in CIS:
  - Scenarios analyzed concrete expansion due to thermal, ASR, and varying combinations
  - Other effects analyzed such as cycling fatigue, stress concentrations, and creep

- The studies used best estimated actual loads (in-situ loads) to simulate the observed conditions



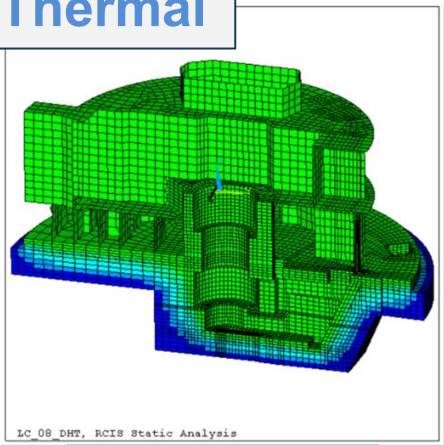
Heat Transfer Model



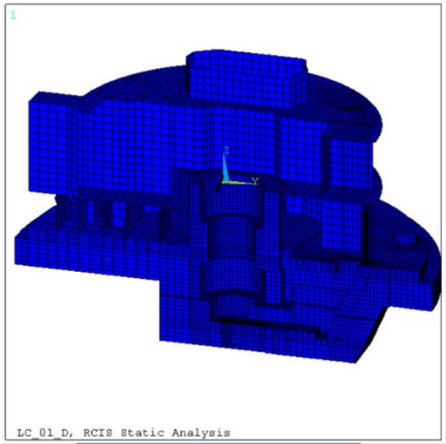
Global Stress Model

Review of Containment Internal Structures

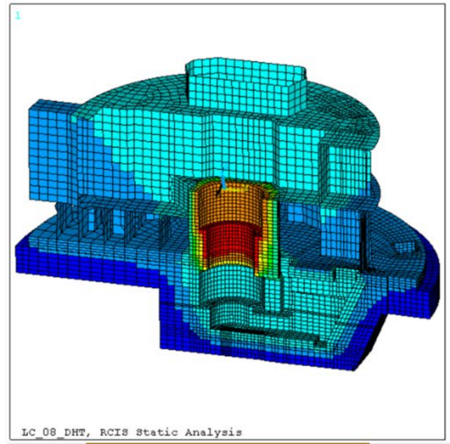
Thermal



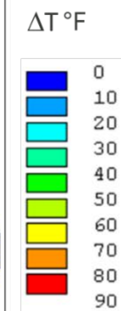
Scenario 1, Variation A
(Design basis thermal)



Scenario 2
(no thermal)

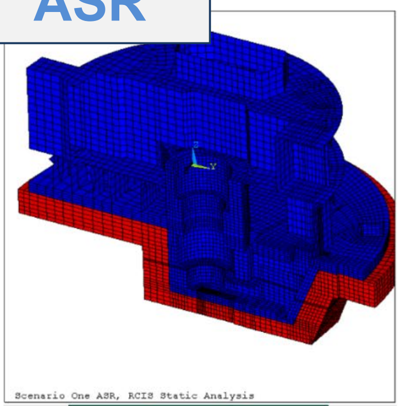


Scenario 3, Variation E
(operating thermal)

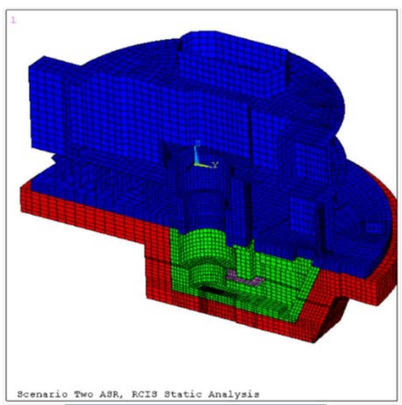


Parametric Study Example Scenarios

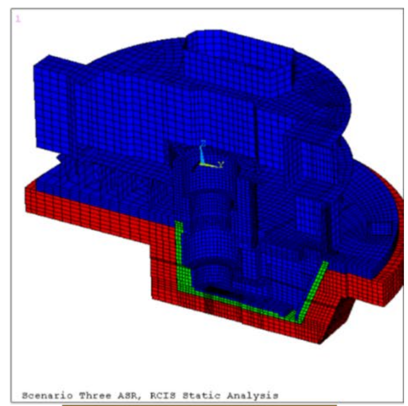
ASR



Scenario 1
(representative case)



Scenario 2
(representative case)



Scenario 3
(representative case)

Legend
- No ASR
- ASR in CB Foundation
- ASR in Reactor Pit
- ASR in Slab at EL. (-)44'-9"

Review of Containment Internal Structures

Parametric Study Example:

<table>
<tr><th rowspan="3"></th><th rowspan="3">Thermal Variation</th><th colspan="6">Thermal Load Boundary Conditions</th><th rowspan="3">ASR</th><th rowspan="3">Compare to Observed Conditions?</th></tr>
<tr><th colspan="2">Steady-State Operation</th><th colspan="2">Shutdown</th><th colspan="2">Reference</th></tr>
<tr><th>Inside CIS</th><th>Bedrock Interface</th><th>Inside CIS</th><th>Bedrock Interface</th><th>Inside CIS</th><th>Bedrock Interface</th></tr>
<tr><td rowspan="4">Scenario One</td><td>A</td><td>120 °F</td><td>40 °F</td><td rowspan="4">80 °F</td><td rowspan="4">40 °F</td><td rowspan="2">70 °F</td><td rowspan="2">40 °F</td><td rowspan="4">CB foundation only</td><td rowspan="4">No</td></tr>
<tr><td>B</td><td>110 °F</td><td>40 °F</td></tr>
<tr><td>C</td><td>120 °F</td><td>40 °F</td><td rowspan="2">60 °F</td><td rowspan="2">40 °F</td></tr>
<tr><td>D</td><td>110 °F</td><td>40 °F</td></tr>
<tr><td rowspan="3">Scenario Two</td><td rowspan="3">NO Thermal Load</td><td colspan="2" rowspan="3">N/A</td><td colspan="2" rowspan="3">N/A</td><td colspan="2" rowspan="3">N/A</td><td>CB foundation only</td><td>No</td></tr>
<tr><td>CB foundation and reactor pit</td><td>No</td></tr>
<tr><td>CB foundation, reactor pit, and slab at EL. (-) 44’</td><td>No</td></tr>
<tr><td rowspan="4">Scenario Three</td><td rowspan="3">E</td><td colspan="2" rowspan="4">90°F inside CIS reactor pit
150°F on the inside face of the primary shield wall
90°F on the outside face of the primary shield wall
77°F on the top surface of the fill mat slab at EL. (-) 26’
40°F at bedrock interface</td><td rowspan="4">80 °F</td><td rowspan="4">40 °F</td><td rowspan="3">60 °F</td><td rowspan="3">40 °F</td><td>CB foundation only</td><td>Yes</td></tr>
<tr><td>CB foundation and fill mat slab at EL. (-) 26’</td><td>No</td></tr>
<tr><td>CB foundation and reactor pit</td><td>Yes</td></tr>
<tr><td>F</td><td>70 °F</td><td>40 °F</td><td>CB foundation only</td><td>No</td></tr>
</table>

Review of Containment Internal Structures

Additional Parametric Studies:

- NextEra reviewed conditions in other areas of the CIS to determine the cause of the observed distress:
  - El. (-) 26ft fill mat slab and adjacent to the sump at azimuth 80^{o}
  - Personnel elevator with base at El. (-) 26 ft azimuth 335^{o}

- Studies analyzed combinations of in-situ load conditions (thermal and ASR) with results documented in two root cause reports

- NextEra concluded the conditions were likely due to effects other than ASR

Review of Containment Internal Structures

Ongoing Monitoring

- Continued monitoring inside the CIS to see whether any ASR expansion is occurring:
  - Fill Mat Slab at EL (-) 26 ft
  - Secondary Shield Wall at EL (-) 26 ft
  - Walls adjacent to Sump at AZ 80° on EL(-) 26 ft
  - Elevator at AZ 335° on EL (-) 26 ft and 0 ft
  - Reactor Cavity Pit Area

- Retrieve reactor pit temperature data during the upcoming fall 2024 refueling outage

Review of Containment Internal Structures

NRC Review:

- Inspectors verified that NextEra completed the CIS structural evaluation in accordance with their revised licensing basis and the Stage 2 analysis process in the NRC-approved methodology document

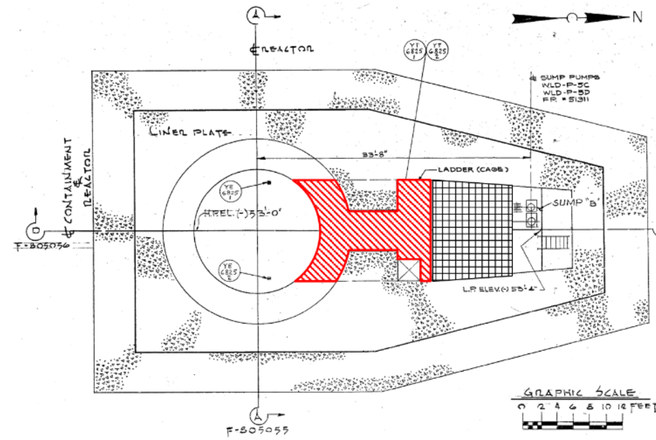
- The inspectors concluded that root cause studies were of appropriate technical detail to develop insights and provide for upper limit estimates of possible ASR/swelling expansion that reasonably correlated with CIS field observations. Plans were in place to continue to monitor and refine results.

- Inspectors noted there was an upper limit of ASR expansion of 0.2mm/m in foundation mat with 30% margin for expansion

- NRC identified one violation because NextEra did not evaluate the reactor pit slab in the revised structural evaluation

Review of Containment Internal Structures

Violation:



NextEra did not verify the adequacy of the reactor pit slab at el. (-) 44'-9" for the applicable design loads, including ASR, in the revised structural evaluation for the CIS.

Corrective Actions:

- NextEra performed a prompt operability determination to confirm the integrity of the reactor pit slab for the unusual load combination with a safe shutdown earthquake
- Engineering change in-progress to reclassify the slab as non-Seismic Category I and perform the necessary analysis for SC II over I
- Designed a preventative modification as contingency for protection of components below the slab in the event of concrete spalling

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Summary

- NRC inspections determined Seabrook structures remained capable of performing their safety functions as intended, including under limiting conditions

- NRC inspections identified some findings and violations of very low safety significance which have been addressed in the corrective action program

Summary

<u>Next Steps</u>:

- NRC will continue inspections under the Reactor Oversight Process that includes NextEra's ASR-related activities to:
  - Bring structures into compliance with licensing basis
  - Monitor ASR and building deformation
  - Complete modifications (physical and/or reanalyze) on affected structures.
  - Validate thermal and ASR loads in CIS
  - Implement ASR related license conditions

- Monitor NextEra's plans that may potentially involve large-scale testing

Summary

Public References:

NRC Webpage on Concrete Degradation at Seabrook

https://www.nrc.gov/reactors/operating/ops-experience.html

NRC Webpage on Seabrook License Renewal

https://www.nrc.gov/reactors/operating/licensing/renewal/applications.html

Documents Available are at https://adams.nrc.gov/wba/:

- ASR license amendment request
  - NRC staff safety evaluation: ML18204A291
  - ACRS letter: ML18348A951
- License renewal application
  - NRC staff safety evaluation report: ML18362A370
  - ACRS letter: ML18353A954

End of Staff Presentation