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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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728TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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WEDNESDAY

SEPTEMBER 3, 2025

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The Advisory Committee met at 11545 Rockville Pike, Rockville, Maryland, and via Videoconference, at 8:30 a.m., Walter L. Kirchner, Chair, presiding.

COMMITTEE MEMBERS:

WALTER L. KIRCHNER, Chair

GREGORY H. HALNON, Vice Chair

DAVID A. PETTI, Member-at-Large

VICKI M. BIER

VESNA B. DIMITRIJEVIC*

CRAIG D. HARRINGTON

ROBERT P. MARTIN

SCOTT P. PALMTAG

THOMAS E. ROBERTS

MATTHEW W. SUNSERI

ACRS CONSULTANT:

RONALD BALLINGER

DENNIS BLEY*

DESIGNATED FEDERAL OFFICIAL:

CHRISTOPHER BROWN

WEIDONG WANG

ALSO PRESENT:

ALAN BLIND

STEVEN BLOOM

LARRY BURKHART

ANDREW COOKE

THOMAS DASHIELL

MICHAEL KEEGAN

PAUL KLEIN

PATRICK KOCH

KAMAL MANOLY

KEIKO MORITA

TAMARA SKOV

GEORGE THOMAS

IAN TSENG

SHANDETH WALTON

*Present via video-teleconference

C-O-N-T-E-N-T-S

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

8:30 a.m.

CHAIR KIRCHNER: Good morning. The meeting will now come to order. This is the first day of the 728th meeting of the Advisory Committee on Reactor Safeguards, ACRS. I'm Walt Kirchner, Chairman of the ACRS. ACRS members in attendance in person today are Vicki Bier, Greg Halnon, Craig Harrington, Robert Martin, Scott Palmtag, Dave Petti, Thomas Roberts, and Matt Sunseri. ACRS member Vesna Dimitrijevic is participating virtually via Teams.

ACRS consultant Ron Ballinger is participating in person. And I see that we have Dennis Bley, also a consultant, joining us remotely.

If I have missed anyone, either ACRS members or consultants, please speak up now. None? Okay.

Christopher Brown of the ACRS staff is the Designated Federal Officer for this morning's full committee meeting. Member Sunseri is recused from this afternoon's Seabrook topic due to a potential conflict of interest.

I know we also have a quorum. The ACRS was established by the Atomic Energy Act and is governed by the Federal Advisory Committee Act. Under

the Atomic Energy Act, ACRS shall advise the Nuclear Regulatory Commission on the hazards of proposed and existing reactor facilities and the adequacy of proposed safety standards.

Following Executive Order 14300, the Committee has narrowed its focus to only those activities necessary to fulfill its statutory obligations. As a result, ACRS is prioritizing the review and reporting of new reactor facilities and proposed safety standards, with particular attention to those issues that are unique, novel, and noteworthy. And the Committee will also consider other nuclear safety matters at the direction of the Commission, and that includes this morning's topic.

Please note that the ACRS speaks only through its published letter reports. All member comments should be regarded as only the information opinion of that member and not a Committee position.

Information about the ACRS activities, such as letters, meeting rules, and transcripts are on the NRC public website and can be found by searching for "About Us ACRS" on NRC's homepage.

The ACRS provides an opportunity for public input and comment during our proceedings. For this full committee meeting, we have received no

written statements this morning. Written statements, however, may be forwarded to today's Designated Federal Officer. We have also set aside time at the end of this morning's session for public comments.

The 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 on Teams. If you are participating by phone, press star-6 to mute your phone and star-5 to raise your hand on Teams.

The Teams chat feature is only for communicating IT issues or brief meeting logistics. Please do not use it for comments or questions on topics under Committee discussion. For everyone in the room, please put all of your electronic devices in silent mode, and mute your laptop microphone and speakers. In addition, please keep sidebar discussions in the room to a minimum since the ceiling microphones are live.

For our presenters, your table microphones are unidirectional, and you'll need to speak into the front of the microphone to be heard online and also

for the court reporter. Finally, if you have any feedback for the ACRS about today's meeting, please fill out our public meeting feedback form on the NRC's website.

Today, we will consider this morning the Palisades Nuclear Plant restart activities. Just looking ahead this afternoon, we will take up the Seabrook ASR. Tomorrow morning will be our planning and procedures, and we will probably continue report writing tomorrow afternoon and perhaps into Friday morning. And with that, unless there's any comments from members, I'm going to pass the mic on deliberations to Greg Halnon who is our subcommittee chair for plant operations. Greg?

VICE CHAIR HALNON: Thank you, Walt. Just a quick pause. Okay, good morning. Again, my name is Greg Halnon, the subcommittee chair for the Palisades restart effort.

We're here to -- this morning to discuss the restart efforts at Palisades. Overall, the governance of the restart is sound. And I think the Committee is relatively comfortable with the restart panel and how the produce the products that they have done so far.

But the most significant issue on the

table is the ability of the steam generators to be an effective RCS pressure boundary. There have been circumstances during the shutdown. Did some repairs that caused us to pause and ask more questions of our staff experts on the effectiveness of the steam generators and the inspections.

We had asked the staff to come here today to have a conversation about the steam generators for a cycle of operation and the short- and long-term risk of their present condition. With that Paul Klein and Andrew Johnson from the staff are here. And I'll turn it over to them to start. I think you guys want to star addressing some follow-up from the subcommittee that we had three weeks ago. So Paul and Andrew, I'll turn it over to you.

MR. KLEIN: Thank you. Good morning. I'm Paul Klein from Nuclear Reactor Regulation, the Corrosion and Steam Generator Branch. And I did have one follow-up item from the subcommittee meeting.

One of the questions we were asked was related to what I think inspection in circa 1997 which would've been the last steam generator inspection for Indian Point before the tube rupture versus the practices today. And so between the subcommittee and today's meeting, we had a chance to do a little bit of

homework. And I wanted to elaborate a little bit on our response of the subcommittee.

And so in terms of a couple of topics, we wanted to compare that inspection practice back in '97 versus Palisades in 2024 with the benefit of doing some additional research in the meantime. So it appeared from reading the lessons learned at Indian Point that one of the biggest issues that happened at that time was noise in the steam generators and U-bend and just the ability to detect cracking in the U-bends. And so an indication was missed that it eventually led to the tube rupture.

And so if you compare the '97 inspection at Indian Point to they were using a Cecco-5 bobbin combination probe at that time. And that probe had limited field experience. Of course, the bobbin probe did now.

But the Cecco probe was relatively limited field experience for that point. And it was chosen over a plus point for the speed of the inspection. Palisades in 2024, they also used a variety of probes. But including the bottom probe again and the plus point which is a service writing probe and has much greater detection ability than the Cecco-5 probe, for example, which is a send-receive rate type probe.

I think more importantly the industry practices have become more much mature over that time. And in terms of noise, the '97 time frame, there wasn't as much of a criteria for noise in industry inspections. And data quality was something that was treated site by site at the time.

I think now the noise in data quality criteria are well established with the EPRI guidelines. And in terms of inspection analysis, in the '97 time frame, it would've been manual analysis. And today's world, you use typically a combination of manual analysis and automated analysis.

And the automated analysis programs have come quite a long ways in that interim. And one of the things they will typically flag or be noise exceeders. So if you get an area of high noise in a generator, the automated analysis will flag that for resolution analysts to make sure that they follow up with that site.

And the tube integrity engineer who ultimately is responsible for tube integrity sets a threshold for noise. And so if you have a noisy part in a generator, typically you would go in with a plus point and use your best probe in order to examine that area to try and avoid the kind of situation that

occurred back in Indian Point. I think also programmatically-wise and tech spec framework-wise, there's been a transition from that late '90s time frame where the tech specs are very prescriptive.

You typically did some sample and categorized results. And then based on that initial sample in that category, you may or may not do additional tests. And so what industry and NRC came to realize over time is that the tech specs became outdated.

And so in the early 2000's leading up to test at 447 and the I-9706 when it was implemented, the focus now is on tube integrity. And so it's a whole different approach. It's performance based and tube integrity is the goal in everyone's mind when they performance inspections in a steam generator. And they do the analyses coming out of those inspections. I think those are the main highlights that we wanted to hit.

VICE CHAIR HALNON: I'm going to ask Ron. You had those questions.

MEMBER BALLINGER: We talked earlier.

VICE CHAIR HALNON: All right. So we don't have a design presentation this morning because we're focused solely on the steam generators. I know

that there's been -- individuals have had -- on the Committee have had questions specific to it. So I would just ask that members with questions, go ahead and start, for lack of a better term, peppering Paul and Andrew with your questions and let's get the conversation started. Scott?

MEMBER PALMTAG: This is Scott Palmtag. So I appreciate your presentation from August. I thought that was really useful. I understand the inspections really well.

I'm kind of concerned with the rate of change. So excuse me. My understanding is these steam generators operated for several years with deferred maintenance. And that allowed accumulation of crud on the tubes and the stress corrosion cracking to start.

There were several recommendations were made for chemical cleaning as far back as 2015-2016 time frame. And all the recommendations for cleaning were deferred. After this, they were shut down and the steam generators were placed in a wet state with unknown chemistry and in time that this stress corrosion cracking progressed.

The previous inspection has 56 stress corrosion cracking indications. The shutdown outage

inspections showed 1427 indications. And from the transcripts, this is a significant jump in cracking.

Compare this to something like Beaver Valley. Beaver Valley has a very good inspection routine. They had to do chemical cleaning. They've been adding sleeves. And at Beaver Valley, the number of sleeves, my understanding is that the number of sleeves has surpassed Beaver Valley just in one inspection.

So I'm kind of concerned about the rate of increase. My first question is, are we outside of our operating experience? Or have there been other steam tube generators or steam generators that had this large increase in rate of cracking?

MR. KLEIN: I think that's a good question. So we have gone back and tried to benchmark the Palisades experience to other CE steam generators. And so St. Lucie Unit 2 had very similar models of steam generators to Palisades, almost the same number of tubes, not quite the same model but similar enough to provide a good benchmark.

Actually, it had more plugs and two consecutive outages compared to what Palisades would've had to plug had they plugged every two that had an indication instead of sleeved in this past

outage. So there is some precedent. However, if you trend the three or four outages leading up to where you had the big jump in the number of tubes that was plugged at St. Lucie 2, Palisades had a more dramatic jump compared to those two.

And so one of the questions we anticipate getting and it's hard to address is, was there cracking occurring at low temperature after the shutdown? Or was this all done at elevated temperature prior to the final shutdown before the extended shutdown? And I think that's something that we try to put a lot of thought into.

But it's hard to provide an answer to that with certainty. When you look at the distribution of cracks per elevation of Palisades, the highest number of cracks at support plates are at the two lowest elevations. So about 60 percent of the OESEC cracking in support plates is Hot Support No. 1 and Hot Support No. 2 which would be the highest temperature which would make you think that's typical of cracking and support plates.

SCC is temperature driven, and that would make you think it's more geared towards occurring and operating temperature. However, when you look at -- if you benchmark it compared to St. Lucie and you see

that there was a more rapid jump, that would be suggestive of maybe another factor in addition to operating temperature and cracking. We did try to go back and look through the literature for stress corrosion cracking of Alloy 600.

And there are a few papers that address low temperature cracking. Those papers, though, for example, there's a 2004 corrosion paper that talked about cracking in the presence of hydrogen that occurred at room temperature after the samples were removed from the autoclave.

And so that would indicate cracking could occur at a low temperature. But it's difficult to benchmark that to the Palisades operating conditions because you have a much higher stress sample in this case. Compared to a steam generate tube, it's at ambient temperature in the Palisades generator.

So there's things when you analyze them would suggest maybe there was a contribution from low temperature because it is known that they had uncontrolled chemistry. They had a lack of oxygen. The pH got lower in Steam Generator A. It ran out of hydrazine earlier in Steam Generator A, and there's more cracking in Steam Generator A.

So at the end of the day, I don't think we

can answer with certainty whether there was ambient temperature cracking as a contribution to the Palisades steam generators. We certainly can't rule it out.

MEMBER PALMTAG: But I'm more concerned about the rate. I mean, it doesn't sound like there's -- anywhere else had this high rate of --

MR. KLEIN: There's been a significant jump. And so if you're a tube integrity engineer and you're projecting an operational assessment forward, you have to account for that. So that's one of the things that needs to be done is that they will develop a model and that model will have to be benchmarked according to past experience which will now include the previous outages before, including the 2024 inspection with a huge jump in cracking.

MEMBER PALMTAG: But that's my concern. You're projecting, but we've never had this much crud just because the deferred maintenance.

MR. KLEIN: I might take exception to we've never had this much crud in steam generators because if you look at what Framatome is projecting might be removed by the chemical cleaning that's happening this month at Palisades, its less and has been removed from other generators in the fleet. And

so there's no doubt that the plant will benefit from a chemical cleaning. But I don't think in terms of crud that it's unprecedented condition.

MEMBER PALMTAG: It's a large chuck. But if it's not due to crud, do you think it was due to the low temperature?

MR. KLEIN: Well, I think the crud established the necessary conditions in order to have crevice chemistry, both at operating temperature and after shutdown that was quite different than the wall chemistry.

MEMBER PALMTAG: I just want to make a comment that I understand that this isn't Holtec. Holtec inherited these steam generators. It has nothing -- I'm not questioning their operations. I realize that this is something they inherited from the previous operators. So thank you.

MEMBER BIER: What consideration has been given to just replacing the steam generators? And if that's not being considered seriously, is it due to cost or schedule or sort of technical assessment that things will be fine enough going forward? What's the thinking on that?

MR. KLEIN: I think that's better addressed to Holtec. It's really -- our focus in

terms of a LAR that's under review is can sleeves be installed in the steam generator and they maintain tube integrity? And so that's been our focus. And I think the other part is commercial consideration. It's more appropriately addressed to licensing.

MEMBER BIER: Okay. That's fine. So from a safety perspective you're just looking at is sleeving going to be adequate?

MR. KLEIN: That's correct.

MEMBER BIER: Okay. Thank you.

MEMBER BALLINGER: This is Ron Ballinger, consultant. Does the staff have a criteria of which or beyond which you would recommend a mid-cycle outage?

MR. KLEIN: So I think that the important input to that is going to be the operational assessment coming out of the most recent inspection. And so when you look at the cracking and the other degradation that occurred that was measured by that inspection that they'll be able to take that and we'll need to account for it, as I mentioned. And so we anticipate that for cracking and supports, it'll be a probabilistic analysis that's performed because due to the probability detection of cracking at support plates, deterministically, it just wouldn't work.

So once you go through that probabilistic analysis, there's very specific criteria that would need to be met in order to demonstrate tube integrity. And if the analysis shows that that can't be met for a full cycle, then one of the options for the tube integrity engineer would be to shorten that cycle or time to inspection to that it allows less time for cracking till the next steam generator inspection.

MEMBER BALLINGER: Thank you.

MEMBER HARRINGTON: And this is Craig Harrington. Just to follow up on that, all of that is built into the guidance documents and the process that they routinely go through, correct?

MR. KLEIN: That's correct. We would not expect a process to change. It might be more difficult to model given the rapid step increase and degradation as was already mentioned here.

But the process itself should not change. And so if you do -- say you do probabilistic analysis for cracking at the support plates, you might run 10,000 or 15,000 cases. And then each one would plot the worst case burst pressure from that rod.

And then you have a cumulative distribution of those probabilities. And you go to the lower 95th percentile and you compare that first

pressure from the lower 95th to the 3 delta P criteria. And if it's greater than the 3 delta P, then that would be success. And if it's less, then you would need to adjust your analysis in order to make it meet the criteria.

MEMBER HARRINGTON: So it's less a matter of NRC recommending a mid-cycle outage or anything like that. It's more a matter of you're reviewing the work that they did in their operational assessment. And evaluating that against the criteria and then either accepting or pushing back if you feel like it's conservative or inappropriately done or they haven't made the reasonable assumptions and accounting for this unknown chemistry period and those kinds of considerations. Is that --

MR. KLEIN: That's exactly correct. So when we get a copy of the CMIA, the NRC won't be -- we typically aren't in the business of telling licensees how long to operate the generators until the next inspection. However, as part of the review of that CMIA, we'll be trying to look at the assumptions that go into that analysis relative to the last time that it was done, try to understand the differences. And if we have questions about changes that were made in those assumptions, then it would be appropriate to

have discussions so that we clearly understand what went into that and make sure that we're in agreement.

MEMBER HARRINGTON: This is Craig. What happens if you're not in agreement? What is the process after that?

MR. KLEIN: The process would be probably additional calls between the NRC and the licensee and their vendors. And till we reach a point where any misunderstandings are cleared up just at a disagreement between the conclusions, in that case then we would typically elevate that type of concerns to our management.

MEMBER HARRINGTON: So at that point, you would start using the processes established in the NRC to either convince us or order them. I guess ultimately you could order them to do a mid-cycle or some smaller outage based on wherever the agreement or disagreement came out.

MR. KLEIN: Yeah, and I don't expect us to get to that point. I think we have a very experienced vendor that's providing these services. I think the processes are pretty mature.

This is unusual circumstance maybe. But it's in everyone's best interest that tube integrity be maintained on whatever operating cycle comes out of

this current shutdown at Palisades. And so I would expect that we'd be able to resolving questions over time.

MEMBER HARRINGTON: Okay. Well, I think that the main point is that there is a process beyond this that you all maintain governance and oversight, just don't allow the licensee to go forward with whatever they say. There has to be either agreement or you all can take action and will take action if necessary.

MEMBER SUNSERI: This is Matt. I have maybe a question or two here. So I mean, once the plant goes into service, it's not as if they're just riding without headlights, right? There's a primary, secondary leakage monitoring requirement that should provide some early indication of degradation of the tubes, at least to the threshold which would require shutdown if it becomes to excessive. Is that fair?

MR. KLEIN: That is correct. There's radiation monitors on the main steam line. And you have condenser off gas analysis and steam generator blow down that all provide indications of a primary, secondary leak if you were to get it.

And that type of thing is trended over time to make sure there's no changes. And if there

are any changes, it would get immediate attention from the regulators. And there are well-established primary-secondary leakage guidelines in the industry that provide action levels and steps that are needed to be taken dependent on the amount and the rate of change of leakage.

MEMBER SUNSERI: Would you anticipate any, for lack of better words, I'll say increased scrutiny of that requirement by resident inspectors or something based on the conditions that we know exist with the generators?

MR. KLEIN: I would think that the -- our regional counterparts will be paying particular attention to that.

MEMBER SUNSERI: Thank you.

MEMBER MARTIN: This is Rob Martin. At our last meeting we had extensive public comment, genuine public comment. And first, the comments were less concerned of the science behind tube integrity and more of a consequence.

I think it'd be valuable for the public record for you to kind of step through consequence analysis and maybe provide some perspective with what appears in the safety analysis report and relative to maximum hypothetical accidents. I think this set you

up here. Steam generator tube ruptures have happened, right?

And of course, they did make local news. But it doesn't really go beyond that. But it's a serious event, but it's not, say, a Three Mile Island. Again, this is all on record. Kind of step through what happens when things break.

MR. KLEIN: I think as you mentioned before there has been a number of steam generator tube rupture events in the past. And of course, the goals is to never get there, right? So the whole tech spec program and the regulatory framework is to try to prevent that from ever happening again.

But it is an analyzed accident. And there are specific criteria for allowable dose at the boundary of the plant and also to the control room operators. And so a full guillotine break of a steam generator tube is analyzed and shown to meet all the criteria.

I think the most recent high profile tube leakage that we had was at SONGS. And the site boundary dose for that leak, it was not a tube rupture. It was a leak was negligible.

MEMBER MARTIN: Okay. So that wasn't exactly -- I mean, that's all valuable. And I know

you're probably not a Chapter 15 person, and --

MR. KLEIN: I'd like to phone a friend for my dose colleagues.

MEMBER MARTIN: Right, if you have some. But I mean, I could step through this for you. I think it's value for you to go through it or somebody from the staff to go through it if you have phone a friend.

Otherwise, I will go through it. But I think it's important because we did get a lot of public comment about worrying about consequences. And some perspective would be valuable here. Do you have a phone a friend here?

MR. KLEIN: I don't believe we do. So I'm not prepared to --

MEMBER MARTIN: Okay.

(Simultaneous speaking.)

MR. KLEIN: -- the accident analysis in detail.

MEMBER MARTIN: So the way it works out, of course, you get the event. And you may or may not get -- again, depending on the leakage rate, it may be a while before the safety systems can kick in, right? Leakage occurs.

There's some depressurization on the

primary side. At some point, you'll trip and you'll get some safety injection at which point the secondary will respond by isolating the event so that it's contained. But prior to that, there will be some release beyond the isolation valve.

And you get some release. In safety analysis, there are some very conservative multipliers that incorporated to the consecration of activation of fission products that are conservatively estimated. It's a source term going in there.

And typically, you're going to have a deterministic treatment of the timing of all these things. So you get a very conservative outcome. I did check a relatively recent -- there's no recent Palisades analysis.

But you're going to be under, at worst case, a rem, which of course is well within any safety limit. Now one of the things I think you need to talk just a little bit about last time related to the PRA and the status of the PRA. And one of the factors that might play into here is the estimates on the frequency of the event, right?

And you start shifting that, say, to the left meaning a more frequent event. Of course, if it happens, it happens once and then you're shut down for

three years or whatever it takes to get a new one. Or you're just shut down, period.

It's because it's more of an economic or investment protection kind of question. If they've made no plans on investment, that's not our purview here. That's their business.

But it's from the perspective of, say, relative to the worst case events, we're dealing with a large break LOCA, what have you, this is somewhere in the middle of the spectrum of things. And it would be nice to have a more mature risk profile. And there are certain methods that are really coming mature as we speak to do a better job here.

And maybe it's not the forum at this time to get a better perspective. But I do think there's a little bit of science there today that I think needs to be incorporated and mature that PRA model beyond just what was done before. I think it's harder because I think that is new for them to deal with a steam generator that is, you know, more sleeve-y than any steam generator before.

Throw in a little bit more data, right? So Beaver Valley, you mentioned that. Now I'm going back and looked at -- now I guess when they started their extensive sleeving, they were given a five-cycle

clock.

And that was in, like, 2018. And then here we are in 2025. I think that hits the five-cycle. And then we're going to take that out. There's going to be some inspection.

And I wonder if that is playing into any kind of decision making. Is it providing information, which can give insight risk and consequences and that all sort of stuff? I know the timing is a little funny because that's happening now. This is happening now. And whether they're crossing, you know, at other intersections right now, only you all can answer, and will that play into any decision-making down the road.

MR. JOHNSON: This is Andrew Johnson. So regarding Beaver Valley Unit 2, when they were initially approved for sleeving, that was more like the 2009, 2010 time frame. And they were initially approved for five years.

And they didn't install sleeves right away. They waited -- I think it was about one cycle or two cycles before they actually were going to install sleeves. And so once they did, right, they had already run out of a number of years.

So they came in with another amendment. We approved them to go five cycles since they hadn't

installed them originally. They did they five cycles, then they did additional testing in the interim while sleeves were installed.

They came in with a subsequent license amendment request where they requested to have permanent installation of the sleeves. And that was the one that was approved in 2018 that you referenced. So that's just kind of --

(Simultaneous speaking.)

MEMBER MARTIN: -- they had a pretty good experience with the sleeving process.

MR. JOHNSON: Yeah, they have not had significant issues that we're aware of. And they have been just steadily installing sleeves every outage as they need to. And I think just last year, they went over 1,000 sleeves in their generators.

MEMBER MARTIN: Okay. I didn't realize it was that high for Beaver Valley. It is kind of the order of magnitude we're talking about, maybe a little bit more than that with Palisades. But still that's not unprecedented to be in a balanced state.

MR. JOHNSON: And do you want to talk about preventative and corrective?

MR. KLEIN: Yeah, I did some numbers while you were speaking. So even though Palisades has

installed almost 3,000 sleeves, if you look at the number that actually went over cracks, it's just over 900. So the other 2,000 sleeves are what they call preventative sleeves.

And they were done as part of a strategy of running the unit because once you install a sleeve at the lower elevations which tends to be the hottest and the ones that had the most cracks, you can never put a sleeve at a higher elevation behind that because it won't fit through the tube anymore, right? So typically when they decided a sleeve of a tube, they would do up to the fifth elevation and then four, three, two, one in each tube. So the number of sleeves is very high, but the number of sleeves that are over a crack is about a third of that total.

MEMBER MARTIN: So how do you decide which extra 2,000 to do? I mean, it must've been some logic just based on temperature profiles.

MR. KLEIN: It's based on you know you're already sleeving a tube at a lower elevation. So you're going to try to do the most susceptible locations by temperature.

MEMBER MARTIN: So they have a good idea what's most susceptible based on, say --

(Simultaneous speaking.)

MR. KLEIN: Well, it's based on temperature, right? So you're hot leg, first few. And maybe we can show a schematic of the steam generator that might be helpful.

MEMBER HARRINGTON: Hello, this is Craig. Did they ever find any cracks about the fifth support plate?

MR. KLEIN: Yes. So if you go above the fifth support plate, I think there's maybe about one percent of the total at each of the other elevations, maybe not all the way down into the cold leg. But there were some cold leg cracks as well. But the sleeving is limited to the hot leg.

There's no sleeving on the cold leg because you need to inspect from the cold leg if you put sleeves into the hot leg. And so if you look on the screen here, the first elevation support on the hot leg is labeled don't go on to each. And so if you look at 01-H through 05-H on the left side here of the horizontal supports, those would be the locations of sleeves. And there's also in the lower rows, there's restrictions that are presented by the diagonal supports because you can't install a sleeve adjacent to a support, for example.

MEMBER HARRINGTON: Yeah, the sleeves

don't preclude the inspection. So it means there was cracking at some point above the fifth support plate on one of these sleeved tubes. You'll find it on the next inspection.

MR. KLEIN: Yes, so you would --

MEMBER HARRINGTON: You have to take action relative -- probably plugging I guess at that point if it exceeded the criteria.

MR. KLEIN: That's correct. So your tech specs would require you to inspect all active portions of the tube and sleeve tube assembly. So you might do from that uppermost sleeve to the end of the cold leg from the cold leg plenum where you can reach it. And then where you have sleeves into the hot leg at the highest elevation and down, you would need to use the sleeve probe that's a rotating probe but much slower than a bobbin probe, for example.

MEMBER HARRINGTON: I know you haven't seen the operational assessment yet. Do you sense, predict, guess they'll be doing 100 percent at a current or in a little while in future cycles?

MR. KLEIN: I would expect or ensure that the next refueling outage, if, for example, they meet a full cycle, it'd be 100 percent inspection of all tubing and sleeves.

MEMBER HARRINGTON: Okay. So we'll get a good trend on all the tubes.

CHAIR KIRCHNER: Could you repeat that? Also, that was a question I was going to ask. But looking ahead to this operational assessment, what's your estimate of what will come out of that? Can you summarize?

MR. KLEIN: I wouldn't want to speculate.

CHAIR KIRCHNER: You don't want to speculate?

MR. KLEIN: It could be a full cycle. It could be a partial cycle. At this point, the staff doesn't know and --

CHAIR KIRCHNER: Okay.

MR. KLEIN: -- we're expecting to get that answer later this month.

CHAIR KIRCHNER: When you say a full cycle, you mean the next refueling outage then?

MR. KLEIN: The full cycle would be the next refueling outage.

CHAIR KIRCHNER: Right.

MR. KLEIN: So for example, when this is --

CHAIR KIRCHNER: And that's in two years?

MR. KLEIN: -- just speculation, if they

may a year and a half fuel cycle with an inspection, I would anticipate 100 percent inspection at that point.

CHAIR KIRCHNER: I would too.

(Simultaneous speaking.)

MR. KLEIN: -- and it's not based on any knowledge.

CHAIR KIRCHNER: That's a reasonable anticipation given the extent of the years. Could you address for the Committee and the public what kind of startup testing is going to be done when they bring the plant back up online with the NRC's approval for the steam generators in particular? And second question related to that is there was, I think, on the order of, what, 300 tubes that were originally plugged that are being unplugged.

Now they were plugged for a purpose I would think initially. But now I'm just estimating or speculating that they want to recapture heat transfer area by unplugging those tubes. Is there any vibration or other concerns that you would be looking for in startup testing as a result of all the sleeving and also unplugging tubes that previously had been plugged initially?

VICE CHAIR HALNON: And Paul, to answer

that, April Nguyen is on Line 2. So if you need to phone a friend, she was pretty friendly.

MR. KLEIN: April, if you want to, I can address the unplugged tubes, if you want to speak more to the startup testing. But in terms of the tubes that were unplugged that were plugged prior to service, it was about 300 tubes in each steam generator that are above that central stay cylinder. You see the lower plenum of the steam generator.

So above that area, the tube bundle can be susceptible to high vibration. So prior to service, about 600 tubes total were plugged. Roughly, a little over 300 in each generator.

So they deplugged those and did eddy current to see if they could be returned to service. And of the roughly 300 in each generator, it looks like 139 in Steam Generator A and 136 in Steam Generator B were returned to service after eddy current. And the other ones were deemed not able to be returned to service where they were replugged, taken out of service again.

MEMBER HARRINGTON: This is Craig. Is there any expected difference or change in vibration behavior, water flow through the tubes as opposed to being plugged?

MR. KLEIN: So there may be some differences. I think we're comfortable that those tubes will be able to survive for a cycle until the next inspection. We all spent such a big differences that it would create unprecedented wear rates in one cycle.

I suspect that those tubes -- and this is speculation. But I would think they would remain in service for quite some time because what the vendor told us on the recent call that we had was it became pretty clear that there were certain zones in the low rows that were susceptible to vibration. And so that was readily detected by eddy current. And then once you moved outside those zones, the tubes looked to be in very good shape.

VICE CHAIR HALNON: April online, can you address the startup testing, the focus that the resident inspectors will have during pressurization of this plant and beyond?

MS. NGUYEN: Yeah. So good morning. This is April Nguyen. I'm the lead for the recert efforts in Region 3, and we have the primary responsibility of the inspection and oversight of the activities at Palisades.

So for the startup testing sequence, very

similar to coming out of a refueling outage. The site will be doing testing at normal operating pressures and temperatures. They're going to be look at system flow balances, right?

So especially related to the steam generators and to the feedwater systems that go there which I believe will help inform some of the questions about heat transfer capabilities, right, and balancing the two steam generators. And then also as they work through that sequence of going up in power, right, there'll be a variety of system tests that will be performed as they work through those different power levels as well to ensure that the systems are operating as expected. The resident inspectors do have a plan on how they're going to approach these activities by observing specific pieces that are of higher risk significance and then also then prioritizing the system restorations as the systems are needed to be brought back into service and verifying the operability of those components.

CHAIR KIRCHNER: So going back to Matt's question about leak rates and such. So would that be part of -- April, would that be part of the initial startup? Would you look for a leak rate from primary to secondary being within the tech specs which I

believe is on the order of, what, 50 gallons a day, something on that order?

MR. KLEIN: I believe there's a 150-gallon a day tech spec limit. There is a site administrative limit of 72 gallons a day.

CHAIR KIRCHNER: And if that were exceeded, what would happen next?

MS. NGUYEN: So this is April Nguyen again. So yes, as they raise, again, pressure, temperature, and also power levels, they will be more closely monitoring those leakage rates. Generally, in a sort of sequence, you do calculate those more often just to verify that there isn't some sort of inner system leak or other unknown source of leakage that's occurring. And if they do hit any of those limits, right, as required by the tech specs and the operating licensing basis, they would be required to either go down in mode or shut down, depending on what those values were. But as you mentioned, those generally are very small numbers on the order of 0.0-something gallons per minute, or as you all had it, in gallons per day.

CHAIR KIRCHNER: Thank you.

MEMBER MARTIN: Bob, one more thing here. Again, I think it's valuable for the record.

Palisades operates their hot leg temperatures lower, I guess characteristic of CE designs.

It's around 585 in the conversation I had with Ron earlier today. I'm looking at an IE report right now and talking a little bit about stress corrosion cracking, international experience. And of course noting that certainly for hotter plants, the rates of stress corrosion cracking, much, much, much higher.

I'd just note here that with can dos, again, still talking about Alloy 600 which operates typically in the range of what Palisades is. It really has been little observation. Is this consistent with your experience?

Are you bringing that experience, that particular detail into the assessment? Does that imply a de facto? I mean, I probably already have a tech spec for other reasons. But does that come into play when you're thinking about limiting their operation with regard to temperature and its influence in stress corrosion cracking?

MR. KLEIN: I think the primary place that will be considered is in the CMS because there are well-established Alloy 600 SCC growth rates in industry. And it is benchmarked to temperature. So

the lower you're T-hot, the more favorable for your plant.

MEMBER MARTIN: Is it something that you think is a basis for setting a tech spec on T-hot?

MR. KLEIN: No, the tech specs wouldn't really be set on T-hot. The tech specs we're seeing are set on maintaining to integrity.

MEMBER MARTIN: Right.

CHAIR KIRCHNER: And so the focus is meeting the performance criteria which is the structural integrity of the performance criteria that accident induced leakage performance criteria and the operational leakage criteria.

MEMBER MARTIN: Is it just a coincidence that looking at the CANDU experience a way Palisades has been operated the last however many cycles just happens to be right around the same temperature, low compared to rest of the PWR fleet.

MEMBER HARRINGTON: This is Craig. The temperature -- I'll say the temperature is going to be -- or the temperature effect is going to be factored into the evaluation assessment processes, guidelines that are established that all plants implement their steam generators. And you plug in your T-hot value, you go through that analysis.

And it accounts for the benefit or the detriment of your T-hot conditions. So it's not something that NRC on the back end would say, well, they've got a low T-hot. So we get more latitude.

It's built into the process. Now I don't have the data for this. But I would imagine in the instances where steam generator tube ruptures, these are hotter plants.

I mean, we, of course, have a general experience at Indian Point. And that's a Westinghouse BWR. Typically, T-hot is highest probably 610, 620, in that range. So you would expect them, the cracking rates would be higher than a case where you're dealing with 585 or less. Do you know the operating experience for other cases where they've been isolated to Westinghouse clients or higher T-hot plants?

MR. KLEIN: I think Member Harrington characterized it pretty well. They're well-established within the guidelines and integrity assessment -- I mean, integrity assessment guidelines, et cetera. There's a well-established Alloy 600 cracking growth rates.

And so within those guidelines, they're normalized at, I think, 611 temperature. And then there's an adjustment factor that's applied depending

on your site specific temperature. So if you're running at, say, 620 or 621 degrees, you need to multiply that growth rate.

If you're running at 583, you would reduce that growth rate. And so plants that run at higher T-hot have higher growth rates. And so they need to account for that in the operational assessment.

MEMBER HARRINGTON: This is Craig again. The T-hot, because of the growth rates, all things equal, a hotter plant might be more prone to faster cracking. But there's other factors, chemistry factors. And the other thing, it's not the only issue involved.

MEMBER MARTIN: Right. I mean, the phenomenon may be different on the primary side versus the secondary side, right, that sort of thing. But anyway, I think it's important to kind of get that out of the Palisades. At least it's kind of in a way they operate the plant. And it may be a better situation than, say, other plants.

MR. KLEIN: You'd asked about our experience with other units as well with temperature. And if you look at the Alloy 600 thermally treated tubing fleet which is about 16 units or so, the two units that have the most cracking within that fleet

are also the two units that operate at the highest temperature.

MEMBER PALMTAG: This is Scott. I guess I take a different twist on that. The Palisades plant operates at a lower temperature. I wouldn't expect lower crack rates.

But we're actually seeing the entire crack rates. Yes, so that kind of tells me that we are outside of our operating experience. This is something else that's going on that we're not comfortable with that.

I understand. The inspections sound great. I have all the confidence in the inspections and operational assessment. But if we're outside of our operating experience, we don't know what's driving these crack growth rates for this particular plant.

I'm worried that this operational assessment won't have the right rates. I think it would be helpful if we saw this operational assessment. If we saw the operational assessment, everything came back normal, everything is in the normal operating range, it would help give us confidence.

Everything is working well. If there's things came back that weren't, one suggestion I have

is to delay this until we can see the operational assessment. My understanding is the whole plant startup has been pushed back.

So I don't think we have a -- there's not a time constraint on this. And so some we can consider is maybe waiting. But we can discuss later.

VICE CHAIR HALNON: So this is Greg. I guess this question is to April. What is the present thinking of the restart pressurization for this plant?

MS. NGUYEN: Yes, this is April Nguyen again. Currently, the restart activities are scheduled to begin in the fourth quarter of this year, closer to the end of the calendar year 2025. And the restart sequence itself will be a longer process than what you would expect for a standard restart, right? It's going to be a little bit targeted to go to certain power levels, certain parts of the process, pause, do testing, et cetera, and then continue to work the way up slowly.

VICE CHAIR HALNON: Thank you. What is the time frame you expect when you receive the operational assessment that your review will be complete?

MR. KLEIN: Good question. I think it would be a matter of weeks but not many weeks.

CHAIR KIRCHNER: So is that still set for -- I think it was September 23rd was the date we were given.

MR. KLEIN: That's our understanding based on our last communication with the licensee.

CHAIR KIRCHNER: Thank you.

MEMBER HARRINGTON: Hello, this is Craig again. And maybe you guys have covered this before and I just don't remember. But it was roughly 3,000 tubes -- roughly 3,000 sleeves. How many tubes since there's multiple sleeves in many cases?

MR. KLEIN: I think we can provide that number.

MEMBER HARRINGTON: Okay, yeah.

MR. KLEIN: But if my math is correct here, it's about 732 tubes.

MEMBER HARRINGTON: So in one sense, it's maybe to Scott, your rate question. It's not so much that they went from no sleeves to 3,000 tubes being affected. It's a much less significant jump. That's still significant.

MEMBER PALMTAG: So my understanding, and Ron, please correct me if I'm wrong. But stress corrosion cracking, it starts -- there's some starting point of the stress corrosion cracking. You get the

small cracks. They're untenable.

Small cracks grow at some growth rate until they become detectable. A related concern is the rate of the growth and what's causing the rate of the growth. It seems like there's been a large increase from one inspection to the next inspection.

So it's really the rate that I'm concerned about. It's not so much the number of tubes. It's how fast are they growing. How fast are the undetected cracks growing. And then even how fast the detected cracks are growing too. There's two pieces of that.

MEMBER HARRINGTON: But I think that rate piece is less evident from all of the discussions that we've had since, correct me if I'm wrong. My sense is we're all looking at zero sleeves, now 3,000 sleeves, and kind of backing into a sudden increase in a rate of cracking. And there probably has been an increase in rate of cracking. But exactly what's driving that --

MEMBER PALMTAG: Yeah, and so the number I was looking, I came out from the September 21st was there was 56 stress corrosion indications at the previous inspection. And then at the latest inspection, there was 1,427. So I think that's sort

of we went from 56 to 1,427 over one session period.

CHAIR KIRCHNER: Post-layup.

MEMBER PALMTAG: Post-layup, right.

CHAIR KIRCHNER: Post-layup.

MEMBER HARRINGTON: Then, you know, stress corrosion cracking, also you're dealing with, back to Paul's discussion about hot growth, cold growth, those kinds of questions. It doesn't necessarily mean you continue at the same rate. If you've had odd circumstances, you could have a step change. And then it could slow down. It could speed up.

MEMBER SUNSERI: But wait a minute. So it's probably -- and let me check this again with your all's experience. But I mean, when you plug a tube, it's not because it's leaking. It's not because it's about ready to break. It's because it's not going to make it to the next expansion interval without being above the rejection criteria. So a lot of it -- a lot of the plugging is preemptive in nature.

MR. KLEIN: With stress corrosion cracking, it's plug-on-detection. So it's not related to structural significance at all. The cracks detected, it's taken out of service. And that's based on just the challenges of sizing a tight stress corrosion crack that's branching and very tiny and

trying to get the -- interpret the eddy current signal.

So I should probably emphasize that despite the high number of indications in cracks and there's no doubt it's a significant jump that the tubes all did maintain tube integrity. And that was established through analysis and also in situ pressure testing on the worst flaws of each steam generator. And so we said there's uncertainty about exactly when this change occurred.

And I think that's the appropriate way to characterize it. But moving forward, I would expect the growth rates to be more typical of Alloy 600. The chemistry had been established in the steam generators for quite some time now. We do know that they're doing chemical cleaning of steam generators that try and remove as much of the deposits of the support plates as possible. And so I think the conditions moving forward for the next operating period are going to be improved relative to what they were even for the last operating cycles.

MEMBER MARTIN: Take it a little bit different, still on the chemistry side. So -- and there's still this report up here. Some time ago, probably four years ago, did some work on zinc

injection which, I guess, apparently mitigates -- slows down the rate stress corrosion cracking.

That would seem, of course, a unique chemistry treatment for the specific purposes. Now is that coming into play? Is that a general practice? Is that something that --

MR. KLEIN: That might be a practice employed to slow down. I don't think it'd be relevant in this case. I think the biggest thing that they could do and that they are doing is to try to remove as much of the support plate deposits as possible because we know -- despite the uncertainty about the cracking, we know that that is a necessary condition.

And the more deposits you have, I believe the worst the condition inside that support plate. So I think that's the biggest change that they could make. And I don't think we'd want them crediting the type of additions in terms of change in the known Alloy 600 crack growth rates that are well-established and measured for decades at operating units.

MEMBER MARTIN: That data would already be biased by zinc injections. If that's been something that's been done --

(Simultaneous speaking.)

MR. KLEIN: I don't think so.

MEMBER HARRINGTON: This is Craig. Ron, zinc is --

(Simultaneous speaking.)

MEMBER HARRINGTON: -- initiation, right?

MEMBER SUNSERI: I think it's on the reactor coolant.

MEMBER HARRINGTON: And it's initiation.

MEMBER BALLINGER: Right.

MEMBER HARRINGTON: Not growth.

MEMBER BALLINGER: Right.

MEMBER HARRINGTON: Okay.

MEMBER BALLINGER: You've got to distinguish between ID and OD SCC. The bad stuff, you're talking about OD SCC.

MEMBER ROBERTS: I was wondering if you can comment on the reliance of Palisades on atmospheric dump valves. We got a public comment that was concerned about the fact that the safety analysis reports the consequence of about a rem if you had a tube rupture and you followed the plant's procedure to depressurize the atmospheric dump valves, which would trigger the emergency accident level. And presumably, have people start thinking about evacuating, it would potentially have public consequence.

Can you talk about how that affects the

evaluation? Does that affect it? Is your basis that you're going to have no measurable or no detectible increase in probability and frequency of tube rupture? Or is it part of the overall risk analysis? And because of the degradation, you've got a higher frequency, therefore, need to consider things like emergency action levels and the consequence of the event.

MR. KLEIN: I don't -- I guess in -- I'm not a dose person again. So I don't want to speak to accident analysis. I think that you have more potentially undetected cracks in the steam generator than you had before just because you had more detected cracks. However, I think the appropriate focus and the processes are in place to maintain tube integrity. So I would not say that you have a much higher risk coming out of the extended outage than you had before of a steam generator tube rupture.

MEMBER MARTIN: Okay. So your view is the risk profile had not changed?

MR. KLEIN: That's my view as a steam generator person. I understand the inspections that were employed during the extended outage and the in situ pressure testing that was done and the sleeving that was done. And so I think even though there's

greater uncertainty about what remains, there are mature processes that account for that and model for that. And that'll be one of the things that we're paying particular attention to in the operational assessment once it becomes available to the staff.

MEMBER MARTIN: Okay. Thank you.

VICE CHAIR HALNON: Any other questions? Doesn't mean you can't ask them later. So at this point, I'm going to ask for public comment. Because we're in Full Committee and we have a strict time, limit the comments to a total of 15 minutes.

And there's going to be two minutes per comment. If you hit the two-minute mark, then I'm going to ask the DFO to mute you. We're going to continue on. So get your points out succinctly. If a comment comes on off topic from Palisades restart, we'll mute you and go on to the next commenter.

CHAIR KIRCHNER: We should note for the record as well that there's a transcript with the extensive comments from the September 21st subcommittee meeting.

VICE CHAIR HALNON: Correct.

CHAIR KIRCHNER: So --

VICE CHAIR HALNON: We don't need to restate.

CHAIR KIRCHNER: -- we don't need to restate the same comments. Those are part of our record and can be found on our website.

VICE CHAIR HALNON: So the DFO and Larry Burkhart will call on commenters. And they'll start the time clock. So the way you do this is raise your hand on Teams.

I think it's star-5 for if you want to raise your hand on the -- if you're just on the phone. And Larry will take them in the order that they come in. And again, to reiterate, we'll stop at 15 minutes and limit you to 2 minutes. And if it's off topic, we'll mute. So go ahead, Larry.

MR. BURKHART: This is Larry Burkhart from the ACRS staff. Thank you, Vice Chairman Halnon. So yes, please, if you do have a question, raise your hand as I see you're doing already, star-5 if you're on the phone. I will take you sequentially. So with that, Mr. Blind, please provide your comments in two minutes.

MR. BLIND: Yeah, do you hear me okay?

MR. BURKHART: Yes, very, very well.

MR. BLIND: Thank you. Okay. Thank you to the ACRS. You obviously read my comments. As a reminder, I was the Vice President Nuclear at Con

Edison when we had the tube rupture.

And that too, I'd like to compare that to Palisades quickly. When we had that, we did not have an offsite release. I know there's some dispute on that, but we were able to contain the steam from the rapid depressurization to the condenser hot wells.

Unlike Palisades is -- and I would ask the Committee take a look at task interface agreement 2009-003. That came from a component design basis inspection in 2009 where the inspector questioned whether the atmospheric dump valves needed to be powered from offsite -- from the diesel generators. They are not.

And in there, you can see all of the analysis that goes back to the SCP pre-general design criterial of Palisades that they rely primarily on the atmospheric dump valves. And thank you for bringing that up. So this goes to the consequences.

There's so much discussion on the technical aspects of this. And it's all well done. But it has to be put in context of the consequences.

At Palisades, they will rely on the atmospheric dump valves for the rapid depressurization. And that's with or without offsite power. In fact, if they lose offsite power, they will

have to go to an open pressurizer power operator relief valves for the rapid depressurization.

In other words, they feed and bleed because the atmospheric dump valves are not powered from onsite sources. So the Committee must be informed by the consequences of this to the public.

MR. BURKHART: Thank you, Alan. Alan, that's two minutes. Thank you. Next.

MR. BLIND: Thank you.

MR. BURKHART: And you can send comments written to the DFO, myself, and Christopher Brown. Thank you, Mr. Blind. Okay. The next commenter is on the phone, 240-462-3216. Please hit star-6 to unmute yourself.

(Simultaneous speaking.)

MR. KAMPS: Hello. Can you hear me?

MR. BURKHART: Yes.

MR. KAMPS: Hi, this is Kevin Kamps with Beyond Nuclear and Don't Waste Michigan. I'm speaking to you from Kalamazoo which is 35 miles down from Palisades. I have one question.

And NRC staffer, I believe, mentioned heat treated Inconel 600. My understanding is that the current generators of Palisades are un-heat-treated which is either unique or very rare in the United

States. And I wondered what difference that makes in terms of your safety analyses.

And I just wanted to communicate to you all that I attended the FEMA meeting on August 5th near Palisades. And the local residents who attended from Palisades Park Country Club, from South Haven made it very clear that despite an intense period of activity out there involving FEMA, the state police, et cetera, they had no idea what to do during a general site emergency. They really just had no specific instructions.

Very late in the game here. So I'm kind of startled to learn of the one rem dose rate projections triggering a general site emergency if a steam generator tube burst at Palisades, let alone a cascading failure. So there's some real disconnects going on between the various approvals needed for this restart.

And so yet again, we encourage that this be slowed way down. And that question about replacing instead of repairing, back in 2006, the previous owner, Consumers Energy, testified to the Michigan Public Service Commission that these very steam generators needed to be replaced --

MR. BURKHART: Thank you, Kevin.

MR. KAMPS: -- in a short period of time. That was 20 years ago.

MR. BURKHART: Thank you. Thank you, Kevin. That's two minutes. I appreciate it. Please continue with your comments in writing back to the DFO.

VICE CHAIR HALNON: Yes, thank you, Mr. Kamps. Mr. Stein, please.

MR. STEIN: Hello, this is Dr. Adam Stein from the Breakthrough Institute. I appreciate the opportunity to make a comment today for the ACRS Committee. I appreciate the detailed look that the ACRS Committee is taking to this licensing action and the diligent work that the staff has done up to this point.

I followed this process since the beginning. Today I've heard that ACRS has some lack of confidence in what would result in certain growth rates of cracking and want to understand that better. I appreciate the ACRS wants to take a detailed look at that and I think they should do so with the time that is necessary for them to complete that accurately.

I think the staff has already done this work. But the ACRS should have confidence. But it's also challenging in this particular format of meeting

for ACRS to review that in detail and have a discussion on it.

There was a report mentioned that the ACRS Committee might want to review. And if they think that's necessary, I think they should. But I do not think that ultimately, although I'd like them to make their own conclusions, that they will find a mismatch of what the technical information says, what the staff has already recommended. Thank you for taking a look at this and providing confidence to the public.

MR. BURKHART: Thank you, Dr. Stein. The next commenter is on the phone, number 616-540-7027. Please hit star-6 to unmute yourself. Please proceed.

MR. SCHULTZ: Hello, can you hear me?

MR. BURKHART: Yes.

MR. SCHULTZ: My name is Kraig Schultz. I'm an environmental health advocate with Michigan Safe Energy Future. I live 50 miles from Palisades. What we are seeing here is not safety first.

It is safety first unless it costs too much. Holtec has the money. They're receiving hundreds of millions in federal and state support. But instead of using those funds to replace degraded steam generators, they chose the cheaper shortcut of sleeving.

Everyone here knows the risk. Sleeving means leaks, shutdowns, and public exposure. If the tubes fail in a cascade, the consequences could be catastrophic.

This is not about lack of resources. It's about priorities. Holtec is choosing to save money, not to safeguard the public. I urge the Committee to put that on the record. Palisades is being restarted on shortcuts, not on safety. Thank you.

MR. BURKHART: Thank you, Mr. Schultz. And the final commenter is Mr. Michael Keegan. Please unmute yourself and provide your comment.

MR. KEEGAN: Hello. Can you hear me?

MR. BURKHART: Yes.

MR. KEEGAN: Yes, thank you. Michael Keegan with Don't Waste Michigan. There were 600 tubes that were plugged and some of them got unplugged and some of them couldn't be unplugged. But my understanding, they were plugged because they were batwing vibration concerns, an eggshell lattice.

If you could speak to that. How are we going to avoid the batwing vibration again? And then I have a question on the inspections. Is this going to be done by sampling? Or is every single tube going to be inspected?

Because if you miss one, that could start it off. So with that, and I'm concerned that with all the sleeving, it's my understanding that for every 10 to 12 sleeves, you lose about one tube. And so you've got 3,000 sleevings going on. So you're losing about 300 tubes there.

What's the efficiency of the steam generator on that? I also have concerns that historically the steam generator was manufactured in the early '70s with Alloy 600 and not heat treated. It is an outlier.

So the remainder steam generator is known to be a faulty, a lesser alloy not heat treated, the only one that wasn't. So take that into consideration. Thank you for the meeting. Thank you.

MR. BURKHART: Thank you, Mr. Keegan, for your comments. Okay. I see Mr. Blind, you have your hand raised again. Is there anyone else who would like to make a comment? Okay. So Mr. Blind, please, two minutes. And then the Committee needs to go to deliberations. So Mr. Blind, please.

MR. BLIND: Yeah, so you hear me fine?

MR. BURKHART: Yes, sir.

MR. BLIND: Just one quick comment for the Committee. There was no discussion about the

condition evaluation. And I bring that up. A leak we know will be detected from the condenser off-gas.

But what about a tube rupture? And that's what we're concerned about. There was no discussion of the actual inspection results. There were, I think, about four tubes that had in the area of 1.7 inches longitudinal crack in length, 1.7 inches, not a pinhole and over 90 percent through-wall.

So there was no discussion of that by the staff. So I think that needs to be probed further because that seems like the structural integrity in the last operating cycle, we were very close to having a tube rupture in the last operating cycle from Entergy. That needs to be explored more. Thank you.

MR. BURKHART: Thank you, Mr. Blind. Anybody else who would like to make a comment? Okay. So Vice Chair Halnon, I'll turn it back to you.

VICE CHAIR HALNON: Thank you, Larry. At this time, we're going to close public comment and we're going to move into Committee deliberations at that point. The Committee will hold a discussion. I'm sorry. Could you hit -- you need to be muted.

MR. BURKHART: This is Larry Burkhart from the ACRS staff. Okay, very good. Thank you.

VICE CHAIR HALNON: So at this point,

we're going to take a break. It's 9:55. So we'll be back at 15 after 10:00 and we will continue with the Committee meeting. So at this point, we'll be in recess until 10:15.

(Whereupon, the above-entitled matter went off the record at 9:52 a.m. and resumed at 10:15 a.m.)

CHAIR KIRCHNER: Okay. We're back in session and we're taking up Palisades restart. And I'll turn it back to Greg Halnon.

VICE CHAIR HALNON: Thank you, Walt. At this point, there was a couple issues that we wanted to have Paul and Andy readdress or address. So if you guys -- I think you have the short list.

MR. KLEIN: Thank you. This is Paul Klein from the NRC staff. I think the one item that we wanted to address was a comment that there was a near steam generator tube rupture at Palisades during the last operating cycle.

And so in order to address that, we should talk a little bit about the process. It's performed when the tube inspection is done. So when the 2024 tube inspection was performed at Palisades, they go through condition monitoring process or CM.

And the goal of that condition monitoring is to demonstrate that tube integrity was maintained

up into the period till that inspection. And so as part of that process, you typically take your eddy current results and sort through those. And for some indications, you may find that more eddy current information is needed.

So more sophisticated flaw profiling may be done to provide additional crack dimensions that would then be put into analytical evaluation for that particular degradation mechanism to demonstrate tube integrity. If that can't be done analytically, the final step would be to do in situ pressure testing where an individual tube would be pumped up to elevated pressure in order to demonstrate or not demonstrate that tube integrity was maintained. So as part of that 2024 inspection subsequent to the eddy current date, they did in situ pressure test 17 tubes in Steam Generator A and 5 tubes in Steam Generator B.

That included the -- I think the tube that was referenced was the 1.79 inch crack. And so for that particular tube, the in situ pressure test requirements would've been three times the normal operating pressure differential between the primary and the secondary side plus some margin for gauge error and correction for ambient temperature testing versus elevated temperature material properties. And

so all of those in situ pressure tests passed.

There was no tube leakage and no tube rupture. So that would demonstrate that all tubes within the generator maintained tube integrity to that point. And subsequent to the in situ pressure testing, those 17 tubes will be plugged and taken out of service. They will no longer be in service moving forward and the five senior review as well.

VICE CHAIR HALNON: Thank you, Paul. Was there any other follow-up questions? Okay. So given the level of questions that we had in the -- I wouldn't say open items but conditions that we hope to return the operational assessment.

I'm going to suggest and I want just either a head nod or any comments that we wait and finish this letter in October, Full Committee, and have Paul and Andy come back for a little while and give us a short presentation on what the operational assessment has said, what the conditions of the operational assessment are setting for testing down the road and if any issues are with it going forward. Does anyone have any comments on that one way or the other, thumbs up, thumbs down? I see you got a green light.

MEMBER SUNSERI: Yeah, well, I'll just

comment. I don't support delay personally. Our purpose is to identify potential safety concerns with the operation or design of the plant.

I think we fulfilled that mission and we placed it into a governance process that I have a lot of confidence in. We'll look at the input we provided, look at the results from the condition assessment. There are people that will be involved with us that work with us every day. And I'm competent they'll make the right decisions on the operational assessment. And so therefore, us delaying does nothing but -- no benefit for us delaying because we will have nothing additional to add in my opinion.

VICE CHAIR HALNON: Thanks, Matt. Well said. Anybody else have an opinion one way or another? Scott, you're going to go the other way?

MEMBER PALMTAG: Yeah, I appreciate the comments. But I do think that we're kind of in uncharted territory. And the OA -- it sounds like the OA may answer a lot of questions.

So if the OA comes back, everything is within normal specs, that'd give me a lot more confidence. We are talking about steam generator tube rupture. So I do think there are safety consequences on it.

VICE CHAIR HALNON: Others? I'm going to call you out if you don't raise your hand.

MEMBER SUNSERI: I would just add, just part of my point is, is that even if we're not completely accurate with our assessment of these tubes going in, there is sufficient operational constraints like secondary leakage measurements, a primary system inventory requirements. You have tech specs that say that there's no pressure boundary leakage. There's tech specs for less than one gallon per minute, unidentified leakage, ten gallons per minute identified leakage, 0.1 gallon primary-secondary leakage.

These are all very tight criteria that will be extremely monitored. And it precludes the steam generator rupture or even a significant leak in my opinion. So I just think there's sufficient defense in depth even in addition to the inspection campaign that had been done to prevent an issue going in operation.

MEMBER HARRINGTON: This is Craig. I'm not opposed to delaying. But I'm kind of with Matt in the operational assessment is not going to have tremendous new insights.

It's a matter of the assumptions that they

invest in that document relative to the cracking that's occurred. And they have to inspect as I understand it the next outage regardless. It's just a question of whether that assessment will allow them to get to the next outage without taking a mid-cycle to do an earlier inspection.

And so really the operational assessment follows a well-established process to develop it. The wild card here is the assumptions that the utility makes about this uncertain period of chemistry really. And that's it.

There's well-established monitoring processes. It's a well-established evaluation process. And we're not going to change that process or provide some totally brilliant evaluation of it. It's already been well vetted and implemented across the fleet for a long time.

MEMBER MARTIN: This is Bob. I do think it's due diligence, completeness question on our part. We obviously haven't seen that. And obviously, there seems to be an opinion that everything will be fine.

But we've got to do our job. And I think our job involves vetting that ourselves and then drawing our own conclusions. So certainly, I support the delay.

VICE CHAIR HALNON: And that's great. It's two to two. Come on, Tom.

MEMBER ROBERTS: I agree with Scott and Bob on this that to me the open question I understand with the staff position is and I appreciate it is the assessment of the likelihood of the sudden rupture occurring between the inspections, between the start of operation whenever the first inspection is. And I think it's very incumbent on us to see that assessment. Because everything else seems to be clear. And I appreciate Matt's point about the monitoring and the -- as long as you don't have a sudden rupture. So I think it's useful for us to see what assessment the applicant does to show the likelihood of that is low enough to not change the risk profile.

PARTICIPANT: I would just say if the criteria is set out and Craig kind of said it. It's a very mature program. And the industry, the criteria that's set out is designed partially to prevent getting to that point.

So it's not like we're going into it blind to start up and see what happens. There has been extensive inspections already performed that every plant in the country does with the same criteria that

prevents and minimizes the risk of a rupture. That's the whole point of the program.

So to say that we don't take any credit for the industry program that has matured over the years, we're thinking there may be a rupture. I think it only comes back to what Scott was talking about earlier. What's the rate?

And that should be taken care of from an operational assessment perspective, not knowing that. And I assume that's one of the emphasis you'll be look at when you' look at the operational assessment is to ensure that the next inspection period takes into consideration the uncertainties of the rate of change in those tubes. Is that correct?

VICE CHAIR HALNON: I think that's correct. The focus of our review is going to be on how they model and the most recent inspection results moving forward since they have to be accounted for. And we know that the criteria will be a 95 percent probability with 50 percent confidence that the calculated lower 95th calculated burst pressure will be greater than 3 delta P. So we do know the acceptance criteria going into it.

And to point out we've talked a lot about Holtec. But Framatome is doing the inspections and

analysis. They'll be the authors of the operational assessment. And that is their business. I mean, they are good at it.

So it's not a first time member for Holtec. It's really Framatome is very experienced at doing it. So I just wanted to make sure that we don't put a question on the industry program and the maturity and the knowledge and science behind finding tube rupture -- I'm sorry, tube degradation. That is well-established in the industry from that perspective. Others? Dave?

MEMBER PETTI: No, I unfortunately didn't attend the subcommittee meeting. My concern is more if we go forward today and write the letter, will there be added confidence? Because I don't think that does us -- I don't think it's good for us at that point if, in fact, waiting a month would remove that.

I tend to agree with Matt and Craig. I mean, this is Framatome. This is their bread and butter. They know what they're doing. But at the same time, I'd hate to see a letter with that occurrence to the contrary.

CHAIR KIRCHNER: I just wanted to ask Paul, not to rescue us from our decision-making process, but clarification. So we're talking about

the EPRI guidelines. The operational assessment that we're talking about will be according to the EPRI guidelines for NEI 9706.

And if I understand the major thing that will come out of the operational assessment is a projected time through the next complete to integrity assessment. Is that the major thing? We already have the condition monitoring part of the equation that's been completed.

So that's been the inspection. And you just elaborated on what they found and what they did in the case of those 17 and 5 tubes. So if I understand it correctly, they'll project the condition of the tubes, the tube integrity to the time of the next scheduled inspection outage. And so what will we see from the operational assessment then? Is that projection of tube integrity over time integral and then an identified period for the next inspection?

MR. KLEIN: That's correct. So the operational assessment will take each active and potential degradation mechanism within the steam generators. And then on a per degradation mechanism basis demonstrate tube integrity to the next steam generator inspection, whatever that may be.

CHAIR KIRCHNER: And a typical integral

for that is in the fleet at large. It's usually the next refueling outage or can it be longer in some cases?

MR. KLEIN: It's very dependent on tubing material. So for the Alloy 600 high temperature yield that Palisades has, there's only two units, Palisades and Beaver Valley. Their most recent inspections going back probably 15 years or so have been at the next refueling outage. Tubes, steam generators that have the Alloy 600, thermally treated or Alloy 690 might go longer dependent on the particular condition of the steam generator.

CHAIR KIRCHNER: Right. Thank you for the clarification.

VICE CHAIR HALNON: Okay. So I guess you're looking to me to make a decision, right?

CHAIR KIRCHNER: Did we hear from Rob?

VICE CHAIR HALNON: Yes. Okay. So we're going to push forward with the letter. We're going to see if we can get the language that either captures the concerns. If we can't get to that language, then we will address it at that point.

But we've got the letter that reflects -- draft letter that reflects, I believe, concerns that we've raised. And given that the operational

assessment will provide a few pieces of information. What we just heard, it really will not necessarily make a decision on -- I mean, it's going to come out that the steam generator will stay safe on startup because either the interval will be very, very short of a cycle or it'll be 100 percent the next refueling outage.

Somewhere between zero and 18 months is where it's going to end up. And that's what we're going to learn. So sounds like we got all the data that we need from the standpoint of which tubes are which and what happened and which ones were sleeved and unplugged. And there's going to be a sufficient startup testing to make sure that all provides for operating the steam generator.

So at this point, we're going to go forward. I think at this point, we can ask the court reporter that she come back at 1:00. There's no need to be transcribing from here on the rest of his session.

(Whereupon, the above-entitled matter went off the record at 10:33 a.m. and resumed at 1:02 p.m.)

CHAIR KIRCHNER: Good afternoon. The meeting will now come to order. This is a continuation of the first day of the 728th meeting of

the Advisory Committee on Reactor Safeguards, ACRS. I am Walt Kirchner, Chairman of the ACRS.

ACRS members in attendance in person are Vicki Bier, Gregory Halnon, Craig Harrington, Robert Martin, Scott Palmtag, David Petti, Thomas Roberts, and Matthew Sunseri. ACRS member Vesna Dimitrijevic is participating virtually via Teams. ACRS Consultant Ron Ballinger is here participating in person and I believe our consultant Dennis Bley participating remotely.

If I have missed anyone, either ACRS members or consultants, please speak up now.

Hearing none. Weidong Wang of the ACRS staff is the Designated Federal Officer for this afternoon's Full Committee meeting. And I know that we have a quorum.

The ACRS was established by the Atomic Energy Act and is governed by the Federal Advisory Committee Act as well. Under the Atomic Energy Act, ACRS shall advise the Nuclear Regulatory Commission on hazards of proposed and existing reactor facilities and the adequacy of proposed safety standards. Following Executive Order 14300, the Committee has narrowed its focus to only those activities necessary to fulfill its statutory obligations.

As a result, ACRS is prioritizing the review and reporting of new reactor facilities and proposed safety standards, with particular attention to issues that are unique, novel, and noteworthy. The Committee will consider nuclear safety matters as referred to by the Commission.

Please note the ARCS speaks only through its published letter reports. All member comments should be regarded as only the individual opinion of that member and not a Committee position. Information about the ACRS, such as letters, meeting rules, and transcripts are on the NRC public website and can be found by searching for About Us ACRS on the NRC home page.

The ACRS provides an opportunity for public input and comment during our proceedings. Please note that portions of this meeting may be recorded for internal purposes. For this Full Committee meeting, we have received no written statements. We have one, yeah, from Dr. Saouma. So we'll get to that later in the meeting. Other written statements may be forwarded to today's Designated Federal Officer. And we have also set aside time at the end of this meeting for public comments.

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 on Teams. If you are participating by phone, press star-6 to mute your phone and star-5 to raise your hand on Teams.

The Teams chat features only for communicating IT issues and briefing logistics. So please do not use it for comments or questions on the topics under Committee discussion or deliberation. For everyone in the room, please put your electric devices in silent mode, and mute your laptop microphone and speakers. In addition, please keep sidebar discussions in the room to a minimum since the ceiling microphones are live.

And then we'll remind our presenters that your table microphones are unidirectional. So you'll need to speak into the front of the microphone to be heard and recorded online. Finally, if you have any feedback for the ACRS about today's meeting, please fill out the public meeting feedback form on the NRC's website.

This afternoon, we are going to consider a wrap-up of current ACRS activities on the Seabrook

alkali-silica reaction topic. And so if there are no comments from members, I'm going to pass our deliberations over to Greg Halnon, subcommittee chair of our fine operations committee. Greg?

VICE CHAIR HALNON: Thank you, Bob. Good afternoon. We're here to get information regarding the issue of the alkali-silica reaction at Seabrook Nuclear Plant.

We have been following this issue for several years as a committee and have received numerous presentations and have received much information regarding the issue, including a plant tour and the discussion with the NextEra staff on the progression and mitigation of the issue. We've also had significant meetings for public engagement, most notably the C-10 organization. At the last meeting that we had this topic on this for discussion, it'd be about time for Dr. Saouma who's representing C-10 to present his views on testing performed at NIST.

This presentation resulted in an open question to the Committee on whether or not the previous conclusions that we had made were still valid. That is I'm going to paraphrase that the testing, called a large-scale testing program undertaken by licensee was sufficiently representative

of the site to ensure that programs in place to manage the ASR are sound. We're going to focus on this question today.

And I'd encourage any questions that you guys have. You should raise them up. Hopefully, we've got the right people here to answer them. So we've asked the staff for a presentation along with their expert opinion and assessment of this question. So with that, I'll turn it over to the staff and we're all ears. Thank you.

DR. THOMAS: Thank you, Greg, and good afternoon, ACRS members. My name is George Thomas. I'm a senior civil structural engineer in the Office of Nuclear Reactor Regulation, Division of Engineering and External Hazards. Also joining me on the table are my colleagues.

MR. MANOLY: Kamal Manoly, Senior Technical Advisor for Structural Mechanics, Division Engineering and External Hazards at NRR.

MR. TSENG: Ian Tseng, Chief of Civil, Structural, and Geotechnical Engineering Branch in NRR.

MR. COOKE: Andrew Cooke. I'm also in structural engineering.

CHAIR KIRCHNER: Pull those microphones

closer to you. It will help the court reporter.

DR. THOMAS: So I'm here to discuss an open issue from previous ACRS Full Committee in May with regard to findings from the ASR research sponsored by NRC at the National Institute of Standards and Technology. And specifically the Task 3 study as it relates to addressing an ASR issue at Seabrook. And then the NIST research we're going to talk about here is the Task 3 study which is documented in Technical Note 2180 entitled Assessing Cyclic Performance of ASR-Affected Concrete Shear Walls.

And this report is publicly available at the link on my slide, and it's reflective of the NIST report. Task 3 involved in-plane cycle lateral loading tests under constant axial load of three ASR-affected and one control wall specimen. The NIST study was generic ASR and not specific to Seabrook.

The large scale test program conducted by NextEra or LSTP conducted by NextEra was Seabrook-specific. I'll discuss the relevance of the NIST study to Seabrook structural safety with regard to ASR in three areas. One is relevance of the NIST test configuration and results in regard to representativeness to Seabrook. The second is

relevance of the NIST tests to in-plane shear capacity at Seabrook. And the relevance of NIST tests to past and pre-instrument expansion estimate.

MEMBER PALMTAG: This is Scott Palmtag. And I just had a question on this, kind of a high level question. But NIST -- to put this in perspective, in NIST, they're studying ASR. Is this related to NRC or is the NRC communicating with them? Or is this something they're doing independently?

DR. THOMAS: Well, it was NRC sponsored in response to a user need request.

MEMBER PALMTAG: So this was NRC sponsored. So you're very familiar with what they were doing?

DR. THOMAS: Yeah.

MEMBER PALMTAG: Okay.

DR. THOMAS: And it was connected between 2014 and 2021.

MEMBER PALMTAG: Okay. That's good to know that all their tests were NRC sponsored. So you're aware of the details. So thank you.

DR. THOMAS: So I start off with some background regarding Seabrook criteria for ASR-affected structures. So the Seabrook structures, concrete structures there other than containment, a

design to ACI 318-71 and supplemented by the LSTP. The containment is, the code of record is the ASME Section 8, Division 2. And for that, the ACI 318, the acceptance criteria is the nominal capacity of the design loads, a capacity reduction factor to be greater or equal to load factor times the demand, including the ASR, the design loads for applicable limit states.

So the large scale test program from a technical basis only for the capacity side of the equation. And it's valid within the levels of expansion achieved in the large scale test program. The expansion limits are monitored using the ASR monitoring program which is a subset of the Seabrook structural monitoring.

The second thing that LSTP formed the basis was for the monitoring methods used. So the monitoring methods used to monitor in-plane expansion in two directions and through-thickness expansion in one third direction. The methods used are the same as those used in the large scale test program which is for in-plane expansion.

It's crack indexing and pin-to-pin measurements using division calipers. And to measure through-thickness expansion, it's use of extensometer

that is installed in specific locations.

So when an extensometer is installed, it gives you the expansion only from the point of installation of the extensometer. And therefore, there needs to be a method to estimate the expansion to date until the instrument was installed. And the large-scale program developed a method to do it.

The demand side of the equation is determined by structure-specific structural analysis on the design loads, which includes ASR and load combinations. And the analysis determines structure-specific pressure limits monitored and accounts for future ASR expansion. This is monitored using the Building Deformation Program, which is another subset of the site -- the plans, such as monitoring.

Now the distribution of the force components, such as axial or membrane, flexure, out-of-plane shear, in-plane shear. And so the structure analysis is performed. And the first component that's checked against applicable acceptance criteria in the ACI 318 Code.

Now, moving the first area, which is the limits of the test configuration and results to Seabrook. This figure is a schematic. It's taken out of the NIST report. It shows the effect of all aspect

ratios, which is height and length, and predominant behavior and failure mode. The lateral loading suggests (inaudible). And as we can see, as the height of our aspect ratio moves from lower amounts, lower value to higher, the dominant behavior tends from shear to diagonal tension to flexure, and you may have something intermediate in between. So our technical and nuclear power plant shear walls aspect ratio is one of the tests.

Now, in this test, ratio walls specimens had a wall height to length ratio of two. And therefore, (inaudible) and relatively low reinforcement ratio, approximately .31 percent. And the observed failure mode was in flexure or bending and not in plane. As I mentioned --

VICE CHAIR HALNON: This is Greg. This raises the question. What were the LSTP? What was their ratio?

DR. THOMAS: The reinforcement ratio?

VICE CHAIR HALNON: Well, you had this -- back one slide, the aspect ratio. You had that as the NIST test. It's the blue dot area, I assume.

DR. THOMAS: Yeah.

VICE CHAIR HALNON: Where was the LSTP test?

DR. THOMAS: So, the LSTP did not include a shear wall test for in-plane shear.

VICE CHAIR HALNON: (Inaudible.) Okay. Go on. It's fine.

DR. THOMAS: As I mentioned, typical nuclear power plant structural walls, including Seabrook, have low aspect ratio. They are of 1 or less. They have significantly larger reinforcement ratio. And the expected failure mode is diagonal shear cracking or diagonal tension.

So the NIST test specimens were not representative of Seabrook structural walls and test results do not apply to Seabrook. Nevertheless, even for the observed flexural failure mode, the appropriate comparison is between the measured flexural capacity, M'-max, to the nominal capacity, Mn, which is calculated using coefficients. And observed that all shear walls had this ratio of greater than 1. And so what it means is the tested walls, they reached the code, nominal ultimate flexural capacity, with some margin, although the margin was lesser than for non-ASR wall.

So the NIST test showed no reduction in maximum observing plane capacity compared to code nominal capacity. It doesn't pose any contradiction

with the LSTP out-of-plane flexural test. And the NIST test also made a comparison of yield moment to the nominal code moment capacity and the results were less than 1.

However, it's the staff's opinion that it is not an apple-to-apple comparison because the nominal moment capacity is calculating the code based on concrete reaching its maximum strain, or crushing strain, of 0.003. And for a tension control design, at that point, the steel is well beyond yield, whereas the yield moment in the NIST test were based on onset of yielding and the extreme reverse. So actual nominal capacities calculated based on rebar strain being well over the yield.

MEMBER HARRINGTON: George, this is Craig. Just to clarify in my mind, I guess, the terminology. So the structural walls, shear wall in these pieces, does that include the containment building wall itself? Or is that other walls within the structure? Because I'm not seeing the aspect ratio of a containment wall being anywhere near 1. So, help me understand that.

DR. THOMAS: So in containment wall, the aspect ration, they are 1 or less.

MEMBER HARRINGTON: They're very tall and

very slender relative to their height. So what am I missing?

DR. THOMAS: No, I mean, the diameter is specific.

MEMBER HARRINGTON: So, in that case, you take the entire diameter, not just the thickness of the wall alone?

DR. THOMAS: Right.

(Simultaneous speaking.)

MEMBER HARRINGTON: Right, okay. So it's taking account --

DR. THOMAS: And the height is up to the strain level.

MEMBER HARRINGTON: Yeah, yeah. Not just the thickness of the wall versus the height in a local area. It's the entire structure.

MR. KOCH: It's the length of the wall, not the thickness.

(Simultaneous speaking.)

MEMBER HARRINGTON: That helps. Thanks.

MEMBER ROBERTS: Hey, George, just wondering. Given this was an NRC-sponsored test, why wasn't an aspect ratio more reflective of Seabrook chosen?

DR. THOMAS: So, that was the original

intent. And the staff's plate tests were collected late in the research period. It went into during COVID. So, when they came to design experiments, they found that if you only test a shorter aspect ratio it requires more load, higher capacity of the equipment to apply the load. And that was very reasonable thickness and size of the wall. And the lab capabilities did not have sufficient lab equipment to conduct higher load tests. I think that was the reason they went with a wall that's already published in literature and they had some results on it and determined that it was within the load capabilities of the lab.

MEMBER ROBERTS: Okay. Thank you.

CHAIR KIRCHNER: George, could you go back? So, you make a strong conclusion there in that previous slide, the next one, that these test specimens were not representative of Seabrook's structural walls. Is that because they didn't have the larger reinforcement ratio or --

DR. THOMAS: In respect to both the --

(Simultaneous speaking.)

CHAIR KIRCHNER: I'm having problems with the aspect ratio being the dominant consideration here. As Craig said, you've got a tall cylindrical

vessel. These are much larger structures than any of the test specimens.

DR. THOMAS: It's both the aspect ratio as well as the reinforcement.

CHAIR KIRCHNER: Go ahead.

DR. THOMAS: It's both the aspect ratio as well as the reinforcement.

(Simultaneous speaking.)

MEMBER HARRINGTON: If you test it this way, it's going to have a different failure mode.

CHAIR KIRCHNER: No, no. I get that, right, right. But I think the samples in this test aren't near anything representative of the containment wall structure, are they? I mean, they're reinforced concrete, but not the same reinforcement. I'm just trying to understand how you say -- you end that second bullet with test results do not apply to Seabrook.

Is it primarily the aspect ratio or the reinforcement? Because you've highlighted the aspect ratio. It seems to me that the really different and more important thing here is the reinforcement. But, explain.

DR. THOMAS: Yeah, it's both. So, nuclear power plants are often subject to lateral loading or

in-plane shear. The failure mode is the in-plane shear, the diagonal cracking. In this case, it was by flexure. The lower the reinforcement ratio, the structure is likely to have more flexure. And the higher the aspect ratio, the failure mode is likely more to flexure.

CHAIR KIRCHNER: Makes sense to you, Craig?

(Laughter.)

CHAIR KIRCHNER: Okay. Go on, George.

MR. KOCH: This is Patrick. Just if this clarifies things. So what the aspect ratio was talking about there is, like, that drives what the failure mechanism is. And the failure mechanism in the NIST test is not the failure mechanism you would expect in Seabrook structures. So that's why that's an important point, that the aspect ratio of the test is not representative.

CHAIR KIRCHNER: Right, okay.

DR. THOMAS: And then the LSTP test specimens, they were not conventional beam specimens. They were a slice of the representative reference location of a Seabrook structural wall, with two-dimensional reinforcement on each face. And that provides both horizontal and vertical or biaxial

confinement to ASR expansion. And there was no through-thickness reinforcement.

For the load test, the vertical wall slice was oriented horizontally, the reinforcement layers on the top and bottom faces. The load was applied normal to the top face. Essentially, the specimen was a full-scale slice of the reference location.

MEMBER PALMTAG: This is Scott again. I'm having trouble visualizing. What does that mean? It was the -- you have two rebars inside the wall, right?

DR. THOMAS: Yeah, two layers, one on each face.

MEMBER PALMTAG: So it was oriented --

DR. THOMAS: Typically, the wall is vertical with the reinforcement on the outer face. But on our test it was rotated.

(Simultaneous speaking.)

MEMBER PALMTAG: It was rotated and then the force was down?

DR. THOMAS: Yes.

VICE CHAIR HALNON: George, I read somewhere that -- and I don't know where, but there's a delamination in the test that caused -- did that cause any results issues with the delamination?

DR. THOMAS: So, there was some edge

cracking around the specimens. So these specimens do not have through-wall reinforcement. Therefore, in the allegory to ASR expansion, the edges are the weak link areas of least resistance. So the cracking initiated there. But it was really localized. I mean, it was around the specimen, but it didn't propagate through the specimen.

(Simultaneous speaking.)

MR. THOMAS: Yeah. It was only a few inches in two. And that was determined by, after the tests, some of the specimen, the cross-section was cut and examined to --

(Simultaneous speaking.)

VICE CHAIR HALNON: Okay, thanks.

DR. THOMAS: So the LSTP was specific to Seabrook and representative of a typical Seabrook wall configuration, and addressed some of the critical limit states for that configuration without through-thickness reinforcement. And those were flexure and reinforcement anchorage, studying the effect of ASR on flexure as well as the bond between rebar and concrete. Because the maximum moment was applied at rebar splice, the lab splice.

And then a study of shear and effects on anchor bolts and capacity, as well as an instrument

study which determined which was the best instrument to be installed for through-thickness expansion measurement. And both the specimens were almost full-scale of the reference location.

Regardless, the LSTP provided the technical basis as well as limitations with regard to expansion limits and continued applicability of the ACI 318-71 of ASME III-2 codes of record for the ASR-affected structures at Seabrook. As I said, the LSTP did not include in-plane shear tests.

And I'll explain why. Now, Seabrook structures are subject to design basis loads and load combinations, which are defined in the UFSAR, and the analysis conducted on a structure-by-structure basis. So if it's a cylindrical structure, the analysis models the cylindrical shape.

And these walls have physical configuration and bounding conditions or loading conditions that also result in out-of-plane or radial shear forces, out-of-plane moments, in addition to membrane and in-plane shear forces. So one or more element force components may dominate the response over the others. And that falls out of the structural analysis.

Now, elemental section, design checks are

made for each limit state based on the analysis response results, as well as applicable interactions between the forces. While relevant and evaluated seismic load combination, due to relatively large margins for the Seabrook structures, in-plane shear forces typically do not control.

As I mentioned, the physical configurations are bounding conditions resulting in out-of-plane shears and moments. Examples of those are seismic. Most of these structures are, a significant portion of them, were built below grade level and are subject to hydrostatic forces, such as lateral loading. They are subject to -- many of them are -- most of them have concrete infill surrounding them, which are also undergoing ASR expansion. So they exert a lateral force. So the distribution of the force on the design loads falls out from the structural analysis.

The Seabrook structure walls, for in-plane shear, they have (inaudible) lot of reinforcement on each case. That is resistant in-plane shear in addition to contribution from the concrete. But the containment building structure has a layer of orthogonal, diagonal reinforcement in addition to the (inaudible) directions. And those are designed

specifically to the seismic in-plane shear forces. The ASME Section III-2 code requires the in-plane shear capacity of concrete to be zero.

MEMBER PALMTAG: This is Scott again. I'm sure you've already got them a long ways, but the in-shear, if I understand it, it would be the equivalent of, like, if you had a containment kind of twisting the in-shear. Is that what you're talking about here?

DR. THOMAS: Yeah, it's the tangential.

MEMBER PALMTAG: Right.

DR. THOMAS: (Inaudible.)

MEMBER PALMTAG: Speak into the microphone, please. It's kind of like a twisting. Would that cause in-shear -- in-plane shear?

DR. THOMAS: In-plane shear, yes.

MEMBER PALMTAG: And then what you're saying that's also -- it's not just concrete. It's also reinforced by additional reinforcements to prevent that?

DR. THOMAS: Yeah, reinforcement designed to resisting plane shear.

MR. BURKHART: You need to speak into the microphone for the court reporter.

MEMBER PALMTAG: So going back to the LBNT test. Those were aligned correctly for the forces you

would have on an earthquake, right? Because the earthquake wouldn't be -- I'm using my own words -- twisting. It'd be more swaying of the whole building, right?

DR. THOMAS: So the in-plane shear tests were not done in the last gain test.

MEMBER PALMTAG: Right. Because they're not forces you would expect.

DR. THOMAS: Because there's reinforcement available to resist it. And reinforcement provides confinement, the ASR expansion. So effective ASR is reduced.

MEMBER PALMTAG: So this supports that the LBNT was the correct orientation because you didn't have to look at the other orientation because of these additional reinforcements?

DR. THOMAS: Right. So in the out of plane direction, there's no reinforcement resisting the shear force. And you can expect more available.

MEMBER PALMTAG: And that's the one that was tested?

DR. THOMAS: Yeah.

MEMBER PALMTAG: LBNT. All right. Thank you.

DR. THOMAS: So in-plane shear failure

mode is expected to be more ductile because of reinforcement available versus non-ductile out of plane shear failure. That's resisted primarily by concrete. And this is from the Seabrook configuration which had no through-thickness.

So there's also corroborating evidence from experimental work cited to here, Habibi et al. at the University of Toronto that was done by Kojima Corporation in Japan. These tests lateral cyclic loading tests simulating great loads of low aspect ratio ASR walls. So the Habibi et al. had an aspect ratio of 0.71 and the reinforcement ratio of 0.8 percent. So an aspect ratio of 0.83 and reinforcement ration of 1 percent. These are the data range of plant walls.

And these experience show that ultimate in-plane shear capacity of the test walls would not adversely affect ASR. The observed failure mode was diagnosed as cracking and rebar yielding and not flexure. The staff in our safety evaluation for the license amendment use the evidence from Habibi et al., an earlier publication of the same one that support our conclusions.

So based on these, there is reasonable assurance that Seabrook's structure walls remain

capable of resisting design-basis lateral seismic loads by in-plane. And yes, they call it in-plane shear and cylindrical tangential shear. Just to give some perspective, the containment is the most significant.

And a lot of the reasonably sized model test conducted by Sandia National Labs. It's documented in NUREG-CR-6906 which is titled An Overview of Containment Integrity Research. So these containment structures were tested. There were sealed containments, concrete containments, resisted concrete containment.

And there was a test that included reinforced concrete containment. One of these models is similar to Seabrook. And under accident conditions, the failure mode for the containment was significant cracking and diagonal tension resulted in functional failure between leakage and not catastrophic structural failure.

And the functional failure occurred at pressures that were of the order of three times the design pressure. So that indicates there's significant margin in these designs. Also, there's another NUREG-CR-6707 which discusses shake table tests, seismic shaking of several earthquake records.

And the tests show that although this damage that accumulates, the catastrophic failure of the containment that occurs. Again, these containments have significant seismic load carrying capacity.

The third area is the relevance of the NIST test to past expansion estimate. Like I said, when -- from the time it's installed. So there needs to be an approach to estimate the relevance of past expansion. And in the last scale test program NextEra developed the method which is based on the relationship between normalized modulus and the functional expansion.

The normalized modulus is the elastic modulus at the time of installation of the extensometer divided by the elastic modulus at time zero. So in this approach, the Et, which is the modulus at the time of installation, was determined by testing cores at the time of extensometer installation, however, E naught (phonetic), there was no modulus test results available from original construction. They only tested the concrete compressive strength.

So to determine E naught, in an empirical equation in the code which is Ec equals 57,000 square

root of fc prime, where fc prime is the compressive strength. So this equation is really used, the modulus 57,000 square root of fc prime is only used at the time zero, at which time there was no ASR degradation in the concrete.

The elastic modulus empirical equation is not used for determining concrete modulus of elasticity (Et) of ASR-affected concrete. It's, like I said, it's determined directly by testing of cores. Since there is no ASR degradation at the time of construction, the use of this modulus empirical equation is reasonable and justified.

So there is variability associated with the empirical equation. So that's accounted for by using a reduction factor on the normalized modulus in the modulus-expansion correlation. Now, the modulus-expansion correlation, the higher the value of normalized modulus, the smaller the expansion. The lower the value of the normalized modulus, the higher the expansion.

So in the event the empirical modulus or E naught over-predicts the original elastic modulus, which means the denominator is higher than actual, that gives you a lower normalized modulus which means higher expansion. Likewise, the other way, if the

empirical equation under-predicts the original modulus, you get a larger value of normalized modulus, which means smaller expansion. And the reduction factor adds conservatism to account for the variability.

Regarding the empirical ACI equation, the NIST report states that the trend indicates that the compressive modulus of the reactive concrete degraded faster with ASR expansion than did the concrete's compressive strength. So it's the elastic modulus material property that degrades more, degrades faster than compressive strength. So the empirical modulus equation does not apply to the ASR-affected concrete.

And that observation is consistent with the data from the large-scale test program too. As I said before, the empirical modulus equation is not used for ASR-affected concrete in the LSTP methodology. So the NIST findings do not invalidate the modulus-expansion correlation used at Seabrook to calculate the expansion to-date at the time of extensometer installation.

In conclusion, the NIST Task 3 wall test specimens are not representative of Seabrook structural walls, and so the test results do not apply. The NIST Task 3 Study does not refute the

overall conclusions of the Seabrook LSTP or the License Amendment 159.

And the NRC will continue to inspect Seabrook's performance in the management of ASR under the reactor oversight process. And this includes the licensee's actions and compliance with the six license conditions associated with the ASR issue at Seabrook. And these license conditions are all intended to confirm the continued applicability of the LSTP conclusions to Seabrook's structures. That concludes my presentation.

VICE CHAIR HALNON: Thank you, George. Questions or comments, thoughts from the Committee?

Under the reactor oversight process, I would put the word augmented in front of that just because it's a focus.

I mean, it's a long term equipment issue at Seabrook. And I know that plants that have long-term equipment issues with the resident inspectors and the regional folks are focused on that when they get to the site. It's not just when we get to it next on our inspection schedule, we'll inspect it. But they look at -- the residents, I see most of the resident reports have some mention of --

DR. THOMAS: So it's been a PI&R, Problem

Identification and Resolution sample. So thus far, we have been doing two in a year.

VICE CHAIR HALNON: So I have just one question and I'm going to look to my impartial consultant, Ron, Dr. Ballinger. With all the technical information and the code information that's bouncing around, is there room in this topic for professional opinions that are different and still result in an acceptable approach to managing ASR? Because as you know, we have a pretty big divide between opinions of experts in this area. Is there room for that difference of opinion and still say that we can come out with an adequate, safe program?

MEMBER BALLINGER: You're asking a question.

VICE CHAIR HALNON: I'm asking you the question as a --

MEMBER BALLINGER: There always is. You look at their last slide.

PARTICIPANT: Ron, can you get closer to the microphone?

MEMBER BALLINGER: You look at their last slide, the last bullet, that's really the key.

VICE CHAIR HALNON: We're not just waiting it out, leaving it alone --

MEMBER BALLINGER: All these codes, by the way, have factors of safety built into them already. So there's a lot of margin that's in here. And as long as they monitor what's going on, I think everything -- you can say in theory that this is bad. But they'll know in practice if it actually is.

VICE CHAIR HALNON: Okay.

MEMBER PALMTAG: Scott Palmtag, just a comment, but I think your presentation did a really good job of laying out the reasons that the NIST test isn't available. And I think it's important to stress that the NIST test was NRC sponsored. So these people would know.

They would know whether it's applicable or not. I think the conclusions, the height over length was not the right one. It's pretty clear that the NIST tests weren't representative of a structural wall. So I'm satisfied that the NIST tests do not invalidate the LSTP.

I do have one question. It's not really related but it kind of might be related to our letters. My understanding is that the LSTP data, Seabrook is reaching the end of that. How long before Seabrook reaches that validation basis? You may not know. I'm just springing it on you.

DR. THOMAS: I believe it's sometime in, around 2035. They may approach that value.

MEMBER PALMTAG: Right. So if they do approach that, there'll be an opportunity for more testing.

DR. THOMAS: Yes.

VICE CHAIR HALNON: And don't they already have a test planned in the early 2030s?

DR. THOMAS: Yeah, that's my understanding. They're already thinking about another set of testing to increase the expansion.

MEMBER PALMTAG: So everything you've learned with LSTP and the NIST test and I assume there's also a Task 1, Task 2. I think that should all get rolled into the future tests if there are any, so I think that would be good.

VICE CHAIR HALNON: I think it's important to go back with what Ron said that the real proof is in the actual physical structural monitoring programs. We're not doing -- that's not theoretical space. We're actually physically monitoring it. So it's not, we're sitting back on calculations and projecting. That's always good. But the actual physical monitoring is ongoing, which is better than any analytical approach, I would assume.

MEMBER HARRINGTON: This is Craig. Can you comment on the role of the double wall containment at Seabrook and whether that makes all this better or worse or changes the environment for ASR, effect on the -- I don't know. Does that play any role at all?

DR. THOMAS: So the arrangement at Seabrook, it's unique. No other reinforced concrete containment plant in the U.S. has an enclosure building around it. Usually, it's the steel containments that have an enclosure building, referred to as shield building. And that is to protect the steel containment from environmental effects like tornados or missiles. My understanding of why an enclosure building is provided here is from aircraft impact findings.

MEMBER HARRINGTON: Right.

DR. THOMAS: So the other, the enclosure building, it's exposed to -- the portion that are below grade, they are exposed to groundwater. And the portions above grade, they are exposed to the elements. So the enclosure building is more susceptible to ASR development and progression than the containment building because it's protected.

MEMBER HARRINGTON: So does the enclosure building then have a -- from a safety perspective, a

lesser role or a greater role in earthquakes? Obviously, it takes a predominant role in aircraft strike. But for earthquakes, seismic loadings, is it more expendable, I guess, in that sense? I'm just trying to figure out the relevant roles.

MR. MANOLY: The enclosure building is similar to the BWR-6 where you have an enclosure outside the containment. And it's basically, what George explained, it's to deal with aircraft or any outside loads. But it's still designed for the SSE.

DR. THOMAS: It's designed as a seismic category I structure.

MEMBER HARRINGTON: It seems like that gives the inner -- the actual containment more margin, less exposed to the elements, less susceptible to ASR. And being on the outside in an earthquake, you mainly don't want it to collapse and damage the containment vessel. Again, trying to weigh what all that means in this particular context.

DR. THOMAS: Also, the containment is designed to ASME Section III, Division 2 which is a more conservative code than the ACI 318. Because it's designed to the pressure vessel. The containment is also subject to integrated leak rate tests required by Appendix J. And the most recent one was conducted in

2023 and met all the acceptance criteria. There was nothing observed out of the ordinary.

MR. MANOLY: And one more thing. The work that was done by NIST on the wall testing, you're talking about plane structure, the wall. The results there give us some information. So we did it with similar -- you have plane structures and you have the containment. Containment itself has inherently far more capacity than plane structure. So if you use the same analogy to apply to the containment, no matter what kind of loading you're talking about, applying to a plane structure versus containment shell structure has an entirely different capacity than a wall.

CHAIR KIRCHNER: So George, could you tell us and the public what you do in reviewing? You started -- well, let me back myself up. You started with an equation. Capacity needs to be greater with margin than the load and the demand, load factors times the demand. So I presume then for all the critical structures, you audit or review calculations that are presented by the applicant to demonstrate that kind of margin is there under seismic and other loads. Is that correct?

DR. THOMAS: Yes. So --

CHAIR KIRCHNER: And so what is the --

what kind of margin do you see in those calculations, capacity versus demand?

DR. THOMAS: As it stands right now, there are six structures that are in the operability space, which means they are operable and perform the intended function but they're not conforming with the licensing basis, meaning in some cases it doesn't meet the code acceptance criteria. And margins can be assessed based on the pressure factor in the analysis which is what accounts or allows for future ASR expansion from the data they had at the time from the calculations. And these structures that are in operability space, they are all around 1.2 with some localized areas exceeding the code acceptance criteria.

CHAIR KIRCHNER: Does that include the containment building?

DR. THOMAS: The containment building has much higher margins.

(Simultaneous speaking.)

CHAIR KIRCHNER: It's important to know.

DR. THOMAS: It's on the order of 1.8 or so.

CHAIR KIRCHNER: So what do you do to provide confidence that those remaining -- those six structures then are -- can you just describe what your

oversight process is to assure that those six structures remain capable of providing whatever function -- safety function is required.

DR. THOMAS: So until corrective actions are taken to bring it into compliance, licensee, they have an operability evaluation that sends -- that has monitoring limits. And typically, the monitoring is done at the smaller intervals, example would be like, every two months. The licensee's plan is to bring it all in compliance with a pressure factor of 1.5. And their plan is to do that to refine -- more refined analysis using the latest data or performing retrofit in areas where there are exceedances to increase capacity, or a combination of the two.

VICE CHAIR HALNON: Any questions, comments? Okay. At this point, we're going to open it up for public comment. George, if you'll take your slides down so we can see the gallery up there.

We're going to take the same rules as I put out this morning. So many of you probably were not on. I'll restate them. I'm going to give each commenter two minutes.

If you're still commenting after two minutes, I'll cut you off. If you won't stop, I'll mute you. I would appreciate if you can keep your

comments to the subject of this meeting and not past meetings where we've talked different topics or different scopes of topics.

We're talking about the applicability of ASR testing at NIST to the ASR at Seabrook. We're going to stop at 15 minutes. If you did not have a chance to get in line to give your comment after 15 minutes, you can submit those written comments to the DFO, designated federal officer.

So with that, if you would raise your hand on Teams to get in line. Larry Burkhart will manage the lineup of commenters. Again, we'll be timing, so two minutes.

MR. BURKHART: Okay. This is Larry Burkhart from the ACRS staff. So yes, if you do wish to make a public comment, please raise your hand. I will take you sequentially. Okay. Ms. Sarah Abramson from C-10, please provide your comment.

MS. ABRAMSON: Thank you. And just a practice question. If I don't finish my comment within two minutes but after all the comments have been submitted, there's more time, can I come back to the queue?

VICE CHAIR HALNON: On a case basis, Sarah. But since you've been the most prominent,

we'll certainly give you priority.

MS. ABRAMSON: Okay. Thank you. So I'll begin. My name is Sarah Abramson. I am executive director of the C-10 Research and Education Foundation, a group near the Seabrook Station Nuclear Power Plant. And I live in Stratham, New Hampshire near the plant.

I want to comment that I heard an emphasis on the ASR inspections at Seabrook being the best indicator of ASR's implication of safety of structures. That sounds rational. But you may know that there are soon to come overhauls of the reactor oversight process, including revisions to inspection schedules in response to the ADVANCE Act and recent executive orders.

I've attended a lot of those public meetings. This specifically included proposals to decrease the frequency of the PI&R inspections. So I hope that you consider those soon to come possible changes when you draft your letter, if it's at all possible to enshrine the current frequency of ASR inspections and protect that from any decreases that might be coming.

I also took note of comments that a pending large scale testing program is coming in the

early 2030s to increase the expansion limits. And I appreciated the ACRS member comment that the NIST finding should be incorporated into future testing. Again, enshrining this opinion in some way in your letter would be important so that it can be translated into some type of regulatory action in the seven years or so when that comes to pass.

The fact is that the NRC will not undertake any review of any LSTP until it's complete. I've talked to the inspectors about this. And they've made clear that they won't inspect the testing program plans. They'll inspect the program and its result after it's complete.

So I think the licensee and the public and the NRC would kind of all deserve and be best served by knowing what the expectations are. And I think you probably hold the key more than anyone to enshrine that in some type of letter. And the NRC sponsored study of the NIST study, it's a little confusing that they didn't expect it to be applicable to Seabrook which is the only U.S. commercial plant to suffer from ASR.

And to put this into perspective, C-10 submitted a petition for formal rulemaking to the NRC to promulgate rules on ASR testing and management.

But it was denied on the grounds that Seabrook is the only plant with ASR. So it seems to me a little wasteful to sponsor a multimillion dollar study and then have its findings not apply to the one plant that can benefit from its findings.

And I recall reading in transcripts that it was meant to be confirmatory. And so I guess we're really looking for who is doing a good job of dissecting those study findings to see if it was confirmatory of the LSTP. Thank you.

MR. BURKHART: Thank you, Sarah. Anybody else from the public that would like to make a comment at this time? Okay. I see no others. Turn back to Vice Chair Greg Halnon.

VICE CHAIR HALNON: Sarah, did you complete your comments? I know you talked fast. Was there anything else?

MS. ABRAMSON: I could give more if you're willing to provide more time. I was trying to be respectful of the time limit.

VICE CHAIR HALNON: No, I appreciate that. We'll give you a few more minutes. Go ahead.

MS. ABRAMSON: Okay. Thank you. So we do understand and have heard comments indicating that it's difficult to understand the two totally different

takeaways that qualified scientists are having on this point. So one, we would ask that if you're having a hard time understanding the literature that Dr. Saouma has supplied compared to what the NRC staff is supplying, perhaps the NRC can have them independently reviewed.

Also, we believe that the comments that Dr. Saouma has supplied are all in reference to peer reviewed papers. And I didn't see a lot of that in the presentation. Perhaps, I maybe did see a few.

But I think, again, an independent review of the two kind of contradictory findings on this could reveal why there is sort of a mismatch between the two findings. Those are the two major comments I didn't make the first time that I just wanted to make a point of. Thank you.

VICE CHAIR HALNON: Thank you, Sarah. Okay. With that, we'll close public comments. And let's go ahead and take a -- I'm sorry. We're going to take a 10-minute break -- a 15-minute break. We'll be back here at 2:30 p.m. The meeting will be recessed until 2:30.

(Whereupon, the above-entitled matter went off the record at 2:16 p.m.)

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Presentation to the ACRS Full Committee

Seabrook Alkali-Silica Reaction (ASR) Issue – Relevance of NIST Task 3 Study

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September 3, 2025

NIST Study – Task 3 on Cyclic Performance of ASR-Affected Shear Walls

- The National Institute of Standards and Technology (NIST) Research Task 3 Study is documented in Technical Note 2180, "Task 3: Assessing Cyclic Performance of ASR-Affected Concrete Shear Walls," publicly available at https://www.nist.gov/publications/structural-performance-nuclear-power-plant-concrete-structures-affected-alkali-silica-1 (NIST Report)
- Task 3 involved in-plane cyclic lateral loading tests under constant axial compression of three ASR-affected and one control wall specimens.
- The NIST Study was **generic ASR research** and not specific to Seabrook, whereas the Large-Scale Test Program **(LSTP)** conducted by NextEra was **Seabrook-specific**.
- Discuss relevance of the NIST Study to Seabrook ASR structural safety:
  1. Relevance of NIST Test Configuration and Results (Representativeness)
  2. Relevance of NIST Tests to In-plane Shear Capacity
  3. Relevance of NIST Tests to Past (Pre-instrument) Expansion Estimate

Seabrook Criteria for ASR-Affected Structures: Background

- Seabrook Acceptance Criteria based on ACI 318 Code and LSTP is:

  ϕ x Capacity ≥ Load Factor x Demand (including ASR) for all applicable limit states

- The LSTP forms a technical basis only for:
  - The **"Capacity"** side of the equation within the LSTP expansion limits, which is monitored by the ASR Monitoring Program; and
  - **Monitoring methods** used, including determination of through-thickness expansion-to-date (pre-instrument) at the time of extensometer installation.
- The **"Demand"** side is determined by **structure-specific structural analysis** of design loads (including ASR) and load combinations, with threshold limits allowing for future ASR expansion monitored by the Building Deformation Program.
- The distribution of force components (axial or membrane, flexure, out-of-plane (OOP) shear, in-plane shear etc.) is a result of the structural analysis and is checked against applicable acceptance criteria.

1. Relevance of NIST Test Configuration and Results

(Extract: NIST Report Fig. 2.2)

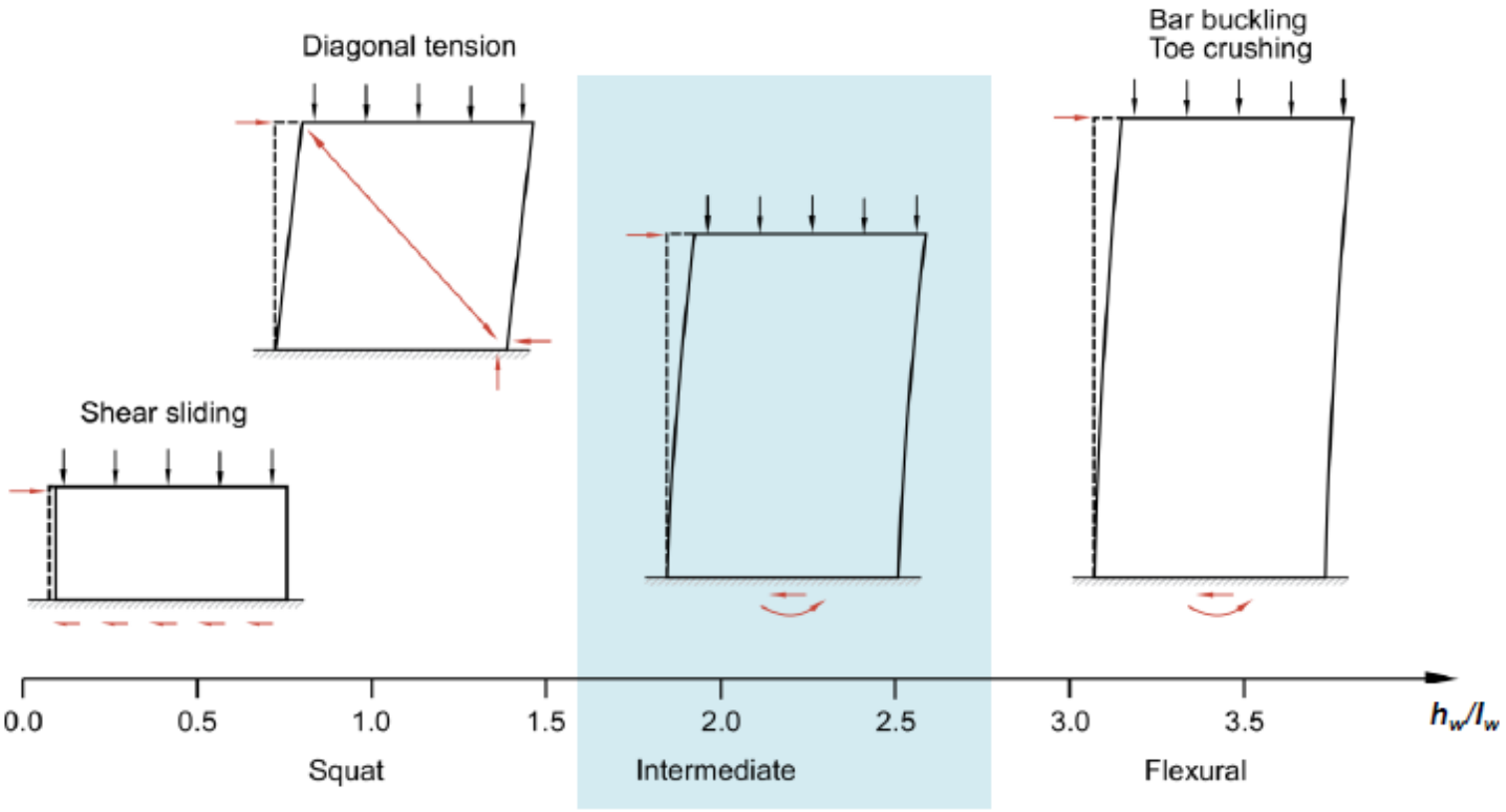

Figure 2.2: Schematic showing the effect of wall aspect ratio on predominant behavior and failure mode under lateral loading

1. Relevance of NIST Test Configuration and Results (contd...)

- NIST Task 3 shear wall test specimens had **a wall height to length** (**h/L) aspect ratio of 2** and therefore not "squat," relatively low reinforcement ratio (**0.31%**, #3 @ 8.8") and **failure mode was in flexure** (NIST Report p171) **and <u>not</u> in-plane shear**.

- Typical Nuclear Power Plant (NPP) structural walls (including Seabrook) have **low aspect ratio** (**h/L of the order 1 or less)** and larger reinforcement ratio for which the expected failure mode is **diagonal shear cracking (diagonal tension)**. Thus, NIST test specimens **were not representative of Seabrook structural walls and the test results do not apply to Seabrook**.

- Nevertheless, for the observed flexural failure mode, the measured normalized peak flexural capacity, **M'_{max}/Mn,** for all ASR-walls **are greater than 1.0** (1.132, 1.141, 1.104 for ASR vs 1.311 for non-ASR; Ref. NIST Report Table 6.1). Therefore, the tested **ASR walls reached code nominal ultimate flexural capacity, Mn, with margin** although lesser than for non-ASR wall.

- The NIST test results thus showed no reduction of maximum observed in-plane moment capacity compared to code nominal moment capacity. It poses no contradiction with LSTP out-of-plane (OOP) shear or flexural (rebar anchorage or bond) tests.

1. Relevance of NIST Test Configuration and Results (contd...)

- The LSTP test specimens were not conventional “beam” specimens. They were a slice of a representative reference location Seabrook structural wall with 2D orthogonal reinforcement on each face (providing horizontal and vertical or biaxial confinement to the wall) and no through-thickness reinforcement. For the load test, the vertical wall slice was oriented horizontally, with the 2D reinforcement layers on the top and bottom faces and loading applied normal to the top face.

- The LSTP (MPR-4273, public ML16216A242) was specific to Seabrook and as representative or bounding of typical Seabrook wall configuration as practical, and addressed the more critical limit states (flexure, out-of-plane shear, flexure and reinforcement anchorage (bond between rebar and concrete), effects on anchor bolts capacity, and instrument study) at a large scale than data available in the literature. Overall, the results of the LSTP provide the **technical basis and limitations** (e.g., expansion limits) for continued applicability of the ACI 318-71 and ASME III-2 codes-of-record to ASR-affected structures at Seabrook. The LSTP did not include in-plane shear tests.

2. Relevance of NIST Tests to In-plane Shear Capacity

- Seabrook concrete structures are subject to **design basis loads (including ASR) and load combinations** defined in the UFSAR, and physical configurations/layout that result in out-of-plane (OOP)/radial shear forces, OOP moments, in addition to membrane/axial forces and in-plane (tangential) shear forces. One or more element force components may dominate the response over the others.

- The element or sectional magnitude and distribution of these force components falls out from the structural analysis. Element or sectional design checks are made for each limit state along with applicable combined interaction. While relevant and evaluated for seismic, due to relatively larger available margin at Seabrook, in-plane shear forces typically do not control.

- Seabrook structural walls (including containment enclosure building or CEB) have **2D orthogonal reinforcement** on each face that resist in-plane shear in addition to contribution from concrete. The Containment Building (CB) has a layer of **orthogonal diagonal reinforcement** specifically designed to resist seismic tangential shear forces with zero concrete contribution.

2. Relevance of NIST tests to In-plane Shear Capacity (contd...)

- In-plane shear failure mode is expected to be relatively more ductile (due to reinforcement resisting it) versus non-ductile OOP shear failure, which is primarily resisted by concrete for the typical Seabrook configuration with no through-thickness reinforcement.

- Corroborating evidence from experimental work by Habibi et al (2018) [1] and Sawada et al (2021) [2] of lateral cyclic loading tests of ASR-affected low-aspect ratio shear walls (h/L = 0.71, web reinforcement ratio, ρ_t = 0.8%; and h/L = 0.83, ρ_t = 1%, respectively, which are more in the representative range of typical NPP structural walls) show ultimate in-plane shear capacity (strength) of the tested walls was not adversely affected by ASR. Observed failure mode was diagonal shear cracking and rebar yielding.

- There is **reasonable assurance** that Seabrook structural walls remain capable of resisting design-basis lateral seismic loads by in-plane (straight) or tangential shear (for cylindrical).

[1] Habibi et al, Effects of Alkali-Silica Rection on Concrete Squat Shear Walls, ACI Structural Journal, Sep 2018.

[2] Sawada et al, Structural Performance Evaluation and Monitoring of Reinforced Concrete Shear Walls Affected by Alkali-Silica Reactions, Journal of Advanced Concrete Technology, Volume 19, May 2021.

3. Relevance of NIST Tests to Past Expansion Estimate

- For Seabrook, the **empirical ACI code equation Ec = 57, 000 sqrt(f'c)** is used only for calculating **nominal value of concrete modulus of elasticity at time zero (Eo)** from measured compressive strength (f'c) at the time of original construction (@ 28-days, no ASR). This is used to determine value of the **normalized modulus (En = Et/Eo)** in the modulus-expansion correlation equation developed in the LSTP. This correlation is used to calculate the through-thickness expansion-to-date (pre-instrument expansion) at the time of extensometer installation. (Report MPR-4153 (public ML16279A050), p3-4)

- The elastic modulus empirical equation is **NOT used** for determining concrete modulus of elasticity (Et) **of ASR-affected concrete** at the time of extensometer installation. Et is directly measured by testing of cores removed from the location at the time of extensometer installation. There is no ASR degradation mechanism present at the time of construction; therefore, use of empirical modulus equation to determine Eo is **reasonable and justified**.

- For Seabrook, variability or uncertainty in the calculated value of the concrete elastic modulus using the empirical equation is conservatively accounted for by a **reduction factor** applied to the **normalized modulus (En = Et/Eo)** in the modulus-expansion correlation (Report MPR-4153 (public ML16279A050), p4-2)

3. Relevance of NIST Tests to Past Expansion Estimate (contd...)

- In instances where the empirical modulus equation over-predicts the original elastic modulus, use of the modulus-expansion correlation adds conservatism to the approach. In instances where the empirical equation under-predicts the original modulus, application of the normalized-modulus reduction factor adds sufficient conservatism to account for variability. (Publicly Available Report MPR-4153 (ML16279A050), p4-4)

- Regarding the empirical ACI equation for Ec, NIST Report states on page 72: ***"… This trend indicates that the compressive modulus of the reactive concrete degraded faster with ASR expansion than did the concrete's compressive strength. The non-reactive Wall 4 cylinders, on the other hand, remained within the +/- 20% range of the ACI equation."*** This is consistent with the modulus data and scatter from the LSTP (MPR-4153, p3-3, 3-6). The staff agrees that empirical modulus equation is not applicable to estimate elastic modulus in ASR-affected concrete, and it is not used for ASR-affected concrete in the LSTP methodology.

- Thus, the NIST findings **do not invalidate** the modulus-expansion correlation used at Seabrook to calculate expansion to-date at the time of extensometer installation.

Conclusion

- The NIST Task 3 wall test specimens are not representative of Seabrook structural walls (h/L, ρ_t) and thus the test results do not apply.

- The NIST Task 3 Study does not refute the overall conclusions of the Seabrook LSTP and License Amendment 159 (ML18204A291 public).

- The NRC will continue to inspect Seabrook's performance in the management of ASR under the Reactor Oversight Process.

Questions

White Paper

Technical Response to NRC's Rebuttal of My Analysis of NIST ASR Findings for Seabrook Safety

By

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Submitted to

The Advisory Committee on Reactor Safeguards. *Washibgton, DC*

August 31, 2025

About the Author

Victor E. Saouma with over 40 years of research experience, including nearly 15 years dedicated to Alkali Silica Reaction (ASR) has made significant contributions to the field. His ASR research encompasses 11 major funded projects, two books (Saouma and Hariri-Ardebili, 2021), (Saouma, V.E., 2013), 9 major reports, 9 short courses, 13 peer-reviewed papers.

He chaired an international committee through RILEM (International Meeting of Laboratories and Experts of Materials, Construction Systems, and Structures), focusing on the diagnosis and prognosis of structures affected by ASR. He was the editor of a RILEM report with over 450 pages and 30 top researchers contributing, his expertise is evident.

He is a past President and Fellow of the International Association of Fracture Mechanics for Concrete and Concrete Structures, and is thus well-versed in concrete cracking issues. He has advised the Tokyo Electric Power Company (TEPCO) on nonlinear dynamic analysis of large arch dams and on ASR-related problems for massive reinforced concrete structures. He conducted shear tests for them (and for EPRI).

He was a key contributor to EPRI's report *Structural Modeling of Nuclear Containment Structures*.

Saouma's research on AAR (Alkali-Aggregate Reaction) has been funded by various organizations including the Nuclear Regulatory Committee, Oak Ridge National Laboratory, and the Bureau of Reclamation. His technical reports are accessible online.

His research interests extend to theoretical, numerical, and experimental fracture mechanics, chloride diffusion in concrete, real-time hybrid simulation, and centrifuge testing of dams.

His international collaboration includes France, Spain, Switzerland, Italy and Japan.

In addition to his scientific expertise, Saouma is a trained civil engineer. He has taught linear and nonlinear structural analyses as well as reinforced and advanced reinforced concrete design, providing him with a broad perspective on engineering challenges.

In studying ASR over fifteen years, he has found that ASR is an extraordinarily complex and nefarious reaction. While it has been known since the 1940's, only recently have we witnessed an emergence of structures suffering from this problem (as it may take many years to manifest it- self). As a result, ASR has attracted the attention of researchers from many disciplines: chemists, mineralogists, geologists, material scientists, mechanicians, experimentalists, and yes structural engineers. Not a single one of those disciplines can provide a definite answer to questions posed by ASR. However, those who have taken a comprehensive view to the problem are best positioned to opinionate.

Given his diverse research background encompassing theoretical, experimental, numerical, and field work, as well as his leadership in addressing ASR globally, he is well-positioned to evaluate the adequacy of the work conducted at Seabrook Nuclear Power Plant.

Contents

Executive Summary

I have reviewed the NRC's 22-point rebuttal to my earlier analysis of the implications of the NIST report for the structural safety of the Seabrook Nuclear Power Plant. I approached this review with the same rigorous standards I would apply to a peer-review assignment: demanding credibility, accuracy, and completeness in technical arguments.

The NRC's response fails to meet basic scholarly standards. Of the twenty-two points presented, only one cites supporting peer-reviewed literature. More troubling, the rebuttal contains demonstrable technical errors, including a fundamental misunderstanding of how elastic modulus predictions affect expansion calculations—claiming that over-prediction "adds conservatism" when the physics dictates the opposite.

Two critical deficiencies render the NRC's safety assessment unreliable:

1. **Inappropriate testing methodology:** The Large-Scale Test Program (LSTP) relies on out-of-plane shear testing, which is scientifically inappropriate for cylindrical containment analysis where in-plane membrane forces govern structural response.

2. **Flawed expansion reconstruction:** The methodology for determining historical ASR expansion depends on empirical equations with well-documented statistical inadequacies, compounded by undisclosed "reduction factors" that cannot be independently verified.

Rather than addressing these fundamental technical concerns, the NRC deflects with irrelevant discussions and unsubstantiated assertions. The response reads more like advocacy than technical analysis, relying on professional judgment rather than documented evidence.

Based on my expertise in ASR research spanning nearly fifteen years and documented qualifications (page i), **I conclude that the current approach compromises Seabrook's ability to safely resist seismic loading.** The NRC has not demonstrated that their methodology can reliably assess structural integrity under ASR degradation.

Given the magnitude of these technical deficiencies and their implications for public safety, I strongly recommend that the ACRS submit both this analysis and the NRC's response for independent review by a panel of recognized experts in structural engineering, seismic analysis, and ASR effects on nuclear containment structures.

1 Introduction

> Because what is ultimately at stake is the structural safety of Seabrook under seismic excitation, I strongly recommend that the ACRS subject this document—like the one submitted by C-10—to external independent review by a panel of recognized structural engineering scholars

In this review I will examine, one by one, the 22 points raised by the NRC (presented in grey boxes). My approach is that of a scholar, applying a rigorous standard in which each argument must be substantiated, documented, and evaluated against accepted principles and peer-reviewed evidence. This stands in clear contrast to the engineering approach taken by the NRC, which relies heavily on professional judgment and unsubstantiated assertions.

This difference in method gives rise to a fundamental clash: on the one hand, an engineering response rooted in intuition and selective references to the American Concrete Institute (ACI) design code[1], even though the code does not address AAR and at times is invoked with a permissive interpretation; on the other, a scholarly review that insists on documented support. I encourage the reader to keep this distinction in mind when reading what follows, as my analysis necessarily employs a fine-comb approach that may appear exacting but is essential when the structural safety of Seabrook under seismic excitation is at stake."

With this framework established, I now turn to a detailed examination of each of the 22 NRC points, evaluating them individually for credibility, rigor, and relevance.

2 Test configuration

> The NIST Study was generic ASR research and not specific to Seabrook, whereas the Large-Scale Test Program (LSTP) conducted by NextEra was Seabrook-specific.

- Both the NIST study and my contract with the University of Colorado were necessarily generic in scope. The NRC commissioned them precisely to improve its understanding of ASR in reinforced concrete, not to provide site-specific case studies.
- It is difficult to see why the NRC would have invested millions of dollars in ASR research unless the intent was to address gaps in its technical knowledge relevant to plant safety.
- The NRC itself described my Colorado contract as "confirmatory." Had either the Colorado or NIST studies produced outcomes consistent with the NRC's earlier (and poorly conceived) Texas tests, those results would have been treated as valid and directly applicable to Seabrook.
- In any event, even if these studies had been framed as Seabrook-specific, the essential findings and implications would not have differed materially.

[1] The ACI code is a prescriptive design standard developed for conventional reinforced concrete structures. It does not address material degradation from Alkali–Aggregate Reaction (AAR), nor does it provide a research framework for evaluating such deterioration.

SLIDE 3

The discussion of this (questionable) approach to safety assessment is entirely irrelevant in this context.

SLIDE 4-8

Slide 5; Squat

NIST Task 3 shear wall test specimens had a wall height-to-length (h/L) aspect ratio of 2 and therefore were not "squat," had a relatively low reinforcement ratio (0.31%, #3 @ 8.8"), and failed in flexure (NIST Report p.171) rather than in-plane shear.

I agree with the NRC that the aspect ratio of the NIST shear wall tests (2.0) places them in the "intermediate" category (between squat and slender), but this does not in itself disqualify them.

> Shear walls (or wall segments) shall be considered slender if their aspect ratio (height/length) is >3.0, and shall be considered short or squat if their aspect ratio is <1.5. Slender shear walls are normally controlled by flexural behavior; short walls are normally controlled by shear behavior. The response of walls with intermediate aspect ratios is influenced by both flexure and shear.
>
> Elwood, Matamoros, Wallace, et al. (2007) *"Update to ASCE/SEI 41 concrete provisions"*

Turning to the LSTP tests, two distinct failures are evident:

Unintended: The absence of vertical reinforcement in the center of the beam allowed vertical expansion, producing a large delamination crack *before the test even started*. This flaw rendered all subsequent results highly questionable (Fig. 1).

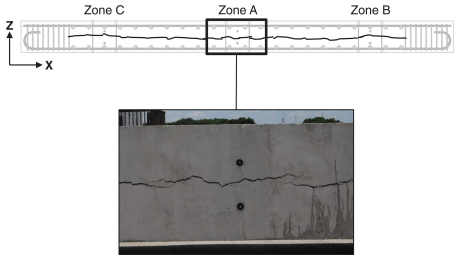


Figure 1: Unanticipated pre-test delamination

Intended: Since the purpose of the LSTP was to investigate the impact of AAR on shear strength degradation, the specimen should have been designed to maximize shear and minimize flexure. The NRC itself recognizes this necessity (see slide 4). However, even setting aside the pre-test delamination, the governing mode was not pure shear but rather *flexure-shear*[2]. In the critical zone, significant shear was accompanied by a non-negligible moment—precisely the situation for shear walls with larger aspect ratios that the NRC criticizes in the NIST tests (Fig. 2).

[2] A flexure-shear crack is a flexural crack that, under significant shear, rotates into a diagonal crack and propagates from the tension zone toward a support or load point across the web.

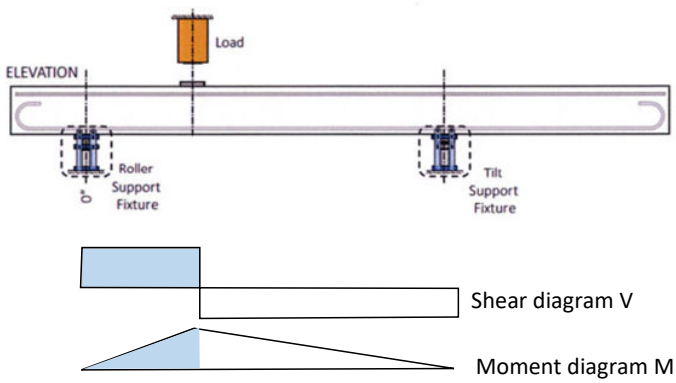


Figure 2: LSTP specimen with shear and moment (flexure) diagrams

Slide 5; Aspect ratio not applicable to an NPP

Typical Nuclear Power Plant (NPP) structural walls (including Seabrook) have low aspect ratio (h/L of the order 1 or less) and larger reinforcement ratio, for which the expected failure mode is diagonal shear cracking (diagonal tension). Thus, NIST test specimens were not representative of Seabrook structural walls and the test results do not apply to Seabrook.

To claim that an NPP has a *low aspect ratio* is **conceptually flawed**. The containment is a **thin cylindrical reinforced-concrete shell**, not a planar shear wall. For such a structure, the definitions of h and L used for wall aspect ratios are not meaningful, and applying them to the Seabrook containment is therefore **inappropriate**.

Slide 5; Nominal moment M_n

Nevertheless, for the observed flexural failure mode, the measured normalized peak flexural capacity, M'_{max}/M_n, for all ASR walls was greater than 1.0 (1.132, 1.141, 1.104 for ASR vs. 1.311 for non-ASR; see NIST Report Table 6.1). Therefore, the tested ASR walls reached code nominal ultimate flexural capacity, M_n, with margin, although the margin was smaller than for the non-ASR wall.
The NIST test results thus showed no reduction of maximum observed in-plane moment capacity compared to the code nominal moment capacity. This poses no contradiction with LSTP out-of-plane (OOP) shear or flexural (rebar anchorage or bond) tests.

This segment of the NRC review is essentially irrelevant, since in the NPP containment we are dealing with membrane action; bending moments develop only locally at the juncture between the cylindrical and spherical shells.

1. It is correct that Table 6.1 shows $M'_{max}/M_n > 1.0$ for the NIST tests. However, Swamy and AlL-Asali (1989) report that *ASR can create large irreversible concrete and steel strains that affect the overall serviceability, strength, and stability of reinforced concrete beams. The maximum recorded loss in flexural capacity due to ASR was about 25%.*

2. The NIST report itself (p. 184) notes several important reductions:

   - The presence of ASR decreased normalized peak flexural capacity, M_{max}/M_n, by about 10–11%.
   - The presence of ASR decreased normalized yield moment, M_y/M_n, by about 26%. For seismic analysis, this indicator is more relevant than maximum moment, as it reflects the ductility[3] required to dissipate energy without brittle, sudden cracking.
   - When the four NIST measurements of M_{max}/M_n were combined with three additional measurements from Oh, Han, and Lee (2002)[4], the presence of ASR was found to reduce the mean M_{max}/M_n by about 7%.

In light of this evidence, the NRC's assertion that the NIST tests show "no reduction" is **misleading**: while ultimate flexural strength may have met code values, ductility—**critical under seismic loading**—was demonstrably compromised by ASR. The NIST results (Table 6.1) clearly show that the ASR-affected walls exhibited reduced ductility, with normalized yield moment M_y/M_n decreased by approximately **26%** compared to the non-ASR wall. **Ignoring such reductions in ductility is unacceptable for any credible seismic safety evaluation of Seabrook.**[5]

Slide 6; Design of the LSTP beam

The LSTP test specimens were not conventional "beam" specimens. They were a slice of a representative reference location Seabrook structural wall with 2D orthogonal reinforcement on each face (providing horizontal and vertical or biaxial confinement to the wall) and no through-thickness reinforcement. For the load test, the vertical wall slice was oriented horizontally, with the 2D reinforcement layers on the top and bottom faces and loading applied normal to the top face.

I agree that the configuration is indeed **unconventional** and **unrepresentative** of the shear resisted in a cylindrical containment subjected to lateral load. It is universally accepted that in this case, the lateral load is resisted by *membrane in-plane* shear,[6] not by out-of-plane shear.

In fact, the NRC itself acknowledges that the LSTP does not capture in-plane shear and does not treat the specimen as representative of membrane action.

- The LSTP setup is illustrated in Fig. 3(a). Note that pre-test delamination occurred in the center segment of the specimen. What is actually modeled is a narrow vertical strip

[3] In structural engineering, *ductility* is the ability of a material or structural element to undergo significant plastic deformation before failure. In seismic design, ductility is essential because it allows a structure to dissipate earthquake energy through controlled inelastic deformations rather than collapsing in a brittle manner (Paulay and Priestley, 1992).

[4] The geometry and reinforcement ratios of the NIST wall specimens were selected to match non-reactive walls previously tested by Oh et al. (2002) to facilitate independent comparisons.

[5] In structural engineering, *ductility* is the ability of a material or structural element to undergo significant plastic deformation before failure. In seismic design, ductility is essential because it allows a structure to dissipate earthquake energy through controlled inelastic deformations rather than collapsing in a brittle manner (Paulay and Priestley, 1992).

[6] Global response is membrane-dominated; in-plane resultants $N_\theta, N_\phi, N_{\theta\phi}$ govern. Transverse (out-of-plane) shear Q_θ, Q_ϕ is generally small and localized near supports, junctions, penetrations, and under non-axisymmetric actions; it is not zero.

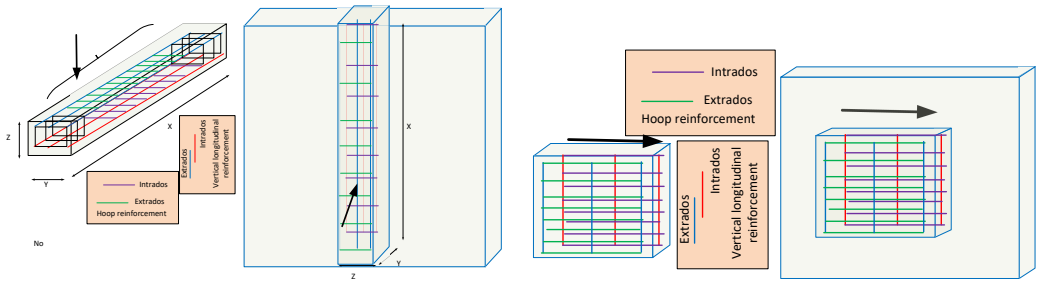
of the wall by "punching" through it. This strip is artificially **laterally constrained** by adjacent concrete; such edge effects could have been reduced had the LSTP tested a plate rather than a beam. Key differences are:

- **Dimensionality:** Beam theory is 1-D; plates are 2-D with coupled curvatures (κ_x, κ_y) and twisting M_{xy}. Beam tests cannot capture M_y or M_{xy}.
- **Shear field:** Beams develop primarily Q_x (τ_{xz}), whereas plates carry bidirectional shear Q_x, Q_y (τ_{xz}, τ_{yz}). Beam tests miss through-width shear flow.
- **Crack mechanics:** Beams transition flexure → flexure–shear. Plates develop **two-way crack fields** (diagonal + transverse) governed by biaxial bending and in-plane shear.
- **Torsion/biaxial coupling:** Plates experience twisting moments and Poisson coupling; beams omit these effects.
- **Size (width) effects:** Plate response reflects finite-width phenomena (shear-lag, stress redistribution). Beams lack a mechanism to capture this.

Thus, while I strongly disagree with the design philosophy of relying on out-of-plane shear, the handicap would have been at least partially mitigated had a plate been tested instead of a beam.

- By contrast, an **in-plane shear test** configuration, Fig. 3(b), would be far more representative of the structural response of an NPP containment under lateral seismic load. By limiting the height, flexural effects are minimized and shear clearly dominates.

Bottom line: to the best of my knowledge, the LSTP is the **only test program in the world** that relies on **out-of-plane shear** (Fig. 3(a)), in stark contrast to other researchers who correctly test **in-plane shear**.



(a) LSTP (b) Other researchers

Figure 3: LSTP vs. in-plane shear tests by others

This acceptance of LSTP by the NRC stands in direct contradiction to established principles of structural mechanics and to the testing practices followed worldwide.

Slide 6; Test configuration trying to satisfy multiple demands

The LSTP (MPR-4273, public ML16216A242) was specific to Seabrook and as representative or bounding of typical Seabrook wall configuration as practical, and addressed the more critical limit states (flexure, out-of-plane shear, flexure and reinforcement anchorage (bond between rebar and concrete), effects on anchor bolts capacity, and instrument study) at a large scale than data available in the literature. Overall, the results of the LSTP provide the technical basis and limitations (e.g., expansion limits) for continued applicability of the ACI 318-71 and ASME III-2 codes-ofrecord to ASR-affected structures at Seabrook. The LSTP did not include in-plane shear tests.

- The test configuration indeed does **not explicitly refer to Seabrook in particular**; it represents a generic NPP. At the very least, the test should have attempted to satisfy the requirements of the Buckingham Pi theorem (Buckingham, 1914)[7], and clarified how the **reinforcement of the test specimen correlates with that of an actual NPP containment structure** as I have attempted to elucidate in Fig. 3(a).
- The single test configuration attempts to cover **too many limit states simultaneously**:

  1. flexure
  2. out-of-plane shear
  3. flexure and reinforcement anchorage
  4. anchor bolts capacity
  5. instrument study

  I agree that the beam test may have been adequate for the last three objectives. However, in an NPP **flexure is not predominant**, and with respect to out-of-plane shear—**which is not the relevant shear mode for containment**—a plate should have been tested instead of a beam.

- The assertion that the LSTP results "provide the technical basis and limitations (e.g., expansion limits) for continued applicability of the ACI 318-71 and ASME III-2" is **irrelevant**: **bad data can also be forced into a good model**. What is needed is an **in-plane shear test**, which would have provided reliable and representative shear data.

Slide 7; Mixing correct but irrelevant with erroneous details

Seabrook concrete structures are subject to design basis loads (including ASR) and load combinations defined in the UFSAR, and physical configurations/layout that result in out-ofplane (OOP)/radial shear forces, OOP moments, in addition to membrane/axial forces and inplane (tangential) shear forces. One or more element force components may dominate the response over the others.
Seabrook structural walls (including containment enclosure building or CEB) have 2D orthogonal reinforcement on each face that resist in-plane shear in addition to

[7]Buckingham Pi (scaling) rule, in plain terms: many physical problems can be described by a few key *ratios* with no units (for example, height/length or load/strength). A small model represents the real structure only if those ratios are the same in both. If the ratios are not matched, the model's results cannot be trusted for the full-size case. Think of it like a recipe: halving every ingredient works; changing only some does not.

contribution from concrete. The Containment Building (CB) has a layer of orthogonal diagonal reinforcement specifically designed to resist seismic tangential shear forces with zero concrete contribution.

- This discussion is **completely irrelevant** in the context of responding to criticism of the LSTP test configuration.
- The NRC claims—without clarification or substantiation—that the configuration results in membrane/axial forces. A simple free body diagram[8] demonstrates that there are **no such forces**.
- The statement that "one or more element force components may dominate the response over the others" is a **gratuitous claim**, offered without identifying which force component dominates (I maintain it is the out-of-plane response) or providing any scientific substantiation.
- In short, much of this passage combines **true but irrelevant assertions with erroneous ones**, a ploy often used when substantive arguments are lacking. The reader should be on guard against such tactics so as not to be **misled or obscured**.

Slide 8; Irrelevant detail given delamination compromise

In-plane shear failure mode is expected to be relatively more ductile (due to reinforcement resisting it) versus non-ductile OOP shear failure, which is primarily resisted by concrete for the typical Seabrook configuration with no through-thickness reinforcement.

This statement shifts the discussion away from the real failure mechanisms relevant to a cylindrical containment shell, where global response is ***membrane-dominated*** *rather than governed by through-thickness shear.*

I agree that the in-plane response is **more ductile** than the out-of-plane response, where shear is resisted directly by the concrete (since there are no stirrups). **However, the assertion that Seabrook containment is governed by a "non-ductile" out-of-plane shear mode is questionable: as a cylindrical shell, its global response is membrane-dominated, and through-thickness shear is not the critical failure mechanism.**

Let us also not forget that the LSTP test specimens cracked along their length *prior to testing*. At that point, the LSTP was no longer testing one beam of depth h, but effectively two beams of depth $h/2$ stacked on top of each other. **This is summarized quantitatively in Table 1**, which highlights the differences.

Bottom line: once delamination occurred, the experimentalist had no way of knowing whether the behavior corresponded to Case 2 or Case 3. **One cannot credibly assess the safety of an NPP subjected to AAR on the basis of such clumsy and compromised results.**

[8] A free body diagram is a schematic representation of a body or structure isolated from its surroundings, showing all external forces and moments acting upon it.

Table 1: Comparison of original deep beam with two-beam arrangements: well-connected (ideal) versus delaminated (weakened). The delaminated case (Case 3) **invalidates the LSTP results as a basis for any credible safety assessment of Seabrook**.

Configuration	**Deflection**	**Bending Stress**	**Shear Stress**
1. Original beam (depth h)	small (stiff section)	low (spread over deep section)	low (spread over deep section)
2. Two beams, fully connected (acting as one)	essentially the same as original	essentially the same as original	essentially the same as original
3. After delamination (two beams, depth $h/2$ each, unconnected)	about **4× larger**	about **4× higher**	about **2× higher**

Slide 8; Conclusions

There is reasonable assurance that Seabrook structural walls remain capable of resisting design-basis lateral seismic loads by in-plane (straight) or tangential shear (for cylindrical).

Ideally, the qualifier **"reasonable"** should have been omitted altogether. This assertion can be **easily challenged**, because:

In-plane: The specimen was already **damaged prior to testing**, rendering all subsequent results highly questionable.

Tangential: If by this the NRC means "in-plane shear," then the claim is **unsupported by the test data**. The LSTP provides no credible basis for such a conclusion.

Bottom line: this statement is part of the NRC's rebuttal of my previously submitted analysis—and as shown above, it is **fundamentally unfounded and scientifically indefensible**.

3 Relevance of NIST Tests to Past Expansion Estimate

Slide 9; Applicability of the equation

For Seabrook, the empirical ACI code equation $E_c = 57,000\sqrt{f'_c}$ is used only for calculating the nominal value of concrete modulus of elasticity at time zero (E_0) from measured compressive strength (f'_c) at the time of original construction (@ 28 days, no ASR). This is used to determine the value of the normalized modulus ($E_n = E_t/E_0$) in the modulus–expansion correlation equation developed in the LSTP. This correlation is used to calculate the through-thickness expansion-to-date (pre-instrument expansion) at the time of extensometer installation. (Report MPR-4153 (public ML16279A050), p. 3–4)
The elastic modulus empirical equation is NOT used for determining the concrete modulus of elasticity (E_t) of ASR-affected concrete at the time of extensometer in-

stallation. E_t is directly measured by testing of cores removed from the location at the time of extensometer installation. There is no ASR degradation mechanism present at the time of construction; therefore, use of the empirical modulus equation to determine E_0 is reasonable and justified.

- I agree that the equation is applied only to concrete at an early age. However, reliance on the ACI expression is **problematic**, as it is purely empirical.
- The NRC does not address my central concern: the NIST data (not disputed) **unequivocally show that reliance on this equation is not only incorrect but also unconservative**.
- The absence of an alternative method does not make the current approach correct by elimination. At a minimum, one should provide **error bars or uncertainty bounds** to delimit its range of applicability.
- ASR is highly heterogeneous: expansion may occur at point A and be entirely different (or absent) a few feet away. Reliable reconstruction of past expansion would require **fine-grained historical data**, but only a very limited number of cores were tested at construction. It is therefore highly probable that the assumed expansion history is **poorly correlated with the actual local behavior**.
- **Historical context matters:** Based on Pauw's (1960) foundational research, which forms the basis for the current ACI 318 equation, the statistical relationship between compressive strength and elastic modulus is inherently poor. Pauw observed "a poor statistical relationship between compressive strength and the elastic modulus" and recommended "a future reassessment of the role of compressive strength in estimating the elastic modulus." Puttbach, Prinz, and Murray (2023) note that despite six decades since Pauw's recommendation, this fundamental weakness persists. The equation essentially conflates two mechanistically different properties: elastic modulus is primarily governed by aggregate properties and the aggregate-paste interface, while compressive strength is controlled by paste strength. Therefore, the assertion of high variability in predicting elastic modulus from compressive strength for normal weight concrete remains valid

In short: the NRC's position does not resolve the fundamental problem—**past expansion cannot be credibly reconstructed using empirical equations with substantial scatter in the underlying data**.

Slide 9 Applicability of the equation

For Seabrook, variability or uncertainty in the calculated value of the concrete elastic modulus using the empirical equation is conservatively accounted for by a reduction factor applied to the normalized modulus ($E_n = E_t/E_0$) in the modulus-expansion correlation (Report MPR-4153 (public ML16279A050), p4-2)

Indeed, ML16279A050 states that "*a normalized modulus reduction factor of XXX is applied so that the final calculated through-thickness expansion is conservative*". However, **this approach raises fundamental concerns about transparency and adequacy**:

- **Why is the reduction factor redacted?** Transparency in safety calculations is essential for public confidence and technical review.
- **A reduction factor cannot remedy a fundamentally inadequate baseline calculation.** If the original value X is unconservative (as established by NIST and not challenged by the NRC), applying an undisclosed reduction factor provides no assurance of safety. For example, if X = 100 but NIST demonstrates the correct value should be 70, **how can we verify that the NRC's reduction factor is sufficient?** If only a 20% reduction is applied, yielding 80, this remains 14% higher than the NIST-established value of 70.
- **The logic is circular**: The NRC acknowledges uncertainty exists (hence the need for a reduction factor) but simultaneously claims the result is "conservative" without demonstrating that the reduction adequately addresses the identified uncertainty.

Slide 10 Under/over estimate of the modulus

In instances where the empirical modulus equation over-predicts the original elastic modulus, use of the modulus-expansion correlation adds conservatism to the approach. In instances where the empirical equation under-predicts the original modulus, application of the normalized-modulus reduction factor adds sufficient conservatism to account for variability. (Publicly Available Report MPR-4153 (ML16279A050), p4-4)

The NRC's statement contains a fundamental error that undermines their entire safety analysis:

- **The NRC's first claim is categorically wrong.** The NRC states: *"In instances where the empirical modulus equation over-predicts the original elastic modulus, use of the modulus-expansion correlation adds conservatism to the approach."* This is the exact opposite of reality[9]. As clearly shown in Figure 4, when the initial modulus is over-predicted ($E_0 \nearrow$), the normalized modulus decreases ($E_n \searrow$), which leads to **increased predicted expansion** ($\varepsilon^{AAR} \nearrow$). This is **non-conservative**, not conservative.
- **The NRC's second claim is equally flawed.** They state: *"In instances where the empirical equation under-predicts the original modulus, application of the normalized-modulus reduction factor adds sufficient conservatism to account for variability."* Again, the physics is backwards: when E_0 is under-predicted ($E_0 \searrow$), the normalized modulus increases ($E_n \nearrow$), leading to **decreased predicted expansion** ($\varepsilon^{AAR} \searrow$). The system naturally becomes more conservative in this case, **without any reduction factor**.
- **No scientific justification exists for the adequacy of the reduction factor.** As previously discussed, there is no evidence that the (redacted) reduction factor is sufficient to compensate for the systematic errors in the empirical equation. The NRC

[9]Such a fundamental error from the NRC is incomprehensible and suggests inadequate technical review. This type of error should not occur at an agency entrusted with safeguarding public safety from nuclear accidents.

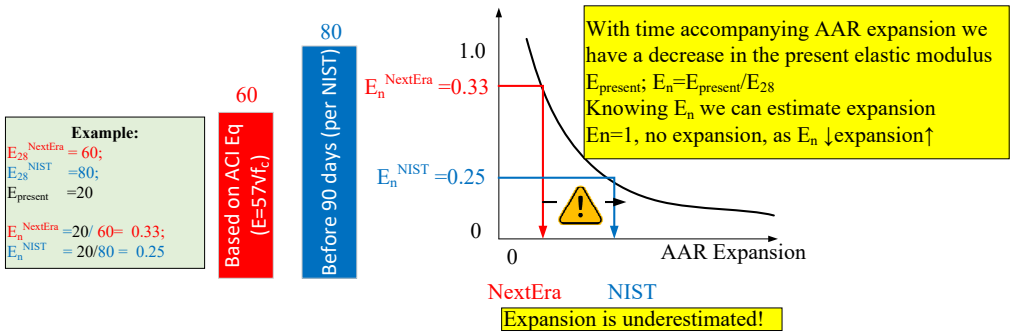


Figure 4: Effect of underestimation of E_0 in expansion prediction

provides no quantitative analysis demonstrating that their adjustment addresses the magnitude of uncertainty identified by NIST.

Slide 10 Applicability of the ACI equation

Regarding the empirical ACI equation for E_c, NIST Report states on page 72: *"... This trend indicates that the compressive modulus of the reactive concrete degraded faster with ASR expansion than did the concrete's compressive strength. The non-reactive Wall 4 cylinders, on the other hand, remained within the +/- 20% range of the ACI equation."* This is consistent with the modulus data and scatter from the LSTP (MPR-4153, p3-3, 3-6). The staff agrees that empirical modulus equation is not applicable to estimate elastic modulus in ASR affected concrete, and it is not used for ASR-affected concrete in the LSTP methodology.

This statement appears to be addressing a non-existent issue and creates unnecessary confusion:

- **No one has claimed the ACI equation applies to ASR-affected concrete.** The fundamental issue is not whether the ACI equation works for degraded concrete—of course it doesn't. The issue is whether the ACI equation can **accurately estimate the original, undamaged elastic modulus** (E_0) needed for the LSTP methodology.
- **The NRC is conflating two different applications.** There is a critical distinction between:
  - Using the ACI equation to estimate current modulus of ASR-degraded concrete (inappropriate and never proposed)
  - Using the ACI equation to estimate the original modulus of concrete before ASR degradation (the actual concern raised by NIST)
- **The statement deflects from the real problem.** By focusing on the obvious fact that the ACI equation doesn't work for damaged concrete, the NRC avoids addressing the core issue: **the high variability and poor statistical correlation between compressive strength and elastic modulus in normal concrete**, as identified by (Pauw, 1960) and confirmed by the NIST report.

- **The reference to LSTP data scatter actually supports the critics' position.** The NRC mentions "modulus data and scatter from the LSTP" as if this validates their approach, but scatter in the data actually demonstrates the unreliability of empirical predictions—exactly the concern being raised.

Slide 10 Applicability of the ACI equation
Thus, the NIST findings do not invalidate the modulus-expansion correlation used at Seabrook to calculate expansion to-date at the time of extensometer installation.

This conclusion is fundamentally flawed and contradicted by the NRC's own analysis. The NRC's dismissal of the NIST findings is particularly troubling given the multiple technical errors and omissions in their presentation:

- **The NRC made a blatant error in analyzing the impact of elastic modulus under-prediction.** Their claim that over-predicting the original elastic modulus "adds conservatism" is physically incorrect—it actually leads to non-conservative (higher) expansion estimates, as demonstrated in Figure 4.
- **The NRC ignores fundamental statistical limitations.** They fail to acknowledge that the substantial randomness and scatter in the empirical data warrant the inclusion of uncertainty bounds and error bars in any safety analysis. The high variability identified by Pauw (1960) and confirmed by subsequent data from the NIST report, as well as scatter reported in my original analysis of Bureau of Reclamation data (Dolen, 2005), demonstrates that single-point estimates from empirical equations are insufficient for safety-critical applications without proper uncertainty quantification.
- **None of the NRC's six arguments refute the core technical concerns.** Rather than addressing the fundamental statistical inadequacy of using compressive strength to predict elastic modulus—the central issue raised by NIST—the NRC deflects to tangential issues and mischaracterizes the problem.
- **The NRC relies on an unquantified "reduction factor" to compensate for unquantified errors.** This approach provides no scientific basis for confidence in safety margins. Without transparency regarding both the magnitude of the error in E_0 determination and the adequacy of the correction factor, the public and technical community cannot evaluate the safety implications.
- **The burden of proof remains unmet.** The NRC has not demonstrated that their methodology can reliably reconstruct historical expansion values with sufficient accuracy for safety-critical applications, particularly given the empirical equation's inherent limitations and the consequential nature of potential underestimates.

In short: the NRC's position does not resolve the fundamental problem—**past expansion cannot be credibly reconstructed using empirical equations with substantial scatter in the underlying data**.

4 Conclusion

This detailed examination of the NRC's 22-point rebuttal reveals fundamental flaws in both their technical analysis and regulatory approach to ASR-affected structures at Seabrook Nuclear Power Plant.

Critical Technical Deficiencies The NRC's response contains several **egregious technical errors** that call into question the competency of their safety assessment:

- **Physics misconceptions:** The NRC incorrectly claims that over-predicting the original elastic modulus "adds conservatism" when the opposite is true—it leads to non-conservative expansion estimates. This represents a fundamental misunderstanding of the modulus-expansion correlation.
- **Inappropriate test methodology:** The LSTP's reliance on out-of-plane shear testing contradicts established structural mechanics principles for cylindrical containment analysis, where membrane-dominated in-plane shear governs. The NRC's acceptance of this approach is scientifically indefensible.
- **Compromised experimental data:** The pre-test delamination that compromised the LSTP specimens fundamentally altered the structural behavior being measured, yet the NRC continues to rely on these flawed results for safety-critical decisions.

Methodological Inadequacies Beyond specific technical errors, the NRC's approach suffers from systemic methodological problems:

- **Empirical equation limitations:** The NRC dismisses well-documented concerns about the ACI elastic modulus equation's statistical inadequacy, first identified by Pauw (1960) and confirmed by NIST data. The equation's high variability and poor correlation between compressive strength and elastic modulus make it unsuitable for safety-critical historical reconstructions.
- **Lack of transparency:** Critical safety factors are redacted without justification, preventing independent verification of their adequacy. The use of an undisclosed "reduction factor" to compensate for known systematic errors provides no scientific basis for confidence.
- **Circular reasoning:** The NRC simultaneously acknowledges uncertainty in their calculations (hence the need for reduction factors) while claiming the results are "conservative" without demonstrating that their adjustments adequately address the identified uncertainties.

Regulatory and Scientific Standards The contrast between approaches is stark and concerning. Where rigorous scientific analysis demands:

- Documented evidence and peer-reviewed support
- Proper uncertainty quantification with error bounds
- Transparent methodologies subject to independent review

- Conservative assumptions when dealing with public safety

The NRC instead relies on:

- Unsubstantiated engineering assertions and professional judgment
- Single-point estimates without uncertainty bounds
- Redacted safety factors that cannot be independently verified
- Claims of conservatism that are demonstrably incorrect

Safety Implications The documented technical errors and methodological inadequacies raise serious questions about the NRC's ability to ensure public safety at Seabrook. Specifically:

- **Unreliable expansion estimates:** The methodology cannot credibly reconstruct historical ASR expansion with sufficient accuracy for seismic safety assessment.
- **Non-conservative assumptions:** Key aspects of the analysis underestimate potential structural degradation, contrary to standard nuclear safety practice.
- **Inadequate structural testing:** The reliance on inappropriate test configurations fails to capture the actual failure mechanisms relevant to containment structures under seismic loading.

Recommendations Given the magnitude of technical deficiencies identified and the stakes involved, I make the following recommendations:

1. **Independent expert review:** Both this analysis and the NRC's response should be submitted to a panel of recognized structural engineering experts with expertise in ASR, seismic analysis, and nuclear containment design for independent evaluation.
2. **Comprehensive testing program:** A properly designed testing program focusing on in-plane shear behavior of ASR-affected concrete specimens representative of containment wall configurations should be conducted.
3. **Transparent methodology:** All safety factors, reduction coefficients, and calculation procedures must be disclosed and subjected to peer review to enable independent verification of adequacy.
4. **Conservative interim measures:** Until these fundamental technical issues are resolved through credible scientific analysis, additional monitoring and potentially enhanced seismic restrictions should be considered to ensure public safety.

Final Assessment

The NRC's 22-point rebuttal fails to address the core technical concerns raised about Seabrook's structural integrity under ASR degradation. Indeed, the response reveals additional technical errors and methodological flaws that further undermine confidence in the current safety assessment.
Two fundamental deficiencies remain unresolved:

1. **The testing program is inadequate:** The LSTP's focus on out-of-plane shear is scientifically inappropriate for cylindrical containment analysis. A proper shear wall test examining in-plane shear behavior—the actual failure mechanism relevant to seismic loading of containment structures—should have been conducted.

2. **The methodology to determine past expansion is fundamentally flawed:** Historical expansion cannot be credibly reconstructed using empirical equations with substantial scatter in the underlying data, particularly given the systematic errors and lack of uncertainty quantification identified in this analysis.

The NRC has failed to properly address my earlier technical contentions, instead deflecting with irrelevant discussions and demonstrably incorrect assertions about the physics of structural behavior. As documented in my credentials and expertise (outlined on page i) **it is my professional opinion that if the current approach remains unchanged, the safety of Seabrook to resist even small seismic events is compromised**.
Public safety in nuclear facilities demands the highest standards of technical rigor and transparency. The documented deficiencies in the NRC's approach fall far short of these standards, necessitating immediate independent review and corrective action before any conclusions about Seabrook's continued safe operation can be drawn.

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Peer review of the technical white paper authored by Professor emeritus Victor E. Saouma (Univ. of Colorado – Boulder)

By Prof. emeritus Jacky Mazars (Polytechnic Institute –Grenoble Alpes University, France)

Some words About Prof. J. Mazars

Being from the same generation as V. Saouma, I, like him, have over 40 years of experience in research on concrete and concrete structures, mainly focused on analyzing damage to materials caused by mechanical effects and the effects of time. The objective is to analyze the risk in extreme situations for structures when they are subjected to earthquakes, explosions, impacts or other environmental actions.

Background : Doctor from Paris University, Professor at ENS Paris-Saclay then at INP Grenoble (currently emeritus).

Scientific advisor:

- at Electricité De France (EDF) : Vercors research program performed around a mock-up of a Nuclear Enclosure Building and Co-chairman of TINCE'23 (Technological Innovations in Nuclear Civil Engineering -Paris-Saclay 2023) and in this capacity Guest Editor of a special issue TINCE of the European Journal of Environmental and Civil Engineering (Vol. 28, n°13, Nov. 2024)
- at OXAND group providing consultancy and software solutions to help clients optimize their Asset and infrastructures

Committee and expertise - Civil Engineering expert at : ANR France, FRS Belgium, NSRC Canada, NCN Poland; Advisory Board on Structural Earthquake Engineering - European Commission (JRC Ispra Italy), Committee on Testing methods in Fracture Mechanics (RILEM). He was member of several institution: ASCE, ACI, RILEM, AFPS…. and led numerous collaborations: Associated Prof. at Univ. Sherbroooke (Canada), invited Prof at UC Berkeley, Northwertern Univ. and several university in Europe (UK, Italy, Spain, Switzerland & France).

Scientific responsabilities : in France CEOS.fr research program - ARVISE program and in Earthquake Engineering CASSBA, CAMUS, CAMUS 2000 programs – in Europe PREC8, ICONS, SAFERR, ECOEST, LESSLOSS all in the field of Earthquake Engineering.

Scientific production : About 80 peer-review papers and contribution to books chapters and 8 books acting as editor or author including : "Damage and Cracking of Concrete Structures" from J. Mazars & S. Grange – ISTE-WILEY 2022.

Peer review of the technical white paper authored by Dr. Victor E. Saouma

This white paper (WP) aims to alert the relevant authorities to a questionable analysis of the effects of Alkali Silica Reaction (ASR) on the Seabrook power plant.

Based on this WP, four questions are asked by C-10 Research and Education Foundation, which are answered below.

1- Is the paper review technically sound ?

The answer is clearly “yes”. Professor Saouma is a respected figure in the international scientific community, and the presentation of his skills and expertise in the WP confirms this (15 years devoted to ASR, 11 major funded projects, 2 books, dozens of rapport and peer-reviewed papers and the chairmanship of an international RILEM committee on the subject). He is also a leader in the field of concrete structure behavior and has served as president of the International Association of Fracture Mechanics for Concrete and Concrete Structures (FraMCoS), he has advised the Tokyo Electric Power Company (TEPCO-Japan) and was a key contributor to EPRI’s report “Structural Modeling Of Nuclear Containment Structures”. All of this involves numerous collaborations particularly in Europe and, as mentioned above, Japan.

The subject addressed by the WP is the result of his position as an expert witness with C-10 since 2019, as he indicates “His testimony resulted in the implementation of stronger measures for monitoring the state ASR over 20-year license renewal term” and then he is well positioned to evaluate the adequacy of the work conducted at Seabrook NPP.

2- Does the white paper make valid scientific arguments ?

In my opinion there are two key points in the arguments put forward in the WP :

a/ the fact of relying on results from tests carried out on a type of structure that is not representative of the situation in which a Containment Enclosure Building (CEB) is, when it is subjected to ASR and a seismic type loading. The resulting problem is that the results contradict those obtained elsewhere on shear walls representative of in-plane loading (the situation experienced by the CEB), whereas the beam test carried out by NextEra is representative of out-of-plane loading.

Tables 1 & 2 of the WP are clear on this subject :

Results on RC beam (NextERA) : no reduction of shear capacity in ASR–affected concrete

Results on shear walls (NIST) : the presence of ASR caused a # 20% reduction in the shear strength

It should be noted, however, that while this downward trend is confirmed by tests carried out at the University of Colorado (-22%), it is not confirmed by tests carried out in Japan

(Kajima), which did not find a significant decrease, or by tests carried out at the University of Toronto, which found a slight increase in this resistance. All of this confirms that the conclusion drawn from the NextEra's tests is incorrect and that this point should be further investigated.

That said, it is also the use of NextEra's results that poses problems. In summary, the idea is to consider that after the ASR effect, a new concrete with modified mechanical characteristics is obtained and that it is sufficient to use the ACI 318 formulas to move forward, particularly in estimating ASR expansion, which is a major indicator for predicting the behavior of CEB over time.

To fully understand the subject of concrete damage, its internal microstructure is modified by expansions and microcracks and no longer reacts in the same way to traditional stresses. This is reflected in particular in the tests carried out and presented in Figure 5 by a wide dispersion of results, which has a significant impact on the determination of past expansion (Figure 7). The wide dispersion of results leads us to say that conclusions that do not take this wide dispersion into account will produce erroneous results. As stated in the WP "the relationship between compressive strength and elastic modulus cannot be reliably captured by a single equation" (equation (1) in the WP).

3- Are Dr. Saouma's conclusions supported by scientific evidence?

The answer here is also "yes". The WP is the result of an analysis based on the experience and expertise of a man of culture on the topic of ASR, but also on many others subjects (he is very knowledgeable about finite element structural analysis, and his presentation in the WP on the particularities of membrane action (Appendix B) is that of a man experienced in the theme).

His analysis is based on a rich and solid bibliographic knowledge from institutions recognized for the quality of their work.

Thus, in the conclusions presented on page 10 of the WP, I fully endorse what is said :

a/ on "LSTP erroneous test configuration", especially on the points 1 (on the NextEra's tests), the point 4 (the non-use of the membrane theory) and point 5 (the failure to take into account the biaxial confinement present in the CEB).

b/ on "Relevance of NIST report on shear strength", especially on point 1 related to the shear strength of ASR-concrete, which is a major point.

c/ on "Relevance of NIST tests on past expansion". I totally agree with points 1 (inapplicability of the ACI Code equation relating compressive strength to elastic modulus) and 2 (the NextEra's procedure to estimate past expansion) and I confirm Dr. Saouma's opinion on the fact that the current structural monitoring program is fundamentally flawed and presents a significant safety risk (point 3).

4- Do you see a weak link in Dr. Saouma's argumentation?

Following on from what I said above (Q 1, 2 and 3) I can only answer "no" to this question, and I would add that, in my opinion, the arguments developed in the WP and the conclusions drawn from them lead, in my view, to the need to revisit the studies carried out on the basis of the NextERA trials and to incorporate the results of other experiments more suited to the context in order to move forward with a new analysis.

And above all, it is important to be very vigilant when transferring the observed effects to structural calculations, for which the ASR-related risk analysis must be based on nonlinear finite element calculations, which are the only method capable of accurately determining the consequences of ASR development in this plant. In any case, that is what we would try to do in Europe.

Dr. Jacky Mazars, August 2025

Written Statement for ACRS – Palisades Restart

I urge the Committee to reflect carefully on the precedent being set at Palisades. The issues here are larger than one plant. They go directly to the credibility of the nuclear industry, the legitimacy of the NRC itself, and the long-term health of our ecosystem. If nuclear energy is to play any constructive role in addressing climate change, decisions must be guided by caution, discipline, and stewardship — not speed or cost alone.

First, the risk to the industry and the regulator. If the restart of Palisades proceeds along the current path and fails, the result will not be contained to Michigan. A major failure will undermine confidence in nuclear power worldwide, accelerate premature plant closures, and increase global carbon emissions at a time when the world can least afford it. Just as damaging, public trust in the NRC's independence and credibility will erode further if the agency is seen as endorsing shortcuts over safety.

Second, the lack of a process. Once a plant has been placed into decommissioning, the existing license no longer authorizes operation under 10 CFR 50.82(a)(2). Palisades is the first case in U.S. history where a reactor decommissioned under this process is being restarted. That makes it a defining test case. Treating restart as a "reactivation" assumes continuity that no longer exists. The only credible options are: (1) pause Palisades and develop a new, dedicated process for restarts of closed plants, or (2) apply the only proven process we already have — a full license review as if the plant were new. Anything less invites unnecessary risk.

Third, the implications for the rest of the fleet. If NRC sets the precedent that plants in decommissioning may be reopened without full review, then every operating plant must now assume that decommissioning is not final. That means every facility would need to be maintained as if it will run indefinitely, with major increases in preventive maintenance and cost. Otherwise, operators are simply shifting risks and costs onto the public. Palisades is not just a local issue; it resets the regulatory and economic framework for the entire industry.

Finally, the responsibility across generations. Nuclear decisions endure. The choice made at Palisades will shape public trust and industry practice for decades to come. This Committee has a duty not only to today's Commission but also to the generations who will live with the consequences.

For these reasons, I respectfully urge the Committee to place on the record that the current restart approach is inadequate, that Palisades requires either a new process or a full license review, and that this precedent must be addressed now to protect public safety, the credibility of the nuclear industry, and the long-term health of our ecosystem.

A pause and redirection now would not weaken nuclear energy — it would strengthen its credibility. The conservative path is not anti-nuclear; it is the only way nuclear can sustain public trust and play a meaningful role in addressing climate change.

Respectfully,
Kraig Schultz
Environmental Health Advocate
Michigan Safe Energy Future
Grand Haven, Michigan

An Open Letter to The Advisory Committee on Reactor Safeguards
Concerning the Safety of the Palisades Nuclear Plant
September 10, 2025

In January 1986, two NASA contract engineers identified that the Challenger Space Shuttle was endangered if it were to be launched in cold weather. Those engineers used all the professional channels available to prevent the launch. But the bureaucratic inertia within NASA to maintain the launch schedule caused those NASA engineers to be overruled. We all know the outcome of that safety lapse. I write to you today in the spirit of those two NASA engineers as I continue to express my safety concerns to the members of the ACRS. You provide the last possible public safety oversight before resurrecting the Palisades nuclear plant.

First, I wanted to thank you for allowing me to share my concerns about the condition of the diminished integrity of the Reactor Coolant System at Palisades for five minutes during the Palisades subcommittee hearing on August 21, 2025. And I also want to thank you for your thoughtful Steam Generator questions to the NRC staff during the full committee meeting of September 3, 2025. I appreciate that the ACRS appears to be taking its oversight of the Palisades "resurrection" precedent seriously.

That said, new information just placed on the Palisades docket has amplified my previously expressed concerns. I know the NRC staff has not been forthcoming with information for me to analyze as an expert. I fear that the NRC staff has not been forthcoming to the ACRS either. Never in my 54 year professional career have I been more concerned about the integrity of the reactor coolant pressure boundary than I am about the condition of Palisades. Please let me explain.

All operating nuclear reactors are required to provide detailed Steam Generator (SG) Tube Inspection Reports to the NRC identifying flaws discovered during eddy current inspections. Six months after the inspections are completed, these detailed tube inspection reports become available to experts like me in the Public Document Room (PDR). Based on my prior industry experience, I knew that prolonged corrosive chemical exposure from extended shutdowns is deleterious to the metal components in both the Reactor Coolant and Secondary systems. I suspected that degradation was occurring at Palisades after it was permanently closed by Entergy in May 2022 and acquired by Holtec in June of 2022. But I had no hard data from the PDR to support my concerns. The last detailed Palisades SG tube Inspection Report in the PDR is from the 2020 SG inspections performed by Entergy. Five years of tube inspection data on both the primary and secondary systems is lacking from the PDR.

Since Holtec acquired Palisades, it appears to have used regulatory loopholes to avoid filing years of detailed Steam Generator Tube Inspection Reports indicating the extent of the damage. The NRC Staff has even acknowledged that Holtec has failed to provide some Steam Generator inspection details, which is why the NRC staff delayed issuance of the SG sleeving LAR. Here is the NRC's statement about the cause of that schedule delay:

> NRC staff has estimated that this licensing request will take approximately 940 hours to complete. The NRC staff expects to complete this review by September 30, 2025. Due **to**

the eddy current qualification data not being provided by the licensee, the review date is beyond their originally requested date of August 15, 2025. (March 20, 2025, https://adamswebsearch2.nrc.gov/webSearch2/main.jsp?AccessionNumber=ML25076A177)

There are only two publicly available documents that discuss the condition of Palisades SG tubes. The first is the September 18, 2024 Preliminary Notification of Occurrence (PNO) (ML24262A092) issued by the NRC staff based on their concerns after the shocking August 2024 Holtec SG inspection results. The second is a letter containing meeting notes from October 1, 2024 (ML24262A092) between Holtec and the NRC that summarize the August inspection and make vague promises about follow-up analyses. That's it. If additional information is in the possession of the NRC staff, it should also be in in the PDR, and there is no such information. That leads me to the conclusion that the NRC staff is not in possession of some critical Steam Generator tube inspection data from 2024 and 2025.

In your September 3, 2025 meeting, the NRC staff told the ACRS that approximately 3,000 sleeves were inserted into about 700 tubes since May of 2025. Each sleeve is 18 inches long, which means that 4,500 feet of sleeves (0.85 miles!) were installed. That is an astounding length of sleeving and is not supported by the publicly available flaw data from the September 18 and October 1, 2024 PDR documents. For an expert like me, it would be a simple matter to compare the existing 2020 Entergy Inspection with both the 2024 and 2025 Holtec Inspections to search for trends and their root cause of the increased cracking indications, but none of the 2024 and 2025 inspection data is available. However, it appears likely that the tube damage that was identified and sleeved in 2025 exceeded the tube damage that was identified in 2024.

The general rule for plugging is that tubes are sleeved or plugged when an indication has reached or exceeded 40% through wall. So a 20% indication will not be plugged but will be reexamined during the next refueling outage based on Electric Power Research Institute (EPRI) water chemistry guidelines. But the chemical hideout at Palisades is anything but normal. When Holtec did examine the tubes in 2024, it found some previously unaffected tubes had Stress Corrosion Crack indications exceeding 80% through wall cracks after remaining in cold unpressurized water for two years. Slow, anticipated crack growth that EPRI assumes is not realistic for Palisades. Hence 3,000 sleeves, already a huge number, may be inadequate to prevent additional tube failures because of hideout before the next Palisades Steam Generator inspections.

Traditionally, eddy current testing begins several inches above the tube sheet. The tube sheet is part of the reactor coolant pressure boundary which is where chemical hideout would be expected to be most prevalent. Because of this hideout, it is not clear that either the SG tubes or the SG tube sheet will survive for even half a year after Palisades "resurrection" is complete.

Now, new information of degradation has become available. In addition to all the steam generator tube and tube sheet indications indicating both SCC and PWSCC in the steam generator, on August 20, 2025 Holtec filed a series of relief requests (ML25232A195) indicating that it has discovered Primary Water Stress Corrosion Cracking (PWSCC) in at least eight dissimilar metal welds within Palisades Primary Coolant System. The affected welds include indications in two hot leg welds, four cold leg welds and two pressurizer welds.

The record indicates that Holtec did not take samples of either primary or secondary water chemistry at Palisades for two years and also that it is aware that Palisades was not in compliance with EPRI water quality guidelines. Clearly the absence of adequate water chemistry control at Palisades and its effect on the primary coolant system boundary are issues that deserve the thorough attention of the ACRS before allowing Palisades to set a new licensing precedent. This is a generic issue, as there are other decommissioned reactors now in the queue to be resurrected that have also not maintained adequate water chemistry during closure.

The existing evidence suggests that the reactor coolant pressure boundary degradation detected was caused by inadequate water chemistry control at Palisades, which places the facility in violation of two General Design Criteria:

> *Criterion 14—Reactor coolant pressure boundary*. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.
>
> *Criterion 15—Reactor coolant system design*. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

The last time a steam generator tube completely ruptured was at Indian Point more than two decades ago. The condition of both the Primary Coolant System and the Steam Generators is even worse at Palisades with extensive SCC and PWSCC already identified. Luckily Indian Point's design allowed it to dump the radioactive steam into the condenser where it was contained. Palisades does not have this feature and would use Atmospheric Dumps to discharge radioactivity directly into the atmosphere.

Previously, I have seen the ACRS advise the NRC staff and vendor (General Electric) of its concerns that regulatory expediency was placed before public safety. About two decades ago, I was one of a few experts who petitioned the ACRS to evaluate Net Positive Suction Head concerns relating to the request for regulatory relief on Containment Overpressure during Boiling Water Reactor Power Uprates. The ACRS did the right thing then by refusing to allow for the containment overpressure relief which was championed by the NRC staff and GE. I have previously applauded the ACRS personally for making that decision.

My concern initially started with SCC and PWSCC discovered in Palisades' SGs but new Holtec relief requests have identified significant PWSCC corrosion at eight other locations within the reactor coolant system. The loss of the reactor coolant pressure boundary can lead to previously unimaginable impacts to the general public. The ACRS must be keenly aware of what could happen in the event of primary coolant system failure or a Steam Generator tube failure due to years of neglect from improper wet layup by Holtec at Palisades.

I pray that you will thoroughly question the integrity of the reactor coolant pressure boundary and steam generator tubes caused by Holtec's failure to meet EPRI primary and secondary water chemistry standards before allowing Palisades to set a new licensing precedent.

Thank you,

Arnie Gundersen
Expert Witness for Beyond Nuclear, Don't Waste Michigan, *et al.*