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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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NUSCALE SUBCOMMITTEE

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OPEN SESSION

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TUESDAY

FEBRUARY 4, 2020

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ROCKVILLE, MARYLAND

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The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2D30, 11545 Rockville Pike, at 8:30 a.m., Walt Kirchner, Chair, presiding.

COMMITTEE MEMBERS:

WALTER L. KIRCHNER, Chair

RONALD G. BALLINGER, Member

DENNIS BLEY, Member

CHARLES H. BROWN, JR., Member

VESNA B. DIMITRIJEVIC, Member
JOSE MARCH-LEUBA, Member
DAVID A. PETTI, Member
JOY REMPE, Member
PETER RICCARDELLA, Member
MATT SUNSERI, Member

ACRS CONSULTANT:

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

MICHAEL SNODDERLY

ALSO PRESENT:

BRUCE BAVOL, NRR
SARAH BRISTOL, NuScale*
MARTY BRYAN, NuScale
LARRY BURKHART, NRR
MARK CHITTY, NuScale*
JEFF EHLERS, NuScale
SARAH FIELDS, Public Participant*
STEVE HAMBRIC, Consultant*
JOHN HONCHARIK, NRR*
PAUL INFANGER, NuScale
ATA ISTAR, NRR

DAN LASSINGER, NuScale
GREGORY MAKAR, NRR
SCOTT MOORE, Executive Director, ACRS
TONY NAKANISHI, NRR
JOSH PARKER, NuScale
REBECCA PATTON, NRR
SHEILA RAY, NRR
JOE REMIC, NuScale
SUJIT SAMADDAR, NRR
THOMAS SCARBROUGH, NRR
JEFF SCHMIDT, NRR
KEVIN SPENCER, NuScale
OMID TABATABAI, NRR
CARL THURSTON, NRR
KENT WELTER, NuScale
BRIAN WOLF, NuScale
YUKEN WONG, NRR
BRADEN WUTH, NuScale
PETER YARSKY, RES

*Present via telephone

AGENDA

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P R O C E E D I N G S

8:31 a.m.

CO-CHAIR KIRCHNER: Okay. Good morning. This meeting will now come to order. Let me start one more time. Okay. This meeting will now come to order. This is a meeting of the Advisory Committee on Reactor Safeguards, NuScale Subcommittee.

I am Walt Kirchner, Chairman of the subcommittee. Member Ron Ballinger will co-chair this morning's session. Members in attendance today are Ron Ballinger, David Petti, Peter Riccardella, Joy Rempe, Matt Sunseri, Jose March-Leuba, Charlie Brown will be with us in an hour, Dennis Bley, and Vesna Dimitrijevic.

We also are joined by our consultant, Stephen Schultz. And I think on the line we will have Mike Corradini as well, a consultant.

Mike Snodderly is the designated federal official for this meeting. The subcommittee will conduct an area focus review on the NuScale steam generator design, turbine missile analysis, inadvertent actuation block valve, and electrical system reliability.

Today we have members of the NRC staff and NuScale to brief that --

PARTICIPANT: And for those that have dialed on, unless you're speaking, please put your phones on mute. Thank you. We're waiting for the ACRS to dial back in.

MEMBER MARCH-LEUBA: They don't hear us.

CO-CHAIR KIRCHNER: They mustn't hear us.

The ACRS was established by statute and is governed by the Federal Advisory Committee Act, FACA. The NRC implements FACA in accordance with its regulations found in Title 10 of the Code of Federal Regulations, Part 7.

The committee can only speak through its published letter reports. We hold meetings to gather information and perform preparatory work that will support our deliberations at a full committee meeting.

The rules for participation in all ACRS meetings were announced in the Federal Register on June 13, 2019.

The ACRS section of the U.S. NRC public website provides our charter, bylaws, agendas, letter reports, and full transcripts of all full and subcommittee meetings, including slides presented there. The meeting notice and agenda for this meeting were posted there.

Portions of this meeting can be closed as

needed to protect proprietary information pursuant to 5 USC 552(b)(c)(4).

As stated in the Federal Register notice and in public meeting notice posted to the website, the members of the public who desire to provide written or oral input to this subcommittee may do so and should contact the designated federal official five days prior to the meeting, as practicable.

We also have set aside ten minutes for comments from members of the public attending or listening to our meetings. Yeah, NuScale probably in Corvallis cannot hear us. But if one of you could text them, just have them mute their phones so they're not broadcasting.

PARTICIPANT: They are. I did do that --

CO-CHAIR KIRCHNER: Okay. Thank you.

We have not received written comments or requests for time to make oral statements for the members of the public regarding today's meeting.

A transcript of the meeting is being kept and will be made available on the ACRS section of the U.S. NRC public website.

We request that participants in this meeting please use the microphones located throughout the meeting room when addressing the subcommittee.

Participants should first identify themselves and speak with enough volume and clarity so that they can be readily heard.

A telephone bridge line, as we've noticed, has been established for the public to listen to the meeting. To minimize disturbance, the public line will be kept in a listen-in-only mode.

To avoid other disturbances, I request that attendees put their electronic devices, such as cell phones, in the off or noise-free mode.

We'll now proceed with the meeting. I will look first to Ron for this morning's session. We have a full day today. This morning is given over for the steam generator discussions. Ron, have you any further comments?

MEMBER BALLINGER: Not really, no.

CO-CHAIR KIRCHNER: Okay. So, with that, I'm going to turn to the NuScale people. Thank you. Marty, are you going to kick it off?

MR. BRYAN: Yes.

CO-CHAIR KIRCHNER: Please.

MR. BRYAN: Yes, I'm Marty Bryan, the NuScale licensing project manager for Chapter 3. And in the open session today, we're going to cover steam generator design, the steam generator inspection

program. We're going to talk a little bit about density-wave oscillation, and then the ITAAC closure path for density-wave oscillation or DWO, then some results from our preliminary scope of study, and then plant closure activities through ITAAC.

So, with me this morning, I have a number of gentlemen. So, right now, I'll turn it over to Kevin Spencer. And he's going to take us through the first part of this.

MR. SPENCER: Hi. Good morning. I'm Kevin Spencer. And I'll be speaking a little bit this morning about steam generator design.

So the NuScale steam generator is designed to be an integral helical coil steam generator. The primary side fluid is on the shell side of the steam generator. And the secondary fluid is inside the tubes.

The two bundles are made out of Alloy 690 thermally treated. There are about 1,380 tubes. They range in length from 77 to 87 feet long. And we have a 5/8-inch outer diameter of the tubes.

Because of natural circulation on the primary side, we have very low velocities, approximately one foot per second on the outer diameter of the tube.

Within the tube sizing, we have included a ten mils allowance for corrosion and erosion or erosion degradation. Alloy 690 thermally treated --

CO-CHAIR KIRCHNER: May I ask anyone on the phone lines to mute their phones? We are getting interference with the speakers here in Washington. Thank you.

MR. SPENCER: Thank you. As I was saying, there is, in the tube wall sizing, a 10 mils allowance for corrosion/erosion. Of course, Alloy 690 thermally treated material is very corrosion and erosion resistant. Nevertheless, we include some additional wall thickness above the ASME min wall requirement to cover the corrosion/erosion throughout the life of the plant.

The tube design will support 100 percent volumetric inspection of the tubes. So every tube will be able to be volumetrically inspected during outages.

And, additionally, because of the design during refueling, when the NuScale power module actually separates into multiple segments, we will have access to the shell side of the tubes from below the bundle during refueling. This will assist with any evaluations on the shell side degradation, as well

as any foreign material or other inspections that need to happen on the outer diameter of the tubes.

MEMBER BALLINGER: I'm not sure whether this is a proprietary question or not or whether this is the right time to ask it. But you say there's a .01-inch allowance for corrosion and erosion. You know, Alloy 690 is really not susceptible to cracking.

MR. SPENCER: Right.

MEMBER BALLINGER: But its Achilles heel is wear. And in fact, it's about five times higher than Alloy 600, round numbers.

And so, if you, at some point, couple that with potential instabilities where the vibrations and things like that can really get going, is that enough?

MR. SPENCER: Well --

MEMBER BALLINGER: Have you looked, have you really looked at the issue of wear with respect to the overall design and operation?

MR. SPENCER: Yes. We will be speaking a little bit to wear during, a little bit later in our presentation. As far as wear due to DWO, you know, most of the oscillations during DWO are considered to be thermal stresses.

So, other than, you know, maybe some wear due to axial growth of the tube and wear of the tube

along the supports, which we will address a little bit later, we don't expect to have any sort of, you know, high velocity. We don't expect to have any high velocity flows in the tubes.

The primary fluid around the tube is very slow. The secondary fluid on the inside of the tube is well within kind of nominal or -- that we don't expect to have high velocity secondary flows either.

MEMBER BALLINGER: Yeah, I understand that. But the wear issue is coupled, is more dependent on relative movement --

MR. SPENCER: Right.

MEMBER BALLINGER: -- for things like that. And that's independent of some of that --

MR. SPENCER: Right. Wear due to fretting at the supports --

MEMBER BALLINGER: Yeah.

MR. SPENCER: -- would be a concern that we will address.

MEMBER BALLINGER: Okay.

MR. SPENCER: I guess at this point, I'll just jump to the next slide real quick, even though we haven't quite finished with this one.

MEMBER MARCH-LEUBA: Are you going to come back to this one?

MR. SPENCER: I'll come back to this one, yeah --

MEMBER MARCH-LEUBA: Okay.

MR. SPENCER: -- and cover the points. But I'll just going through --

MEMBER MARCH-LEUBA: Well, while you have it there, you think incorporation of operating experience, almost every coal fired plant in the world uses a helical steam generator, right.

How does this one differ from the existing ones, I mean, the ones where we have a lot of experience with, the coal fired plants? Is there anything special about NuScale steam generator that makes it different?

MR. SPENCER: I can't speak too much to coal, coal fired designs. I guess in terms of, you know, previous PWR experience, you know, normally we would have a u-tube recirculating steam generator, that's very common, or a one-screw steam generator, which tends to be stray axial tubes.

MEMBER MARCH-LEUBA: But in the PWRs, the steam is on the shell side.

MR. SPENCER: I don't think on, I wouldn't imagine in the coal plants they would use Alloy 690, but maybe they do.

PARTICIPANT: That is the difference.

MR. SPENCER: Yeah.

MEMBER RICCARDELLA: As well, also I think the diameter of the tubes is significantly different in coal. Coal plants are, you know, much bigger.

MEMBER MARCH-LEUBA: Yeah --

MEMBER RICCARDELLA: Yeah.

MEMBER MARCH-LEUBA: There is a lot of experience. There are thousands of those steam generators working in the field. And we should have gained some experience from them. I mean, from what we should use, what do you know?

MEMBER BALLINGER: Yeah, you recognize that our concern is the fact that minor changes in dimensions and design at San Onofre caused a lot of wear issues.

And it's kind of a threshold phenomena when you come right down to it. Nobody expected that it would exist. And so you're very different in that respect.

MR. SPENCER: Right. To be fair, that was a flow-induced vibration, though. And we don't have the same, you know, we don't have the same fluid velocities that they were experiencing. But I do understand your comment there.

MEMBER BALLINGER: But the source of the vibrations are different.

MR. SPENCER: Right.

MEMBER BALLINGER: But you might have some conditions which --

MR. SPENCER: Right.

MEMBER BALLINGER: -- result in vibrations which --

MR. SPENCER: It increased wear --

MEMBER BALLINGER: -- increased wear.

MR. SPENCER: Yep. So, to the next slide, I'll just go through a little bit about the actual steam generator design here pictorially.

On the left side of the slide, you can see the helical coil steam generator. It's kind of color-coded here.

Now, each NuScale power module will include two independent steam generators. Each steam generator has its own steam isolation valve. And they are completely independent. They have an independent feedwater train, independent steam plenum, et cetera.

The way that these are linked here, you'll see at the bottom, there are three colors. There's a red, a blue, and a yellow. And then behind and not pictured in this would be the green coil. The blue

and the green coils are the same steam generator. And the red and the yellow are the same steam generator.

As each -- at the bottom is the feedwater plenum. So that's where the feedwater is introduced.

There are 21 columns of tubes. Each tube column alternates in the direction of the helical coil. So they go counterclockwise. And then the next row, the next column would go clockwise.

You'll notice at the top of the bundle where we have four steam plenums, you'll notice that while the red, the yellow, and the blue and the green are still segregated into two different steam generator common plenums, you'll notice that the tubes themselves are intermixed. This is to keep the --

So sometimes the steam generator requires an additional half coil or the subtraction of a half coil in order to remain, that that column remains in the overall same length range as the other columns.

Also here in the middle and at the, on the right here we show a little bit about our steam generator support system. Our support system is accomplished by the steam generator support plates. On the bottom middle inset here, you'll see the steam generator tubes and their interaction with the steam generator support plate.

The steam generator support plate has three tabs of support for each tube. Then there is common tabs between the tubes. Those tabs are bent out from the plate material. And they provide a three-point support method for the tube as it curls through and contacts the plate.

In addition, those plates are sandwiched together, which you can kind of see in the upper middle side view of the drawing there, where two plates and their associated tabs will be sandwiched together and in contact to prevent the tubes from moving back and forth or to constrain their growth related to the axis of the helical coil.

The steam generator support plates are held in place by the support assembly that you see on the right. There's an upper plate, or there's an upper t-bar that will be used to hang the plates. And there's a lower steam generator support, which will keep them aligned.

And then the plates are free to move in the radial direction with respect to the ax helical coil. This allows this -- thermal expansion and growth during heat-up and cool-down.

MEMBER RICCARDELLA: Would you give a little more detail? You said three-point support. I

don't quite understand how the tab provides three-point support.

MR. SPENCER: So there are three tabs. So there's going to be, there's -- one tab is either going to be on the top or the bottom of the tube and then two tabs on the remaining side.

MEMBER MARCH-LEUBA: Marty, you can use the mouse to point to the tabs. Is that what you're trying to do?

MR. SPENCER: So down in the lower picture, Marty, on the left-hand side where the yellow and green bundle are, you can see there that we have on -- the top tube is held by those two top tabs and then one tab on the bottom.

MEMBER RICCARDELLA: Got it.

MR. SPENCER: And the next tube on the, below it is held by that, the bottom side of that upper tab, and then two other tabs.

MEMBER RICCARDELLA: Thank you.

MR. SPENCER: Any other questions about the steam generator design? Okay.

So back to the previous slide, Marty. I'll talk about incorporation of operating experience. So we did follow the guidance of NEI 97-06 and EPRI, the COL Item 5.4-1, which is to develop and implement

a steam generator program.

With regards to operating experience, you know, we want to determine what's applicable and not applicable. But we also want to see what operating experience might apply with modifications and any deltas that we may see to any operating experience, any differences and any additional information or any ways in which our tube bundle would be different than previous operating experience.

MEMBER MARCH-LEUBA: But based on your previous, I'm sorry, you did not incorporate any coal fired plant operating experience because those are more applicable to yours.

MR. SPENCER: I can't speak to, I can't answer that question myself. We could look into whether --

MEMBER MARCH-LEUBA: There's --

MR. SPENCER: -- there was operating experience from coal fired plants.

MEMBER MARCH-LEUBA: You'll send an email to Mike --

MR. SPENCER: Yeah.

MEMBER MARCH-LEUBA: -- whether you -- most steam generators in the nuclear industry have the steam on the shell side, where is --

MR. SPENCER: Right, right.

MEMBER MARCH-LEUBA: Coal fired plants have it in the inside. And they manage to run them very well. You need to figure out what's their secret.

MR. BRYAN: Yeah, I think the plan is to look at the relevant experience. And so we'll get back with you on that answer.

CO-CHAIR KIRCHNER: Just for the record, Fort St. Vrain had helical steam generators like this. Of course, that was helium cooled system. But there is some operating experience that's relevant.

MR. WELTER: Hi, this is Kent Welter. So I can make a -- we'll look on the question regarding the coal fired plant. I know from looking at modeling, and we'll talk about that more in the closed session, from thermal-hydraulics, we definitely looked at a wide range of literature for thermal-hydraulics modeling.

From a, I would say from a wear perspective, an operational perspective, we've also taken a look at the Otto Hahn reactor, which is a ship-based reactor. There are older reports. But there's a number of reports affirming what's a degradation, what's destructive and non-destruction of

testing.

So we get a lot of information from, or insights from the Otto Hahn reactor, which actually has a very similar steam generator plenum and stepped-orifice design that the NuScale has.

And it's not the exact conditions. It's a little bit lower pressure, a little bit lower temperature. But that reactor is quite similar to ours. And so we are looking at using some of that experience --

MEMBER MARCH-LEUBA: Maybe you are expecting me to ask the question. That reactor, was it stable?

MR. WELTER: What's that?

MEMBER MARCH-LEUBA: Was that reactor stable?

MR. WELTER: Yeah, and so that actually gave us a lot of information and lessons learned in terms of --

MEMBER MARCH-LEUBA: So --

MR. WELTER: And we can get maybe more of that in the closed session. But, yeah, that gives us a lot of lessons learned.

MEMBER MARCH-LEUBA: Yeah, so everybody else wants us to make the steam generators stable. I

think that is a fact of life.

MEMBER BALLINGER: But the Otto Hahn did not have 690.

MR. WELTER: In terms of a material perspective, yeah, you are correct, yeah.

And so what we did is we looked at -- you know, they have a number of, when you're looking at the non-destructive testing and look at the cracks that formed on the tube-to-tube sheet welds and looking at the number of cracks and the type of cracks and what the potential causes are, we can learn from that and take that into consideration in our design.

MR. SPENCER: So Kent did mention the flow restrictor design at Otto Hahn. And we do have a flow restrictor designed into our steam generator at the tube inlet.

It is a stepped loss, which produces a large inlet pressure drop due to the expansion and contraction of the fluid as it passes through the flow restrictor. This prevents and mitigates density-wave oscillations.

MEMBER MARCH-LEUBA: I need to step in here. The whole FSAR is full of the statements like this, which are very good aspirational goals. But they're completely unproven.

I mean, you say the flow restrictor ensures acceptable criteria are satisfied. And when we get to the operator slides, you'll tell me that at least one of the criteria at least with the preliminary concern with the calculations is not satisfied.

So, I mean, you're contradicting yourself. And we'll talk about this on the classified, not classified, proprietary session. But the whole FSAR is full of statements like this, completely unproven and undocumented.

At best, you'll be able to prove the statement when you finish your ITAACs. But right now, you don't have a leg to -- I mean, you have -- it's a good aspirational goal. But it's not a statement.

MR. WELTER: Yeah, as we get to the presentation, you know, we agree that the work to prove or show that we're going to meet these allowable stress limits is currently under development and part of downstream activities. So --

MEMBER MARCH-LEUBA: But I'm going to recommend to the staff that they review their SCRs with respect to a statement like this and Chapter 5, which say don't bother with it, everything is good. And you have an honest mistake Okay.

CO-CHAIR KIRCHNER: Kevin, for the record, how big of a pressure drop are you taking there? Your nominal steam conditions are what? Our inlet feedwater pressure is what?

MR. SPENCER: I guess if we were to speak to actual data like that, I probably should hold off to the closed session.

MEMBER MARCH-LEUBA: It is on the open session of the FSAR. But let's hold on.

CO-CHAIR KIRCHNER: Okay --

(Simultaneous speaking.)

CO-CHAIR KIRCHNER: Yeah. Okay.

MR. SPENCER: So I'll go to two slides, Marty. So we are required to have a steam generator inspection program. The purpose of a steam generator inspection program is to monitor the performance and the condition of the steam generators during in-service inspections. It is required by the ASME, Section 11 of the ASME code, as well as 10 CFR 50.55(a), bullet g.

We do have to implement the steam generator inspection program according -- and it needs to follow the NEI 97-06 and EPRI guidance regarding steam generator management programs. It must include a lot of different things, such as primary and

secondary water chemistry, evaluations and such.

But particularly today, we would note that the steam generator inspection program will include degradation assessments. This will include wear due to fretting that may be caused by, or that may be attributed to density-wave oscillations, as well as other degradations that may occur by any sort of cracking or volumetric flaws that are discovered.

There will be a tube integrity assessment. The tube integrity assessment will look at, it will be both backwards looking and forwards looking. So it will look at past wear and compare current wear to past wear, as well as it will look ahead for operation and look at wear rates to determine that, you know, to determine safe operation until the next inspection can be performed.

MEMBER BALLINGER: Now, I would -- again, not to beat a dead horse, but 97.06 and the EPRI guidelines and all that are basically evolved historically based on u-tube steam generator design.

And so the frequency of inspection is, would be different or might be, need to be different for this steam generator. So have you given thought to a pattern of inspection times so that you can validate things initially?

MR. SPENCER: Well, I think that we would still be in the planning stage of what we would recommend for the steam generator inspection program, which would be incorporated into the technical specification of the plant.

But preliminary wear assessments that we have are probably going to be pretty, or preliminary wear assessments that we perform show that we have a lot, because we have a lot less wear due to other phenomena, such as turbulent buffeting and fluid loss -- instability, that we have, we probably, we expect less wear.

MEMBER BALLINGER: Well, yeah, but there's no database.

MR. SPENCER: There's no database.

MEMBER BALLINGER: Yeah. Okay.

MR. SPENCER: Right. We would --

MEMBER BALLINGER: So you have to get a database.

MR. SPENCER: Right.

MEMBER BALLINGER: And the only source of a database is your first customer.

MR. SPENCER: Right.

MR. SCHULTZ: Kevin, is this time to be done as part of the ITAAC? The program, the way

you've presented it so far, you've got even calculational work to do and analyses development to do before you determine what the inspection program needs to be. And if you haven't defined the extent of the program --

MR. SPENCER: That is correct. And we will address that in the following slides. We have a path forward that other people will be presenting that will kind of go through that.

In terms of the steam generator inspection program, however, because it's part of the technical specification of the plant, our recommendations to the steam generator inspection program will follow ITAAC. So we would have the, any work that we're, as we're going to discuss that will be performed to close ITAAC will inform the development of the steam generator inspection program.

MR. SCHULTZ: Thank you.

CO-CHAIR KIRCHNER: Kevin, have you, so when you take the module apart, so to speak, and you pull this upper section with the containment vessel and the upper part of the reactor with the steam generator, the riser and the steam generator, you would do visual inspections for wear with probes, remote probes or something.

Have you considered, unlike in a large LWR where the steam generator is some distance away from the reactor, have you given any consideration to what the dose would be for people doing these inspections?

MR. SPENCER: Well, go ahead, Kent. You can handle that.

MR. WELTER: Before I answer that, I'd like to clarify real quick the last statement on slide 4, the steam generator inspection program is developed as part of a COL Item. So that work would be prior to the finalization enclosure of the ITAAC.

Regarding your question on -- forgot it in my brain. Sorry.

CO-CHAIR KIRCHNER: Visual inspection --

MR. WELTER: Visual inspection.

CO-CHAIR KIRCHNER: -- for wear.

PARTICIPANT: And dose.

MR. WELTER: And dose, yes. Thank you. Yeah, as part of the FSAR, we looked at dose calculations for all normal operations and maintenance and inspections. And so we did do that calculation for dose of getting in there and putting eddy current probes up the tubes and inspecting it. We did do that.

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: All these inspections are performed under 40 feet of water. Do you send in divers?

MR. WELTER: No, some are when you're looking at the lower part. The upper assembly gets put into what's called a dry dock area. And we can flood up that dry dock to certain levels to reduce the dose primarily around the flange, which gets kind of hot there at the bottom. So we can flood it up to there. But then we have access ways to get into the plenum where we can inspect not under water.

MEMBER MARCH-LEUBA: So the flow restrictors that you have to remove and put back in, are they put by hand or remotely?

MR. WELTER: It's not a robot, no, if that's what you're saying.

MEMBER MARCH-LEUBA: So --

MR. WELTER: We have a set of rigging and equipment that will be designed --

MEMBER MARCH-LEUBA: No, but there are human eyes.

MR. WELTER: Correct, getting in there and being able to look and take those out and inspect it.

MEMBER BALLINGER: Current LWR or PWR steam generators, they're all, it's all robotic.

There's --

MR. WELTER: There's a certain amount of --

MEMBER BALLINGER: -- specialized tools that --

MR. WELTER: Sure.

MEMBER BALLINGER: -- have been unveiled so you can do a couple of seconds per inspection. Would you anticipate a robot kind of implementation for this kind of inspection, because that's -- if you have to do it by hand, boy, it's going to take a, ooh, long time.

MR. WELTER: Those, the exact way that we're doing it with robots are not as under development. But the way that we did the dose calculations is we had a fair amount of manual inspection.

But we don't currently have a fully robotic system in mind that goes in there and attaches to the plates and inspects them. So --

MR. SPENCER: With respect to the IFR particularly, they would be treated like any sort of removable hardware. We'd have to, like a manway, where you'd have to have someone pull that off by hand. And then once the hardware was removed and the

tubes had entrance, then you could, you know, begin the eddy current inspection using a robotic technology.

MEMBER MARCH-LEUBA: Yeah, my concern is more when you put it back in. That plate has a couple hundred IFRs, maybe 300.

MR. SPENCER: One IFR for every tube. So --

MEMBER MARCH-LEUBA: Yeah.

MR. SPENCER: -- it would be about 1,380 total.

MEMBER MARCH-LEUBA: But each plate --

MR. SPENCER: Right.

MEMBER MARCH-LEUBA: -- right?

MR. SPENCER: Right.

MEMBER MARCH-LEUBA: It's going to be very difficult. I mean, I could not possibly align that many tubes in there. I mean, do you have an aligning tube or something?

My concern, which I'll bring up later, is when you do this operation, you're going to be scratching the IFR and you're scratching the tube. And you are going to change the clearances around the IFR and changing the k-effective, by k-effective I mean the --

MR. SPENCER: -- efficient.

MEMBER MARCH-LEUBA: -- not the neutronic k-effective.

MR. SPENCER: Right.

MEMBER MARCH-LEUBA: And I want to ask you later if you have it all the way to how much degradation you're going to have on the IFR, effective IFR over 60 years of operation.

MR. SPENCER: That would be a good topic to discuss during the closed session, yes. So I think that wraps up my side. And I'll turn it over to Kent.

MR. WELTER: Thank you. Again, my name is Kent Welter. I was going to introduce briefly the phenomenon of density-wave oscillation. We were asked by the ACRS to present on that. So I have a few slides in the open session, and then we have more opportunity in the closed session.

This is -- so, as we've touched on, density-wave oscillation is a phenomenon that is unique to a once-through steam generator or any channel where you have a two-phase flow in that channel from subcooled, from a subcooled liquid to some sort of a boiling.

And the idea is with the right set of conditions, you can have a self-sustaining cycle where

you have flow oscillating. And this picture shows just an example of how that flow oscillation could begin and become self-sustaining.

The reason why it's called a density wave is the idea is if you're starting out with boiling in the channel and you excite the system in some way, say you reduce the inlet flow, you can have a change in the thermal-hydraulics in the two-phase flow if you transfer such that voids would form.

And as that void travels up that channel, or in this case the helical coil steam generator, that can impact the local pressure drop, and that changes the total pressure drop. And under the right set of conditions, you can have a self-sustaining cycle where the flow and the pressure are completely out of phase.

And so it's an important phenomenon to try and design for when you're looking at once-through steam generators. It's a known phenomenon. It's been studied for quite a while.

And I think we touched on briefly that the things that we need to make sure are, first off, from a flow perspective, that you can control the system under those conditions. If you have them, you need to be able to understand when you have the density-wave oscillations and under what conditions.

You try -- as steam generator designers, the idea is to eliminate them and reduce the consequences of them. And that's what the inlet flow restrictor designs tend to do.

And in the closed session, we'll give you a little bit more detail on some of the competing parameters associated with designing an IFR for this particular steam generator. But this is the basic phenomenon.

And what we are concerned about or make sure we design for also is the thermal oscillation. So the flow in itself is not necessarily the most important thing from a once-through steam generator perspective. It's the temperature fluctuations and what does that do on fatigue.

And we've also touched on the wear aspects. We'll also talk about how the velocities inside the tubes, on the outside of tubes are not very high. And we don't really expect any additional wear characteristics that we haven't considered already due to density-wave oscillations.

So what we're primarily focused on is the thermal-hydraulics and the thermal fatigue perspective.

MEMBER MARCH-LEUBA: Let me just bring up

a concept. And we'll discuss it in the closed session.

Whenever you've got these flow-induced oscillations, the steam wave, they always grow very large. I mean, there is no such thing as a small oscillation because there is nothing to stop them. Once they want to go, they go. And you always end up having reverse flow, yeah.

And the inlet, you have the IFR, the inlet flow restriction, which is a very tight flow area and expansion and contraction to get -- where you have a very high velocity during normal operation.

When you have reverse flow, you have that same velocity or maybe twice as much. But now with the same mass flow rate, you have 40, 80, 90 percent void fraction going down the flow restrictor, meaning that the liquid velocity and the flow restriction on the reverse side is probably 20, 40 times larger than on the forward side.

And not only that, it has void. So, in a sense, your IFR is cavitating. Okay. I mean, you have voids. You have 20, 30, 40 times the 3,000 percent velocity that you had in a normal direction. And all that is impinging on that plate that is holding that IFR in a cantilever way from a bolt.

And when we go to the proprietary section, I'll ask you to give me the confidence that you have in that analysis, that that thing doesn't vibrate.

MR. WELTER: Understood.

MEMBER MARCH-LEUBA: But the concept is perfectly open. And then we'll talk about numbers.

MR. WELTER: Sure, yeah. And we'll talk about some of the, from the data we have from the TF-1 and TF-2 experiments that provides us information on what we consider to be a bounding density of oscillation transience.

You need to assume something in the thermal fatigue analysis and the assumptions on the film coefficient right where that inlet flow restrictor is and talk about --

MEMBER MARCH-LEUBA: We'll --

MR. WELTER: -- the details of that in the closed session.

MEMBER MARCH-LEUBA: Let's going over this so we can go into the closed session --

MR. WELTER: Understood.

MEMBER MARCH-LEUBA: But my question on the open is that the IFRs are hundreds of cantilever bars held by a bolt into a common plate which has been impinged by cavitating flow or high velocity.

MR. WELTER: Understood. Okay. Next slide.

As mentioned in the FSAR, density-wave oscillations are mentioned a number of times. In Revision 3 of the DCA, Section 3.9.1 discusses the possibility of secondary flow oscillations.

In our analysis that we've been doing so far, it's primarily in the power ascent and descent or in the low power range, around 0 to 20 percent. That's --

MEMBER MARCH-LEUBA: Keeping it at a very level, and we'll talk about it later.

MR. WELTER: Sure.

MEMBER MARCH-LEUBA: The TF-2 tests have been analyzed by two groups of people that I know of, you and the staff. And you guys disagree with the conclusions. And for the record, I trust your calculation better than the staff because yours was a little more detailed.

But just say that there are two opinions on that statement you just made, that it only happens on the ascent. There is an opinion on paper on your SCR --

MR. WELTER: True, yes.

MEMBER MARCH-LEUBA: -- that this, that

your full power operation is unstable. And we definitely need to address that.

MR. WELTER: Understood, yes. And we recognize that disagreement. We'll talk about that in the closed session. Thank you.

MEMBER MARCH-LEUBA: It has to be cleared up because --

MR. WELTER: Understood.

MEMBER MARCH-LEUBA: -- when I see two different opinions, what I tell anybody is that one of them is wrong and most likely both are wrong because otherwise you would have agreement.

MR. WELTER: Understood. In the FSAR in Section 5, we talk to the inlet flow restrictor design, that stepped orifice design, or not orifice, stepped flow restrictor design and that limits DWO under acceptable limits where acceptable limits are ASME stress calculations or fatigue limits.

And Kevin will talk a little bit about more what the ASME stress calcs that we're doing.

PARTICIPANT: Joe.

MR. WELTER: Joe. I'm sorry.

MEMBER MARCH-LEUBA: On this point, I'm going to reinforce what I said before. The FSAR has the blanket statements about what may or may not

happen after you finish your calculations five years from now. And I don't think it's appropriate to have that language on the, without any basis.

MR. WELTER: Understood. Well --

MEMBER MARCH-LEUBA: Section 5.41 --

MR. WELTER: Yeah.

MEMBER MARCH-LEUBA: -- I mean, it's wishy-washy. I mean, everything is just one sentence -- everything is good.

MR. WELTER: Understood. It has those --

MEMBER MARCH-LEUBA: And I don't have any document or any calculation that backs up that statement.

MR. WELTER: Understood. And we'll talk a little bit about the preliminary work that we're doing to meet those limits as part of the downstream activities.

MEMBER MARCH-LEUBA: Very simple, FSAR said as part of ITAAC so-and-so, we will ensure that the flow restriction is doing that, instead of saying the flow restriction, that's it. We're wanting to demonstrate it, but we have not yet.

MR. WELTER: And I think we recognize that. And we did visit with the staff regarding the, to my last bullet there, the specific language in the

DCA.

MEMBER MARCH-LEUBA: I would strongly encourage that language to be changed, because as it is, it's not accurate. It's likely to be. But it's not accurate.

MR. WELTER: Right. And --

MEMBER BLEY: That's true. But there's a lot of other places in the FSAR the same thing applies.

MEMBER MARCH-LEUBA: I'll leave that to you.

MEMBER BLEY: I can't do the whole thing. But there are quite a few places where this, not, this is not what we expect, this is the way the world is, and nothing to back it up.

MR. WELTER: No, but I agree. And in terms of the language related to the IFR design and the use of RELAP, we're planning to revisit that and see if we need to make some changes. And we've started that discussion with the staff.

MEMBER MARCH-LEUBA: Let me put another concept, which I don't think is proprietary. The use of RELAP for secondary side instability calculations is explicitly forbidden on your topical report that we will be reading in a couple of weeks. I mean, it's a

limitation of the topical report. So we need to address that, too.

MR. WELTER: Yeah, and we --

MEMBER MARCH-LEUBA: You're using a tool that that guide explicitly forbids you from using for this application.

MR. WELTER: Right --

MEMBER MARCH-LEUBA: And I keep forbidding not because it's bad but it's because he didn't review it for this application.

MR. WELTER: Understood. And so that's why we have here in the third bullet this, I believe you're talking about the stability topical report.

MEMBER MARCH-LEUBA: The non-LOCA --

MR. WELTER: The non-LOCA. I'm sorry. Yes.

MEMBER MARCH-LEUBA: -- has a limitation on ASA. And RELAP is good for everything except the secondary side instability. Is it the non-LOCA or is it LOCA? It's one of the two, both.

MR. WELTER: Right. There is some certain limitations on RELAP. And in the closed session, we'll talk about the planned use for RELAP forward, analyzing the density-wave oscillations, some of the reassessments we did, and the limitations of RELAP and

how those need to be addressed. Correct?

Okay. So you -- so I think that's it for me on the open session. So I'd like to turn it over to Joe now.

MR. REMIC: Thank you, Kent. Good morning. My name is Joe Remic.

I will be talking to you a little bit today about NuScale's plan associated with DWO in regards to our ITAAC closure plan. I'll then be going into briefly ASME design specification and design reports. And then I will be closing out my portion of today's presentation with some of our preliminary results.

So, first, ITAAC 02.01.01 requires the inspection of an as-built ASME design report for a variety of different ASME code class components, 1, 2, 3, as well as a variety of other things.

So what does this mean in relationship to the steam generator for us? Well, Table 2.1-2, and as part of our tier 1 document, identifies the ASME code class of numerous components.

And as you can see here from this snippet, both the RCS-integral RPB and the steam generator are classified as ASME code, Section 3, Class 1 components. Therefore, ITAAC 02.01.01 will require

the inspection of an ASME as-built design report for the steam generator, as well as the corresponding components of the RPD.

MEMBER RICCARDELLA: What do you mean by the word inspection in those bullets?

MR. REMIC: So --

MEMBER RICCARDELLA: I mean, you have to do preparation of the, of a design report. You have to have a third-part review. But the inspection --

MR. REMIC: So the inspection --

MEMBER RICCARDELLA: -- by the staff you mean?

MR. REMIC: Yes.

MEMBER RICCARDELLA: Oh, review, okay.

MEMBER MARCH-LEUBA: In that spirit, and I would ask the staff to prepare for their presentation, when I was reviewing this slide -- because this is the first documentation I have of how this is going to be solved is this slide. And there is no documentation anywhere else. It's not on the SCR. It's not on the FSAR. It's on this slide.

I was looking at this. And I just don't feel -- and this is my gut feeling, which is not approved for use in the laboratory environment.

But my gut feeling is that this topic is

not an ITAAC topic. This is more like an open item. I mean, I would call this an open item. This is not an inspection and test. This is we have not finished our analysis.

And you move to ITAAC items like you don't know where the cooling tower is going to be because in some plants it will be in the northwest and some plants, some sites it will be in the northeast.

But this has still been an issue we haven't decided -- is finalized. There is no excuse not to have the analysis of the -- certification.

I mean, this is an open item in my opinion. And I just said my opinion. While I say this, this is a subcommittee meeting. All you're hearing here is individual members' opinions. If you are in this room, you have this oral --- it's good to put it on the record.

CO-CHAIR KIRCHNER: No, I did say in the opening statement that we only, our conclusions, et cetera are done through our letter reports.

MEMBER MARCH-LEUBA: Yeah.

CO-CHAIR KIRCHNER: So, yes.

MEMBER MARCH-LEUBA: Whenever you hear some of these things and when we write a letter, we remember what we said. We might include it. And then

there will be a vote among the committee whether we, the committee agrees or not. So what you're hearing from me is my opinion.

MR. REMIC: Okay.

MEMBER MARCH-LEUBA: But my opinion is that this is not an item, this is not an item.

MEMBER SUNSERI: Is this ITAAC a common to all design certifications, or is this a specific thing for NuScale?

MR. BRYAN: Yeah, this is, my understanding, common to all from the ones I've looked at --

CO-CHAIR KIRCHNER: This is an ASME code requirement.

MR. BRYAN: Right, right.

CO-CHAIR KIRCHNER: This is not a design issue per se. It's, as you just presented, this is -- I'm not an expert on the code. But I'm familiar with it. And this is a Section 3, Class 1 vessel. So this is a requirement to meet the ASME boiler and pressure vessel code. It's not unique to NuScale.

But I share Jose's concern that this is not addressing the design issues. This is just meeting the requirements of the ASME code.

MEMBER BLEY: It is. But --

CO-CHAIR KIRCHNER: Yeah.

MEMBER BLEY: -- they are going back many years on these design certs. We have argued that at other times. There is a whole series of IEEE requirements that have to be documented in the reports. No reason some of those couldn't have been done earlier. But they're done as ITAACs. So there's, ITAACs are full of this sort of thing --

MR. REMIC: Just --

MEMBER BLEY: -- whether we like it or not.

(Simultaneous speaking.)

CO-CHAIR KIRCHNER: No, this -- I think Jose's point is subtly different. This slide, this ITAAC to do the ASME code requirement would apply to anyone who had this kind of vessel under Section 3, Class 1.

MEMBER BLEY: It's no different than the --

CO-CHAIR KIRCHNER: It's no different, yeah. But I think what you're getting at --

MEMBER BLEY: In the ITAAC, there should be a report that documents it.

MEMBER MARCH-LEUBA: There is an expectation that the pressure vessel would satisfy the

ASME code, Class 1. Right now we don't have any evidence that the steam-to-flow instability would satisfy ASME 1.

Before I certify your steam generator, I should have some expectation that it will work. And therefore, I should have some of these calculations done.

I mean, you know the difference, because when you have a steam line, there is an expectation that we know how to do them and there is no problem. We are raising issues about very large flow oscillations in the steam generator which don't happen in any other steam generator or in the majority of the steam generators in the world. And therefore, this requires a little more effort.

MR. REMIC: No, I understand. And I appreciate that. I just -- the level of pedigree of our analysis is consistent with other applications historically in the past, in which what we plan on doing and what I'll be discussing here today is, as you indicated, we are going to be performing the full Section 3 evaluations of these components.

And we will be documenting all of those requirements in an ASME design report. But as you can see here, in regards to the DWO loading specifically,

Subsection NCA of the ASME code provides a series of different requirements on what needs to be included in both ASME design specifications and ASME design reports.

So, in order to connect the dots, the ITAAC requires the inspection of an ASME as-built design report. In order for an ASME as-built design report to occur or to be constructed, there needs to be an ASME design specification.

Within that ASME design specification, as you see on this slide here, according to NCA-2142.2, a subsection of the ASME code, all corresponding loads need to be defined, both thermal, mechanical, the corresponding cycles, the service level conditions, as well as the load combinations.

So, within our ASME design specification for the RPd and the steam generator, we will have to have the DWO event defined as part of that.

MEMBER MARCH-LEUBA: Right. But this is what -- go ahead.

MEMBER RICCARDELLA: You know, if I go back early, I spent the first ten years of my career preparing ASME Class 1 stress reports. And that, of course, was under the Part 50 applications.

But, I mean, they weren't done until the

vessels, the steam generators were being built. I mean, it's the vendor who -- you know, you issue a contract. You're going to build the vessel. They're the ones that are responsible to prepare the design report.

But what is your responsibility I think is to prepare the design specification, as you said, that defines the loads. And you would have to define these density-wave oscillations as part of that design spec. Is it the intent to do that prior to certification, the design spec? No?

MR. WELTER: Currently, it's not part of -- it's downstream activity.

MEMBER RICCARDELLA: Okay.

MR. WELTER: Yeah, it's separate from that. It's part of the downstream activities that would need to be completed for the ITAAC.

MEMBER MARCH-LEUBA: Let me oversimplify. People get upset when I do that. But basically what I hear from you is stress test will do a good job later. That's what I hear.

MR. WELTER: And that is the basis of the ITAAC concept, where you are making future commitments that need to be addressed.

MEMBER MARCH-LEUBA: What I see missing

here from the point of the staff is there has to be an agreement on what the methodology will be to define those loads.

For example, you never even thought about the reverse flow on the IFR with two-phase flow going through it. I mean, this is a concept you just realized right now. I'm making a statement. If I'm wrong, let me know.

MR. BRYAN: I believe that's an inaccurate statement.

MR. WELTER: That's incorrect. And we have considered reverse flow and --

MEMBER MARCH-LEUBA: And you -- the reverse flow with very high void fraction and all those loads in the plate.

MR. WELTER: It's part of our methodology to look at that, yes. Yeah, again, we're --

MEMBER MARCH-LEUBA: So --

MR. WELTER: We're looking at the data and the operating experience from other things that we have lessons learned from and incorporating that in the methodology. And we are considering reverse flow.

MEMBER MARCH-LEUBA: I haven't seen it on your slides. Do your slides say I put a swear word in temperature and it failed?

MR. WELTER: On one of them, to be fair. On the other one, it passed and one where it did not. So --

MEMBER MARCH-LEUBA: But all I've seen on the FSAR is this IFR is a very stiff rod. It has very high frequency and musn't vibrate.

MR. WELTER: Right. There's a level of detail that's not in the FSAR regarding the methodology. The methodology is not --

MEMBER MARCH-LEUBA: My point where I'm trying to get to is that the use of RELAP for those calculations or TF-2 data, if it's from banding or TF-1 or TF-7, whatever it is, should be, in my opinion, it should be a topical report like it's agreed -- the methodology that you would use to calculate the ASME code limits should be agreed upon with the staff. And you cannot make it up and just give me the results.

Somebody that actually is overseeing over your shoulder and has no concern about how much it costs should look at it and ensure that the methodology is sound. And the way I see the path foreclosure, there is no review of the methodology.

MR. WELTER: So I do have a slide in the closed session. And there's a bullet. I think it's slide 21, that I can talk in more detail about an

opportunity for the staff to review the methodology prior to that ITAAC closure.

MEMBER MARCH-LEUBA: Yeah, it mentions they have an opportunity --

MR. WELTER: They have an opportunity to do that, correct, yeah.

MEMBER MARCH-LEUBA: Yeah. I would rather to see a commitment on the topical report for review. I mean, this is important. This is the only steam generator in the world that is planning to operate in a unstable mode.

MEMBER BALLINGER: But to circle back a little bit on the materials issue, in the, and with respect to the code, what got the San Onofre people was the fact that there were some words, and I'm paraphrasing, in the code that says as long as your design is within a certain envelope of history you don't have to do extensive experiments. And I'm paraphrasing.

What happened was the design was outside the, it turned out to be outside of the experience. And under those conditions, the code requires additional experiments and validation.

Is your design within the envelope of existing data for tube wear and things like that? Are

you making that claim, and therefore, you don't have to worry about doing additional experiments to verify the wear models? I could probably find the exact words --

MR. WELTER: No, I'm try to get --

MEMBER BALLINGER: -- in the code. But --

MR. WELTER: So we are doing -- so I think what we're trying to do in the conversation is make distinctions between what we need to consider from a flow-induced vibration perspective, and we are doing testing for that right now, and the tests that we have done already for density-wave oscillations that again are primarily looking at thermal-hydraulic conditions at which they would occur, and the effectiveness of the IFR design, and making sure that we have the right thermal transient for fatigue calculations.

MEMBER BALLINGER: But this is --

MR. WELTER: They're two separate --

MEMBER BALLINGER: This is kind of a little gray area where the cause of the wear might be different than the cause of the wear in a u-tube steam generator. And so --

(Simultaneous speaking.)

MR. WELTER: And we've done --

MEMBER BALLINGER: Okay.

MR. WELTER: -- significant testing to look at, in considering wear and providing the allowances for this specific steam generator.

MEMBER RICCARDELLA: And that's TF-1 and TF-2. And you're pursuing TF-3, right?

MR. WELTER: Currently TF-3 for the FIB testing. That's correct.

CO-CHAIR KIRCHNER: I propose that we let the NuScale staff go through the remaining slides because I think they will address some of the questions or at least tee them up for the closed session. Could we do that? Is it --

(Off mic comments.)

CO-CHAIR KIRCHNER: All right. Please.

MR. REMIC: Thank you. So once the DWO transient definition as well as all the other transients and service conditions are defined within the design specification, an ASME design report will be constructed.

That design report, as you can see from the various subparagraphs identified on this slide, will require that a registered professional engineer certify that all loads and all load combinations defined within the design specification are appropriately met.

And that registered professional engineer has to be competent in the field in which they're performing the evaluation that's being performed.

And as well, as you can see on the third sub-bullet there underneath design reports, that in order for us to be able to apply our N stamp to our RPD, an ANI inspection will also be occurring on the associated design reports.

So all this together means that the DWO loading will have to be defined within the design specification, the corresponding loadings, the load combinations associated with that event, as well as a variety of other events on the steam generator, will then have to be analyzed successfully in a design report.

And then that design report will have to consider any as-built deviations. And the inspection of the completed as-built design report will be handled by ITAAC 02.01.01.

I now would like to go over a little bit about our preliminary results that we have handled thus far.

So specifically, the steam generator tube, so we have performed a preliminary evaluation on the steam generator tubes. We have modeled the entire

length of the steam generator tubes.

We applied the primary side conditions on the primary side. And then we assumed DWO full flow reversal. We assumed the entire length of the tube was in the subcool region. And then the entire length of the tube was also then in the super-heated region. And we oscillated that for a variety of different power levels.

And what we were able to conclude for the steam generator tubes in this preliminary evaluation is that the resulting, or excuse me, the resulting alternating stress of the, due to DWO on the steam generator tubes is below the endurance limit.

And what that effectively means is that the steam generator tubes can withstand an infinite number of cycles of DWO without incurring any fatigue damage.

MEMBER SUNSERI: This doesn't address Professor Ballinger's questions about wear.

MEMBER BALLINGER: That's the gray area.

MR. REMIC: That is true. This evaluation was highly focused on thermal fatigue. We had performed separate evaluations concerning the mechanical stresses imparted due to vibration and other thing, other such loadings due to DWO. And we

concluded that they were negligible in nature.

However, wear on the steam generator tubes outside of DWO, as Kevin and others have indicated, will be considered and has been considered. And we are --

MEMBER RICCARDELLA: We had a whole day-long presentation on that by the professor from Penn State. Do you remember that? I mean, they've done extensive flow-induced vibration work. And --

(Off mic comments.)

CO-CHAIR KIRCHNER: Well, Pete, that was flow-induced vibrations, which --

MEMBER RICCARDELLA: Yeah, but I think we're conflating two different subjects here. DWO is different. That's a thermal-hydraulic phenomenon versus the wear --

CO-CHAIR KIRCHNER: No, but the concern for the DWO in the tubes is that they will shrink and expand. And then you have a three-point clip that is holding up the tube.

So, you know, normally if it was just steady state, you would expect that the helical coil tube would expand very tight fit against those support structures and clips. But if it's oscillating, then you have to be concerned about whether that becomes a

migratory or a wear issue --

MEMBER RICCARDELLA: Wear due to DWO.

CO-CHAIR KIRCHNER: -- at the support. And that was not discussed --

MEMBER RICCARDELLA: Yeah.

CO-CHAIR KIRCHNER: -- previously.

MR. REMIC: Yes, so I know that we do have plans to finalize evaluations concerning wear due to thermal expansion. I do believe that our conclusions, as indicated by Kevin and others on the NuScale staff, are such that the wear due to DWO will not be -- we will -- we are considering that, but I cannot comment on the pedigree of that evaluation at this time.

MR. WELTER: Yes, we have done some preliminary calculations to look at that and we'll see if we can't get someone on the line to talk a little bit about that in the closed session.

CHAIR KIRCHNER: And that would have to go into your specifications for the ASME report?

MR. WELTER: Yes, we didn't prepare the slides associated with that, but I'm going to see if we can't get some more information on that specifically.

And just at a high level, although we have this oscillatory nature, I mean when you look -- and

we'll talk -- Kevin will talk about from a fatigue perspective it's a quasi-steady state kind of a thing. It's not -- the frequency at which is oscillates is not something that we would be particularly concerned with from a wear perspective. It would be more of a second-order effect when you're comparing it to wear associated with flow-induced vibration. I mean it's not -- we have looked at it. It's not something we're concerned about because of again the quasi-steady state nature of those temperature transients and because -- yes, anyway.

MEMBER MARCH-LEUBA: It was a second -- it's not --

MR. WELTER: Yes.

MEMBER MARCH-LEUBA: It's not the 100 hertz, right?

MR. WELTER: I know, but it's not like you're chugging in an -- it's a different type of phenomenon, the wear associated with the outside of the tubes and the flow-induced vibration and the wear that potentially could be caused by the fluid in the columns going up and down and creating a thermal transient. And so they're different. And so I'm going to see if we can't get someone on the line to try and make that distinction a little bit more --

(Simultaneous speaking.)

MR. SPENCER: I can speak to some of the general effects of it.

MR. WELTER: Okay. Maybe Kevin can.

MR. SPENCER: With respect to wear due to DWO, one thing that we -- I've identified during our preliminary scoping analyses is that the temperature fluctuations happen kind of along the length of the tube. Because of the helical design in a rough sense each tube is making circular orbits. The supports that we have are -- act on a very small part of the tube, so there's a very large span from support to support where the tube is loose and able to displace.

Thermal axial growth is rather limited by the temperature fluctuations that we actually see during the DWO transient. It's also kind of counterbalanced since the whole is in place -- is growing axially at one time rather than just a portion of the tube. We tend to see that there is some sort of balance loading that you get between tube segments. And additionally, because of the flexible length of tube between the supports most of that additional stress is taken up as additional bending.

So we see that -- we expect generally that given thermal loadings -- changes in temperature of

the tube on two sides of the support will provide some sort of balancing effect with respect to the reaction forces at that support, and most of that thermal stress will be taken up as additional bending in the free span of the tube.

MEMBER MARCH-LEUBA: Maybe it's because I don't know anything about this; and I will let Dr. Ballinger worry about it, but I'm not worried about the -- I don't think we're worried about the stress or strain on the tube. We're talking about the tube is loose, but it's rubbing. So there is nothing preventing it from moving. And you're talking quasi-steady state, 100-150 degrees change in temperature every second. And if it's not -- if that support is not very tight, it will move and rub. It can.

MR. SPENCER: Yes, it can. It certainly can move and it can cause wear. Wear due to motion tends to be self-limiting. With regards to interactions between tubes and tube supports a lot of times the wear that you see there is self-limiting, meaning that it will wear and then it will stop wearing. So it's not a continuous wearing through the wall of the tube.

And then that again goes back to our steam generator inspection program where we'd be monitoring

the wear rates and we would be looking for increased n rate, or we may see wear that happens once and as the steam generator kind of wears into its support.

But in terms of the wear due to DWO, because the thermal axial growth of the tube is occurring on both sides of the supports, we don't see that there's very much differential motion there between the two, especially considering the long unsupported span which can take up the additional strain in terms of a bending stress.

MEMBER MARCH-LEUBA: The important thing is you tell me have analyzed it, which I haven't seen an analysis.

MR. SPENCER: Right, that's been included in our preliminary work that we're kind of discussing today, but that preliminary work has not been subject to inspection at this point.

MEMBER MARCH-LEUBA: So and your expectation is if the staff wants to audit your work under Corvallis, or is this going to be part of the certification?

MR. SPENCER: I'm sorry, could you --

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: Yes, your expectation about this is you plan to do the work in Corvallis and

if the staff wants to go and audit it, they're welcome to come, or is this part of the review?

MR. BRYAN: Well, I think from our perspective the review is completed. That would have to be addressed elsewhere. I think we would have to discuss that. It's not our plan that it would be inspected per se, but it's part of our normal design completion activities.

MEMBER MARCH-LEUBA: Yes, but you will agree that -- everybody will agree that it's not desirable to have this oscillations happening every one second inside your steam generator. It's not. So it's an unusual circumstance in this particular design. So whereas the staff is not going to audit the ASME Code for the steam line because they trust you. On this one they should. I mean it's just an unusual circumstance. I'm just leaving it for --

MR. SCHULTZ: We're talking to the staff next, but they would -- the other alternative would be for the staff to conclude that this just isn't a problem and therefore the place where NuScale is with regard to the design is not of concern.

But I agree with you, Jose. More needs to be done.

And you've talked about inspection in a

number of different ways: professional engineer evaluation, ASME requirements, audits, inspections and so forth. It would be good to get all of that clear so we know who's going to see what when with regard to work that's still ongoing.

MEMBER BLEY: It's not completely relevant, but it is to an extent. Back in the '60s and '70s when we started having problems in U-tube steam generators, we solved it and we changed the chemistry. And then it happened again. And then we changed the chemistry. We thought we solved it at least two or three times. Eventually they got there. We had condenser tube problems with corrosion. Changed the materials. No more corrosion problems, but all of a sudden we got fretting problems ripping the tubes apart.

I think Ron's point earlier about how you lay out the surveillance and inspection programs once one of these starts up and we track it real closely is really crucial to not having events that damage your reputation and the industry's. And that doesn't say anything about trying to get this analysis part right, but that might not solve the whole problem.

MEMBER BALLINGER: And I would -- again basically your ejection seat in this thing is that you

can change the inspection interval. And if it's well designed to start with, you'll avoid -- head off issues.

MR. WELTER: Yes, and we recognize that. And this is information that's used for developing the steam generator inspection program, and because it is a unique steam generator and although we're doing a lot of work from a design perspective to ensure its full 60-year life, we will have to get operating experience and inspect every cycle to figure out -- looking where the wear is. It's 100 percent visual inspection. The eddy current probe is up there and every cycle we might have to update the inspection program for the next cycle.

So we fully expect -- which is a COLA item, the inspection program. So we fully expect we're going to be taking this apart and looking at it every cycle and then adjusting the inspection program as needed. So we don't I think claim to have all the information today, but that's the order of activities, the downstream activities to ensure that we put a steam generator in operation and that we're looking at it and making the necessary changes as needed when we get operating data.

And that's the plan that we've outlined

here in the open session to talk about that through designing the steam generator, completing ASME design specs, the ITAAC reports and developing a steam generator inspection program.

CHAIR KIRCHNER: Would you please proceed to 13 and -- I think we have two slides to go, and we can finish this and then follow up the detailed questions in the closed session, please.

MR. SPENCER: So our plant closures activity is through ITAAC. We have the design report that we'll be issuing. So here's kind of how we will accumulate all of the data that we'll need to inform the ASME Code Section 3 Design Report.

So entering that five, we'll provide thermal hydraulic time histories of the design transience to support our fatigue analysis. These include fluid temperatures, pressures, nominal flow rates, quality. And this is the general design transient inventory not specific to but inclusive of DWO. We'll use the TF-1 and TF-2 test data to provide a basis for the parameters that we'll use to calculate the film coefficients and resulting stresses during DWO.

Because the thermal response of the tube -- in other words, the thin wall of the tube lends a

very fast thermal response to temperature changes due to changing fluid conditions inside the tube. That thermal response is much faster than the DWO frequencies that we've seen as a result of the TF-1, TF-2 testing. So we can feel confident that we can perform a quasi-static analysis on the stresses of the tube.

In other words, we can take the peak -- the stress correlating to the peak of one BWR oscillation and the stress state corresponding to a trough of a DWO oscillation. We can just look at the changes in stress between the peak and the trough of the oscillation as a quasi-static state. We can let the tube fully come to those thermal conditions and look at the differences in stress.

When we combine that stress with all other loadings and all other transients, we're going to be comparing that to allowable stress limits as prescribed in the ASME Code. And then the goal of our final ASME calculation of course is to confirm that any alternating stresses on the steam generator tubes, the tube to tube sheet weld and the tube sheets, all steam generator components due to DWO stresses will be below the ASME endurance limit. Therefore, with respect to DWO we would be able to withstand infinite

cycles of thermal oscillations.

MEMBER MARCH-LEUBA: Reading in between lines that you only plan to evaluate the expansion and contraction of the bulk of the two because of changes in temperature. That's what you're telling me there. And we were talking before about reverse flow through the IFR with significant -- and tremendously high velocities impacting on the plate that supports those IFRs.

Have you done any analysis? You told me a month ago that you considered it and that you've done a significant analysis and decided that was not a problem. I think it is.

MR. SPENCER: So I can -- I will specifically say that we have considered that we would have full reversal flow at the tube sheet even though that has not been -- even that kind of flow has not been seen in any of the previous testing that we've done. We haven't seen --

MEMBER MARCH-LEUBA: And you will consider the impact on the expansion of the tube for thermal -- the temperature change?

MR. SPENCER: With respect to the actual tube analysis.

MEMBER MARCH-LEUBA: I'm worried about the

localized cavitation --

MR. SPENCER: Right.

MEMBER MARCH-LEUBA: -- of the --

(Simultaneous speaking.)

MR. SPENCER: So I think --

MR. WELTER: Can I jump in real quick? So, yes, we actually have an activity that we need to look at and disposition the -- from a CVAP perspective making sure that that high velocity doesn't introduce any additional flow-induced vibration and wear. We have to do that and it is part of our program. That is correct.

MEMBER MARCH-LEUBA: Because that wear --

MR. WELTER: Yes.

MEMBER MARCH-LEUBA: -- will have -- your IFR effectiveness?

MR. WELTER: Right, and that's a small tolerance there and we've designed that such that it -- we have confidence that it won't be an issue, but that is in our program to evaluate flow-induced vibration right there.

MEMBER MARCH-LEUBA: I don't see it there.

MR. WELTER: It's not this slide, but in the -- I have to double check, but on previous presentations there is -- when we talked about flow-

induced vibration, there's that bypass flow, and that's on some of that information. I don't know if we -- it's not here today, but we have --

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: My point; to my colleagues more than you, is that I see the necessity of developing a methodology to evaluate this problem, have you review by the staff, by somebody else, not just by you. Because the plan I see for you is you're going to calculate the strain and you're going to say it's 22, which is less 43, and therefore it's okay. And that's all they're going to see.

I think we need to agree on a methodology by which you can play that 22. And I don't see a path or a desire on the part of the staff of doing that, certainly not yours -- because you're the one paying for it. But I see a need for that.

MR. WELTER: Well, just to reiterate, we do plan on developing a methodology and we're providing some preliminary results of that methodology. Staff will have an opportunity to audit that as downstream activities.

MEMBER MARCH-LEUBA: All right. We'll talk to the staff about their plans for that.

While I have the microphone and we're only

five minutes over, we have not talked about -- you don't seem concerned out of operational concerns. The fact that the flow is going to oscillate -- I will call it dramatically. I call it violently. You can it small. But it's a large flow space, okay, by any standard.

You don't think that's an operational concern and your main argument -- let me apologize. Only Brian and Peter are going to understand what I'm going to say. The rest try to keep up. Your main argument is that the oscillations are going to be out of phase. And physics wants to do that because the left channel likes to borrow flow from the right channel because it doesn't cost any money in friction terms. And if you have to come from the pump, it costs friction, costs money. So it's cheaper to borrow money from the right than from the pump.

However, that is in normal systems where the inlet flow restriction is not outrageously large. To borrow money from the right channel I have to go through the IFR, which is an outrageous friction. And therefore this fact that we all know that out-of-phase oscillations are preferred by nature over in-phase oscillations, I don't buy it in principle. You need to prove it.

And we'll go into the; I keep saying classified, I mean proprietary, on the TF-2 design. I'd like to know what type of pump that we're using. Because with this flow -- you're following me, right? The flow restriction is not what we have in a BWR or a normal experiment -- It's infinite. To borrow flow from the right channel I have to pay a lot, whereas the interest from the pump might be less, especially when the flow control valve is fully open at 100 percent power, and there is no friction there. And maybe at low flow when the valve is almost closed there will be a lot of friction and then the out-of-phase will be preferred.

So this blanket statement that is on the FSAR, and it went conducted on the SAR, that you don't have to worry about in-phase oscillations, I don't buy it. It's unproven.

MR. WELTER: I understand and when we get into the proprietary section, we'll talk about the data and how we did not -- we observed out-of-phase oscillations.

MEMBER MARCH-LEUBA: While we're waiting for that can you -- if Corvallis is listening, can you find out what that pump was the TF-2 experiment using and what type of flow, feedwater control --

MR. WELTER: Yes.

MEMBER MARCH-LEUBA: -- and you can tell us in an hour?

MR. WELTER: Sure. And we'll also talk -- I'll talk more specifically about the type of feed reg valve we have in the plant, which is an anti-cavitation device, and the variable speed feedwater controller we have and provide some more detail on that for --

MEMBER MARCH-LEUBA: You have a variable speed feedwater --

MR. WELTER: Yes.

MEMBER MARCH-LEUBA: It used to be a valve.

MR. WELTER: Well, we have both. I'll talk about the feed reg valve and the feedwater pump and how we control it at the low powers and so forth.

PARTICIPANT: Yes, I think they need at the --

MEMBER MARCH-LEUBA: Yes, absolutely. But they keep telling us that my valve will do it.

CHAIR KIRCHNER: So we left off with Joe and Kevin.

Marty, are we pretty much through? You have one slide.

MR. BRYAN: Yes, this is the last slide. We'll deal with the rest of it in the closed session.

So what we try to prepare is the process by which the successful completion of the ITAAC by the licensee constitutes the basis for NRC determination to allow operation of facilities certified under 10 C.F.R. 52. So that's our conclusion. We'll talk more in the closed session of how we go about that.

Any other questions?

CHAIR KIRCHNER: Go ahead, Pete.

MEMBER RICCARDELLA: Just to help me with my understanding. My understanding is that this density wave oscillation, if it occurs is only going to happen in certain regimes of operation, right? It's not a full-power thing. It's only as you go through a start-up or something.

If it's occurring and if it's as violent as Jose seems to think it's going to be, will the operators know? Is there any indication that you'll have to know that it's going on? Any instrumentation or I don't know --

PARTICIPANT: Noise.

MEMBER RICCARDELLA: Noise. Yes, vibration.

MR. WELTER: First off, I think it's not

as -- NuScale doesn't believe this is as violent as maybe others think, and we'll try and talk to that in the closed session, and that it doesn't really propagate beyond the feedwater plenum. And then our reg valve and our variable-speed feedwater pump can accommodate that.

In terms of instrumentation, we don't have any specific instrumentation currently in the plant that would detect DWO at a high level of confidence, if you will, that it's occurring or not, and frankly because from an operational perspective DWO is not really an issue based on the current design and the IFR concept. And in addition, if we show that we're under the fatigue limit, it's not a fatigue issue as well.

So we have considered it in our design. And remember our control system is not designed yet, but we need to take into consideration the presence of DWO in the development of our control system. But, yes, currently there is no specific instrumentation for monitoring or detecting DWO. And we haven't identified a need yet.

MEMBER RICCARDELLA: Well, part of defining what the fatigue cycles are going to be in the technical specification will be what percentage of

an operating cycle does it occur? Does it occur one day out of a year, or does it occur 50 percent of the time?

MR. WELTER: And we'll get a little bit more in the closed session, but just if we show that it -- we can sustain an infinite number of cycles of a wide range of conditions, we wouldn't necessarily need to track those fatigue cycles. But if we said to meet some of these limits we needed to identify some of these cycles, then that would be potentially a different approach.

MEMBER RICCARDELLA: I'm a little --

MR. WELTER: Lead us down a different path.

MEMBER RICCARDELLA: -- skeptical of endurance when it's --

CHAIR KIRCHNER: Okay. Let me take the prerogative of chair and let's stop here, because we now several times have gone into material that I think we had expected to cover under closed session.

So we need to change out. Thank you, NuScale and to the staff for the open session. And then I think at that point -- and we'll take a break after we hear from the staff and then we'll go into closed session.

We're back in session and we're still on the record and I'm going to turn to Bruce Bavol of the staff.

Bruce?

MR. BAVOL: Thank you very much. My name is Bruce Bavol. I'm NRC staff with the Office of NRR. I'm also facilitating for -- this will be for Chapter 5 review from the staff perspective and also Chapter 3. I'm filling in for my colleague Mary Liz Johnson.

This is the open session for steam generator. The slides that I provided to the members have what we call backup slides, and that will be several topics that we'll cover: flow-induced vibration, NRELAP steam generator modeling, and secondary wave oscillation. So hopefully we can answer your questions in the closed session also.

With that, to my right Yuken Wong and Greg Makar will be going over the Chapter 5 and Chapter 3 parts of the review. And there were a couple of questions from the NuScale side that I'll sure they'll be able to answer for you.

So with that, I'll turn it over to Greg.

And, Greg, just let me know when you need to change slides.

MR. MAKAR: Thank you. My name is Greg

Makar. I'm a materials engineer with the Division of New and Renewed Licenses in the Corrosion and Steam Generator Branch. And as Bruce said, we have prepared material today that touches on different chapters and topics. Some of it is for open discussion; some if it will be closed discussion.

I'm going to begin with the type of material we review in Chapter 5 related to materials of the steam generator and the inspection program. Yuken's going to present some Chapter 3 items on the ASME Boiler and Pressure Vessel Code, stress calculations and flow-induced vibration. And then it will return to me and I'm going to touch on some -- the other topics of the stability topical report, secondary side density wave oscillation and accident analysis.

And we have other staff members here that can provide answers and present material in more detail than I can.

On this slide I just wanted to show -- I don't want to spend a lot of time on this, but I wanted to give a review of the kind of information that we review in Chapter 5. This is related to the selection of materials and the requirements for processing, code classification, water chemistry, and

then access for -- to inspect the steam generators on both sides. And our focus here on these topics is tube integrity; that is leakage and structural integrity of the tubes.

And so this -- and the inspection here is for progressive forms of degradation, rapid forms of degradation. The intent is to design -- is to address them in the design phase. And so something like flow-induced vibration is addressed in Chapter 3.

Next slide, please, Bruce.

There are many similarities between these steam generator --

MEMBER MARCH-LEUBA: Sorry. You got me sleeping. The FSAR Section 5, I believe -- I read so many documents; correct if I'm wrong, it basically says you do not cover the secondary side instability in Chapter 5. It's covered in 15.9, correct? The AREVA --

(Simultaneous speaking.)

MR. MAKAR: Yes.

MEMBER MARCH-LEUBA: Okay. So you're not going to talk about density wave?

MR. MAKAR: I'll talk to that. I have some slides on that later.

MEMBER MARCH-LEUBA: But your SER does

not?

MR. MAKAR: The SER, Chapter 5 does not.

MEMBER MARCH-LEUBA: And then you refer to 15.9, which doesn't either. Just for the record.

MR. MAKAR: There are a lot of similarities between these steam generators and the ones that are at operating plants. I want to point out some of the differences that we're familiar with mainly so that I can then point out how they affected our review.

One is the primary pressure outside the tube, so the tendency is more for a collapse than a burst of the tube. There's the helical coil shape. The first of a kind flow restrictors, support structures and steam and feed plenums. And first of a kind means developing the appropriate design and inspection requirements.

Next slide, please, Bruce.

And these are how these differences affect our design B. All plants are required to perform a degradation assessment to assess what types of degradation they expect to find in the steam generator. This would be different than the operating fleet. A big difference is that they are looking at tube collapse rather than tube burst as the primary

failure mode.

There are also maybe secondary -- effects of the secondary coolant on steam generators that we haven't seen in others because we now have the less-pure water inside the tubes rather than on the outside. There may be some beneficial effects of having it arranged that way, but there may be things like deposits that form in the tubes that affect inspections in a way that they don't in operating reactors now.

There are new flow effects and loose parts potential inside and outside the tubes because we have different types of support structures. We have flow restrictors inside the tubes. And the inspection methods require development because of the different tube shape and potentially internal deposits which can affect the -- both the ability to inspect and potentially the results as well.

And these inspection analyses require qualification as well as developing the procedures to do it. They're going to have to be qualified to show that they are -- that they do what they intend. And so there may be things like cleaning the insides of the tubes that may -- could be required that we don't -- that plants don't worry about now.

There's also the question of inspection interval. All tubes will be -- 100 percent of the tubes will be inspected after the first operating cycle. After then we have performance-based tech specs which NuScale is adopting which -- where the subsequent inspection would be determined by the results of the first operating --

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: Out of curiosity the 100 percent inspection is for the first module or for each of the --

(Simultaneous speaking.)

MR. MAKAR: Every steam generator tube.

MEMBER MARCH-LEUBA: Every single one?

MR. MAKAR: Yes.

MEMBER REMPE: So I have a question about this, too. It seems from the discussions even earlier today that there's some information that's going to come as we gain operating experience, and clearly NuScale has to do some investigations. What about the staff? Is research needed by the staff or is your approach going to be just stand off and look and see what the folks from NuScale, or whoever the COL applicant is, learns? Do you need to have some sort of user need to research to have them get engaged to

help you define what parameters should be the ones that you're going to focus on, or what's your plan of attack to close this out?

MR. MAKAR: We have included an item on inspection of these steam generators in our discussion, and I have to be honest, I don't remember what we included, what's included in a user need for --

(Simultaneous speaking.)

MEMBER REMPE: Well, when we talked to the folks from research about it, which was several months ago, they said, no, we've not seen anything coming from the folks that are reviewing the NuScale reactor. And so you think you've got enough? You don't need to do anything else? Do you think maybe this might be something where it would be good to have some sort of research effort engaged in it?

MR. MAKAR: I think it would be supplemental in the sense that when they do inspections and have their condition monitoring and operational assessments to look at what's going forward, it's going to be based on the information available from their steam generators and operational experience from other steam generators as best that can be --

(Simultaneous speaking.)

MEMBER REMPE: So I'm going to paraphrase your response is that the Agency can basically dictate what the COL applicant needs to provide and then we'll decide. You're basically saying, no, I don't think the Agency should be spending research dollars on it. Is that what I'm hearing from you? I'm just trying to -- understand what your response is.

MR. MAKAR: I think that we can proceed without that.

MEMBER REMPE: Okay.

MR. MAKAR: And again, I think we also have had discussions about what we might need for this, what we might want to supplement our reviews on this.

MEMBER REMPE: But you think that the --

MR. MAKAR: So I don't think -- I think we haven't determined that it's necessary.

MEMBER REMPE: Okay. You're going to make the COL applicant pay for it? You're not thinking that the --

MR. MAKAR: We're not --

MEMBER REMPE: -- research dollars are needed to do it?

MEMBER MARCH-LEUBA: But following on

that, I mean right now you, Greg; and I'm going to use you as an example of all of the staff, you are the number one expert on what areas of this design were on more certainty, where we would benefit from additional research. It would be worthwhile if you guys have a half a day off-site meeting and bring the map and pass them to management and say, okay, if you are going to spend research dollars, I would like you to spend research dollars on fretting on the steam generator. That would be worthwhile. That would have value for me. Yes, and if you get together, you're going to go out with 100 topics, okay, because everybody has a favorite one.

But right now you guys are the number one expert in the world on what assumptions you have to make where data would have been valuable. And before you forget, it would be really worthwhile to get together and then let management decide whether they have the money.

MEMBER REMPE: And as I recall what Dennis was saying earlier, there were a lot of lessons learned over the years with the traditional steam generators. We didn't have all the answers. And it might be good to get ahead of it.

MR. MAKAR: Okay.

MEMBER REMPE: But that's just one member's opinion.

MR. MAKAR: Thank you. Whatever we have decided or have -- maybe haven't decided yet about user need research, we also have National Lab people available who support us in steam generator work as well, data analysis and testing.

MEMBER MARCH-LEUBA: And don't concentrate only on the steam generator. Think about pumps, amount of circulation, boiling. I mean what would you have liked to have more data to complete this review, and just have a list. That's a recommendation from one particular member.

MEMBER BLEY: I want to take you back to the tech specs you mentioned. They require 100 percent testing the first time around and then it's based on the results. Those requirements came out of following the U-tube steam generators for 50, 60 years as it evolved. Given that these are completely new, are the requirements applicable or does something more aggressive need to be considered? Have you thought about that?

MR. MAKAR: Some of requirements that are not -- that are associated with the tech specs, the guidelines are still applicable and fairly clear about

how to say take your inspection results and assess going forward when you need to inspect next. The difficulty for a new design like this is they -- that requires things like what is -- if there is some wear of the tubes, what wear rate is going to be assumed for the next cycle? And in most cases for operating plants they can use values that they -- we agree are conservative --

MEMBER BLEY: You've got pretty good expectations for that.

MR. MAKAR: -- and they can still get through the next cycle or more.

In this case I think it's going to require conservative assumptions until they have more data. So that's --

(Simultaneous speaking.)

MEMBER BLEY: Is that something that can happen within the current tech spec, or do you need to look for a change?

MR. MAKAR: Right. The current tech specs and framework that they're using will require them to do that kind of assessment. We inspect at the plan the work they're doing during the outage. And they also submit the reports to us as part of the tech specs. And then we ask questions about the findings

and the decisions they make. So that's something that would have to be -- whatever decisions they make about the next cycle is something we can question.

MEMBER BLEY: Okay. Thanks. I opened this and that leads me to another place where I didn't really have a disagreement with Jose earlier, but I had said we had pushed for things to be moved forward in other design certs and cited the IEEE requirements that have reports that get done as ITAAC, but those have methodologies that the staff has already approved. From what we were hearing this methodology is evolving. And I don't know, the argument that there ought to be a topical report kind of makes sense to me.

Have you guys thought about that aspect? How do you look at this methodology? You just go down and do a -- the audits may or may not dig deep enough to find problems. I mean it's an interesting issue.

MEMBER MARCH-LEUBA: And let's look at human nature. If we wait to review the ITAAC after they've done all the work, the only option you have is to review it, to approve it. You're not going to reject it. So the time to agree on the methodology is before they spend all the money to do the calculation.

MR. MAKAR: We've had that discussion.

Some questions have been asked by other members of the staff. I think this ITAAC gives us the ability to dig down and do whatever review we need to do on this, but it's a -- but how long is that going to take, at what point in the process? And it's not defined.

MEMBER MARCH-LEUBA: And it's reasonable that the data is going to go to a region somewhere. And then we have a poor lonely mechanical engineers saying does it satisfy us, anyone? Yes, yes, yes, yes. Done. Well, it won't come back to you.

MR. MAKAR: I think we are able --

MEMBER MARCH-LEUBA: You are able to, but --

MR. MAKAR: -- to do that if we --

MEMBER MARCH-LEUBA: -- the default, it goes to a lonely mechanical engineer in a region.

MR. MAKAR: But it's an area of uncertainty and therefore schedule risk.

MEMBER BALLINGER: Again, what you've I think adequately described is this competition that's been going on the steam generator, and that -- the earlier steam generator designs we learned by painful experience. And so we've evolved to using Alloy 690. That took is a while. But that was presuming certain degradation mechanisms that we were finding a solution

for, which we did. And that's the other -- on the other side you've got a steam generator that's very new in this design. It's using 690, which is a great material, but you got to be careful that the inspection interval initially -- the initial inspection interval is cognizant of the fact that there may be something going on that we didn't account for or find in the U-tube type or recirculating steam generators.

MEMBER BLEY: We solved them eventually, but --

MEMBER BALLINGER: Yes, but --

MEMBER BLEY: -- we had plants that were never -- I mean we didn't even have a way to get the steam generators tubes out. We had to cut holes in the --

MEMBER BALLINGER: Yes, well, I think that --

MEMBER BLEY: -- containment to get things out and replace them. It was a big deal.

MEMBER BALLINGER: I don't think it's 100 percent, but it's darn close to 100 percent of initial steam generators did not make it past 20 years. They had to be replaced.

MR. MAKAR: We agree completely that the

degradation assessment for a plant like this is different based on the material and design factors, and that was the nature of a lot of our discussions with the applicant.

And so just the status of our review in this area is that we -- there are some confirmatory items to clarify language. And as you also heard from the applicant, there is -- they are looking at the language in Chapter 5 about some of the statements that were made, and they made some changes. I think we will have some corresponding changes to make in our SER.

MEMBER MARCH-LEUBA: Are you planning to add some confirmatory items related to the methodology for DWO?

MR. MAKAR: Currently we do not.

MEMBER MARCH-LEUBA: You don't plan to?

MR. MAKAR: We don't plan to.

MEMBER MARCH-LEUBA: Then do you know of any steam generator in the world that operates in -- stable condition?

MR. MAKAR: No.

MEMBER MARCH-LEUBA: That's an unusual circumstance. I think it certainly deserves a little more inspection.

MR. MAKAR: Well, to address the stability issue that -- I don't think a -- we wouldn't have a confirmatory item now because there's nothing to confirm. It's more of the methodology and the --

MEMBER MARCH-LEUBA: But we're going back to the topical --

MR. MAKAR: Yes.

MEMBER MARCH-LEUBA: -- the need for a topical report. Call it technical report. Call it an off-site meeting. But there is a need to agree on what methodology we'll use before you send me that final number.

MR. MAKAR: So we agree that this is not resolved. It's -- rather than a confirmatory item, it's another regulatory process.

MEMBER MARCH-LEUBA: Talk to the lawyers and figure out how to do it, but it needs to be done.

MR. MAKAR: Next slide, please, Bruce.

Well, we've talked a lot about the inspection already, so I'll just reiterate that this is -- that they are adopting a mature -- framework would have been a better word here than program. We think of program as the individual plant's inspection program. But this combination of the NEI 97-06 program with the research guidelines and technical

specifications has requirements on -- which define structural and leakage integrity and the required -- it requires that they do inspection and operational assessment and condition monitoring.

And as we said, it will include the full length of the tube and for -- during each inspection the full length of the test -- we have to inspect the full length of the tube with a technique that can find what they're looking for, and the objective is to find any kind of degradation from one tube end to the other.

The tech specs also have limits on primary and secondary leakage, and it serves as the ASME Code inspection on the tubes and it limits the allowable through-wall degradation. As we said, the first outage is going to be 100 percent of all of the tubes and that these inspection reports are submitted to the staff.

Next, please, Bruce.

Now I just wanted to go into a little more about how the inspections are done. So the plant is to do what -- operating plants do it in spectrum. The tube -- look for tube degradation by putting eddy current probes inside the tubes that look for volumetric degradation that could be on the inside or

outside of the tube and internal or external cracking. Most of this applies to applies to NuScale with some modifications. Like operating plants, they are using eddy current testing.

In general plants have found that they can manage wear from support structures and foreign objects over time. It depends on the nature of the cause and the type of characterization they're able to do with the eddy current and also the projected growth rate.

Cracks generally are plugged when they're detected. There are some exceptions that some plants have through license amendments. And then there's a -- on the shell side there is a visual inspection to look for things like loose parts, foreign objects, deposits and things of that nature. And the NuScale program is described in the FSAR. There's a COL item for the -- to actually submit the program.

MEMBER MARCH-LEUBA: Are there any requirements with respect to loose parts? Because keeping the primary system clean is very -- not very easy, but it's doable. Now keeping the secondary side where it's all balance of plant -- or all those things, keeping that clean us much harder. And that's why BWRs have problems and PWRs don't typically have

that many problems.

Any flake of oxide that comes off from the -- pack will stuck into the IFRs and will produce additional loads, flow restrictions. Any fiber; that clearance is one of my hairs, and I have very thing hair, you're the same thing, will block it. Okay? So has there been any effort to minimize loose parts on the secondary?

MR. MAKAR: I don't know what all they do to minimize loose parts on the secondary. I think in some -- I can't say. What I can say is that from an inspection standpoint there -- you would -- if you have something, you would assess whether it's --

MEMBER MARCH-LEUBA: My experience is --

MR. MAKAR: -- what the consequences are, whether it's okay to leave it there, if you remove it, where it came from and what you need to do to prevent it from happening again.

MEMBER MARCH-LEUBA: We're talking about a piece of hair. I mean it's minimal. You can hardly see it. My experience with BWRs that most of the fuel leakage, the failure, the clad failure is because a small loose part that sticks into the spacer. And here we have a filter, which is the inlet on the steam generator that will catch every loose part that's

small enough to fit inside, which are the EC1 steam generator because they're the small ones.

And whenever you have one of those in BWR, you fail the claim, because the conditions are different and much more severe than here. But the potential is there. I mean you have the mechanism to trap every single flake of oxide that moves to the secondary, and this is the balance of plant model.

MR. MAKAR: That may not make it to inspection, so that would affect operation and --

MEMBER MARCH-LEUBA: It would if it breaks a tube.

MEMBER DIMITRIJEVIC: Well, I have a fundamental question related to Chapter 19. What is the failure mode we are most concerned here? Because we are not talking to burst, but to collapse. So is that going to result in the leakage or rupture? What is our -- because I notice they don't say steam generator tube rupture in PRA. They say steam generator tube failure.

MR. MAKAR: That's right.

MEMBER DIMITRIJEVIC: What's a failure mode?

MR. MAKAR: Failure mode is -- in the primary mode is collapse and we don't --

MEMBER DIMITRIJEVIC: So what, like a plugged tube, a vacuum tube or rupture tube or --

MR. MAKAR: We wouldn't necessarily -- there would not necessarily be primary/secondary leakage in --

MEMBER DIMITRIJEVIC: Right.

MR. MAKAR: -- this mode of failure, but we don't know. So when we look at the tube plugging criterion, for example, how thin can the tube can get before it would collapse? We are approaching that with preventing collapse.

MEMBER DIMITRIJEVIC: So from the safety point of view is our main concern losing effectiveness or is our main concern leakage within primary/secondary? What is the main concern?

MR. MAKAR: For safety it would still be leakage because we don't know if there would be a rupture or a break in the tube during a collapse.

MEMBER DIMITRIJEVIC: So leakage is still our main safety concern?

MR. MAKAR: Right.

MEMBER DIMITRIJEVIC: All right.

MEMBER BALLINGER: With respect to the visual inspection with a U-tube steam generator you have limited inspection visual capability, but you do

have a tube lane. You go inside, you can do the outside, but in this steam generator how much of the system can you actually see with a visual inspection? It sounds to me that it's pretty limited.

MR. MAKAR: Very. I can't --

(Simultaneous speaking.)

MEMBER BALLINGER: Very limited, yes.

MR. MAKAR: But you would have access to the tightest spins and the -- be able to see the --

MEMBER BALLINGER: The tightest spins?

MR. MAKAR: Yes, which are the -- yes, because they're at the top and the bottom. And things that are going on at the tube sheet. And some ability to see the support structures. And tabs, looking for the condition of tabs and spacers.

MEMBER BALLINGER: Which is the outside?

MR. MAKAR: Right. And I should mention also that the steam generator supports, which are the horizontal bars that are welded to the reactor vessel, since they're -- that's a Class 1 weld, so that has an ASME Code requirement for those welds, I assume.

But that's all I had for Chapter 5, and so I'll -- Yuken will now present Chapter 3.

MR. WONG: Thanks. My name is Yuken Wong from the Mechanical Engineering and -- Testing Branch.

I've reviewed the reactor internals vibration program with assistance from Dr. Steve Hanbric. He's on the phone. I will present the flow-induced vibration review and the ASME Section 3 stress analysis area.

Dr. Steve Hambric and I have previously presented the review of the steam generator vibration evaluation and the TF-3 testing audit. The review involved the flow-induced vibration, fatigue evaluation and wear for the flow-induced vibration mechanism involving vortex shedding, fluidelastic instability and turbulence buffering.

NuScale has updated the analysis based on the mode of testing from the TF-3 specimen. The margins have improved. The staff reviewed the analysis and the margins and they are acceptable.

In the interim letter the ACRS recommended some Tier 1 commitment such as ITAAC for the completion of the TF-3 test, and NuScale has included the TF-3 statements in Tier 1 of the FSAR and also Tier 2 in Section 14.2 regarding initial start testing. The test acceptance criteria are also included. The staff finds the description and acceptance criterias are acceptable.

Next slide. The inlet flow restrictor is susceptible to liquid flow instability and the design

is evaluated by testing. The three concept designs are tested and one of the design is selected, and it has adequate margins against flow-induced vibration as well as adequate pressure drop.

MEMBER MARCH-LEUBA: While you have this here, the argument that was making earlier this morning -- first, I'd like the arguments, I love this. First, look at the device. There is a Christmas tree. You have couple hundred, three hundred IFRs protruding from this plate and each of them is occluding from the back of the plate. So it's cantilever on the plate. And somebody, a robot or a person, will grab it and will try to align each of those things into the steam tubes, all of them at once.

I know I've tried to do that. I mean I was replacing some door handles and I couldn't put the screw in the right hole this weekend. Putting 300 of them in the right hole is going to be very difficult. And you are going to end up scratching the tube or the IFR therefore increasing the clearance, the gap. Over 80 years, 60 years of operation that gap is going to be twice as big after you've taken it and put it back 100 times. That's number one.

Number two, whenever you have reverse flow with high void fracture, it's not just reverse flow.

It's reverse flow at 10, 20, 40 times a velocity that is normal and with a dynamic load consistent with cavitation. You have voids and liquid and an uneven force. So it's -- basically the thing is cavitated. Plus, when it comes out, it's impinging on the plate at this high velocity.

I would not want to have to analyze the mechanical integrity of that plate and those devices myself, number one. I couldn't do it for a single cylinder myself. But I don't think Pete can analyze the mechanical integrity of that thing under those conditions. Hopefully we have a lot of margin, but to me it seems that this reverse flow with high void fraction at a tremendously high speed; and I'm surprised it not choke flow, does not produce vibrations -- Okay. Yes, and now that you have visualize it, you see the --

MEMBER BLEY: If you really have two-phase conditions coming in, we got to get steam cutting, too, I would expect on these things. Yes.

MEMBER MARCH-LEUBA: Yes, I -- if I was going to do this in trace, I would put a K factor and a frequent factor, an effective frequent factor forward and an effective frequent factor backward and do the analysis like that. That's not what's

happening. What's happening is in each of those you have an acceleration, a deceleration. Voids are going randomly producing uneven dynamic loads. And true the rod is pretty stiff, but it's held by a single screw at the bend at the end and it's been there for 60 years. I'm not convinced that that thing doesn't vibrate and produce fretting.

And basically my concern from the point of view of stability, which is my concern, is that the IFR is going to lose its effectiveness with time and the gap will be larger with time. I mean you're going to relax. One way or another it's going to relax and instead being a friction factor of 100, it will be 90 or 80 or 60 or 40, and therefore it will become more and more unstable at full power. Just wanted to put this in the open record.

MR. WONG: Okay. We will discuss more in the -- regarding reverse flow in the void fraction later on in the closed section.

And regarding the scratching on the side of the wall due to the installation for the hundreds of IFRs, in the test for the concept design they did have misalignment tests that have the FORs in IFRs touching the side of the tube. And they performed the bore scope inspection and find the wear is acceptable.

MEMBER MARCH-LEUBA: What inspection they perform?

MR. WONG: They inspected the inside of the tube after the test and they found the wear is acceptable.

MEMBER MARCH-LEUBA: And this test had reverse two-phase flow on it?

MR. WONG: I do not --

MR. HAMBRIC: This is Steve Hambric. No, it did not.

MEMBER MARCH-LEUBA: I didn't want to offer the answer for you guys because you always tell me I'm wrong, but I was 99.9 percent sure that they didn't. Okay.

MR. WONG: Okay. From the -- the concept is -- concept testing, NuScale selected the final design for the IFR and the final design is similar to one of the tests that the -- tested design, and this testing, this final design will be validated by a test and this test will be performed after design certification. The test plan and acceptance criteria are described in the NuScale technical report and the staff finds the description and acceptance criteria reasonable.

MEMBER BLEY: What does it mean that

they'll be similar to the ones that were tested? How will they be different?

MR. WONG: They are all -- they are step design with five different steps, as you see on the sketch. And the step length is within 10 percent. The spacing between the steps is also within 10 percent. And also there's a tip at the end of the inlet flow restrictor. That length is also within 10 percent.

MEMBER BLEY: But they changed it as a result of the test? I'm curious as to why it's not exactly like the ones they tested.

MR. WONG: I don't recall the specific reasons. I cannot answer the question.

MEMBER BLEY: So it's almost the same? And just before we try to start this up we'll get some confirmation that it's okay?

MR. HAMBRIC: Yes, that's correct.

MR. WONG: Yes, the test will be performed --

MEMBER BLEY: Thank you, Steve.

MR. WONG: -- prior to initial start testing. And that is a part of the -- completing the comprehensive vibration assessment program.

MEMBER RICCARDELLA: And that is covered

by an ITAAC?

MR. WONG: It's covered by the Chapter 14.2 initial start testing activities.

MEMBER RICCARDELLA: Okay.

MR. WONG: Even though it's performed in a lab, but it's covered in the FSAR.

MEMBER BALLINGER: A little bit of a nitpick on this slide. Alloy 690 is not a nickel-chrome-iridium alloy.

MR. WONG: Thank you.

MEMBER BALLINGER: That would be kind of cool, but --

MR. HAMBRIC: That's my fault. I should have -- Apologies.

MR. WONG: Okay. Density wave oscillation was observed in the TF-2 testing. The test data showed a temperature and secondary flow oscillation. train gauges was installed on the tubes and the test data indicate long oscillation periods. So the oscillation frequencies and the steam generator tube resonance frequencies differ by an order of two magnitude, so the density wave oscillations would not excite the steam generator tube resonance frequencies.

MEMBER MARCH-LEUBA: Yes, but -- I'm not a mechanical engineer, but in my mind clearly the

limiting case is where you weld the tube to the plate, not the middle of the tube. So it's the connection of the tube to the inlet plate is where you want the strain. That would be the limiting case, wouldn't you agree? So in the middle of the tube -- those tubes are strong. That's not going to be a problem. It's going to be the welding.

MEMBER RICCARDELLA: It's going to be a fatigue problem. It will be a discontinuity, not in the middle of a tube like that.

MEMBER MARCH-LEUBA: And these measurements will depend a lot of how prototypically they clamp the tubes with those three --- elements. So if you let them flex, if you have it relaxed, it will stand and there won't be any strain. If you have one that is more clamped, it will -- so, yes. It's good that we have an example that it wasn't bad, but I don't think it's limiting.

MR. WONG: Yes, the steam generator tube as well as the welds will be evaluated according to the ASME Section 3 requirement according to the ITAAC.

There are strain gauges mounted on the outside of the tubes and using the strain gauge measurements they are in the order of hundreds of microbe strains. So the staff evaluated alternating

stress intensities and found the alternating stress intensities below the endurance limit.

The maximum surface stress are below the tube yield and ultimate strength. And we know that the TF-2 test is not identical to the MP operating conditions, so NuScale will perform the ASME stress evaluations as required by the ITAAC to confirm this preliminary stress analysis.

MEMBER PETTI: So was the calculation done -- is it done for the free tube or was it done where there's constraint? I can see that there's not a problem where the tube was allowed to move, but was the calculation done either at the -- where it's welded or where it's clamped, where there's --

MR. WONG: We have not seen --

MR. HAMBRIC: This is Steve Hambric. Yes, we're just taking the measured surface strains as implemented. It's not really an analysis. We take the strains and multiply it by e. That's pretty much it. So it's a measurement of the strain. And I agree that the welds are probably more important, but we don't have access to that information, and as Yuken said, that should be part of the upcoming ASME calculations under the ITAAC. They should do the assessments of the stress at any welds.

MEMBER PETTI: Okay. Thank you.

MR. WONG: Okay. Next slide, please. The oscillating thermal loads can cause the tubing to expand and contract. Because of the long oscillating periods, this should not cause vibration, but it cause the tube to slide along the tube supports. And you can see the sketch on the lower right. The thickness of the tube support is much bigger than the thickness of the tube. So the tube supports should experience even lower stresses. And as Greg had indicated earlier, the long-term wear of the tubes will be monitored by the steam generator inspection programs.

Next slide, please. Okay. Regarding the ASME Section 3 stress analysis, the steam generator tubes are ASME Class 1. ASME Section 3 Code requires the analysis to address fatigue and the relevant design loads. And there's an ITAAC to require NuScale to perform the ASME Section 3 analysis. And for the Class 1 component is listed in DCD Tier 1 table 2.1-2. And the steam generator is listed as one of the components in that table.

MEMBER BALLINGER: We keep going round and round about Section 3 and Class 1 and all that stuff. Has anybody on the staff taken a look at FFS-1? Which is fitness for service, and it's an ASME Code. And

it's a recent; not so recent now I guess, adaptation of API-579 for the petrochemical industry. And it explicitly deals with environmental effects, corrosion issues, which the Section 3 doesn't. Just a thought. It's only 1,100 or 1,200 pages. You can read it in an afternoon, right?

MR. WONG: There's no reference to that.

MEMBER BALLINGER: No, I'm just -- no, there will -- eventually it will fold its way in. The Code operates on a geotechnical time scale.

PARTICIPANT: Geological.

MEMBER BALLINGER: Geological time scale. Excuse me. Maybe both.

MR. WONG: Thanks.

Okay. During initial start testing one tube in the steam generator will be instrumented with strain gauges. If density wave oscillation occurs, the strain gauges will be able to measure the change in strain in the tube. However, the strain gauge measurement is not intended to detect density wave oscillation because strain gauges in multiple tubes is needed -- are needed to detect the relative phase between the same two tubes.

So as I mention earlier, the steam generator tubes are qualified by the ASME stress

analysis, the ITAAC requirement.

MEMBER MARCH-LEUBA: Since you're talking about instrumentation, the issue was brought earlier that we don't want to require any instrumentation to take DWO on this reactor. And it sounds unwise to me. Clearly operation in -- under unstable conditions is undesirable. Everybody will agree that this is undesirable. It may be tolerable from a strain position, but certainly undesirable.

And we expect -- and I'm siding with NuScale's evaluation that full power it's likely going to be stable.

Sorry, Pete.

I would still like to see the analysis. But I am also postulating that the IFR effectiveness is going to age. You're going to have binding, you're going to have movement, you're going to have misalignments, and 10 percent of the tubes will have an IFR that is not as effective as it should be -- full power operation eventually. Okay? And there's this likelihood that it will do. And we don't need strain gauges on every tube -- I know that the theory will tell you --

CHAIR KIRCHNER: Just pardon the interruption. Someone on the line, would you mute

your phone? We're hearing your paperwork scratching against something. We're picking up static.

MEMBER MARCH-LEUBA: It's static. Okay. We'll go over it. I'll talk louder.

So I know the theory tells you that the inlet and outlet pressure are constant during these out of phase oscillations, but they're not. So if the oscillations developed, the inlet and outlet pressure of the plenum will have one to 10 hertz -- 0.1 to 1 hertz oscillations depending on the flow and will be detectable with simple instrumentation.

And if we are going to go into allowing unstable operation, which is an undesirable condition, and we do that because we believe that during normal operation it won't happen, but with respect that degradation of the IFR will get you there in 10 years, we should be monitoring it. Because if we start detecting significant noise on the inlet or the outlet pressures, we can do something about it. We can fix the IFRs. I mean we know that in this outage we need to align them or tune them, or whatever you need to do to make them work. So the fact that we don't have instability instrumentation, I consider that negligent.

And I realize that it's not your area of

expertise, but I don't know whose area of expertise it is? Who will require that? This agency is moving away from defense-in-depth and just risk-informed regulation. This is where defense-in-depth would be very helpful. I mean you say you're not unstable. Prove it. And we suspect -- I mean everything relaxes. Those IFRs are going to relax. They're going to degrade. I guarantee it. Will they degrade one percent or 100 percent? I don't know.

MR. WONG: Okay. Next one. This is a summary slide. I talk about the ITAAC for the ASME, the commitment to perform the TF-3 test commitment in the DCA. The density wave oscillation does not induce significant steam generator tube vibration. The ITAAC to perform the ASME Section 3 analysis and also the steam generator tube inlet flow restrictor testing will confirm the IFR final design.

And that's the end of my presentation. I'm going to turn it back over to Greg.

CHAIR KIRCHNER: Before we turn it over to someone else, Yuken, let me ask you a question on the second bullet. You say density wave oscillation does not induce significant steam generator vibration. What is that based on? Just a single tube or -- what we have, as Jose has outlined some scenarios,

significant instabilities in multiple tubes? Is that going to lead to a vibration issue for the steam generator and hence --

MR. WONG: No significant vibration is because the differences in frequency, the DW frequency versus the steam generator --

(Simultaneous speaking.)

CHAIR KIRCHNER: No, I understand that. Yes, that's kind of a static analysis. I misspoke. Not static, but you're just looking at the fundamental structural resonant frequency?

MR. WONG: Right. And we don't believe it cause vibration. It will cause the -- there's a sliding of the tubes along the supports. And the -- we have not seen the wear analysis from NuScale and this --

CHAIR KIRCHNER: Okay.

MR. MAKAR: -- the long-term wear due to this -- the slow oscillation will be monitored by the steam generator inspection program.

CHAIR KIRCHNER: By the inspection program? Okay. That goes back to Ron and my earlier --

MR. WONG: Yes, the --

CHAIR KIRCHNER: -- comments.

MR. WONG: -- vibration review primary focus on the rapid degradation mechanisms such as the vortex shedding and fluidelastic instability. Some of the slow degradation mechanisms can be managed by the steam generator inspection program.

CHAIR KIRCHNER: Well, I wouldn't see problems on the primary side; the flows are so low anyway, but certainly I would see problems with the tubes themselves. And the -- you were contrasting the -- for the wear issue the fact that the clips that hold the steam generator tubes in place, those are much thicker, stiffer I would guess is kind of -- then the actual tube itself. They're much -- so they're bent? Those clips are bent out of that --

MR. WONG: The support plate.

CHAIR KIRCHNER: -- support piece? So that's not a loose parts issue then?

MR. WONG: We're not concerned with loose part issues just because it's -- the clips are punch out from the --

CHAIR KIRCHNER: It's a punch-out?

MR. WONG: -- support columns. They are one continuous piece.

CHAIR KIRCHNER: Yes.

MR. WONG: And as the figure indicates the

supports are much stronger than the tubes.

CHAIR KIRCHNER: Right. Okay. What about with the inlet flow restrictors? Have you looked at failure of those so that -- you potentially don't loose parts going up this steam generator tubes and then exiting into the valves.

MR. WONG: We look at the vibration for these inlet flow restrictor, the vibration aperture of the flow restrictors versus the clearance between the flow restrictor and the tubes and we have not specifically addressed the likelihood of those parts in --

CHAIR KIRCHNER: Right, my concern would be you've got this long rod protruding into the tube sheets. It's vibrating. And then if you fracture off a small piece of that, it possibly could go right up the steam generator and out. And there are valves, important valves on the exit side of the steam generator.

MR. MAKAR: I think there are a couple of ways to deal with that, and they don't involve what to do at the valve, but that would have to be addressed if it were damaged or the parts were in there, I think. But I think the -- if for example, during an inspection you find that a piece is missing from the

flow restrictor, then aside from the damage to the flow restrictor, then you're going to be looking for damage to the tubing --

CHAIR KIRCHNER: Right.

MR. MAKAR: -- for your inspection process and/or that there's a part captured in the --

CHAIR KIRCHNER: That's my concern.

MR. MAKAR: -- tube.

CHAIR KIRCHNER: Downstream you have -- of the tubes you have some important valves --

(Simultaneous speaking.)

MR. MAKAR: Right, I think if there was something missing and you didn't find it, then you would have to go and inspect those things downstream.

MEMBER MARCH-LEUBA: In their defense, the first half of the tube is a steam flow --

CHAIR KIRCHNER: Yes.

MEMBER MARCH-LEUBA: -- and it's going uphill. It would be hard to carry anything that's not a snowflake. I mean it's steam and it's a gas.

CHAIR KIRCHNER: Yes.

MEMBER MARCH-LEUBA: It's not a significant flow of water going uphill.

CHAIR KIRCHNER: Yes, but it's at a good velocity.

MEMBER MARCH-LEUBA: Yes.

CHAIR KIRCHNER: So, but you have done an analysis of the actual potential fatigue of those inserts that are part of the flow restrictor?

MR. WONG: The inlet flow restrictors are tested. There are no analysis performed for flow-induced vibration.

CHAIR KIRCHNER: Well, I'm just concerned that the fatigue life of those individual restrictors may lead to mechanical failure and then a loose part issue.

MR. WONG: We can confirm with NuScale if the ASME Section 3 stress analysis -- will also --

CHAIR KIRCHNER: So that if the code would --

MR. WONG: -- provide a fatigue analysis.

CHAIR KIRCHNER: -- require them to analyze that as well?

MR. WONG: Yes, ASME Code requires fatigue analysis.

CHAIR KIRCHNER: We talked about the plate itself vibrating, but the individual flow restrictors as well would be analyzed for fatigue?

MR. WONG: The ASME Section Code analysis will --

CHAIR KIRCHNER: Okay.

MR. WONG: -- require fatigue analysis, and we need to have NuScale to confirm the flow restrictor will also perform ASME Section 3 analysis as well.

CHAIR KIRCHNER: Okay. Thank you.

Bruce?

MR. MAKAR: I have just a couple of slides to address, some of the other topics that have come up in these discussions. One is the accident analysis. We recognize that steam generator design affects accident analyses of the -- it affects the temperature and pressure at the initiation of an event. The geometry of the steam generator would affect the flow and the other conditions, the supports and the steam generator itself. And that the -- but overall the steam generator design features don't have large effects on accident analyses and the biggest effect is on the tube break event. And the flow restrictor design and the density wave oscillation does not have a large effect on the accident analysis either.

MEMBER MARCH-LEUBA: Since you brought it up; I was going to bring it during the closed session, but one thing I'm concerned about, and it will become apparent why I'm concerned about two weeks from now or

-- no, next month, is that if you have density wave oscillations, you have large flow oscillations, meaning that you are doubling or tripling the liquid velocity and covering parts of the core that past the tubes that are not -- that are blanketed with the steam and now they're blanketed with water.

So in the analysis that NuScale performed they were concerned about reducing the effectiveness of the heat -- the steam generator, the heat transfer coefficient of the steam generator because the steam boundary would also go down actually.

I'm concerned that it will increase it significantly, because now instead of having a velocity of one, you'll be oscillating at a velocity of plus three minus three. Plus three minus three. And a change of 300 percent in liquid velocity, when you put it in liquid vol terms, which is the heat transfer coefficient, it's a significant change in heat transfer.

So my suspicion is that any accident in Chapter 15 that involves overcooling will be even more overcooled if there is instability on the secondary side. And I would want to make sure that somebody addresses that. Typically we think that if your heat exchanger is better, it's better, right? Most of the

time it is. But any event that involves overcooling I have the gut feeling; again, not an approved methodology for analysis -- a gut feeling that the heat transfer coefficient will increase during instability significantly; just putting on the record, you don't need to answer, which will contradict your it does not affect the Chapter 15.

MR. MAKAR: Understand. This density wave oscillation was also -- or secondary side flow instability was also assumed in the stability topical report and there was not -- it did not affect the primary flow, the primary side's stability. And there's the -- but it wasn't looked at directly. The stability topical report didn't evaluate the secondary side density wave oscillation directly.

And so next slide, please. So we've recognized that density wave oscillation occurs and that it may affect component integrity, that it's sensitive to the design of the flow restrictors.

As we said -- I said earlier that our inspection program for the steam generators is designed to address progressive forms of degradation, and we tried to design rapid forms of degradation out at the beginning. And so we come back to this idea that the density wave oscillation is a -- could

generate a type of cyclic load which would have to be addressed in the ASME Section 3 code calculations and that we would have access to that through the ITAAC. And that the applicant is developing that methodology to analyze DWO, and we'll have to assess that methodology for that purpose.

MEMBER RICCARDELLA: I'm assuming we're going to hear more on that last bullet in the closed session?

MR. MAKAR: If you want to hear, then we can do that, yes.

MEMBER RICCARDELLA: I think we would.

CHAIR KIRCHNER: Okay. We are a little bit over the schedule. I'd like to stop here and now turn for the opportunity for public comment.

And if there is -- while we are opening up the bridge line, if there is anyone in the room who wishes to make a comment, please come forward to the microphone, state your name and make your comment.

Seeing no one, we'll wait for the bridge line to be opened to the public. If there is anyone on the public bridge line who wishes to make a comment, please state your name and make your comment.

MS. FIELDS: This is Sarah Fields. I have one comment, and that is throughout the discussion

there's a certain assumption that NuScale would be involved at the construction/testing phase. I mean, when there is an actual construction and then initial testing that NuScale would be somehow involved in that. And yet my understanding is that another entity, a utility in this case, the Utah Associated Municipal Power Systems, intends to submit a COL application.

And I don't know if there any legal requirement that in their use of the NuScale design that means they would in fact be involved at that site. I know they're planning to, but I don't know if -- for any COL applicant referring to this design if they are required to involve NuScale during the testing and operational phases.

CHAIR KIRCHNER: Thank you, Sarah.

MS. FIELDS: Thank you.

CHAIR KIRCHNER: We will note that comment.

Is there anyone else on the public line who wishes to make a comment? Hearing none, we'll close the bridge line.

MR. SNODDERLY: I'm sorry. Before we close the bridge line, for those interested members of the public, we have a change in the agenda. We are

going to go into closed session and we will not open back up in open session until about 1:45 this afternoon. NuScale's presentation on turbine missile analysis will be a completely closed session. The public will have an opportunity to hear the staff's evaluation of turbine missiles starting at about 1:45.

If we're much later than that, I'll send an email to those members of the public that have shown an interest thus far. And if you would like me to notify you when we start, please send an email to michael.snodderly@nrc.gov and I'll let you know if it's going to be later than that. But we'll be in closed session until at least 1:45.

Please close the phone line now.

CHAIR KIRCHNER: Okay. Thank you, Mike.

So we're going to take a break here and recess for about 10 minutes. Let me see, I think we use that clock. Let's reconvene at 20 after 11:00 and we'll continue in a closed session.

(Whereupon, the above-entitled matter went off the record at 11:06 a.m. and resumed at 1:50 p.m.)

CO-CHAIR KIRCHNER: Okay, we're in open session. We're going to hear from the staff on turbine missiles.

Let me try that again. We are now in open

session, and we are going to hear from the staff on their evaluation of turbine missiles.

MR. BAVOL: Okay.

CO-CHAIR KIRCHNER: Please proceed, Bruce.

MR. BAVOL: My name's Bruce Bavol, with staff at Office of NRR. To my right is Sujit Samaddar, and I believe we have John Honcharik on the phone bridge line. John, could you?

MR. HONCHARIK: Yes, it's John Honcharik, I'm on the line.

MR. BAVOL: Very good, thank you. And with that, I'll just, I'll turn it over to Sujit.

MEMBER BLEY: Let me, before you even get started, did you get a chance to review their analysis or do any audits on it?

MR. SAMADDAR: We actually looked at the calculations in the ERR, sorry, electronic reading room.

Okay, so I'm Sujit Samaddar, as I've been introduced. Okay. So in this part of the presentation, I will address the NuScale selection of turbine missile parameters used in the barrier design assessment and the staff's conclusion. That's the first part. And then I'll move on to the barrier design itself. Next slide, please.

The missile parameters must first be determined to evaluate the adequacy of the barriers. NuScale proposes to use half the last stage rotor with blades attached as the bounded missile. This is consistent with EPRI votes and experience referred in NUREG 1275, Volume 11. The bounding missile is a 24-inch radius half-circle of the rotor weighing 3568 pounds.

The staff finds the Applicant's proposed bounding missile acceptable since it is based on a typically sized 50-megawatt turbine with 24-inch radius and 12-inch wide last stage rotor made of typical turbine rotor material. Speed of the missile was determined to be 476 miles per hour based on a 190% destructive oversuit, which is consistent with EPRI reports and NUREG 1275, Vol. 11. And this may answer the question you have raised.

Also to provide reasonable assurance, the staff confirmed that at this speed, the centrifugal stresses exceed the tensile strength of the material, causing ductile failure.

To ensure bounding analysis for a selected turbine design, a COL item, 3.5-1, has been added to the COL Applicant to confirm that the selected turbine design is bounded by the parameters used in the

NuScale FSAR analysis for size, weight, and speed of the postulated low trajectory turbine missile for the last stage of the turbine rotor.

Therefore, based on the above the staff concludes that the bounding missile is acceptable and provides a reasonable assurance when used to evaluate the barriers which protects safety-related SSCs. Next slide, please.

So this is a little bit of the regulatory basis and the user barriers. Let us look over it. GDC-4 requires SSCs important to safety to be appropriately protected against dynamic effects, including the effect of missiles that may result from equipment failure, such as the turbine.

In the NuScale design, these safety-related SSCs are in the reactor building and the control room building, which are in the low trajectory turbine missile hazard zone. Traditionally, safety-related SSCs have been protected from turbine missiles by reducing the probability of a turbine missile strike on these SSCs to 10 to the power -7 by a combination of turbine missile integrity, oversuit protection, and favorable orientation.

However, safety-related SSCs can also be protected using barriers, as provided in Regulatory

Guide 1.115 and DSRS 351.3. NuScale chose to use barriers instead as providing the required protection.

So now I'll go into talking about the evaluation of the barrier itself. Next slide, please. So this part of the presentation will address NuScale's selection of turbine missile barriers and the acceptance criteria, the methodology assessment, analysis results, and staff's conclusions. Next slide, please.

A barrier is deemed as effective only if missile impact there are no objects dislodged from the back face of the barrier, creating secondary missiles, or that the barrier is not perforated, allowing the missile to eject from the other face. Next slide, please. Sorry, I skipped a slide, let's go back one.

NuScale considered the following as barriers providing protection to the safety-related SSCs. So it's five-foot thick reinforced concrete for the reactor building, and three-foot walls and flow slabs for the control room building. So next slide, please. Okay.

The user barriers for protection against turbine missiles is a first-of-kind approach. And there is no standard review process. Therefore, the staff considered for the review of this application

the analytical approach, the reliability in simulating the impact mechanism, and the rationality of the results.

A barrier is accepted for missile protection by the staff if the barrier is not perforated, not generate secondary missiles from scabbing off back face. Imperial relationships, such as the modified NDRC formula, establishes the minimum thickness of a barrier which would meet the acceptance criteria based on how deep a missile would penetrate on impact.

NuScale's approach was to determine the depth of penetration using a finite element code that was suitable for a high energy, small contact impact on concrete walls and the use of the NDRC relations to establish the minimum thickness required for adequate protection. Next slide, please.

NuScale used a well-established, non-linear finite element-based computer program Teragram to provide a reasonable assurance that the code was appropriate for impact of high analysis, high energy, low contact area missiles. The code was used to simulate actual turbine missile tests performed by EPRI.

Comparison of the simulated and test

results showed a close prediction capabilities of the software with respect to missile penetration depth and exit velocities in the case of missile perforation. Next slide, please.

NuScale's analysis considered missile impact at the center of the modeled wall, which was smaller than the dimensions of the actual wall. The selection of the location was based on the location that would render the maximum damage to the wall. Near edge and corner strikes low transfers to the heavy support elements, reducing the damage to the wall. Next slide.

The plotted strength contours on the analysis show that the strain drops rapidly to zero with distance from the impact location. Thus, the closeness of the boundaries of the modeled wall do not impact the strength profile of the impacted site. A very localized impact effect was shown, with the missile energy being dissipated by the penetration or the perforation of the missile.

This reduction in missile energy was evidenced by the much reduced missile exist velocity from the control room building wall. For the reactor building, the depth of penetration was near the full thickness of the wall. All SSCs behind the wall was

considered as lost from secondary missile impact.

For the control room building, the wall was fully perforated, and the penetration of the slab was small, and the slab meets the acceptance criteria discussed before. NuScale therefore did not consider any global effects from the missile impact in the building member design. Last slide.

Based on the simulated results and the ability of Teragram to replicate the results from the actual turbine missile impact, the staff concluded that there is reasonable assurance that the simulated results reflect the turbine missile impact, and that barriers considered will protect the safety-related SSCs in the NuScale design, as required by GDC-4.

And that concludes my, the staff evaluations.

MEMBER BLEY: Thank you but would you show your first backup slide and tell us what that's about.

MR. SAMADDAR: Okay, so this slide was meant to demonstrate one thing and one thing only. That shows an ordinance going through three walls --

MEMBER BLEY: It's a military test.

MR. SAMADDAR: Yes. And the idea is there is no global impacts. The whole wall does not come crumbling down, it's only just localized portions are

getting blown away.

MEMBER BALLINGER: For reference purposes in GPU57, is that's what is right here, will go through 200 feet of concrete before exploding. We don't have to worry about that.

MR. SAMADDAR: This is before that.

MEMBER BLEY: What we'd seen this in earlier work is if the missile gets through easily the first barrier but can't get out of the second, it does weird stuff inside. It's like a pinball machine.

MR. SAMADDAR: So I'm open for questions if there are any.

CO-CHAIR KIRCHNER: This is a great picture. I'm familiar with this stuff. I think we can see the projectile coming out. It's unlikely you'll get a projectile like that designed for, specifically for penetration, from a turbine rotor disintegrating. Just a, so apples and oranges. This is specifically designed to go through concrete. It's a good picture.

Members, any other questions or the staff?

MEMBER RICCARDELLA: Did the staff have the complete analysis document, or did they just review what was in the DCA?

MR. SAMADDAR: No, we looked at the CAT

computations and the pictures of the plots in the electronic reading room, so we did look at it.

MEMBER RICCARDELLA: Oh, you did, okay.

MR. SAMADDAR: And the code is a well-established code because it's extensively used in AIA work and it's familiar with ACRS too because we have presented it before.

CO-CHAIR KIRCHNER: Dennis, is that?

MEMBER BLEY: Nothing more from me.

CO-CHAIR KIRCHNER: I neglected to announce that you were chairing, or co-chairing this part of the meeting.

MEMBER BLEY: I'm pleased, thank you for sharing that.

CO-CHAIR KIRCHNER: Well, that's a hint that Dennis is also going to draft a letter on this topic. Right? Yes, thank you.

Okay, if there are no further questions, then I thank the staff and the Applicant for your presentations.

MR. SAMADDAR: Thank you.

CO-CHAIR KIRCHNER: Just a minor housekeeping. We're going next to open session on the inadvertent actuation block valve. And as several more open sessions are, I remind everyone that you

have proprietary material. If you do not wish to keep it, please return it to Mike so we can take proper care of it.

Thank you, Bruce. Okay.

(Off-record comments.)

CO-CHAIR KIRCHNER: We're back in session. We are going to hear from NuScale on the emergency core cooling system valves, and in particular the inadvertent actuation block design. And I'm going to turn to Paul Infanger from NuScale to start.

MR. INFANGER: Good afternoon. Paul Infanger from NuScale Licensing, and we're going to present some information about the emergency core cooling system valves and the inadvertent actuation block designs.

So with me here is Dan Lassinger from NuScale Engineering, and he's going to start off the presentation.

MR. LASSINGER: All right, thanks, Paul. So we've had a couple of presentations about the ECCS valve design and operation previously. This presentation is really to summarize the DCA demonstration test program, which was designed and carried out in 2019.

MEMBER BLEY: That is complete.

MR. LASSINGER: It's complete, right. And specifically we'll focus on the performance of the inadvertent actuation block feature of the ECCS valve design, and we won't talk too much about the rest of the valves at this point.

Just to quickly recap the function of the ECCS valves and the role of the inadvertent actuation block feature, I'll refer to it as the IAB from here on out. The ECCS valves receive a demand to open on a actuation signal from the module protection system or on a loss of DC power, a loss of power to the ECCS trip solenoid valves, pilot-actuated valves, so those would create a demand for the ECCS main valves to open.

However, if the pressure in the reactor vessel is sufficiently high, then the IAB will interfere or engage to prevent the opening of the ECCS valves. Effectively, this prevents the opening of the ECCS valves due to spurious signals or equipment failure at normal operating pressures, but permits opening of the ECCS valves at LOCA conditions at lower pressure.

In the initial DCA submittal, the IAB pressure range was 1000-1200 psi. And what I mean by that is that over 1200 psi, the IAB would interfere to

block the main valves from opening if there was a spurious signal or equipment failure.

(Off-mic comments.)

MR. LASSINGER: What's that?

MEMBER BLEY: I said or a --

MR. LASSINGER: There are no demands of the ECCS in which the IAB would be, would interfere.

CO-CHAIR KIRCHNER: But no, no, the signal. You have two signals from your module protection system. You've got containment level and low pressure, low, low pressurizer level --

MR. LASSINGER: But a signal from module protection system would only be on a high containment level, which in that condition, the reactor pressure would be lower. It would be below the 1000 psi. So the IAB would never be interfering in our LOCA situations in which we have a, I'll say a valid demand for ECCS by the module protection system.

CO-CHAIR KIRCHNER: Do you know that for sure?

MR. LASSINGER: We know that for sure. I'd have to defer this a little to the scope of the safety analysis considered because it's dependent on many --

CO-CHAIR KIRCHNER: I'm thinking of the

containment level signal. If you got a spurious signal there or you actually had water accumulating in the containment.

MR. LASSINGER: I think the best thing for me to say at this presentation is that that's supported by the scope of safety analysis performed in Chapter 15.

CO-CHAIR KIRCHNER: Okay.

MR. LASSINGER: So over 1200 psi, the IAB was specified in the valve performance specs to block to prevent the ECCS main valves from opening. And this supports meeting CHF criteria for postulated spurious openings of the ECCS valves where there was a rapid depressurization. The lower level, 1000 psi, you know, to permit the ECCS valves to open, was based on protecting coolant levels for small breaks.

In other words, some small break that occurs for a long time where reactor level is getting lower but reactor pressure is not reducing significantly. So that was the basis for that range. Next slide.

A little more background, and we did do some proof of concept testing of the ECCS valves in 2015, we've talked about this before. But this was done with air and room temperature water. This was

primarily to support design development of the ECCS valves, specifically unique aspects like the IAB function of the concept and the fact that the actuation, the pilot actuators are remotely positioned from the main valve. Next slide.

During the DCA review, it was determined that additional demonstration of the ECCS valve functionality was required to satisfy 10 CFR 5034. So a test program was designed and executed up to operating and temperatures and pressures. So with hot water and steam at the reactor operating temperatures and pressures.

And tests were carried out in pressures to actuate ECCS below the IAB range, so below 1000 psi, and above the IAB range, which would be representing a spurious signal or a failure to see if the IAB blocked and blocked in time. Which it did for some pressures above.

However, in the execution of the test program, we did observe that the kind of two-phase effects slowed the IAB response and prevented the IAB from reliably blocking and releasing within that 200 psi window. It did block and release as expected, but the range was greater than was previously specified.

So it was initially a range of 200 psi.

We did a series of sensitivity tests at that point and made a few adjustments to the IAB valve performance, which had to do with a spring force in internal orifices of the valve assembly, and determined that a minimum range to satisfy functionality of the valve was 400 psi.

So we need a 400 psi window to reliably block and then reliably release with operating margin to, you know, that was, in the current state of the valve design, that was the limitation that we needed to work with. So --

MEMBER BLEY: How many test trials were done to come to this conclusion? You know, I'm not surprised that the range is bigger than you'd hoped. But it might even be bigger than you think, unless you did a fair number of tests.

MR. LASSINGER: Maybe somebody on the phone can help identify the exact number, but it was on the order of 20 to 40 test runs at --

MEMBER MARCH-LEUBA: How many valve, how many pieces of equipment, only one?

MR. LASSINGER: This is only one piece of equipment.

MEMBER MARCH-LEUBA: One piece of equipment around 40 times.

MR. LASSINGER: One 40 times. We'll point out Paul has a slide to this later that we still have qualification, functional qualification to be performed on these valves. So this is not the end of the testing before the valves go into service, this is just the testing to support functionality for DCA. So there's still an entire qualification test program that's required prior to any valve going into service.

MEMBER BLEY: I haven't seen the details of your qualification testing, but I've seen some others. And often you qualify if you get it to work once. You can fail a few times, and then if you finally get it to work, it's qualified. Are we like that, or do you have any idea?

MR. LASSINGER: I don't have a great sense I think off, you know, off the cuff of what the number of -- I know there are margins which are required to be applied to the qualification program. So we have margin between what you expect to happen and what the limit is. So what the valve spec limit is. So that is supposed to account for operational uncertainty and fabrication uncertainty of the valve itself.

MEMBER BLEY: Let me ask it another way. In the test that you did do, did you find open and closing was over exactly a 400 psi range, or was it

less than that --

MR. LASSINGER: It was less than --

MEMBER BLEY: And you went to 400 to give you some margin.

MR. LASSINGER: That's correct.

MEMBER BLEY: Margin.

MR. LASSINGER: That's correct, yeah. There was around --

MEMBER BLEY: Do you know the minimum and max? How close are we?

MR. LASSINGER: It's difficult to, well, it's difficult to determine the exact minimum but we did have closing pressures at around 1275. So that's a 25 psi -- or sorry, let me, I didn't get to the last bullet of this slide.

So the new range is 900 psi to 1300 psi. So we established multiple tests to show the, demonstrate the blocking at a pressure below 1300. It blocked it around 12, you know, a little bit of variation, 1270-1275, multiple test. And the release was near 950, 925 to 950. So there's a 25 to 40 psi range.

MEMBER BLEY: Okay, so pretty close.

MR. LASSINGER: Yeah.

MEMBER BLEY: So you tuned it pretty close

to what you saw in a 1000.

MR. LASSINGER: That's true, but yeah, there was also --

MEMBER BLEY: I asked a question four times ago when we talked about these, and I don't remember I got an answer that made me comfortable. Something led you to really worry about spurious operation of ECCS valves to go to this kind of elaborate function. And they don't spuriously operate very often in my experience. What is it that pushed you to do this?

Because this, you know, you don't want to have an unreliable system, and that's what you're protecting against. But I'm surprised that it drove you to this, which challenges the safety role of these valves.

MEMBER MARCH-LEUBA: It has to do with our next topic. The power to the valve is unreliable because it's not safety-related.

MEMBER BLEY: What do we mean by unreliable?

MEMBER MARCH-LEUBA: Not safety-related.

MEMBER BLEY: Well, it's highly reliable.

MEMBER MARCH-LEUBA: But these guys behind you will force you to assume it fails twice a week

because it's not safety-related. You have to be to survive that.

MR. INFANGER: You wouldn't want ECCS inadvertently actuating at that full pressure.

MEMBER BLEY: No, you wouldn't, but you sure don't want it not to work when you want it to work.

MR. INFANGER: Yeah, the timing is not real critical on the ECCS function. That's why the delay is not a problem in the analysis.

MEMBER BLEY: Delay as long as it works, yeah.

MR. INFANGER: Yeah. And it's, the release is before you hit the 900 pounds.

MEMBER BLEY: You really have no reason to expect that you would have inadvertent operation very often, if ever.

MR. INFANGER: Yeah, we don't expect it, but it is required to be postulated by the Chapter 15 analyses. And the limit is the critical heat flux depressurization criteria that we have to meet. So that's --

MEMBER BLEY: So you're it's a Chapter 15 issue you're protecting against?

MR. INFANGER: Yes.

MEMBER BLEY: Although there's no probability associated with Chapter 15 events.

MR. INFANGER: That's right.

MEMBER BLEY: I still don't get it. Go ahead.

MEMBER MARCH-LEUBA: If you have safety-related power you have to, that means you can live without it. It can fail every Monday.

MR. INFANGER: Deterministically we assumed DC power fails. So then you would --

MEMBER MARCH-LEUBA: Yeah, if it's not safety grade, you have to demonstrate that you need it.

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: Or the ECCS valve would open every Monday.

MEMBER BALLINGER: Does the operator have to demonstrate operability before startup each time?

MR. INFANGER: Yes, it's a in the --

(Simultaneous speaking.)

MEMBER SUNSERI: Dan, I don't know if this makes any difference or not, but in your testing sequence for this determination, this three to four hundred range, did you do anything to account for the fact that you're essentially preconditioning valve?

Like stroking it so many times, and in actuality this valve is probably going to be sitting for long periods of time before it has to do its function?

MR. LASSINGER: Well, it is sitting for long periods of time, but it's sitting at operating pressure and temperature. And the temperature really actually was a contributing factor, for example, to the IAB spring force. Maybe as expected, the temperature is a --

MEMBER SUNSERI: But all I'm saying is, you know, if you were going to test it 20 times and you did it today, versus the fact that it might only operate once every 18 months, that's a different story. We've seen on containment isolation valves, you test them and they work fine. Then you go to test them again, they don't.

MR. LASSINGER: I don't have any, it's a good question, I don't have any specific mechanism that I have a concern about that would be, that would change its, the IAB characteristic in the plant if it was sitting for a period of time. And actually as this member just identified, we do have tech specs to address the functionality of the ECCS valves and specifically the IAB function.

I believe there's a slide on that towards

the end about the in-service testing aspects that were actually updated as a result of this test program also. So maybe that will help.

MEMBER MARCH-LEUBA: I'm confused, can we backtrack on what we're talking about? You're talking about block and release. Block means if my pressure is above 1300, it will never open. Or does it have the specific -- tell me what this block, what this means.

MR. LASSINGER: The IAB is, you know, this differential pressure valve with the spring inside the, you know, I'll say attached to the main ECCS valve. If there's a significant enough differential pressure, so the reactor vessel pressure is too high compared to containment, the IAB rapidly moves to close, yeah.

We do have some reference figures available that we looked at previously.

MEMBER MARCH-LEUBA: No, no, just tell me --

MR. LASSINGER: So it, so the --

MEMBER MARCH-LEUBA: What's the difference between the block --

MR. LASSINGER: Yeah, the IAB will move --

MEMBER MARCH-LEUBA: Tell me the

difference between release and not block.

MR. LASSINGER: Release is if there was an initial block. So say we're at operating conditions. A solenoid valve fails. The trip solenoid valves open, the main valve control chamber tries to start venting, but the differential pressure is so high that the IAB closes and prevents the depressurization of this relief valve of the ECCS valve --

MEMBER MARCH-LEUBA: So release, the IAB valve is closed, and it has to open.

MR. LASSINGER: It's normally open. It will go to the closed position to stop the main valve from opening. So it'll --

MEMBER MARCH-LEUBA: So that's blocking.

MR. LASSINGER: That's blocking.

MEMBER MARCH-LEUBA: That's blocking.

MR. LASSINGER: And the main valve will stay closed.

MEMBER MARCH-LEUBA: So what is release?

MR. LASSINGER: Release is if there is an actual LOCA occurring, pressure will decrease in the reactor vessel to, you know, this 900 psi. And then the IAB will open and allow the ECCS valves to open. And the difference between a release and a not block is that if you were at, if you were over 900 psi

before the scenario started, basically, if you're below 900, the IAB will just never interfere and the ECCS valves will open.

MEMBER MARCH-LEUBA: I understand below 900 and above 1300. I don't understand what happens between 900 and 1300. I still don't understand.

MEMBER BLEY: I think what you need to do is walk him through if the pressure comes done, when will you get an ECCS actuation.

MR. LASSINGER: It's specified now at 950 psi, plus or minus 50 psi.

MEMBER MARCH-LEUBA: But if at 12 --

MR. LASSINGER: You'll, there's a slide on that Paul will get to on this, but we determined that the --

MEMBER MARCH-LEUBA: I give up. I give up, I don't get it. I don't -- I mean, if at 1299 you failed to block, why doesn't the ECCS open?

MR. LASSINGER: Okay, the safety analysis --

MEMBER MARCH-LEUBA: The pressure is going down --

MR. LASSINGER: The safely analysis does consider the possibility of the valves opening over a range. This slide in particular doesn't address the

release of pressure by itself. We determined that a 400 psi range window was required for blocking the release.

MEMBER MARCH-LEUBA: So 400 is the uncertainty on the set point.

PARTICIPANT: Yes.

MR. LASSINGER: That's how, that's how it was previously specified. Now we have established what we call the threshold pressure, which is 1300 psi. It will always block over 1300 psi. Now we have a release pressure of 950 psi, plus or minus 50. So --

MEMBER MARCH-LEUBA: Is that an analytical limit, or does it have anything to do with reality?

MR. LASSINGER: It's used in the safety analysis and that's supported by the test program --

MEMBER MARCH-LEUBA: Analytically.

MR. LASSINGER: And it's supported by the test program with margins --

MEMBER MARCH-LEUBA: I was basing some kind of probability that one in a thousand would always be blocked, and 1300, and one in a thousand would always be open below 900. But most of them would open at 1150.

MR. LASSINGER: No, the --

MEMBER MARCH-LEUBA: I don't get it.

MEMBER SUNSERI: I think it sounds analogous to like a safety valve that has blowdown and accumulation, right. So you got to get above a set point before it gets fully open, and you got to get way before it before it's going to fully close. And that's kind of how this thing's working.

MR. LASSINGER: That's correct, yeah. There's, the period between 1300 and the 1000 is the, that additional I'll say Delta P we need to reliably hold the IAB closed, which to your point that's how it works.

MEMBER MARCH-LEUBA: It is not normally down, and if only it goes up or down depending, when you lose power and you try to open ECCS. So there is -- is it a -- I give up. It's an uncertain.

MR. LASSINGER: Well, let's, I think I'll turn it over to Paul to talk about how this, these results were incorporated into the safety analysis. Because this is a change, you know, from the 1200 to the 900/1300. And this may, I think it may help answer a few questions with performance of ECCS in the NuScale plan.

MR. INFANGER: Yeah, and before we move on, I believe Corvallis has some information about the

number of tests that were done for the IAB. If you'd unmute your.

MR. WUTH: Yes, this is Braden Wuth speaking from the NuScale office. We, at each test condition we performed at least three repeatability runs to make sure that there wasn't any variation caused by minor changes in temperature or pressure of the condition.

And Dan spoke correctly, when we were investigating and doing sensitivity studies, we did approximately 20-40 individual runs at the facility to try and determine the block and release pressures.

MEMBER MARCH-LEUBA: So let me see if I can understand what you said and maybe helps the previous question. In this test you set up a Delta P across the ECCS valve of 1400, and then you remove power, and then see the slope. Then, and you do it three times to make sure it's different. And you do the 1350, doesn't open. Then you do it, call it 1299 so it's below 1300, and it opens?

MR. LASSINGER: We did a series of tests, you know, from 1300, 1275, you know, 1260, 1250 --

MEMBER MARCH-LEUBA: And what happens at those intermediate pressure between 1300 and 900? It sometimes opens and sometimes doesn't open?

MR. LASSINGER: Below, well, it either blocks or it doesn't block.

MEMBER MARCH-LEUBA: Yeah.

MR. LASSINGER: And right, so we establish that it will always block around, you know, in that test configuration, and which is basically identical to the IAB final valve design. It blocked at around 1250 psi, and so we did test repeatedly at 1275 to establish that it blocks at 1275, 1275 each time, which is below the 1300 which we established as the limit.

So it, so in terms of safety analysis credit, we can always say that it blocks over a pressure of 1300 psi. Safety analysis does still take account for the fact that it may not block between that range that you're talking about that range that you're talking about --

MEMBER MARCH-LEUBA: So there's an uncertainty whether it will block or not --

MR. LASSINGER: That's right --

MEMBER MARCH-LEUBA: Between 1275 and 950.

MR. LASSINGER: And in fact, I think you're talking about different considered staggered releases or blocking that were considered because of that effect.

MEMBER MARCH-LEUBA: For a spring and a well very defined surface area, that's a very large uncertainty. I mean, it's a regular spring --

MEMBER BLEY: It's not untypical of these kind of valves. We've, I've never tested one, and it's been a long time since I tested them, never, ever tested or had them tested and they came anywhere close to their set points. So these are better than I'm used to seeing.

MEMBER MARCH-LEUBA: No, no, there is only one valve that has one set point.

MEMBER BLEY: The set point was, I assume, 1200.

MEMBER MARCH-LEUBA: No, he's telling me that sometimes it opened, sometimes it didn't.

MEMBER BLEY: Yeah, when it was set to operate at 1200, if I understood you guys right.

MEMBER MARCH-LEUBA: You're going to change the set point. It's given by the spring and the surface area. There is no adjustment, it's whatever the set point is.

MEMBER BLEY: I'm sorry, but they all have adjustment.

MR. LASSINGER: The issue is the amount of Delta P between the reactor and containment to

reliably hold the IAB in the closed position. When you start getting closer and closer to the release set point, then the force balance across the disk is approaching zero.

So it does not reliably hold it closed and it could start to leak as you're approaching, you know, the spring force, the force balance per spring force is going to zero across the disk. So we need that, you know, when I say 400 psi, I mean that's the limit. But we're a little bit inside of that in terms of operations.

MEMBER MARCH-LEUBA: So the issue is the valve actually moves, but it leaks. You need more force to hold it.

MR. LASSINGER: That's why we need a wider range, and you can't, you know, that's why we need the range from 900 to 1300 psi, correct.

MEMBER MARCH-LEUBA: And you tested it 1000 times to make sure that that's a range, or you tested it three times. Oh, 45 times under pressure. They only tested three that it worked.

MR. LASSINGER: Well, we tested a number, it was more than three. There was a number of test series, maybe more like ten test series to establish that blocking pressure. Establish that it blocks

below 1300 psi. So any pressure above 1300 psi is more pressure to close the valve. So there's no, that's better performance --

MEMBER MARCH-LEUBA: What I understand, this is a probabalistic assessment that you did. Sometimes it moved, but it didn't, but it leaked, and it wasn't really closing. Sometimes it moved when it closed and sometimes it didn't, in between 900 and 1300. If you're above it, it always worked.

MR. LASSINGER: That's right.

MEMBER MARCH-LEUBA: But how many runs did you run above 1300? More than three?

MR. LASSINGER: Well, you know, from my perspective, running any pressure below 1300 is much more conservative because that's the less pressure to hold the IAB closed. Any pressure higher holds the IAB closed better, I'll say. So there's no --

MEMBER MARCH-LEUBA: It's of no safety concern. I, yeah, no, I -- you guys need to understand what you're doing. I mean, if you cannot explain to me what you're doing, I don't think you understand what you're doing. You cannot explain to me what this means. I'm sorry, must have half-caffeine in here. I mean, do you understand what I'm saying?

MR. LASSINGER: Yeah.

MEMBER MARCH-LEUBA: I mean, we've being going on for 20 minutes, and nobody at this table understands what you did.

MR. INFANGER: Okay, we'll take a look at the analysis that we did before we did the testing and afterwards. So prior to getting the new range for the IAB, we had assumed that the ECCS valves would open between 1000 and 1200 PSID. So, and above 1200 PSID it would always block. And that was the assumption we had made, and it was supported by the initial proof-of-concept testing.

So, and we assumed that all of the valves opened at the same time. It was a pretty small range so we, at the time we didn't look at staggered opening. So that was the analysis that we had done and submitted in August of 2019.

MR. CHITTY: Hey, Paul, this is Mark Chitty. We have the test engineers stand by here that he can maybe add a couple of things.

MR. WUTH: Ah, yes, this is Braden Wuth speaking again. We performed 24 quality test runs to, above the IAB set points, showing that it blocked each and every time. We did it at a variety of pressures, some well above and some, as Dan said, right at the

1275 PSID mark, to show that it closed and blocked the main valve from opening each time.

On top of that, as Dan stated, we performed a number of sensitivity studies while we were trying to determine the, where the mark was. But then once we had determined that we were shooting for 1300 psi as the limit, we did 24 other tests to support that.

MEMBER MARCH-LEUBA: How about the lower limit? The 900?

MR. LASSINGER: Yeah, every test that we performed in which the IAB blocked initially we depressurized the test facility. And the release was very consistent. It was initially, you know, with the, closer to the 1000 limit that we had. But there was, we adjusted the shim for the IAB, and it released in 950, plus or minus 10 to 20 psi every time. So that's 24, you know, plus the initial tests that we did, so.

MEMBER MARCH-LEUBA: So the uncertainty is with the blocking.

MR. LASSINGER: That's correct. The release is very consistent.

MEMBER MARCH-LEUBA: Are you doing a ramp, or are you doing steps? For the release.

MR. LASSINGER: It's a ramp. It was a specified ramp to correlate --

MEMBER MARCH-LEUBA: Because if the issue is with leaking because it's not closing properly, the slope on the ramp will give you a different answer.

MR. LASSINGER: That's right. We did establish the slope, yeah, the rate of pressure decrease --

MEMBER MARCH-LEUBA: It's irrelevant, it doesn't make any --

MR. LASSINGER: Taken from safety analysis results, so. Thanks, Braden.

CO-CHAIR KIRCHNER: So just to clarify then. So for the release part of the experimental runs, these were all ramps, not steps.

MR. LASSINGER: That's right.

CO-CHAIR KIRCHNER: I think that's the right thing. It's, because that's what the valves are going to see in the actual blowdown conditions.

MEMBER BLEY: Although it's back to what Matt said.

CO-CHAIR KIRCHNER: Yeah.

MEMBER BLEY: For the first one, yeah. After you've operated it one time, unless you let it sit for months without pressurizing, you're going to

be really consistent. I'm surprised they're not consistent on the high end too, given consecutive sets.

CO-CHAIR KIRCHNER: Okay. Keep going, Paul.

MR. INFANGER: Okay, so the analyses that were of interest was the containment peak pressure in Chapter 6, loss of coolant in Chapter 15, and the inadvertent opening of an ECCS valve in Chapter 15.

So we revised the accident analysis assumptions to match the, what the testing had determined. And the, so we assumed that the IAB blocks above 1300 PSID and opens between 900 and 1000 PSID.

And we submitted the revised analysis in September of 2019, and then the staff came out to Corvallis with a group of about ten people. And the primary subjects were to review these calculations and review the testing and a few other issues. But this was one of the focus areas.

And so then we submitted, so the NRC, you know, they reviewed the calculations and saw what changes we were going to make to the FSAR, we had markups and had submitted them. And then we submitted the formal DCA revision 4 with the new updated results

in January, just a month ago, a few weeks ago.

Okay, so the impact of these changes on the IAB to the analysis resulted in increased peak containment pressure. The reason the containment pressure went up was that the limiting event is an inadvertent opening of an RRV. So it's one of the recirc valves, so you're releasing fluid to the containment.

And the second, in the new analysis, the second RRV opens at 1000 PSID, and the RVVs don't open for another 100 pounds. So what happens is you get a second recirc valve opening up. So you get more fluid into containment, the greater amount of mass then results in a slightly higher pressure. So that's was a difference.

Before, we had assumed that the valves, the vent valves opened at the same time as the recirc valves. You get a, the containment pressure and the RCS equalized pressure very quickly. And therefore you, it, then that decreases the amount of mass that gets out into containment quickly, so.

CO-CHAIR KIRCHNER: So that one, Paul, I reviewed Chapter 6, Jose reviewed Chapter 15. On Chapter 6, I think I must have been looking at Rev 3. So how much, you had from Rev 2 to Rev 3, to, at least

on paper, redesign the containment vessel pressure rating. It was originally 1000.

After you analyzed one RBV opening, you were close to that and you redesigned to 1050. How close now are you with this scenario to the design limit of 1050 psi?

MR. INFANGER: I have the results on the next page. But it went up by about eight pounds. So it went from 986 to 994, if that gets the various results. But it's less than one percent increase, so it was relatively small.

And that, again, that was just due to the large amount of water that gets out into the containment prior to the pressures equalizing.

For the --

CO-CHAIR KIRCHNER: Did you do a sensitivity analysis, since you have a range now?

MR. INFANGER: Yes.

CO-CHAIR KIRCHNER: That's broader than your initial range of where actually it releases. So your assumption is it holds until 1000 psi, the second valve?

MR. INFANGER: Yeah, we looked at a 900 pound opening, a 1000 pound opening and place in --

CO-CHAIR KIRCHNER: Well, I would be

worried about if it opened at 1100 or higher.

MR. INFANGER: Well, that's, the testing showed that it always opened between 900 and 1000, so we used 1000 as the analytical limit.

CO-CHAIR KIRCHNER: Okay.

MR. INFANGER: It's the highest pressure it would open.

CO-CHAIR KIRCHNER: Okay, thank you.

MR. INFANGER: In the LOCA analysis, what we're concerned about is the water level above the top of the active fuel. And the staggered IAB opening did not make a difference in that analysis. The limiting case was the IAB releasing at the lowest pressure, which was 900 PSID. So that resulted in the lowest water level above the top of active fuel.

And in the, for the inadvertent opening of an ECCS valve, the change in IAB set point did not affect that analysis. The MCHFR occurs when the valve opens within half a second of 0.3 seconds, so that it happens so quick that the, the staggered in the change in the IAB set point was not, did not have any impact.

But since we had redone the analysis, we had for other reasons we had made some changes in the model, and that did result in a slight change to the MCHFR. But again, it was not related to the IAB

change, but it was, we did this calculations over again, so the change did impact the values in Rev 4 of the FSAR.

CO-CHAIR KIRCHNER: What did drive those changes?

MR. INFANGER: There was, they were pretty much one was an error correction in flow resistance. And then there was a conservative reactivity feedback that we did that was beneficial for reload analysis. So it was something that we decided that we wanted to make the change. But then that did impact a little bit of our market.

CO-CHAIR KIRCHNER: Other decision making and analyses were involved in the reasons for those changes.

MR. INFANGER: Right, and they just happened to be at the same time --

CO-CHAIR KIRCHNER: Okay.

MR. INFANGER: As making this change. And we verified that the 1300 was adequate to ensure that the ECCS valves remained closed until the reactor trips, it's one of our acceptance criteria.

So as you can see here what the actual values were. So the one, the containment pressure design is now 1050 PSIA, and the results went from 986

to 994. The minimum water level above the top of active fuel and a small break LOCA went from 1.7 feet to 1.5 feet, and that's just the transient condition. And it goes up to about ten feet after things steady out.

And we invert an ECCS valve. The MCHFR limit is 1.13, and the value dropped about 0.09. So it's a small change to that.

So one of the things that came out of the audit is we looked at how we were going to do the IAB testing. So we established, in order to ensure the safety functions are met, there's a technical specification that has a surveillance that says to do the testing in accordance with the in-service testing program.

We added a section in FSAR 3.9 for an alternative authorization, which basically shows that we will for pre-service testing, we will verify the set points for the RRV and RVVs for all inadvertent actuation block devices on all of the NPMs. So in the pre-service they all get tested to make sure they block at the above-the-threshold pressure and they open in the release window of 1000 to 900 pounds.

And then after the initial pre-service testing, they'll be operational testing required by

the tech specs. And for the first refueling of the first NPM, we're going to do the same tests as the pre-service test where all the valves are tested. And then the second NPM refueling, we're going to test one RRV and one RVV. And subsequent outages are per the ASME OM code and the mandatory appendix.

MEMBER BALLINGER: Now, what if between the first startup and the first refueling, you have to shutdown and depressurize? Do you restart the plant? Do you have to verify that the IAB works? In other words, do you have to verify operability at each startup?

MR. INFANGER: No, the testing is just as stated there, that we're just going to test it just for the first NPM we'll test all the set points. Not something that's verified every shutdown.

CO-CHAIR KIRCHNER: Indirectly, you're going to do a test on each of these towels when pressurize the primary system. At least you'll know the upper set point is working.

MEMBER BALLINGER: That's true, that's true.

CO-CHAIR KIRCHNER: Just think about the sequence they have to go through. They've got to pump down containment, and then you're going to have a

differential pressure of, what's your normal operating pressure, 18 --

MEMBER BALLINGER: 1850.

CO-CHAIR KIRCHNER: Versus subatmospheric. So you'll at least know the upper end is working.

MEMBER BALLINGER: Yeah, but BWRs and stuff like that with pilot-operated relief valves, they have to verify operability before they go above 150 psi or something like that when they start up. I think.

CO-CHAIR KIRCHNER: Well, they have to reset their valves each time.

MEMBER BALLINGER: Yeah.

MR. INFANGER: And then, I think Dan mentioned, as Dan mentioned earlier that we will be doing the functional design and qualification testing for ASME QME-1 to verify the valves function as designed.

So on the conclusions, what we've, this change to the IAB range resulted in a containment peak pressure slightly more limiting, about 8 psi, due to the explicit evaluation of ECCS opening at different release pressures. The LOCA minimum water level drops slightly due to the lower IAB release pressure of 900 pounds.

The inadvertent ECCS valve opening at a slightly more limiting MCHFR due to other changes made to the model but not to the IAB change. All of the results showed that we had margin to the acceptance criteria, and that we established pre-service and operational testing programs to assure that safety functions are maintained.

Any other questions?

MR. LASSINGER: I might add just another interjection, you know, explanation of the IAB valve testing. It's not that, you know, I think maybe there's an impression that we don't know how the valve behaves between 1,000 psi and 1,300 psi. We always expect, and in the test program that we carried out, the IAB to block around 1,200 to 1,250 psi every time.

You know, like was said, it's a spring and a disk. It's nothing really complicated going on there. But it does not block at something less than 1,200 psi. Or, I guess let me say it differently.

Given a block, it won't release until 950. But, if you start with an initial pressure of less than 1,200 psi, you know, testing shows that it may not block, it doesn't block, it won't block below 1,200 psi initial pressure. Even though if it does block initially it won't release until the 950.

And, you know, that, I think that's a nuance. We understand how --

MEMBER MARCH-LEUBA: If I'm sitting at 1,200 psi, how I don't -- if I sit at 1,200 psi for a week and I will turn the power off the ECCS valve, so I demand an opening, you're saying it may remain -- what does the IAB valve, the IAB valve moves, or it fails to move, or is it running?

MR. LASSINGER: At 1,200 psi it's right on the edge. IAB may not block, and safety analysis accounts for that fact that may not block.

MEMBER MARCH-LEUBA: So, ECCS would open?

MR. LASSINGER: Right.

MEMBER MARCH-LEUBA: But, if the valve is blocking --

MR. LASSINGER: But if it does block.

MEMBER MARCH-LEUBA: -- then the valve now is blocking position. You need to lower the pressure, otherwise --

MR. LASSINGER: Yes, yes.

MEMBER MARCH-LEUBA: - 950, whatever it is, otherwise it will not move.

MR. LASSINGER: If it does block it will not release until that 950 psi plus or minus 50. That's the delta between which it doesn't block given

an initial pressure to the release given, given an initial block.

MEMBER MARCH-LEUBA: So, when it is blocking, the surface area is different because part of the surface area is covered, and therefore you need a higher delta P to produce the same force. Whereas, if it's not closed you have a larger surface area and, therefore, your behavior is different. Is that the mechanism?

You know what I mean, you have this, I mean, if it's hitting against the surface, and it's a ring, you do the analysis more.

MR. LASSINGER: It's the surface area is on the bottom of this bellows of the IAB.

MEMBER MARCH-LEUBA: Those are different, there are two.

MR. LASSINGER: Right. And so the surface area is really still the same.

MEMBER MARCH-LEUBA: Okay. I think what I said, think about it.

MR. LASSINGER: Yeah.

MEMBER MARCH-LEUBA: Is the reason why this the status is.

MR. LASSINGER: Okay. So, I just wanted to add that.

CO-CHAIR KIRCHNER: Members, further questions? Okay, thank you, Paul and Dan.

Thank you, Paul and Dan. We will now turn to the staff. Again, we are in open session. And we're pretty close to being on time. Thank you.

Okay. I am going to turn again to Bruce Bavol, and he will make an introduction.

MR. BAVOL: Okay. My name is Bruce Bavol. The name of the presentation is The Inadvertent Actuation Blocked IAB Valve. And it's going to be discussed from our technical reviewer Tom Scarbrough.

Just let me know when you want to change.

MR. SCARBROUGH: Okay. Thank you.

I'm Tom Scarbrough. I'm currently the acting chief of the Mechanical Engineering Branch until a permanent replacement's assigned.

Your questions about the IAB valve are terrific because that is what we asked about when we first starting looking at this design back in, like, 2017. We started with some initial audits and found out more about the valve, you know, how it worked, and what testing they had done to demonstrate it.

And from that we went even further and discussed it with them.

So, first of all this is what Dan was just

talking about in terms of it's a first-of-a-kind valve system design, for the emergency core cooling. There's three reactor vents to the top and two 2-inch reactor recirculation valves on the side.

And each RVV and RRV has a, it's a main valve, an IAB valve, solenoid valve, and solenoid reset valve that's connected by small tubing.

I'll try to stay away from any numbers, right, because I don't want to get into proprietary.

And during 2019 NuScale conducted ECCS valve design demonstration testing. And I'll talk about what that is. And it was intended, it's intended to satisfy 10 CFR 5247(c)(2) and 50.43(e), Echo, for this first-of-its-kind design.

So, let me show you what are in those requirements.

5247 requires that a DCA that uses a reactor, has a reactor design that uses passive means to accomplish functions must have an essentially complete design and must meet 50.43(e).

Now, 50.43(e) is a very long requirement. But just boiling down to just the main part here, has to do with if it's a reactor, a passive reactor and its safety -- means to accomplish its safety functions will be approved only if the performance of each

design safety feature has been demonstrated -- and it's not qualified, it's demonstrated -- through either analysis testing, experience, combination thereof. And since it's the first time, there was none of that available.

And there has to be sufficient data to support any analysis.

So, what I did when we first started looking at his review was what had NuScale performed and to support this first-of-a-kind design, and did it satisfy 50.43(e).

So, I went back to the notice, which is August 28th, 2007, and in it it talks, the Commission talks about what it intended regarding this status finding. Now, it used the term "qualification." But when you read it, it wasn't reading qualification as, like, Appendix B to Part 50, that type of qualification. It was more of a design demonstration aspect.

And I talked to OGC about it and what their interpretation was, and we came up with agreement of what we would expect for them for a design demonstration aspect.

Okay. So, so what did we do? Okay. We started our evaluation. We reviewed the design

description BCA. And that was very general, as you know, it's very general. So, we had meetings with NuScale. And also we stated conducting audits of their designs and their design documentation.

And part of that was their Failure Modes and Effects Analysis. We looked through that, looked through what they assumed were all the possible ways that this valve system could fail. We had a number of comments on that.

We conducted our initial audit, onsite audit at Carderock in 2018, in May of 2018.

MEMBER BLEY: Excuse me.

MR. SCARBROUGH: Yes, sir?

MEMBER BLEY: Did they make any changes from your review of Failure Modes and Effects Analysis.

MR. SCARBROUGH: In the Failure Modes -- yes, they did.

MEMBER BLEY: Okay.

MR. SCARBROUGH: They went back and --

MEMBER BLEY: They agreed with the things you raised?

MR. SCARBROUGH: Yes. Yes. They went back and modified it in response to we had a specific audit report on the Failure Modes and Effects

Analysis. And as part of the follow-up they went back and revised that and expanded some of those discussions they had.

MEMBER BLEY: Okay.

MR. SCARBROUGH: Okay. So, in May of 2018 we went to Carderock and looked at what they had developed so far. And Dan had mentioned that back in 2015 they had conducted what they called proof of concept testing.

It was, it was pressurized water but it was tap water; right? It was just 70 degree Fahrenheit water. There was no flashing, no boration, anything like that. And they indicated based on their discussions that it worked fine.

Now, they didn't have the hydraulic lines weren't the same length. It was more, it was a proof concept; right? Do you want to put more money into this design or do you want to go a different route. Right?

So, based on that we reviewed the proof concept report. And the conclusion was, yeah, we think we can span this discuss -- this design. We can, you know, do more testing, more evaluation. So that was the conclusion.

So, we came back and we look at that. And

we talked about the various aspects of that. And decision of the NRC was that was not sufficient to meet 50.43(e). You needed to have high temperature, pressurized water, boration. You needed to show flashing. You needed to show these things because you're demonstrating this safety feature. This is the safety feature of the plant; right?

So, so that's what our decision was. We went back and relayed that to NuScale. NuScale was very receptive. They understood our concern, our interpretation of 50.43(e). And that, at that point they started the more detailed design demonstration testing that they conducted over the summer. And so, they conducted that at Carderock on Long Island. And we were there many times to watch the testing.

Also, back in May of 2018 and in June of 2019 we had the NRC Vendor Inspection Branch participated with us. They actually conducted a sort of parallel inspection. While we were there conducting our audit, they were conducting an inspection of NuScale's oversight of the Carderock testing activities. What would happen when they had issues, right, when you'd actually find issues. Right?

And every time they were making sure that

Carderock was following the provisions that NuScale set up for when you have testing issues, how you deal with them, corrective action, that sort of thing. And those were successful. There were inspections, Vendor Inspection were there for that.

So, after those sets of audits and such, our audit report, which is in December '19, 2019, we reached our status of where we are with the design demonstration testing to satisfy 5040 -- 5247(c)(2) and 50.43(e).

MEMBER BLEY: Tom.

MR. SCARBROUGH: Yes, sir?

MEMBER BLEY: You brought up boron.

MR. SCARBROUGH: Yes.

MEMBER BLEY: And if you get any kind of leakage of the reactor coolant, in the reports are heat and boric acid crystallizes, that could really change how this valve operates. Now, they didn't do any testing to evaluate that. That sometimes is over a long period of time.

MR. SCARBROUGH: We did, we did -- we talked to them quite a bit about that.

MEMBER BLEY: Did you? Can you tell us what --

MR. SCARBROUGH: And one of the things we

talked about was that whenever they shut down for refueling they flush all those lines. All those lines have to be completely flushed out with new CVCS water.

And so, and then when they shut down they're going to disassemble --

MEMBER BLEY: De-min water without any boron?

MR. SCARBROUGH: Well, actually they have to, they have to refill it with CVCS borated water.

MEMBER BLEY: Borated water.

MR. SCARBROUGH: Because they have to match what's in the rest of the coolant system.

MEMBER BLEY: So, the flushing is done with borated water?

MR. SCARBROUGH: Yes. Right, right. Yes.

So, but they're going to flush it every outage as part of the whole process. So that was the concern we had. That was part of the Failure Modes and Effects Analysis, you know, what could happen if you had precipitated boron in your system? Since they have these actions they're going to take to make sure they do not have that happen.

All right, so --

CO-CHAIR KIRCHNER: To follow up on that one, Tom, that all sounds very good, but if the

refueling interval is 24 months -- if I have this right -- then you're not likely -- that valve system, and I'm glad you're calling it a system because it really is, the valve system then is going to sit stagnant at hot conditions. Let's assume it runs at full power for 24 months, these valves are sitting in the hot containment environment, quite a bit different than a valve sitting up on the top of a pressurizer.

So, was there any -- I assume you did, you mentioned prototypical testing at temperature and pressure --

MR. SCARBROUGH: Uh-huh.

CO-CHAIR KIRCHNER: -- with boron in the water.

MR. SCARBROUGH: Right.

CO-CHAIR KIRCHNER: But this effect that Dennis alludes to I think you won't see until the lines sit there stagnant for a considerable period of time.

MR. SCARBROUGH: Right. Now, they did, they did have -- stop me if I get to far -- that they had heat racks; right? So, they made sure that this was all up to temperature, right, throughout this entire --

CO-CHAIR KIRCHNER: Okay.

MR. SCARBROUGH: -- testing process. And part of the discussion was could any of this boron start to precipitate.

But they'd done analysis to show that they're going to stay above the precipitation temperature, the point where it might come out of solution; right?

So that, that's their, that's what they're seeing right now.

But, as part of the qualification process on QME-1, right, it's going to be even more detailed in terms of looking for things like that. Right?

CO-CHAIR KIRCHNER: Okay.

MR. SCARBROUGH: This is, this was -- this, the intent of this design demonstration testing is to demonstrate the concept.

CO-CHAIR KIRCHNER: Proof of concept, yes.

MR. SCARBROUGH: Yes. It's an aspect to demonstrate the physics, that the physics will work. But they still have to go through all the qualification testing of QME-1.

And part of what they learned from this -- and I'll mention this later in the slides -- the final test report has a lot of very good lessons learned from the testing; right? That they have to, they

indicate they have to fold this into our QME-1 qualification program.

And so, part of that is they need to make sure that they don't have any issues regarding temperature effects of sitting for a long period of time. They may have to have testing concepts where they let it sit for quite a while, that sort of thing.

So there's the QME-1 qualification is the next step. This was to demonstrate the physics, did it satisfy 50.43(e) in terms of will it, will it work.

And based on that, yes, they have demonstrated that it will work but they still need to qualify it for its, you know, the entire application for the valve. And so, that's the next step actually.

MEMBER BLEY: When does that happen in the process?

MR. SCARBROUGH: Well, I'm sure they're going to start soon because they're going to need to have all of that completed. They'll need to prepare their qualification plan.

We saw a draft while we were there during our, our audit. But there are other places they need to update it based on all these lessons learned. So, what they'll do is -- I understand Carderock's already working on their, their sort of prototype valves and

cells and that sort of thing.

So, I'm sure as soon as the next step comes they will start processing this qualification.

MEMBER BLEY: When does it have to be done, by the COL time or?

MR. SCARBROUGH: Yeah, it -- oh yes, it has to be completed. There is an ITAAC that's for qualification.

MEMBER BLEY: So it's an ITAAC?

MR. SCARBROUGH: For QME-1 qualification. And so they will have to have that completed.

But, as we've done with Vogtle 3 and 4, we don't wait till the end to close out or review an ITAAC, especially when it comes to valves. Right? We, you know, like, we're currently looking at Vogtle 3 and 4 for their qualification of their valves. And they're not going to start up for quite a while.

MEMBER BLEY: I'm sorry, this doesn't have anything to do with this one, but so you're getting ITAACs completed well before, a year or two before?

MR. SCARBROUGH: We can review them. Or we could close them out if we wanted to --

MEMBER BLEY: Okay. And you're actually reviewing, yeah.

MR. SCARBROUGH: -- but we're, we're

reviewing them. RRVs; right?

MEMBER BLEY: Yes.

MR. SCARBROUGH: Thank you.

Okay. As the summary of their, of their actually their design demonstration testing. So, they conducted, they used a flow valve, a 2-inch flow valve that's sort of a mock-up of the RRV. And it was the same valve they used for the proof of concept 2015. But they're basically saying it has the same sort of internal dimensions and such.

And so, they had three main test objectives. They were: demonstrate the main valve functions at the operating temperatures and pressure fluid conditions; they wanted to show that the IAB valve functions at operating temperature and pressure; and then they had it set up for the IAB valve and the trip lines will function under the fluid temperature conditions with the boron.

So, they wanted to have -- and a lot of times when Dan was talking about the IAB valve not operating properly, a lot of that was in the IAB valve functionality testing that they saw.

So, so the main valve they wanted, you know, they wanted the main valve to remain closed, right, until the pressure was reduced to a specific

value and have it open.

The IAB, they wanted it to close quickly. It sits there open -- like, I'm not going to say how far it has to sit there open -- but it sits there, it's open. Right? And it's a seal quickly, very quickly, within very, very short amount of time. Right? Has to seal, right, and hold the pressure so that the pressure for the main valve doesn't bleed off and allow the main valve to open prematurely. So, it has all those functions.

And then, and then later as the pressure drops, so like if it's a real low pressure, the pressure drops and then the spring will pull it back open.

So, all that has to happen. So they wanted to, and then wanted to show that in the IAB valve they wanted to make sure that the flashing of the borated water. Because one of our concerns was when this borated water starts flashing, is it going to come out and clog up that line and clog that. Well, they ran a number of tests and they did not see it happening.

So, you know, based on that, I mean, they showed it didn't work, i.e., it was kind of guessing whether or not it would block it or not, but it seemed

to work okay even with the flashing of borated water.

CO-CHAIR KIRCHNER: Did you folks get to observe the test?

MR. SCARBROUGH: Yes, absolutely. Yes. We went there in June and went through a walk-down of their whole -- their test set-up. We watched them run the test.

Then we went back later again in July to watch them run the addition of borated water test. So we, we made sure that we had enough eyes on what they were doing that we were comfortable with the whole process.

MEMBER SUNSERI: And that sequence of events or that, all that work is well documented --

MR. SCARBROUGH: Yes.

MEMBER SUNSERI: -- in the audit report?

MR. SCARBROUGH: Yes. My audit report is probably one of the longest audit reports you've ever seen.

MEMBER SUNSERI: Yeah.

MR. SCARBROUGH: But it's --

MEMBER SUNSERI: Well, not the longest but it's a good one.

MR. SCARBROUGH: Yeah. If you look at the -- I put the ADAMS number right here. And you can

feel free to look at it. I mean, we want to make sure that in the future if there's another reviewer picking this up that they'll have the whole story.

MEMBER SUNSERI: Yeah. It's in ADAMS, your report.

MR. SCARBROUGH: It's in ADAMS, yes, sir.

All right. So, they tested. Test conditions were saturated steam, cold water, nominal temperature water. And the pressure range went all the way from 30 to 1850 psi. And so, they had a whole series. And as they mentioned on the phone, you know, there were three for each set of tests; right? So, they had three runs to show that it was set.

Okay. All right. So, the test results. Right.

Okay. So, in August they reported that the testing had been completed. But the test revealed that certain aspects of the original valve system design did not perform as expected. And that's some stuff that Sam was talking about.

Like, for example, when the IAB valve, one of the issues was the IAB valve did not close; right? It was supposed to close at a certain pressure, and it didn't close, right? And so the main valve opens immediately. Right? It loses all its pressure and it

just opens. So, that's not a good thing; right?

Another time the main valve did not open at all; right? The IAB valve shut; right? And then when the IAB valve later reopened, the main valve still stayed closed, which is not a good thing. Right? And because they have to get the max circulation going; right?

And so --

MEMBER SUNSERI: Did they, did they want to close that?

MR. SCARBROUGH: Yes. It has to do with it's very important in terms of its rate, right. What's going on is your -- the pressure is being relieved, right, and there's an orifice in the main valve disk that's feeding from the reactor. So, you have to bleed out this line.

That's one of the reasons why we were very interested in what would happen with the flashing in the borated water because you have to bleed out that fast enough to be able to drain that, that main chamber to get that pressure out of there, you know, to allow the main valve to open like it's supposed to at the right pressure. Right?

So, so that was one of the things that they found, you know, from their issue.

So, so the design changes -- and I won't give any numbers because that's all proprietary -- but they had to reduce that main valve control chamber orifice diameter; right?

MEMBER SUNSERI: Uh-huh.

MR. SCARBROUGH: They had to reduce the IAB valve portlet because there's, there's between the main valve and the IAB valve there's a port going through there. And that port needs to be the right size to be able to bleed that pressure off at the right rate to be able to have the main valve operate properly. And so they had to adjust that.

MEMBER MARCH-LEUBA: If the problem was this drain orifice was this much, then it will never open.

MR. SCARBROUGH: Yes.

MEMBER MARCH-LEUBA: It never did open?

MR. SCARBROUGH: Well, yes.

MEMBER MARCH-LEUBA: Or was it one time, a random fail?

MR. SCARBROUGH: No. What happened with the main valve not opening was it just eventually it opened. Because there's a small spring in the main valve that tries to open it; right?

MEMBER MARCH-LEUBA: That's when you get

to 100 Psid.

MR. SCARBROUGH: Yeah, right. So, yeah, that's right. It's very, very small.

MEMBER MARCH-LEUBA: Yeah, that wasn't opening.

MR. SCARBROUGH: It's very small. But, yeah, eventually that little spring opened it. Sure.

MEMBER MARCH-LEUBA: But it all failed.

MR. SCARBROUGH: Yeah. So they, so they needed to adjust all that relationship.

And then the last change there is the valve trim. Because we talked about set points, but there really aren't any set points in this valve, it's a design valve; right? They, they established the spring time, the spring rate of how much load the spring's going to provide by a shim in the design; right?

It's not something like a normal release valve, you can go out and do the rings and adjust them; right? So, you know, and actually that's one point we've asked them to make. If you looked at the latest version of the Rev 4, they go through and they take out places where they set checkpoints; right? Because it's kind of misleading.

They use the word checkpoints when they're

talking about this type of valve.

CO-CHAIR KIRCHNER: It's hard wired.

MR. SCARBROUGH: Yes, it's hard wired. Right. It's part of the design. Right.

CO-CHAIR KIRCHNER: Part of the design.

MR. SCARBROUGH: You can't, unless you pull the thing off and disassemble it, you cannot adjust where that spring is going to go.

MEMBER PETTI: Is the valve in a high radiation field?

MR. SCARBROUGH: Yes. Yes, well, yes, it's sitting -- well, it's a valve system so it's sitting -- but the valves are sitting on top of the reactor vessel or on the side of the reactor vessel. So, yes, it's very high.

MEMBER PETTI: I mean, I just worry about the spring constant change. It's on, yes. The spring constant changing because of VPA on the steel changing its modulation.

MR. SCARBROUGH: Yes.

MEMBER PETTI: So, it's, you know, it's K.

MR. SCARBROUGH: Right. Right. And that's, and that's one that arised the first time they shut the plant down, right, because you can't really operate this during the plant operation, they're going

to pull all five of those IABs off and test them for their press -- you know, the release pressure and threshold pressure. And if they see any changes they're going to have to adjust it.

They might if they see, if they see differences they're going to have to go in and maybe do mid-cycle outages or something like that. But that's the first, that's the first thing they're going to check: is there something like that happening to these springs?

MEMBER PETTI: Some sort of aging effect, --

MR. SCARBROUGH: Yes.

MEMBER PETTI: -- if you will.

MR. SCARBROUGH: Right. Right. And so that's something they're going to have to look into.

Now, what they did was -- and as Dan was talking about, in the original DCA it was 1,100 Psid plus or minus 100. And they determined that that was not possible. So now it's basically 1,100 plus or minus 200.

But, as Dan was kind of telling you, the wording in the DCA is more complicated than that. I won't read it, but it has to do with 1,300, and 950 plus or minus 50. So, but just from a slide

perspective it's roughly that.

So, then they also had to do one of the things we asked them about was, okay, how does the RVV operate, right, because that's a 5-inch valve, right. So they tested the 2-inch. So, they need to relate that.

So, in their final test report they went through and talked about the discharge volumes and how that would be different. And they calculated, you know, the short time differences that might occur from that. And they did run some tests with steam which would be the RVV condition.

So, based on that they were able to show that from a physics perspective, you know, that they demonstrated from the 50.43(e) perspective that this was a reasonable extrapolation of the testing.

So, they still have to do the QME-1 qualification testing of the RVV.

Okay.

CO-CHAIR KIRCHNER: One valve set sits in a different environment than the other. The RRVs are going to be sitting in a steam environment. The RVVs are going to be sitting in a water environment.

MR. SCARBROUGH: Well, no, actually all of them are --

CO-CHAIR KIRCHNER: Oh, I misspoke. Maybe I got it --

MR. SCARBROUGH: Well, no. The --

CO-CHAIR KIRCHNER: The RV on the top --

MR. SCARBROUGH: Yes. Those are RVVs are on the top.

CO-CHAIR KIRCHNER: But the RRVs on the bottom and they'll sit in a water environment.

MR. SCARBROUGH: Not normal. Only if an accident occurs.

CO-CHAIR KIRCHNER: Only in the accident.

MR. SCARBROUGH: That's what kills. Yeah, they'll be sitting there just in the containment, you know, sort of environment.

CO-CHAIR KIRCHNER: No, what I meant was the valve seat is sitting in a water environment.

MR. SCARBROUGH: No. No, it -- No, that area won't be wet. It will still be dry.

CO-CHAIR KIRCHNER: No, I'm talking about the valve that's on the inside.

MR. SCARBROUGH: Oh, on the inside.

CO-CHAIR KIRCHNER: Yes. That's going to sit in a water environment, and it's hot. And it's been in a hotter environmental --

MR. SCARBROUGH: Yes.

CO-CHAIR KIRCHNER: -- situation than upstairs in the, where the RVVs.

MR. SCARBROUGH: Right, right.

CO-CHAIR KIRCHNER: And the RVVs are going to be sitting in a steam environment, which is more typical of the target rod valve on a pressurizer.

MR. SCARBROUGH: Right. So, you know, these are sort of typical flow valves, yes. So, you're right, they'll -- that one at the RRVs will be sitting down below. Right.

CO-CHAIR KIRCHNER: There's two on the bottom.

MR. SCARBROUGH: And they'll have that water temperature.

So they're going to have to evaluate that as part of the qualification testing to demonstrate that from that specific application they won't have, you know, leakage issues and such with that seat sitting there.

CO-CHAIR KIRCHNER: I worry less about leakage than it's just welding itself in position.

MR. SCARBROUGH: Well, it's --

CO-CHAIR KIRCHNER: And "weld," I use that word loosely.

MR. SCARBROUGH: Yes. Yes. Yes.

CO-CHAIR KIRCHNER: The boron environment, hot conditions, radiation field, the valve sits into its seat and doesn't open.

MR. SCARBROUGH: Yeah, they're going to have to look at that as part of qualification that it's going to set there for a while under that environment; right? So they're going to have to qualify it for the conditions.

CO-CHAIR KIRCHNER: There will always be testing for leakage.

MR. SCARBROUGH: Uh-huh.

CO-CHAIR KIRCHNER: They've got a tech spec on there. And they won't be able to keep a containment vacuum if they're leaking any significant amount.

MR. SCARBROUGH: Yes.

CO-CHAIR KIRCHNER: So, that part of worry less about.

I worry more about the valve functioning and opening --

MR. SCARBROUGH: Right.

CO-CHAIR KIRCHNER: -- on demand.

MR. SCARBROUGH: Yes. That's a very good point. So, we'll make sure that that's part of their qualification because they're going to have to make

sure that they qualify it for those conditions and they don't have an issue with the boron valve in it.

CO-CHAIR KIRCHNER: Right.

MEMBER BALLINGER: I forget, what's the seat and disk material?

MEMBER PETTI: I was going to ask you that same question.

MR. SCARBROUGH: Oh.

MEMBER PETTI: It's all, it's all, the whole thing is all stainless steel. Right? The disk material that will all be stainless steel material.

MR. SCARBROUGH: Maybe NuScale can help us with the exact material.

(Simultaneous conversation.)

CO-CHAIR KIRCHNER: Go ahead, Dan.

MR. LASSINGER: There's a hard facing weld build-up for the seat and the disk. Or, I'm sorry, excuse me. Only for the seat, not for the disk. The disk is the 16 stainless steel.

MEMBER BALLINGER: I didn't get it.

MR. LASSINGER: It's a hard facing.

MEMBER BALLINGER: We couldn't hear him.

CO-CHAIR KIRCHNER: You're soft-spoken. You'll have to say it again, Dan. Speak louder.

MR. LASSINGER: The disk -- excuse me, the

seat of the valve, of the main valve is a welded hard facing. It's not stellite, it's a no-rem, which has no cobalt. But the disk is a 316 stainless steel material.

MR. SCARBROUGH: Okay.

MEMBER BALLINGER: Okay.

CO-CHAIR KIRCHNER: Thank you. Thank you, Dan.

MR. SCARBROUGH: Okay. So, I mentioned the final test report that came, and we looked at that. And there were a number of key observations from the report. And some of, Dan mentioned some of these in his presentation, was the ratio of inflow and outflow for the main valve it's critical to have the main valve open properly. Right? So, that was one of their lessons learned or key observations.

And also the IAB valve threshold and release pressures are impacted by room temperature. This whole flashing effect causes the valve. That's why it's so complicated to come up with the parameters for the IAB valve.

MEMBER BLEY: Tom, can you clarify that first bullet for me? When you say inflow and outflow you're talking about into the ports that connect up with the IAB?

MR. SCARBROUGH: Right, yes. The main, the main valve has an orifice that communicates with the reactor coolant system.

MEMBER BLEY: And that's the input and the output?

MR. SCARBROUGH: Yeah, it is inflow trying to fill up that chamber. And when you want it to open you have to evacuate that main chamber faster than what the reactor coolant system is trying to resupply.

And if the pressure in that main chamber doesn't get down to a certain percentage, I won't say it, just a certain percentage of the reactor coolant system pressure, it won't open. You know, so it has to be relieved off.

Yes, so that's where that -- that's where that comes from.

And so, and then also, as we talked about, the port that feeds from the main valve into the IAB valve, that port is very important. And to get the IAB valve to open and close you have to, you have to remove that pressure out of there so that the IAB valve can perform its function.

MEMBER BLEY: We didn't ask NuScale but from I'm certain the tests, if this, these two were found to be really important did they have to change

anything in the design of those --

MR. SCARBROUGH: Well, yes. Yeah.

MEMBER BLEY: -- to make it work?

MR. SCARBROUGH: Right, right. Remember, on the previous slide I was talking about they had made those changes to the main valve control chamber orifice diameter.

MEMBER BLEY: Okay.

MR. SCARBROUGH: The port inlet diameter, the shim set. All those things --

MEMBER BLEY: That all came out of the tests?

MR. SCARBROUGH: Yes, yes.

And then there's, as Dan was talking about, there's, because there's a IAB valve trying to operate under flashing conditions and such in 2-phase flow, it's hard to come up with a very specific parameter in terms of its opening and closing pressure; right? As you all were talking about, it has to, you have to have enough oomph, right, for it to get it to seal because it has to seal immediately first time. Right.

And so, because it has to hold, all five have to hold, right, because if you lose one, eventually they're all going to open because you're

going to be losing, right, pressure in your reactor coolant system. So, they all have to seal. And so you have to have enough pressure force from the reactor coolant system to overcome that spring to get it to seal tightly.

And so that's why they weren't able to come up with a very sort of simple plus or minus. It's more complicated than that.

All right, so those were the lessons, the report's conclusions.

So, so in the audit report that we wrote, so based on our audit we found that the testing revealed that they needed to modify the valve design and the performance assumptions. The lessons learned need to be incorporated into the QME-1 standard qualification. All these things we've talked about were excellent, good ideas you provided you to us.

And then, also, the in-service testing provision. And let me -- I talked to you about the in-service testing is on the next slide.

So, so what has to happen is the DCA has to be modified to include -- and that's what you see now in the draft Rev 4 that may have already bene issued as Rev 4 -- they have to update the DCA. They have to update design documents, which is why the

design documents that include the internal orifice, shim sizes, things like that, that all has to be updated. We reviewed the previous versions and they all, all those have to be updated.

So, and then the COL holder will need to address all these lessons learned, right, and that's going to actually be happening probably soon, right, because you still want to have this all taken care of. So, they'll probably be developing this, you know, in the relatively near future.

MEMBER SUNSERI: Excuse me, Tom.

MR. SCARBROUGH: Yes?

MEMBER SUNSERI: The COL, the COL responsibilities are to assure that testing and qualification of the valve is appropriate? When you say --

MR. SCARBROUGH: The QME-1 qualification.

MEMBER SUNSERI: In terms of the lessons learned?

MR. SCARBROUGH: Yes. Because they're going to have to satisfy the ITAACs which have QME-1 qualification requirements listed with them. And so, as the process gets closer --

MEMBER SUNSERI: Right.

MR. SCARBROUGH: -- to building the plant,

they're going to have to have all these valves qualified. So, they have to go through the qualification process.

MEMBER SUNSERI: The elements of what needs to be done that needs to be integrated into that program and meets the ITAAC?

MR. SCARBROUGH: Yes.

MEMBER SUNSERI: In other words, this is -- as opposed to the others where NuScale has already done a lot of work to get it modified, this is, this is somewhat new?

MR. SCARBROUGH: Yes. Yes.

MEMBER SUNSERI: In other words, has to be developed and the it has to reflect the lessons learned.

MR. SCARBROUGH: Yes, exactly.

MEMBER SUNSERI: Okay, thank you.

MR. SCARBROUGH: So, anyway, once we address the follow-up items we'll reach a conclusion on whether or not they satisfy 5247(c)(2) and 50.43(e).

MEMBER BLEY: Tom.

MR. SCARBROUGH: Yes?

MEMBER BLEY: This catalog of things they learned from the valve manufacturer who has been doing

this for a long time, and the combination of sitting there for a couple of years without being touched, what's your level of confidence that, you know, after a couple years these things are going to work the way they've been tested?

MR. SCARBROUGH: Well, we're --

MEMBER BLEY: And will that be covered in the qual?

MR. SCARBROUGH: Well, they're going to have to have qualification that shows that over a long period of time they don't see this type degradation. That's going to have to be part of their whole process in terms of thinking about qualification.

And then when they, they start the plant up, right, they're going to run, you know, 2 years and then they'll pull it off, right. And if any of these valves don't still match their threshold and release pressures they're going to have to rethink how they conduct this entire process. They might have to shut down and do something.

MEMBER BLEY: These things are very delicate. And, you know, we saw in the last time when we saw, you showed pictures of the valves --

MR. SCARBROUGH: Yes.

MEMBER BLEY: -- how, how precise

everything --

MR. SCARBROUGH: Yes.

MEMBER BLEY: -- has to be to work right.

MR. SCARBROUGH: Yes. Yes. And I think that's what happened here with their testing when they found, you know, they had to make design changes. Because everything is so fine in terms of the dimensions and stuff.

Yeah. So, but qualification to me will provide sufficient qualification support that they could install the valves. And it should be sufficient to show that over a good period of time they do not have this sort of problem.

Because they could run this test because they could pressurize it and have it sit there, right, and then come back, you know, you could have a good time later, right, and then and run the tests to make sure it still works properly. So, they're going to have to think about that.

MEMBER BLEY: If that stuff is still occurring.

MR. SCARBROUGH: Yes.

MEMBER BLEY: I asked NuScale in previous meetings and today what led them to worry so much about inadvertent operation of the ECCS valves that

they'd go to this complex arrangement to avoid the problems. From a regulator's point of view can you answer that question for me?

MR. SCARBROUGH: Well, you know --

MEMBER BLEY: Did you drive them to this?

MR. SCARBROUGH: Well, the regulations are very clear, right, in terms of --

MEMBER BLEY: Well, let me, let me tell you where I'm coming from.

MR. SCARBROUGH: Okay, sure.

MEMBER BLEY: Back in the '60s and '70s we had reactor trip breakers. And our regulators at the time decided we want that to be passive. And they made all of the operating plants disconnect the shunt trips on those breakers from PRAs that were then back then predicted that we're going to have some SCRAM failures because of that because the shunt trip is much more effective than just the dropout of power.

And sure enough, we did. And then we changed our regulatory approach on these things.

I'm wondering if we're driving ourselves into a spot here that we should have -- I don't know where the biggest risk is. And I think that likelihood of inadvertent operations is probably pretty darn small. And I don't know if we're

challenging our ECCS system unnecessarily.

MR. SCARBROUGH: Well, I agree with that thought. I mean, we've had issues like that, you know, in my other life, where we'd operate valves with some overloads; right? We always told everybody to bypass it from overloads.

MEMBER BLEY: Yes.

MR. SCARBROUGH: And valves died trying. Right? And we found that that wasn't such a good idea.

MEMBER BLEY: Not such a good idea.

MR. SCARBROUGH: Yeah. So, so and here because of the way the current regulations are, they are required by the regulations to assume that any non-safety related power supply is not available.

Now, they could have come in and asked for --

MEMBER BLEY: For an exception.

MR. SCARBROUGH: -- some exception.

MEMBER BLEY: And started elsewhere.

MR. SCARBROUGH: Right. And so they could have come in and asked for something like that.

And, actually, in looking at their proof of concept reports and such, they had various designs that they were thinking about. Right? This is the

one they came up with. Right? But they could have come in and asked for some type of, you know, exemption, or exception, or something.

And, yeah, because I, I agree. You know, there is some risk involved here because you do want this valve eventually to reopen. And you don't want this, this main, this IAB to close and then something happens that the main valve wouldn't open.

So, yeah, I think there's --

MEMBER BLEY: They're hanging on that.

MR. SCARBROUGH: I agree with your concern.

MEMBER MARCH-LEUBA: And I would like to reinforce my colleague's opinion.

MR. SCARBROUGH: Uh-huh.

MEMBER MARCH-LEUBA: We are supposed to be moving into risk-informed regulations. And we have here a regulation that in my opinion, and this isn't going away, that this is against regs, I mean it's bad. It wasn't for the IAB and certain parts --

MR. SCARBROUGH: Yes.

MEMBER MARCH-LEUBA: -- and it can only get in the way.

MR. SCARBROUGH: Yes. And there were other ways that you could have looked at this from,

you know, a perspective, but this was the way they came up with because that's what we were reviewing.

MEMBER MARCH-LEUBA: I'm not against risk-informed regulation. I think that we should use it properly.

MR. SCARBROUGH: I agree.

MEMBER BALLINGER: I think I heard that the reasons for this is inadvertent ECCS valve operation with respect to CHF. Is that what I heard right?

MEMBER BLEY: Assuming electric power goes away because of not one -- even though it's been certified as highly reliable by the staff, which they've defined.

MEMBER MARCH-LEUBA: I mean, multi-power.

MEMBER BLEY: My prediction is sometime in the next 5 years we're going to revisit this.

MEMBER MARCH-LEUBA: Look at it, the first time we tested the valve by the manufacturer which has been building this valve for 50 years, it didn't work.

MEMBER BLEY: It didn't work the way they planned.

MEMBER MARCH-LEUBA: No. It didn't, it didn't open. It didn't open period because the holes were the wrong size. So, the testing in the facility

didn't work.

Well, they make it a little bigger and then, woo, now it works, woo.

MR. SCARBROUGH: That's right.

CO-CHAIR KIRCHNER: To wrap up some of what you were doing, Tom, the qualification program, that becomes an ITAAC, and that's a COLA responsibility.

MR. SCARBROUGH: Yes, yes. There is an ITAAC which says that all the valves, and these are some of them, have to be, you know, qualified, you know, demonstrated. And in the Tier 2 section of the DCA it talks about QME-1 is a process that's going to take place to meet that. So, that's what they're going to have to do.

And so they're already thinking about how they're going to do that.

So, in terms of the audit, what solid guidance we have is, one, we have to have the IST program updated in DCA because, you know, what we were talking about was these valves, these IABs sit there in the open position all the time; right? And when they go to shut the plant down they're planning to lower pressure; right. So, the IAB valve never operates, right, when they're shutting down. Right?

And so, because, you know, why, why have a high pressure and then force the valve to perform its function to stop a high pressure situation. So they're going to lower pressure down. And the IAB valve will just sit there; right? So, it will never have been stroked.

Well, the owned code, right, ASME owned code says that every valve has to be stroked quarterly, unless you can't do it quarterly, then you do it every refueling outage or full shutdown.

And so, which means that they would, every time they shut down, they have to take all five valves off on the bench and test them, all five. So, what they did was they came in with an alternative request under 50.55(a)(z) to say, okay, we'll do all five for the first module for pre-service testing. We'll show they all work properly.

Then, for the first time we shut down for a refueling we'll pull all five out and we'll show that they all work in the right amount. If they all do, then the next time would be a sampling of one RVV and one RRV. Right?

But, if they ever have any failures, they're going to have to go back and rethink that whole process; right? If one of them doesn't operate

within that range that they have, they have to go back and rethink that entire process. Right?

CO-CHAIR KIRCHNER: Is that on a module basis or 12 module basis?

MR. SCARBROUGH: Well, the first one, for the first module would be --

CO-CHAIR KIRCHNER: All five valves.

MR. SCARBROUGH: All five.

CO-CHAIR KIRCHNER: One module.

MR. SCARBROUGH: Right. And then, and then after that it would be a sample on whatever module it is. Right? But for pre-service they all would be done; right? But then for each module for the refueling they would do one.

But then because they're under mandatory Appendix 4 of the code, it has a performance assessment test schedule where they can show once they have -- because they're going to have 12 modules, there's quite a bit of data, they might be able, hopefully, be able to show that, yeah, there's been no change on this over time, right, and we'll be able to kind of lengthen these out. Right?

But for right now the request is that they would do this sampling. And once they do it the first time it's for all five. So, that's the request.

MEMBER SUNSERI: You say that as if it's something new. Isn't that the fundamental a risk-informed IFP program works?

MR. SCARBROUGH: Well, you could, you could request a risk-informed IFP program. The only code has risk-informed aspects to it. But when it comes to the quarterly stroke time testing, it's just a pure requirement. It's deterministic. And that's why they're going to come in and ask -- they come in and ask for this alternative to include this PRA risk--informed aspect to it. That's, that's the difference.

Okay. So, all right, so we'll do that. That's item one.

Then we'll have item two in which they have to update the DCA for all the performance requirements. And then they have to update all these design documents for all the modifications they made, or if there's just a shim placed. Okay?

All right. So, so where are we right now? Okay. So, in October last year they submitted a letter with the status of what they are doing. And one of it is a discussion of the updated Rev 4 which has the new working for the IAB valve parameters.

Then in November they sent a letter which

included the proposed alternative for the IFP program. And we're going to be reviewing that as part of the next phase of the SBR.

And then once they notify us that all of the other items have been completed we'll, we'll follow up with that and decide how much we need to look at that. It may be very simple, just to make sure that they updated all these documents.

So, then when we get to that point, during the next phase we'll reach the conclusion regarding whether or not they satisfy 5247(c)(2) and 50.43(e). And then, and then we'll be done.

And then what we're doing, we'll be sort of monitoring what the plan for, you know, the COL stage at that point, where we're going to go from there.

But it's been a very interesting review. You know, I've enjoyed it, you know. I think, I think we demonstrated that the original proof of concept testing was not sufficient because of all the design changes they had to make based on the testing they did under actual conditions.

So, I think it was the right decision that we made that they needed to do extra testing. And now, now they're sort of close to where they have

actually real design that meets 50.43(e). And once they start conducting their qualification testing they'll have to show that it operates under the more extensive conditions of a qualification project.

CO-CHAIR KIRCHNER: So just so I understand or we, not me, but the committee and for the record, when you say the next phase, that's we are scheduled to complete our review by the June time frame. You're talking about when you write up the final SER --

MR. SCARBROUGH: I had --

CO-CHAIR KIRCHNER: -- in September or summer or?

MR. SCARBROUGH: Yes. Yes, sir.

CO-CHAIR KIRCHNER: Is that --

MR. SCARBROUGH: Yeah.

CO-CHAIR KIRCHNER: -- what you mean by the next phase?

MR. SCARBROUGH: Yes. Yeah, we're sort of in phase 5 now because we're briefing you.

CO-CHAIR KIRCHNER: Right. In phase 5.

MR. SCARBROUGH: And phase 6 is when I, I take all these items and I close it all out in the SER.

MEMBER BLEY: But the qualification

testing won't have been done yet?

MR. SCARBROUGH: No. No. That's going to be the process for the COL, you know, process as we go forward, yes.

MEMBER BLEY: Tom, I want to thank you for a very good presentation. And I just starting your audit report. It's very interesting.

MR. SCARBROUGH: Oh, thank you.

MEMBER BLEY: I wish we had seen that a little earlier. But, yes.

MR. SCARBROUGH: Well, great. You know, if you have any questions or follow-up, I'll be happy to come back and talk to you, you know, as a group or individually, however you'd like to do it.

CO-CHAIR KIRCHNER: Members, further questions?

Dennis, any more?

I'll echo Dennis' comment. Thank you for your thorough presentation. And I like the way you keyed it up as a system because that's really what we're dealing with here.

MR. SCARBROUGH: Right.

CO-CHAIR KIRCHNER: And a very interest one --

MR. SCARBROUGH: Yes.

CO-CHAIR KIRCHNER: -- at that.

All right, thank you.

MR. SCARBROUGH: Thank you.

CO-CHAIR KIRCHNER: We are scheduled now to take a break. Mike, we're a little bit behind schedule but not bad.

Can I ask that you come back at 10 of 4:00. That's a quick 10-minute break. Thank you.

So, we are in recess.

(Whereupon, the above-entitled matter went off the record at 3:38 p.m. and resumed at 3:50 p.m.)

CO-CHAIR KIRCHNER: We'll reconvene.

Marty, am I going to turn to you again?

MR. BRYAN: Yes, sir.

CO-CHAIR KIRCHNER: Okay. We're getting familiar.

MR. BRYAN: Yes.

CO-CHAIR KIRCHNER: Please.

MR. BRYAN: This is the last one. Marty Brown. This is Chapter 8. And then, Jeff Ehlers is going to take us through it. Ours is pretty short, pretty straightforward. We'll be happy to entertain any questions, but this should get us back on schedule.

So, with that, I'll turn it over to Jeff.

MR. EHLERS: Yes, we only have two slides here to present, but the majority of the time is an opportunity to write, ask any questions. We're not covering any new material here. This has been discussed before at a previous meeting.

Just a reminder, I'm the Electrical Engineering Supervisor at NuScale. So, I own the DC system.

This is a high-level refresher. The DC system is not a safety-related system. The design of it was based on the NRC-approved Topical Report. And so, we went through the conditions of applicability for the system design to show that it could meet the needs that we needed to meet. And this was also further discussed in Chapter 1 of the Safety Evaluation Report, what didn't meet the Topical Report.

In Chapter 8 of the DCA, to make it convenient, we did provide a table, sort of a Rosetta Stone. It showed where in the DCA, so someone didn't have to go through and find all the pieces scattered throughout the chapter of where we addressed the Topical Report items in the DCA. That's in Table 8.3-9 in the DCA and, also, Table 8.3-10 shows the cross-reference for the EDSS Augmented Provisions.

So, that's basically the list of the brunt of the presentation for you, but I am happy to take any questions you have at this time for me.

MEMBER REMPE: So, I have a question.

MR. EHLERS: Okay.

MEMBER REMPE: I was looking through your Topical Report, and I'm paraphrasing based on my memory.

MR. EHLERS: Okay.

MEMBER REMPE: So, if I say it wrong, correct me. But, basically, you talk about, yes, we aren't going to have -- even if we don't have power for instrumentation, it's okay because we don't have to have any operator actions. But we have concerns about defense-in-depth, and that's why we think it's good to have those measurements. And I guess I'm not sure I agree with that philosophy, especially when we talk about the things we've talked about today. I think for post-accident monitoring, it would be good to know whether valves are open or not.

But if I look at Chapter 20, and Table 20.1-1, it talks about that you need to have these parameters to assure the cooling function is established, and then, it's good to have it for post-accident monitoring. Could you elaborate about, if

you were to lose power, why suddenly is there this kind of disconnect between Chapter 20 that says you've got to have assurance that the cooling function is initially established if you don't need power? I mean, there seems to be a disconnect. Do you understand my concern?

MR. EHLERS: Yes, I think I understand the question you're asking. So, I think some of the confusion comes in some of the IEEE guidance associated with PAM variables and what the requirements are for safety-related power for them. So, there's really kind of two parts to it, which is, one, this is information that operators need to assess the plant and, two, this information will be used for them to go take operator action based on some procedure, or whatever.

Now, for Type B and C variables, for the NuScale design, we say, yes, we have these variables so the operators can assess the status of the plant, but the second condition of that is not met because there are no operator actions to take as a result of that. So, we didn't meet both conditions to need the safety-related power.

So, what we did say, we still think it's good for the operators to understand what's going on.

So, we provided an augmented design to help prevent the loss of those indications. That is really the EDSS's function, is to make sure that they have power for the PAM instrumentation.

MEMBER REMPE: I get that that part is consistent.

MR. EHLERS: Yes.

MEMBER REMPE: But what seemed a little inconsistent is, if I look at Chapter 20, in that table they talk about the post-accident monitoring, but, then, they have a column that says, yes, you need to have the parameters to assure that the cooling function is established initially. And will you have that capability if you lose power?

MR. EHLERS: To ensure?

MEMBER REMPE: That the cooling function is established initially. It seems like you need it initially, and then, okay, afterwards there's no operator actions. But it seems like the initial time, and then, what do I do if I don't have the power?

MR. EHLERS: So, I'm not aware of the Chapter 20 statement you're talking about. But what I do know is that there are no operator actions. So, this might be the use of the words to say it's necessary for them to know the status of the plant,

but there's still no operator actions associated with that information.

MEMBER REMPE: That's true. That's what I'm getting.

MR. EHLERS: Yes.

MEMBER REMPE: But why do they need to know then? I guess I'm going further. I mean, it kind of sounds like, why bother having that in Chapter 20 saying that you want to establish initially? If there's no operator actions, why put that in there? It's a little inconsistent. I think you would want the operator to know, and that's been a problem with a lot of accidents that have occurred, is the operators don't have a way to monitor it. I mean, all throughout this Topical Report and other places in the DCA they've emphasized no operator actions are needed. But, yet, why is it in Chapter 20, in this particular table, they have that the operator needs to establish that the initial cooling or determine that the initial cooling has been established, function has been established?

MR. EHLERS: Like I said, we could probably follow up to figure out why we put that in Chapter 20 for you, but --

MEMBER REMPE: I think, yes, it seems a

little distant.

MR. EHLERS: I think the answer is the same, but I think we'll have to do a follow up as to exactly why it was put in there.

MR. BRYAN: Yes, it's a good question. I think we understand the question. I've got several people trying to give me an answer, but I would prefer to try to sort it out --

(Laughter.)

MEMBER REMPE: Well, the reason why is that I've been involved in looking at Chapter 20 for the meetings that are going to occur later this week. But I saw that after I had been preparing for this meeting. I was a little curious because it seemed the one place that you've guys weren't quite consistent.

MEMBER BLEY: Yes, I agree it's not a consistent thing. But, back when we reviewed the Topical on electric power, we really thrashed about this quite a bit and kind of agreed that, given the highly reliable electric power they put it instead of 1E, we would have what we needed for PAM in almost all conditions. And the staff had argued that same thing. So, we bought into it at that point.

MEMBER REMPE: I understand.

MEMBER BLEY: And now, there's an

inconsistency problem that affects Chapter 20.

MEMBER REMPE: There's an inconsistency if we're going to be talking Chapter 20. It's just I was maybe amused by it. And I thought I'd bring it up today because this is such a short presentation.

MEMBER BROWN: If you go back and read the -- that's the EDSS I think you're talking. That's the highly reliable when you're referring to --

CO-CHAIR KIRCHNER: Yes, that's correct.

MEMBER BROWN: If you go back and read under the conditions and limitations and applicability, they go through a list where the EDSS, you could call it 1E if you wanted to, but they don't bless it that way. I mean, you have to meet all the typical requirements. But do they get an exemption that says they don't need it because you don't have to do anything? So, it's still they're building pretty reliable stuff. And I mean, we really did beat this horse to death, if I remember correctly.

CO-CHAIR KIRCHNER: On that topic, in preparation for this meeting, we had a teleconference with NuScale and the staff on electrical systems on Thursday, January 16th. And one thing we learned, I think it was following some questions from Dennis, is that there is an internal engineering report on the

reliability of your system. That was something that we hadn't had access to.

Do you want to say a few words about that, Jeff? I thought it would have shown up on your slides in some manner.

MR. EHLERS: A report concerning the liability of our design --

CO-CHAIR KIRCHNER: Reliability.

MR. EHLERS: Reliability?

CO-CHAIR KIRCHNER: Yes, estimating, demonstrating the reliability, an engineering analysis to demonstrate that it would, indeed, achieve the highly reliable objectives that are set out.

MR. BRYAN: Yes, I believe you're referring to the one that we did and the staff audited for us to do that --

CO-CHAIR KIRCHNER: Yes. Yes. So, I'd like to just put that on the record. So, could you say, Marty, for the record, what was done and what the staff audited?

MR. BRYAN: Well, Sarah, do you want to address that? Or someone?

MS. BRISTOL: Yes, I'm here. This is Sarah Bristol.

And I think we did an engineering study

that looked at a typical warning system at an existing plant, and then, compared that to the NuScale EDSS highly DC power system and showed that the systems were equivalent in reliability. And then, the staff had taken a look at that in one of their audits.

CO-CHAIR KIRCHNER: And we'll ask the staff the same question. Because it was my feeling, at least, when I read Chapter 1 of the SER, that it was more just the statement, repeating a statement that it is highly reliable, almost ipso facto. But, then, we learned that there was actually some engineering analysis that would support that claim. But that doesn't come through in either the FSAR and/or the FSER, or at least this reader did not get that impression that it had been demonstrated.

MEMBER DIMITRIJEVIC: And my question is a little broader than this highly reliable system. I just have a general question. I brought it up, also, in these discussions. I completely have no issue that this is like a normal 1E Class system. There's a PRA. However, I do have an issue if this is considered not risk-important non-safety system, because those systems can be completely forgotten in any of these Tech Specs, user testing, and things like that. And usually, those systems, in several NEI documents are

considered to have a reliability four times more. I know that you are not applying 5069 and it's a risk-informed application, but I think that there should be some measures which make sure that this system is performing its function. It means that you cannot take this generator and keep it all year out of service, I mean and things like this.

Because, even this system, it did not show up. That shows you the PRA is not -- it should never be just a number. That system shows it is not safety-significant because it doesn't satisfy Dr. Vasily (phonetic) and weak risk measures. But, at the same time, looking at the action to start charging systems shows significant because these guys are higher affiliated, so it satisfies Dr. Vasily (phonetic).

So, with regard to the action to start charging these risk-significant, then charging has to be risk-significant, because it actually cannot be performed without charging, right? Any of the charging is risk-significant, and the life system also has to be important, because if you don't have a life system, the operator cannot start charging.

So, we cannot play these number games that we claim that something is not risk-significant which is risk-significant. That's why I was sort of more

interested in how this risk significance of the systems was calculated. Then, I have a billion questions in the common cause, but I decide this is not a question about the importance measure. It's a question, how do you determine that it's a significant system? And if that system supports important operator action, then it has to be significant. So, therefore, I think there should be some measures to make sure that that system is available and operable. That's my point.

MEMBER BLEY: I don't know if NuScale is going to tell us anymore. And the staff used a slightly different approach on this. When we wrote our letter, we actually said we didn't think it was a useful exercise. And the real reason we said that is there is no typical 1E electric power system.

Back in the eighties, NRC had a program -- I think it was called IREP -- and they were trying to do a generic PRA for PWRs and for BWRs. And the idea was things that were really plant-specific you would build models for those systems, but things that were in all of them, we would have a single model that would work for everybody. And after trying that, we found out electric power systems and ultimate heat sink service water systems -- at that time, there were

50 different plants; there were 50 different systems. It wasn't a single system. So, I don't think it's very useful.

But I think what the staff did might be helpful because they took a more not reliability analysis, but kind of bounding approach to it, although they didn't tell us a whole lot about it. Maybe they will today.

CO-CHAIR KIRCHNER: You thought you were going to get off easily with two slides, didn't you?

(Laughter.)

Members, any other questions or comments?

MEMBER REMPE: Just to follow up, Table 20.1-1, it's called "Core Cooling Parameters," and it lists the functions such as RCS inventory control, reactivity control. And for each of those functions, it lists between two and four parameters that are required to assure that the function is established. And there is a footnote, "By design, once these functions are established, they're maintained indefinitely." And again, that's why I was puzzled, because if you go to the third column, it says, "Parameters for assuring the function is maintained," and often they say, "None." So, it's just it seems like it's important to know initially, and I don't

quite understand that table.

MR. BRYAN: Yes, someone's getting me the same table. So, we're trying to run that to ground, but I understand the question.

MEMBER REMPE: Okay. I'd appreciate an answer --

MR. BRYAN: Yes.

MEMBER REMPE: -- because I have to do some Chapter 20 work this week. Thank you.

MR. BRYAN: And the answer to your other question was, apparently, 13 on the data points.

MEMBER REMPE: Oh, 13 data points?

MR. BRYAN: Yes.

MEMBER REMPE: Thank you.

(Laughter.)

MR. BRYAN: You're welcome. And centrifugal is the pump.

CO-CHAIR KIRCHNER: Of course. Yes, yes, of course. Yes.

Okay. With that, then, if there are no more questions from the members, thank you, Marty and Jeff.

And we'll now hear from the staff, please.

MEMBER BLEY: Mr. Chairman? Walt?

CO-CHAIR KIRCHNER: Yes?

MEMBER BLEY: While they're coming up, on that phone call, I had said there was a new Topical Report and we didn't know anything about it. Well, it turns out the new Topical Report is pretty much the old Topical Report with a Section A, which is a transmittal letter; a Section B, which is the SER; C, which is the Topical and D, which is a summary, or something like that. So, there wasn't anything really --

CO-CHAIR KIRCHNER: There's no new information there?

MEMBER BLEY: And the things they pointed out -- the staff will talk about this -- were really the conditions in the staff SER which they addressed.

CO-CHAIR KIRCHNER: Yes.

MEMBER BLEY: And instead of calling it the SER, they called it the Topical, which left me very confused because I hadn't seen it yet.

CO-CHAIR KIRCHNER: Okay. Omid, should I turn to you to make the introduction?

MR. TABATABAI: Sure.

CO-CHAIR KIRCHNER: Please.

MR. TABATABAI: Thank you. Good afternoon, Members.

We're here today to talk about the way

that NuScale basically addressed all the conditions and limitations of the Safety Evaluation Report for Class 1E or Electrical Topical Report in the DCA. That's the purpose of our briefing today.

Just as a refresher, to talk a little bit about the background and timeline. The NRC's staff, we issued the Safety Evaluation for the Topical Report in June of 2017. We approved the Topical Report with six conditions and limitations. We briefed the ACRS Subcommittee and full Committee in July of 2017. We received your letter and you recommended us to approve the Topical Report, but make it only specific to NuScale design, not generically to any passive design, to all passive designs.

We incorporated your comment, responded to your additional recommendations, and we docketed the approved version of the Topical Report in February of 2018. Because NuScale has incorporated this Topical Report by reference in its DCA, we requested NuScale, in January of 2018, via an RAI, to basically describe how the DCA meets all of the conditions and limitations of the Topical Report-Safety Evaluation.

As a result of your recommendation in your letter, since we made it specific to NuScale design, we originally had six conditions, but the revised SE

we dropped one of the conditions that would make it applicable to generic designs, passive designs. So, we have only five conditions in the SER. In March of 2018, NuScale responded to the RAI and provided a markup of the DCA, how they have revised the DCA to incorporate or to address those conditions.

So, in the following slides, we're going to go through those conditions and describe how we made our finding in terms of applicability of those conditions to the NuScale design.

With that, I'm going to turn the microphone to Sheila.

MS. RAY: Good afternoon. My name is Sheila Ray. I'm in NRR, the Electrical Engineering, New Reactor, and License Renewal Branch.

The highly reliable DC system is non-Class 1E and has augmented design and quality assurance provisions. There are two DC subsystems. The EDSSC serves common loads and the EDSSMS serves loads specific to each module.

Based on staff's review of the applicability of the Topical Report to the DCA, staff concluded that the onsite and offsite power systems are non-Class 1E. Staff approved NuScale's exemption request from the requirements of GDC-17 on electrical

power systems.

Condition 4.1 of the Topical Report is to address guidance in Reg Guide 1.155, Appendix A, on the quality assurance of non-safety systems and equipment. The DCA states that an augmented QA program is applied to the EDSS and meets the provisions in Reg Guide 1.155. Since the DCA addresses these quality assurance provisions, staff finds the Applicant has met Condition 4.1 and documented the findings in Chapter 8 of the Safety Evaluation.

Next slide.

Condition 4.2 is to confirm the VRLA batteries and structures are Seismic Category 1, to ensure the batteries will perform as intended.

Next slide.

The DCA states that the EDSS is classified as Seismic Category 1 and that the design accommodates the effects of environmental conditions. Since the EDSS batteries and structures are Seismic Category 1, and the EDSS has augmented design requirements for qualification, the staff finds the Applicant has met Condition 4.2.

MEMBER BLEY: On your previous slide, Sheila, you talked about the qualification testing

plan. You didn't mention that now, and I didn't see that directly in the SER. Is that called out in the augmented requirements table?

MS. RAY: Yes.

MEMBER BLEY: Okay.

MS. RAY: Yes, it's augmented qualification provisions.

MEMBER BLEY: Okay.

MS. RAY: It's not a testing plan for --

MEMBER BLEY: Because it's just that it was said to be seismic requalified, there will be testing done?

MS. RAY: It's an augmented qualification.

MEMBER BLEY: Okay. That's not complete, though?

MS. RAY: No.

CO-CHAIR KIRCHNER: And the COLA action to demonstrate it meets the Seismic Category 1 --

MS. RAY: It's in the design. It's in the design that it's Seismic Category 1. So, I don't believe there's a COLA action needed because it's already in the design.

CO-CHAIR KIRCHNER: Okay. I misunderstood your statement.

MEMBER BLEY: And it's in the license

because of this condition.

CO-CHAIR KIRCHNER: Yes. Okay.

MEMBER BLEY: Yes.

MR. SCHMIDT: Okay. I'm Jeff Schmidt with Reactor Systems. I'm going to cover the last three conditions.

So, Condition 4.3 was basically operator actions necessary to mitigate a Chapter 15 event. And the Chapter 15 review concluded that no operator actions are necessary to mitigate a design basis event out to 72 hours.

Condition 4.4 was evaluate the frequency of an AOA and actuation of the CCS system is realistically expected to occur, and show that such a combination of events is not expected to occur in the life of the model.

MEMBER BROWN: Can I interrupt you just for a second?

MR. SCHMIDT: Sure.

MEMBER BROWN: May I, please?

MR. SCHMIDT: Sure.

MEMBER BROWN: When we say 72 hours, it effectively means we have to have power back, some power back in 72 hours. That's the way I read this in both the TR and the SER, when I read it.

MR. SCHMIDT: Not necessarily. I mean, it's obviously --

MEMBER BROWN: Well, it says they don't have to, operators don't have to take any actions for 72 hours. That implies you have to have some capability of taking actions, and I don't think that means blacksmith hammers and anvils.

MR. SCHMIDT: As far as like what happens after 72 hours, is that what you're --

MEMBER BROWN: No, it's after 72 hours. The statement is they don't have to take action for 72 hours. Therefore, that implies that they have some means, and you hardly have any means if you don't have electric power back.

MR. SCHMIDT: Right, right.

MEMBER BROWN: Okay. That's the way I was reading this stuff.

MR. SCHMIDT: Right, right.

MEMBER BROWN: Through all the earlier meetings as well as this.

MR. SCHMIDT: Right.

MEMBER BROWN: And I just never thought to clarify to make sure I was correct.

MEMBER REMPE: Well, in other chapters, I thought it was mainly that the staff just isn't going

to review beyond that point.

MEMBER BROWN: No, that's not what it says. It says they don't have to have power for 72 hours.

MR. SCHMIDT: They don't, from a Chapter 15 standpoint, they do not have to have power up to 72 hours.

MEMBER BROWN: Up to 72 hours.

MEMBER REMPE: But his second corollary, that they have to after 72?

MEMBER BROWN: No, they haven't --

MEMBER BLEY: This kind of thing has been in regulation a long time, and there's an embedded underlying assumption that in 72 hours you can get help.

MEMBER BROWN: Yes, that's right.

MR. SCHMIDT: I think that's more in Chapter 20.

MEMBER BROWN: Even if you bring in batteries and generators on the back of a truck, that's power. Okay. It doesn't matter. So, this whole idea that electric power is not needed, at some point it is. And if you walk away that it isn't, that's not the case. I was just trying to make sure everybody understood that clearly.

MEMBER BLEY: Yes, it may not be the case.

MEMBER BROWN: Well, yes. It's hard to believe if you had no idea what's going on in the plant.

MR. SCHMIDT: All right. Next slide.

MEMBER BROWN: Thank you.

MR. SCHMIDT: Sure.

So, 4.4 is what I consider like the escalation clause. You don't want an AOO to turn into something worse, kind of a defense-in-depth philosophy here.

CO-CHAIR KIRCHNER: So, Jeff, this is where I have a little problem.

MR. SCHMIDT: Okay.

CO-CHAIR KIRCHNER: Chapter 6 defines -- and it was the Applicant, and your SER just reiterates what the Applicant states -- that inadvertent opening of an ECCS valve, and the one that they're worried about is the recirculation valve --

MR. SCHMIDT: Right.

CO-CHAIR KIRCHNER: -- is an AOO. Now the definition of an AOO in 10 CFR 50 is one or more times in the life of the plant.

MR. SCHMIDT: That's right.

CO-CHAIR KIRCHNER: So, this is, at least

from my reading, it seems to be a contradiction in Chapter 1 based on what is stated in Chapter 6.

MR. SCHMIDT: So, the Applicant chose to evaluate it as an AOO. I think that's the distinction. For instance, you were just talking about the block valves, right?

CO-CHAIR KIRCHNER: Yes.

MR. SCHMIDT: And it was pointed out that like spurious signal or operator error could lead to -- and that is classified, typically, in Chapter 15 as an AOO event. And that's why they put it in the block valves. But the expectation is, with the combination of the block valves, that would prevent that from occurring in the life of the plant. That's why the word "realistic" is in there. So, the Applicant chose to analyze to AOO criteria, but it's not expected to occur on an AOO frequency.

MEMBER BLEY: Jeff, from a regulator's point of view, if it happens once in the life of a plant, they must treat it as an AOO, if it's expected to happen once a month.

MR. SCHMIDT: Yes. If it's classified --

MEMBER BLEY: The fact that it isn't there, they can still analyze it as --

MR. SCHMIDT: Yes, they can analyze to a

more conservative criteria. And I think I talk about that a little bit on the next slide.

MEMBER BLEY: Yes.

MR. SCHMIDT: But, yes, you can go down in classification, but not typically up. So, does that answer your question?

MEMBER BLEY: I just felt that the SER wasn't internally consistent or clear on that. Maybe you might look at the wording again.

MR. SCHMIDT: Okay.

MEMBER BLEY: It just seemed to me an inherent contradiction with this particular limiting condition No. 4.

MR. SCHMIDT: So, the keyword in this condition is "realistic". It's more like a PRA and not necessarily like an assumption that you would make in Chapter 15. So, this is more of, like I said, PRA-based. But they chose to analyze all the events in Chapter 15 as AOOs, even though I think we could argue that LOCA is probably not an AOO, right? But they're analyzing AOO criteria.

CO-CHAIR KIRCHNER: Personally, would it define inadvertent operation of ECCS valve as a LOCA? I know there are reasons they don't want to go there, because it seems they want to keep a distance from a

large-break LOCA --

MR. SCHMIDT: That's right, and this is a pipe break.

CO-CHAIR KIRCHNER: -- in Chapter 15.

MR. SCHMIDT: Right.

CO-CHAIR KIRCHNER: But, by calling is an AOO, then you open the door to one or more times in the life of the plant and inconsistency. And I know I did read the word "realistic," but I would just ask you, then, to go back and look at your SERs and --

MR. SCHMIDT: And this was in Section 6?

CO-CHAIR KIRCHNER: Offline. I can get the SER.

MR. SCHMIDT: Okay.

CO-CHAIR KIRCHNER: And it's Chapter 6.

MR. SCHMIDT: Chapter 6.

CO-CHAIR KIRCHNER: And I'll get you the page number.

MR. SCHMIDT: Okay.

So, as we kind of already alluded to, these are the means of uncontrolled reactor coolant pressure. That's coolant losses, LOCA, inadvertent operation of an ECCS valve, which we'll basically have an operator error or a valve failure, or even a spurious signal, I probably should have added there.

And then, AOO coincident with the loss of the non-safety-related DC powers, we talked about it's not safety-related. So, on an AOO event, we assume it's not there in Chapter 15.

Again, LOCA is classified as postulated accident. Operator error, opening the ECCS valve is basically prevented by the block valve. Mechanical failure of the safety-related ECCS valve is not expected to occur in the lifetime of the plant.

AOA coincident with the loss of non-safety-related DC power, again, the loss of the highly reliable non-safety-related DC power is not expected to occur in the lifetime of the plant, based on the attributes of the system.

For a longer-term loss of AC power, DC power is returned long enough that the subsequent opening of the ECCS valves does not challenge containment integrity.

Condition 4.5, this is demonstration of shutdown. They have chosen to use the alternate and, alternatively, an applicant referencing this Topical Report may provide justification for NRC review for a less restrictive condition. As we've discussed many times, shutdown may not be maintained, assuming the design basis conditions at or near end-of-cycle

conditions. NuScale provided a less restrictive condition, demonstrating that the SAFDLs are not exceeded for a potential return to power with a stuck rod. This is consistent with SECY-18-0099. Applicant's Chapter 15 analysis demonstrated that the SAFDLs are met.

MEMBER BLEY: Is there an SRM associated with that SECY?

MR. SCHMIDT: There is.

MEMBER BLEY: There is?

MR. SCHMIDT: There is. That's correct.

CO-CHAIR KIRCHNER: One more time, Jeff, I would look at the SER write up in Chapter 1 there for this particular condition. You have the word "stable" in there. A return to criticality and power is not, personally, it's not stable. Whether it challenges the SAFDLs or the reactor pressure coolant boundary is a different matter, but I would hardly describe that as a stable condition. Your definition, you, along -- I don't want to reopen the GDC-27 exemption, but I guess I am. But it's really like a hot shutdown. It's not --

MR. SCHMIDT: I can't remember the exact number, but in 15.0 we reach a stable condition, and return to power is considered a safe stable condition.

You will see that in Chapter 15.

CO-CHAIR KIRCHNER: Yes, I saw that. Yes. Uncontrolled return to power is not a stable condition.

MR. SCHMIDT: Well, I mean, when I call the return to power, I call it -- it is the natural sequence of events that could occur that will reach an equilibrium condition. So, we're considering that a safe stable condition.

CO-CHAIR KIRCHNER: I would agree with you. I'm quibbling on words, Jeff. I would agree that it doesn't challenge the SAFDLs, or as far as we know it doesn't, and it doesn't challenge the reactor coolant pressure boundary. So, it meets the conditions of a safe shutdown, as you've agreed with the Applicant. But it's not a stable condition. I am just quibbling with your words.

MEMBER BLEY: Perhaps "equilibrium," as he just used, would be accurate.

CO-CHAIR KIRCHNER: Okay.

MR. TABATABAI: Well, this concludes our presentation. If there are other questions for us, we'll be happy to --

CO-CHAIR KIRCHNER: Could you, Omid, and your team, talk to the audit of their reliability

analysis that came up in the teleconference last month? I mean, basically, what you found or just, again, for the record. So, you audited an internal engineering report that basically went through the EDSS system to demonstrate that the design was, quote-unquote, "highly reliable". Is that correct?

MR. TABATABAI: We did audit some of NuScale's documentation, internal documents and analysis, but I'm not sure whether we did it in the context of finding the system highly reliable, you know, putting that label on it or not. But I think our audit was as part of going through the conditions of applicability that was part of the Topical Report, to go through those.

But I guess I'm going to need help from my colleagues who formed that --

CO-CHAIR KIRCHNER: Because, for me as a reader, I felt like it was in your SER rather circular. Ipso facto, this is a condition of applicability, and, yes, they meet the condition of applicability, but the evidence, compared to other sections of the SER, is not -- you know, this engineering report, for example, we were provided --

MEMBER BLEY: You had a paragraph on their engineering report, reliability of their system. And

you didn't speak in the language you used a minute ago, but you did make some conclusions based on your looking. And it was kind of terse in the SER, and maybe it can be addressed up here.

MR. NAKANISHI: This is Tony Nakanishi with the PRA Branch.

And what we could say is that we looked at the reliability study in the context of the whole PRA. That was the primary purpose of that review. But I guess, in terms of the relevance to this, the application of the Topical Report, I would say that we really focused more on the design attributes, the qualitative aspects of how we find adequate reliability for the system. So, you know, how it's protected, independence, redundancy of the system, programs to monitor, things of that nature. So, in terms of the quantitative reliability aspect, we did look at it, but we really didn't focus on that particular -- you know, for the finding that we had to make.

MEMBER BLEY: After our phone call, during our phone call, we asked that you particularly address this one. This was one of the paragraphs in Section 1.4.3.2, and it addressed the provision that they do a reliability analysis. And you gave your reasons for

deciding it was adequate in there, and I'm trying to find it again. And I was hoping you were going to discuss that. If not, I'll keep looking for it.

MS. RAY: I believe it is on page 1-28 in Chapter 1.

MEMBER BLEY: Oh, okay. I'm not quite there yet. Okay.

MS. RAY: We make a note that they performed a failure modes and effects analysis to evaluate the reliability. However, staff did not investigate that reliability because EDDS is a non-safety system, and we found that they're exempt from GDC-17. So, we did not investigate further on the reliability. We just noted that they did a failure modes and effects analysis.

MEMBER BLEY: Okay. I'm going to look back at it because you did draw some conclusions.

MEMBER MARCH-LEUBA: Yes, but, Dennis, the way it said, I'm thinking -- I always thinking backwards than anybody else. If you have made the analytical analysis and found that this is a highly unreliable system, would that have changed any of these conclusions?

MS. RAY: We don't make any conclusion that it is highly reliable. We make a conclusion that

it is a non-safety system.

MEMBER MARCH-LEUBA: If you mentioned you now know it's highly unreliable, would it change the conclusions?

MS. RAY: No. The conclusion is still non-safety.

MEMBER MARCH-LEUBA: Yes. That's my point, that --

MS. RAY: Which means it is not going to be available in an accident scenario.

MEMBER MARCH-LEUBA: That's my point, that it's good that it is highly reliable.

MS. RAY: Right.

MEMBER MARCH-LEUBA: And we hope to continue to make it highly reliable, but it's not needed.

MS. RAY: Correct.

MEMBER DIMITRIJEVIC: But what you're talking now is not risk-informed discussion at all because you are talking of application to Chapter 15, which is the deterministic principle. You don't talk about the implication on risk at all. So, how are we in a risk-informed area?

MS. RAY: So, our evaluation was following the enhanced safety focus review, which focuses the

review on risk-significant and safety-related items. And for the electrical section, based on those components that we looked at, then, we did the deterministic review. And so, since they met the exemptions, and we granted them the exemptions for GDC-17, we're not going to look further into the EDSS reliability. So, we did use some risk information at the beginning to begin our review, but, then, after that, it is a deterministic review.

MEMBER BLEY: On the page you cited in No. 2, which is the direction to implement a methodology -- and that came from you folks, I believe -- similar to following -- compared the reliability of highly reliable DC systems, that of a typical Class 1E, which I personally didn't think was a useful thing. You come down partway through and you say you reviewed the disposition and found it acceptable because, as you just said, a failure modes and effects analysis was performed to evaluate the reliability. Therefore, the staff finds that the Applicant met that condition of having done this analysis. So, you did look at it. And did you evaluate how well they did that analysis? You know, your statement here is pretty vague.

MS. RAY: Correct, and that is because the failure modes and effects analysis, we did not look at

it in-depth because this is a non-safety system. There's no reason --

MEMBER BLEY: Okay. You probably should have said that here because this gives the implication that you looked at it and concluded they did it well.

MS. RAY: We will take that feedback. I appreciate that comment. We can try and make --

MEMBER BLEY: And there's not a hint of what you just said, that we didn't look at it because --

MS. RAY: I understand.

MEMBER BLEY: -- it's a non-safety --

MS. RAY: I understand your comment.

CO-CHAIR KIRCHNER: Dennis, I have to admit I read it the same way. That's why --

MEMBER BLEY: The way I did?

CO-CHAIR KIRCHNER: Yes.

MEMBER BLEY: Yes.

CO-CHAIR KIRCHNER: Yes.

MEMBER BLEY: It's what it says. That's why you read it that way.

(Laughter.)

MEMBER DIMITRIJEVIC: Well, would the DRAP take care of some of those issues?

MS. RAY: I will refer to my colleague.

MR. NAKANISHI: So, the EDSS system is not in the RAP. But the way we sort of look at this is it sort of has its own RAP program that's got the augmented --

MEMBER DIMITRIJEVIC: Well, I just have to correct you. It's not in DRAP as directly put from the PRA. They have an expert panel. I guarantee you it is going to make it to DRAP because CVCS operates that. I mean things are there. So, it has to make it to DRAP.

CO-CHAIR KIRCHNER: But the CVCS system did not make it the RAP, either.

MEMBER DIMITRIJEVIC: No, some of the -- that's possible. It's part of the containment process --

CO-CHAIR KIRCHNER: It was judged to be not risk-significant.

MEMBER DIMITRIJEVIC: No. Okay. Some part of CVCS is, but not when I look at just the safety containment isolation part of the system. However, the CVCS actuation human action was related as safety-significant. And systems which support safety-significant actions also had to be treated as safety-significant. The PRA input is just one part of DRAP. There is all kinds of other questions they have

to respond to when they make finally --

MEMBER BLEY: But the point you just raised --

MEMBER DIMITRIJEVIC: It's not just the PRA.

MEMBER BLEY: The point you just raised ties these back together again. We've been there before. If that action is important, then the system must be important. And if the system is important, then the electric power going to that system must be important.

CO-CHAIR KIRCHNER: Yes.

MEMBER DIMITRIJEVIC: So, I think that electrical system is definitely going to make it to DRAP. And if it doesn't, something isn't in relation to --

CO-CHAIR KIRCHNER: It's not in there now.

MR. NAKANISHI: So, the operator action, the risk-significant operator actions are addressed through the operator -- you know, there's a separate program that looks at risk-significant operator actions, and they'll be treated accordingly.

MEMBER BLEY: But wait. If an action required this to operate a system, the system has got to be there or you can't do that operation. These

things are daisy-chained together.

MR. NAKANISHI: No, I'm not necessarily disagreeing with you. Now, you know, NuScale has established a Topical Report that determines the risk significance. So, the risk significance, the quantitative approach did not lead those systems to be risk-significant.

MEMBER DIMITRIJEVIC: Right.

MR. NAKANISHI: Now it becomes, like you said, the expert panel approach. And I believe we went back and forth on that particular item, especially CVCS, because you could look at the cutsets, just forget the numbers, but --

MEMBER DIMITRIJEVIC: Yes.

MR. NAKANISHI: But, you know, everything relies upon the ECCS. And one thing that will be a defense-in-depth with that would be a CVCS.

CO-CHAIR KIRCHNER: Especially with exemption to 27.

MEMBER DIMITRIJEVIC: Twenty-seven, right.

MR. NAKANISHI: So, we went back and forth with that, and I was not part of that review. But we accepted their expert panel approach. We audited their meeting minutes. I believe we talked to their expert panel members, and we came to a conclusion

that's not necessary.

MEMBER DIMITRIJEVIC: Well, let me ask you just technically: The DRAP problem, whatever it is now, in the sense of certification is not finally to pass the test, too, right? They have to finalize it after they have expert panel meetings and everything. And they didn't have that now. So, they cannot finalize it.

MR. NAKANISHI: Right. So, yes, the DRAP, upon fuel load, you know, it becomes sort of integrated into existing operational programs, like maintenance rule and things like that. So, it will carry on, but we're making some findings at the DC stage here. And we found the expert panel process to be adequate.

MEMBER DIMITRIJEVIC: Well, I thank you.

CO-CHAIR KIRCHNER: Members, further questions of the staff on the EDSS?

Okay. Somehow, miraculously, we've gotten back close to schedule.

So, at this juncture then, we're going to stop and provide an opportunity for public comment.

I thank the staff and NuScale for your presentations.

So, while we're opening the bridge line,

is there anyone in the room who wishes to make a comment? If so, please come up to the microphone, state your name, and make your comment.

Seeing none, we'll wait for the bridge line. Okay. If there is anyone on the public line who wishes to make a comment, please do so. State your name and make your comment.

I'm just the 5-second rule. I'm not detecting any comment. I think we can go on, then, and close that bridge line. Thank you, Mike.

And we have two pieces of business. The first thing I want to ask of the members, do you feel any need to go back into a closed session to ask any further questions at this juncture?

No? No. Okay. Good. Now we're running well ahead of schedule.

This part, though, may get us back onto the schedule. So, discussion. We need to consider what we would like to hear in a full Committee meeting on the topics that were presented today. And we're open to any suggestions as to either things that you want the Applicant and/or the staff to focus on, on the topics that we covered today.

MEMBER MARCH-LEUBA: Is there a member missing?

Mike can take notes and email them.

CO-CHAIR KIRCHNER: Yes. I mean, this is an internal discussion. I just want to hear what you want to hear or focus on in the full Committee meeting, given that we won't have near this amount of time. I think we will have about a half-day to cover these topics in full Committee.

MEMBER PETTI: I really think only the steam generator topic. I think we can somehow close these others with a letter from a respected -- like we closed the chapters.

CO-CHAIR KIRCHNER: Yes.

MEMBER PETTI: We should be able to close these others, right?

CO-CHAIR KIRCHNER: Yes. Okay.

MEMBER PETTI: I think the steam generator one requires us to -- when are we talking about?

MEMBER BLEY: March the 5th.

CO-CHAIR KIRCHNER: Yes. We were looking at the March timeframe.

MEMBER MARCH-LEUBA: Unless we are asking for original calculations, which I would love to, in which case it might be April or May, whenever they have it.

MEMBER PETTI: Well, the real question is

they're doing some other stuff. Does it make a difference?

MEMBER MARCH-LEUBA: From my point of view, I find it hard to write a positive letter if I don't have a stability map.

MEMBER PETTI: No, that's the question: how close of a --

MEMBER MARCH-LEUBA: A stability map.

MEMBER PETTI: Would delaying a month help? There's two pieces of this, right? I mean, there's the stability of the reactor and the stability of the secondary site. The safety case is based on the stability of the reactor, and we're not saying anything on that, right?

MEMBER MARCH-LEUBA: The reactor is Chapter 15 and the Topical Report. That's done.

MEMBER SUNSERI: So, that's my question: what's the safety significance of these oscillations in the steam generator tubes? Are we worried about creating a containment bypass accident, or what are we really worried about, other than the instability in these --

CO-CHAIR KIRCHNER: No, it's not that.

MEMBER MARCH-LEUBA: It's inoperability.

CO-CHAIR KIRCHNER: Yes, it's

inoperability, but it's also a design issue. To write, as Pete eloquently stated earlier, without that instability map, and then, looking at the operation of the module over the time period, you don't have the total loading to do your design specification or the ASME code case. That's a safety issue because the failure of steam generator tubes would be essentially an accident.

MEMBER SUNSERI: But the only load that I'm understanding that they're not showing to be bounded is the potential for fretting of the tube against the tube support plate, right? That's one of several of those, but now with anything else, that they're all --

MEMBER MARCH-LEUBA: My opinion, with the operator, they're not going to be able to start the reactor is the only problem. The turbine is going to continuously trip whenever they hit 20 percent power.

MEMBER BALLINGER: Yes, but is that a safety issue?

MEMBER RICCARDELLA: If you can't write a design specification for the denominator of the steam generator tubes that has the proper load definitions, then that's something I think we should be concerned about. The level of disagreement between NuScale and

the staff on this subject was surprising.

CO-CHAIR KIRCHNER: Yes, it is surprising because, if you look at the staff's -- if, indeed, it operates primarily in an unstable mode of the lifetime of the steam generator, it's not going to make it. And the safety issue is failure of the steam generator, which is a primary system breach.

MEMBER SUNSERI: And the proposed failure mechanism is wear of the tube.

CO-CHAIR KIRCHNER: No, fatigue. No? Fatigue.

MEMBER RICCARDELLA: They demonstrated with 100 percent oscillations that they meet the endurance limits of the tube from fatigue.

MEMBER BALLINGER: Yes, but I'd like to hear --

MEMBER DIMITRIJEVIC: Yes, but, like Pete says, that doesn't mean -- and Pete said himself that --

MEMBER RICCARDELLA: In the tubes, I haven't seen any analysis of the tube sheet weldings.

MEMBER SUNSERI: Well, that's what I'm trying to understand.

MEMBER PETTI: If I understood what Jose said, it's that the experiment has a boundary

condition because of that pump that is not exactly the same boundary condition that will exist in NuScale. So, there's always going to be an uncertainty in how you extrapolate from TF1 and 2, probably 2, to what you expect in NuScale, the power margin. How that gets resolved, I mean, then, that may be part of a whole issue of why the staff thinks one thing and NuScale this other. I don't know. I haven't been able to close it and put a bow on it yet.

MEMBER DIMITRIJEVIC: What Mark and David are talking about, though, are probability issues, and they're not necessarily safety issues. When it comes to the safety issues, even if they maintain ASME codes, right, because Pete said every failure he saw met those codes. That doesn't guarantee there would not be the fail.

We discussed it in the lunch. This could become a safety issue, if the condition of containment performance requirement doesn't meet this because we've got the frequency of the steam generator, it becomes, let's say, industry level. Probably they will still need that. I have to look at that. But it's higher than industry level because of those issues. That can challenge this safety requirement, but my guess is it's not really likely to happen

because they had all of these other fires, you know, high core damage frequencies which will reduce these.

They calculated steam generator tube failure frequency based on some assumptions. We don't have that work. They just told us. They have a basic thing.

MEMBER BALLINGER: Yes, that's one of the things I'd like to hear them talk about.

MEMBER DIMITRIJEVIC: But this work is much done before all of this discussion happened. So, I have a feeling they have very optimistic assumptions in there, in that report. If they had a more realistic, it will change their frequency significantly. How much, I don't know.

MEMBER BALLINGER: Yes, but, again, what saves them, what may save them is that the failure mode is not first. It's creep collapse.

MEMBER DIMITRIJEVIC: Well, see, okay, but they smartly decide to count that as a failure, as a leak.

MEMBER BALLINGER: Okay.

MEMBER DIMITRIJEVIC: Because you don't what's going to happen when it collapses.

MEMBER SUNSERI: All I'm saying is just -- and this is getting closer to my point --

MEMBER DIMITRIJEVIC: Okay.

MEMBER SUNSERI: -- if we're going to be successful in providing advice or judgment on this design, that's going to be predicated on the safety case, not some owner risk downstream of whatever. Okay? It's the safety case for the plant. We only have a few hours, if that much, at the meeting. We need to get the staff and the Applicant focused on those issues, so we can make a judgment there, is my thought.

MEMBER REMPE: So, along those lines, in March, prior to full Committee week, there's going to be a discussion of PRA Chapter 15 and some other topics. Would it not be wise to defer writing this letter on this topic until we hear that additional information to see if there is some safety cases? What I am wondering is, if we write a letter in March about this, but we learn a lot of stuff at the beginning of the week, if we're making it more difficult for ourselves.

MEMBER PETTI: To me, I'm still concerned, the more I think about Jose's point, if the boundary condition in the experiments that I use to validate NRELAP are not the same boundary conditions in the power module, at least another NRELAP calculation with

a bounding condition that's more typical somehow -- does it change anything? I mean, it could be different, but not have any impact, right? Or it could be different and have an impact. It hasn't been addressed. So, that may be something that we can get.

MEMBER MARCH-LEUBA: And you get into the regulations on what GDC will satisfy. But if we have a design that we certify and sign our names on it, if we believe that it cannot possibly work, and I believe it cannot possibly work, cannot possibly be operated, I don't want to sign my name on it. I mean, it may not violate any GDC, but, at the minimum, you need to tell me how is it going to operate. Is it going to be unstable below 20 percent? Is it going to be unstable between 20 and 100? I mean, you need to know that about your design. How can I certify a design that you, yourself, don't even know?

MEMBER SUNSERI: So, I think they acknowledged that there's a potential for flow oscillations in these tubes, and therefore, that's why they put these flow restrictors in there. And I'm pretty confident that, if the ones that they have aren't sized right away and they learn that through startup testing of their plant, they won't start up until they resize them and get them right. I mean,

that's how plants go through construction and startup testing.

MEMBER MARCH-LEUBA: Or they've just got a hole and go to a void and the thing breaks.

MEMBER BALLINGER: But, again, what Matt is saying is correct. We may need to hold our nose because we're all a bunch of PhDs, at least some of us think we are, sometimes wrong, but never in doubt. It's really, does this design have a significant impact on the health and safety of the public, period? That's our criteria, right? If the steam generator doesn't last three years, well, you know --

MEMBER BLEY: But if we believe that and don't make a strong point of it --

MEMBER BALLINGER: We could say that.

MEMBER BLEY: -- we look as bad as anybody else.

MEMBER BALLINGER: Yes, we could say that for sure.

MEMBER DIMITRIJEVIC: Then, we could say, "We told you."

MEMBER BALLINGER: In the past, when quite a few of these investment risk things actually happened, all of a sudden, everybody's investigating, and it's a big-deal issue throughout the NRC and

everywhere else. It looks like a safety issue at that point. And if we can't agree on experiments and calculations, you've got to do something about it, not just close your eyes and walk away.

CO-CHAIR KIRCHNER: But I think my own opinion is that the first line is to, as Pete suggested, do we have the correct information and confidence in the modeling capability to give that design specification needed for the code? And that is at root a safety issue because failure --

MEMBER BLEY: There's a GDC.

CO-CHAIR KIRCHNER: There's a GDC, yes. So, I think that's the basis, Matt, for having a safety concern. And then, the actual operability of the plant, that may not rise to the level of safety concern, but it should be a concern, and I would expect that we would comment on that in such a letter.

MEMBER SUNSERI: And I don't disagree with that. I mean, I understand what you're saying.

MEMBER MARCH-LEUBA: It is my opinion that, before this meeting, these guys and those guys were not planning to talk about the methodology that we are going to use to define the loads. And because we've been pushing at them, now they say, "Well, we're going to talk to them about the loads, how to define

them." And this Topical Report we've been talking about, at the minimum, our letter should say the staff should get together with NuScale and identify the schedule -- it doesn't need to be for June -- to ensure the methodology used to define the loads is adequate.

Because NRELAP 5 is explicitly forbidden from being used for this application. The staff SER says: thou shall not use NRELAP 5 for this. So, they're violating the law by this.

MEMBER SUNSERI: I've made my point.

CO-CHAIR KIRCHNER: Okay. Others?

So, again, let's just go through quickly. I think I heard a sense that we had sufficient information on the turbine missiles. Dennis, are you --

MEMBER BLEY: I'm not sure we need to do anything beyond an internal letter on that.

CO-CHAIR KIRCHNER: Okay. And then, I believe that the IAV, again, looking at it as part of the system, we will come back to that when we do ECCS focus area.

MEMBER BLEY: And Chapter 19.

CO-CHAIR KIRCHNER: And Chapter 19, right.

My own recommendation would not be to have

a standalone letter on the IAV, but to wait and do the focus area and treat it like the system it is, and not just get focused on the IAV. Is there agreement there? Okay.

Electrical system reliability, an internal letter? Do you think? No?

MEMBER BLEY: Well, we set this meeting up on it by an internal letter on Chapter 8.

CO-CHAIR KIRCHNER: Right.

MEMBER BLEY: We'll just close it.

CO-CHAIR KIRCHNER: Okay. Okay. And then, what we just discussed, the steam generator. So, I'll work with Mike. I think the request to both the Applicant and the staff should be focus on this instability concern as affects providing a design specification that would be used in the ASME code submittal.

MEMBER BALLINGER: Yes, but --

CO-CHAIR KIRCHNER: And then -- go ahead, Ron.

MEMBER BALLINGER: I'm sorry.

CO-CHAIR KIRCHNER: As well as looking at the fretting and wear-related issues.

MEMBER BALLINGER: Yes, but I think, from my point of view, from me understanding the difference

between the staff and the Applicant, we sort of understood that there's a big difference as a result of this meeting, but it sort of tumbled that that was the case. It would be nice to have someone talk to us about what is the difference between your interpretation and my interpretation and why, you know, that kind of thing. So, it's a little bit clearer.

CO-CHAIR KIRCHNER: The cause-and-effect aspect.

MEMBER BALLINGER: Yes. Yes.

CO-CHAIR KIRCHNER: After they figure it out.

MEMBER BALLINGER: Well, maybe they have to say, "We haven't figured it out." But, one way or the other, yes.

MEMBER MARCH-LEUBA: What about if they come here in March and they tell us, "We won't be stable -- we'll be stable about 20 percent," and they come and tell us, "From 20 to 100, I'm unstable," I mean completely diametrically opposed position, our letter has to be, "You guys don't know what you're doing."

MEMBER BLEY: And at that point it ought to be a letter.

MEMBER BALLINGER: Oh, yes, that's for sure.

CO-CHAIR KIRCHNER: Yes, yes.

MEMBER BLEY: Before we get to our last letter saying there's a problem.

MEMBER BALLINGER: I think the steam generator focus area is going to be a letter anyway.

CO-CHAIR KIRCHNER: Yes, that was the intent, that for each of the focus areas, we would have a letter like we did for source term.

MEMBER SUNSERI: I know I was in and out a couple of times today, and I apologize for that. I might have missed something. So, let me ask this question.

I mean, even though I understand that there's a difference of opinion between the staff and the Applicant, they have both arrived at the same conclusion, that with the Safety Evaluation, that there is no safety concern.

MEMBER BLEY: On the primary system.

MEMBER SUNSERI: On the primary? Okay.

MEMBER MARCH-LEUBA: On the GDC, it does not --

MEMBER BLEY: Nobody has challenged that.

MEMBER SUNSERI: Okay, but the staff has

a concern about the DWOs in the secondary side?

MEMBER MARCH-LEUBA: No, we do. I do.

MEMBER SUNSERI: So, that's my question then. We said that there's a big difference, right?

MEMBER RICCARDELLA: If I interpreted the staff's presentation properly, it said that, as far as these DWOs, it's almost universally unstable in all operating conditions.

MEMBER MARCH-LEUBA: Yes.

MEMBER RICCARDELLA: And I don't think you want to do a code fatigue analysis that says that you've got this unstability happening 100 percent of the time while the plant is operating. I doubt that that will survive an ASME code fatigue analysis.

MEMBER BLEY: And in the NuScale open session on this, they agreed they were doing the methodology to examine it, and they were going to include wear and fretting. And that's going to take some time. We haven't seen the plan for that, but it's in progress. So, you know, we might eventually just say that has to be complete and get resolved with agreement with staff, but --

MEMBER MARCH-LEUBA: It may be a confirmatory item for after June. I mean, it doesn't need to be done in May. But we should issue a

recommendation -- when this thing blows up and it doesn't work, they're going to come and say, "Well, you guys approved it."

MEMBER DIMITRIJEVIC: Even I was telling you, not everybody has a safety concern. It's just that I was thinking that maybe through some safety concern we can bring it back because we cannot really -- accountability is not, you know, we cannot truly continue to discuss this.

MEMBER BLEY: So, if we go to the power uprates, we had a big concern about the moisture separators, and that turned out to be real. And there was arguments there that, you know, you shouldn't worry about that; that's an investment risk thing. Well, it could link back to other parts of the system.

MEMBER SUNSERI: Yes, I'm just asking questions to seek to understand.

MEMBER BLEY: I understand.

MEMBER SUNSERI: There's just one more point here, though. I thought what I heard today, though, is that, from a staff's perspective, there is -- maybe I'm not going to use the right language here -- but there is, essentially, no hold on the design certification at this point. There may be some other work and additional things that the Applicant needs to

do that goes into this ASME code report that they're going to inspect later that may hold up the actual building of the plant. But am I interpreting that position properly?

MEMBER BLEY: I think you've got that right.

MS. PATTON: Yes, it's basically what I explained earlier. We are having discussions with them regarding the methodology and the appropriate licensing process for that methodology, but the ITAAC itself is what would ultimately have the whole ASME report and everything. And that's inspection process and everything.

MEMBER BLEY: We have the idea Jose brought up, and I really support that. This is complex enough and confusing enough, at least right now, that a Topical Report to get agreement on how they're going to go ahead with developing the methodology is really in line.

MEMBER BROWN: How can you put this off for an ITAAC after a license is complete? I thought that's what I just heard. Do you understand what I'm saying? I mean, I'm going back 10 years with the I&C stuff. There was a big issue on the ITAACs and the DACs. In other words, there was lists of ITAACs and

DACs all characterized in the licensing, in the DCD, in the licensing document. And we eventually said that's not going to work. We've got to have a framework; we've got to have a design at which you can focus the ITAACs and DACs on. And I'm just listening, and it doesn't sound like you have a design for the steam generators. How do you focus ITAACs on that and settle it after the license is approved? That seems to me to be a disconnect.

I might be wrong because I, obviously, don't know as much about DWO types. It sounds kind of fascinating. But fighting is not a good idea, no matter what you do. You're trying to figure out a way to accept permanent oscillations forever at all power levels, and that somehow just, if you say that out loud, it doesn't sound real crisp.

This is kind of the same as this thing. You've got to have a framework or a basis, a methodology, a Topical Report, something that focuses-in and says these are the things you have to resolve to say this is okay. And it doesn't sound like you've got that, based on listening today.

CO-CHAIR KIRCHNER: For me personally, it feels like a confirmatory item, that the Applicant demonstrate and the staff accept the analysis that

will eventually inform things like the specification for the ASME case.

MEMBER BROWN: That would be more comfortable for me.

CO-CHAIR KIRCHNER: I would be much more comfortable with such an approach, because I have something of a gut-level feeling this is not something that's going to be resolved in the next couple of months.

MEMBER DIMITRIJEVIC: Right. No, definitely not.

CO-CHAIR KIRCHNER: I mean, if we're working on first checking methodology and agreeing on the methodology is acceptable, and then, you're doing the analyses, that doesn't sound like a couple of months to me.

MEMBER BROWN: Nobody was talking that way that I heard.

MEMBER BLEY: By the way, if anybody lost a cover for an Ironkey USB drive, there it sits.

CO-CHAIR KIRCHNER: Okay. We have some homework to do, I think, and preparation for the full Committee meeting on this topic of steam generator.

Further comments at this point? Anyone from the staff wish to make any comment?

Jeff, please. Since we're on record, just give them your name. He knows who you are.

MR. SCHMIDT: This is Jeff Schmidt, Reactor Systems.

I was asked if there was an SRM on the SECY-18-0099 and I said yes. That's incorrect. There is not. That was an information-only SECY. So, you wouldn't get an SRM on that. So, I just wanted to set the record straight. I got confused with the IAV for a second there. That's all I have.

MEMBER BLEY: How could that be?

(Laughter.)

MR. SCHMIDT: There's a lot of SECYs.

All right. Thank you. I apologize for that.

CO-CHAIR KIRCHNER: Okay. Thank you, Jeff.

MEMBER BLEY: Thanks, Jeff.

CO-CHAIR KIRCHNER: All right. If there are no further comments from members, we are adjourned.

(Whereupon, the above-entitled matter went off the record at 5:03 p.m.)

January 31, 2020 Docket No. 52-048

U.S. Nuclear Regulatory Commission
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11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Titled "ACRS Subcommittee Presentation: NuScale FSAR – Steam Generator Design," PM-0220-68568, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) Subcommittee Meeting on February 4, 2020. The materials support NuScale's presentation of the NuScale steam generator design.

The enclosure to this letter is the nonproprietary presentation titled "ACRS Subcommittee Presentation: NuScale FSAR – Steam Generator Design," PM-0220-68568, Revision 0.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Marty Bryan at 541-452-7172 or at mbryan@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
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Distribution: Robert Taylor, NRC, OWFN-8H12
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Bruce Bavol, NRC, OWFN-8H12

Enclosure: "ACRS Subcommittee Presentation: NuScale FSAR – Steam Generator Design," PM-0220-68568, Revision 0

Enclosure:

"ACRS Subcommittee Presentation: NuScale FSAR – Steam Generator Design," PM-0220-68568, Revision 0

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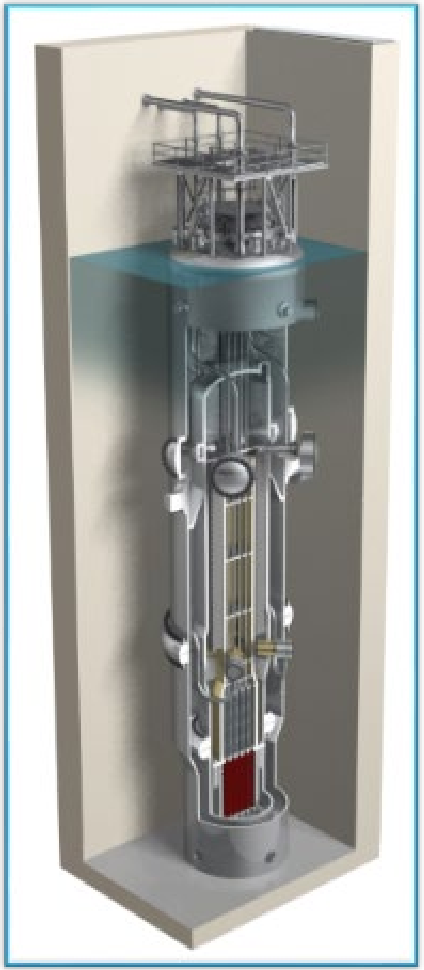


ACRS Subcommittee Presentation

NuScale FSAR

STEAM GENERATOR DESIGN

February 4, 2020

PM-0220-68568
Revision: 0

Presenters

Kent Welter
Engineering Chief, Testing and Analysis

Kevin Spencer
Mechanical Engineer, NSSS Engineering

Joe Remic
Supervisor, NSSS Component Analysis

Brian Wolf
Supervisor, Code Development

Marty Bryan
Licensing Project Manager

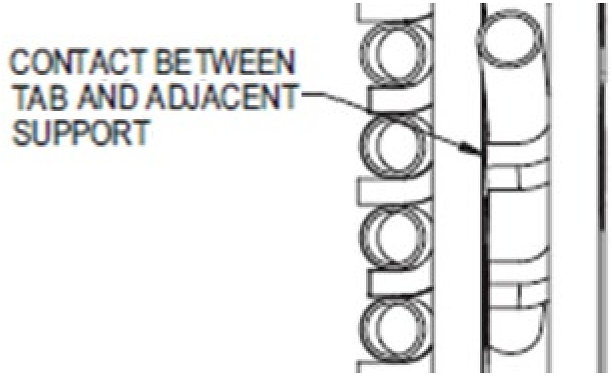
Steam Generator Design - Open Session Agenda

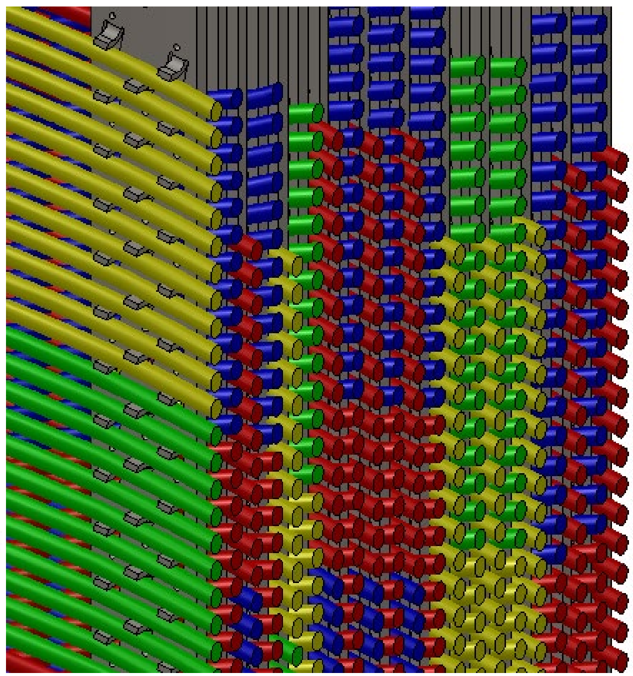
- Steam Generator Design
- Steam Generator Inspection Program
- Density Wave Oscillation Overview
- ITAAC Closure Path for DWO
- Preliminary Scoping Study
- Planned Closure Activities Through ITAAC

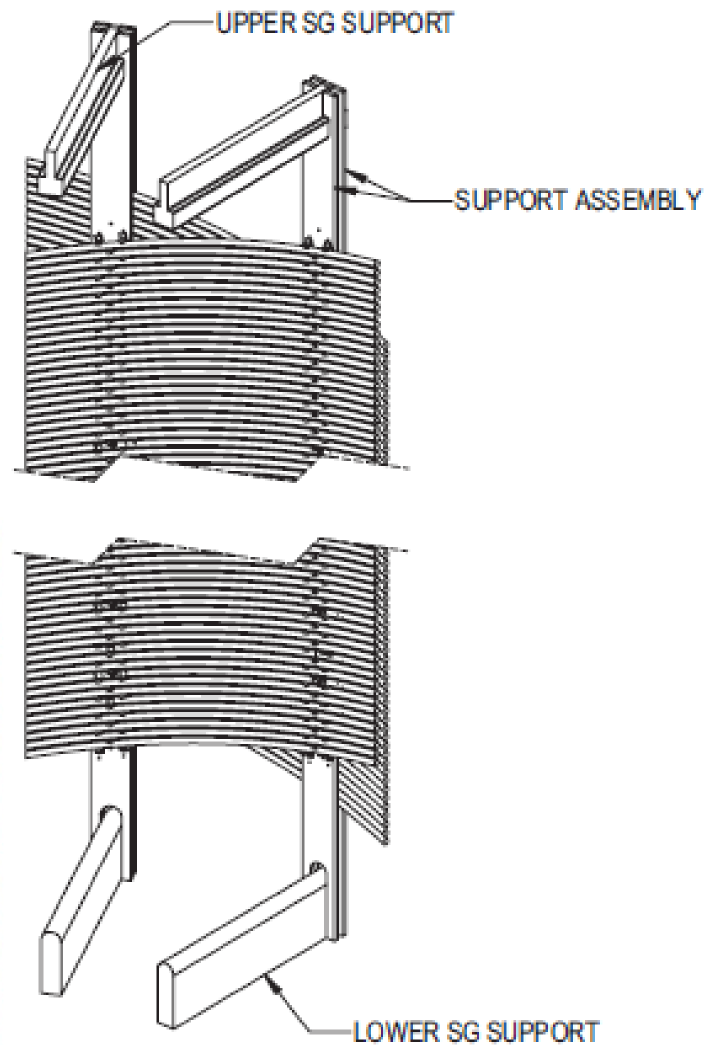
Steam Generator Design

- **Integral Helical Coil SG Design features**
  - Shell side is primary side - Tube side is secondary side
  - Alloy 690 TT (1380 tubes, 77 - 87ft long, 5/8” OD)
  - Low flow in primary (~1ft/sec)
  - Tube wall degradation allowance (0.010” > ASME min wall)
  - Support 100% volumetric inspection
  - Normal access to shell side of tubes from below during refueling
- **Incorporation of Operating Experience**
  - Follow guidance of NEI 97-06 & EPRI (COL Item 5.4-1: Develop and implement a SG Program)
- **Flow restrictor design at SG tube inlet ensures acceptable tube flow fluctuations during operation**

Steam Generator Design (Cont'd)





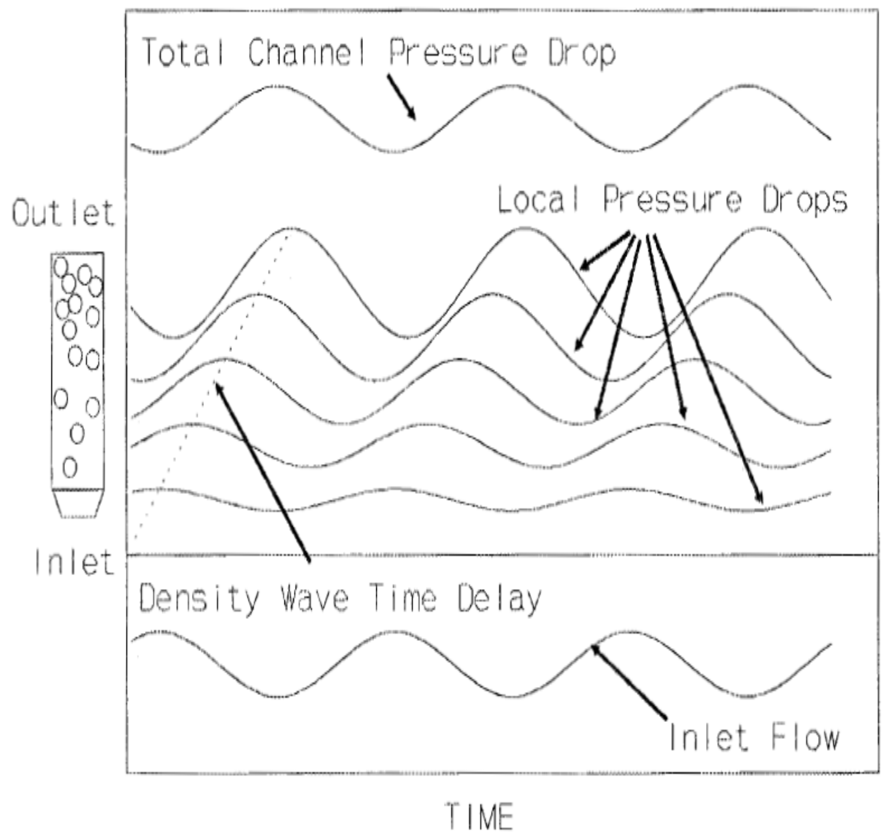


Steam Generator Inspection Program

- Monitors performance and condition of SGs as part of ISI
- Implements applicable portions of B&PV code Section XI and 10 CFR 50.55a(g)
- Appendix B to 10 CFR 50 applies to implementation
- Follows NEI 97-06 and EPRI guidance 1013706 Rev 7 “Steam Generator Management Program, PWR SG Examination Guidelines”
- Includes
  - Degradation assessment (including wear due to fretting)
  - Tube integrity assessment
  - Shell side integrity assessment

Density Wave Oscillation – Overview

- Consider a case for inlet flow reduction
  – sinusoidal flow fluctuation
- Causes increase in void
- Voids move through the channel
- Voids movement also known as Density Wave Motion
- Density wave produces pressure drop fluctuations
- Pressure drop is delayed with respect to the inlet flow fluctuations
- At certain frequency
  - Total pressure drop is completely out of phase with the flow fluctuations
  - Effectively produces the negative pressure drop
  - Causes flow surge
  - Cycle repeats
- DWO can be self sustaining



(March-Leuba, J., "Density Wave Instabilities in BWR", NUREG/CR-6003, Oct. 1992.)

Density Wave Oscillation – Overview (continued)

- DCA Rev. 3 Section 3.9.1 discusses the possibility of secondary flow oscillations during power ascent and descent
- Section 5.4.1.2 addresses Inlet Flow Restrictor (IFR) sizing to limit DWO to acceptable limits
- The NuScale Stability Topical Report TR-0516-49417 concludes that secondary oscillations do not challenge fuel thermal limits
- Additional review of the DCA language is ongoing and adjustments will be made in a final revision as necessary

ITAAC Closure Path for DWO

- ITAAC 02.01.01 requires that an inspection is performed of the NuScale Power Module "ASME Code Class 1, 2, 3, and CS as-built component Design Reports to verify that the requirements of ASME Code Section III are met"
- Tier 1 Table 2.1-2 defines the NuScale Power Module (NPM) ASME Code Class 1, 2, 3, and CS components:

Equipment Name	ASME Code Section III
RCS Integral RPV/SG/Pressurizer	1

- Therefore, ITAAC 02.01.01 requires the inspection of the certified ASME Design Reports for the RPV and the steam generator

ITAAC Closure Path for DWO (continued)

- Subsection NCA of the 2013 Edition of the ASME Code defines requirements of what is to be included in ASME Design Specifications and ASME Design Reports.
  - Design Specifications
    - NCA-2142.2 requires that Design Specifications identify all loadings (e.g. pressure, temperature, mechanical loads, cycles, and/or transients) and the service limits a component will experience
      - » Loading combinations for the RPV (including SG tubes) defined in Table 3.9-3 of DCA
      - » Transient (TH) loads are based on time history of design basis transients, described in DCA Section 3.9.1.

ITAAC Closure Path for DWO (continued)

- Subsection NCA of the 2013 Edition of the ASME Code defines requirements of what is to be included in ASME Design Specifications and ASME Design Reports.
  - Design Reports
    - NCA-3260 requires that the Design Report evaluate the loads and load combinations as defined in the Design Specification to the applicable acceptance criteria are met.
    - NCA-5350(4) requires the Design Report be certified by a Registered Professional Engineer (RPE) competent in the applicable field. The RPE confirms all loads and load combinations specified in the Design Specification have been addressed and satisfied.
    - NCA-8310(a) permits application of the ASME 'N' Certification Mark to the Reactor Pressure Vessel which provides certification by an Authorized Nuclear Inspector that all ASME examinations and testing have been completed.
- Therefore, the resolution of DWO will be achieved through ITAAC 02.01.01 activities related to the RPV and the steam generator

Preliminary Scoping Study

- Steam Generator Tubes
  - The applicable primary side conditions were applied to the exterior of the SG tubes and DWO full flow reversal was considered on the interior (secondary side) for the total length of the tube.
  - The resultant alternating stress due to this oscillation is below the ASME Code endurance limit; therefore, preliminary analyses demonstrate that DWO full flow reversal will not result in any fatigue damage to the steam generator tubes.
- Feedwater Plenum Tube-to-Tubesheet Weld
  - Preliminary analyses using bounding DWO transient definitions, assuming complete tube dry out, and conservative tube to plenum interaction, resulted in alternating stresses above the ASME Code endurance limit.
    - The DWO transient definition in this vicinity is undergoing further evaluation.
    - The preliminary DWO alternating stress in relation to the ASME endurance limit, coupled with the analytical conservatisms applied in this evaluation, enable NuScale to confidently predict the final alternating stress due to DWO in this region will be below the ASME Code endurance limit.
    - ITAAC 02.01.01 requires the inspection of the Design Report in which this evaluation will be documented.

Planned Closure Activities Through ITAAC

- ITAAC 2.01.01 inspection is performed of the NuScale Power Module "ASME Code Class 1, 2, 3, and CS as-built component Design Reports to verify that the requirements of ASME Code Section III are met"
  - NRELAP5 will provide thermal-hydraulic time histories of design transients in support of the fatigue analysis, including fluid temperatures, pressures, nominal flow rates, and quality
  - TF-1 and TF-2 test data will provide a basis for parameters used to calculate film coefficients and resulting stresses during DWO
  - Thermal response of the tube is much faster than the DWO frequency, so a quasi-static analysis can be performed
  - The stresses from all loadings and transients (including those from DWO) will be combined and compared to allowable stress limits
  - The goal of the SG ASME Code calculations is to confirm that the alternating stresses on the SG tubes, the tube-to-tubesheet welds, and the tubesheets due to DWO stresses are below the ASME endurance limit; thereby enabling these components to withstand infinite cycles

NuScale Conclusion

- The successful completion of ITAAC by the licensee constitutes the basis for the NRC determination to allow operation of a facility certified under 10 CFR 52

Acronyms

ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
DCA	Design Certification Application
DHRS	Decay Heat Removal System
DWO	Density Wave Oscillation
EPRI	Electric Power Research Institute
FSAR	Final Safety Analysis Report
FW	Feedwater
IFR	Inlet Flow Restrictor
ISI	Inservice Inspection
ITAAC	Inspection, Test, Analysis and Acceptance Criteria
NEI	Nuclear Energy Institute
NPM	NuScale Power Module
NSSS	Nuclear Steam Supply System
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
SG	Steam Generator
TH Code	Thermal Hydraulic Code

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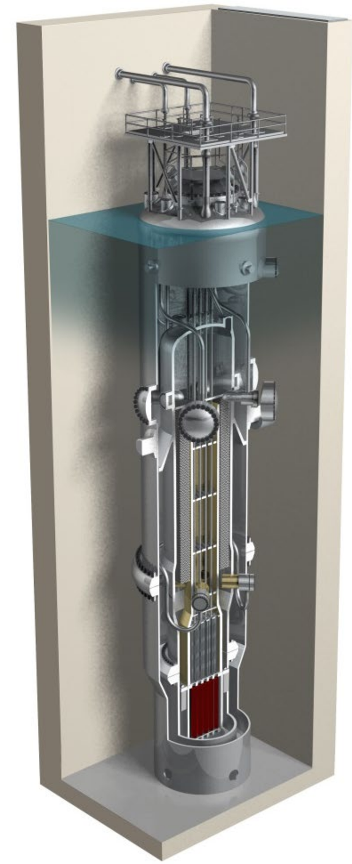

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Protecting People and the Environment

Steam Generator Design and Related Topics: Steam Generator

NuScale Design Certification Application

ACRS Subcommittee Meeting
February 4, 2020

(Open Session)

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
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Agenda

- NRC Staff Review Team
- Summary of the NRC Staff’s Review
- Steam generator materials, design, inspection (Chapter 5)
- Steam generator component design (Chapter 3)
- Initial Test Program (ITP) and Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC); (Chapter 14)
- Accident analysis (Chapter 15)
- Secondary-side flow stability (Chapter 5)

NRC Staff Review Team

- Technical Reviewers:
  - Gregory Makar
  - Yuken Wong
  - Peter Yarsky
  - Raymond Skarda
  - Carl Thurston
  - Leslie Terry
  - Kaihwa Hsu

- Project Management:
  - Marieliz Johnson
  - Bruce Bavol

FSER Section 5.4.1 and 5.4.2 Steam Generator Review

- Selection, processing, testing, and inspection of materials
- Design for limiting susceptibility to progressive degradation
- Fabrication and processing of ferritic carbon and low-alloy steels
- Fabrication and processing of austenitic stainless steel (pressure boundary)
- Compatibility with the coolant (primary and secondary) and cleanliness control
- Access to shell side
- Ability to implement a steam generator program for progressive forms of degradation (primary and secondary)
- Reasonable assurance that tube integrity (structural and leakage) will be maintained

Staff's Evaluation:

NuScale Steam Generators

Key Differences Compared to Operating Reactors

- Primary pressure outside the tube
- Helical coil shape
- First of a kind tube support structure (design and inspection requirements)
- First of a kind feedwater plenum and steam plenum designs
- Inlet flow restrictors with mounting plate (secondary side)

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Staff's Evaluation:

Effect of these differences on staff review

- Degradation assessment different than operating fleet
- Structural integrity evaluation based on tube collapse rather than burst
- Possible new effects of secondary coolant on degradation
- New flow effects and loose parts potential inside and outside tubes
- Tube support and flow restrictor dynamic effects
- Inspection methods require development (tube shape, internal deposits)
- Inspection analyses require qualification
- Inspection interval

Staff’s Evaluation:

Status of the Steam Generator review

Chapter 5 confirmatory items will clarify:

- ASME Code classification of the SG supports and SG tube supports
- Description of the steam and feed plenum design

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Staff's Evaluation:

Steam Generator Program description (FSER 5.4.2)

- Manages progressive forms of degradation

- Mature program for operating reactors
  - Steam Generator Program Guidelines (NEI 97-06)
  - Electric Power Research Industry (EPRI) guidelines
  - Technical Specifications

- Technical Specifications related to tube integrity and inspection
  - Define and require structural and leakage integrity
  - Require inspection, condition monitoring, operational assessment
  - Include full length of the tube using capable techniques
  - Limit primary-to-secondary leakage
  - Limit the allowable through-wall degradation
  - Serve as the ASME Code inservice inspection for SG tubes
  - Require secondary water chemistry program
  - Require inspection reports

Staff's Evaluation:

Steam Generator Program inspections

- Tubes inspected from the inside with eddy current testing (ET)
  - Internal and external volumetric degradation (corrosion, wear)
  - Internal and external cracking
- Cracking – plug tube when detected (some plants have exceptions through license amendment)
- Wear from support structures and foreign objects managed based on the cause, ET characterization, and projected growth
- Visual examination on the shell side (tubes, supports, deposits)
- NuScale program described in FSAR Section 5.4.1.6

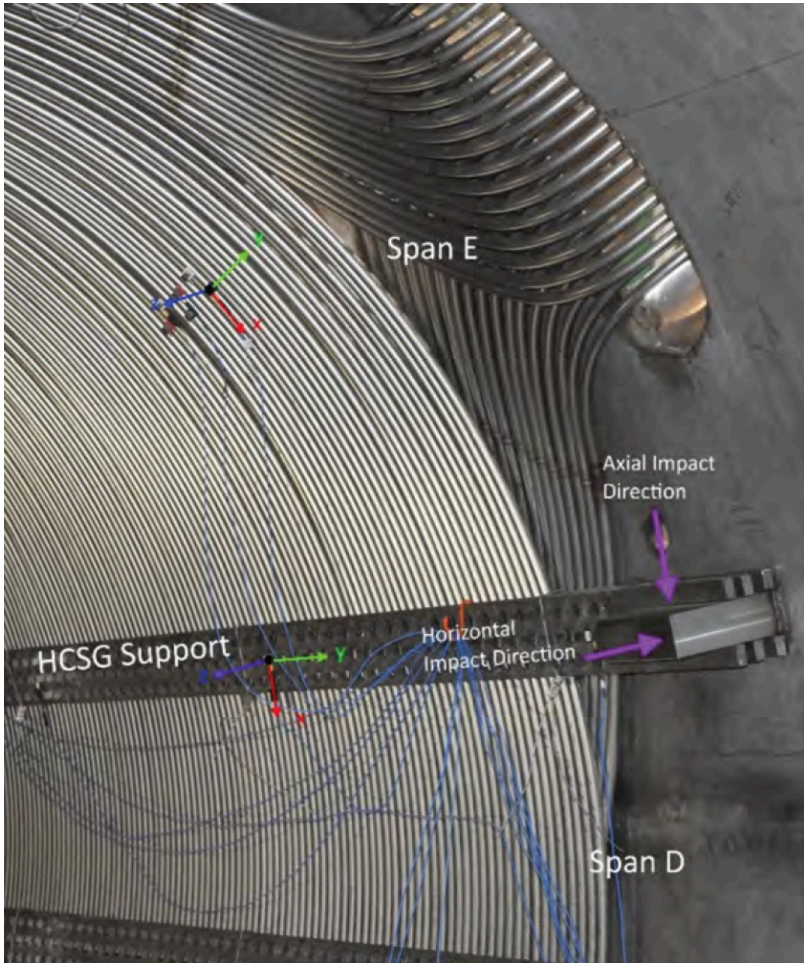
Staff's Evaluation: Chapter 3

Chapter 3 topics:

Flow-Induced Vibration
ASME Section III Stress Analysis

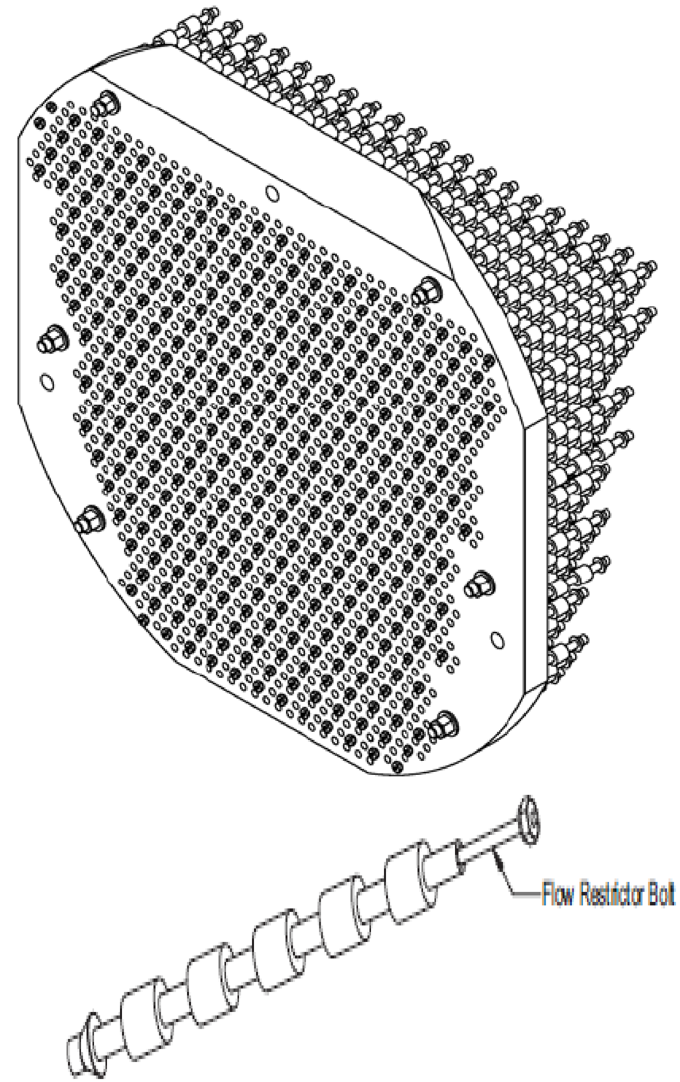
Staff's Evaluation: HCSG FIV

- Mechanisms evaluated:
  - Flow induced vibration, fatigue life, wear
    - Vortex shedding (VS)
    - Fluid-elastic instability (FEI)
    - Turbulent buffeting (very low due to slow flow)
- Analysis methods are reasonable; margins are acceptable
- Design to be validated with testing in SIET TF-3 facility
  - DCA Part 2 Tier 1, Section 2.1.1 "Design Description"
  - Part 2, Tier 2, Table 14.2-72, "Steam Generator Flow-Induced Vibration Test #72"
  - Acceptance criteria are reasonable



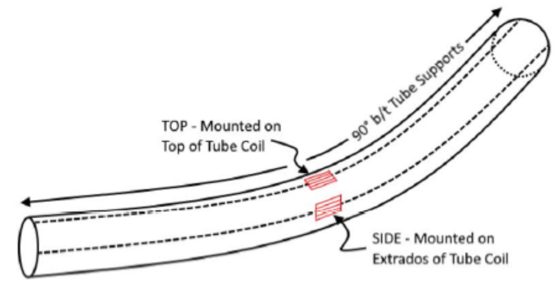
Staff's Evaluation: SG Inlet Flow Restrictor

- Mechanisms evaluated:
  - Leakage flow instability (LFI)
- Design concepts evaluated by tests
  - Significant margin against LFI
- Final design to be validated with testing prior to initial startup
  - Test plan and acceptance criteria described in comprehensive vibration assessment program (CVAP) Measurement and Inspection Plan Technical Report
    - Plan and criteria are acceptable



Staff's Evaluation: DWO FIV

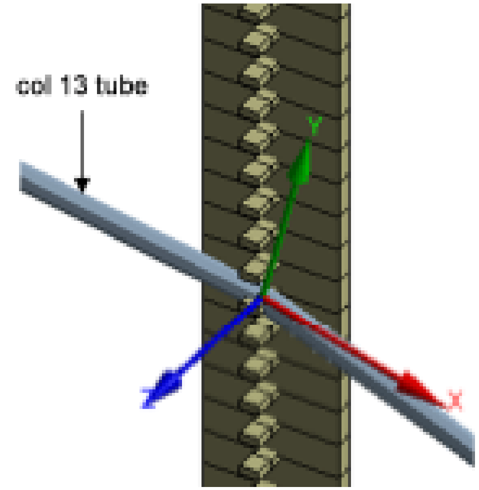
- Density wave oscillation (DWO) observed in TF-2 testing
  - Temperature and flow oscillations in secondary coolant
- TF-2 tubes instrumented with strain gages
- Very long oscillation period
  - Frequency very low
    - DWO will not excite HCSG structural resonances
- Surface strain fluctuations on the order of hundreds of microstrain
  - Alternating stress intensities appear to be below fatigue endurance limit (ASME calculations performed under ITAAC will provide final confirmation)
  - Max surface stresses are below yield and ultimate strengths

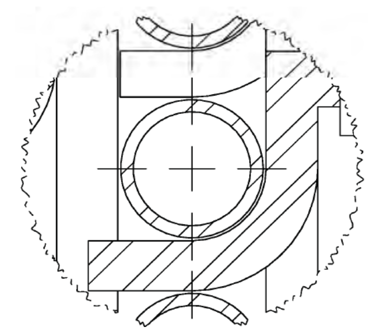


Alloy 690 (Ni-Cr-Ir)

Staff's Evaluation: DWO Wear

- Sliding and wear
  - Oscillating thermal loading could cause tubing to expand and contract, leading to sliding along the tube support tabs
  - Long term wear will be monitored by steam generator program inspections





Staff's Evaluation: DWO ASME III

- SG tubes are ASME Class 1
- ASME Section III Code requires analysis to address fatigue and design basis loads
- ITAAC 2.1-4.2 exists to confirm that ASME Class 1 components in DCA Tier 1, Table 2.1-2 are designed to ASME Section III requirements
    - Table 2.1-2 includes "RCS integral RPV/SG/Pressurizer"

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Staff's Evaluation: DWO Initial Startup

- One tube in prototype HCSG will be instrumented with strain gages at top, middle, and bottom
  - If DWO occurs, will be measurable in strain gages
  - However, measurements not intended for DWO, since SG tube ASME Section III analyses will include effects of DWO

Staff's Evaluation: Summary

- DCA commitment ensures TF-3 testing will be performed and acceptance criteria are met
  - DCA Part 2 Tier 1, Section 2.1.1 "Design Description"
  - Part 2, Tier 2, Table 14.2-72, "Steam Generator Flow-Induced Vibration Test #72"
- Density Wave Oscillation (DWO) does not induce significant HCSG vibration
- The effects of any thermally induced stresses will be addressed in ASME Section III analyses via ITAAC
- SG inlet flow restrictor (IFR) testing will confirm the IFR final design will not have adverse vibration effects due to leakage flow instability

Staff's Evaluation:

Additional Steam Generator topics

- Chapter 15 – Accident Analysis
  Steam generator design affects some accident analyses

- Stability Topical Report (TR-0516-49417-P)
  - Evaluation of power module stability
  - Assumed secondary-side flow instability

- Secondary-side density wave oscillation (DWO)

Staff's Evaluation:

Secondary-Side Density wave oscillation (DWO)

- May generate cyclic loads (must be considered for ASME Code stress calculations)
- ITAAC 2.1-4.2 exists to confirm that ASME Class 1 components are designed to ASME Section III requirements
- Applicant is developing a methodology to analyze secondary-side DWO and assess the methodology for this use

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Abbreviations

ACRS - Advisory Committee on Reactor Safeguards
ASME - American Society of Mechanical Engineers
CVAP - Comprehensive Vibration Analysis Plan
DWO - Density Wave Oscillation
EPRI - Electric Power Research Industry
ET - eddy current testing
FEI - Fluid-Elastic Instability
FSAR - Final Safety Analysis Report
HCSG - Helical Coil Steam Generator
IFR - inlet flow restrictor
ITAAC - Inspection, Test, analysis, and acceptance criteria
ITP - Initial Test Program
LFI - Leakage Flow Instability
RCS - Reactor Coolant System
RPV - Reactor Pressure Vessel
SF - Single Failure
SG - Steam generator
SGIFR - Steam Generator Inlet Flow Restrictor
VS - Vortex Shedding
TB - Turbulent Buffeting

Questions?

BACK-UP SLIDES

Staff's Evaluation: Chapter 3

Chapter 3 topics:

Flow-Induced Vibration
ASME Section III Stress Analysis

See Closed Session

NRELAP5 SG Modeling and it's Application to Chapter 15 events

Carl Thurston
NRR/DSS/SNRB

See Closed Session

Secondary Side Density Wave Oscillations

Ray Skarda, Ph.D.- Reactor Systems Engineer, RES
Peter Yarsky, Ph.D.- Senior Reactor Systems Engineer, RES

See Closed Session

January 28, 2020

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled “ACRS Subcommittee Presentation: NuScale FSAR Chapter 3 – Protection Against Turbine Missiles,” PM-0220-68162, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) Subcommittee Meeting on February 4, 2020. The materials support NuScale’s presentation of protection against turbine missiles.

The enclosure to this letter is the nonproprietary version of the presentation entitled “ACRS Subcommittee Presentation: NuScale FSAR Chapter 3 – Protection Against Turbine Missiles,” PM-0220-68162, Revision 0.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Marty Bryan at 541-452-7172 or at mbryan@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

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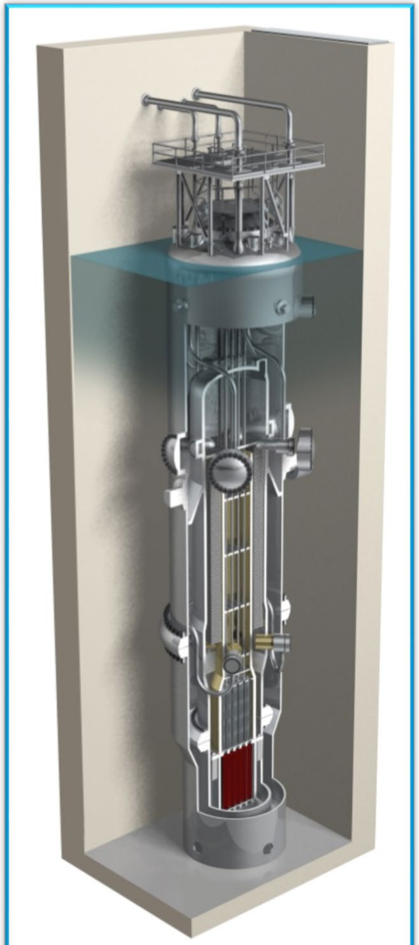
Enclosure: “ACRS Subcommittee Presentation: NuScale FSAR Chapter 3 – Protection Against Turbine Missiles,” PM-0220-68162, Revision 0

Enclosure:

"ACRS Subcommittee Presentation: NuScale FSAR Chapter 3 – Protection Against Turbine Missiles," PM-0220-68162, Revision 0

ACRS Subcommittee Presentation



NuScale FSAR

Chapter 3

Protection Against Turbine Missiles

February 4, 2020

PM-0220-68162
Revision: 0

Template #: 0000-21727-F01 R5

Presenters

Josh Parker, P.E.
Civil/Structural Analysis Supervisor

Marty Bryan
Licensing Project Manager

Turbine Missiles – Overview of Topics

- Turbine Missile Definition – Parameters
- Analysis Approach and Methodology
  - Local analyses
  - Global analysis
- Illustrations of Turbine Missile Trajectory and Barriers
  - Reactor Building
  - Control Building
- Conservatisms in Approach
- Summary

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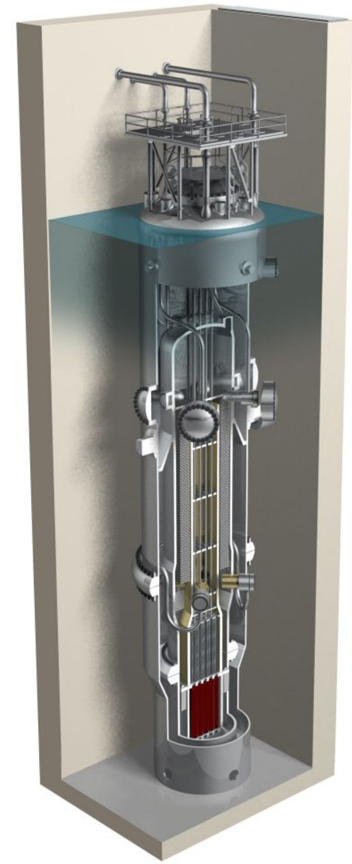

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Turbine Missile

NuScale Design Certification Application

ACRS Subcommittee Meeting
February 4, 2020

NRC Staff Review Team

- **Technical Reviewers:**

  Sujit Samaddar, NRR/DEX/ESEA

  John Honcharik, NRR/DNRL/NPHP

- **Project Management:**

  Marieliz Johnson, NRR/DRNL/NRLB

NuScale
Turbine Missile Spectrum

Assessment of the Turbine Missiles for which protection is required for Compliance with GDC 4

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Determining Turbine Missile Parameters

- NuScale proposes the bounding missile to be 24 inch radius half circle of the rotor weighing 3568 pounds, based on using half of the last stage rotor with blades attached.
- NuScale proposes the bounding missile speed of 476 mph, based on 190% destructive overspeed (6840 rpm).
- COL Item 3.5-1- A COL applicant will demonstrate site specific turbine missile parameters are bounded by the design certification analysis, or provide site-specific turbine parameters.
- Staff concludes the bounding missile described above is acceptable based on past events as documented in EPRI reports and NUREG-1275.

Regulatory Basis and Use of Barriers

- 10 CFR Part 50, Appendix A, GDC 4, requires SSCs important to safety to be appropriately protected against environmental and dynamic effects, including the effects of missiles that may result from equipment failure.
- Safety-related and risk-significant SSCs for the NuScale design are located within the RXB and CRB.
- Turbine generator rotor shafts are unfavorably oriented such that the RXB and CRB are within the turbine low-trajectory hazard zone.
- To meet the requirements of 10 CFR Part 50, Appendix A, GDC 4, NuScale proposes to use installed or existing structures for protecting safety related SSCs.

NuScale
Turbine Missile Barriers

Assessment of barriers providing required protection in compliance with GDC 4

Acceptance Criteria for an effective barrier:

RG 1.115 and SRPs 3.5.1.3 and 3.5.3 recommend sufficient thickness of concrete barrier(s) to prevent perforation and scabbing in the event of turbine missile impact.

Turbine Missile Barriers:

- 5 ft thick reinforced concrete exterior wall of the RXB as the primary barrier.
- 3 ft reinforced concrete exterior wall and the 3 ft reinforced concrete slab at grade, collectively, as the barriers for the CRB.

Analysis Methodology:

- TeraGrande non-linear finite element code with ANACAP reinforced concrete model.
- Estimate the depth of penetration.
- Use modified National Defense Research Council equation to establish minimum barrier thickness to meet acceptance criteria.

Confirming simulation capability of computer code:

- TeraGrande code with ANACAP reinforced concrete is well established and used for AIA.
- Ability to simulate full scale high energy rotor impact test conducted at SNL by EPRI.
- Close correlation between simulation and test results.

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Barrier Model:

- Walls and slab represented by plate elements.
- The model boundaries fixed.
- Modeled area is smaller than actual wall dimension.
- Impact on center of the model.
- Strain contours plotted.

Results:

- Change in stress levels radiating from impact zone rapidly tend to be zero (localized effect).
- Global effect of impact can be ignored.
- RXB wall near perforation.
- Assumed all equipment lost in service gallery from secondary missiles (scabbing), redundant equipment is provide in a separate location to keep the plant safe
- CRB wall perforated – exit-velocity relatively low – minimum thickness requirement met based on computed floor-slab penetration.

Staff Conclusion:

There is reasonable assurance that SSC’s important to safety will be protected as required by GDC 4, from the characterized missile, by the protective barriers considered in the NuScale design.

Abbreviations

AIA - aircraft impact assessment
CRB - Control Building
EPRI - Electric Power Research Industry
RXB - Reactor Building
SSC - structures, systems, and components
SNL – Sandia National Laboratories

Questions?

BACK-UP SLIDES

Protection Options:

- Reducing the potential for missile generation.
- Selecting favorable turbine orientation.
- Using barrier(s) for missile protection.

January 31, 2020 Docket No. 52-048

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SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled “ACRS Subcommittee Presentation: NuScale FSAR – Emergency Core Cooling System Valve and Inadvertent Actuation Block Design,” PM-0120-68441, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) Subcommittee Meeting on February 4, 2020. The materials support NuScale’s presentation of the emergency core cooling system valve and inadvertent actuation block design.

The enclosure to this letter is the nonproprietary presentation entitled “ACRS Subcommittee Presentation: NuScale FSAR – Emergency Core Cooling System Valve and Inadvertent Actuation Block Design,” PM-0120-68441, Revision 0.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Rebecca Norris at 541-602-1260 or at RNorris@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
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Enclosure: “ACRS Subcommittee Presentation: NuScale FSAR – Emergency Core Cooling System Valve and Inadvertent Actuation Block Design,” PM-0120-68441, Revision 0

Enclosure:

"ACRS Subcommittee Presentation: NuScale FSAR – Emergency Core Cooling System Valve and Inadvertent Actuation Block Design," PM-0120-68441, Revision 0

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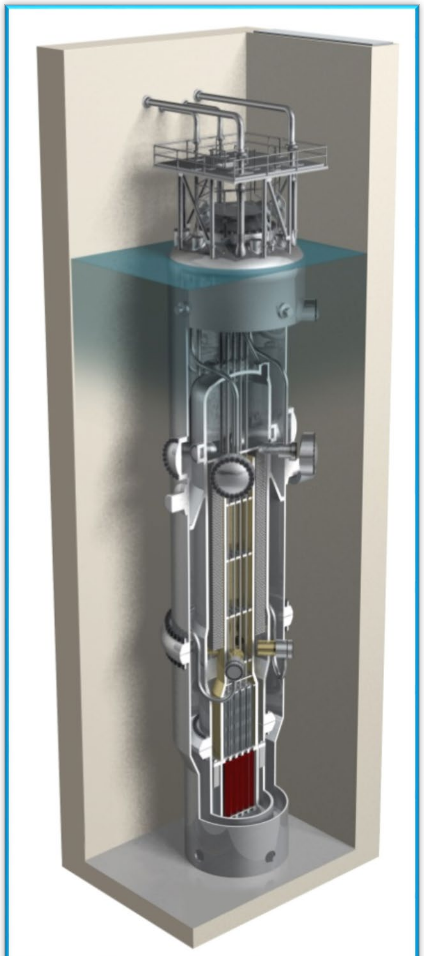
ACRS Subcommittee Presentation



NuScale FSAR

Emergency Core Cooling System Valve and Inadvertent Actuation Block Design

February 4, 2020

PM-0120-68441
Revision: 0

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Template #: 0000-21727-F01 R5

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Paul Infanger

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Engineering

ECCS Valve Operation

- Emergency core cooling system (ECCS) valves receive actuation demand on an ECCS actuation signal or loss of DC power to ECCS trip valves
- Inadvertent actuation block (IAB) feature prevents ECCS valve opening if the differential pressure between the reactor coolant system (RCS) and containment (CNV) is above the IAB threshold pressure
  - This feature prevents opening due to spurious signals or equipment failures at normal operating pressures but permits opening in loss-of-coolant accident (LOCA) conditions
  - Previous IAB range 1000-1200 psi
    - Upper limit supports critical heat flux (CHF) criteria for postulated inadvertent opening or loss of power events
    - Lower limit protects coolant levels for small breaks

ECCS Valve Testing

- Initial ECCS valve testing was done in 2015 for "Proof of Concept"
- Testing was done with air and ambient temperature water
  - Testing verified ECCS valve functionality with remote pilot actuators
  - Testing verified IAB blocking and release functions

ECCS Valve Audit

- Identified need to demonstrate ECCS valve functions at operating conditions
  - DCA Demonstration Test Program performed ECCS valve and IAB function tests at operating pressures and temperatures
  - Two-phase effects in the ECCS valve control chamber and trip line affected operation of the IAB
    - Slower response affected the ability of the IAB to reliably block at threshold (approximately 1200 psid) and release 1000–1200 psid
    - Adjustments were made to the IAB spring force and internal orifices
    - Sensitivity testing determined that 300 psi to 400 psi range (versus 200 psi) was needed for reliable blocking and release
    - IAB test was repeated and demonstrated to reliably block above 1300 psid and release between 900 psid and 1000 psid

Previous FSAR Analysis

- DCA Revision 3 Chapter 6 and 15 analyses prior to IAB changes (August 2019) had the following conditions
  - Assumed all ECCS valves remain closed due to IAB block function above 1200 psid
  - Evaluated ECCS valves opening between 1200 psid and 1000 psid
  - All ECCS valves (RRVs and RVVs) open at the same time

Revised FSAR Analyses

- Impacted FSAR Sections
  - FSAR 6.2 Peak CNV Pressure
  - FSAR 15.6.5 Loss of Coolant Accidents
  - FSAR 15.6.6 Inadvertent Operation of ECCS
- Revised assumptions
  - Assumes all ECCS valves remain closed due to IAB block function above 1300 psid
  - Evaluated ECCS valves opening between 900 and 1000 psid
- Revised analysis results submitted in September 2019 and reviewed in NRC October audit in Corvallis
- DCA Revision 4, including revised FSAR analysis results, formally submitted January 2020

Revised FSAR Analyses

- Impact of revised IAB threshold and release pressures
  - FSAR 6.2 limiting CNV peak pressure (inadvertent RRV opening)
    - » Second RRV opens at 1000 psid (before RVVs – open at 900 psid)
      - » Slightly higher CNV pressure due to larger mass release before RCS and CNV pressures equalize
  - FSAR 15.6.5 LOCA - limiting water level above fuel
    - » Staggered IAB opening pressures not limiting
    - » IABs release at 900 psid – lower minimum water level above fuel
  - FSAR 15.6.6 – inadvertent ECCS valve (RVV) opening
    - » Not sensitive to different ECCS valve opening pressures – MCHFR occurs before 0.5 seconds
    - » No change in MCHFR due to IAB threshold change – other model changes, including more conservative reactivity feedback and flow resistance, impacted MCHFR
    - » Verified 1300 psid threshold adequate to maintain remaining ECCS valves closed until reactor trip

Updated Analysis Results

Event / Acceptance Criteria	DCA Rev 3 Results	Updated DCA Rev 4 Results	Comments
Peak CNV Pressure (RRV Opening) CNV Design Pressure - 1050 psia	986 psia	994 psia	Change in peak pressure due to staggered IAB release (2nd RRV at 1000 psid, RVVs at 900 psid)
LOCA - Minimum Water Level Above Top of Active Fuel	1.7 ft	1.5 ft	Change due to lower IAB minimum release pressure 900 psid
Inadvertent ECCS valve opening – MCHFR limit 1.13	1.41	1.32	Change due to model revisions not IAB threshold change

Operational IAB Testing

- Operational testing established to assure safety functions are met
  - Technical Specification 3.5.1 requires surveillances in accordance with the inservice testing (IST) program
  - FSAR 3.9.6.4 defines IST Alternate Authorization
    - Preservice testing for all RVV and RRV IABs on all NPMs
      - » Verify IAB minimum analyzed closing threshold pressure and opening release pressure is within analyzed range
    - Operational testing
      - » First refueling of first NPM – all IABs tested (same as preservice test)
      - » Second NPM refueling – one RRV IAB and one RVV IAB tested
      - » Subsequent outages – IAB test frequency determined by ASME OM Mandatory Appendix IV
  - FSAR 3.9.6.1 Functional design and qualification per ASME QME-1

Conclusions

- CNV peak pressure results slightly more limiting (8 psi) due to explicit evaluation of ECCS valves opening at different IAB release pressures
- LOCA minimum water level above fuel results slightly more limiting (~0.2 feet difference) due to lower minimum IAB release pressure of 900 psid
- Inadvertent ECCS valve opening MCHFR slightly more limiting due to evaluation of error corrections and more bounding model input, not from IAB change
- All updated event results demonstrated margin to acceptance criteria
- Preservice and operational testing programs established to assure safety functions maintained

Acronyms

ASME	American Society of Mechanical Engineers
CHF	critical heat flux
CNV	containment vessel
DC	direct current
DCA	Design Certification Application
ECCS	emergency core cooling system
FSAR	Final Safety Analysis Report
IAB	inadvertent actuation block
IST	inservice testing
LOCA	loss-of-coolant accident
MCHFR	minimum critical heat flux ratio
NPM	NuScale Power Module
OM	Operations and Maintenance of Nuclear Power Plants
psi	pounds per square inch
psid	pounds per square inch differential
QME-1	Qualification of Active Mechanical Equipment Used in Nuclear Facilities
RCS	reactor coolant system
RRV	reactor recirculation valve
RVV	reactor vent valve

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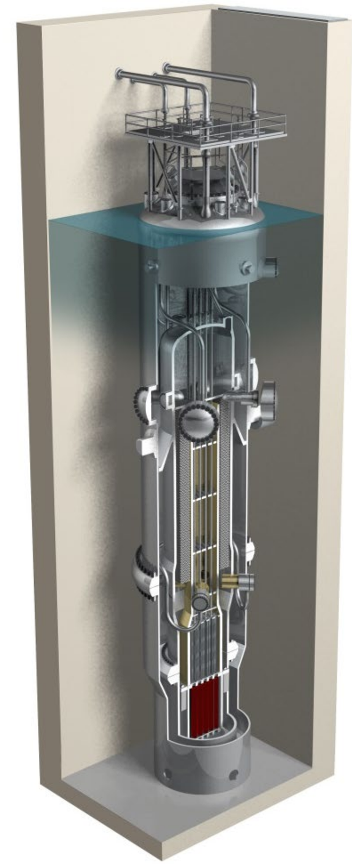

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Inadvertent Actuation Block (IAB) Valve

NuScale Design Certification Application

ACRS Subcommittee Meeting
February 4, 2020

NRC Staff Review Team

- Technical Reviewers:
  - Thomas G. Scarbrough
- Project Management:
  - Marieliz Johnson

NuScale ECCS Valve System

- NuScale reactor uses a first-of-a-kind (FOAK) valve system design for the Emergency Core Cooling System (ECCS).
- ECCS design includes 3 five-inch Reactor Vent Valves (RVVs) and 2 two-inch Reactor Recirculation Valves (RRVs) that open to allow natural circulation for emergency core cooling.
- Each RVV and RRV consists of a main valve, inadvertent actuation block (IAB) valve, solenoid trip valve, and solenoid reset valve connected by small hydraulic tubing.
- During 2019, NuScale conducted ECCS Valve Design Demonstration Testing to satisfy 10 CFR 52.47(c)(2) and 50.43(e) for this FOAK design feature.

NRC Regulatory Requirements for New Design Features

- 10 CFR 52.47(c)(2) requires, in part, that a DCA for a reactor design that uses passive means to accomplish its safety functions must provide an essentially complete design and must meet 10 CFR 50.43(e).
- 10 CFR 50.43(e) requires, in part, that a DCA for a reactor design that uses passive means to accomplish its safety functions will be approved only if the performance of each design safety feature has been demonstrated through either analysis, test programs, experience, or a combination thereof, with a requirement that sufficient data exist on design safety features to assess analytical tools for safety analyses over a sufficient range of normal operating and transient conditions, and accident sequences.

NRC Staff's Evaluation of NuScale ECCS Valve System

- NRC staff reviewed the ECCS valve design description in the NuScale DCA, including meetings with NuScale personnel and audits of design documentation.
- NRC staff conducted on-site audits at Target Rock of ECCS Valve Design Demonstration Testing to review test plans, procedures, results, and corrective action, including walkdowns of test setup and observations of testing activities.
- NRC staff conducted vendor inspections of NuScale QA oversight of Target Rock activities.
- NRC audit report (ML19340A019) dated 12-19-2019 describes NRC review of ECCS Valve Design Demonstration Testing to satisfy 10 CFR 52.47(c)(2) and 50.43(e) with follow-up items.

NuScale ECCS Valve Design Demonstration Testing Summary

- NuScale conducted ECCS Valve Design Demonstration Testing with two-inch globe valve RRV mock-up in 2019 at Target Rock facility.
- Test objectives were to demonstrate:
  - Main valve functionality at operating temperature and pressure fluid conditions
  - IAB valve functionality at operating temperature and pressure fluid conditions
  - IAB valve and trip line functionality at operating chemistry fluid conditions, including boric acid solution
- Test conditions included saturated steam, subcooled water, and nominal temperature water with pressures ranging from 30 to 1850 psig.

NuScale ECCS Valve Design Demonstration Testing Results

- In August 2019, NuScale reported that the ECCS Valve Design Demonstration Testing had been completed.
- Testing revealed that certain aspects of the original ECCS valve system design did not perform as expected.
- Several design changes to the ECCS valves were necessary based on test results including:
  - Reduction in main valve control chamber orifice diameter
  - Reduction in IAB valve port inlet diameter
  - Reduction in IAB valve shim size
- Operating range for IAB valve will be expanded from 1100 psid +/- 100 psid to +/- 200 psid.
- Two-inch RRV test results extrapolated to help demonstrate five-inch RVV performance.

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NuScale ECCS Valve Final Test Report (FTR)

- NuScale FTR concludes ECCS Valve Design Demonstration Testing was successful with the following key observations:
  - Ratio of inflow and outflow for main valve is critical to allow proper depressurization of main valve control chamber.
  - IAB valve threshold and release pressure is significantly impacted by fluid temperature.
  - Location and size of minimum outflow area downstream of main valve control chamber is critical to IAB valve threshold pressure.
  - Dynamic performance limitations of IAB valve do not allow threshold and release pressures to be within a narrow range such that IAB valve pressure range was increased to 900-1300 psid to allow for satisfactory IAB valve performance under test fluid conditions.

NRC Audit Report Conclusions

- NuScale ECCS Valve Design Demonstration Testing revealed need to modify valve design and performance assumptions.
- Lessons learned need to be addressed in ASME QME-1 Standard qualification and inservice testing (IST) programs.
- ECCS performance attributes in DCA not consistent with results of ECCS Valve Design Demonstration Testing.
- NuScale will update DCA and applicable design documents to reflect design modifications and performance characteristics.
- COL holder will address lessons learned from ECCS Valve Design Demonstration Testing for RVV and RRV qualification.
- Upon follow-up item resolution, NRC staff will conclude its review of NuScale ECCS Valve Design Demonstration Testing to satisfy 10 CFR 52.47(c)(2) and 50.43(e).

NRC Audit Follow-Up Items

- Item 1: NuScale will update IST program description for ECCS valves in DCA, including a request to depart from ASME OM Code requirements for IAB valve testing.
- Item 2: NuScale will update DCA provisions for ECCS valve performance (including IAB valve) to reflect lessons learned from ECCS Valve Design Demonstration Testing.
- Item 3: NuScale will update applicable ECCS valve design documents to reflect design modifications.

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Current Status

- On 10-24-2019, NuScale submitted a letter with the status of its actions to address ECCS valve audit follow-up items (ML19297H199).
- On 11-13-2019, NuScale submitted a letter with its draft update to IST program description, including alternative request, for DCA Revision 4 (ML19317E531).
- Upon notification of completion, NRC staff will verify that audit follow-up items have been resolved.
- During next SER phase, NRC staff will finalize its review of the ECCS Valve Design Demonstration Testing Program to satisfy 10 CFR 52.47(c)(2) and 50.43(e) in demonstrating safety features of NuScale ECCS valve system to support NuScale DCA.

Questions?

January 31, 2020

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Subcommittee Presentation: NuScale FSAR Chapter 8 – Electric Power," PM-0220-68597, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) Subcommittee Meeting on February 4, 2020. The materials support NuScale's presentation of electric power.

The enclosure to this letter is the nonproprietary presentation entitled "ACRS Subcommittee Presentation: NuScale FSAR Chapter 8 – Electric Power," PM-0220-68597, Revision 0.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Marty Bryan at 541-452-7172 or at mbryan@nuscalepower.com.

Sincerely,

Zackary W. Rad
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Enclosure: "ACRS Subcommittee Presentation: NuScale FSAR Chapter 8 – Electric Power," PM-0220-68597, Revision 0

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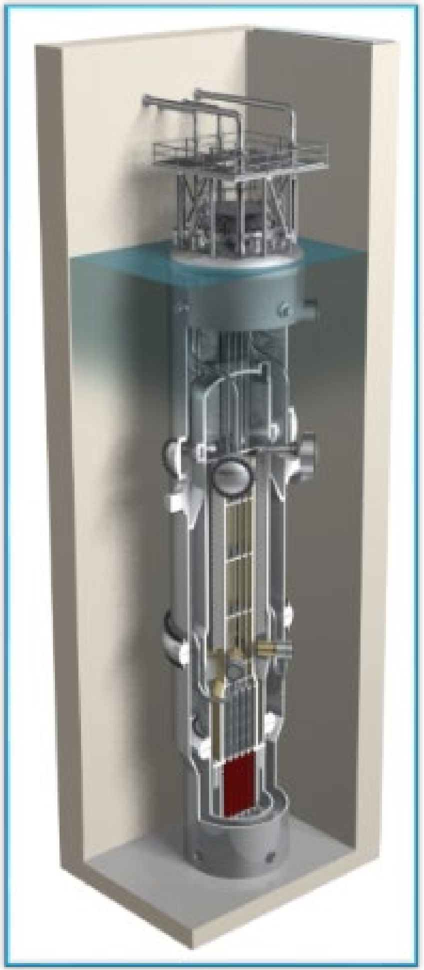
ACRS Subcommittee Presentation



NuScale FSAR

Chapter 8

Electric Power

February 4, 2020

PM-0220-68597
Revision: 0

Presenters

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Electrical Engineering Supervisor

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Licensing Project Manager

NuScale Electric Power

Safety of the NuScale Plant Design is

Not Dependent on Electric Power

- Based on the design that is consistent and compliant with NRC-approved Topical Report.
  - "Safety Classification of Passive Nuclear Power Plant Electrical Systems," TR-0815-16497-P-A, January 2018.
    - Includes Conditions of Applicability specified to justify classification of plant electrical systems as non-Class 1E.
    - Chapter 1 of Safety Evaluation Report (SER) documents NRC staff review and conclusion that design is consistent with Topical Report and appropriately considers electric power non-Class 1E as described in the Design Certification Application (DCA), Part 2, Tier 2.

NuScale Electric Power

- Design Conformance with Topical Report Documented in Part 2, Tier 2, Chapter 8:
  - Table 8.3-9, "FSAR Cross Reference for the Conditions of Applicability and NRC SER Limitations and Conditions for TR-0815-16497-P-A," and
  - Table 8.3-10, "FSAR Cross Reference for the EDSS Augmented Provisions in TR-0815-16497-P-A"
- These FSAR tables map the Topical Report conditions to FSAR locations that demonstrate conformance.
  - Design features
  - Design requirements

Tables includes specific FSAR locations that demonstrate alignment with the five SER conditions

Acronym

EDSS	Highly reliable DC power system

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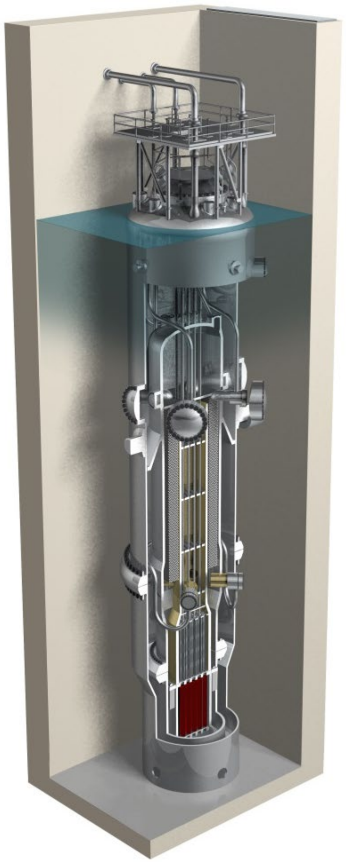

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NRC Staff Presentation on

Applicability of Topical Report, "Safety Classification of Passive Nuclear Power Plant Electrical Systems"

NuScale DCA Review, Phase 5

February 4, 2020

Purpose

- To brief the ACRS SC members on how NuScale DCA incorporated the Conditions and Limitations of staff's safety evaluation report for Topical Report, TR-0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems."

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Background/Timeline

- The NRC staff issued its safety evaluation for NuScale TR-0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems," (TRSE) in June 2017 (ML17170A201). The staff approved the TR with 6 conditions and limitations.
- The staff briefed the ACRS (FC) in July 2017.
- In its July 26, 2017 letter, the ACRS recommended the TR to be approved only for NuScale design and provided additional comments (ML17205A380).
- The staff responded to the ACRS's comments in September 2017 (ML17221A058).
- The NRC docketed the "-A" version of the TR in February 2018 (ML18054B607).

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Applicability of TR to NuScale DCA

- NuScale TR-0815-16497 is incorporated-by-reference in the DCA,
- In January 2018, the staff requested NuScale (RAI 9359) to describe how its design meets the 5* conditions of the TRSE, and asked NuScale to update the DCA accordingly (ML18028A008).
- In March 2018, NuScale responded to the RAI (ML18086B096)
- In the following slides, the staff will describe how NuScale addressed the TR’s 5 conditions in the DCA.

*****_The staff’s TRSE contained 6 conditions and limitations. Per ACRS recommendation, the staff revised its SE to approve the TR only to be applicable to NuScale design. As such, in the “-A” version of the TR, and the associated staff SE, the number of conditions and limitations was reduced to 5._

Highly Reliable dc Power System (EDSS)

- Non-Class 1E dc power system to which augmented design, and quality assurance (QA) provisions are applied.
- two dc subsystems:
  - (1) the EDSS-common (EDSS-C) plant subsystem serves plant common loads that have functions that are not specific to any single NuScale Power Module (NPM) and
  - (2) the EDSS-module-specific (EDSS-MS) plant subsystem consists of up to 12 separate and independent dc electrical power supply systems, one for each NPM.
- Based on its review of the TR-0815-16497 to the DCA, the staff finds that the onsite and offsite electric power systems do not warrant a Class 1E designation and they are not safety-related systems.
  - Staff approved NuScale's Exemption Request from the requirements of GDC 17

Condition 4.1

- Address the guidance in Regulatory Guide (RG) 1.155, “Station Blackout,” Appendix A, “Quality Assurance Guidance for Non-Safety Systems and Equipment,” in sufficient detail to enable the NRC staff to verify that the relevant QA program would meet or exceed the guidance in RG 1.155.

- Disposition of Condition 4.1:
  - DCA Part 2, Tier 2, Section 8.3.2.2.2, states that an augmented quality assurance (QA) program is applied to the EDSS and the program meets the QA provisions of RG 1.155.
  - The staff finds the disposition acceptable because DCA Part 2 addresses the QA provisions of RG 1.155, Appendix A.
  - Therefore, the NRC staff finds that the applicant met Condition 4.1. FSER Chapter 8 documents the staff’s evaluation supporting this conclusion.

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Condition 4.2

- Confirm that the valve-regulated lead-acid (VRLA) batteries and their structures are seismic Category 1. To provide reasonable assurance that the VRLA batteries will perform as intended, an applicant that references the TR shall provide a COL action item to support that the VRLA batteries and their structures are seismic Category 1. A qualification testing plan includes environmental and seismic qualification and a technical functional requirement for VRLA batteries to show they can perform as intended.

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Condition 4.2 (Cont'd)

- Disposition of Condition 4.2:
  - DCA Part 2, Tier 2 states:
    - the EDSS is classified as seismic Category I
    - EDSS design accommodates the effects of environmental conditions
  - The staff finds the disposition acceptable since the EDSS complies with GDC 2 as the EDSS components are located in seismic Category I structures and the EDSS has augmented design requirements for environmental qualification.
  - Therefore, the NRC staff finds that the applicant met Condition 4.2. FSER Chapter 8 documents the staff's evaluation supporting this conclusion.

Condition 4.3

"Demonstrate that operator actions are not necessary to ensure the performance of safety related functions for any postulated DBE (i.e., the design does not include Type A variables as defined in IEEE Std. 497 2002, as modified in RG 1.97, Regulatory Position C.4), as presented in Chapter 15 of its FSAR and the human factors analysis in Chapter 18 of its FSAR."

- Chapter 15 DCA review concluded that no operator actions are necessary to successfully mitigate a DBE out to 72 hours.

Condition 4.4

"Evaluate the frequency for which a combination of an AOO and an actuation of the NuScale ECCS is realistically expected to occur, and show that such a combination of events is not expected to occur during the lifetime of the module."

- Means of uncontrolled RCPB coolant loss are:
  - LOCA
  - Inadvertent operation of an ECCS valve
    - Operator error
    - Valve failure
  - AOO coincident with loss of nonsafety-related DC power

Condition 4.4 (Cont'd)

- LOCA is classified as a postulated accident not expected to occur in the lifetime of the plant (SRP 15.0)
- Inadvertent operation of an ECCS valve
  - Operator error opening the ECCS valves is considered an AOO but the IAB valves prevent RPV coolant discharge
  - Mechanical failure of the safety-related ECCS valve is not expected to occur in the lifetime of the plant.
- AOO coincident with the loss of nonsafety-related DC power
  - Loss of the highly reliable non-safety-related DC power (EDSS) is not realistically expected to occur in the lifetime of the module based on design attributes
  - For a longer term loss of AC power, DC power is retained long enough such that the subsequent opening of the ECCS valves does not challenge containment integrity

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Condition 4.5

"Demonstrate that the reactor can be brought to a safe shutdown using only safety-related equipment in the absence of electrical power following a DBE, with margin for stuck rods. Alternatively, an applicant referencing this TR may provide justification, for NRC review, for a less restrictive condition."

- Shutdown may not be maintained assuming DBE conditions at or near EOC conditions
- NuScale provided a less restrictive condition demonstrating the SAFDLs are not exceeded on a potential return to power assuming a stuck rod
  - Less restrictive SAFDL condition is consistent with SECY-18-0099
- Applicant's Chapter 15 analysis demonstrated, and the staff agrees, the SAFDLs are met.