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

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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665TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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OPEN SESSION

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WEDNESDAY

JULY 10, 2019

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ROCKVILLE, MARYLAND

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The Advisory Committee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B10, 11545 Rockville Pike, at 8:30 a.m., Peter Riccardella, Chairman, presiding.

COMMITTEE MEMBERS:

PETER RICCARDELLA, Chairman

MATTHEW W. SUNSERI, Vice Chairman

MICHAEL L. CORRADINI, Chairman

DENNIS BLEY, Member

CHARLES H. BROWN, JR. Member

VESNA B. DIMITRIJEVIC, Member

WALTER L. KIRCHNER, Member
JOSE MARCH-LEUBA, Member
DAVID PETTI, Member
JOY L. REMPE, Member
GORDON R. SKILLMAN, Member

ACRS CONSULTANT:
STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:
MICHAEL SNODDERLY

ALSO PRESENT:
CLINTON ASHLEY, NRO
RUFINO AYALO, NuScale
ANTONIO BARRETT, NRO
BRUCE BAVOL, NRO
AMBER BERGER, NuScale*
MARTY BRYAN, NuScale
KEVIN COYNE, NRO
TIM DRZEWIECKI, NRO
YUSUF FARAWILA, NuScale*
RANI FRANOVICH, NRO
ANNE-MARIE GRADY, NRO
SYED HAIDER, NRO

STEVE HAMBRIC, NRC
OLIVIA HAND, NuScale*
JOHN HONCHARIK, NRO
BHAGWAT JAIN, NRO
NADJA JOERGENSEN, NuScale
REBECCA KARAS, NRO
SELIM KURAN, NuScale*
RENEE LI, NRO
TIM LUPOLD, NRO
GREG MAKAR, NRO
MEGHAN McCLOSKEY, NuScale*
RYAN NOLAN, NRO
REBECCA NORRIS, NuScale
MATTHEW PRESSEN, NuScale
ZACKARY RAD, NuScale
HANNAH ROOKS, NuScale*
SUJIT SAMADDAR, NRO
PRAVIN SAWANT, NuScale*
THOMAS SCARBROUGH, NRO
JEFF SCHMIDT, NRO
RAY SKARDA, RES
OMID TABATABAI, NRO
BOYCE TRAVIS, NRO
ANDREA VEIL, ACRS
MARIELIZ VERA, NRO

ROBERT WEISMAN, OGC

BRIAN WOLF, NuScale*

YUKEN WONG, NRO

PETER YARSKY, RES

*Present via telephone

T-A-B-L-E O-F C-O-N-T-E-N-T-S

P R O C E E D I N G S

8:37 a.m.

CHAIR RICCARDELLA: The meeting will now come to order. This is the first day of the 665th meeting of the Advisory Committee on Reactor Safeguards. I'm Peter Riccardella, Chairman of the Committee.

The ACRS was established by the Atomic Energy Act and is governed by the Federal Advisory Committee Act, or FACA. The ACRS section of US NRC public website provides information about the history of the ACRS and provides FACA-related documents, such as our charter, bylaws, Federal Register notices for meetings, letter reports, and transcripts of all full and subcommittee meetings, including all slides presented at the meetings.

The Committee provides its advice on safety matters to the Commission through its publically available letter reports. The Federal Register notice announcing this meeting was published on July 1, 2019 and was revised on July 8, and provided an agenda and instructions for interested parties to provide written documents on request, or request opportunities to address the Committee as required by FACA.

In accordance with FACA, there is a Designated Federal Office, DFO, for today's meeting. The DFO for this meeting is Mr. Michael Snodderly.

(Off-microphone comments.)

CHAIR RICCARDELLA: Okay, the DFO for this meeting is Mr. Michael Snodderly. During today's meeting, the Committee will consider the following: NuScale's design certification application Chapters 3, 6, 15, 20, and the Stability Topical Report, and two, preparation of ACRS reports.

As reflected in the agenda, portions of the sessions on NuScale design certification application and stability report made be closed in order to discuss and protect information designated as sensitive or proprietary. There is a phone bridge line. To preclude interruption of the meeting, the phone will be placed in a listen-only mode during the presentations and Committee discussions.

We have received no written comments or requests to make oral statements from members of the public regarding today's sessions. There will be an opportunity for public comment as we have set aside ten minute in the agenda for comments from members of the public attending or listening to our meetings. Written comments may be forwarded to Mr. Snodderly,

the DFO.

A transcript of the open portions of the meeting is being kept, and it is requested that the speakers use one of the microphones, identify themselves, and speak with sufficient clarity and volume that they can be readily heard. And I would also ask everyone to silence cellphones and other devices to avoid interruption of the meeting.

And with that, I will turn the meeting over to Michael Corradini.

MEMBER CORRADINI: Thank you, Mr. Chairman. So to remind the members where we are, this is the final set of chapters from the draft -- I'm sorry, from the DCA, as long, as well as the draft safety evaluation reports. So we're going to go through first the Stability Topic Report, and then we will proceed on to the other chapters.

PARTICIPANT: Anyone here?

PARTICIPANT: Yes, NuScale Corvallis can hear you.

PARTICIPANT: Okay.

(Off-microphone comments.)

MEMBER CORRADINI: All right, so let's try again. So today we have a number of things on the agenda. We first want to talk about the topical

report which was submitted by NuScale and the associated SE, and then we'll follow up with the final four chapters of the DCA Revision 2.

Okay, so why don't I turn this over to Zack. I'm sorry, excuse me, I'm going to turn it over to Matthew from NuScale, excuse me. Matthew.

MR. PRESSEN: Quite all right. Thank you all and good morning, I am Matthew Pressen, Licensing Specialist with NuScale Power and Project Manager for this topical report.

If we could move to the next slide. We are here today to discuss the evaluation methodology for stability analysis of the NuScale power module. Slide 2, sorry.

MEMBER REMPE: You guys, is this the only way we can see the slides? We're going through Internet Explorer. Did they get downloaded to this computer?

MEMBER CORRADINI: No, it's being run by the control room.

MEMBER MARCH-LEUBA: But the control room needs to download the file, not run it on Acrobat.

MEMBER CORRADINI: If we can suffer through this, let's just keep on going.

MEMBER BLEY: Well, the two side screens

are working.

MEMBER CORRADINI: We have hard copy.

MEMBER SUNSERI: Is your mic close to you? Because with this background noise, it's hard to hear. You're quiet.

MR. PRESSEN: So presenting today will be Dr. Yusuf Farawila over the phone. He is the NuScale System Thermal Hydraulic Stability Expert. He will be supported by myself, Licensing, and Zack Rad. With that, we will pass this presentation off to Dr. Farawila. And if we could move to slide 3.

MEMBER SUNSERI: Ask your presenter to speak.

MR. PRESSEN: Yeah, can Corvallis hear us? So looks like the Corvallis line cannot hear us anymore.

MEMBER SUNSERI: Did they mute as requested? They need to unmute.

MR. PRESSEN: They did, but they are not hearing us.

CHAIR RICCARDELLA: Let's take a five-minute recess. We'll reconvene at ten to nine.

(Whereupon, the above-entitled matter went off the record at 8:45 a.m. and resumed at 8:52 a.m.)

MR. FARAWILA: Good morning, everyone,

this is Yusuf Farawila here from NuScale. And I'm going to try to, we are on slide 3, so we are going to try to do this as fast as we can, maybe we can make up some of the lost time.

Our agenda today is just to make an introduction in the new methods and introduce our stability solution and provide and for us to start work that led us to that point. Both critical, numerical, and experimental benchmarking. Some of that will sure have to wait till the closed session. And we'll introduce also our procedure methodology and take your questions.

Slide 4, our main message is that the NuScale Power model design was investigated for stability and found to be stable in its entire range of normal operation. Even outside of normal operation, the reactor can be de-stabilized when the riser flow is boiling. So we have identified stability thresholds.

However, this unseasoned flow is limited by non-linear effects so that even operations of that type when the stability threshold is crossed is also denied.

That stability threshold is protected by scram, a full loss of riser in with the cooling. And

that constitutes region exclusion. So not a detector of resolution type. Because scram was already there for other reasons other than stability, there is no action required for cooling in the hardware that is indicated (inaudible) instability.

PARTICIPANT: Can't hear anything.

MR. FARAWILA: These conclusions are based on extensive analysis with first principles, experimental and computational studies and we'll try to include some of that in the next slide.

Go to slide 5.

(Simultaneous speaking.)

MR. FARAWILA: We looked at the literature and found out that (inaudible) stability was reported in the literature. And you can see certain types there.

PARTICIPANT: I'm sorry. We're being asked to pause and have everyone go on mute.

MR. FARAWILA: Okay, all right. All right, this is Yusuf Farawila again on slide 5. Presenting an outline of an instability --

(Simultaneous Speaking)

MR. FARAWILA: -- literature. You see the location of the heater and the cooler can affect the stability, you can see four regimes there. Only the

first one could go unstable. And our modeling looks like one of the most stable configurations. But --

(Off-microphone comments.)

MEMBER CORRADINI: So if could make a suggestion. My suggestion to the Chairman is we essentially adjourn till 9:15. We get this settled. Keep Farawila on the line as the test case and get it cleared up and not come back until we're --

MEMBER REMPE: And a test case with a staff member acting as the public would be a good idea too.

MEMBER CORRADINI: That's fine. So that's my suggestion so we can clear this up, otherwise we're just going to be fussing around.

CHAIR RICCARDELLA: Okay, the meeting is recessed until 9:15, and hopefully we will be able to reconvene at that time.

MEMBER CORRADINI: I volunteer to be the public if Dr. Farawila --

PARTICIPANT: Okay, NuScale Corvallis is back on the line and we're being requested to ask everyone else to go on mute. We think we heard some things from the meeting.

CHAIR RICCARDELLA: This is Pete Riccardella, ACRS Chairman.

PARTICIPANT: But now we're not hearing anything again.

CHAIR RICCARDELLA: Okay, so tell her we're going to recess the meeting until 9:15 East Coast time. And we'll convene at that time, and hopefully everything will be working.

(Whereupon, the above-entitled matter went off the record at 8:57 a.m. and resumed at 9:30 a.m.)

MR. FARAWILA: All right, this is Yusuf Farawila again, sorry for the problem. You are on slide 4.

Presenting you with the main message, and that is the NuScale Power module has been examined for instability and how to be stable in the entire range of its normal operation. Also outside of normal operations, the reactor can be destabilized if the riser flow (inaudible).

But even then, this unstable flow oscillation amplitude is limited by nonlinear effect and there is no critical heat flux ratio that is, could actually improve. So that kind of instability is also benign.

That stability threshold, which is voiding the riser, is protected by scram, which is actuated upon loss of inlet risers of cooling with a margin.

So that's contextually equivalent to the traditional region exclusion. The region is defined only that line of the riser inlets of cooling, not a two-dimensional one like BWRs.

And because this protection action is there for other purposes, it's not made specially for stability. So there is no hardware solution that is required for stability. So the way we reached these conclusions using first principles for that experiment and computation studies are prudent.

We are in slide 5. I will present you with an example to be found in the literature where metric instabilities were reported in configurations of where the cooling the heat sink and heat source are placed around the loop. And that affects the stability. Only the first kind with the heater in the extreme bottom and the cooler in the very top is the most unstable and instabilities were reported.

And our model is more like the last figure there with a cooler on the upper side and the heater is on the lower position of the other side. But we could not just take this for granted and we went ahead and made this extensive evaluation.

You go to slide 6, and we see how we investigated this starting from the gray area. We

just had a design and operational domain. We are fortified with first principles and theory and experience. We have been done a preliminary purge to start with. And then we moved into constructed models for the main code, which is code 10.

And on both sides of that box you will see that we also did independent models using reduced order modeling and also conducted experimental investigation.

So using thus qualified code to (inaudible) with a steady state of deviations and stability during trending. And we identified where the threshold is and reached the conclusion that the model is stable within operating domain. So that comes, brings us to the bottom box, where we identified the stability solution area, just simply protecting the riser's cooling margins and the hardware is already in place.

Go to slide 7. The theoretical investigation started with the extensive first meeting. And we did a scope and review of thermal-hydraulic instabilities to see which ones applied. And we identified what possible mechanisms of instabilities are, we used direct first principles to study parts of the model, like riser-only mode, what

will happen. And when you put the steam generator, what kind of trends this can be for stability.

And so we could also show, and that part was kind of raised in the Subcommittee meeting, that we could also show that steam generator dynamics are decoupled from the stability of that primary loop. And I think that's important for -- important outcomes from our theoretical investigations.

And these theoretical investigations also used to inform the design of the stability experiment. Mentioned in the third, there's always a subjective judgement which a phenomena of how we rank each phenomena are ranked medium or less. So we, as a rule, rated all medium-ranked phenomena in modeling as if they were highly ranked.

Go to slide 8. Just giving you an outline of what we mean by instability, what are the required conditions from first principles for a system to go unstable. First, we have to have to have -- identify negative feedback processes are there. Positive feedbacks are not allowed.

So the negative feedbacks are present, but they need to be delayed and they need to be sufficiently strong. Because when a negative delay, a negative feedback is delayed, then the system can

overshoot in the other direction. And if that's sufficiently strong, we get oscillatory instability.

So in our particular case, the feedback is negative because any perturbation increase in core flow will give you a force that will reduce it, essentially defeating the initial perturbation. But that feedback is delayed because it takes time for this perturbation, like a changed temperature at the core exit, to fill the riser before it will have an impact on the (inaudible) that drives the flow.

So if the feedback is strong enough then that is when the quantitative analysis is required. So we need to look at all the characteristics of, you know, the riser lens and the possibility of (inaudible) change. Because if you receive a change, then you have a stronger response that may become unstable and this is something -- really something actually.

Let's go to the next slide, slide 9. Just introducing you to the main stability analysis tools, PIM, which was optimistically named PIM, when means pendulum in molasses. Because this module has proven to be very stable, particularly under normal operation.

You see a picture of the module, then see

a more abstract picture, and then that's a diagram that describes the geometry of the code. The core in the middle and risers in the riser. And then in the cold leg, you see the steam generator where the second side acts as a heat sink.

We can go through this equation quickly, we don't have to recite them in slide 10. So the model equations are essentially having the thermal-hydraulic conservation equations for liquid and vapor mass balance and mixture momentum and energy conservation, also assuming that the vapor is always saturated.

So go to the next one, slide 11. We applied (inaudible) kinetics in the core. So because the core is rather small and very much in the bottom, so there are no three-D effects that are of interest. And we also modeled the reactivity input to that -- kinetics, which come from the thermal-hydraulic model and come from also the Doppler Effect.

So there's always this interaction between the thermal-hydraulics and the dynamics in the core. Go to the slide 12. There are other models also that need to be implemented in order for all these phenomena to get coupled. And that's heat conduction in the fuel rods, and we have heat transfer models.

Particularly important is what happens in the steam generator.

Slide 13. For any thermal-hydraulic model like that, (inaudible) relations and correlation for frictional pressure drop. We used the drift flux formulation for the flip between the vapor and liquid stages. Remember here that we are mostly (inaudible). There is the possibility of subcooled boiling, but since the code is applicable to beyond the normal operation, we allow for vapor formation at any amount.

We can go with the riser to any amount. So a general model for two-phase flow is required and that's what is there. And so we have a equilibrium evaporation and condensation model to account for all of that. And the rest of the properties for water and other physical materials and all the reactivity coefficients that also should come from approved neutronics codes.

Important to note that when you model, you have to essentially not model everything. Some of the things here are not modeled. We are not modeling the pressurizer because of the permanent effect on the momentum balance. And we make assumptions also about heat transfers through the riser wall if we assume it’s (inaudible).

And so all these assumptions that we use, and for the -- to model something, it's either shown to be concerned with its full stability, or the auxiliary (inaudible).

Next slide, 14. We examined also the effect of the flow stability in the steam generator tubes. Because now you look at the secondary side and see you have boiling channels inside the steam generator, and will these be unstable or not.

And we could perhaps talk more about that in the closed session, but the flow in these tubes is subject to this fluid instability. And we have experiments, and these experiments succeed to demonstrate that these instabilities are possible.

However, the most important part is that the observations in different tubes. You have over a thousand tubes. They are not phase-locked, so with random phase or out of phase, they do not feed back to the primary.

Actually, the primary does not feed back to that secondary either, because you cannot have a perturbation in the flow in the cold leg in one direction that affects two different tubes in opposite directions. So that's not also possible.

That makes a case for having no impact on

the primary flow for any possible instabilities that can happen in the steam generator tubes.

PARTICIPANT: Before you go, when you're ready to go to the next slide, Matthew says there's a question.

MR. FARAWILA: Okay, I'll take the question.

MEMBER MARCH-LEUBA: That's my cue --

MR. FARAWILA: Slide 14.

MEMBER MARCH-LEUBA: And you will relay it to him, right? The third bullet says that the experiments demonstrate for oscillations and they're generally low flow. However, with -- is this a true statement? Because we've seen other data that indicates that this unstable in the full operating --

(Off-microphone comments.)

MEMBER MARCH-LEUBA: There. If you want to simplify the question is do the experiments demonstrate instabilities outside free flow?

MR. FARAWILA: I'm not sure I understood the question, but it may be a question about when -- okay. I'm going to ask you what the question is. When is the steam generator tube stable or unstable?

MEMBER MARCH-LEUBA: Yeah.

MR. FARAWILA: So basically, the way we set

it in general, is that the question?

PARTICIPANT: Yeah.

MR. FARAWILA: Okay.

PARTICIPANT: That's the question.

MR. FARAWILA: Hold on.

(Off-microphone comments.)

MR. FARAWILA: Can we defer the answer to that question to the closed session?

MEMBER MARCH-LEUBA: Okay. Okay, so we'll wait for the closed session.

MR. PRESSEN: And yes, we do have a number of slides covering the experiments in the closed session, so.

MR. FARAWILA: Let me speak in the general sense right now and tell you that for basically wave oscillations in a boiling channel, you need to have a generally high power to flow ratio. So everything else being equal.

All right, may I proceed to slide 15?

(Off-microphone comments.)

(Pause.)

MR. FARAWILA: All right, this is Yusuf Farawila again in Corvallis. We are on slide 15, and we wanted to present in this slide how PIM is used to calculate stability. So what we do is we perturb a

(inaudible) solution and see how the cold responds. So for several power levels and different modes of operation at different exposures, things like this, we --

PARTICIPANT: We want to move to slide 18.

MR. FARAWILA: All right. Oh right, slide 18, that's summary and conclusions.

All right, so to conclude, the stability of the nuclear module was evaluated using a dedicated code and supported by first principles and experimental data benchmarking, and was found to be unconditionally stable with the normal operation. And the safety boundary was identified and predicted. And best protection is a scram when the risers' subcooling margin is reached within a margin.

And that essentially concludes the presentation. And if you have any questions.

MEMBER KIRCHNER: I have a question. Have you used this set of models to look at ATWS and what is the impact of an ATWS on the stability of the system?

MEMBER MARCH-LEUBA: While you type it in, he means a beginning of cycle ATWS with a very low density of activity coefficient.

MR. FARAWILA: Yeah, I mean it's a

question about the anticipated plan without scram. So essentially, the way we reach the stability threshold, its protected by scram. So we do not have the scram in order to allow the oscillations to proceed. So that's considered essentially a transient without scram. And I refer to that in slide 3 or something.

I think that is in slide 4, when we stated that unstable flow oscillations' amplitude is limited by non-linear effects. And their amplitude does not result in any deterioration of the critical power performance. And so that's considered an ATWS because the real system with a scram, and the code did not scram.

MEMBER KIRCHNER: Well, that doesn't quite answer the question.

MR. FARAWILA: Yeah, there was that question about the moderator feedback, the reactivity feedback. In the beginning of cycle when the moderator feedback, it's least stabilizing with specific oscillation for end of cycle, when the moderator coefficient is very negative. In some calculations we don't get any instabilities.

Or if we do get instabilities when you reach highly voided riser state, the outcome is the same because the stabilization is really in the

hydraulics. Because the swing of oscillations, in the upper part of the oscillation, the void tube collapsed. Once the void collapsed in that part, the downswing cannot exceed that. So that is limiting what we can measure for the oscillation.

MEMBER KIRCHNER: Well, the BWR fleet has a problem with ATWS and operator procedures to respond to that. I, and that's for circulation.

MEMBER CORRADINI: He can't hear you, Walt. You're going to have to formulate it for --

MEMBER KIRCHNER: I'm trying to think of a shorthand way. BWRs have this kind of issue under ATWS conditions, and they are force circulation. I intuitively expect it will be amplified in a natural circulation.

MR. FARAWILA: I just mentioned that ATWS with instability is problem with BWRs. And I will be coming next month to address your committee on one of those. Fortunately, NuScale reactor is not a BWR, and if you allow instabilities to happen, they are so benign that critical heat flux ratio actually improves.

Because when you start moving the riser, the steady state average flow increases so much because you have so much more things to be had that

the operations around that, you're not consumed the increase that you get. So you get an improvement of CHFR (phonetic).

I know that this kind of improvement that we obtained from a generic combination that's not part of the approval process, but it's generic enough. And we know from first principles that this not be correct. That we do not have any challenge to CPR.

In boiling water reactors, for example, we know that severe oscillations can happen if not, if we fail to scram, and you agree to have dry-out definitely, and dry out in cyclical, dry out and waiting.

And the operator has to ask to lower the water level in order to avoid having the clad temperature exceed limits. But we have absolutely no issue in, of similar nature in the NuScale reactor.

MEMBER CORRADINI: Thank you. That's good, that's good enough. Walt, you have a follow-up? Are you okay for the moment?

MEMBER KIRCHNER: Okay for the moment.

MEMBER MARCH-LEUBA: I wanted to apologize to all the members and to the readership of this transcript that we have technological difficulties.

MR. FARAWILA: If there are no other

questions.

MEMBER CORRADINI: Tell him thank you, we'll move on to the staff.

MEMBER REMPE: I think we should also think him and NuScale for helping us deal with the situation. It's more than normal.

MEMBER CORRADINI: Okay, so we can turn to the staff who are actually in the room.

MR. FARAWILA: Well, thank you very much.

MEMBER CORRADINI: Bruce, are you guys ready? Okay.

MR. YARSKY: This is Dr. Peter Yarsky from the Office of Research staff. I suppose we'll just do this presentation in an analog format with the handouts. Today I'm here to talk to you all about the staff review of the NuScale Stability Topical Report.

If we go to slide 2, the review team included Dr. Ray Skarda, myself, and Rebecca Karas from the Office of New Reactors. If we go to slide 3, this presents an outline that is, that shows the content of what we presented at the Subcommittee meeting.

And a few bullets here are highlighted. These are items that we're going to cover in today's presentation with the full Committee. And this

includes discussion of the PIM evaluation model and the impact of secondary side instabilities on primary side stability. Other secondary side stability concerns, and finally to wrap up with Stability Topical Report conclusions.

If we go to slide 4, slide 4 presents a figure that describes the exclusion region-based long-term stability solution approach. And this was described in the earlier session by NuScale that the stability solution works by preventing instabilities. Instabilities are possible only when there's riser voiding.

So the module protection system precludes the occurrence of riser voiding by tripping the reactor if there's a loss of riser subcooling. If we go to slide 5, calculations --

MEMBER CORRADINI: Just for the members that weren't here, so the limit is what you state as the allowed operation. So once they get close to subcooling, the trip is incurred.

MR. YARSKY: I'm sorry, I don't think I understand. So if you look at the figure, I can't point to anything. So the allowable operation is in this bluish area on the figure, and that's where there is subcooled margin in the riser. So there's a trip

subpoint where once the subcooled margin is smaller than that trip subpoint, the module protection system will insert control rods and shut the reactor down.

MEMBER CORRADINI: Thank you.

MR. YARSKY: Okay, if we go to slide 5. The stability analyses are performed using the PIM evaluation model. PIM, as it was described in this morning's presentation, includes simple models for thermal-hydraulics, reactor kinetics, fuel thermal mechanic response and steam generator tube heat conduction heat transfer.

And the PIM evaluation model has been validated against tasks performed at the NIST-1 facility.

The key output from PIM, moving on to slide 6, is the decay ratio. The decay ratio acceptance criterion is determined to account for model uncertainties and model biases such that calculating a decay ratio less than the acceptance criterion indicates that the power module is stable.

The decay ratio itself is insensitive to variations in most of the important phenomena over the PIM application range. And as a part of the staff review, we confirmed that the acceptance criterion affords sufficient margin to account for the bias and

uncertainty in the method. As part of determining that bias and uncertainty, numerical effects were considered.

If we go to slide 7. This is a follow-up to the issue of secondary side instability. As part of the review, the staff determined that the secondary side flow oscillations are not safety-significant with respect to the primary side. And so while there is the possibility for secondary side instability, found that has no impact on the relevant regulatory criteria as they apply to reactor stability.

MEMBER MARCH-LEUBA: Pete, are we going to have a closed session with you? Because I cannot find the enclosed slides.

MR. YARSKY: Oh, the staff is not planning to present in the closed session on the topic of stability.

MEMBER MARCH-LEUBA: So let me ask on the secondary side, instability, generic question. Because we all saw the closed Subcommittee meeting, or some of us saw the closed Subcommittee meeting. It was your evaluation there that the secondary side would, is likely to be unstable under the complete operating domain on the, for NuScale? Basically on low flow?

Because we just heard from NuScale that the test indicated that would only happen at low flows.

MR. YARSKY: I don't know how much in open session we can talk about the specific conditions under which the tubes would be expected to be unstable. However, the staff based its review assuming that the secondary side would be unstable. So the conservative approach there of course is to assume that it would be unstable and potentially achieve like the maximum value of its oscillation.

MEMBER CORRADINI: So you didn't look for where it's unstable, you assume it's unstable and see the effect on the core.

MR. YARSKY: Right. So one thing the staff did was to compare the operating conditions expected during normal operation, including shutdown and startup maneuvering, and compare that to test conditions that were evaluated by NuScale. And from that comparison, the staff was not able to conclude that the secondary side would be stable.

So there are these, of course the potential for certain conditions for the secondary side to be unstable. In that case, the conservative approach and the approach adopted by the staff was to

assume that the secondary side would be unstable.

MEMBER MARCH-LEUBA: Is it safe to say that, at least I haven't seen a calculation by NuScale of the stability range in the secondary?

MR. YARSKY: Correct. So the staff did not pursue requesting information from the applicant to specifically demonstrate at what conditions the secondary side would be stable or unstable. The approach adopted by the staff was to just assume the secondary side would be unstable and to evaluate that instability as it relates to the primary side acceptance criteria.

MEMBER MARCH-LEUBA: And one concern, because ACRS sees all the chapters. Chapter 15?

MR. YARSKY: Yes.

MEMBER MARCH-LEUBA: One concern I personally has is the impact on thermal fatigue. If you have significant oscillations of, in temperature at the inlet of the tube because the boiling boundary goes up and down every three to ten seconds, you will have thermal fatigue. So that has to be translated to the Chapter 3 guys.

MR. YARSKY: Right so the issues associated with like the mechanical performance and thermal fatigue as a result of secondary side flow

oscillations has to be addressed by the Chapter 3 review.

MEMBER MARCH-LEUBA: And just for the record, we have NuScale saying on the record that they will fix this instability. My concern is I don't know how they're going to do it.

MEMBER BLEY: We had a fairly lengthy discussion on this. But I don't think when we did Chapter 3 we addressed this issue at all.

MEMBER MARCH-LEUBA: We did not.

MEMBER CORRADINI: I think if, so we'll come back. We can ask questions of the applicant, but I think they have to address the thermal fatigue as part of Chapter 3, and this is one of the inputs that have to be considered.

MEMBER BLEY: But what I was getting at is we didn't raise it in our letter because we did not talk about that I think that time, I don't think.

MEMBER CORRADINI: Well, and Chapter 3 is part of this letter that you're going to see, and it is mentioned. Okay.

MR. YARSKY: Right. And the cognizant reviewers in Chapter 3 that are doing the staff review for mechanical and thermal fatigue-related issues are aware of this issue and are incorporating that of

course into their review.

MEMBER MARCH-LEUBA: And now let me summarize. The staff assumes the secondary side oscillations will happen, and then propagates that stable condition in the secondary to the core. And because of the multiple (inaudible) close to a thousand do not phase lock, they're kind of --

(Simultaneous speaking.)

MR. YARSKY: So they're, the staff in conducting the review asked a number of questions about the secondary side in relation to the claim that the tubes would not phase lock.

And we weren't able to reach the same conclusion that NuScale reached that it would be impossible for the oscillations to be in phase, because we're just, we have an absence of information about the specific nature of the control system and how the secondary side will be controlled, either with different valves or different control system schemes.

So therefore when we assume the secondary side is unstable and we wanted to propagate that to the primary side, the staff requested an analysis where the secondary side tubes all oscillate in phase. Because that would produce the maximum consequence for the primary side.

MEMBER MARCH-LEUBA: And that's conservative.

MR. YARSKY: That's conservative, and it would bound any potential considerations for the final design of the secondary side control systems.

MEMBER MARCH-LEUBA: And using that very conservative assumption that the control, it fails and locks them. You still don't see propagation to the primary.

MR. YARSKY: So there of course you would see some degree of oscillation on the primary side, but it isn't significant from the standpoint of thermal margin.

MEMBER MARCH-LEUBA: And that is because of the different frequencies of oscillation?

MR. YARSKY: So that is an excellent question. The frequencies are different between the primary and secondary sides.

So in the bounding analysis that was performed to address the impact on the primary side from the secondary side, the secondary side maximum for oscillation was actually adjusted so that it was not at the natural frequency of the steam generator, but rather the natural frequency of the primary system.

So it attempts to put that maximum oscillation magnitude also at the worst possible frequency. And so it's, the analysis that was performed is a perfect storm of what could possibly be the worst condition for the primary side. And we found that it is not significant -- it doesn't significantly impact thermal margin.

MEMBER BLEY: Did the staff do that analysis or did you do it?

MR. YARSKY: No, the staff requested that NuScale perform that analysis, so that analysis was performed by NuScale.

MEMBER KIRCHNER: And then -- and so the steam generator is just treated as one node heat sink?

MR. YARSKY: No. So there are multiple, you need to treat the steam generator with multiple nodes.

MEMBER KIRCHNER: Yeah.

MR. YARSKY: Because there's different heat transfer regimes as you progress axially through the steam generator. However, PIM has the capability to treat -- I think it would be best if we wanted to talk specifically about the calculation method that it be deferred to the closed session.

I don't know how much we can talk about

what PIM does in terms of the specifics of nodalization and subnodal treatments in the open session.

MEMBER KIRCHNER: Even if they have different boiling regimes?

MR. YARSKY: I think in the open session we can say that the methodology does treat that there are different heat transfer regimes as you progress axially on the secondary side of the steam generator tube.

MEMBER KIRCHNER: Okay, that's really the heart of my question. Okay.

MR. YARSKY: And that phenomena is captured. But I think if we want to talk specifically about how that's done, it would be best to do so in closed session.

If we move to slide 8 -- oh, we have the slides now.

(Laughter.)

MR. YARSKY: So in the review of the topical report we found that while PIM is a simple model, its models are anchored to some more high fidelity upstream methods, and this improves its overall accuracy. Because the decay ratio is highly insensitive to variations in most of the important

phenomena, the overall uncertainty in the decay ratio is relatively small.

PIM predictions in the steady state and transience have been confirmed by the staff with independent trace confirmatory calculations that show very good agreement between PIM entries. And so we have concluded that PIM is acceptable for performing the stability analysis for the NuScale power module.

If we move on to slide 9. The topical report not only discusses PIM but also the nature of the long-term stability solution. We found that this solution is also acceptable as the applicant has identified the correct primary instability mechanism, we've confirmed that mechanism through our own independent analysis.

During normal at-power operation, we've concluded that the NuScale power module is very stable, and that the exclusion region-based long-term stability solution is effective in preventing the reactor from becoming unstable during normal operation and including the effects of AOOs.

We also considered worst rod stuck out conditions. This was discussed in greater detail at the Subcommittee meeting. But we concluded that potential instability during return to power with

worst rod stuck out is not an issue that needs to be considered as part of that analysis.

And so therefore GDCs 10, 12, 13, 20, and 29, which are the applicable regulatory criteria, are all met.

MEMBER MARCH-LEUBA: Pete, you performed a calculation with trace of ATWS with a conservatively low MTC model for the temperature coefficient, in which you are in ATWS and now you have boiling for an extended period of time in the riser. Could you describe the stability results you saw with trace within that transient?

MR. YARSKY: Oh, so this is really outside of the scope of the stability evaluation here because the long-term stability solution of course credits the reactor trip. However, if you were to have an ATWS event, there are sort of two possible ATWS cases one could consider. One could consider like an overpressurization-type ATWS scenario.

In that scenario, you don't have a significant decrease in primary system pressure and inventory. In those kinds of scenarios, the high reactor power and the ATWS leads to a loss of inventory through the reactor safety valves, which will just burp off excess pressure, transferring

inventory from the primary side to the containment.

In those scenarios, there's a slow reduction of the level, but the pressure's really high. So because of that high pressure, you don't necessarily see voiding in the riser until you get to a condition where the level drops below the top of the riser.

And when the level drops below the top of the riser, there's a significant reduction in the core flow rate because the natural circulation pattern breaks. At that point, the nature of any flow oscillations, that mechanism that creates those flow oscillations is very different than the mechanism that we're discussing as part of this topical report, because you don't have that same loop, right.

You now have something more akin to like a YouTube manometer, where you have like a liquid level that's separated by the riser and you have a level above the core. Under those conditions, the core flow rate can get quite low. You can have boiling and you can have the level kind of sloshing back and forth between the --

MEMBER MARCH-LEUBA: But before the water level dropped to the top of the riser and went into what I call the percolator mode, when the door was

high and you had circulation, did you see significant boils in the riser or did you see any flow instability?

MR. YARSKY: In our calculations we didn't observe significant flow instability. There is the possibility for there to be conditions where flashing may occur in the riser under certain conditions. That may occur prior to the level dropping below the top of the riser because the RSV is cycling between its lift and its set pressure.

So if the core power remains high enough that it heats up the riser to a relatively high temperature and then the RSV lifts, there'll be a moment where the system pressure may drop below the -- they drop sufficiently low that the temperature is above the saturation temperature in the riser. Now, that would be like a longer period-type phenomena, so that would like over the course of minutes.

MEMBER MARCH-LEUBA: But that would not be --

MR. YARSKY: Where the RSV cycling is happening. But the voiding itself would be, would occur in a much shorter timeframe.

MEMBER MARCH-LEUBA: But that would not be an intrinsic instability, it would be driven by the

SRV or RSV.

MR. YARSKY: Yeah, so you could potentially have this short-term flow response to the RSV cycling. But the RSV cycling frequency is going to be divorced from the natural frequency of the flow loop. So it wouldn't necessarily create a feedback loop that would create a growing flow instability.

MEMBER MARCH-LEUBA: It's safe to use and you used a state-of-the-art code trace.

MR. YARSKY: Yes.

MEMBER MARCH-LEUBA: To analyze this phenomena, and you did not see large oscillations.

MR. YARSKY: Correct, correct. So trace we did an extensive look at the trace applicability to analyze these phenomena and concluded that trace is able to model these phenomena for NuScale, including doing additional trace assessments to address potential information gaps.

So that is something that we've had done, concluded traces applicable and performed those analyses, and we didn't see any issues associated with flow oscillation during our ATWS analysis.

MEMBER BLEY: Peter, this isn't exactly a question for you, but maybe somebody on the staff can address it. And we didn't ask NuScale about it. This

ATWS scenario sounds kind of benign, but I'm not completely sure of that. I don't think it's analyzed in the PRA. It seems like, as if it ought to have been. But I don't, the PRA, your PRA folks didn't come to you about this issue, did they?

MEMBER CORRADINI: We had Pete's presentation, I don't remember which Subcommittee meeting, in closed session about these results.

MEMBER BLEY: On a --

MR. YARSKY: Yeah, so on the chapter, as part of Chapter 19 review.

MEMBER BLEY: On Chapter 19. Okay, I didn't remember that, okay.

MR. YARSKY: One thing to keep in mind is that the acceptance criteria for beyond design basis events like ATWS are different than the acceptance criteria for stability and AOOs. So for an ATWS event, there's the possibility that the, you may lose thermal margin, that the fuel could go into CHF. That would allowable so long as the core remains in a coolable geometry.

MEMBER BLEY: That's why I was asking about the PRA, which should have looked at --

MR. YARSKY: Right so the --

MEMBER BLEY: –- the likelihood of damage

under the specifications.

MR. YARSKY: Right, so in the ATWS scenario, you're looking at possibility for fuel damage, you're not looking at possibility for fuel failure due to CHF considerations.

MEMBER BLEY: Right.

MR. YARSKY: So because of that difference in the acceptance criteria, the event that we analyze for ATWS, we'd say well it looks relatively benign from the standpoint of those acceptance criteria, which are of course very different than the acceptance criteria considered for AOOs.

MEMBER CORRADINI: But just to remind everybody, he did identify and we have the docketed, his complete report on ATWS.

MEMBER BLEY: Oh, I do remember that now.

MEMBER CORRADINI: I sent it to you.

MEMBER BLEY: Yes.

MEMBER CORRADINI: You asked me where is it and I sent it to you.

MEMBER BLEY: You sent it to me, you are absolutely correct.

MEMBER CORRADINI: Just wanted to make sure we're clear.

MEMBER BLEY: And Peter was the right one

to ask that question.

MR. YARSKY: I'm happy to help.

MEMBER CORRADINI: Anything else for Peter? Okay.

MR. YARSKY: Thank you.

MEMBER CORRADINI: Thank you very much. So you don't have anything in closed session to present, but you'll be around if we have questions.

Did we have, I want to make sure, want to ask the members, do we want to now go into closed session, as if we're not already, and talk about details that Yusuf Farawila said he'd have to wait and talk about in closed session? Is that what the members would like to do?

MEMBER MARCH-LEUBA: Yes.

MEMBER CORRADINI: Okay. So I'm going to need to get NuScale folks back up in front and then get Yusuf back on the line.

MEMBER MARCH-LEUBA: The transcript needs to go into closed and we need to verify that a room is available.

(Whereupon, the above-entitled matter went off the record at 10:19 a.m. and resumed at 12:59 p.m.)

MEMBER RICCARDELLA: We will resume the

meeting.

MEMBER CORRADINI: Okay, so, Marty, I think you're going to take us through Chapter 3?

MR. BRYAN: Yes, sir. Thank you.

MEMBER CORRADINI: Okay.

MR. BRYAN: Good afternoon. I'm Marty Bryan, the licensing project manager for Chapter 3 and today we're going to cover three main subjects.

Amber Berger is going to take us through turbine missiles. There was a lot of questions at the last meeting, but I will note this is the open session. Most of the detail for that will be in the closed session.

Olivia Hand is going to discuss CVAP and TF3. Again, it will be open, and more detail in the closed session. Then Hannah Rooks is going to take us through the DWO. I know there was a lot of discussion earlier. We'll try to address some of those questions.

MEMBER CORRADINI: And we'll hold the closed session until we hear after the staff.

MR. BRYAN: Yes, correct, yeah. Okay, so with that, Amber, will you take us through the turbine missiles?

MS. BERGER: Sure, Amber Berger, NuScale.

Chapter 3 or Slide 3 is just a title for the missiles. Slide 4 is basically an overview of what we'll discuss today –- Whoever is shuffling paper, could you put your phone on mute, please? Thank you.

Okay, again Amber Berger. The slide 4 of this slide deck is just the agenda of the turbine missile discussion. We're going to talk about and we'll show an illustration of the turbine missile trajectory and the barriers that we're crediting in our analyses.

We'll go over some additional details of the turbine barrier analyses that we performing, focusing on local analyses and global analysis. We'll also present the turbine missile barrier results and discuss the conservatism in our approach, and that's all I have on turbine missiles right now.

MR. BRYAN: Thank you, Amber, and again, we'll cover it in a lot more detail in the closed session. I know there's a lot of interest in the REI response, and so, previously, so we've extracted a lot of that information.

So now we're going to transition to Olivia taking us through CVAP and TF-3.

MS. HAND: Hi, so this is Olivia Hand. If we move onto slide number 6, that has a brief

discussion of what TF-3 is and the purpose of that testing.

So this TF-3 test specimen is a five-column full height assessment. It has prototypic steam generator tube support. It's a two-part test program. The first part is to characterize the response of the test specimen, specifically looking at the response of the steam generator tubes, and we're going to be performing testing both in air and in water.

Then once we have the (inaudible) testing complete, we'll be moving onto flow testing. Flow testing for this is just focused on the primary site flow. So this is the test specimen and we do not have flow inside of the steam generator tubes.

And the purpose again, we're looking to characterize the frequency response of the steam generator tubes, the mode shapes that we're seeing, and also the damping (phonetic) values, and that's because all of three of those parameters are inputs to our design analysis.

As a part of the flow testing, we'll be looking to characterize the critical velocities for onset of (inaudible) phenomena, which are elastic instability and vortex shedding, and then also we're

going to be looking at what square vibration amplitudes are as we change our primary coolant flow rates.

And we've done some preliminary demonstration testing. The NRC was out to view this last month in Italy, and so far, our results are showing very good agreement in both the frequency and dampening with our design analysis predictions.

But again, those results are just preliminary. We still have to actually complete commissioning of the test facility, and then we're hoping to kick off testing for our first column in September.

MEMBER CORRADINI: So let me ask a question at this point. The flow restrictor tests are a separate set of tests, is that correct?

MS. HAND: Yes, that's correct. So we have actually some validation testing to do before the program for the final flow restrictor design to look for the lack of a leakage flow instability characteristic for that component, but, yeah, there are no flow restrictors in this design. There is nothing inside of the steam generator to (inaudible)

MEMBER CORRADINI: So, I'm sorry, go ahead.

MEMBER MARCH-LEUBA: No, you go ahead. Okay, so this is the TF-3 test. What's the status of the TF-2 test? Because we are going to -- if we go on (inaudible), we're going to strongly encourage that you rerun the heated tube flowing side tests with the final solution. How much of a problem would that be to resuscitate TF-2 or am I asking the wrong person?

MS. HAND: Hannah might be able to -- Hannah Rooks might be able to field that question a bit better because as far as flow and vibration is concerned, we've already used the TF-2 results for our benchmarking, and we got acceptable results, so.

MEMBER MARCH-LEUBA: Yeah, this is a completely different topic. It's not flow and vibration. It's inside the tube circulation, so which is what we tested on the TF-2 test. What I'm asking you is does the TF-2 reading still exist? Because apparently TF-3 will not be able to do this.

MS. ROOKS: Yeah, this is Hannah Rooks. So the TF-2 test facility is currently, it's still in existence, but it's, you know, it's closed down and we weren't planning on doing any more testing at that facility.

MEMBER CORRADINI: But can you remind us what TF-2 is? It's a heated test facility, is that

correct?

MS. ROOKS: Yes.

MEMBER CORRADINI: Okay, and --

MS. ROOKS: It has primary and secondary side flow, and the main purpose of it was to characterize (inaudible).

MEMBER CORRADINI: But it was that test facility that at least exhibited flow, two-phase flow oscillatory behavior on the secondary side, is that not correct?

MS. ROOKS: Yes, so we had tests that were specifically designed to reduce oscillations, and then we also had other tests that had stable operations, so we had kind of a variety of tests in that facility.

MEMBER CORRADINI: Okay, thank you. That helps me.

MEMBER MARCH-LEUBA: Are you documenting those? Which conditions were the ones that were stable? Are those -- I don't think -- are they on the area response? I forget the number. I don't think I've seen it. It would be very interesting for me to see what condition resulted in stable flow.

MEMBER CORRADINI: I think what Dr. March-Leuba is asking for really is to get some sort of mapping of where things were stable versus where

things were not stable on the secondary side, and the flow conditions that essentially drove those differences.

MEMBER MARCH-LEUBA: Yeah, basically this issue of the tube flow instability, which has nothing to do with vibrations. It's a two-phase flow oscillation has gotten so much visibility that you have to solve it, and you have to prove that it's solved.

So I'm just putting it in there that it needs to -- you need to start thinking about how to fix it and then how to prove that you fixed it, and I'm thinking here that TF-3 will not be the proper vehicle. Resuscitating TF-2 might be a better path to a solution.

MR. BRYAN: We are going to talk about DWO here in the third presentation as part of this.

MEMBER MARCH-LEUBA: Okay, so we'll postpone it then.

MR. BRYAN: Yeah.

MEMBER MARCH-LEUBA: You guys know --

MR. BRYAN: Yeah.

MEMBER MARCH-LEUBA: -- where we're going.

MR. BRYAN: Yeah.

MEMBER RICCARDELLA: Regarding CVAP and

TF-3, I'll point out that, you know, we addressed this at our last meeting and actually included it on our letter, in our prior letter, and one of our recommendations related to that is that there be an ITAAC regarding TF-3.

MR. BRYAN: Yes, and we have proposed some wording to be put in tier one as opposed to an ITAAC to address and ensure TF-3 is done, so we'll probably be working with the staff to evaluate that, so we did see that in the letter, yes.

MEMBER RICCARDELLA: Okay, great.

MEMBER CORRADINI: Go ahead.

MR. BRYAN: Go ahead, Olivia.

MS. HAND: That was it for CVAP and TF-3, so we can move on to Hannah.

MR. BRYAN: So Hannah is going to cover the DWO discussion.

MS. ROOKS: Okay, so we'll move onto slide 8. So this is an overview of the work that has been done so far in our design base. One of the design objectives for the steam generator is to withstand all loadings for the design type of the components, and that includes oscillation loading.

So what we've done is we've done RELAP, NRELAP5 simulations to compare to the TF-1 and TF-2

test data, and we compared it to both the cases with stable results and cases with (inaudible).

And so the test data shows that the frequency of the oscillations are much slower than natural frequencies of the components, so there is no resonance of (inaudible) concerns, but they do show that we have (inaudible) oscillations at our low power operating conditions.

So the design objective of our flow restrictor is to create pressure drops in the (inaudible) to limit those oscillations (inaudible) set the lower limits.

So the analysis that we've done with our full 21-column steam generator model is to analyze the conditions at (inaudible) readings of power levels and to (inaudible) on that last coefficient in the unit flow restrictor to get below our second criteria for the flow.

MEMBER CORRADINI: So let me just ask a question at this point. The flow restrictor that you now are testing is not what was used in the TF-2 testing, is that not correct?

MS. ROOKS: Yes, that's true. So the TF-2 test, the last coefficient was lower than our current design, and it was also just like an orifice, and our

current design has several steps of constricting area (inaudible) in it, so it's a different design.

MEMBER CORRADINI: Thank you.

MEMBER MARCH-LEUBA: Okay, and I see you using NRELAP5 for these models. Did you benchmark it, I mean, the TF-2 experimental data, and how were --

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: I'm telling you because my experience with RELAP is, you know, very bad.

MS. ROOKS: It showed very stable results at the 100 percent power operating conditions, but we did see some oscillations in our lower power levels, so we've done some, you know, combination of operational changes. So we've increased our steam pressure at the lower power levels to provide more subcooling, and we are also looking into some other options for, you know, operational changes as well.

MEMBER CORRADINI: So can I just follow up on Dr. March-Leuba's question? Is there on the docket a report that compares the calculation to the TF-2 experiments or is that internal to NuScale?

MEMBER MARCH-LEUBA: And since I know the answer is internal, can we see it in a couple of weeks?

MEMBER CORRADINI: Well, let her confirm what you're thinking. If Hannah could at least, if you could please at least explain? You said you did a comparison of NRELAP to the TF-2 data. Is that in some form of a report or can it be discussed with us? That's what I think we're having some interest in.

MS. ROOKS: Yes, we do have those written calculations. They're proprietary class one calculations because they have, you know, test data in them.

MEMBER CORRADINI: Okay, so we'll take that up and see what, between the staff and NuScale, if we're allowed, if we can see those in some manner or form because that confirmatory calculations we were looking for relative to you predicting it under full power conditions.

MS. ROOKS: Okay.

MEMBER CORRADINI: Go ahead. I'm sorry for interrupting.

MS. ROOKS: All right, so part of our revision three was just going to clarify some terminology that's in our SR section 5412, so we just wanted to make clear that the design intent of the inward flow restrictor is not to fully preclude (inaudible) oscillations, and that there will be

naturally some small amount of oscillations, but to ensure that they're within acceptable limits.

And then the other ongoing activities right now are we are developing a stability map based on the TF-1 and TF-2 data for comparison with the stability map that the NRC has presented as part of the stability work.

So moving onto the next slide, I want to go over the activities that are in our ITAAC related to the oscillations, so we have a simulator tube. The tube (inaudible) and the tube sheet are class one components.

Their value was evaluated to the rules of subsection (inaudible) of the ASME code, and these components will be included in the design report, so will be submitted and reviewed by the NRC, and these are actions that are currently in our ITAAC.

So the analysis has been completed and is under NRC review. The tube sheet (inaudible) were not included and the (inaudible) was not considered yet, but maximum result usage in the vicinity of the tube sheet weld is very low.

So the next step is the (inaudible) analysis. And we'll consider all of the (inaudible) and that will help kind of circle back to the design

work that was done to confirm that the (inaudible) to reduce those oscillations to acceptable levels, so that was it.

MR. BRYAN: Yeah, I'm starting to take questions. Is that all? That's all we had for the open session for Chapter 3.

MEMBER SKILLMAN: Yes, I'd like to ask this question. The presenter communicated that secondary pressure was raised to increase subcooling, and the question is did that increase in pressure result in any magnification of the pressure waves at the mouth of the tube where the restrictor orifice is located?

It seems to me that driving up the pressure will increase subcooling, but it could also enlarge a rarefication wave if there's a pulse.

MR. WOLF: This is Brian Wolf, development supervisor at NuScale. I'm not sure I understand the question, but here (inaudible) oscillations (inaudible) waves, not pressure waves. What we know is that obviously inlet flow restriction helps to reduce oscillations.

If you increase exit losses, that will potentially increase oscillations. Increasing pressure typically helps DWO, as well as increasing

the inlet subcooling to some extent, depending on where you're at, and then also increasing power is destabilizing, and decreasing flow rate can be destabilizing depending on the operating conditions.

So I'm not sure I understand the question, but increasing the pressure, considering all other things constant, would typically be used to help reduce DW oscillations.

MEMBER SKILLMAN: Thank you.

MEMBER CORRADINI: So are we -- so we'll move on to Chapter 6?

PARTICIPANT: Yes.

MEMBER CORRADINI: Okay.

PARTICIPANT: I think we're going to --

PARTICIPANT: Fifteen.

PARTICIPANT: Yeah.

PARTICIPANT: NuScale doesn't plan to present on Chapter 6 publicly, but they will in the closed session.

MEMBER CORRADINI: So we will talk about six in the closed session?

PARTICIPANT: Yes.

MEMBER CORRADINI: All right, thank you, to Chapter 15.

MR. PRESSEN: And just to confirm, do we

have Meghan on the line?

MS. McCLOSKEY: Yes, I'm here.

MR. PRESSEN: Excellent.

MEMBER CORRADINI: Go ahead.

MR. PRESSEN: All right, so thank you all again. I'll introduce myself. I'm Matthew Pressen, licensing specialist with NuScale Power and project manager for Chapter 15.

We're here today to discuss Chapter 15 of the NuScale design certification application, and the presentation today will be provided primarily by our technical experts in Corvallis.

Meghan McCloskey, supervisor of system thermohydraulics, will be our speaker today, and joining us as needed will be Dr. Pravin Sawant, Dr. Brian Wolf, Dr. Selim Kuran, and Mark Shaver also on the phone in Corvallis.

So we wanted to provide a quick scope of the information that we will be presenting today as there is a lot of information summarized in Chapter 15 where we deal with postulated transients and events, and a number of topical reports that define methodologies that we use in Chapter 15 but have not yet presented on.

So in our presentation today, we focus on

material provided in the FSAR and the results of those methodologies. So we have the summarized topical reports here, but any detailed discussion will wait until we get those in November, I believe.

A quick overview of what we'll be going over today, as mentioned, will be the high level of the NuScale module design, including an overview of Chapter 15 assumptions and analyses.

We'll cover the limiting cases seen in Chapter 15, and for the Chapter 6.2.1, containment response, and the last portion of our presentation will be focused on long-term cooling and how Chapter 15 assumptions look in 72 hours.

So this is a high level map of the technical and topical reports that developed methods used in Chapter 15 and Section 6.2.1, containment. NRELAP5 was developed from the RELAP5 3D code to address NuScale specific phenomenon and levels used to evaluate the NuScale power module when developed following Reg Guide (inaudible), evaluation model development and assessment process, which is that acronym there.

These topicals are sorted by several additional methods, which the ACRS had a chance to review, which are the nuclear analysis codes and

methods, the critical heat flux, and the subchannel analysis listed there.

And while we will discuss this in more detail in the later portion of our presentation, we did want to kind of put up front our exemption to GBC 27 since it has an impact on the acceptance criteria for the NuScale BCA essentially to account for the postulated return to power correction that can occur under Chapter 15 assumptions as part of that NuScale design, assure fuel cladding integrity for all design basis events, so we're decking our fuel for all of those Chapter 15, well, Chapter 15 pretty much.

So again, this is a fairly dense slide. The point of that is to open our discussion with Chapter 15 with a quick summary of our dose consequences.

The bold numbers on the right-hand of the slide represent our limiting cases for the postulated design basis and beyond design basis events, and we felt it was a good slide to show we are creating a safe, and passive, and functional nuclear design.

MEMBER MARCH-LEUBA: Could you describe the scenario for what you call the core damage event?

MR. PRESSEN: Yeah, do we have anyone in Corvallis that can talk to core damage events?

MR. SHAVER: Sure, this is Mark Shaver, supervisor of radiological engineering at NuScale. The core damage event is what's typically called the maximum hypothetical act or MHA, and the guidance used to develop it was the NEA white paper on mechanistic source terms for small modular reactors, so it has substantial core damage.

MEMBER CORRADINI: But give us a little bit --

(Simultaneous speaking.)

MEMBER CORRADINI: Can you give us a little bit more about the initiating event and how it progresses?

MEMBER MARCH-LEUBA: And what fraction of the core is assuming failure?

MR. SHAVER: It's actually a surrogate event that's representative of a spectrum of severe accident beyond design basis events, so it's not one particular scenario.

MEMBER CORRADINI: But it's a radio --

(Simultaneous speaking.)

MEMBER CORRADINI: It's a radiological accident, so I'm assuming you have essentially a gap release? I'm still trying to understand because I don't think -- we're thinking thermohydraulic, but I'm

thinking it's more a radiological design basis event, is that not correct?

MR. SHAVER: It's a beyond design basis radiological event, but the inputs are thermohydraulic inputs, and if you'd like the details of the different events that go into that, we can get our PRA group on, but we do have a gap release and release timing, and those are informed by different scenarios that are severe accidents.

MEMBER CORRADINI: Okay, go ahead. We'll catch up. Keep on going.

MR. PRESSEN: All right, so on slide seven, this was primarily provided here for reference for any questions that we had --

OPERATOR: The chairperson has disconnected. The conference will now end.

MEMBER MARCH-LEUBA: Do we want to break off the record?

MEMBER CORRADINI: Well, obviously we're on break for right now.

(Whereupon, the above-entitled matter went off the record at 1:27 p.m. and resumed at 1:31 p.m.)

MEMBER CORRADINI: Okay. Let's keep this going. That was pretty quick.

MR. PRESSEN: All right. So back to slide

seven. It's provided here as a reference for use for any questions. It's a quick diagram reminder of what the MPN looks like.

And it provides a schedule of highlights on the unique aspects of the NuScale design, such as the integrated reactor design, passive safety decay heat removal, and automated module protection system for handling event mitigation.

So, and with that, we are on slide eight. Which is where I will pass this over to Megan McCloskey.

MS. MCCLOSKEY: Thank you. This is Meghan McCloskey with NuScale. Before we get into the overview of the Chapter 15 events and results, we have two slides on the module protection system functions and the types of functions expected in response to the Chapter 15 design basis event.

The module protection system actuations for the NuScale plant were designed with a focus on our events and our specific module response. And the actuations needed to support a passive plant response to design basis events without operator action.

So, when we consider the scope of events, the functions needed from the module protection systems can be broadly classified in four areas.

Reactivity control, through reactor trip, you know, or -- and isolation of the blind emission source from the demineralized water systems.

RCS and secondary inventory controls where containment isolation assures that position inventory is maintained for ECCF cooling. And containment isolation also mitigates loss of inventory outside of the module for breaks of small lines to limit dose consequences and mitigate the event.

So, secondary isolation also mitigates dose consequents for events such as the steam generator tube test layer. And assures that second -- appropriate secondary site inventory is maintained and at least one train of DHRS for the decay heat removal.

A dual secondary side, if cooling is unavailable, the module protection system actuates DHRS. If necessary the emergency core cooling system to provide heat removal.

And finally, as was discussed this morning, primary side sub-cooling and stability are protected by a reactor trip. Going to slide nine.

MEMBER MARCH-LEUBA: Okay. Can we hold up, please? I don't know which particular revision you have been -- NuScale has been changing the number of trips, or the number of initiation signals that

result on a trip, eliminating some of the redundancy.

Could you explain what the philosophy was on -- for example this actuation only happens on high level in the containment.

MEMBER CORRADINI: And there was a second trip that you indicated to us in the subcommittee meeting that you were going to eliminate.

MEMBER MARCH-LEUBA: That's correct. And you mistake that for losing DC power where that is not an automatic trip.

Could you --

MS. MCCLOSKEY: Right. Right, so we --

MEMBER MARCH-LEUBA: Could you explain the philosophy behind, I mean, we don't -- I don't feel comfortable eliminating the redundancy. I would like to have as much redundancy as possible.

Could you explain what your philosophy was?

MS. MCCLOSKEY: In the original design as submitted, we had ECCS actuation by the module protection system on either low level in the riser inside the vessel.

MEMBER MARCH-LEUBA: Um-hum.

MS. MCCLOSKEY: Or high containment levels.

MEMBER CORRADINI: That makes sense.

MS. MCCLOSKEY: And the change that we had made is to eliminate the trip on the high, I'm sorry, on the low riser level. And so we are relying on the containment level trip to actuate ECCS.

The way these signals were originally designed, in most cases that riser level and containment level were happening around similar times. And we determined that we wanted to delay the ECCS actuation.

And so we focused on the containment level signal. And it's a little bit higher than it was previously.

From Chapter 15 perspective, the module protection system is our safety-related system. There are redundant instruments measuring the containment level.

And so from that perspective, we felt there was sufficient robustness in the riser to achieve ECCS.

MEMBER CORRADINI: So, I don't think I understand your timing argument. Can you repeat that? Because that -- I remember you saying that to us during the subcommittee meeting.

And I'm not following, because I think I'm

with Dr. March-Leuba that the redundancy on two different locations gives you a diversity that you now have removed.

And I guess I'm not understanding your timing argument. Can you repeat that please?

MS. MCCLOSKEY: When we have those two different signals, in terms of the event progression -- in terms of the progression, removing the riser signal doesn't significantly change the timing of the event progression, because we're getting the actuation signal on the high containment level around the same time that you, we would have been getting the actuation on the low RCS riser level.

MEMBER MARCH-LEUBA: Yeah. Well, your point though of why we're trying to talk about diversity and redundance is that this is correct on a simulation when everything works perfectly.

But in real life, having back up, I mean, in my car I like to have four breaks instead of only one. You know, just in case one breaks, one fails, you have the -- the ones in the back still work.

So, your only excuse for the lack of available water is that in the simulation you were always tripping on containment. So, why have the inside the vessel level?

I mean, it is a perfectly acceptable answer, only that that assumes that all those other instruments are going to work perfectly. And all events will progress the way you modeled them.

MEMBER BLEY: This is Dennis Bley. I sort of got a hint you were implying there was some kind of interference you could have if these trips are coming in about the same time.

But, I don't understand that. This -- is the only reason you're getting rid of it is because they're replicates and you don't need to -- you don't think you need them both?

MS. MCCLOSKEY: From Chapter 15 perspective, we didn't think we needed them both. I don't think -- we have our I&C Group here.

And we can provide some additional information with respect to diversity, the diversity aspects with the containment level signals.

MR. AYALO: Thank you Meghan. So my name is Rufino Ayalo, I'm with the I&C Review Group. And just one of the things that isn't specifically, you know, modeled and taken into account within Chapter 15 is some of the diversity that we already have within some of our measurements, our sensor measurements.

So for example, the containment water

level, within, you know, regarding redundancy, you know, we have four separate containment water level sensors. And the -- it would specify that two of the four has to be diverse, from different vendors.

So that provides a different level of diversity. So, we're not relying on a single sensor from the same vendor across all four separations.

As we said, we have two from one manufacturer and two from another. And that provides an additional level of diversity.

MEMBER BLEY: Can I -- before you leave that one, can I push you a little bit? I remember in the past we had different kinds of DP cells, we thought from different manufacturers.

But when you really dug into it, you found that the actual operable unit was only by one manufacturer, and put in different housings and had other names on them.

So, are you sure you have that diversity you're claiming?

CHAIR REMPE: Well, I'd even go further. Let's remind ourselves that this is close to a first of a kind application in a containment or a reactor vessel.

And I'm real curious, I mean, I wasn't

here when you guys discussed this, because I had to leave for the subcommittee. But, at the time somebody from NuScale popped up and said oh, yeah, Chapter 7 talks about diversity.

And I looked it up before this meeting. So I'm glad you at least acknowledged your diversity definition is two different vendors.

But why didn't you explore something like a multipoint thermal couple on the side of the containment? Or within the containment or within the vessel as a really truly diverse sensor that's not a first of a kind application?

MR. AYALO: So, that's a lot -- there's a lot of questions there. So let me see if I can try to answer a few of those.

And just as a reminder as well, for later this month, there will be a presentation related to sensors as well. But, regarding your -- the first comment about the use of the different level measurements.

You know, one of the approaches that we're taking is it was a phased approach that we started and we described in our technical report associated with sensors.

You know, we focused with the --

essentially a down place of various concepts. And you know, based on the conditions that were, based on our scenarios, based on environmental conditions, we decided to use the levels selected.

And while it does appear to be, you know, first of a kind in terms of a nuclear application, there are applications such as Dean Drum's (phonetic) where you see the use of level sensors in this particular application.

So our -- in sort of this style of environment, so --

CHAIR REMPE: Well, it has a radiation, a high dose radiation level in the other applications? I mean, I know you -- they've been used maybe in a split fuel pool or something.

But, within the reactor vessel you see -- I don't know what your design has. But some of them have some sort of glass that will darken and would recalibrate.

And you've seen some sort of application where they've done this, and they don't have to worry about the radiation levels, fluences?

MR. AYALO: Yeah. I mean, I guess that -- I guess maybe I should clarify. I'm not saying it's an exact match of our situation with the high

radiation, the high pressurized temperature.

But there are, and we will demonstrate as part of our program, the environmental qualification of the center with the radiation, with the pressure, with the temperature, but from a -- from a purely that there is no examples in other industries where this is being used in a similar application.

And that -- and by that I mean, the pressure and temperature, I would say that that -- that there are examples of that. And that would be Dean Drum where there is a level measurement taken with the use of radar. And in particular, guided wave radar.

CHAIR REMPE: Well, it will be interesting to hear more about it when we travel to Corvallis. But again, now we hear you're going to have two vendors helping you do this first of a kind technology or when you finally deploy it.

And again, you could use other technologies that are different that would be more diverse and maybe provide more confidence that you don't have to worry about common cause failures, which is, I believe, what Jose was emphasizing in the transcript when I was reading through it.

MEMBER MARCH-LEUBA: Well, at a minimum

the diversity from two vendors gets you away from -- if it truly is diverse, and they're not just two different labels on the same instrument. It gets away from the digital common cause blue screen of death. That's good.

CHAIR REMPE: But now we have to worry about fluence not only for design basis conditions, which is what is in that, the open version of your technical report.

It clearly says that is what you are going to do for the radiation levels. But now, we might want to worry about it from MND type events which might have a different radiation level too.

But anyway, it's just a question that I keep asking.

MEMBER MARCH-LEUBA: My personal sense in how they -- it's up to the seven guys up there on the microphone, is that this sensor will measure a level during initial startup as you empty the containment.

And then it will be out of the scale for two years. It will not be giving you a signal. It will be giving you noise. Hopefully not much noise.

But, there won't be any bounce back for two years. So, if there is a failure of those sensors, it's likely it will go unnoticed for two

years until you need it.

So, that's like, I mean, you have to increase their liabilities significantly. When you're -- every second you're measuring and getting a signal that this is changing and it's oscillating, you have confidence that the sensor is alive.

When for two years you have it completely off scale, you've got to make sure that thing is working. Just something for you to think about.

MR. AYALO: So what we -- so I would like to respond to that. But I think that in terms of some of the details involved, I'd like to discuss that in the closed session.

But, I would disagree with your statement that it would be out of calibration. Because what you're essentially saying is if I have an empty tank, you know, a level sensor is always out of calibration, which is not necessarily true.

But in the closed session I'd like to go into more details on how you can demonstrate that the level sensor itself, it's still working, even with no level in any tank, or even in our containment.

MEMBER MARCH-LEUBA: And you can save that for our visit to Corvallis in a couple of weeks if you want to. You know, I just want to bring -- the fact

that you thought about it, that's -- you're 90 percent there.

MR. AYALO: Yeah, thank you.

MS. MCCLOSKEY: Okay. So this is Meghan McCloskey again. Moving onto slide nine. When we look at the functions of the module protection system and look at our design basis events mitigation, this slide summarizes broadly the types of actuation that we expect to see for the different event types.

Of course reactor trip is common. To provide reactivity control and reduce heat loads to the decay heat level.

For the vents that are increasing heat removal transients, secondary site isolation isolates the module from the malfunction causing most of the event. And then BHRS heat removal can be provided as necessary.

For the decrease in heat removal events, the decay heat removal system provides any removal following malfunctions that indicate normal -- secondary type cooling is unavailable.

The reactivity and power transients, the reactivity control is most important. It's provided by reactor trip and the demineralized water isolation.

The increase in RCS, in reactor coolant

system inventory transients, is mitigated by containment isolation. Which controls the inventory -- which controls the additional inventory from the chemical and volume control system.

And finally, the decrease in reactor coolant system inventory transients is mitigated by containment isolation to control the inventory and mitigate dose consequences.

For LOCA and valve opening events, and ECCS, emergency core cooling system actuation establishes the cooling path. And finally, quality of the light water reactor trip protection.

Going to slide 10, the Chapter 15 design basis initiating events consider internal events that could affect a module while operating at power conditions from hot zero power to hot full power conditions.

And the Chapter 15 analysis are performed for a single module response. For the NuScale we primed chill systems, the reactor pool ultimate heat sink is the key safety related shared system.

And it's most relevant in the long-term cooling analysis. And therefore the long-term cooling analysis would consider the effects of up to 12 modules rejecting heat to the reactor pool ultimate

heat sink.

In order to identify these Chapter 15 design basis events, we used a systematic process to evaluate the NuScale system. And components that could affect the module response and cause reactor trip or other type of plant transients as well as considering industry experience and other advanced module PRAs in order to identify the skillful design basis events.

And we evaluated the heat scale systems and components to identify failures that could affect the module process, the module response, focusing as well as on NuScale system designing system -- design features such as the containment evaluation system and the emergency core cooling system.

When we had identified the failures we categorized them based on their effect in the module. And those categories are consistent with the operating white water plants.

And we also have NuScale specific phenomena and event progression that are analyzed as part of Chapter 15, namely feedwater flexibility that was discussed this morning, and return to the power analysis that we have later in this presentation.

Going to slide 11, this slide highlights

the NuScale unique Chapter 15 events. Because our other events are consistent with light water reactors, or are not applicable to our design because of the natural circulation operation of the plant.

So any unique events, the first is discussed in Section 15.1.6, the loss of containment vacuum. During normal operation, the containment is operated at vacuum conditions in order to minimize heat transfer to the coolant and allow for leakage detection.

And the loss of vacuum is postulated due to failures in the evacuation system. Containment flooding is postulated due to a break in piping that can carry its reactor component cooling water to the control rod drive mechanism.

And either of this loss of vacuum or flooding increases the heat transfer from the reactor pressure vessel to the containment in terms of normal operating conditions. And so it's a -- we're hoping a very slow increase in a heat removal event that's really non-limiting compared too other over cooling events analyzed tentative to inspection.

The inadvertent operation of the decay heat removal system is another unique NuScale AOO. A place for the heat removal system is inside for decay

heat removal if an inadvertent operation causes a decrease in heat removal while that power operation.

And there are several variations of this event. It may be a single valve opening, which results in a diversion of feedwater flow from the steam generator too over the decay heat removal system condenser.

Or it may be an inadvertent signal that actuates the DHRS valve and closes the secondary isolation valve on one or both frames in the system.

In the event of 15.6.6 in the decreased inventory events, the inadvertent operation of the emergency core cooling system is a NuScale unique event. It also works without the valve opening event.

And then although opening of a recirculation valve, or a vent valve is not expected to occur during the life of the plant, this was conservatively classified as an anticipated operational occurrence.

And we extended the LOCA evaluation model to the valve opening events and used the inner related pod code to demonstrate that margin two critical heat flux is maintained during the event.

Now moving on to slide 12. For Chapter 15 analysis, there are a number of assumptions that are deterministically applied.

For the -- as a passive plant in the NuScale design, we assume no operation actions to demonstrate that none are required to achieve safety functions for at least 72 hours after the limiting event.

In addition to the -- during the initiating event, the worst single failure of a safety-related component assumes, and then the scalability of, and power considered to determine whether loss of power is more conservative with respect to margin two acceptance criteria.

In terms of the scope of the event progress, the Chapter 15 analysis are performed until a safe, analyzed condition is reached, by which we mean that the module protection system responses to the initiating events have occurred.

We've demonstrated that there's margin to the acceptance criteria in system parameters such as inventory, temperatures, pressures, are trending in a favorable direction.

After the end of that short term response and demonstration of the safe stabilized condition, the longer term progression is considered. And the long-term cooling analysis addresses emergency core cooling systems, long-term decay, and residual heat

removal.

The return to power analysis demonstrates that this provides acceptable fuel design limits are met. This hopefully continues in the module with running control water is struck out of the core, and a return to power occurs.

And if power is available but no operator actions are assumed, then extended DHRS operation is good and considered.

Going to slide 13, this slide and the next slide illustrate the effect of loss of power, AC and DC power functions on the Chapter 15 event progression.

This slide is for the non-LOCA events that are initially mitigated by decay heat removal system. And so occurring these events relatively early in the transient, the module protection system will detect the upset condition and actuate reactor trip, and if necessary the decay heat removal system.

On the search side, with AC and DC power available, that plat is the end of the module protection system actuation. And typically stable, the completely mobile system is established by a half hour into the event.

If both trains of decay heat removal

system are operating and that cooling is assumed, a wide recovery may occur in the first two hours. If this occurs, it's a self-limiting condition that is maintained over the 72-hour time frame.

In the schematic in the middle of the slide, its AC power loss, but DC power remains available. And the main difference in the event progression occurs at 24 hours when the module protection system timer actually BCCS in module transition from the decay heat removal system to the emergency core cooling system cooling.

And then finally at the bottom of the slide, if neither AC nor DC power are available, then the emergency core cooling system valves are actuated early in the transient progression due to the assumed loss of power.

And this occurs while the reactor coolant system is at high pressure conditions, and the inadvertent actuation block actuates to hold the emergency core closed and valves closed while decay heat removal system cooling is established.

MEMBER MARCH-LEUBA: Sorry, can you go ahead and tell us, the last one, no AC, no DC, what tapes the ACC is about opening?

MS. MCCLOSKEY: The assumed loss of DC

power -- actually it's the ECCSL.

MEMBER MARCH-LEUBA: So it's an instantaneous opening of the solenoids and the I -- whenever you reach the IAD pressure it will open.

MS. MCCLOSKEY: Yes. Yes. So the loss of DC power, or power in the solenoid --

MEMBER MARCH-LEUBA: Okay. Got it.

MS. MCCLOSKEY: Yeah. Okay. And the decay heat, still on that last one, the decay heat removal system cools the module until -- until the module is sufficiently depressurized that the IAD releases. And then the ECC itself opens and the module transition.

So, on slide 14, this covers the same scenarios as previously, but for a look at PSI. And in those cases with power available, the module protection system actuates a reactor trip contained in isolation.

And the normal response of the design is to actuate the decay heat removal system as well. When the containment level is reached, then the emergency core cooling system is actuated and in a realistic event progression, the emergency core cooling system valves are expected to open at that time.

The -- if AC power is lost, but DC power is available, then the event progression is essentially the same. If DC power is lost in addition to the AC power, then the ECCS valves are actuated while the reactor cooling system is still at high pressure condition.

And again, the inadvertent actuation block acts to hold the valves, the ECCS main valves closed until the reactor coolant system is sufficiently depressurized through the break to which the inadvertent actuation block relief comes.

We're ready to do to --

MEMBER MARCH-LEUBA: What we're asking -- sorry. I don't remember, what is the IAB actuation pressure?

MS. MCCLOSKEY: The range is the -- the threshold pressure range is -- and the DCA is 1,000 to 1,200 PSI. And the release pressure range is 1,000 to 1,200 PSI.

MEMBER MARCH-LEUBA: Yeah. So the releases are Delta P, between the core and the containment, correct? Yeah.

MS. MCCLOSKEY: Not technically.

MEMBER MARCH-LEUBA: Okay. Thank you.

MS. MCCLOSKEY: Okay. Slide 14. The

eliminating transient results. I'm sorry, slide 15. Because some of the limiting Chapter 15 transient results for the various acceptance criteria that are addressed in the Chapter 15 analysis.

In the NuScale design, the maximum reactor coolant system pressure is mitigated by the reactor safety valve. And so there are several events that result in safety valve lift and have similar maximum pressures around 21 steps to get PSIA.

And that's analyzed considering the safety valve set points, and accounting for direct. And it remains well below the acceptance criteria of 2,315 PSIA or 2,520 PSIA depending on the event classification.

In the maximum steam generator pressure, the steam generator design pressure is equal to the reactor coolant, with some design pressure at 2,100 PSIA. And the maximum pressure in our transient analysis occurs after decay heat removal system actuation as the heat transfer for the primary too secondary generate steam and begin condensing in the decay heat removal condenser.

So from a physical standpoint, the secondary pressure is limited to the saturation pressure for the primary site hot temperature. And as

a result in all of our cases, the margin to second criteria is on the order of about 700 PSI.

And for the minimum critical heat flux ratio, the limiting non-LOCA event that's analyzed with VIPRE and the subchannel analysis methodology is the single rod withdrawal case.

And reactivity events, the submitting result is 1.61, which remains above the 9595 acceptance criteria of 1.284.

For the inadvertent opening of a valve, our limiting case is the opening of a recirc valve which then launch with NRELAP5. And the results are shown on the slide here, 1.41 compared to the limit of 1.13.

And then finally for a LOCA we also, the NuScale evaluation model demonstrates that margin to critical heat flux remains and level remains above the top of the core. And so in both of these cases, we continue to show margin.

MEMBER MARCH-LEUBA: And both the IRV opening, and the LOCA, use the LOCA methodology? Which does not use the VIPRE subchannel CHF correlation, correct?

MS. MCCLOSKEY: So -- correct. The VIPRE subchannel methodology is not used for either the

valve opening event or the LOCA analysis.

The Appendix D of the LOCA topical report describes the valve opening methodology and how the margin of critical heat flux is demonstrated there.

MEMBER MARCH-LEUBA: Yeah. One concern that I --

MS. MCCLOSKEY: And it continues to PSI.

MEMBER MARCH-LEUBA: One concern I raised during the subcommittee meeting, and I assume we're still having a closed session Chapter 15, correct, Mike?

MEMBER CORRADINI: Yes.

MEMBER MARCH-LEUBA: The answer is yes. I will talk more about that then, was the differences in MCHFR estimates for this various state by the two methodologies, the VIPRE channel and the LOCA methodology, which were very large.

And indeed this number that you show her, 1.796 for LOCA is higher than the steady state MCHFR that you show later.

So we'll -- just remember that we'll talk about this during the closed session. Keep going.

MS. MCCLOSKEY: Yeah. Okay. All right, that finishes the overview of Chapter 15 for the committee's presentation.

And the next key slides summarize the Chapter 621 containment response analysis. So, going to slide 17, the containment pressure and temperature analysis is performed for a spectrum of primary and secondary sides, mass and energy release events to demonstrate that there's margin to the design pressure and temperature limits considering the worst single loss of failure and loss of the power scenarios.

The containment analysis technical report describes the methodology that we use to analyze the con -- pressure and temperature response. And due to the design of the module with the small high pressure containment and the design of the emergency core cooling valves opening on high containment level or the release pressure, the mass in energy released from the primary is very tightly coupled to the containment pressurization response.

And therefore, we use NRELAP5's model, the integrated response of the reactor pressure vessel blow down and the containment pressurization and heat removal.

The qualification of NRELAP5 for this analysis was based on the qualification and the scale plant modeling approaches that are described in the LOCA and the non-LOCA evaluation models. Because the

evaluation models were developed considering containment pressure as a figure of merit from the codes developed for VDM.

And therefore for the containment sent out, the overviews are based on the LOCA and the non-LOCA event analysis models with the initial boundary condition modified to biased mass and energy releases from the reactor pressure vessel to maximize the containment pressure and temperature response.

Our limiting results for peak pressure is a recirculation valve opening event. And the maximum pressure of 986 PSIA is just below the design pressure limit of 1050.

And the maximum temperature event is a break in the reactor coolant system injection line, with a maximum core temperature of 526 degrees Fahrenheit less than the 550-degree design temperature.

The slide 18 summarize the peak heat of events. And shows the containment pressure for the maximum pressure case.

So at time, this is initiated by the inadvertent research valve opening at time zero with assumed loss of both AC and DC power. Which results in a reactor trip and secondary site isolation.

Very rapidly you get a containment pressure signal. And the plot shows the containment pressure increases rapidly due to the blow down through the valve.

And when the IAD release pressure is reached, the ECCS valves open at 74 seconds into the event. And the containment pressure increases rapidly from that point including the additional mass and energy release.

And peaks around 91 seconds after the event initiation. That pressure is rapidly turned around as the energy is absorbed into the containment wall.

After 700 seconds in this case, heat transfer to the reactor pool is established. And the heat removal through the containment wall to the pool exceeds the decay and residual heat transfer through the emergency core cooling valves to contain them.

It's after this point there's no driver for any subsequent increase in keeping the heat times crossed. And the pressure is reduced below 50 percent of the peak value within the first hour of the event initiation.

All right, hearing no questions, we'll move onto slide 20 to summarize the long-term cooling

analysis results.

In the NuScale design there are different events and analysis assumptions that can result in emergency core cooling systems cooling. Obviously the LOCA and the valve opening events initially transition to ECCS cooling.

And for long-term cooling, the full bright spectrum is evaluated, as well as the valve opening event. If a loss of power is assumed, the non-LOCA event will transition from decay heat removal system to ECCS cooling.

And the non-LOCA events specifically considered are steam generator tube failure, because that event results in a decrease in RCS inventory. And therefore there is a lower inventory event on the reactor coolant system when emergency core cooling system valves open.

And so we evaluate that to demonstrate that there is sufficient inventory maintained to support ECCS cooling.

We analyze a loss of feedwater flow because that's a heat shift event. And so there's a small increase in reactor coolant system energy due to the loss of cooling.

And certain events didn't have any

significant impact on the long-term cooling through ECCS. And we also considered a generic decay heat removal system cool down.

To find the long-term cooling calculations we -- the transient calculations were performed for 12 and a half hours after emergency core cooling system actuation. Which is sufficient to capture the minimum level of what occurs during the long-term phase, and demonstrates a long-term level of recovery.

To evaluate the 72-hour design basis coping period, the state point calculations, we ran state point calculations to establish the quasi-steady condition. And that's really a function of decay heat.

For those state point calculations the decay heat was set to a constant value in the RELAPS model. And the models run until the temperatures are converged.

There are several different acceptance criteria that are of interest in the long-term cooling analysis. And in general to evaluate these different acceptance criteria, we have two different biasing approaches.

In the minimum level cases, these are performed to demonstrate that liquid level remains

above the top of the core. Because in the long-term phase, the minimum level occurs in scenarios where the containment heat transfer is maximized, and the reactor coolant system energy is maximized.

And this results in the maximum amount of vapor generated in the core that needs to be vented through the vent valves, increasing the Delta P over the vent valves.

And this requires the highest liquid level in containment to maintain sufficient driving tests to balance since that pressure drop results in a minimum level over the top of the fuel.

The minimum temperature cases are evaluated to demonstrate margin to boron precipitation limits. And the maximum temperature cases are performed to demonstrate that the under conditions of high decay heat and high cool temperatures, and lower cool levels, the containment provides sufficient heat removal to maintain decrease in climbing temperatures.

MEMBER MARCH-LEUBA: Let me interrupt you here. You keep talking in this slide about boron precipitation, like boron plating. A different mechanism that has been proposed, doesn't result in plating or precipitation, but it results in homogeneous distribution of boron between the core and

the containment.

And what I mean by -- and this applies more to Chapter 20 when the modules will be under ECCS operation for weeks at a time. And so.

So, when you're in ECCS operation, you're steaming steam through the upper valves, the RSVs or RVVs. And that steam is essentially the steam water. It does not have any boron or any boron to speak of with them.

So you are dumping distilled water into the containment. And one place or another, you are concentrated boron in the core.

Now, after a couple of weeks of this operation, you will have essentially pure distilled water in the containment, which makes the down comer to be pure distilled water.

The riser will have a higher concentration of boron, whether liquid or solid, I don't know. And then the possibility exists that that cold distilled water in the riser will, during one of these passes, will make it into the core and regain criticality.

So, a concern I have, I'm not as much concerned about precipitation as the homogeneous distribution of the boron concentration or -- in the core is what I'm going to worry about.

But, clearly after you operate for two or three weeks on the distilled mode, you have clean water on the containment and the down cover. And it's just waiting for any transient to make it into the core.

That's the concern I have more than plating. I know you guys have done some boron transport analysis that you showed us.

MS. MCCLOSKEY: Um-hum.

MEMBER MARCH-LEUBA: I just wanted to put into the record that the homogeneity is on the boron distribution after a couple of weeks of distilled operation, need to be a list of, a list for that.

And you are -- you -- you're driving too much precipitation as being the only problem. But, there are other ways to put high reactivity into your core. Which are unlikely to happen.

MS. MCCLOSKEY: Well, if we're operating in ECCS cooling conditions for long-term, because our decay heat rates are very low, then the recirculation rates from the containment is the down comer into the bottom of the core are also quite low.

And so under those conditions, we have not identified any realistic mechanism that can result in a large slug of dilute water entering the bottom of

the core and causing a return to power.

MEMBER MARCH-LEUBA: But then yes, describe one for you. I mean, it -- granted decay heat is not that much. But in all the safe reactions we had, it was sufficient to melt the core.

So there's energy in there. And you are operating at one --

MS. MCCLOSKEY: Well, we have water over the core. So, --

MEMBER MARCH-LEUBA: Right. You're not going to melt the core. But you have 1 percent at the beginning and then you have 0.5 percent. And then you establish at some level, which is non-reachable, you're boiling a lot of water.

And all that water ends up in the containment, is distilled water. Okay. And you're then driving it into the down comer over weeks, right.

And this is not two hours. This is a week or two. And eventually the steady-state condition, if you continue to operate distilled water and put it in the containment, the containment will be distilled water.

The down comer will be distilled water. And the riser will have an increased concentration of boron. Now hopefully that boron is concentrated in

the core.

But, if you've been flushing on the top of the riser like some calculations show, you might be concentrating the boron in the top of the riser and it stays there, dissolved.

So, there are -- there are -- go ahead.

MS. MCCLOSKEY: What we've seen in the literature for large scale tests is that there's no significant gradient of boron concentration in large scale tests.

We've also considered the -- if you postulate the boron that's concentrating in the riser above the core, then the fluid density in that region would increase due to the boron concentration.

And we considered the really tailored instability mechanism to evaluate how large of a concentration gradient could develop with the type of geometry in from the riser.

And based on that analysis, we are seeing only a very small grad -- a concentration gradient that could develop.

MEMBER CORRADINI: I don't understand the last thing you said. Could you repeat that please?

MS. MCCLOSKEY: We considered -- if you're concentrating -- if you postulate that boron is

concentrating above the core region some place, then the fluid density would increase slightly due to the boron concentrate, the difference in boron concentration there.

And you have either postulating heavier fluid, a more dense fluid over the top of lower dense fluid with a slightly lower boron concentration. And that's an inherently unstable situation that would tend to resolve itself based on the raised tailored instabilities.

MEMBER CORRADINI: So, you're basically saying that you would create a circulation pattern that would remix?

MS. MCCLOSKEY: Yes.

MEMBER CORRADINI: Okay. All right. Thanks, I didn't appreciate what you were saying. Thank you.

MEMBER MARCH-LEUBA: But my point is, let me give you the core and riser will be uniformly mixed. And through the raise stability you have circulation.

And your boron in the core and riser will be uniform. The boron in the down comer and containment will eventually go down to zero. Because there is nothing there to replenish it.

And there are mechanisms to remove it. Which is making it -- driving it through the core. If you have any reservation, like for example a return to criticality, when you suddenly increase the flow through the core, you will sack, cold, and break the water from the down comer. And that will give you a positive feedback.

MS. MCCLOSKEY: I think the following scenario --

MEMBER MARCH-LEUBA: I'm just saying that there is --

MS. MCCLOSKEY: The scenario that's being postulated that would cause such a large probation, because --

MEMBER MARCH-LEUBA: A return to power?

MS. MCCLOSKEY: We're shut down at the -- we're shut down at this point and we're concentrating boron in the core and riser regions, which will prevent a return to power.

MEMBER MARCH-LEUBA: Yeah. I'm just -- my -- I just wanted to put it on the table. That there is a possibility.

I mean, there is clearly, the design of the ECCS results in a nonhomogeneous distribution of boron across the vessel and containment. And high

reactivity water concentrates at the core inlet.

It may just be the down comer. It may not be the lower plenum. The potential exists that it has to be looked at. That's what I'm saying there.

You're going to just dismiss it that's impossible because it is there. It's waiting to happen.

MR. RAD: Yeah, well we have looked at it.

MS. MCCLOSKEY: I think that I --

MR. RAD: Yeah, I'm sorry. Meghan, this is Zack. Yeah, so we're sensitive to your concern. And I don't think anybody is saying it's not something that should be considered or looked at.

In fact we've spent a lot of time looking at it. And a lot of discussion with the staff on it. I don't think we're quite resolved on agreeing on all of the assumptions.

But it's certainly something that we're looking in. So, it may not show up on this slide, but it's certainly a topic of discussion with the NRC, and something that we've been spending a lot of time on.

MEMBER MARCH-LEUBA: As I said before, if you thought of the problem, you're 90 percent over the solution. The problems happen when you have not thought of what would happen.

MR. RAD: right.

MEMBER MARCH-LEUBA: And we need to be thinking about that. And this, we'll discuss this issue during the Chapter 20, a little longer, because I certainly believe that reactivity, subcriticality, now there's a criticality if there's a down should be one of the safety goals.

And we keep saying it's not. I mean, you keep saying it's not that concerned, because this cannot possibly happen.

There are physical forces driving you towards a return to power. There is cold un-braided water in the region of your core.

We'll discuss more of that. As I said, this happens when you're abating in these conditions for two, three, four, or five weeks. You start concentrating un-braided water, a lot of it.

All right. So we'll follow up later.

MS. MCCLOSKEY: Okay.

MEMBER MARCH-LEUBA: Keep on going.

MS. MCCLOSKEY: I think we are -- we go to slide 21. This slide presents results for the minimum level case for the injection line break.

And the plot shows the riser collapse level as a function of time for both the 100 percent

injection line break, and a 5 percent injection line break.

And for the 100 -- for the larger break the emergency core coolant system actuates early in the transient. And then as the inventory redistributes the biasing conditions to maximize the reactor coolant system energy and minimize the emergency core cooling system valve capacity, results within a higher pressure difference between the reactor pressure vessel and the containment vessel.

And that requires a higher driving head and containment to maintain circulation and continue venting vapor from the reactor pressure vessel to containment.

So for this case, in long-term, the minimum level occurs between three to five hours after the emergency core cooling system actuation.

For the smaller break, per the LOCA evaluation model that NuScale has developed, the decay heat removal system is not credited. And therefore there's an extended time of break flow into the containment before the ECCS valves are calculated to open.

And that minimum level, that's contained in the 5 percent break occurs during the short term

LOCA phase as the emergency core cooling system actuates. And then longer term, the decay heat is lower. And the long-term cooling level disk is less pronounced compared to the large break scenario.

But for both break scenarios, the long-term level recovers towards equilibrium as the pressure difference between these reactor pressure vessels and the containment reduces with decay heat.

Going to slide 22, this slide presents results from the injection line maximum temperature case to demonstrate that even under conditions with low reactor pool level assumed and high pool temperature.

The containment provides effective cooling. And the clotting temperatures continue to decrease long-term.

In these cases, the system pressure is running higher than they do in the minimum level cases. And the level depression during the long-term is not observed.

So, slide 24, the last section of the presentation discusses Section 15.06 and return to power analysis. The NuScale design includes an exemption to the GDC-27.

So we have principal design criteria 27 to

ensure that safety related reactivity control system is designed to achieve and maintain a subcritical core. And ensure that fuel integrity for an extended over cooling in combination with a partial cell of reactivity system, namely of stuck rods as assured.

The compliance with PDC-27 is demonstrated with three parts of analysis. One the immediate shutdown from hot conditions is -- the control rod self-sufficient work to protect the reactor coolant pressure boundary. And SAFDLs takes the margin for the worst rod stuck out of the core.

Two that cold shut down conditions are achieved with all control rods fully inserted. And third, that the loss of shutdown margin consequences for a case where the highest worth control rod is stuck out of the core, that those consequences are benign.

And that's demonstrated by showing that the return to power does not challenge SAFDL, and the critical power level is sufficiently low that the decay heat removal system, or the emergency core cooling system continues to provide sufficient removal capacity.

And then finally, the NuScale exemption request describes the probability that the combination

of conditions that results in a loss of shutdown and return to power with a single rod stuck out of the core is a small less stained one E to the minus six per reactor year.

MEMBER MARCH-LEUBA: And that's -- hello, hold on a minute. Does your PDC-27, we learned the other day that there is a requirement that every single transient that could result in sort of pow -- in a return to power, which is all of them, cannot have -- must not have CHF violations, even at the beginning of the transient.

Do you remember that discussion?

MS. MCCLOSKEY: Yes.

MEMBER MARCH-LEUBA: Yes. That was a surprise to me. Because I was not aware of it. So, is that part of our PDC-27? Or is that part of your analysis? Second one raised.

MR. RAD: I can answer that question for you Meghan. Yeah, so that's built into our licensing basis. It's a function of the acceptance criteria and the methodologies we've chosen.

So at this point we don't have methodologies that analyze transients post CHF to demonstrate that there is no fuel failure.

So that's the natural limit based on the

methodologies that we use. And so it is a functioning of our licensing basis.

There's the potential that we could analyze further in the future, and understand what happens to the fuel post-CHF. And then analyze any subsequent consequences of a potential postulated return to power.

MEMBER MARCH-LEUBA: You can also go into my next question. Which is, you're happy today with this analysis, because you can do it.

But let's say you do a power upgrade next year. Then you have -- you will have to come back and get another exception, because I believe the exception includes that requirement.

MR. RAD: Our exempt -- or our goal is to not have that specific requirement be in the PDC. But to have it be part of the licensing basis subject to NRC review it changes in the future.

MEMBER MARCH-LEUBA: And what does the staff think about it?

MR. RAD: I'm not sure what the final wording is. But I think we're pretty close to alignment.

MEMBER MARCH-LEUBA: Okay.

MEMBER CORRADINI: We should ask the staff

when they're up.

MEMBER MARCH-LEUBA: Yes. Because it really should be part of the PDC. And if it is a requirement, it's like one.

And it should be harder for you to remove it in other words.

MR. RAD: And it is an administrative exercise to request a separate exemption. But, there's also a lot of work to do to change our methodologies, to get NRC approval for that as well.

MEMBER MARCH-LEUBA: Well, okay.

MS. MCCLOSKEY: Okay. Going to slide 25 now. In the DCA submittal, we had a bounding dynamic return to power analysis to demonstrate that there was no challenge to minimum critical heat flux ratio limits, the reactor coolant pressure boundary, or the containment boundary.

And as we've worked through the DCA review, we've worked to develop an alternate approach to analyze the return to power analysis that allows for a clearer demonstration that the range of conditions under which a return to power could occur, are analyzed.

And so this slide and the next demonstrate that the new app -- the approach and it's consistent

with what was presented to the subcommittee last month.

The mechanisms that lead to a loss of shutdown margin and return to power are first moderator cool -- overcooling, which can be postulated from either decay heat removal systems cooling, or emergency core cooling systems cooling.

And under cold reactor pool conditions particularly, either of these systems can cause a fairly rapid temperature decrease in the first several hours after the event. Which causes the reactor reinsertion from the increased moderation.

And then second, xenon decay after shutdown cause a slow post-shutdown reactivity insertion. So for the return to power characterization, NRELAP5 is run for several state points with a specified power level to characterize the steady-state reactor coolant system temperature and a function of that power level.

And we've run those state points for decay heat removal systems, cooling with the riser remaining covered, decay heat removal system cooling with the riser uncovered, and core emergency core cooling system cooling.

And then simulate is run to calculate the

critical power level assuming the worst rods stuck out as a function of reactor coolant system temperature. And where these state points from the NRELAP5 and simulate calculations intersect, gives the power level and reactor coolant system temperature where a return to power could occur.

And the conclusions from this analysis are summarized here. Currently based on best system, it simulates results. But a loss of shutdown margin may occur around 40 hours with, if you're assuming the third one in the RCS. And so end of cycle conditions and moderator temperature feedback when this occurs after xenon decay.

The resistant coolants in temperatures are quite low. Based on best estimates, simulate results less than about 200 degrees Fahrenheit.

ECCS cooling can drive the module to lower temperature conditions. And therefor slightly higher power results.

There are -- if you have a sufficiently high pool temperature, this limits -- it is a self-limiting nature of the design limit goal the module temperatures will reach. And that can preclude a return to criticality with worst stuck rods.

But our power levels remain very low,

around 1 to 2 percent. And at pool boiling CHF evaluation demonstrates the parts into minimum critical heat flux ratio.

MEMBER MARCH-LEUBA: How confident are we, or basically let me rephrase the question. How we run some uncertain analysis on that 1 to 2 percent power level, could it be 15 to 25 percent?

How do we know it's not 15 percent power?

MS. MCCLOSKEY: We are placing analysis that consider uncertainties in the simulated results. And what that shows, is it shifts the critical power level.

It shifts the temperature up about 25 degrees. Which results in full power level that's still less than 2 percent.

So we're nowhere near 15 or 20 percent power level. Even the bounding dynamic return to power analysis that was performed in the DCA system, I think the peak power was approximately 14 percent under that, the conditions of those analyses.

And that was based on a RELAP analysis and point kinetics modeling.

MEMBER MARCH-LEUBA: Yes. Which use a constant MPC value for the whole analysis, correct?

MS. MCCLOSKEY: Yes.

MEMBER MARCH-LEUBA: Yes. Which is very, very valid core submission. So, the simulate is a much better approximation.

Okay. Thank you.

MS. MCCLOSKEY: Okay. Slide 26 shows the results from this return to power characterization. And again, this is what was presented to the subcommittee.

The X axis is the critical power level. And the Y axis is the reactor coolant system average coolant temperature.

And the orange, blue, and purple dots are the results of the NRELAP5 state point analysis, where we specified the power level in NRELAP5, and calculated the steady-state reactor coolant system temperature for either DHRS or ECCS cooling.

And the black line is the critical power level calculated by simulate as a function of the reactor coolant system temperature. And so for ECCS cooling, the critical power level in these cases was about 1 percent power, because the lower temperatures are reached.

But there's -- at the cold, low power conditions there are significant margins of CHF and the emergency core cooling system or the decay heat

removal systems have sufficient capacity to continue cooling the module.

MEMBER MARCH-LEUBA: Yeah. The concern we have with this figure is related to my previous question of uncertainties.

That black line, if you have chosen to put temperature in the X axis and power in the Y axis, that black line is almost vertical. Or in this case, almost horizontal.

And a very tiny change in temperature results in a very large change in critical power. If you've run any sophisticated analysis with that type of sensitivity, you will find that -- it's a very large degree with it. You better not miss it by much.

Anyway, just an observation.

MS. MCCLOSKEY: We've got a -- yeah, we've got a fairly tight frame on the X axis as well here. And what we've seen is if you -- if you shift that simulate line up by about 25 degrees, then the ECCS critical power level increases to about 2 percent power.

MEMBER MARCH-LEUBA: Okay. Keep going.

MS. MCCLOSKEY: That concludes the NuScale Chapter 15 presentation.

MEMBER CORRADINI: Questions by the

members?

MEMBER RICCARDELLA: What is the relative peaking then for the bundles that don't have the -- for the stuck rod assembly?

What's the relative peaking radially primarily for those assemblies versus the average? I think what you're presenting here is the average power.

MS. MCCLOSKEY: We'd like a couple of minutes to discuss, please.

MEMBER CORRADINI: You can also come back -- we can also come back in closed session or later on.

MEMBER RICCARDELLA: Yeah.

MEMBER CORRADINI: You don't have to sit here and wait. Why don't you take time to look for the response and we'll come back to it.

MS. MCCLOSKEY: Thank you.

MEMBER CORRADINI: Okay. We're moving onto Chapter 20, is that correct? And then we'll take a break as a reward.

MS. MCCLOSKEY: Okay. Thank you.

MEMBER BLEY: Are we taking a break?

MEMBER MARCH-LEUBA: Not officially I don't think.

MEMBER BLEY: Okay. I'm going to see if we can get some coffee then.

MEMBER CORRADINI: If you guys are ready. Go ahead.

MS. JOERGENSEN: Good afternoon. My name is Nadja Joergensen. I'm a licensing engineering with NuScale Power.

Next to me is Zack Rad, Director of Regulatory Affairs. And with me I have Chris Maxwell to do the majority of the presentation on Chapter 20, mitigation strategies and beyond design basis external events.

With that I'll turn it over to Chris.

MR. MAXWELL: Good afternoon. The new Rule 10 CFR 51.55 separated into requirements for mitigation strategies for design basis events, which is covered by paragraph 501. And requirements for a spent fuel pool level indication, which is covered by paragraph Echo.

The initiating event is not defined by the Rule, but the outcome is. And that is that the results in an extended loss of AC power concurrent with the loss of normal access to the normal heat sink for a passive design such as NuScale's.

And the Rule requires that the three key

safety functions of core cooling, containment, and the spent fuel pool cooling will be maintained throughout the ELAP.

To evaluate the NuScale power plant scope and capability, it's necessary to establish first what the minimum installed coping duration is. NuScale considers a 14-day minimum coping period to be sufficient to establish an alternate means of removing heat.

And this duration is based on the operating experience from Fukushima where at that event, without a preplanned strategy or staged equipment, or a hardened makeup line and with access to the site significantly restricted by the earthquake and tsunami, they were able to add water to the Unit Four spent fuel pool after nine days using offsite resources.

And were able to restore onsite capability to inject water to the fuel pool using the fuel pool cooling system at the 14-day point.

Beyond this minimum coping duration, the continued use of installed plan equipment, as well as ad hoc resources and equipment repairs, could be used to continue coping indefinitely.

Now, I'd like to talk about the NuScale

power plant's response to an ELAP current with the loss of normal access to the normal heat sink, and the status of the three key safety functions during that event.

As we just mentioned, NuScale power plant was informed by the events at Fukushima and sought to provide a design that would cope during an ELAP without the use of AC or DC electrical power, without a need for inventory addition, or supplemental onsite equipment, without the use of offsite resources, and without a need for operator action. And therefore without a need for operator monitoring during the event.

And in short, the extended coping duration is provided by the design through automatic response of installed plant equipment alone. As I mentioned in the subcommittee, the strategy provides a significant advantage of allowing the operators to focus on restoring site capabilities, including electrical power rather than the deployment of resources, equipment, and strategies, just to address the three key safety functions.

So, now we'll look at the response of the NuScale power plant to the extent of loss of AC power, concurrent with the loss of the normal heat sink,

without operator action and without the use of offsite resources.

So, within one minute of the event initiation, a reactor trip, decay heat removal system initiation and containment isolation are received by all modules. This makes the reactor subcritical and places in service the decay heat removal necessary to establish safe shutdown conditions.

This establishes the core cooling and the containment functions. Twenty-four hours into the event, the module protection system de-energizes the trip solenoids for the ECCS valves, allowing those valves to open and placing the emergency core cooling system into service, again, at that 24-hour point.

With this, each of the modules is in safe shutdown with passive decay heat removal and containment cooling, while the spent fuel in the spent fuel pool continues to be passively cooled by the inventory of the ultimate heat sink.

During this entire period, the ultimate heat sink is receiving heat and is heating up. And after more than five days, the ultimate heat sink will begin to boil. At this point, the ultimate heat sink level is lowering.

And again, with no operator action, so no

operator actions will lower to 45 feet in the ultimate heat sink after more than 50 days. Now 45 feet is a level that's just below the top of the DHRS passive condensers.

And during this entire period again, the core cooling and containment and safety functions are maintained. Forty-five feet, the significance of this is, there's no significance regarding the ability to remove core heat or to cool the containment. This is just where the long-term cooling analysis stopped. It was suspended at 45 feet.

MEMBER MARCH-LEUBA: And just for the record, the ultimate heat sink during all this 50 days, is really the environment, because you're venting steam out of the building. Is that correct?

MR. MAXWELL: That's correct.

MEMBER MARCH-LEUBA: Through a, can you say in open session, how you vent it?

MR. MAXWELL: We have rupture, a safety-related rupture disk in the reactor building.

MEMBER MARCH-LEUBA: Thank you.

MEMBER BLEY: It just starts boiling at five days.

MR. MAXWELL: That's right.

MEMBER BLEY: So, how many days later

before you actually pop this blow out test?

MR. MAXWELL: Oh, we didn't analyze that for this piece of it. Instead we assumed that into our passive cooling calculations we assumed that we retained all of it because it's a worst case scenario to retain that heat in the building, causing the rooms adjacent, the module protection system rooms to heat up.

MEMBER BLEY: To heat up. Okay.

MEMBER MARCH-LEUBA: And because in reality, it's a very large building that is cold.

MEMBER BLEY: Yeah. It would take quite a while.

MEMBER MARCH-LEUBA: Yeah. My concern is not losing the steam. It's what would happen to all the instrumentation and all the electricity that is not flowing through there?

MEMBER BLEY: Right.

MR. MAXWELL: And again, the DC equipment is in rooms that are outside of the pool area. They're in rooms that are adjacent to the pool area, but not in communication with the pool area themselves.

MEMBER MARCH-LEUBA: Are they isolated? I mean, they're -- I mean, I really want condensing

humidity on all my electric systems.

MR. MAXWELL: Right. But the --

MEMBER MARCH-LEUBA: But since you lost power, you don't really care that much.

MR. MAXWELL: Right. So again, the core cooling and the containment functions are maintained during this period.

But with respect to the spent fuel cooling function during this, we said that heat continues to be added to the ultimate heat sink, and the level continues to lower.

And after about more than four months, level will reach the bottom of the opening in the weir wall between the spent fuel pool and the other pools of the ultimate heat sink.

And that point's significant, because before this point the reactor pool and the refuel pool are acting as a makeup source to this spent fuel pool as it's evaporating. And the, again, up until this point they act as a single volume.

Beyond this, now only the spent fuel pool volume remains to cool the spent fuel in the spent fuel pool. And it requires still another month, or more than five months total for a level to lower to the spent fuel -- to the top of the spent fuel in the

spent fuel pool. Again, assuming no operator --

MEMBER KIRCHNER: Excuse me, for the record Chris though, is that sent fuel in the pool at the maximum capacity? Or for this calculation?

MR. MAXWELL: This calculation goes beyond our actual maximum capacity.

MEMBER KIRCHNER: Okay. Right.

MR. MAXWELL: That's 18 years' worth of spent fuel. This assumes that the reactors were at 102 percent power, all of them when the event initiated.

None of the inventory that evaporates off returns to the pool. All very conservative conditions when calculating this pool boil off.

MEMBER MARCH-LEUBA: And just a reminder that 18 years of spent fuel is according to nine cores of spent fuel per core. So you have nine times more fuel in the spent fuel then you do in the cores.

And those are real old. But still, this spent fuel pool will produce more heat --

MR. MAXWELL: That's correct.

MEMBER MARCH-LEUBA: Then the fuel cores themselves, right?

MR. MAXWELL: That's absolutely correct. And the pool boil off reveals that. And that's why it

was suggested that the other two pools of the ultimate heat sink act as the makeup source. Because it is the spent fuel pool that's driving the reduction in level.

MEMBER MARCH-LEUBA: Okay.

MR. MAXWELL: The significant driver. And the next slide, please?

During the subcommittee there was some discussion about a shutdown margin. And we wanted to present this information.

I want to point out that this data doesn't come from a -- it's a preliminary calculation. And it doesn't come from a docketed analysis.

But what this is showing us here on the Y axis is we have boron concentration. And on the X axis is a time to fuel cycle.

So the blue line represents the nominal boron concentration during operation over the cycle. And I'll go straight to the green line, the lower line, which is criticality at a temperature of 420 degrees, or safe shutdown.

So you see the margin from between the blue line and the green line to reach criticality. And the red line represents 4-0 degrees, or 40 degrees to reach criticality.

So you can see at the beginning of the

cycle you have to dilute more than 600 ppm to reach criticality.

MEMBER MARCH-LEUBA: And you didn't do a very good job explaining what that is. Is time to deliver.

MR. MAXWELL: Okay. Y axis, boron concentration. X axis time in the cycle. And my blue line is my normal operating boron concentration.

MEMBER MARCH-LEUBA: That's an AF15 of 1.0?

MR. MAXWELL: That's correct.

MEMBER MARCH-LEUBA: Okay. So what's the green line?

MR. MAXWELL: The green line is with all rods in, and cooled down to 420 degrees.

MEMBER MARCH-LEUBA: Okay. So the difference between green and blue is the role off and whatever the temperature is. And that will depend on whether it is the beginning or end in the cycle.

MR. MAXWELL: That's correct.

MEMBER MARCH-LEUBA: What is this?

MR. MAXWELL: You see at the beginning of the --

MEMBER MARCH-LEUBA: Oh, is this time of cycle?

MR. MAXWELL: It is. That's our X axis. And again, the difference there, the red line is, instead of 420 degrees or safe shutdown, the red line is 4-0 degrees, 40 degrees in RCS picture.

MEMBER MARCH-LEUBA: And this is with all rods up, or all rods in.

MR. MAXWELL: Yes, all rods in. That's correct. Which is one of the boundary assumptions in ELAP space.

MEMBER BLEY: Your blue line to the right side of the curve is that's essentially the end of cycle?

MR. MAXWELL: That's correct. Although we -- the X axis is from beginning of cycle to all the way to the end of cycle.

And you can see that the other two lines stop because the boron concentration essentially would become, you know, give you a negative boron concentration. That's why those lines terminate. Okay, next slide.

The next portion of the Rule is paragraph Echo, which address spent fuel pool level, or correction, spent fuel pool monitoring.

The requirement is to provide reliable, remote indication for five years after the last fuel

that's in the spent fuel pool was used in an operating reactor.

The NuScale ultimate heat sink system includes four remote level indications for the following, for the reactor pool, the refueling pool, and two for the spent fuel pool.

These instruments are seismically qualified and environmentally qualified for the conditions in the pool area. And they're normally powered by our highly reliable DC power system via the plant protection system.

They also include a replaceable battery power source, which is independent of the plant AC and DC power systems. And that replaceable battery source has a known capacity of 14 days.

MEMBER MARCH-LEUBA: And we discussed this, this -- the readout is accessible during a severe accident, correct?

MR. MAXWELL: That's correct.

MEMBER MARCH-LEUBA: We still don't know where they are, but they're not in the pool.

MR. MAXWELL: Yes, sir. They're just -- that's the one thing we know for certain, is they aren't in the pool area. They're remote from that location.

MEMBER PETTI: As well as the battery.

MR. MAXWELL: That's correct. Both the battery and the indication, correct.

MEMBER MARCH-LEUBA: Oops, forgot that one.

MR. MAXWELL: So, in summary again, the two parts of the Rule, the first part is the mitigation strategy.

NuScale's mitigation strategy is to rely on the automatic response of permanently installed safety-related plant equipment to establish and maintain those three key safety functions of core cooling, containment, and spent fuel pool cooling.

And to provide an extended coping capability superior in the 14 days. Again, just to point out that the -- question?

CHAIR REMPE: Well, I want to finish what you were saying. And then --

MR. MAXWELL: That just a reminder that the mitigation strategy itself doesn't rely on AC or DC electrical power inventory addition or require any operator action to establish and maintain those safety functions.

And regarding paragraph Echo and our spent fuel pool level indication strategy, we're relying

upon our installed instrumentation with a 14-day batter back up power supply.

CHAIR REMPE: So, I'm thinking back on the days from when we were dealing with the Near Term Task Force. And Steve was around then too. The intent I thought, the stuff I've been reading has reminded me of the fact that the intent of the rule was to try and give the plants an option to deal with something they didn't expect.

And because there's uncertainties. And so we've got this stylized event. And you seem to have a lot of confidence, and we didn't try to explore, you know, where's the big uncertainties in some of the things that you've presented for this scenario.

But, I think at the beginning of your discussion you were saying you were trying to satisfy the intent of the rule. And I'm just kind of wondering, and again, I seem to harp a lot about instrumentation and operators not having data if we're taking away some flexibility to the operator.

So, I mean, you say yeah, this stuff will be there. But they're not supposed to do anything. But again, how much confidence do we have after 72 hours, until the 14 days that this instrumentation will really be there?

Because that's really, you meet the requirements of the Rule, but have we met the intent, of the Rule which hasn't been published by the way, you know?

MR. MAXWELL: I would step it back a step. And say that what we were looking to do, and we're talking about FLEX and the strategies that went into that.

Was that what we saw was that that was the response to the operating experience at Fukushima. So we said, we don't want to be in a situation where we require this FLEX equipment.

And I don't mean to be flippant about this, but rather than a Band-Aid, we want the design to address the conditions. We went through the use of safety-related equipment, and only safety related equipment, to be able to respond to the conditions that are postulated by this event.

So, again, not knowing the initiating events, but knowing the results of that that must be addressed, that's what the design did. It provided a, without an operator action, without a requirement for operator response, the passive ability to shut down the reactor.

To establish and maintain core cooling.

To cool the containment and to cool the spent fuel and spent fuel pool for an extended period of time, extended coping duration.

I'm very sensitive to the other piece of it, what the operators will want to do as well.

CHAIR REMPE: Right. You said you were an operator.

MR. MAXWELL: Absolutely. And so I have a high level of confidence in the beyond the 72 hours. We've talked about 72 hours.

But we have redundant, 100 percent redundant, highly reliable DC power system, which is another 72 hours on top of it just from our installed highly reliable DC power system.

We've got redundant back up diesel generators that can supply all of the charges for that system. We've got an auxiliary AC power source.

The instrumentation, I can't speak into too much detail about it, but I can say that the Chapter 15 events are more bounding then the ELAP event as far as the conditions that the instrumentation experiences.

And that they're required to be qualified for a far greater period then 14 days. And beyond. So, with that, I have a high level of confidence that

the operators will have information available to them.

But what they -- what I want to emphasize is that it's just a check that the key safety functions are continuing to be maintained while they address the initiating event and establish normal site function and capability.

That's what's really the important part of this strategy to me, is that while those safety functions are maintained, it's getting back to normal. And that's what this allows.

MEMBER RICCARDELLA: Is there any concern about this return to power scenario, under the ELAP event?

MR. MAXWELL: We have our -- let me ask Meghan to respond to that question. And NuScale, Meghan, can you answer that question for me?

MS. MCCLOSKEY: Sure. I think one of the key things is to consider in terms of return to power, under the ELAP conditions, our operating in ECCF cooling mode, which as we discussed earlier today, will tend to concentrate boron in the core and riser region.

Which will be, with the function of all control rods inserted, will likely prevent any return to power scenario.

MEMBER KIRCHNER: They don't assume a step though.

MEMBER RICCARDELLA: I understand that. Let me --

MR. MAXWELL: Let me -- can I just answer in a licensing perspective?

MEMBER RICCARDELLA: Yes.

MR. MAXWELL: So, our PDC-27, principal design criteria, and the related exemption note that with all rods in, we maintain the reactor subcritical, period.

So, --

MEMBER RICCARDELLA: Regardless of the boron con -- or regardless of where you are in the cycle? And --

MR. MAXWELL: That's right.

MEMBER RICCARDELLA: And the boron concentration?

MR. MAXWELL: That's correct. So the boron concentration is a reflection of where you are in the cycle. And that's why we showed you that graph to demonstrate how much margin we have.

So the postulated scenarios of boron dilution or concentration and relocation, which are a little more complicated then simply distilling. But

the margin to that is exceptional.

So, it -- we demonstrated at 40 degrees, well that's not even the postulated scenario, right? We'll postulate that the spent -- the ultimate heat sink heats up.

And so while we don't have a specific analysis for this, this scenario under ELAP, we wanted to demonstrate what we have based on our Chapter 14 postulations.

MEMBER RICCARDELLA: As long as all rods are in, there's no condition that gives you a return to power.

MR. MAXWELL: That's correct.

MEMBER DIMITRIJEVIC: Can you go to the -- back to slice four so we can discuss your 14 days and the 72?

Because then that's where I have a couple of questions. One is what means ad hoc resources? What do you mean by ad hoc resources?

MR. MAXWELL: That we can procure without prearranged contracts.

MEMBER DIMITRIJEVIC: So basically how you designed that, you don't want to rely on some outside source of agreement. You want to be completely independent of the rest of the world, so.

MR. MAXWELL: That's right.

MEMBER DIMITRIJEVIC: All right. And so my question now, can you arm the water to your pool from fire truck?

MR. MAXWELL: We have an externalized -- the short answer is yes. But the way we do it is we have an assured makeup line to the spent fuel pool, seismically qualified with an external connection on the reactor building, with a standard fire protection thread.

MEMBER DIMITRIJEVIC: Okay. And let's say that you don't get any power. And this with the diesel generator which you mentioned a couple of times, how they cool? Your -- those back up diesel generators?

MR. MAXWELL: There's skid mounted diesel generators with their own cooling water and I believe they're -- they haven't been selected yet, but I believe they're actually air cooled.

MEMBER BLEY: I was going to say, that's perfect.

MEMBER DIMITRIJEVIC: All right. But my other question is, so you don't really get any power at, you know, after 14 days back.

And maybe there is another calculation.

Maybe you splash some water somewhere. Maybe you get some leaks somewhere.

And then you have to worry about -- your main thing here is this pool.

MR. MAXWELL: Right.

MEMBER DIMITRIJEVIC: This is the thing that keeps all of this together. So, if you don't get any power, don't anything change. So, how would you makeup to the pool?

MR. MAXWELL: Well, one of the principal assertions is that our pool is seismically qualified as well. So that -- within the bounds of this event, we don't assume a pool leak.

However, we do have, we've got that external connection. It's a gravity feed to the pool. And though they're not credited, we have onsite resources, fire protection, water.

We've got a pool surge tank that's a million gallons, essentially of water, sitting in a berm. And then there's the --

MEMBER DIMITRIJEVIC: So you will have some like diesel driven fire pump or something?

MR. MAXWELL: We do have a diesel driven fire pump that we don't credit. But we do have one onsite.

MEMBER DIMITRIJEVIC: And you can just pump the water in?

MR. MAXWELL: Because it's gravity, it's gravity feed. So, as long as we're able to deliver to that -- to that assured makeup line, we can. Pumper trucks or trucks were procured ad hoc with water. We can add water to the pool that way as well.

MEMBER BLEY: I hadn't thought about it, but you've talked about this gravity feed so often, on the outside of the building, at what elevation is that hook up?

MR. MAXWELL: At ground level.

MEMBER BLEY: Ground level. So you really could drain from a truck with no assistance.

MR. MAXWELL: Yes, sir.

MEMBER BLEY: And then it's piped to the spent fuel pool?

MR. MAXWELL: To the spent fuel pool. And it's sized to ensure the worst pool boil off rate that it has the capacity in excess of worst boil off rate.

MEMBER BLEY: That's gravity flow?

MR. MAXWELL: Correct.

MEMBER DIMITRIJEVIC: So how is this line normally isolated?

MR. MAXWELL: It has a manual isolation

valve. Again, external to the reactor.

MEMBER DIMITRIJEVIC: And so that was so it can be manually opened?

MR. MAXWELL: Correct. Without accessing the pool.

MEMBER DIMITRIJEVIC: And this would be accessible, you know, even though you've got steam coming out?

MR. MAXWELL: Right. That being external assures that.

MEMBER DIMITRIJEVIC: All right.

MEMBER RICCARDELLA: You might not be the right person to ask, but somebody will. When your seismic PRA, at whatever point it's done, look at the fragility of your pool building, and confirm that the probability of a serious, a serious rupture of that is very, very low.

MR. MAXWELL: Can you answer that question?

MR. RAD: I think once you -- sorry. I believe the answer to your question is yes. But I don't have that right in front of me.

MEMBER RICCARDELLA: Okay.

MEMBER KIRCHNER: Chris, I was looking at your diagrams that you used for the heat sink and the

height of the weir. How did you come about determining that height?

MR. MAXWELL: The height of the weir wall itself?

MEMBER KIRCHNER: Yeah.

MR. MAXWELL: I believe that it's ten feet above the top of the spent fuel rack, the weir wall is. And that's the shielding requirement. Ten feet of water above the top of the spent fuel pool rack at all times.

But that's not an official answer. That's just what I recall.

MEMBER KIRCHNER: No. I was just thinking, you're going to be doing refueling 24/7. The height of that makes the lift and then the transit and drop.

MR. MAXWELL: So we tried --

MEMBER KIRCHNER: And actually you have to be underwater obviously when you're doing a fuel element transfer, but.

MR. MAXWELL: And we calculated that as well. And what we saw was that with the fuel bundle in the weir, in the opening, so its highest point, --

MEMBER KIRCHNER: Right. But you still have ten feet under normal conditions.

MR. MAXWELL: Right. And ten feet for the

pool level to lower within ten feet of the top of that bundle that's in the weir wall is about 74 days.

MEMBER KIRCHNER: Yeah. No, I was just thinking minimalizing the amount of up and down on the crane.

MEMBER PETTI: So, I have a different question. It sounds like there's still some conservatism is built into that cartoon graphic time line.

What's your best estimate?

MR. MAXWELL: We only use the conservative values.

MEMBER PETTI: You've got to really looked at --

MEMBER CORRADINI: They have numbers. I thought you were going to quote, I thought you were going to quote 50/50/150?

MR. MAXWELL: Well he says that's conservatism built into that. And that's true.

MEMBER CORRADINI: At 50/50/150?

MR. MAXWELL: Yes, sir.

MEMBER PETTI: But he talked about, you know, more fuel then would ever be in there. Some of the boiling.

And he had to do -- just all these

consumptions, if you took all those out, --

MR. MAXWELL: Right.

MEMBER PETTI: You know, how much time do you really -- do you think you have in a best estimate?

MR. MAXWELL: And --

MEMBER PETTI: What's the sense of margin?

MR. MAXWELL: Because we looked at what our installed coping duration needed to be, you have a minimum duration of 14 days in.

With all that conservatism built in, we saw that again, nothing changes. We don't lose core cooling at 50 days. It's just the calculation stops there at 45 days.

But we said with that margin there was no reason to go further.

MEMBER KIRCHNER: Just Chris, out of curiosity more than anything, when does the estimated boiling of the heat sink start? How many days out?

MR. MAXWELL: It's about a little more than five days.

MEMBER KIRCHNER: Just beyond five. I was looking at the schematic and couldn't tell. Okay.

MR. MAXWELL: Okay.

MEMBER DIMITRIJEVIC: Based on this graph,

you're okay for five months actually, before you even start to worry about the spent fuel pool, right?

Or that you would never worry about anything else?

MR. MAXWELL: Well, let me address that in a couple of different stages. The first one is that we have a tech spec requirement ultimate heat sink level.

So our operators are still bound by tech specs and working to maintain the ultimate heat sink level within tech specs.

MEMBER DIMITRIJEVIC: But what's measuring this? What's --

MR. MAXWELL: The spent fuel pool level indication.

MEMBER DIMITRIJEVIC: And that will be available during this?

MR. MAXWELL: That's correct.

MEMBER CORRADINI: And that's the new battery there.

MEMBER BLEY: They have those replaceable batteries every 14 days they can stop it now.

MR. MAXWELL: But the -- the two safety, key safety functions of core cooling and ultimate -- and I'm sorry, core cooling and containment are

assured to a level, ultimate heat sink level of 45 feet.

Beyond that, we're talking just about the spent fuel pool key safety function. And it's, you know, again well past four months, 150 days before we reach the top of the spent fuel pool and the spent fuel rack.

MEMBER DIMITRIJEVIC: Well, you're going to lose the job as and the efficiency of this. But you will still have ACCS cooling, right?

MR. MAXWELL: That's correct.

MEMBER DIMITRIJEVIC: So, basically --

MR. MAXWELL: We have unofficial calculations, non-docket information that suggests air cooling beyond 45 -- beyond 50 days is sufficient.

MEMBER DIMITRIJEVIC: Right.

MEMBER BLEY: And from what you've told us, the level will be going down faster in the spent fuel pool then it will out in the main pool.

MR. MAXWELL: That's correct.

MEMBER CORRADINI: Further questions? Okay. Let's take a break. 3:20.

(Whereupon, the above-entitled matter went off the record at 3:10 p.m. and resumed at 3:20 p.m.)

MR. TABATABAI: Goof afternoon, everyone.

Thank you very much for your time.

We'll be presenting to you Chapter 20 and Chapter 6 with this crew up here. We're going to start with Chapter 20. This is the full Committee presentation. We had the Subcommittee presentation yesterday and, basically, you asked to come back with to just talk about mitigating strategies and the staff's approach to review NuScale's mitigating strategies. For today, we will be talking about a little bit of background information for members who were not present yesterday and, also, after that, we'll get right into the review strategies.

I don't want to spend too much time on this slide, just in the interest of time. But the purpose of this slide is to highlight for the members that the review of Chapter 20 has been kind of a dynamic review. We received Revision 2 last year in October. But, since then, there have been quite a few developments. The bottom line is that our SER is based on Revision 2 of the DCA, and the findings that we have in Chapter 20, SER, are limited to 72 hours after initiation of a beyond-design-basis. We will get into more details as Ryan will take over to talk about the staff's regulatory framework and the staff's review.

But, first, Ryan?

MR. NOLAN: Okay, yes. Thanks, Omid.

So, the recently-approved Regulation 5155 for Mitigation Strategies for Beyond-Design-Basis Events does not apply to design cert applicants. However, NuScale is voluntarily seeking the NRC approval to use dome of the installed design features for the mitigation strategies.

Go to the next slide.

Again, this is the rule text for your reference.

MEMBER CORRADINI: Just to slow you down, since I know you're going to once we get you on a roll -- but there are other Design Certification Applicants that have done this.

MR. NOLAN: That's correct. We have in the past reviewed Design Cert Applicants and provided finality on the installed equipment when performing certain capabilities.

MEMBER CORRADINI: All right. Thank you.

CHAIR REMPE: Can I go further? I know the APR1400 was one, but there have been others beyond it? There's more than one?

MR. NOLAN: It's primarily the APR1400.

MEMBER CORRADINI: I think all the rest

predated it.

MR. NOLAN: Exactly.

CHAIR REMPE: Right. So, there's just one.

MR. NOLAN: So, when you look at the orders, say for the AP1000, there were statements -- and I don't recall if it was in the orders or within the SECY paper itself -- that made statements to the effect that AP1000 for 72 hours for the onsite coping capability is sufficient, and that the COL-holder, the licensee is responsible basically for the offsite resources and, then, development of the strategies beyond 72 hours.

CHAIR REMPE: So, there's wasn't a Chapter 20 for the AP1000, but, somehow or another, it got approved for 72 hours, if I ever get asked on this. So, thank you. Okay.

MR. NOLAN: Yes.

MEMBER CORRADINI: I think the AP1000, just to be clear on the record, is the AP1000 had already been certified prior to Vogtle and --

MR. NOLAN: That's correct. So, Vogtle --

MEMBER CORRADINI: They were in between the two operating --

MR. NOLAN: Well, Vogtle got an order.

MEMBER CORRADINI: Correct.

MR. NOLAN: Yes.

CHAIR REMPE: It's really just one CDA Applicant, and that's the APR1400. So, maybe you do have a singular word in my draft letter. Thank you.

MR. NOLAN: Yes, oh, yes.

So, the staff's approach for the Chapter 20 review, there's a few considerations. The primary consideration was that we maintain consistency with the level of detail that was performed for the operating reactors as well as the previous Design Certification where we reviewed this.

So, for an operating reactor, where, say, phase 1, coping with installed plant equipment was on the order of magnitude of eight hours, you know, the use of onsite, portable equipment up to 24 hours may be a little bit beyond that. And so, that's where my bullet here that talks about, you know, the staff really focused on the most critical time-sensitive actions. There wasn't an extensive review performed really beyond 72 hours for the operating fleet.

And so, for NuScale, the staff, what we're planning on doing is looking at the installed design features, reviewing their capabilities and capacities, as well as addressing any transient, credible

transient phenomena, for the first 72 hours. So, that's a review of the design aspects of the SSCs. Any of the operation aspects of mitigating strategies would be deferred to the COL.

And so, speaking of the COL, if there is no credible transient phenomena, we do not expect the COL would have to address any additional design aspects beyond 72 hours. What we would expect to see is the development of the strategies, identification of any resources, and the level of detail of those resources would be commensurate with the time available.

And so, yesterday what I mentioned was, if makeup to the spent fuel pool is needed, we would expect to see identification of where the water is coming from, but we wouldn't expect to see necessarily preplanned contracts in place to acquire that, because of the time available for the site to acquire that resource.

MEMBER MARCH-LEUBA: You might have already said that, but what I'm missing in this review is a set of emergency operating guidelines. At the COL, you will be reviewing the EOPs, the actual procedures for that particular applicant. But we are seeing how a reactor might behave if we decide not to

do anything until it is completely passive, which has nothing to do with reality. In reality, all these transients are going to be significantly better.

I think we all would have been served much better if we said, yes, we have a passive, but here's what really is going to happen at here and my EPGs. And I think that's what you're saying. Is that what you're saying, that to go beyond the 72 hours, you would really like to see the EOPs or, in our case, RPGs?

MR. NOLAN: Yes, that's certainly part of it, but any incredible transient phenomena is a large piece of it as well. We're resolving a lot of those issues as part of Chapter 15, but we're only reviewing it out to 72 hours. I'm not saying, or I can't say, whether it's an issue or not. It's not something that we have looked at. And to do so, to go beyond 72 hours is not necessary to make a finding at this point. Or it could be seen as an unnecessary regulatory burden on a future COL. It could limit their ability to come up with their own strategies. On top of it, we don't know where the site is located. But, as you get further out in time, you get more uncertainties.

MEMBER BLEY: You've got help me with this

unnecessary burden. I don't buy that one.

(Laughter.)

MEMBER BLEY: I come in with a COL. They've already done it for me. That's great. If they haven't or they did and I want to change it, I can still do that. COLs do that. They don't have all that many, but the few we've had do that without any extra burden.

CHAIR REMPE: Well, we went through this a little bit yesterday, the fact that we explored with the staff, have you identified some things in addition you would need? Yes, they have the right to say, "I'd like you to go further." I mean, they can say that. But, on the other hand, the staff can say, if you want that, I'd like to see X, Y, and Z and documentation; I've got some RAIs to think about.

MEMBER BLEY: That has nothing to do with the question I just asked.

CHAIR REMPE: Yes, it's a --

MEMBER BLEY: No.

CHAIR REMPE: -- burden that they will have to assume, if they want to do that, is what I would contend.

MEMBER BLEY: No. These guys will do it or won't do it, however it turns out. But when a COL

comes along, it's going to be less burden on the breadth of all the COLs, assuming they sell lots of these things, if it's already resolved. If it's not, they would have to do the same thing. If it is resolved and they want to do something different, they're back kind of where they started from. You've really got to explain why that's extra burden on them.

MR. NOLAN: I think we understand the comment. Our position at this point is we have the information we need to make a regulatory finding that's needed --

MEMBER BLEY: I get that, but --

MR. NOLAN: -- and to go beyond that --

MEMBER BLEY: But I don't understand this point that you've raised, and it's been raised elsewhere, that it's additional burden on the COL applicant.

MEMBER CORRADINI: Keep on going. I think Dennis has expressed himself.

MR. NOLAN: I understand.

So, the last bullet here is on the instrumentation piece. In that, NuScale's position is that instrumentation is not needed to support the strategies. And so, they do not rely on the instrumentation, and to verify certain parameters is

not a direct requirement in the rule, and that it's only there as like an added benefit or additional assurance.

MEMBER CORRADINI: Let me ask you the question, let me ask you a different question. Is 14 days indefinite?

MR. NOLAN: So, it's an interesting question.

MEMBER CORRADINI: Well, if I understand the crux of NuScale's argument, it is they want to go there for that reason.

MR. NOLAN: Yes.

MEMBER CORRADINI: And so, my question is that versus all this other folderol.

MR. NOLAN: I think if we were to go down that road, we would need additional engagement with NuScale to better understand their position. Because when you look at the rule --

MEMBER CORRADINI: I'm sorry to pick on you, but you're the one up there. If that's the case, they're willing to go through the extra engagement. What I'm sensing from staff is you don't even want to go there. I'm sorry.

MR. NOLAN: And I think that, personally, I need clarity on what the position is. Is it just to

acquire offsite resources? That's one thing. Or is it to demonstrate sufficient site functional capability? That's a very different thing. Are you assuming that you are fixing and recommissioning normal systems to bring the site function capability back? Because that, I don't think it would be reasonable to assume non-seismic systems could be repaired in that short amount of time.

MEMBER CORRADINI: Okay, but what I guess I'm hearing from you is there are things that your guys are going to have to think about and ask and get clarification, either by analysis or commitment, if it's beyond the 72 hours. And at this point in time -- I'm picking my words carefully -- you don't want to claim that anything is indefinite until you check into the questions you would ask and the answers you would get?

MR. NOLAN: Yes. I don't believe that 14 days would be indefinite. Now can 14 days be used to justify ad hoc use of offsite resources, that's another question.

MEMBER CORRADINI: Okay. So, it's not so much a matter of that it's more than three days? It's a matter of whatever it is -- call it "X" -- how is "X" arrived at and what happens after "X"?

MR. NOLAN: Right. I mean, if you're looking at Fukushima as the example, 14 days is essentially where they've transitioned approximately from phase 2 to phase 3. And one can argue they're essentially still in phase 3 indefinitely.

MEMBER CORRADINI: One could argue that.

MR. NOLAN: Right. They haven't --

MEMBER CORRADINI: That's fair.

MR. NOLAN: Yes.

MEMBER CORRADINI: That's a fair comment.

CHAIR REMPE: I want to go back to what Dennis said, just to make sure that I can find a way through this. If one says that, if the staff had said that it could place additional burden for a finding at the design certification stage -- for example, they may need to provide additional analysis testing or mitigating measures -- is that a problem with you, Dr. Bley? Because it could, for a design certification finding --

MEMBER BLEY: I don't have a problem. So, I'm not sure what the question means. Try me again.

CHAIR REMPE: Okay. So, if the staff would say that a finding for performance beyond 72 hours could place unnecessary additional regulatory burden at the design certification stage because they

may require additional analysis testing or mitigating measures, for, again, a finding at the DCA stage, I think that's a true statement. It does give them more burden for a finding at the DCA stage. It could if they decide they need that.

MEMBER BLEY: I don't get that. I don't. So, that doesn't convince me.

CHAIR REMPE: Why does that not make sense to you? Because, to me, it makes sense.

MEMBER BLEY: Well, it's --

CHAIR REMPE: And they say something --

MEMBER BLEY: Why does it make sense to you?

CHAIR REMPE: Because they could get a finding, if they go with 72 hours, perhaps based on the available information -- if they want to go beyond 72 hours, the staff may say, well, I've got a lot of RAIs on what you're going to do.

MEMBER BLEY: But that's at this stage. That's at the design cert stage.

CHAIR REMPE: Right.

MEMBER BLEY: Yes.

CHAIR REMPE: And so, this could be additional --

MEMBER BLEY: Oh, it could be. It's

additional burden for this applicant who said we want to take on that burden.

CHAIR REMPE: Yes.

MEMBER BLEY: Now we don't manage the staff. But if the argument is it will cost the NRC more to evaluate this 14-day issue than it would not to do it, I mean, that's true, but that's true for what we're already doing for Chapter 20 which isn't required at this stage.

CHAIR REMPE: They've never had to do anything.

MEMBER BLEY: Well, we've taken on the burden. The staff has taken on the burden of looking at Chapter 20. They've stopped it where they want to stop it. The Applicant has asked to go further. It's going to cost the Applicant more.

CHAIR REMPE: It's going to take more time.

MEMBER BLEY: It's going to take more time.

CHAIR REMPE: And they've got all this push to try to do this expeditiously.

MEMBER DIMITRIJEVIC: It's their choice. It's their choice, Joy. That's where the difference. It's Applicant's choice.

CHAIR REMPE: Yes, I understand it's their choice.

MEMBER BLEY: But what I didn't understand is why this would make the COL more difficult.

CHAIR REMPE: Oh, I didn't --

MEMBER BLEY: I think it will make the COL easier.

CHAIR REMPE: Oh, I definitely agree with you on that, but I didn't know that the staff would be saying that this is more burden for the COL applicant. I thought it was the same as --

MEMBER BLEY: Go back and read the transcript. It was said.

CHAIR REMPE: Well, what's in the Safety Evaluation and what is in the SECY that we're not supposed to know? I thought it was pretty vague. Or it has not been publicly released. It just said it could place additional regulatory burden. I don't think they said "on the COL applicant," did they?

MR. TABATABAI: That's true. Actually, the NuScale position is to resolve as much during the DCA stage with the installed equipment, rather than posing it to --

CHAIR REMPE: To reduce regulatory uncertainty for the COL applicant. That's been pretty

clear. And the staff response back saying that this is more regulatory burden, I thought was basically saying for the NuScale folks, not the COL applicant. Did you say for the COL applicant? I don't think you said that.

MR. TABATABAI: No, I think that this comment was that you were talking about DCA stage. And I think Dr. Bley was referring to the COL stage burden. I think that that was the disconnect.

CHAIR REMPE: But, okay --

MEMBER BLEY: Which is what I thought I heard Ryan say.

MEMBER DIMITRIJEVIC: Right, yes, he did.

CHAIR REMPE: Did you say that, Ryan?

MR. NOLAN: Yes. I mean, I probably should have used a different word.

CHAIR REMPE: Does the SECY say that, though we can't --

MR. NOLAN: No.

CHAIR REMPE: The SECY does not say that?

MR. NOLAN: I should have said flexibility.

MEMBER MARCH-LEUBA: I don't want to read it, but --

CHAIR REMPE: I'm right or he's right? Or

Dennis is right?

MEMBER MARCH-LEUBA: You're right.

CHAIR REMPE: That's what I thought. Thank you. Because, for other reasons, I just want to make sure I've got it clear. Thank you.

MR. NOLAN: We're at a point now in the review where we can make a finding; no additional information is needed from NuScale. It's just documenting in the Safety Evaluation. To go beyond that would require certainly more resources on our end, possibly more resources on you folks.

MEMBER CORRADINI: I know you're near done, so I don't want to hold you back. You're getting excited now.

(Laughter.)

MEMBER CORRADINI: But, on the other hand, it's a cost-benefit question. If the benefit is you would term that indefinite, they might be willing to do the cost. If you're going to term it, well, it's just four times more days, but it's not indefinite because I don't like the term, it may not be worth the cost. But you're not even willing to entertain --

MR. NOLAN: Well, yes, like I said, that term "indefinite," I think we would have to look at that closer and better understand what it actually

means. Is it, do you mean just an alternate means to remove heat, ad hoc use of offsite resources, or do you actually mean mitigation strategies ends at 14 days?

MEMBER CORRADINI: So, I know -- I think I saw the draft SECY, but I don't remember it anyway. So, it doesn't matter. It seems to me, if it doesn't address the concept of what's indefinite and what terms, what is necessary to determine indefinite --

MR. NOLAN: It doesn't, no.

MEMBER CORRADINI: -- then there's no point, because it's just a cost-benefit process question that just sits out there.

MR. NOLAN: Yes, the SECY does not address what "indefinite" means.

MEMBER BLEY: We haven't reviewed the letter that NuScale has sent asking for the 14 days.

MEMBER CORRADINI: No, but, I mean, based on the --

MEMBER BLEY: At least personally, I don't have a clue what it actually says.

MEMBER CORRADINI: Correct. No. But, based on the discussion we had yesterday, that's what we were told was the key.

MEMBER BLEY: And if they wanted to argue

that at that point they wouldn't need anything special, they could have responded through normal sources and --

CHAIR REMPE: But, Mike, I would contend that the draft SECY has a lot of information regarding that, beyond 72 hours, the staff is going to need information that is site-specific to make that determination. They have, in their own way, answered the question you're raising: is 14 days enough or not? Because they've raised -- perhaps the draft SECY would point out several examples where site-specific information is needed.

MEMBER BLEY: That doesn't seem to be in dispute.

CHAIR REMPE: Yes.

MEMBER BLEY: The staff has said it would need more information and NuScale has said, yes, we'd have to provide more information.

CHAIR REMPE: Yes, but it may require something that's site-specific, and NuScale can't do that because they don't have a firm site. And so, this thing about 14 days -- and I think that was what Ryan was trying to say back to you -- is, was 14 days enough or not? Well, it depends on the site.

MR. NOLAN: That's certainly part of it,

yes.

MEMBER DIMITRIJEVIC: It may not be the case.

MEMBER CORRADINI: He thanks you for that reasoning.

(Laughter.)

CHAIR REMPE: Well, I spent a lot of time last night thinking about it, okay?

MEMBER CORRADINI: I'm sure it's true, but --

(Laughter.)

MEMBER CORRADINI: Go ahead.

MR. NOLAN: I'm done.

MEMBER CORRADINI: Oh, you're done? Oh, okay.

MR. TABATABAI: Okay. Well, the last slide for the Chapter 20 presentation, as you're aware, we have issued a SECY paper that describes our plan for competing the review. The SECY paper will be publicly available on Friday, and NuScale plans to submit Revision to the DCA in August of this year. So, our plan is, basically, to review Revision 3 based on SECY papers, guidance, and approach, and compare the information, evaluate the information compared to the requirements of the new rule 10 CFR 51.55.

So, this actually concludes our Chapter 20 presentations. If there are any comments?

(No response.)

MR. TABATABAI: Okay. So, we are going to go to Chapter 6.

MEMBER CORRADINI: So, the Chapter 20 person escapes.

MR. TABATABAI: Okay. Thank you. We're going to start our Chapter 6 presentation. We don't have any closed session, any slides for the closed session. So, we are going to talk about Chapter 6 here.

During the June 18th Subcommittee for Chapter 6 presentation, we presented Section 621 -- I'm sorry -- 6364 and 65. And after the Subcommittee presentation, we were asked to focus our full Committee presentation on, basically, exemption requests in Chapter 6. So, this is, basically, the topic of our full Committee presentation.

Section 6.3 will be discussed as part of Chapter 15. ECCS valves will be discussed as part of Chapter 3, and other topics that the ACRS members had expressed interest in will be discussed by NuScale as part of their presentations during the closed session.

And I think this morning when you

discussed some of the issues, I think they referred to the visits at Corvallis for that additional information.

So, with that, I'm going to turn the microphone over to Clint Ashley to talk about exemptions related to 6.24.

MR. ASHLEY: Next slide, please.

I'll be presenting the staff's review of Section 6.24, exemption requests related to containment isolation requirements. There's four requirements that NuScale is seeking an exemption from. And these four are GDC-55, 56, 57, and a TMI-related requirement at the bottom.

The provisions in GDC-55 require, in part, that each line that is part of the reactor coolant pressure boundary and penetrates primary reactor containment shall be provided with isolation valves. And GDC-55 specifies that these valves will be located one inside containment and one outside containment.

The provisions in GDC-56 are nearly identical to GDC-55, except that GDC-56 applies to lines that penetrate containment and connect directly to the containment atmosphere. For both of these GDCs, redundant valve barriers required to account for a single act of failure in the isolation provisions,

and this is achieved by putting those isolation valves in series.

NuScale's GDC-55 and 56 penetration lines are designed with two automatic isolation valves, but instead of one inside and one outside containment, they're both located outside containment. And as a result, the Applicant is seeking an exemption for those GDC-55 and 56 elements. And it's just those elements, and it's just a part of the requirement.

The provisions in GDC-57 require, in part, that each line that penetrates primary containment is neither part of the reactor coolant pressure boundary or doesn't connect directly to the containment atmosphere shall have at least one isolation valve. And as specified in GDC-57, the valves shall be outside containment.

The containment isolation function for GDC-57 lines is generally accomplished by a closed system inside containment and that isolation valve outside containment. And that gives you your two barriers.

In the NuScale design, there are two independent decay heat removal systems. Each decay heat removal system has a steam line and a condensate return line, and the Applicant proposes to use that

closed-loop DHRS system outside containment as an alternative to the isolation valve requirement. And as a result, they are seeking an exemption.

And the last item on this slide, 10 CFR 5034(f)(2)(xiv)(E) -- and hopefully, I won't say that too many times; I'll just say TMI-related item, and you'll know what I'm talking about.

(Laughter.)

MR. ASHLEY: But it requires containment isolation systems that include automatic closing on a high radiation signal for all systems that provide a path to the "environs". That's the word they use in the rule. I'd prefer to say "environment". So, I might skip between "environs" and "environment".

But, in this case, NuScale design provides signals to automatically close all systems that provide a path to the environment without using a high radiation signal. And a result, they're seeking an exemption.

Next slide, please.

MEMBER RICCARDELLA: This business of the two isolation valves outside containment versus one outside and one inside, is that -- it's kind of related to the break exclusion, the significance of the break exclusion zone, isn't it?

MR. ASHLEY: That is correct.

MEMBER RICCARDELLA: Because, at Subcommittee, I asked you, had this been abandoned before, and you said, yes, it has, but it's not that common. And as I understand it -- and correct me if I'm wrong -- if you assume a break in that exclusion zone, if you have valves both inside and outside, then you can still postulate one active failure. But if they're both outside and you have a break, you can't postulate one active failure. Okay?

MR. ASHLEY: Well, that's a good question.

MEMBER RICCARDELLA: Yes.

MR. ASHLEY: If I postulate the single active failure is the one that's inside containment, and I have a break in that piping between the containment and the outside isolation, I still have that same flow path.

MEMBER RICCARDELLA: Yes. Yes.

MR. ASHLEY: So, I do think in NuScale's exemption request they do acknowledge that -- and this is also consistent with the design-specific Review Standard 624 guidance, which leverages some ANS standards as well. They talk about, if you've got severe conditions inside containment -- and the NuScale design does have severe or harsh conditions --

you can have both valves outside containment, but you need to take into account, you want some high confidence that that piping system -- I've heard some people refer to it as "super pipe" -- but you're going to apply Standard Review Plan 362 and Branch Technical Position 3-4, which gives you additional confidence that you can prevent a break from happening in that passive pipe system.

MEMBER BLEY: Can you give me a one-minute tutorial on this phrase "break exclusion zone"? Is that an ASME term that says, if you do it this way, it won't break there, or what is that?

MR. ASHLEY: Is Renee Li in the audience?

(Laughter.)

MR. ASHLEY: Maybe Renee could speak to that better than I could.

MEMBER BLEY: Thank you.

MR. ASHLEY: She's a 362 reviewer and is well-steeped in Branch Technical Position 3-4.

MEMBER BLEY: Or is it an NRC term? I'm not even sure.

MS. LI: Yes, I'm Renee Li.

You are correct, the staff position, as described in Branch Technical Position 3-4, the break exclusion actually is defined from the inside

containment isolation valve through the containment penetration up to the outside containment penetration. And this region, portion of piping, people refer to "super pipe" because the design and the inspection requirement is upgraded to preclude a pipe break in this portion of piping.

So, for the NuScale design, as it expands -- they have different configuration. There is no inside isolation valve, but, rather, the two isolation valves is built in one single valve body. And that portion of the valve actually is welded to -- I think the configuration is you have the containment vessel, and then, the nozzle, and then, the safe-end, and then, the valve.

So, the break exclusion as defined for the NuScale configuration is different from what the staff guideline in BTP 3-4. So, with that, we have asked a very detailed question. It is that, for the NuScale configuration, with the way that they define the break exclusion, how you can demonstrate with the NuScale design the probability of piping failure within the break exclusion is still extremely low, such that it satisfies the requirement. It says you don't have to evaluate many effects if one can demonstrate the failure is extremely low.

So, in that, NuScale has demonstrated their piping stress design, stress as well as fatigue meet the intent of the BTP 3-4, such that the stress is about 80 percent -- and I'm talking about the stress -- 80 percent of the SME's event stress limit, and the fatigue is .1, which is well below the SME1. But most important is that, for all the welds that they talk about within the break exclusion, they are all subject to augmented ISI requirement such that their volumetric examination is for every 10 years.

And there are other design upgrades, but those are the two main points. So, with that, the staff thought NuScale has demonstrated appropriately that the failure would be extremely low.

MEMBER RICCARDELLA: So, it's super pipe?

MS. LI: Yes. Yes, it's different, but --

MEMBER RICCARDELLA: It's different?

MS. LI: I mean --

MEMBER RICCARDELLA: Extra-super pipe?

(Laughter.)

MS. LI: No, the configuration is different.

MEMBER RICCARDELLA: Yes, I understand.

MS. LI: But the essential point of low stress, low fatigue, augmented inspection, yes, it's

the same.

MEMBER RICCARDELLA: Okay.

MS. LI: Okay?

MEMBER RICCARDELLA: Okay. Thank you.

MS. LI: You are welcome.

MR. ASHLEY: Thank you, Renee.

So, for GDC-55, 56, and 57, on this summary slide, we found that the NuScale design accomplishes the containment safety function by providing a containment isolation capability comparable to that required by the GDCs, and therefore, the underlying purpose of the GDCs is met. So, we found that in this case the underlying purpose of the rule was being satisfied, which supported an exemption request. And the staff recommends that the Commission approve the exemption.

Next slide, please.

MEMBER BLEY: Let me ask you one question. This should have been directed at NuScale, and I think maybe we did during the Subcommittee, but I don't remember for sure.

What is the main reason they don't want to have any valves inside on these lines?

MR. ASHLEY: I'm sorry, were you directing it to NuScale?

MEMBER BLEY: No, I was directly it to you.

MR. ASHLEY: Okay.

MEMBER BLEY: I already missed NuScale. So, I thought maybe you could help me.

(Laughter.)

MEMBER RICCARDELLA: Well, we could ask if anybody --

MEMBER BLEY: We could, but did they justify it to you?

MR. ASHLEY: Well, I think the exemption request outlined a scenario that made a lot of sense, especially given the harsh environment, a thousand pounds of pressure, really high temperatures, and you've got a design of a single valve body, two isolation valves. You've got hydraulic controls. You've got a non-1E power system. So, all these things, if you've looked at other ways to achieve isolation, I think that harsh environment inside containment poses a significant challenge.

MEMBER BLEY: I guess I'm not sure. When is the environment that extreme inside containment?

MR. RAD: This Zac Rad, NuScale.

Harsh environment was certainly one of the considerations, but if you think about how you would

have to operate those containment valves, in order to protect containment we have to put additional penetrations in containment with actuation controls and things like that. So, there's also --

MEMBER BLEY: Oh, no, I'm sorry. Yes, okay. So, it's not only that, it's where do you put them, too?

MR. RAD: That's correct.

MEMBER BLEY: Yes, yes.

MR. RAD: That's correct, yes. And there's also --

MEMBER BLEY: I was thinking up in the top you have some room, but maybe not.

MR. RAD: Right. Space considerations are also a factor. So, there's quite a few considerations that weren't part of the initial requirement considerations.

MEMBER BLEY: Yes. Thanks, Zac.

MR. ASHLEY: So, this particular slide summarizes the staff's findings for the Three-Mile-Island-related requirement. And I'll speak to this a little bit more.

The NuScale Design Certification Application requests an exemption from this TMI-related requirement as applied to the containment

evacuation system, or the CES. That's the system that ties directly to the environment during normal service.

In the DCA, the Applicant describes that the design meets the underlying purpose of the rule by isolating the containment evacuation system using two automatic containment isolation signals. And these two signals are high pressure in containment and low pressurizer level.

The Applicant explains that the NuScale design differs from traditional large lightwater reactor designs because the reactor core uncovery and resulting core damage cannot occur without reaching a low pressurizer level containment isolation setpoint. And therefore, an event similar to Three Mile Island Unit 2 is precluded from the NuScale plant design.

The Applicant also states that the pressurizer is located well above the level of the reactor core and not connected to the reactor vessel by piping. Any decrease in reactor vessel inventory to the level of the core would result in complete emptying of the pressurizer and operation of the level containment isolation signal.

So, as such, they believe that they meet the underlying purpose of the rule with these

alternate containment isolation signal provisions, and the staff found that acceptable. And we also recommend that the Commission approve the exemption request.

That concludes my discussion.

MR. TABATABAI: Any questions for Clint?

(No response.)

MR. TABATABAI: No? So, the next presentation is by Anne-Marie Grady, and she will be talking about the exemption requests in 6.25.

Anne-Marie?

MS. GRADY: For combustible gas control, there is one exemption request. And the exemption request is to 10 CFR 50.44(c)(2), which requires that a plant accommodate hydrogen generation up to 100 percent fuel clad-coolant reaction while limiting containment hydrogen to less than 10 percent and maintain containment structural integrity and other accident-mitigating features. That's the requirement.

NuScale has requested an exemption from that to not provide a system that would control the hydrogen concentration to less than 10 percent. Pursuant to 10 CFR 50.12, the Commission is authorized to grant exemptions, and the exemptions that NuScale has requested, as long as there are special

circumstances. And there are six different circumstances, any one of which is sufficient.

NuScale has requested the exemption based on two of the special circumstances. One, the application of the regulation 10 CFR 50.44(c)(2) is not necessary to achieve the underlying purpose of the rule. And secondly, that the compliance would result in undue hardship or other costs that are significantly in excess of those incurred by others similarly situated.

The staff review on 10 CFR 50.44(c), which applies to advanced lightwater reactors, focused on the hydrogen conditions in the containment during postulated accident scenarios, demonstration of adequate mixing in containment post-accident, equipment survivability following postulated hydrogen combustion. NuScale's analyses of these scenarios demonstrated that the containment would survive a bounding combustion event inside the containment at 72 hours.

Staff did confirmatory calculations and confirmed that the calculation for limiting the pressure pulse inside the containment resulting from the hydrogen combustion -- actually, detonation -- at 72 hours was such a very short duration, the

containment integrity would be maintained. And the containment integrity is the requirement, is what we're trying to protect in 10 CFR 50.44(c).

MEMBER BLEY: Anne-Marie, in looking at these things, where does one assume the oxygen comes from to it?

MS. GRADY: Well, as you know, there is some oxygen to begin with, but very little --

MEMBER BLEY: Very little, yes.

MS. GRADY: -- because, of course, it's being operated at a vacuum.

MEMBER BLEY: Yes.

MS. GRADY: If there's a break in the containment, ECCS line break, for example, the hydrogen would be released into the containment, but, also, radiolysis would begin. And radiolysis generates not only hydrogen, but also oxygen. And NuScale did an analysis and showed how much hydrogen will be generated, and they got up to, I think, about 49 percent of the hydrogen that would be generated from the failed fuel.

MEMBER BLEY: Okay.

MS. GRADY: And coupled that with the oxygen we produced from the radiolysis, it would just barely make 5 percent oxygen in containment, which is

just enough to support combustion. And then, they postulate that that combustion could lead to flame acceleration, and the flame acceleration could be reflected, and the reflection could, therefore, lead to DDT, which means deflagration -- which is burning -- to denotation transition. So, they took a truly conservative approach and --

MEMBER BLEY: With that little bit of oxygen? I'm surprised.

MS. GRADY: -- assumed that they could survive all of those things and come up with a very high-pressure pulse in containment. Now the high-pressure pulse in containment lasts for a very short duration, and other branches in the agency have looked at the pressure pulse and the duration and have determined that that pressure pulse would not exceed the strain on the containment vessel.

MEMBER BLEY: How short is that time period?

MS. GRADY: Pardon?

MEMBER BLEY: How short, roughly, is that?

MS. GRADY: .445 milliseconds.

MEMBER BLEY: Milliseconds? Yes, that's fine. Okay.

MEMBER RICCARDELLA: Half a millisecond.

MEMBER CORRADINI: Doesn't even know it's there.

MS. GRADY: I beg your pardon?

MEMBER CORRADINI: It wouldn't even know it's there.

MS. GRADY: I don't know. But, based on that, based on the fact that there is, to begin with, there's the containment vessel that is oxygen-starved, and even though there could be, if you wanted to postulate a combustion event actually occurring at 72 hours, with all of the conservatism that would lead that to detonation, the containment integrity is maintained. And on that basis, we believe that NuScale's analysis and position satisfies the underlying principle of the rule, which is to maintain containment integrity during a combustion event in containment.

Okay. Do you have any questions on that?

(No response.)

MS. GRADY: All right. Then, there's containment leak rate testing. And containment leak rate testing also has an exemption request, and actually has a request to not meet 10 CFR 50, Appendix A, GDC-52, which is to provide in their design the capability for containment integrated leak rate

testing, which is often referred to as Type A testing, at the containment design pressure.

They've also asked for an exemption for 10 CFR 50, Appendix J, which is the requirement to perform the Type A testing. Now, typically, a COL applicant would be asking for that exemption because they're the ones who have to ask for that. But, since NuScale is asking not to provide the means in their design to perform the testing, they're also asking if we would consider that they don't have to do the testing. Since they're logically connected, we certainly accepted the premise of their argument that they could ask for them together.

NuScale makes their argument that it's not necessary to meet the intention of the rule by performing this Type A test. And they base it on many factors. One of them, because the NuScale containment design is an ASME Section 3, Class 1, pressure vessel; because they have done an analysis for leak tightness of the containment vessel, which we have reviewed; that they have, unlike the operating fleet, 100 percent inspectability of the containment vessel, both inside and outside the containment. And, of course, they're required to inspect it for the ASME Code.

And they have also, in addition to that,

in addition to doing their analysis to show that in their belief the containment vessel at accident pressure will not leak, the vessel itself at the mechanical joints, at the bolted flanges, they have proposed to provide a preservice design pressure test, which is in addition to the ASME Code hydro test that's performed on every ASME vessel.

And they have done that for a couple of reasons. One, the ASME Code hydrostatic test is done really to see if the vessel is strong and to see if there are any flaws in the material in the vessel. However, the mechanical joints, if they leak, the vessel still passes the Code hydro. So, that test wouldn't demonstrate an essentially leak-tight containment vessel. So, they propose this other preservice design pressure test, and the pressure test that they have proposed would be in addition to the Code hydro. It would be a hydrostatic test, not pneumatic. It would be pressurized up to the design pressure. It would be held for 30 minutes. And any leakage in the containment vessel would be a failed test, and therefore, the vessel would not go into operation until that situation had been remedied.

The preserve design pressure test is going to be ITAACed. And that, of course, means every

containment vessel will have this one-time preservice design pressure test.

MEMBER RICCARDELLA: But the bolted joint is taken apart --

MS. GRADY: I'm sorry?

MEMBER RICCARDELLA: The bolted joint is taken apart and put back together at every refueling, right?

MS. GRADY: I couldn't hear you. I'm sorry.

MEMBER RICCARDELLA: The bolted joint --

MS. GRADY: Yes.

MEMBER RICCARDELLA: -- in the containment, isn't that taken apart and, then, put back together at every refueling?

MS. GRADY: Absolutely. There are 21 bolted joints in the containment actually. You're thinking of the main flange? Yes, it's opened and reclosed after every refueling, but there are 20 others, and about half of them are also opened and reclosed.

MEMBER BLEY: But you don't have to do another hydro? Just one in the beginning?

MS. GRADY: Pardon?

MEMBER BLEY: I thought you said you only

have to do the hydro test once on a containment when it's brought in.

MS. GRADY: The ASME Code hydro test, once.

MEMBER RICCARDELLA: Yes. No, but this leak test --

MS. GRADY: The preservice design pressure test, once, that's correct.

MEMBER BLEY: But they do a leak test of some sort every time?

MS. GRADY: But I haven't gotten to the other tests that are part of the Appendix J requirements.

MEMBER RICCARDELLA: Okay.

MS. GRADY: On the mechanical joints, NuScale is going to produce, provide the results of the Type B tests, which are on the mechanical joints, as they are required by Appendix J. They're going to do that at every refueling.

MEMBER RICCARDELLA: Okay.

MS. GRADY: When they start to refuel, they've got a Type B, which is a pneumatic test of the design pressure. They're going to be do a Type B test, an as-found test, which means they're going to test the vessel before they take it apart. Then,

they're going to take it apart, do their refueling, put it back together again. And for any joints that have been opened, mechanical joints, they will do as-left testing as well. So, they will confirm that the mechanical joints are tight.

They will do, as expected, Type C testing, which is the leakage test on the piping penetrations, which has the containment isolation valves in it. They will do that, per the ASME -- per the Appendix J requirements and at the periodicity that's required by Appendix J.

So, no change in those local pressure tests, those local pneumatic pressure tests at all. They're just asking not to test the overall vessel --

MEMBER RICCARDELLA: Got it.

MS. GRADY: -- pneumatically.

MEMBER RICCARDELLA: But, they mean parts of Appendix J, but not all of it?

MS. GRADY: That's correct. That's correct.

Okay. So, as I said before, it's an ASME Section 3, Class 1, pressure vessel. They performed a leakage analysis. They have design specifications. They have capability for 100 percent vessel inspection and examination and testing. And all of these things

provide assurance that the leakage integrity of the containment is maintained.

NuScale analyzed the flange bolts to the flange openings using ANSYS. And based on the seal design and specification, they calculated flange contact pressures at every one of these and corresponding flange gaps, and based on the internal accident pressure and temperatures.

Staff reviewed all of these calculations during an audit. Based on the fact that the design and the analysis and the Types B and C testing would confirm the leak tightness of the containment vessel, staff recommends that the exemption request did not require Type A test, nor to require the design capability for the Type A testing.

MEMBER KIRCHNER: Anne-Marie, you know, once they are ready to go, whether it's a new module or refueled module, then they're going to pull a vacuum on the containment. What happens next if they have water ingress?

MS. GRADY: I'm not exactly sure what your question is, and I think maybe somebody --

MEMBER KIRCHNER: Well, it's an inverse leak test. It's not necessarily a design loading internally, but by pulling a vacuum on the containment

when they're ready to operate, if they had a seal problem, that would show it.

MS. GRADY: They do the as-found -- the as-left testing before they would pull the vacuum.

MEMBER KIRCHNER: Yes, I understand that. But when they pull a vacuum, if, for whatever reason, some seal somewhere flexed or something, then they're going to suck in water from the heat sink.

MEMBER RICCARDELLA: But that's only 15 psi.

MEMBER KIRCHNER: I know. I know that. But it's --

MR. ASHLEY: So, the containment evacuation system would be able to pick up that inventory.

MEMBER RICCARDELLA: They should pick that up.

MR. ASHLEY: And you don't necessary know where that inventory came from. So, you assign to the reactor coolant system, and you would have to meet those leakage criteria, which are very tight. So, you would enter into a tech-spec condition very quickly, I would imagine.

MEMBER KIRCHNER: Right. Okay.

MEMBER RICCARDELLA: But that's at much

lower pressure.

MEMBER KIRCHNER: No, I know that.

MEMBER RICCARDELLA: That's an Appendix J test, right?

MEMBER KIRCHNER: I know that.

MR. ASHLEY: Well, certainly the vacuum from a differential is -- I think their vacuum was 1 psi, roughly. But if there was a leak into containment from any source, your containment evacuation system leakage detection would pick that up.

MEMBER CORRADINI: Keep on going.

MR. ASHLEY: Well, that actually concludes our presentation. Are there questions?

MEMBER BLEY: I wanted to go back to hydrogen.

MS. GRADY: Yes.

MEMBER BLEY: I think you told me that one of the potential sources of oxygen is if you have a line break.

MS. GRADY: No, I didn't.

MEMBER BLEY: I thought you said --

MS. GRADY: I said from radiolysis.

MEMBER BLEY: Well, you did say that, but I thought you also said --

MS. GRADY: Oh, you said hydrogen? Absolutely.

MEMBER BLEY: No, oxygen getting in.

MS. GRADY: No, radiolysis is the source of the --

MEMBER BLEY: Is the only source?

MS. GRADY: -- additional oxygen in the containment.

MEMBER BLEY: Hydrogen gets through --

MS. GRADY: I'm sorry, there is a small amount of hydrogen in the RCS, and if that gets out, yes, that would be accounted for, too. But that's not a significant amount --

MEMBER BLEY: No, where I was going, because I thought you said you had to also think about pipe breaks, it's a very rich mixture. I'd be surprised if it burns in containment. But if it could somehow leak out into the reactor building, that's a different issue.

MS. GRADY: Out into the reactor?

MEMBER BLEY: Yes. Reactor building.

MS. GRADY: Oh --

MEMBER BLEY: But outside of containment. If it gets outside of containment outside of the building, I don't care, but if it gets out inside

of --

MS. GRADY: While it's burning, are you talking about or what?

MEMBER BLEY: No. Before it burns.

MS. GRADY: Well, it --

MEMBER BLEY: I don't think it's going to burn. With 95 percent hydrogen, it would be pretty lucky to get anything burning. But if it could somehow get out -- we don't have any way it can get out?

MS. GRADY: Are you thinking of the scenario where NuScale evaluated the containment bypass accident where there was, in fact, a CVCS line break outside containment?

MEMBER BLEY: I didn't know they did that. I didn't remember that.

So, did they assume all of the hydrogen goes out into the reactor building?

MS. GRADY: They did a containment bypass scenario. CVCS has line break outside containment.

MEMBER BLEY: Okay.

MS. GRADY: They did a MELCOR analysis, and they did determine that, yes, hydrogen would leave outside containment and would be under the bioshield.

MEMBER BLEY: Ah, and this was the thing

that could disrupt the bioshield? All right.

MS. GRADY: Yes, that's correct.

MEMBER RICCARDELLA: So, they redesigned the bioshield to address that. Remember?

MEMBER BLEY: I do remember that, but you can't get enough of a burn out there to do anything else.

MS. GRADY: Their analysis said that there would be sufficient hydrogen and oxygen to support combustion, but that --

MEMBER BLEY: Out there, I'm sure there would be, yes.

MS. GRADY: But they didn't go beyond that to characterize what that really meant. We looked at that further and did determine that there would be enough analysis to support combustion.

MEMBER BLEY: Okay.

MS. GRADY: But now there's a new design for the bioshield, and we're addressing how that would contain the combustion event and prevent it from going into the greater reactor building. However, the new design for the bioshield means it's not air-tight anymore. There are fixed openings in the bioshield.

MEMBER BLEY: Okay. So, it could leak hydrogen?

MS. GRADY: Yes. Yes, it could. But it's a big building and it's a small amount of hydrogen. So, the concentration, I have not seen a number in the reactor building, but I would suspect it's really low.

MEMBER BLEY: If it thoroughly mixed. If it didn't --

MS. GRADY: Yes. Yes.

MEMBER BLEY: -- it would burn where it comes out. Okay. Thank you.

MS. GRADY: You're welcome.

Any questions?

(No response.)

MR. TABATABAI: Okay. This concludes our presentation for Chapters 20 and 6.

MEMBER CORRADINI: Now you're moving on to Chapter 3?

MR. TABATABAI: Yes.

MEMBER CORRADINI: Are you ready?

MS. VERA: We're ready.

MEMBER CORRADINI: All right. Proceed.

MS. VERA: Good afternoon, everyone.

My name is Marieliz Vera. I'm the Project Manager for Chapter 3 of the NuScale DC Application review. Today, we're going to present Chapter 3, design of structure, components, equipment, and

systems.

And the first couple of slides is the staff review team. And we have the list of the sections that have no open items. And this slide here, we have the summary of the open items in Chapter 3 without 3.92 that was already presented and 3.11 that has 18 open items. Nine of those open items have been resolved up to date. And then, you can see the summary of the open items that are still open.

MEMBER BLEY: The turbine missile barrier analysis, you haven't reviewed that as yet, right?

MS. VERA: So, yes, we're going to present a little bit of it.

MEMBER BLEY: So, you have reviewed it?

MS. VERA: We're still on the review.

MEMBER BLEY: Started to review it? Okay. You haven't finished your review?

MS. VERA: Exactly.

MEMBER BLEY: Okay.

MS. VERA: So, we're going to focus our presentation today in the turbine missile, turbine missile barriers, TF3 testing, steam generator tube density, wave isolation, thermal fatigue, and ECCS valve design.

For the turbine missile, we're still

reviewing the information NuScale submitted and we're going to discuss with NuScale the information we presented and the questions in a public meeting that is scheduled for next week, July 16th.

MEMBER BLEY: Is that going to be webcast?

(Laughter.)

MS. VERA: No. You're welcome to call in. It's a public meeting.

MEMBER BLEY: Right.

MS. VERA: So, we're going to start with, now I'm going to go to John Honcharik.

MR. HONCHARIK: Yes. I was a reviewer for turbine missiles and I'll just go over it real quick. The regulatory basis, GDC-4 requires that central SSCs be protected from missiles.

And right now, the safety-related and risk-significant SSCs in the NuScale design are located in the reactor building and the control building. The turbine generator rotor shafts are in unfavorable orientation with respect to the reactor building and control building.

So, they are within the low-trajectory hazard zone for turbine missiles. So, to meet the requirements of GDC-4, NuScale proposed to use barriers, basically, the walls of the existing

structures. Next slide.

So, I'll be talking about the turbine missile spectrum, the spectrum of missiles. And Sujit here will talk about the actual missile barrier analysis.

With that, we had one open item. And basically, requested that the staff determine that the NuScale had not used a full spectrum of the turbine missiles, basically, the size, weight, and speed, which included up that half-stage of the last rotor. So, NuScale provided some additional information, just before the Subcommittee meeting. At that time, we had not reviewed it. So, currently -- next slide.

As Marieliz stated, we'll be discussing some of these issues --

MEMBER SUNSERI: Hey, John. John, could you hold on a second, we have three meetings going on and I want -- I think we deserve to pay you respect here. So, are we ready? Okay. Please continue.

MR. HONCHARIK: All right, thank you. So, as Marieliz stated, we'll discuss some of these issues with NuScale in a meeting next week. So, I just wanted to give you some highlights of some of the things that we wanted to discuss with them.

First item would be the destructive

overspeed of 160 percent. Basically, what was your basis for that, when other things, especially the EPRI report, has different destructive overspeed? And also, that destructive overspeed should be the bounding speed that you use for all your analysis for all the different types of missiles that you have.

MEMBER BLEY: Yes, the one thing I worry about is, if it's better than assumed, so it goes to even higher overspeeds before it comes apart --

MR. HONCHARIK: Right.

MEMBER BLEY: -- you get bigger --

MR. HONCHARIK: Right.

MEMBER BLEY: -- not bigger, but you get faster missiles.

MR. HONCHARIK: Right. And that's why, I think, I'm asking the question about the basis for the 160, 180 is probably more realistic, from the materials standpoint and such. Okay. So, that's why I wanted to discuss a little bit more about this, because in the RAI responses, I didn't see enough information in there about that.

MEMBER BLEY: At Subcommittee, we talked about it. To date, I don't know if we've had any monoblock rotors come apart anywhere.

MR. HONCHARIK: No, well, there have been

monoblock rotors that were done previously, earlier.

MEMBER BLEY: That have --

MR. HONCHARIK: That have failed.

MEMBER BLEY: -- come apart? Okay.

MR. HONCHARIK: Right, due to inclusions and such.

MEMBER BLEY: Okay.

MR. HONCHARIK: So, it's mainly sulfide inclusions. And part of the issue here is that, NuScale is using an off-the-shelf turbine, okay? Therefore, there's no testing done and everything else, okay?

I mean, yes, they're not going to buy something that's -- but part of the, if we do the turbine missile probably analysis, part of that is also turbine rotor integrity, which includes material chemistry, what you do with tramp elements, how you process it, fracture toughness, and such. If you don't have that, fracture toughness in particular, you could fracture even a monoblock rotor.

MEMBER BLEY: I see.

MR. HONCHARIK: And the next issue that we'd like to --

MEMBER BLEY: I guess, I wasn't necessarily arguing you couldn't --

MR. HONCHARIK: Right.

MEMBER BLEY: -- I was arguing, it might be going a whole lot faster than we expect when it actually comes apart.

MR. HONCHARIK: Yes, right. But I think, typically, I think that probably would be a higher range. I mean, it could go higher, but I'm not sure, I mean, because, here, we're talking about 200 percent of the -- that's pretty fast. I'm not sure exactly even the design of the actual rotor either.

The next would be the need to discuss the methodology that they used to determine the speed of the half-rotor. Because, basically, we wanted to make sure it included half of the -- with the blades, because part of it would be that some of the speed would come from that outer portion of the blades, right?

So, we wanted to look at also to see what the dimensions were, just to have some confirmation. So, we want to discuss that with them a little bit more. I know they'll be presenting later, but what I'm talking about is basically what they're going to be presenting later, so, we --

MEMBER BLEY: Are you looking --

MR. HONCHARIK: -- didn't want to discuss

--

MEMBER BLEY: -- at an exchange of more RAIs or you thinking you'll go audit some of the actual calculations they've done?

MR. HONCHARIK: Well, right now, I think the plan is, next week, we're going to have --

MEMBER BLEY: This public meeting.

MR. HONCHARIK: -- a meeting with them to discuss these things.

MEMBER BLEY: Okay.

MR. HONCHARIK: From that meeting, I think we'll discuss whether or not we do another audit or ask more RAIs or such. So, I think that's more of, to get this out. So, this is the first time, I think, they are also hearing this, too. But we wanted to let you, since I think ACRS requested that we talk about responses, so this is what we found so far, up to date.

And the next item that we want to discuss with them is the -- they had a couple of reports in there, what we looked at during the audit, and I think what we're looking for is that they include some of the analysis and information that we requested from the audits and also, the RAI responses to be factored into those reports.

And then, also, have those be part of the DCD, because that's basically part of their basis for saying that the turbine missiles are prevented, based on this missile barrier.

MEMBER BLEY: That is their basis.

MR. HONCHARIK: Right. Yes. This is their basis and should be documented as such. So, so far, that is what we have for the turbine missile spectrum. If there's no question, I'll turn it over to Sujit, who will talk about the turbine missile barrier analysis.

MR. SAMADDAR: I'm Sujit Samaddar and I'm speaking on behalf of the staff who's actually performing the review, but feel free to ask me the questions. I'll skip the first slide, because John has already presented the regulatory requirements and the SRP section that we are using as the guidance for the barrier.

So, essentially, NuScale has characterized the turbine missiles as A, B, and C, which have the weight, the velocity, and equivalent damage and the associated overspeed. So, these are the three types they have actually looked at.

Then, the acceptance criteria for the concrete barrier itself is pretty standard. We have

looked at the back face scabbing, which is the limiting criteria for the barrier. So, there should nothing be spalling up from the back of the barrier.

And the acceptance methodology that we have allowed, or we allow, is the empirical equations based on test results, which is in our SRP. And also, the use of the finite element analysis, with validation.

MEMBER CORRADINI: Can the analysis --

MEMBER BLEY: That's what --

MEMBER CORRADINI: I'm sorry.

MEMBER BLEY: That's what they're doing, is finite element analysis?

MR. SAMADDAR: That's what they've chosen.

MEMBER BLEY: Yes.

MR. SAMADDAR: Yes.

MEMBER CORRADINI: Can the finite element analysis or empirical equations determine spalling off the backside?

MR. SAMADDAR: Yes. They use penetration as a criteria. And on that, they have additional factors that they use to quantify that.

MEMBER CORRADINI: As I remember these from long ago, they're usually penetration correlations. But the penetration correlation then has some sort of

empirical connection to back spalling?

MR. SAMADDAR: Correct.

MEMBER CORRADINI: Okay.

MR. SAMADDAR: They do it in two steps. One is for perforation, can be checked --

MEMBER CORRADINI: Sure.

MR. SAMADDAR: -- based on penetration, and so is the scabbing part.

MEMBER CORRADINI: Okay.

CHAIR REMPE: I hate to interrupt you, but periodically, you put your papers on the microphone and that causes the transcriber to lose his hearing. So, thank you.

MR. SAMADDAR: Oh, sorry. Sorry.

MEMBER BLEY: Mike, as I recall, is that those empirical equations, some of them came from military projectile tests.

MR. SAMADDAR: Yes. And now, there are a wide range of these types of empirical relationships.

MEMBER BLEY: Yes, they're part --

MR. SAMADDAR: It's not just the NDRC, but there are plenty others.

MEMBER BLEY: Yes, okay.

MR. SAMADDAR: So, the NuScale has used NDRC to start their evaluation and they have used, for

limited cases, the Mann and Whitney relationship. And both these empirical relationships have a limitation based on the test data. And both assumes the missile and the target are both hard, or rigid, that's the term really to be used, not hard, but rigid.

The finite element method which NuScale has chosen allows for a variation of the stiffness of both the missile, the hardness of the missile, as well as the target. Okay. But the criteria for acceptance is still continuing to be the same, which is based on the scabbing. Next slide, please.

MEMBER CORRADINI: Does the FEM analysis actually predict the scabbing or is it a correlation given the penetration?

MR. SAMADDAR: It's a correlation given the penetration.

MEMBER CORRADINI: So, in other words, they do a more mechanistic penetration, but then use a correlation to determine if I have scabbing?

MR. SAMADDAR: They do a more realistic, I would say, they use the finite element to find the penetration and then, they use the penetration and proceed to the scabbing.

MEMBER CORRADINI: Okay, thank you.

MR. SAMADDAR: So, the FEM procedure was

not previously reviewed by the staff for a high-speed turbine missile impact. So, this is the first time we are trying to do that, so there are limitations associated with that.

The staff was looking for benchmarking of the computer results. So, they want a low end for benchmarking, because we don't have the high end test data. So, we want to see that the FEM produces similar results as would have been from one of these empirical results.

We also expect that the deformable missile will give lesser penetration depth than hard missiles. And larger the kinetic energy of the impact, the greater will be the penetration depth. So, these are the types of things we want to see, once they do the validation between the FEM and the empirical methods.

MEMBER SKILLMAN: Sujit, how is the missile orientation addressed? For instance, a portion of the low pressure, last stage low pressure blading plus a portion of that rotor entering perpendicularly, with the blades going in first, will probably result in a different outcome than if the rotor portion enters first versus the blade. So, how is the geometry of the missile addressed?

MR. SAMADDAR: Actually, in the empirical

relationships, there is a factor put in if the missile is very pointed to very blunt. So, we use a coefficient to account for what you just said.

MEMBER SKILLMAN: Okay, thank you.

MR. SAMADDAR: Okay. So, this -- we are still continuing with the review. And so, the next step from the local part of the missile impact, we look at the global assessment, of what it does globally when a larger body or mass hits the target. So, that is all captured in the global analysis. Okay.

And the global assessment, the staff review is inconclusive. So, right now, the analogy is that since the barrier has been designed for automobile impact, from tornado-born --

MEMBER SKILLMAN: Yes, this is a tornado analysis.

MR. SAMADDAR: -- that logic is not quite sitting right with us, because the mass that the assumptions for the missile, currently for the turbine blade, is about 3,000 pounds. And a normal automobile engine should not weigh more than a few hundred pounds. It's a big mismatch there.

So, that, we have to look at the analysis really what they performed and we are looking into

what they have given us to make sure that we can convince ourselves that we are looking at equivalent things.

MEMBER BLEY: I wanted to follow up on Mr. Skillman's question. I don't know if you've been able to look at their actual finite element analysis, but do they just assume a head-on impact or do they consider it could come from many different angles and be of different sizes and sort of do a probabilistic look at those different alternatives?

MR. SAMADDAR: No, to answer your question, I have not looked at their --

MEMBER BLEY: Okay, you

MR. SAMADDAR: -- finite element analysis.

MEMBER BLEY: -- haven't seen that?

MR. SAMADDAR: But for them to compare apples-to-apples, they will have to look at perpendicular strike. Because that's how --

MEMBER BLEY: Say that again, I'm sorry?

MR. SAMADDAR: Perpendicular strike.

MEMBER BLEY: Okay, yes.

MR. SAMADDAR: Because that's how all the test missiles have happened.

MEMBER BLEY: Okay.

MR. SAMADDAR: So, to compare with values

from the testing, they will have to do a strike.

MEMBER BLEY: Okay, thank you.

MEMBER RICCARDELLA: Wasn't there an RAI response that had a bunch of analytical curves compared to data? I thought I saw something like that.

MR. SAMADDAR: There were plots, there are plots, I mean, that has been presented, but we have to analyze it and we are still in the process --

MEMBER RICCARDELLA: Okay.

MR. SAMADDAR: -- it is very difficult for me to openly --

MEMBER RICCARDELLA: Okay, sure.

MR. SAMADDAR: -- talk about it. The other point to note is, for the global damage or for the large piece, that the Item Number C Missile, okay, the velocity is around, what?, 350 miles per hour, of the missile impact, while tornado and hurricanes are about 90 to 200 miles an hour.

MEMBER BLEY: Yes, 200 is completely inappropriate.

MR. SAMADDAR: So, these are the two major differences in the global analysis part that the staff has right now seen.

MEMBER BLEY: Okay. Will you just be

reviewing their analysis or will you be doing some of your own?

MR. SAMADDAR: We will, for reasonable assurance, we will do something on our own, since we will be looking at rational approaches, we will be looking at things as engineers. If it does not hurt our senses, we will most probably accept it, but we will not look into numerical details of everything.

MEMBER KIRCHNER: It seems to me that there's a lot of experience in doing this kind of analysis and you could just go for, bound their analysis by just looking at past work done on this with appropriate -- an automobile is not an appropriate surrogate. But there -- you see where I'm going with this? I mean, there's been a lot of analysis on turbine missiles.

MEMBER BLEY: Actually, from what I've seen, it's been pretty simplistic analysis.

MEMBER KIRCHNER: Oh, simplistic.

MEMBER BLEY: There was -- I only know of one analysis like this was ever done and it was a long time ago.

MEMBER KIRCHNER: Okay.

MR. SAMADDAR: I think that was not a question, but --

MEMBER KIRCHNER: I was just --

MR. SAMADDAR: -- we agree with that statement, there is quite a number of, amount of literature available. And we are already looking into those to bound our conclusions.

MEMBER KIRCHNER: There have been a lot of things thrown at concrete walls, of all different shapes and sizes, including armor.

MEMBER BLEY: That's true.

MR. SAMADDAR: Yes. Some of that data is kind of restricted. Okay. So, that's a problem, trying to bring it out into the open, some of them are quite restricted. Okay. Carrying on.

So, in conclusion, the staff is continuing with this review and we have a meeting with NuScale, it's a public meeting, on July 16, where we convey our findings that we have so far to them and hear their responses on it. Okay. And we'll take it from that point on.

So, that concludes my presentation. If there are no further questions, I'll turn it back to --

MR. WONG: My name is Yuken Wong.

MR. SAMADDAR: That was me again.

MR. WONG: Dr. Steve Hambric and I

previously had presented to the full Committee Section 3.9.2, including the comprehensive vibration assessment program, and we were request to report feedback on the TF-3 demonstration modal testing, as well as the steam generator tube regarding density wave oscillation and thermal fatigue. Next slide.

The purpose of TF-3 is to obtain fully induced vibration data for prototypical helical coil steam generator for validation of fluid elastic instability, vortex shedding, and turbulence buffeting design analysis. And NuScale had previously discussed about the objectives, in the earlier session, so I'm not going to go over it again. Next slide, please.

There are some figure here, show the configuration of the helical coils and the support SOLs, those tabs. And for the TF-3 demonstration modal testing, there are five -- two columns will be installed for TF-3. The two outer columns were installed. And performing the modal testing on Column 12.

And from inspecting the tubes, we find not all spans of the support configuration are equal. That's because the supports are not spaced equally, they are eight supports, they're not on the

circumferential direction, they are not spaced equally.

So, there are some longer spans and there are some shorter spans. On the longer spans, generally, half of them appear loose. And what we call loose is, they're not fixed supports. And the other half, the long span, they're generally tight, they're fixed supports. So, half are loose and half are tight. For the shorter spans, they are generally tight, or what we call, have a fixed support configuration. Next slide, please.

The TF-3 was originally intended to replicate the thermal expansion of the tubes and support systems, and a stiffer boundary condition at the support. So, tighter boundary conditions will lead to higher frequencies and lead to higher fluid elastic stability also.

As we show on the figures, there are gaps between the tubes and the supports and the tabs. And the preload system cannot close those gaps between tubes and the tabs and the supports.

So, what the preloading system can do is successfully tighten the fit between the supports, the vertical support column, but not press the tubes into the supports.

So, however, as I mentioned, it does tighten the supports, so increase of tube frequencies by less than ten percent. And TF-3 will now test the looser bounding conditions, without simulating the thermal expansion of the tubes.

MEMBER MARCH-LEUBA: And do I understand correctly, the tubes will be -- there won't be any flow through the tubes and they will be air filled?

MR. WONG: For TF-3, there will be primary flow. However, there --

MEMBER MARCH-LEUBA: Secondary?

MR. WONG: -- are no secondary side flow.

MEMBER MARCH-LEUBA: And they'll be air filled, there won't be any mass in there, will that change the frequencies? I've heard air filled, water filled, what is it going to be?

MR. WONG: Yes, it's empty.

MEMBER MARCH-LEUBA: Empty?

MR. WONG: It's empty.

MEMBER MARCH-LEUBA: So, then, the mass is lower, right? Will that change the frequency?

MR. WONG: Yes, it will, I believe it will, for the analysis, they will compensate.

MEMBER MARCH-LEUBA: Oh, so this will be used to benchmark some analytical tool and then, you

extrapolate results? Okay, got you.

MR. HAMBRIC: Yes, this is Steve Hambric. So, yes, they're going to have to account for the fact that they're missing the secondary coolant, in their margins that they infer from the TF-3 test. But -- and that should be defined in their test plan when it's finalized.

MEMBER MARCH-LEUBA: Okay, thank you.

MR. WONG: Yes, so --

MEMBER RICCARDELLA: Excuse me, would you repeat what you said earlier about the looser tubes lead to lower natural frequencies than if they were in there tight? Is that what you're saying?

MR. WONG: That's correct.

MEMBER RICCARDELLA: And that's conservative?

MR. WONG: And it will be conservative. So, we're hoping, with the preloading system, it can replicate the tube condition in the normal operating condition, because the test is performed at room temperature. So, in a reactor operation, it heat up, to expand.

MEMBER RICCARDELLA: Yes.

MR. WONG: However, the preloading system will not duplicate that thermal expansion of the

tubes.

MEMBER RICCARDELLA: Yes.

MR. WONG: So, therefore, the TF-3 will not be a bounding test. It will have a looser --

MEMBER RICCARDELLA: Because we expect the excitation --

MR. WONG: -- support condition.

MEMBER RICCARDELLA: We expect the excitation frequencies to be low relative to the natural frequencies, is that what you're saying?

MR. WONG: For TF-3, it will -- the measured frequency will be lower than --

MEMBER RICCARDELLA: Yes.

MR. WONG: -- what would be in --

MEMBER RICCARDELLA: Which separates it further --

MR. WONG: -- normal operation.

MEMBER RICCARDELLA: -- from the expected excitation frequency?

MR. HAMBRIC: No, no, they get closer.

MEMBER RICCARDELLA: Okay.

MR. WONG: Yes.

MR. HAMBRIC: So, it's a conservative test, they're more likely to experience FEI because of the loose supports.

MEMBER RICCARDELLA: Understood, because the excitation frequencies are even lower than expected natural frequencies?

MR. HAMBRIC: Well, that's to be decided. They're going to tell us where the excitation frequencies actually occur when they run the test.

MEMBER RICCARDELLA: Okay.

MR. HAMBRIC: So, we don't know where they are now, we have estimates based on historical literature and experiments, but this is a new --

MEMBER RICCARDELLA: Yes.

MR. HAMBRIC: -- structure that hasn't been tested before, so we don't know where the excitation frequency is.

MEMBER RICCARDELLA: But if those excitation frequencies turned out to be higher, then the looser test would be non-conservative, right?

MR. HAMBRIC: No, no, no, then they'll find fluid elastic instability, right? They'll --

MEMBER RICCARDELLA: Yes, okay.

MR. HAMBRIC: Yes, they'll experience it and they'll tell us at what speed that happens.

MEMBER RICCARDELLA: Okay. All right, thank you.

MR. WONG: Okay. Next slide, please. So,

now, NuScale will analytically demonstrate the tube and support conditions. They presented thermal expansion calculations. Thermal expansion will push the tubes downward and radially out onto the tube supports.

During the audit, the staff asked how other forces will affect the tube to support contact? Such as the primary flow and gravity will also push the tubes down. Secondary flow will push the tubes radially out, due to the Coriolis effect. NuScale will quantify these effects of gravity and flow and the staff will review the thermal expansion and gravity and flow effects. Next slide.

During the audit, we also examined the steam generator tube, due to vortex shedding. In this figure, it shows a shot of the feedwater transition tubing.

Only the lower tubes are susceptible to vortex shedding, because there's no downstream structure to disrupt the formation of the vortex. So, the feedwater transition tubing is at the bottom. And those tubing are bent in two planes. So, they are stiffer and so, the frequency is higher.

And the measured frequencies are much higher than previously calculated. And NuScale will

soon submit an updated steam generator tube vortex shedding assessment, which should show increased margin against vortex shedding.

MEMBER KIRCHNER: Yuken, are you referring now to what looks like Span E on this diagram, or are you talking about the bottom of the bundle, if I'm looking at this correctly, which is Span D?

MR. WONG: Near the top of this picture, you see where the cursor shows?

MEMBER KIRCHNER: Yes.

MR. WONG: Those are the feedwater transition --

MEMBER KIRCHNER: Yes, so those would be stiffer, yes.

MR. WONG: Yes, stiffer, because they are bent more --

MEMBER KIRCHNER: Sure.

MR. WONG: -- than the remaining of the tubing.

MEMBER KIRCHNER: And they're closer to the tube sheathe.

MR. WONG: So, the measured frequency is a lot higher than previously --

MEMBER KIRCHNER: Sure.

MR. WONG: -- estimated.

MEMBER MARCH-LEUBA: And when you say the frequencies that are measured are much higher than calculated, you're talking a factor of a thousand or a factor of ten percent? I mean, how much higher?

MR. HAMBRIC: I'd say on the order of a factor of four or five.

MEMBER MARCH-LEUBA: Okay.

MR. HAMBRIC: It's substantial.

MEMBER MARCH-LEUBA: Substantial? Okay.

MR. HAMBRIC: Yes. Yes, they were very conservative with their initial estimates.

MEMBER MARCH-LEUBA: And does it put the frequencies in resonance with something that it's not supposed to? Or is it not high enough?

MR. WONG: Previously, it was about 20 percent margin. So, right now, it should improve --

MEMBER MARCH-LEUBA: Better?

MR. WONG: -- by those factors.

MEMBER MARCH-LEUBA: Oh, so, it went from a 20 percent margin to a 400 percent margin, is that what you're saying?

MR. WONG: No, this --

MEMBER MARCH-LEUBA: I wouldn't understand the figure, let's keep going.

MR. HAMBRIC: Yes, they're going to submit

the new margins shortly and they will be much improved, that's all we can really say now.

MEMBER MARCH-LEUBA: Okay, thank you.

MEMBER CORRADINI: So, we're going on to thermal fatigue.

MEMBER RICCARDELLA: So, again, we reviewed this last meeting as part of our review of 3.9.1, I think it was, and -- whatever, 3.9.something. And one of our comments in our letter was, we recommended that -- you said the staff was considering requiring an ITAAC for this and we recommended that we do that. But then, I heard this morning that they're working on perhaps something different than an ITAAC to address this.

MR. WONG: Correct. NuScale's position is that ITAAC is required for every module. And since this TF-3 testing is only a one-time test --

MEMBER RICCARDELLA: Okay.

MR. WONG: -- maybe there's some other vehicle that's more suited to this need, such as on the statement in Tier 1 and Tier 2 of the SEA.

MEMBER RICCARDELLA: Okay.

MEMBER CORRADINI: Oh, I see. Otherwise, it would be for everybody.

MR. WONG: Correct.

MEMBER CORRADINI: Okay, I see your point.

MR. WONG: Yes.

MEMBER CORRADINI: Okay, thank you.

MEMBER RICCARDELLA: Okay, thank you.

MEMBER CORRADINI: Thank you.

MR. WONG: Okay. Next slide. I'm going to talk about the density wave oscillation. The steam generator inlet flow restricters will limit the density wave oscillation during normal operation, by limiting the flow to within plus/minus ten percent. The TF-1 and TF-2 test --

MEMBER MARCH-LEUBA: Wait, you said they were limited to plus/minus ten percent, guaranteed? Can you repeat what you said, I was sleeping?

MR. WONG: According to the DCD, Chapter 5.4, the flow restricters will limit the flow to within plus/minus ten percent.

MEMBER MARCH-LEUBA: The flow oscillation will not be more than ten percent and that's a commitment?

MR. WONG: The flow through the steam generator tubes.

MEMBER MARCH-LEUBA: But is that a commitment in the DCD? And so, somebody then will measure it and make sure that happens? Because it

didn't, I mean, when it was measured in TF-2, it was not limiting to ten percent.

MR. WONG: Well, TF-2 did not have the inlet flow restricters installed.

MEMBER CORRADINI: So, I think what he's asking is, how are you going to verify, how is the applicant or the staff with review, going to verify that they meet what they just said they're going to meet? Granted, they want it plus or minus ten percent, but how do we know that it's going to be plus or minus ten percent?

MR. WONG: For the inlet flow restricters concept testing, they selected the design that provides the adequate pressure drop to limit the density wave oscillation. And the final design is based on the selected design and there's some minor differences between the final design and the selected design. However, the differences are small and should not affect the pressure drop.

MEMBER KIRCHNER: May I ask a question? I'm trying to put two things together here. So, in the previous section, you talked about how the helical tube bundles are going to lock themselves in as you bring the unit up to power.

One of the things that's happening is

expansion, if I got this right, the thermal expansion of the tube bundle will kind of lock it in place with the tube support things. But as soon as you put a cold water slug in, it's going to be the other direction. Is it going to sit there and rattle and vibrate?

If you have density wave oscillations, what we talked about this morning is the concern about thermal mechanical fatigue, but also, if it's changing the flexure of the tubes against the support tabs, then you're going to have vibration there associated with the thermal cycling of the tube, or am I missing something?

MR. WONG: From the TF-1 and TF-2 data, because they did not install the inlet flow restricters, those tests did show density wave oscillation. However, the frequency of the density wave oscillation is so low, compared to the steam generator tube frequencies, and the tensity wave oscillation --

MEMBER KIRCHNER: No, I understand that, but when you couple the thermal effect and the fact that you're cooling the tube and heating it, so to speak, by changing the thermal conditions, then it's going to change how well it sits in the support

mechanism, in terms of the tabs.

So, if all things are going well, and it's stable at full power, and the system heats up and it's just going along, then as I understand it, the design is such that the expansion of the tubes will kind of help it lock into its support tabs.

But as soon as you start thermally changing that, you're going to induce -- you're going to loosen up its support, then you're going to have periods of cooling and heating of the tube bank and that's going to change how well it sits in the tabs.

MR. WONG: The feedwater will be preheated.

MEMBER KIRCHNER: Yes.

MR. WONG: And this test is conducted at room temperature.

MEMBER KIRCHNER: Okay, what's the --

MR. WONG: It's still bounding.

MEMBER KIRCHNER: What's the inlet feedwater temperature?

MR. WONG: I do not --

MEMBER RICCARDELLA: It's way above room temperature.

MEMBER KIRCHNER: Oh, I know that.

MEMBER RICCARDELLA: so, it's not going to loosen it that much.

MEMBER KIRCHNER: No, no, you -- it's heated up, it's running, you design it for whatever the normal inlet feedwater temperature is, that's several hundred degrees, I don't know what it is off the top of my head --

MEMBER CORRADINI: Two hundred and fifty C.

MEMBER KIRCHNER: -- and the thing locks in because the tubes expand.

MEMBER RICCARDELLA: Yes.

MEMBER KIRCHNER: But now, if you start having oscillations, the tube will cool down and the tab, the way it's locked into the tabs is going to loosen up.

MEMBER RICCARDELLA: But I assume those oscillations aren't going to take you down close to room temperature, you would have --

MEMBER KIRCHNER: I didn't say that.

MEMBER RICCARDELLA: Well, but the locking in occurs between room temperature and 400 degrees, if --

MEMBER KIRCHNER: I don't know that.

MEMBER RICCARDELLA: -- you come down to 350, you're still, I would assume, still pretty well locked in, right?

MEMBER PETTI: It's in a limited number of

tubes, the boiling --

MEMBER KIRCHNER: Yes.

MEMBER PETTI: -- right? So, it's not the entire flange, it's just --

MEMBER KIRCHNER: No.

MEMBER PETTI: -- somewhere in the middle.

MEMBER CORRADINI: Your microphone is off.

MEMBER KIRCHNER: I'm just speculating, I don't know what that differential thermal expansion and how well the tube will then sit there in the course of the density wave oscillation. Maybe it's not enough --

MEMBER CORRADINI: Depends on how big the delta T is, right?

MEMBER KIRCHNER: Yes. Yes, maybe it's not a big enough delta T to make a difference --

MEMBER CORRADINI: The design delta T --

MEMBER KIRCHNER: -- as that boiling front moves.

MEMBER CORRADINI: The design delta T from inlet feed water to saturation is 80 C.

MEMBER KIRCHNER: Yes, so 100-something degrees, 160, okay. I just --

MEMBER CORRADINI: You're sitting with an inlet temperature about 250 C. So, you're going from

250 C to saturation and then, you're coming back down. But I think what, Walt I think, is trying to get at is, he's asking a quantitative question of a qualitative thing.

Sure, you're going to expand out, but then, you're going to get to some sort of oscillatory behavior somewhere in the region that is going to be -- it's going to see an oscillation in temperature and that oscillation in temperature is going to cause an oscillation of the tube temperature, which is going to cause it to essentially start shrinking and swelling, shrinking and swelling. And the question is, can you do this in perpetuity and no problem or not?

MEMBER RICCARDELLA: Well, I mean, fatigue is a different question than whether you lose the firmness of the support or not, right?

MEMBER CORRADINI: Right.

MEMBER RICCARDELLA: I mean, right? They're two different questions.

MEMBER KIRCHNER: Two different questions, if you lose the firmness of the support and the ability for tube to rattle --

MEMBER RICCARDELLA: Yes.

MEMBER KIRCHNER: -- and vibrate and fret is going to increase.

MEMBER MARCH-LEUBA: But the solution that we check is, NuScale has committed to is, we will not have these oscillations, so we don't have to worry about it.

MEMBER CORRADINI: Within some --

MEMBER RICCARDELLA: That's what the ten percent means?

MEMBER MARCH-LEUBA: I was going to make the second comment. I'm an expert on density wave oscillations, that's how I made my career, there is no such thing as a small density wave oscillation. If it's unstable, it will grow until you hit reverse flow, and that's the first non-linearity you hit on that instability is reverse flow.

So, when you have a thermal hydraulic density wave oscillation, you always have at least 100 percent oscillation, because it will grow until it gets reverse flow. There is no such thing as a small DWO, it just doesn't exist.

So, plus, another thing, when we reviewed the stability topical, we were told that the TF-2 had very significant inlet restricters. They have modified it a little bit, but the inlet restriction in TF-2 was fairly large. I cannot say the number, but I call it infinity.

MR. WONG: Yes, they had orifices for TF-2.

MEMBER MARCH-LEUBA: Well, I'll look at the record, but they told us what the velocity losses were and they were equivalent to what they're designing now. So, this is an issue that has to be resolved.

And my concern, I'm going to just put it here out in the open, is that NuScale is going to tell you, we have a RELAP model that predicts we don't have any oscillations, get out of here. And you'll say, okay, they have a model, they have calculated it, and it's going to be okay. And it's not going to be okay.

And it goes back, Mike, to what we were talking about for the -- let me finish my comment. You need to have the right people reviewing the right things.

And this particular item is a cross-discipline item, you have to have a thermal hydraulic guy that knows density waves review what they're doing and a mechanical vibration guy that tells you what the consequences of that is. So, it's a cross-discipline, you cannot do a Chapter 3 review and a Chapter 15 review independently.

MR. MAKAR: Thank you. This is Greg Makar from Materials and Chemical Engineering Branch. I just wanted to clarify that the information in the

design certification on the limitation of the oscillation, I think it's a limitation on the amount of mass flow, on the mass flow rate, not on the oscillation itself, not the density wave oscillation, but the mass --

MEMBER RICCARDELLA: That ten percent number --

MR. MAKAR: -- mass flow rate.

MEMBER RICCARDELLA: -- is just on mass flow rate, yes.

MEMBER MARCH-LEUBA: So, you're predicting the steady state mass flow that you expect to have?

MR. MAKAR: I'm not, but that's what the --

MEMBER MARCH-LEUBA: No, that's what --

MR. MAKAR: -- I think that's --

MEMBER MARCH-LEUBA: -- plus/minus ten percent --

MR. MAKAR: -- the flow restricters design is intended to limit the mass flow rate --

MEMBER MARCH-LEUBA: No, it's not.

MR. MAKAR: -- to within a --

MEMBER MARCH-LEUBA: The flow restricters is there to prevent oscillations. I mean, otherwise, you wouldn't pay for the pressure drop. It is to prevent oscillations.

MR. MAKAR: Agreed, but that's -- the number in the DCD is not --

MEMBER MARCH-LEUBA: Okay.

MEMBER CORRADINI: But I think -- let's just clarify. I think what you're saying is the total feedwater flow, you're worried about the individual tube behavior? Am I getting it correctly? Because they have no way of knowing what the tube flow is.

MR. MAKAR: I don't have the DCD with me, but that was -- I was just trying to --

MEMBER CORRADINI: Okay.

MR. MAKAR: -- explain what I --

MEMBER MARCH-LEUBA: That would make sense, the error on my steady state flow that I predict is plus/minus ten percent.

MEMBER CORRADINI: But I think -- I want to make sure that the staff understands with Dr. March-Leuba is worried about.

He's not worried about the total mass flow, he's worried about the fact that, if I have the density wave oscillation, I'm going to, in certain channels, cause a reverse flow, and he's worried about then -- that may only happen in one channel every once in a while, and may parade around the thousand tubes, but on the other hand, you're causing some sort of

temperature swing, which would then cause a thermal fatigue swing. A thermal fatigue, which you've got to make sure you can survive. That's your point.

MEMBER MARCH-LEUBA: And if they're not sufficiently clamped --

MEMBER CORRADINI: Yes, that part I'm not --

(Simultaneous speaking.)

MEMBER CORRADINI: I guess, personally, I'm not worried about that, I'm more worried about the fact that they minimize what you're worried about.

MEMBER MARCH-LEUBA: I'm not worried about it, I like to make sure that you can look at it and say, yes, it does not detach when it cools down, which it probably won't.

MEMBER CORRADINI: Keep on going.

MR. LUPOLD: This is Tim Lupold, from the Mechanical Engineering Branch. One thing I'd like to say, you're asking questions that the staff's not going to be able to answer regarding --

MEMBER CORRADINI: We're not expecting a answer, as much as just make sure you understand our question.

MR. LUPOLD: We understand the question.

MEMBER CORRADINI: Okay.

MR. LUPOLD: I'll also say that, when you look at the steam generator, you do have to look holistically on everything. And there are requirements for inspection of the steam generators. Yuken already reviewed, a month ago, what the inspection requirements would be for the initial inspections for the steam generators.

And then, there's ultrasonic -- well, there's not. There are examinations of the tubes, which can identify if there is a wear associated with the tubes vibrating in their supports and things like that. So, also consider that those things do exist.

MEMBER CORRADINI: Keep on going.

MR. WONG: Okay. Next slide, please. Regarding steam generator tube thermal fatigue, the steam generator tubes are ASME Section III, Class 1, and fatigue analysis is required.

The staff has not reviewed a specific steam generator tube fatigue analysis. The staff is reviewing the steam generator feedwater plenum fatigue analysis. NuScale has provided an ASME fatigue screening report identifying most critical fatigue locations. Those selected locations have been reviewed, as the analysis there.

In the DCD ITAAC 2.1-4.2, we will confirm

the ASME data reports have been prepared for ASME Class 1 components listed in Table 2.1-2. This table includes RCS components, including the steam generator. And this is the end of my presentation.

MEMBER CORRADINI: Great, thank you.

MR. SCARBROUGH: Hello, this is Tom Scarbrough. I'm going to give you an update on the emergency pool cooling system valve review that we have ongoing. In the open session, of course, we have quite a bit more detail to talk about in a closed session, but I'll give you a high level discussion.

There were three reactor vent valves that are five-inch in size and two reactor recirculation valves that are two-inch in size, and they allow that natural circulation that we've been talking about.

Each RVV and RRV has a first-of-a-kind design arrangement of a main valve, the inadvertent actuation block, IAB valve, a solenoid trip valve, and a solenoid reset valve, all connected by 20 feet of tubing or so.

The NuScale is conducting an ECCS valve design demonstration testing at Target Rock right now, to satisfy 10 C.F.R. 50.43(e), which requires that for new designs, like passive reactors, that they demonstrate their safety features that's referenced in

52.47(c)(2).

We have an ongoing audit of this whole process, designed for the ECCS valves. We were onsite the week of June 3 and watched some of their testing. And we'll be going back next week to look at some of the other testing that's going to take place.

Overall, with the ECCS valves, there's several portions that should provide confidence in the valves once they're fully designed and demonstrated by 50.43(e). First, you have an ASME Standard QME-1 2007 qualification requirement. We've accepted that in Reg Guide 1.100 Rev 3.

There's ITAAC for equipment, qualification for safety-related valve, and it points to the qualification report, which is the QME-1 standard qualification report.

They're also required to meet the code of record that's specified in the ASME Code 2012 edition, that's currently incorporated by reference in 10 C.F.R. 50.55a. Of course, the COL applicant is required to use the most latest version, it's probably going to be like an 18-month time period, but they have to implement the most latest version of the OM code that's incorporated by reference in 50.55a.

The other aspect is that NuScale submitted

an alternative to 50.55a, to apply the Appendix 4 requirements for their ECCS valves. And these include, the Appendix 4 requirements include stroke time testing every quarter, but they can't do it every quarter, so they'll be doing it every outage.

But also, initial and periodic diagnostic performance assessment testing and it's up to a ten-year interval, but they have to demonstrate diagnostically that they can out to those longer intervals.

But based on discussions with NuScale, what they've planned is to bench test, like the IAB, take it off, put it on a bench, make sure that it's operating within the regime of the pressure release and engagement pressures and such as that. But then, you still need to finalize the design first, and that's still ongoing with the 50.43(e) test.

So, our next steps. We have to complete our audit, that's ongoing. We have an open item in the SER regarding ECCS valve design. And we'll be reviewing the final test report for all the testing that's taking place right now. We need to review the design specification updates.

We had an audit previously, they've notified now that those specs have been updated and

they're ready for us to review, so we'll be conducting a follow-up audit to confirm that those changes took place. And then, we'll update the SER as necessary to come up to speed with Phase 4 of the SER.

So, that's where we are right now and we can provide more detail during the closed session about the ECCS valves.

MEMBER MARCH-LEUBA: From the reliability point of view, Chapter 19 and the PRA analysis claims very good performance for this first-of-a-kind valves. Is there any plan for confirming that untested performance?

MR. SCARBROUGH: Well, this testing that's going on right now for 50.43(e) is a design demonstration. So, there's not enough tests to come up with a reliability value for that.

MEMBER CORRADINI: They're functionality tests.

MR. SCARBROUGH: Yes, they're functionality testing for 50.43(e), for design certification. So, that gets them to that point. Then, for the COL phase, they have to satisfy QME-1, which is also a qualification.

But once again, they won't be doing enough testing to demonstrate a specific reliability of the

valves. And then, they're going to have OM code testing and things of that nature. We're metering that.

Usually, it takes several years of -- in the NUREG reports that Idaho puts together, and such, and research, it take years to come up with a value that's reasonable for a valve. But to me, if you look at all the numbers in those NUREGs, they're on the range of 1E-3 per demand.

MEMBER MARCH-LEUBA: There are some that have 14 percent failure probability.

MR. SCARBROUGH: Yes, it's --

MEMBER MARCH-LEUBA: Some types of valves.

MR. SCARBROUGH: Yes, it's a -- valves have sort of a range right around that, little bit more, little bit less. But that's --

MEMBER MARCH-LEUBA: You would expect a more complex one to be in the lower range?

MR. SCARBROUGH: Yes, yes, right. And it's going to -- once they finish all of this and then, actually perform the qualification testing and have some experience, I would expect it to be a typical valve.

It is sort of a typical valve, but it's the application that's so unusual, how they're

actually applying it to this situation. So, I think down the road, they will.

But I don't think, right now, we can say there's a specific reliability, but other than that fact that once they finish all this and they get years and years of experience, it's probably going to be a typical valve that we see, from that perspective. Nothing spectacular and probably nothing terrible, it's going to be somewhere in the typical range of 1E-3 or something like that. That's what I would expect. That's where they are.

MEMBER KIRCHNER: But qualify what you mean by 1E-3, failure to function or does that include opening inadvertently?

MR. SCARBROUGH: Typically --

MEMBER KIRCHNER: Does it mean half open? Does it mean -- what does that mean? What are you saying 1E-3?

MR. SCARBROUGH: Well, when I -- there's -- in the studies that are done on valve performance, they look at failure to close, failure to open, spurious opening, they look at all the various factors.

And from a failure to open, failure to close, it's whether or not it will perform its

function right, without leakage. And that's what I expect it to be able to do and that's what they'll have to qualify for. So, it's going to be in that --

MEMBER KIRCHNER: Well, 1E-3 is an awfully low number for a breach of the primary system.

MR. SCARBROUGH: Well --

MEMBER KIRCHNER: It's a really high number in terms of frequency, is what I'm trying to say.

MR. SCARBROUGH: Yes, well, this valve is -- once they have designed it and they install it, it's -- other valves, like safety relief valves, right?, they --

MEMBER KIRCHNER: I know.

MR. SCARBROUGH: -- also have that sort of similar type 1E-10, like on pressurizer safety valves, right? They're sort of the same range as well. And so, you might have that, right?, you might still have that same potential for that type of failure.

But it's consistent with the other type valves that you have used for these safety valve applications, things of that nature. So, it's not out of the range of sort of typical for safety valves, as such, in terms of the reliability.

So, I think that's what we'd expect them to achieve, once they finish the design and do all

this testing we're talking about and they have some years of experience. I think that's -- that would be the best we would look for for this.

It might be a little more reliable, but it's probably going to be a typical valve. It's a spring-loaded valve that works under differential pressure, so it's going to be along that line. Its unusual nature is its application, in terms of (inaudible). But it's first-of-a-kind, so we'll have to see where they go.

MEMBER DIMITRIJEVIC: So, because it's first-of-a-kind, they didn't assume it's actually usual valves, so that kind of get a little back to reliability performance. I just want to talk to Committee, there is two different failure modes.

One failure mode is fails on demand, and the relief valve and the circulation valve will perform differently in the assumptions of the PRA on this thing.

And then, there is a failure to remain open or failure to remain closed, or spurious operation, which is the failure rate per hour. And those are completely different, they usually 1E-7 or 1E-8 per hour.

So, they're dependent on exposure period,

how long you have to remain closed or open. So, this is what you call sort of like failure rate between the test or something. And they will take forever, given that we will not have hundreds of those valves to get any meaningful data on those.

MR. SCARBROUGH: It'll take longer.

MEMBER DIMITRIJEVIC: Because they will not be challenged so often, and there is -- I don't even know what type of test is proposed to be done during the refueling. So, yes.

MR. SCARBROUGH: And there's different aspects. There's solenoid valves, which have sort of history of their performance --

MEMBER DIMITRIJEVIC: Right.

MR. SCARBROUGH: -- and then, there's the main valve, which is hydraulically closed, right?, I don't want to get into too much detail, we're in open session, but it is hydraulically closed and it's held there by that. So, just very -- it's probably going to very reliable in terms of not inadvertently opening, right?, because you have a pressure holding it.

And then, you have the IAB valve, which is kind of this new valve, but typically, it's sitting there, right?, it's sitting there ready to perform its

function. So, it's typically just sort of hanging out, right?, it's not doing much at that point, until it's called up --

MEMBER DIMITRIJEVIC: There may not --

MR. SCARBROUGH: -- to operate.

MEMBER DIMITRIJEVIC: -- be some standby failure mechanism --

MR. SCARBROUGH: Yes.

MEMBER DIMITRIJEVIC: -- we were discussing moisture or some type of thing which may affect.

MR. SCARBROUGH: Right.

MEMBER DIMITRIJEVIC: Also, the PRA assumed that this valve may not need a solenoid and may open on some small delta P itself. They will not be able to test that, so I mean, if this test is different than what is needed to confirm reliable performance of this valve, it just will confirm performance

MR. SCARBROUGH: Yes. And -- yes, this type of test they're doing now is just to demonstrate the design for the certification. Is the design credible?

MEMBER DIMITRIJEVIC: Right.

MR. SCARBROUGH: Can this combination of valves operate properly in this application? That's what they're trying to show right now. They still

have to go through all the qualification steps and they're going to have to develop their in-service testing program.

And they're going to operating these valves. When they shut down for every refueling outage, they will be opening these valves. And then, when they go into an outage, they're going to have to develop their in-service testing program to stroke them and make sure they operate properly and do an inspection. There's going to be a, these valves are so important, there's going to be quite a bit of activity regarding these valves very outage.

MEMBER BLEY: Just two points, for everybody here. We have been promised, when we go visit in a couple weeks, that they're going to walk us through how they've combined the various failure rates from existing valves to cover this complex of valves that kind of go by single names. So, we look forward to seeing that.

Although it was a completely -- in other design certs, there have been cases where kind of a valve that we've seen before, but expanded greatly in size, where they did substantial numbers of tests, thousands, to actually kind of demonstrate the failure rate of these new valves.

We're not seeing any push to do that here. And I'm not sure if in that other design cert, there was a push from the staff that led to it or if it was just something the vendor decided to do on their own.

MR. SCARBROUGH: Yes, I think, if we go back to the AP1000 and the squib valve discussion and such, I think they were making some assumptions on reliability of those particular valves, which may or may not --

MEMBER BLEY: And there were some vacuum breakers in ESBWR --

MR. SCARBROUGH: Yes.

MEMBER BLEY: -- that had extensive testing, too.

MR. SCARBROUGH: Right.

MEMBER CORRADINI: I think Dennis is remembering that as the example case.

MR. SCARBROUGH: Okay.

MEMBER BLEY: Yes, because they did lots of tests on those.

MR. SCARBROUGH: And we're not asking to do that, because I'm not looking for a specific reliability number, I'm looking to see if they satisfied 50.43(e), which is a demonstration of the safety feature.

And then, once they get past that, the COL applicant is going to have to deal with all the QME-1 qualification and things of that nature. But I'm not looking for a specific value for the reliability of the valve.

MEMBER BLEY: I'm concerned about that, because of these, the real significance of these potentially to the risk of the plant. And the PRA right now isn't showing much, because they're using very strict failures rates for these valves.

MEMBER MARCH-LEUBA: Let me ask you a question. In your mind, do you think it is somebody else's job to look at the reliability and it's the PRA guy's job, not mine?

MR. SCARBROUGH: Well, if --

MEMBER MARCH-LEUBA: That's what you said, you're looking at, does it work?

MR. SCARBROUGH: Yes, right. That's my side of it, right? In terms of the reliability values, in terms of the PRA view of, if they come to me and ask me what I think reliability is, I would say, if you want a number, it's 1E-3 per domain. But --

MEMBER MARCH-LEUBA: You gave me the answer I wanted.

MR. SCARBROUGH: -- they haven't come to me. So, that's --

MEMBER MARCH-LEUBA: But -- yes. I take yes for an answer. And the other thing is, this valve is very complex and it has orifices and restrictions and the flows have to be matched very accurately. Indeed, we were told that the first time they built one, they messed up and they had to go over it with a file and make the orifice larger. So, there are failure mechanisms or fouling of these orifices.

MEMBER CORRADINI: If we're getting into closed session, they're --

MR. SCARBROUGH: Yes, I think --

MEMBER MARCH-LEUBA: This is not closed session, this is --

MR. SCARBROUGH: I think we're getting into that area. I think, I appreciate your question and I agree, but I think we should talk about that in closed session, if you don't mind.

MEMBER MARCH-LEUBA: Okay.

MEMBER CORRADINI: Any questions of the current group? Which will probably become the next group also, since they'll be going first in closed session.

MEMBER MARCH-LEUBA: What's the plan for

Chapter 15?

MEMBER CORRADINI: There is no open presentation for Chapter 15.

MR. SNODDERLY: I'm sorry, Mike, there is a note in Chapter 15.

MEMBER CORRADINI: Oh, I thought there wasn't.

MR. SNODDERLY: There it is.

MEMBER DIMITRIJEVIC: I know I look everywhere.

MEMBER CORRADINI: I was told there wasn't.

MEMBER BLEY: I think everything that could go wrong has gone wrong today.

MEMBER DIMITRIJEVIC: Yes, I think we should --

MR. SNODDERLY: We could go -- so, Mike, do you want to just go into closed? If that's okay with the staff, we could go right to just closed session for 15 and discuss.

MS. KARAS: The presentation is an open presentation.

MR. SNODDERLY: Right. So, do you want to do it in an open session then?

MEMBER CORRADINI: Do we have --

MS. KARAS: I mean, it should be open

session, there's nothing that's --

MR. SNODDERLY: Okay, that's fine.

MEMBER BLEY: We shouldn't close unless we have to.

MR. SNODDERLY: Right.

MEMBER CORRADINI: Okay. My apologies, I thought --

MR. SNODDERLY: Well, when --

MEMBER CORRADINI: So, thank you, and we'll get the next group up.

MR. SNODDERLY: So, Mike, if I could make a suggestion? I'm sorry, can I have the floor, please? I -- Bob Weisman from OGC is here and I think that it would be a very good idea for ACRS leadership and Mike Corradini to speak to him, with regards to the status of the section.

MEMBER CORRADINI: Is this open?

MR. WEISMAN: No, this is Bob Weisman from OGC and if we could take a short break and meet in the ACRS chambers for just a few minutes, I would appreciate it.

MEMBER CORRADINI: Okay. So, let's take a 15-minute break, we'll be back a quarter of.

(Whereupon, the above-entitled matter went off the record at 5:27 p.m. and resumed at 5:43 p.m.)

MEMBER CORRADINI: Okay. So it's my apology that somehow I was under the impression that there was no open part of 15; so that's my mistake. So we did it out of order. So, Jeff, are you going to lead us?

MR. SCHMIDT: I'm Jeff Schmidt with reactor systems, and I'm going to be going through the safety evaluation with open eyes for Chapter 15.

These are the reviewers, the same one that we showed in subcommittee, so I'm going to pass this. So this is the agenda for today. So we're going to have unclear open items which we'll try to define as best we can; staff requirements for SECY-19-0036; Chapter 15 limiting cases; we'll have a slide on that.

We'll talk about critical heat flux ratio variance; long-term cooling, Chapter 15 exemptions, return to power, and then finish up with containment structure and containment (inaudible).

So unclear open items: items without a mutual understanding and clearly-defined path for its resolution. So I think these are the important bullets, because that seems like it's kind of nebulous. It includes items for which there may be an agreement between NuScale and the NRC, but documentation supporting information is pending. I

think there is probably a lot that falls under that category.

OUI lists changes frequently as new material is submitted by NuScale; that's also very true. A list of OUIs has been reduced since Chapter 15 SCR draft which is out a month or so.

So here is the list of unclear open items. I think a lot of these are somewhat repetitive, so they kind of fall under the same category, let's put it that way.

Recriticality: again, this is a return to power with natural circulation interrupted, so that's a loss of the riser that we talked about before and earlier today.

Excluding event scenarios of ejected rod with margin for stuck rod. So this is whether you have to include two stuck rods in the rod ejection analysis. That was an open item.

NuScale RAI response pending, expected to support decision at CRDM housings, robust with enhances features precluding the need to evaluate long-term effects such as recriticality within the design basis of returning the short-term analysis.

So if we proceed down this path which is where we're headed right now, the N minus 2 regarding

recriticality would not be evaluated. This event would really focus on the short-term energy deposition and basically keeping the core intact for coolable geometry over, say, a very short time period.

MEMBER MARCH-LEUBA: But then for return to power you would not evaluate --

MR. SCHMIDT: We will not evaluate, yes. That's right. This is more of a core disassembly calculation, just as it would GDC-28.

Boron redistribution: we've talked about that. This is just a potential return to power scenario. We're having continuing dialogue regarding volatility correlation and mixing assumptions. You kind of alluded to before, the mixing and where the boron goes, so that is still ongoing.

MEMBER MARCH-LEUBA: Yes, the way I describe it is, there are physical mechanisms that result in inhomogeneous distribution of boron.

MR. SCHMIDT: Right.

MEMBER MARCH-LEUBA: I don't know where it is.

MR. SCHMIDT: So we are trying to, from the staff, we are trying to determine what that mixing might look like between the core and the riser and effectively, what the changeover rate is.

MEMBER MARCH-LEUBA: Yes.

MR. SCHMIDT: So far, we've been unsuccessful, but we're still attempting.

Nuclear analysis parameters: return to power; again, this is a return to power situation. What do you assume for, say, MTCs or stuck rod where it's basically feedback terms. Decay heat levels precluding a return to power and addressing EOC-ECCS return to power.

Again, a lot of these are return to power. I think we're making good progress on the EOC-ECCS.

Long-term cooling: there's a statement in the long-term cooling report that the analysis demonstrates adequate cooling for 30 days. A NuScale re-evaluating the analytical basis for that statement and potential revisions to that statement.

Just so you know, the long-term cooling report only really goes to 72 hours. That's a basic assumption. But there is a statement in there that discusses a 30-day coping period, and I think we're trying to resolve whether that should really be in there or not.

MEMBER MARCH-LEUBA: You need to coordinate with the Chapter 20 guys, because the Rule 72 NuScale 114, and now you bring in 30?

MR. SCHMIDT: I did not bring in 30, to be clear. That's in the long-term cooling report.

MEMBER MARCH-LEUBA: From NuScale?

MR. SCHMIDT: That's correct.

MEMBER MARCH-LEUBA: So they changed their mind?

MR. SCHMIDT: Well, this was done well before, I think, the 14 days. So this proceeds the 14 days.

MEMBER MARCH-LEUBA: We have enough problems with 72 and 14.

MR. SCHMIDT: Yes, yes. But he's right that the fact that the finish line may be changing.

MEMBER MARCH-LEUBA: Okay.

MEMBER KIRCHNER: Let me just ask a question. Would your review of this matter, long-term cooling after 30 days, provide sufficient information to your colleagues doing Chapter 20 to make a determination? That 14 is half of 30.

MR. SCHMIDT: So just to be clear, the long-term cooling review is only out to 72 hours currently. That's a base assumption. The staff is unsure what to do with this. A single statement in the long-term cooling report that addresses 30 days, mostly regarding ultimate heat sink level --

MEMBER MARCH-LEUBA: That's typically true. There's always a simple limitation.

MR. SCHMIDT: Well, it's a technical report, not a topical report, so there's no limitations.

MEMBER MARCH-LEUBA: There's no limitations?

MR. SCHMIDT: No, no. There either has to be a justification in the long-term cooling report that staff can review or the sets could be --

MEMBER MARCH-LEUBA: A return to holding?

MR. SCHMIDT: Well, the sets could be struck, and it's really a long-term cooling after 72 hours. That's a base assumption. You read the executive summary, you'll see that.

MEMBER CORRADINI: So just one thing for the members who weren't here on the subcommittee; we saw in closed session, which we'll probably have that if we want it tomorrow, just because of the time of day -- but in closed session of the subcommittee we heard from the staff that their -- I'll call it audit calculation analysis went out to 72 hours, right? Because that's what I remember you showing us in various shapes and forms.

MR. SCHMIDT: For return to power or long-

term cooling? See, I separate those --

MEMBER CORRADINI: I think it was return to power.

MR. SCHMIDT: Yes, I think it was return to power also.

So to be clear, the long-term cooling report is just residual and decay heat, but it also only extends to 72 hours. And the return to power also only goes to 72 hours.

MEMBER BROWN: I got it; it was long-term cooling that we were talking --

MEMBER CORRADINI: That's what he said.

MEMBER BROWN: No, he said --

MR. SCHMIDT: In return to power we were showing graphs to 72 hours. long-term cooling we just had a slide and no graphs.

MEMBER BROWN: No, but you were talking about the emphasis from -- yesterday. They only do their assessments for --

MEMBER MARCH-LEUBA: That was yesterday?

(Simultaneous speaking.)

MEMBER CORRADINI: All I said was, the subcommittee meeting from the staff, when they showed us detailed audit calculations for return to power was for 72 hours. Jeff corrected me that the long-term

cooling was just a slide, nothing beyond that, and that was Tuesday.

MEMBER MARCH-LEUBA: But you're talking about two different subcommittees, he is talking about yesterday's RELAP.

MEMBER CORRADINI: No, I'm not talking about yesterday; I'm talking about the subcommittee meeting in June.

MEMBER BROWN: Okay. All right. I was talking about whether they were talking about the longer-term, they only have to do an audit or evaluation up to a certain time by rule or something like that. That's what I was remembering. I got confused; sorry about that.

MEMBER DIMITRIJEVIC: And this is completely different conditions than for Chapter 47. For RELAP this one for long-term cooling they have something government commissions.

MR. SCHMIDT: So the other unclear open items are topical report methodologies; we talked about those in subcommittee as well as RELAP version 1.4 and revised base model. Staff has those and are actively reviewing it.

Steam generator heat transfer uncertainty, which goes to the non-LOCA topical report; still

awaiting revised NuScale analysis, expected to demonstrate that the results are insensitive to the assumed heat transfer coefficient. Really, the 30 percent, I think, is being removed from the non-LOCA topical report uncertainty.

MEMBER CORRADINI: This is secondary site; heat transfer?

MR. SCHMIDT: This is heat transfer from the primary to the secondary.

MEMBER CORRADINI: So, integrated?

MR. SCHMIDT: Integrated.

MEMBER CORRADINI: Okay.

MR. SCHMIDT: Chapter 15 re-analysis from design and methodology changes: we talked about this before. ECCS logic changes; you guys hit that up a little bit today on the RCS level, ECCS actuation going away.

Those ECCS logic changes are expected by August 30th. Decay heat removal logic changes we've also discussed. Those analyses are expected August 30th as well.

And then the RELAP Version 1.4 is also August 30th.

MEMBER CORRADINI: So the first bullet under the last underline --

MR. SCHMIDT: First bullet under the last underline; got it. ECCS logic changes?

MEMBER CORRADINI: Yes. Are we going to hear about that today?

MR. SCHMIDT: Other than we have to evaluate, and it's coming August 30th. No, you're not going to hear any details.

MEMBER CORRADINI: All right. Because I think some of us are still struggling to understand the logic of the logic change. So we're expecting to ask that of NuScale when we go visit them in Corvallis.

MR. SCHMIDT: Yes, I can't speak to those.

MEMBER CORRADINI: That's fine. I just wanted to make sure.

MEMBER KIRCHNER: Jeff, refresh my memory. What's the DHRS logic change?

MR. SCHMIDT: They're separating out, I think -- and I have a backup slide to remind me -- but I think they're basically changing out when you would isolate secondary side versus activate the decay heat removal system.

Actually, if I could ask Boyce, who is the reviewer of that section, he's probably better to speak to it.

MR. TRAVIS: So originally, all the DHRS actuations were integrated, and now that they've revised the DHRS logic, not only for operational concern, but it also resulted in -- there's isolation signals in DHRS actuation signals that have been separated. And so the isolation signals will occur first. Sometimes they occur at the same time as the actuation signals, but not always.

It preserves the inventory in the DHRS with the isolation signal, and then they actuate the DHRS with the actuation valves to start the circulation on the --

MEMBER MARCH-LEUBA: If I remember correctly, DHRS will trip, because the isolation happened, which will cause the pressure to rise and the DHRS will actuate.

MR. TRAVIS: For some transients, yes.

MEMBER MARCH-LEUBA: It's similar to bullet number 1 under the last underline.

MEMBER CORRADINI: Okay, but since we probably won't discuss this again, I want to make sure this -- I want to get the reasoning. In this case the reason is to preserve the inventory.

MR. TRAVIS: So that's the net result, but the reasoning for the change, and you'd have to speak

to NuScale for the exact reasoning, but I believe it's an operational concern related to startup considerations for how their logic interlocks, and this is a consequence of that. But the functional effect of the change in the DHRS system is not --

MEMBER CORRADINI: So it's a startup logic issue.

MR. TRAVIS: Yes, and so the --

MEMBER CORRADINI: Okay.

MR. TRAVIS: -- the effect on -- we expect the effect on the Chapter 15 analyses to be negligible for the purposes of this analysis because in all cases, the DHRS will be isolated at a similar time when it was previously, but it may not actuate the DHRS if it's not required; for instance, if it's an over pool event.

MEMBER CORRADINI: All right; I appreciate it.

MEMBER MARCH-LEUBA: A concern I also have is that by going in this circuitous route, you increase the probability of not doing something you're supposed to do.

Because it makes not much sense unless you're doing (inaudible) power maneuvering that you isolate the heat sink, isolate the secondary and do

not turn DHRS on. And that's what they do; they isolate the secondary, and they wait to see if the secondary will heat up to turn on the emergency.

It works; it works in the computer very well, but it only works if everything works as designed.

MEMBER CORRADINI: You were going to say something?

MR. TRAVIS: No, I think you've captured the -- that's not how I phrased it, but that does capture the --

MEMBER CORRADINI: But I hear what? I'm sorry.

MR. TRAVIS: That's now how I would have phrased it, but that does capture what is being done, yes.

MEMBER CORRADINI: Okay. Sorry.

MEMBER MARCH-LEUBA: Okay, you beat me to it again.

MEMBER BROWN: Before you shift slides, there were these unclear open items, which is a new revised definition of how we do things, which is just fine. You explained the basis, and you said there were three categories, items for which there may be agreement, but documentation supporting information is

pending.

And I guess my question is, there's five bullets under return to power analysis. Is that the first bullet includes items where you all totally agreed with it, but there's documentation pending? Or is that one where we're still looking for something else, and other disagreements are being put aside?

MR. SCHMIDT: So I think NuScale mentioned in their open presentation that some of the return to power analyses are being redone with different methodology than what is currently in the DCA right now.

Those analyses -- and they showed the same results of those analyses -- the staff needs to review those analyses still. So that is work that NuScale is doing that the staff still needs to review, but I think conceptually, you still have to look at it. We can't make a definitive statement at this point, but at least conceptually I think, regarding the EOC ones --

MEMBER CORRADINI: EOC?

MR. SCHMIDT: End of cycle. So it would be a first bullet. I think the second bullet is just rod ejection in lines 2. So the first and fourth and fifth bullets are heading towards resolution.

MEMBER BROWN: Did we do Chapter 15 this morning? I was --

MEMBER MARCH-LEUBA: No.

MEMBER BROWN: The NuScale?

MEMBER MARCH-LEUBA: Oh, yes.

MEMBER BROWN: Yes, I was missing for other reasons. I got here late, so I apologize. That's why I wanted to ask the question.

MR. SCHMIDT: So I think it's fair to say that while we're still calling these unclear open items, some have more clarity than others.

MEMBER BROWN: That we will revisit. I presume it will be revisited to finalize it.

MR. SCHMIDT: Yes.

MEMBER BROWN: At some point with us as well, right?

MR. SCHMIDT: Yes, it has to be, right.

MEMBER KIRCHNER: Okay. Help me a little, though, because your prior slide said items in bigger font, mutually understood and clearly defined path. Which of these bullets on the subsequent page are without mutually understanding and clearly defined? Are there any show-stoppers here, or is it just a question of edit material?

MR. SCHMIDT: So I would say the one at

least in my mind that is probably the most unclear is the boron redistribution which would be ECCS potential return to power.

MEMBER KIRCHNER: Okay.

MR. SCHMIDT: If you wanted to try and flavor these.

MEMBER MARCH-LEUBA: Because all the others I would not call them unclear. We have called them we haven't had time to review them yet, or mostly because of a late submittal.

MEMBER KIRCHNER: Right.

MEMBER MARCH-LEUBA: Because they changed their mind and send us something new.

MR. SCHMIDT: Yes, some of these are, I think, on a relatively clear path forward.

MEMBER MARCH-LEUBA: Those happen in every review. There are a few where we are waiting for the RAI, and we just keep it in line, and we have to keep moving.

MR. SCHMIDT: Okay. Staff requirements for SECY-19-0036: the inadvertent actuation block valve IAB is part of the emergent core cooling system valve and is designed to prevent high pressure RPB blowdown when the DC power supply is lost to the ECCS trip valves.

So that's just a function of the block valve. I think we probably know that already at this point.

Chapter 15 analyses don't assume single failure of the IAB block, and the key is to block. It does include the failure to open, but at high pressures we need it to block first and then open, so it's a two-step phase.

The Commission has determined that the staff should review Chapter 15 of the NuScale DCA without assuming a single failure of the inadvertent actuation block valve to close.

So what that means to Chapter 15 is there's no change to the DCA relative to the IAB single failure.

MEMBER MARCH-LEUBA: IAB valve to close --

MR. SCHMIDT: To block.

MEMBER MARCH-LEUBA: To block?

MR. SCHMIDT: To block.

MEMBER MARCH-LEUBA: So if it failed to block, you should not assume failure to block.

MR. SCHMIDT: We should not assume failure to block.

MEMBER MARCH-LEUBA: Should you assume a failure to open?

MR. SCHMIDT: Yes, we do assume a failure to open.

MEMBER MARCH-LEUBA: No, no; to allow when the pressure is high.

MR. SCHMIDT: Allowed -- no.

MEMBER MARCH-LEUBA: Well, that would be a failure to block.

MR. SCHMIDT: That's a failure to block.

MEMBER MARCH-LEUBA: Not open when the pressure is low.

MR. SCHMIDT: Not open when --

MEMBER MARCH-LEUBA: The needle goes up, blocks, and it never comes back.

MR. SCHMIDT: And it never comes back.

MEMBER MARCH-LEUBA: So the valve never opens when --

MR. SCHMIDT: No, we only take single-failure of some valves to open. We still take that.

MEMBER MARCH-LEUBA: You take it as a global for the whole valve.

MR. SCHMIDT: Yes, I think that's captured by the --

MR. RAD: Hey, Jose, this is Zack Rad. I want to clarify one thing. Treating the valve of IAB as passive relative to the single failure criteria

means that a failure of that device in addition to the initiating event is not assumed. It could be the initiating event.

MEMBER MARCH-LEUBA: Understood, thank you.

MR. SCHMIDT: So this is the Chapter 15 limit cases. What we tried to do here is just give an overview of Chapter 15, so we looked at basically the figures of merit we use in Chapter 15, RCS pressure, steam generator pressure, critical heat flux, and then we have two critical heat flux; one VIPRE, one RELAP, and then collapsed liquid level above the top of the active fuel.

So you can basically go down; again, we're using AOO criteria. I think that's important to keep in mind. And you can basically look at the percentages, go to the right-hand column and look at the percentages of margin that are available. Any questions on that?

And then I'm going to turn it over to Antonio to discuss CHF correlations.

MEMBER MARCH-LEUBA: We want to talk about this in closed session tomorrow so we can talk numbers?

MR. SCHMIDT: No, no. We hadn't planned

on it, not that I'm aware of.

MEMBER MARCH-LEUBA: So I cannot ask numbers?

MEMBER CORRADINI: You can ask difference in numbers; you can't ask exact numbers.

MEMBER REMPE: Is there a concern that you (inaudible)?

MEMBER MARCH-LEUBA: Very significant -- I have a very significant concern, but Antonio, go for it.

MR. BARRETT: Okay. For Chapter 15 CHF correlations, for non-LOCA events where CHF is calculated, the NSP4-CHF correlation is used in the VIPRE subchannel analysis, this has been approved by the NRC and the ACRS has already seen it.

For LOCA and LOCA-like events where CHF is calculated, there's a high-flow correlation and a low-flow correlation which is used in the NRELAP5 systems analysis code, and that methodology is still currently under staff review.

MEMBER CORRADINI: I was -- do you have more to say?

MR. BARRETT: Yes. This is just an intro to get everybody on the same page.

MEMBER CORRADINI: Thank you.

MR. BARRETT: Okay. Now, with regard to the critical heat flux ration variances, there were a couple of different things that we looked at to try and explain where these variances come from, especially in the initial minimum critical heat flux ratios.

So first we looked at the correlation which is the NSP-4 correlation for non-LOCA and then the high-flow and low-flow for the LOCA and LOCA-like events.

And if you take these correlations and apply the exact same conditions, barring anything else, you'll find that the NSP-4 actually provides a lower CHF.

MEMBER MARCH-LEUBA: By lower, meaning what?

MR. BARRETT: So if you had the same --

MEMBER MARCH-LEUBA: Kilowatts?

MR. BARRETT: Yes. And so -- I can't remember the --

MEMBER MARCH-LEUBA: Kilowatts per minute.

MR. BARRETT: Yes. And if you use the same exact efflux, you would get a lower minimum critical heat flux ratio for NSP-4.

So the takeaway is that the variance that

you would see between a LOCA and non-LOCA event is not squarely on the correlation that's used. It provides some variance, but that's not the key takeaway.

So eventually, if you look at the initial conditions, the analysis inputs, there's some biasing going on that are for specific events: the RCS temperature, the RCS flow, the pressurizer pressure; this all has an impact on your critical heat flux ratio and also for NuScale for some events, they looked at different power levels, which would also have an impact.

And finally, when you look at the methodology and you look at the analysis codes that are being used for VIPRE as a subchannel analysis, and it basically only covers the period of interest.

So if you have the minimum critical heat flux ratio starting at time zero, you'll get the first 50 seconds including time zero. If it occurs at time 100 seconds, you'll get 75 seconds to 125 seconds.

So if you look at that plot, you may see what you think is a difference. But as for RELAP, which is for the LOCA-like events, it continuously calculates a minimum critical heat flux ratio.

So finally, when you look at the modeling itself for how you would model something in VIPRE

versus how you would model something in RELAP, VIPRE is a subchannel analysis code, so you can't inherently model it the exact same way as you would in RELAP, especially in terms of peaking factors and the heat flux assumptions.

So if you took RELAP, for example, and you tried to do something consistent with what was done in VIPRE, but you did something that was consistent, but you used more conservative peaking factors or more conservative modeling of those peaking factors, you could really drive down your minimum critical heat flux ratio.

MEMBER CORRADINI: So let me ask the question a different way. This all makes sense, so has staff taken an example and walked through these to show you can replicate within the variance of the two correlations the same ratio?

I mean, what you're saying to me is, there's reasons; here are the reasons. And my next question is, have you done the detective work to show for an example that these reasons all wash out, and you do get the same result for a condition?

MR. BARRETT: Well, I'm not saying you -- for the correlation, yes.

MEMBER CORRADINI: Okay.

MR. BARRETT: For the correlation, I've looked at both correlations and some conditions that are close to the initial normal operating conditions.

MEMBER MARCH-LEUBA: So you plug in pressure, flow, cooling, on both of them, and you get similar CHF values?

MR. BARRETT: Somewhat similar, with NSP-4 being a little bit lower.

MEMBER MARCH-LEUBA: Okay. That's even more concerning. Let me tell you what my complaint is, then I'll tell you what my concern is.

The complaint is, this is a steady-state. I'm not worried about transients; it's time (inaudible) on the scale.

MR. BARRETT: Correct.

MEMBER MARCH-LEUBA: They're off by a very significant margin, and I'm not allowed to say how much it is here because we don't have a closed session. But the amount is scary.

MR. BARRETT: Right. So the point I'm trying to get at is, that is not necessarily related to the CHF correlation. It's more related to how it was modeled.

MEMBER MARCH-LEUBA: And that's why I'm saying I'm even more scared now. It is not the

correlation; it is the model.

MR. BARRETT: Right.

MEMBER MARCH-LEUBA: What makes you think that the model, that is off by a factor of X with respect to the other, in a steady-state, that's any good during the transient if it misses the steady-state by that much?

MR. BARRETT: I hear you. So I guess the point --

MEMBER PETTI: Can you help the uninformed? It misses it because we're in a natural circulation regime.

MEMBER MARCH-LEUBA: No.

MEMBER PETTI: I mean --

MEMBER MARCH-LEUBA: No, no.

MEMBER PETTI: Yes, we've modeled CHF in reactors many, many times. What's unique here that's causing the model --

MEMBER MARCH-LEUBA: I don't know. I know they have two models, and the starter time CO, they initialize the steady-state. And the answer is a factor of X, X being a large number, not 10 percent.

MEMBER KIRCHNER: That should be of concern, because the CHF correlations, by and large, are not developed with transient data; they're

developed by running the rig, the test rig. You just gradually go until you get a burnout or CHF, whatever you want to call it.

And the thermocouples pick that up, and that's how they do it. So it's a quasi-steady-state correlation, and they should be close. They're off by a significant --

MR. BARRETT: Yes. So the correlations, I think when you look at just the correlations it looks somewhat reasonable.

MEMBER MARCH-LEUBA: So the methodology is incorrect.

MR. BARRETT: So the methodology for VIPRE, the subchannel methodology has been reviewed by the staff, has been approved by the staff, and the ACRS has already reviewed that.

MEMBER MARCH-LEUBA: Based on the heat fluxes of this code would expect the VIPRE numbers to be more representative of reality than the RELAP numbers. RELAP numbers are --

MR. BARRETT: Correct. That is a reasonable observation. So when you go to something different, go to RELAP, and you are using something that is acceptable but more conservative, that also could be okay.

So it doesn't necessarily mean that there's anything wrong, but that is still under review by the staff.

MEMBER MARCH-LEUBA: Let me put something else in the record, a concept. If I make a wrong calculation, I compare it as my benchmark, I have a 50-50 chance of being conservative. You fall to the left, you fall to the right.

And if you a highlight for every time I heard that; We didn't hit the benchmark, but we were conservative; I say, Well, you were wrong. You were not conservative. You were wrong. And in this case, I am asking you, why are they conservative? Can you put your hand on (inaudible) and say, I want to be conservative anytime for every condition and during the transient.

MR. BARRETT: Okay. So this is what I'm trying to say. So for VIPRE and the subchannel analysis in NSP-4, that is acceptable, and it's good. For the RELAP, what was done for RELAP is still under staff review, but that is the one that looks to be more conservative.

MEMBER MARCH-LEUBA: That one is the one I don't believe.

MEMBER CORRADINI: Yes, but I think -- I

want him to finish, because what I'm hearing him say is, it's the starting conditions, the correlation you check, there are differences, but not big differences. It's the starting conditions which could be different. It could be a hot channel assumption versus an average channel. It could be how I tend to do what are traditionally accidents versus AOOs. That's what I'm hearing you say.

What I'm still thinking the only way to satisfy him is, get to a root cause on an example to show that they all come in line if you could do that. That's the only way to satisfy his concern.

MEMBER MARCH-LEUBA: I want you to tell me, we're using a hollow peaking factor of 3.0 as opposed to asserting 1.4 because we want to be conservative. That won't be conservative.

MR. BARRETT: That is the what I'm getting at. That's what I'm trying to say.

MEMBER MARCH-LEUBA: But we need to know what it is. We don't license reactors based on what the (inaudible) says.

MR. BARRETT: I agree.

MEMBER MARCH-LEUBA: It may be a peaking factor; it may be the (inaudible). It may be --

MR. BARRETT: Because it's still under

review is why I don't necessarily want to get too far into the details.

MEMBER MARCH-LEUBA: We will see all of this in October-November.

MR. BARRETT: Okay.

MEMBER MARCH-LEUBA: I will give you a pass now, but I won't give you a pass then.

MR. BARRETT: Understood. The point that I'm trying to make is, the peaking factor is what seems to be, when they try to line it up is making it more conservative.

MEMBER MARCH-LEUBA: Right. And the concern I have. That was called recreational complaining. The concern is that by using this conservative assumption, we reach the wrong conclusions. I heard you guys say that the limiting transient is an event opening of (inaudible).

But that's only because the steady-state was too low. The transient was nothing. I mean, you have to get a microscope to see the drop in CHFR for that one.

MR. SCHMIDT: So if you go back to slide 7; this is Jeff Schmidt. Let's go back to slide 7.

MEMBER MARCH-LEUBA: Let me finish. I venture to say that if we stopped at the high CHFR

level with a LOCA methodology, opening two IABs will be perfectly acceptable. We will have to have the Commission issue an order that you cannot operate. We are shooting ourselves in the foot by adding those peaking factors, whatever they are.

MR. BARRETT: So I understand what you're saying, but we review what we're provided, so if they are being --

MEMBER MARCH-LEUBA: An acceptable outcome of this review is to say, No, you did a bad job. Do it again. On occasion, we should do it.

(Simultaneous speaking.)

MR. SCHMIDT: The staff's job is to determine if these are conservative calculations for Chapter 15.

MEMBER MARCH-LEUBA: What I'm asking you is --

MR. SCHMIDT: Not that they could be done better.

MEMBER MARCH-LEUBA: Prove to me that it's conservative. Why is it conservative instead of --

MR. SCHMIDT: I understand.

MEMBER MARCH-LEUBA: If it was wrong, and I happen to be lucky.

MR. SCHMIDT: Yes, right. So just to go

back one point though is, what we try to do in slide 7 when we talk about the VIPRE CHF and the RELAP CHF is, the way I look at really what the margin is, it's the margin relative to its design limit. They all have different design limits, right?

MEMBER MARCH-LEUBA: Right.

MR. SCHMIDT: So this gives you some idea -- and there's really not a lot of spread here between 25 and 24, so I think that's a point well taken, is that that's not overly limiting, relative to single rod withdrawal.

But when you look at the margins, it's not just the starting value; it's how close you get to your design limit for that correlation that matters.

MEMBER MARCH-LEUBA: Yes, but if your biased in your initial CHFR, and one of them is an American utility but much larger than any other one, if you start with a CHFR of 5, then you will get close to the limit.

The delta; if you look at a delta for the 1566, this is the same; it's 0.05. If you didn't -- it didn't drop. I mean, if you look at the figure with your arms extended, it's a flood light.

MR. SCHMIDT: So if you're saying you could redo that one with a different correlation, you

would have --

MEMBER MARCH-LEUBA: Same correlation, proper initial conditions, or adjusting the right methodology.

MR. SCHMIDT: Or using a less conservative methodology.

MEMBER MARCH-LEUBA: My concern is not the methodology. My concern -- the methodology does not benchmark the data. We haven't done a benchmark, but they would apply the methodology --

MEMBER CORRADINI: I'm not going to try to cut you off, but you told me you were going to talk about it in November-December, so have we talked about it enough now?

MEMBER MARCH-LEUBA: No.

MEMBER KIRCHNER: May I ask a question of clarification? So when you got to the fourth event, the inadvertent operation, the ECCS, you're now switching to the RELAP code to do the analysis.

MR. SCHMIDT: Yes.

MEMBER KIRCHNER: Why is the acceptance criteria 1.13 there?

MR. SCHMIDT: That is the -- probably 9595 plus anything you put in for --

MEMBER KIRCHNER: For the RELAP

correlation?

MR. SCHMIDT: Right. And so --

MEMBER KIRCHNER: Because 1.284 is the NSP-4 NuScale correlation.

MR. SCHMIDT: Right. That's correct.

MR. BARRETT: I think you can look at it as the uncertainty from CHF of critical heat plus ratio of one, so that's the uncertainty; 0.013 is the uncertainty associated with that correlation; 0.284 is the uncertainty associated with the NSP-4 correlation, and then wherever you start out with.

So once you get close to it, since it's your uncertainty, once you hit that you basically assume that --

MEMBER KIRCHNER: I guess intuitively what I'm reacting to is that I would expect the uncertainty to be greater for that correlation than for the NSP-4 correlation. And the reason is, that correlation is being used primarily for steady-state operations for the NuScale correlation, and VIPRE is being mainly used for subchannel analysis which is the simpler problem with less uncertainty.

You've got more uncertainty in transient CHF than you do in steady-state. I just don't get it.

MR. SCHMIDT: Just to be clear, VIPRE is

using transients too. It's a transient --

MEMBER KIRCHNER: I know that, but you benchmark the VIPRE NuScale CHF correlation basically on steady-state operations.

MEMBER CORRADINI: I think Jeff has a lifeline.

MR. SCHMIDT: Yes, I'm going to call a lifeline.

MR. DRZEWIECKI: This is Tim Drzweiecki, (inaudible) staff. I have looked on these correlations. One reason why it's lower; both are based on a 9595 limit --

MEMBER KIRCHNER: I know that.

MR. DRZEWIECKI: -- and they're both based on the same test data set. Now, you're right; the NSP-4 is actually a best fit to data and its benchmark as well, but the (inaudible) correlation, which is then fit to this data account with a limit.

One reason why this limit is lower has to do with the same reason why you're probably seeing a more conservative result when you run RELAP is the fact that that methodology and your peaking factors are going to bias your results in a conservative way, such that when you have 9595 limit, you are actually getting a more conservative result, such that you're

going to predict a lower CHF or lower CHFR, usually, so with the base correlation so you can have a lower limit.

MEMBER MARCH-LEUBA: Because your correlation is making an incorrect prediction of the CHF, you give it a lower uncertainty?

MEMBER CORRADINI: No, no.

MR. DRZEWIECKI: I'm saying it's biased lower. It's biased lower; I'm not saying that the error is lower; I'm saying that it's biased lower.

MEMBER MARCH-LEUBA: Okay.

MR. DRZEWIECKI: I think this will be more clear once you walk through the whole CHF review as part of LOCA, but you'll have to see --

MEMBER MARCH-LEUBA: Right now I want to, when we get off the microphone, I'll tell Walt what the difference between the two predictions is. It's unacceptable in my point of view. It's unacceptable in the 21st century. Now, we get in a steady-state a prediction of the critical heat flux that is off by a factor of X, X being a large number.

MR. DRZEWIECKI: I understand what you're saying, but I think if we go and look at how they're setting what their peak PIM power is, and the powers that would surround it, if that is done in a very

conservative way inside of RELAP, that is what we're seeing here.

If you look at what's inside of VIPRE, that's also conservative, but it's not biased nearly as much. In fact, your peak PIM factor is set by your tech specs, where in RELAP it's more.

MEMBER MARCH-LEUBA: What I said earlier, we're not licensing reactors based on a gut feeling or what it may be. We license reactors because we know what it is. So it is the responsibility of the staff to know what it is.

(Simultaneous speaking.)

PARTICIPANT: Antonio's still digging and is trying to make comparisons to understand, is it the peaking factor? Is it? Are you trying to --

MR. TRAVIS: Yes, that's a fair statement. It's still under review. I intend to get to --

MEMBER MARCH-LEUBA: I can do this too. It's not your responsibility. The first thing you discuss do tomorrow morning is issue -- tell me what the difference is. You're the one that wants to do this correlation. It's not your job to be Sherlock Holmes; it's their job.

As I say, it's a perfectly acceptable result of this is, if you give me such disparity in

results, then you don't know what you are doing. Tell me why or go back to the drawing board.

MEMBER BLEY: We've raised the questions; we don't manage the staff.

MEMBER MARCH-LEUBA: I was giving you friendly advice.

(Laughter.)

MEMBER CORRADINI: Okay. Let's move on.

MR. SCHMIDT: All right. ECCS long-term capability and analysis. Analysis addresses residual heat removal for LOCA and non-LOCA events; focuses on reduced inventory events which challenge the ECCS cooling capability; assumes shutdown, which I think is important; evaluates Chapter 72, which we discussed earlier; figures of merit are decreasing clad temperature, minimum RCS temperature will prevent (inaudible) precipitation, and collapsed liquid level remains above the top of the active fuel.

Staff review focused on validation of NIST test data and RELAP model, assumed analysis conditions. In-vessel downstream effects evaluated was formed in DCA Section 6.3, and the staff found the NuScale evaluation acceptable. That refers to the in-vessel downstream effects.

So anything on long-term cooling?

Appendix K exemption; so ANS-71 is replaces with ANS-73 and RELAP5. That's fairly common in RELAP5, that you typically have the 73 in there.

Post-LOCA PWR reflood and refill phenomenon are not to be encountered, so there's the post-CHF regime heat transfer correlations. Baker-Just water metal-water reactions; obviously, if you don't have uncover, you don't have a significant metal -water reaction; clad swelling and rupture.

GDC-27; I'm going to go pretty quick through that. We talked about this before. The staff took the position that GDC-27 meant that you were in a shutdown configuration.

So following an initial shutdown, the NuScale reactor can return to criticality during a cooldown event on either decay heat or lower ECCS. NuScale submitted an exemption to 27, requested approval of a principal design criteria.

Exemption of (inaudible) also included a lack of ECCS injection, which we should know by now.

So we wrote a SECY-180099, which basically had these three success criteria to determine if the exemption was going to be approved, and the first one is basically preserve the SAFDLs four-year events.

The combination of circumstances and

conditions leading to actual post-reactor trip, return to criticality is not expected to occur in the lifetime of the plant.

And there's incremental risk to the public health and safety from hypothetical return to criticality at the NuScale facility with multiple reactor modules does not adversely erode the margin between the Commission's goals for new reactor designs related to estimated frequencies of core damage or large releases and those calculated for the NuScale design.

So that's largely met; if you can meet the SAFDLs, you're going to be meeting that.

ACR supported the proposed staff criteria with the addition of evaluating the overall risk, which was the third bullet above. Satisfying the three criteria in the SECY would ensure no undue risks to public health and safety.

So we talked about this before; three scenarios can lead to return of power, KD removal of cooldown with keeping DC power. The KD removal cooldown without DC power, which is when you're going to actuate the ECCS at the set point, and then just a valid ECCS signal and cooldown can occur from most Chapter 15 events.

Key assumptions: no operator action; only safety-related equipment is used to mitigate the event. The worst stuck rod is assumed out of the core consistent with the current GDCs.

Return to power analysis: that's in Section 1506 of the DCA. It presents a decay heat removal cooldown, retaining single-phase natural circulation with the ECCS valves opening at the maximum return to power, which is thought to bound all the three scenarios that I listed in the previous slide.

In the DCA analysis, maximum core average return to power is 10 percent rated thermal power, with the equilibrium value at about 2.5 percent, and the CHFR limits are met.

A potential decay heat system return to power when water level drops below the riser still needs to be evaluated to ensure the SAFDLs are met. That was just uncovered as part of the review.

EOC-ECCS return to power still needs to be evaluated to ensure SAFDLs are met, and the boron redistribution of potential return to power at times other than EOC still needs to be resolved. Those are the ones we talked about earlier for the unclear open items.

MEMBER CORRADINI: And this is the one where you guys had done some separate analysis?

MR. SCHMIDT: That's correct.

MEMBER CORRADINI: We heard about it in the closed session, in subcommittee.

MR. SCHMIDT: That's correct. So and then the staff still had an action to complete, confirmatory analysis. We presented some results that were single bay, so the bay heated up, which can limit the return to power. And the staff is currently running with basically almost in a cool calculation to see what the return to power looks like. Those are running currently.

And now I'm going to turn it over to containment.

MR. HAIDER: Next slide? My name is Syed Haider. I am going to give a high-level overview of the staff review for NuScale FSAR Section 6.2.1.1 on containment structure which is ameliorated to the peak containment pressure and temperature that is reserved from the limiting containment design, this is events.

This slide captures some of the staff review of the NuScale FSAR Section 6.2.1.1 in the containment response analysis and methodology technical report that is incorporated by reference.

At present the Applicant's design and the staff's confirmatory analysis reserves for the limiting containment design (inaudible).

As presented during the June 20th ACRS subcommittee meeting, the staff reviewed various conservatisms used in the Applicant's RELAP model. As with its initial boundary conditions to ensure that the containment safety analysis of the NuScale power module was conservative.

The staff also performed confirmatory analysis to make sure that NuScale containment design meets the two key regulatory requirements. One requirement that is rooted in GDCs 16 and 15 to ensure that the peak containment pressure and wall temperature can create it for the limiting mass energy of these events are bounded by the NuScale design values with sufficient margin.

The second requirement given by GDC-38 for containment (inaudible) is interpreted by the SRP as to ensure that the containment pressure reduces to 50 percent of its heat value within 24 hours.

So a NuScale application was presented, and we have five modeling (inaudible) for seven containment DBS scenarios that include three LOCA breaks and two anticipated operational occurrences

involving RRV and RVV rads inadvertent of the events; one main steam line break, and one fatal water line break.

The staff performed confirmatory analysis, and using the MELCORE code for the same spectrum of seven NPM DBS scenarios, using the primary and secondary mass and energy of this data supplied by the Applicant, which has been the standard practice in the containment design reviews.

The confirmatory analyses were performed with the worst case single failure and loss of normal AC and DC power. We evaluated the NuScale pressure and temperature response and confirmed that the required regulatory safety criteria were met by the NuScale containment design.

Both the Applicant and staff also ran numerous sensitivity cases of the peak containment pressure and temperature to investigate several key factors such as the condensation heat transfer modeling on the containment inside surface, effective non-condensable gases, release timing, and composition normalization of containment volume and heat structures and liquid (inaudible) effects inside the containment, as well as the cooling pool.

In general, the peak containment pressure

and temperature were found to be not much sensitive to these factors to significantly reduce the safety margins.

This slide compares the staff MELCORE confirmatory analysis results with NDF-5 results submitted by the Applicant for the limiting containment pressure and temperature design basis events.

The limiting pressure event for NPM is an inadvertent opening of area per the circulation valve, while the limiting temperature event is an RCS injection line (inaudible) break.

The initiation of each of these events is followed by subsequent ECCS actuation, while the key pressure and temperature occurs after the ECCS was opened. On the graph on the left, the two MELCORs in blue and green are based on an eleven node and a single node containment volume analysis.

The graph shows a peak containment pressure of 967 psia, predicted by the staff, using MELCOR versus 986 psia that was predicted by NuScale using NDF-5.

Compared to the 1050 psia containment design pressure value, the licensing basis peak pressure value of 986 has six percent margin, while

the staff considers the margin to be sufficient.

The staff confirmatory value of 967 psia with about eight percent margin is even lower than the NuScale predicted value of 986 psia.

So the graph on the left also shows that the containment pressure drops for 50 percent of the peak value well within an hour after the initiation of the event, which is much earlier than 24 hours that was stipulated by the SRP.

On the graph on the right, the MELCOR predicted a peak containment wall temperature of 521 degree Fahrenheit that is even lower than the NDF-5 predicted value of 526 degrees Fahrenheit for the temperature limiting scenario.

So there is a sufficient margin of 24 degrees Fahrenheit in the licensing basis value of 526 degree Fahrenheit to the containment wall, our temperature design value of 550 degrees Fahrenheit.

So I would also like to mention that the staff also performed dependent confirmatory study using the trace code, and the resulting peak temperature and pressure results were even lower than the ones that were predicted by MELCOR.

The staff also performed a number of sensitivity studies using the NDF-5 containment safety

analysis texts that were obtained from NuScale.

So the containment-related safety findings are informed by three independent computer cores; that's MELCOR, NDF-5, and trace.

In summary, the agreement between the NuScale licensing basis calculations and the MELCOR confirmatory analysis results is a reasonable, independent, confirmatory calculation using MELCOR and trace cores show that NuScale containment peak pressure and temperature predicted by Applicant are using NDF-5 are conservative; now SCR Section 6.2.1.1 and does not have any unresolved issues regarding the containment response analysis and methodology. They are either resolved, closed, or settled as confirmatory items.

However, the SCR section still has several open items related to Chapter 15 and Chapter 3, such as the NIST 1 facility scaling distortion review, experimental validation of NDF-5, and the acceptability of increasing the containment design pressure from 1,000 psia to 1,050 psia.

These open items are reviewed under Chapter 3 and LOCA topical report under Chapter 15 and will be resolved there. Once these open items are resolved, the staff will be able to conclude that all

mandatory requirements for the NuScale CSR Section 6.2.1.1 have been met.

So this pretty much concludes my presentation, and since we don't have any major issues, all open items essentially belong to Chapter 15 and Chapter 3, and they are still being reviewed.

MEMBER CORRADINI: Questions by the committee? So this is the -- -so, thank you to the staff. Given our starting conditions, we've ended in a flurry.

PARTICIPANT: We've got one more slide.

MEMBER CORRADINI: I thought --

MEMBER KIRCHNER: Not there, Mike.

MR. TRAVIS: I'll try to go through this as quickly as possible. There's one exemption related to containment; it's GDC-40 exemption. We've discussed long-term cooling already; I won't discuss the first few bullets.

One exemption related to GDC-40 requested by NuScale for containment heat removal; that's testing of the containment heat removal system as the containment heat removal system in this case is the exterior of the pressure vessel submerged in the pool.

NuScale requesting an exemption to the testing requirement in GDC-40. Staff reviewed the

exemption request and determined that the periodic inspections that are required by GDC-39, in addition to the other periodic testing and inspections required of other systems that will transfer heat to the containment via the containment heat removal system will satisfy the requirements associated with GDC-40, and therefore the underlying purpose of the rule, which is to verify that the performance characteristics of the containment heat removal system would remain within acceptable parameters to ensure operability. That will still be accomplished, and therefore the plan to recommend to the Commission grant the exemption to GDC-40.

There's one open item in 622 related to the performance requirements associated with the containment cleanliness program. Those requirements were deferred to the cleanliness program. Because it's a DC-stage, we cannot make a finding on the cleanliness program itself; that would be for COL. So NuScale explicitly identified the limits in 622. We received that response.

MEMBER CORRADINI: This is really to the cleanliness going forward for this --

MR. TRAVIS: That's correct. The degree limits associated with the cleanliness program for the

COL are now explicitly identified in the DC as limits rather than --

MEMBER CORRADINI: Okay. All right, thank you.

MR. TRAVIS: -- allowable values for the (inaudible).

MEMBER CORRADINI: Now, excuse me. Questions by the Committee? Okay. So I think at this point we'll take public comments, and then I have some questions about the need for a closed session and what we do in the closed session.

So can we open the phone line? Is there anybody left in the room that wants to make a public comment?

So the public line should be open. If anybody is out there, can you please acknowledge by speaking up for a moment, please? Anybody on the public line? Anybody on the public line who wants to make a comment?

Okay. Hearing nothing, why don't we close the public line, and now let me make a few suggestions. I've spoken to NuScale. They are ready to present things if we have questions on Chapter 3 and Chapter 6.

And their anticipation is, a lot of the

other proprietary things we might have with them would be covered in two weeks when we're physically visiting Corvallis.

But I wanted to give the Committee a chance to ask questions on Chapter 3 and 6, as they were ready, potentially, with slides if you have questions.

So I'm looking at you, Bob; missiles? I'm looking at --

MEMBER BLEY: I was expecting them to go through the missile discussion when we're out there, so I haven't prepared by studying the last document they sent.

MEMBER CORRADINI: Which we now have.

MEMBER BLEY: Which we do have.

MEMBER CORRADINI: We have.

CHAIR RICCARDELLA: There were actually two RAI responses. Did you see that? One was kind of general, and --

MEMBER BLEY: We got two; there's actually six or seven.

CHAIR RICCARDELLA: Okay.

MEMBER CORRADINI: So my question is, so let me just lay out the grand plan. The grand plan is, it's very late. We're going to lose some of the

staff, because there's a train leaving for their potential homes, and they're not going to get anywhere else.

So if we want a closed session with the staff, it's got to be tomorrow. If we want a closed session with NuScale, they are ready and willing, but they have stuff only if we have questions. They're not going to come up and start giving us a presentation.

MR. MELTON: So for Corvallis, we were anticipating doing turbine missiles here. This is where we wanted to talk about it. I think we would like to present, if at all possible, tomorrow, on turbine missiles.

Chapter 6, I think we're okay with going with a closed session on 6. But 3, we do want to engage in 3 and present some more information. So if we can do that tomorrow, that would be super.

MEMBER CORRADINI: That's fine.

PARTICIPANT: And that will be in closed session.

MR. MELTON: Yes. That's closed session, and there's details we'd like to present and talk about.

PARTICIPANT: Okay. That sounds good.

MEMBER CORRADINI: I knew that you had something; I didn't know if you wanted to present or wanted to ask questions about.

MEMBER REMPE: Do you know that we can't do this at 8:30? We have a PMP that's going to last for a couple of hours before we can do that. Is that okay still, with your plans for travel?

MR. MELTON: We'll be fine with that. We're good with that, so later in the morning is good.

MS. KARAS: And the Chapter 15 staff, I think we're good until about 7:15, if you have questions.

(Whereupon, the above-entitled matter went off the record at 6:44 p.m. and resumed at 6:45 p.m.)

MEMBER CORRADINI: Well, I haven't finished. Do we have questions of the staff on 3 or 6 that we want to drag them up in closed session now, since we have 30 minutes before we're going to lose them, or do we want to wait for tomorrow?

CHAIR RICCARDELLA: On 3, shouldn't the staff go after NuScale?

PARTICIPANT: Right.

MEMBER BLEY: Well, the staff has told us all they know on the --

MEMBER CORRADINI: The staff is going to

review what they've got --

MR. COYNE: They haven't reviewed it all yet, for missiles.

MEMBER BLEY: I don't think you're looking to tell us anything more on missiles.

MR. MELTON: So our turbine missile analysis staff are not here. But they just recently got an information submittal from NuScale. They've done a preliminary review. They have questions --

MEMBER BLEY: And they told us about that. I don't think they have anything more to tell us at this time.

MEMBER CORRADINI: So unless I hear great objections, we can either do, in the next 30 minutes, before I start losing a lot of people, we can do a closed session on topics on Chapters 3 and 6 if the members want, or we wait until tomorrow at 10:15.

Because I have another assignment for the members now, and that's to take a look at the letter, because we have it ready. We thought we were going to get to it, but I have it printed and ready, based on input by Dennis -- well, hang on. I'm just trying to tell everybody.

So what's your pleasure? My thought is to let go everybody here unless they want to watch us

make sausage.

PARTICIPANT: I think that's a good idea.

MEMBER CORRADINI: All right. And we would pick it up tomorrow in closed session, because then we're going to get some of the input from the members tonight and have a different letter, or hopefully not a too different letter by tomorrow.

MEMBER REMPE: You're planning on doing an entire read-through starting at 6:45?

MEMBER CORRADINI: I'm asking.

MEMBER REMPE: Okay. But that's what you're suggesting.

MEMBER CORRADINI: That's what I'm --

CHAIR RICCARDELLA: Yes, why don't we just read it? Yes.

MEMBER MARCH-LEUBA: You'd have to do this tomorrow. It's mandatory to read it. It's part of --

MEMBER BROWN: He's a speed reader; don't worry. You won't understand a word he says.

MEMBER REMPE: I think we could do a read-through tomorrow. It's 6:45 --

MEMBER CORRADINI: Fine. So you want to get a copy of the letter tonight?

(Simultaneous speaking.)

MEMBER CORRADINI: I've gotten changes

through the day on a continuing basis. So we're now at version 7. I've got input from Dennis and Jose and Joy and Dick and somebody else; I can't remember who.

MEMBER BROWN: Can we get a printed copy now?

MEMBER CORRADINI: Yes, and you can take it home and read it. I've read it a number of times. I'm sick of looking at it.

PARTICIPANT: Let's have it.

MEMBER BROWN: I won't read it until I get it my hot little hands.

MEMBER CORRADINI: So at this point I'm going to turn this over to the Chairman.

CHAIR RICCARDELLA: Okay. So we are in recess until 8:30 tomorrow morning.

(Whereupon, the above-entitled matter went off the record at 6:48 p.m.)

July 9, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Full Committee Presentation: NuScale Topical Report-Evaluation Methodology for Stability Analysis of the NuScale Power Module," PM-0719-66233, 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) NuScale Full Committee Meeting on July 10, 2019. The materials support NuScale's presentation on the "Evaluation Methodology for Stability Analysis of the NuScale Power Module" topical report.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Matthew Presson at 541-452-7531 or at mpresson@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Robert Taylor, NRC, OWFN-7H4
Michael Snodderly, NRC, TWFN-2E26
Samuel Lee, NRC, OWFN-8H12
Gregory Cranston, NRC, OWFN-8H12
Bruce Bavol, NRC, OWFN-8H12

Enclosure: ACRS Full Committee Presentation: NuScale Topical Report-Evaluation Methodology for Stability Analysis of the NuScale Power Module, PM-0719-66233, Revision 0

Enclosure:

ACRS Full Committee Presentation: NuScale Topical Report-Evaluation Methodology for Stability Analysis of the NuScale Power Module, PM-0719-66233, Revision 0

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

ACRS Full Committee Presentation

NuScale Topical Report

Evaluation Methodology for Stability Analysis of the NuScale Power Module

July 10, 2019

Presenters

Dr. Yousef Farawila (*)
System Thermal Hydraulics

Matthew Presson
Licensing Project Manager

(*) On the phone

Agenda

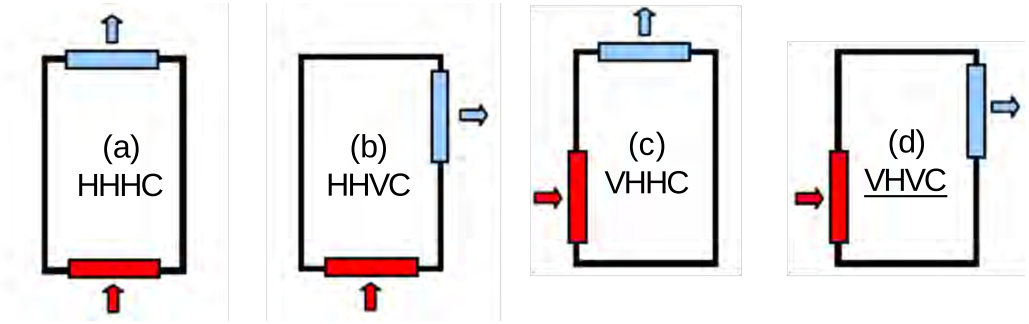
- Introduction and Main Message
- Stability Solution Type
- Stability Investigation Description
  - Theoretical
  - Numerical Using New Code PIM
  - Experimental Benchmark
- Procedure and Methodology
- Summary
- Questions and Discussions

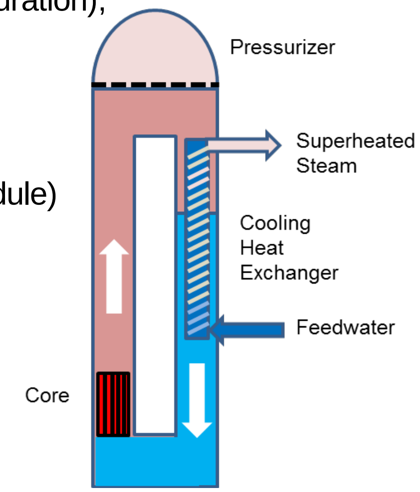
The Main Message

- The NuScale power module design was found to be stable in the entire range of normal operation
- Outside of normal operation, the reactor is destabilized when the riser flow is voided, however
  - Unstable flow oscillation amplitude is limited by nonlinear effects and the critical heat flux ratio actually improves
- The stability threshold is protected by scram upon loss of riser inlet subcooling
  - Conceptually equivalent to a “region exclusion” not a “detect and suppress” solution type
  - No action required to implement a stability solution hardware
- These conclusions are based on extensive first principles, experimental, and computational studies. Details next.

Stability Evaluation

- Natural circulation instabilities were reported in literature
  - See for example D.S. Pilkhwal et al., "Analysis of the unstable behaviour of a single-phase natural circulation loop with one-dimensional and computational fluid-dynamic models," Annals of Nuclear Energy 34 (2007) 339–355.
    - a) HHHC: horizontal heater and horizontal cooler (the only unstable configuration);
    - b) HHVC: horizontal heater and vertical cooler;
    - c) VHHC: vertical heater and horizontal cooler;
    - d) VHVC: vertical heater and vertical cooler (qualitatively like NuScale module)





- Investigation of the NuScale module stability commenced to demonstrate stability, identify threshold conditions, and license stability protection methodology

Stability Investigation Elements

Module Design and Operational Domain
First Principles Theory -- Experience -- PIRT

Construct Models and Main Code (PIM)

Independent Models Reduced Order Model (RADYA)

Experimental Data and Benchmarking (NIST-1)

Comprehensive Analysis & Results
- Steady State Perturbations
- Stability during Transients

Conclusions:
- Stable within Operating Domain
- Threshold is Riser Voiding

Stability Solution:
- Protect riser subcooling with margin
- Hardware already part of MPS

Theoretical Investigation

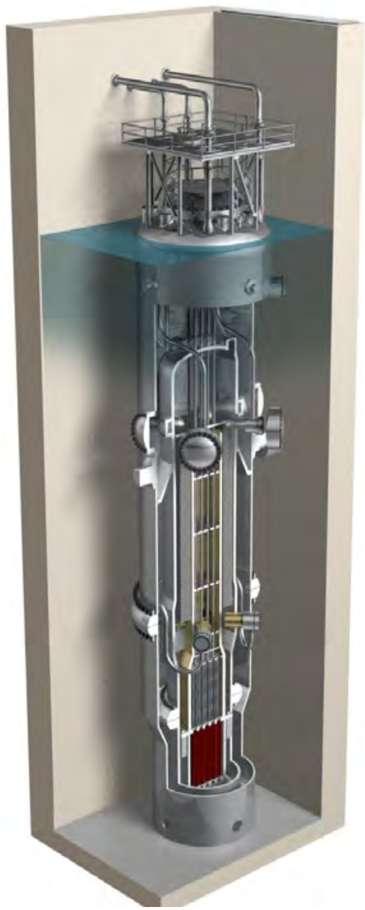
- Kick off with an expert committee to generate a first PIRT
- Scoping review of thermalhydraulic instability modes and contrasting with the NPM design features
- Identification of the possible instability mechanism
- Analysis from first principles
  - Riser-only mode (separate from cold leg)
  - Stability trend with power using a simple SG model
  - Shows decoupling of possible oscillations in SG tubes from primary flow and core
  - Informs design of stability experiments
- All medium ranked phenomena treated as highly ranked

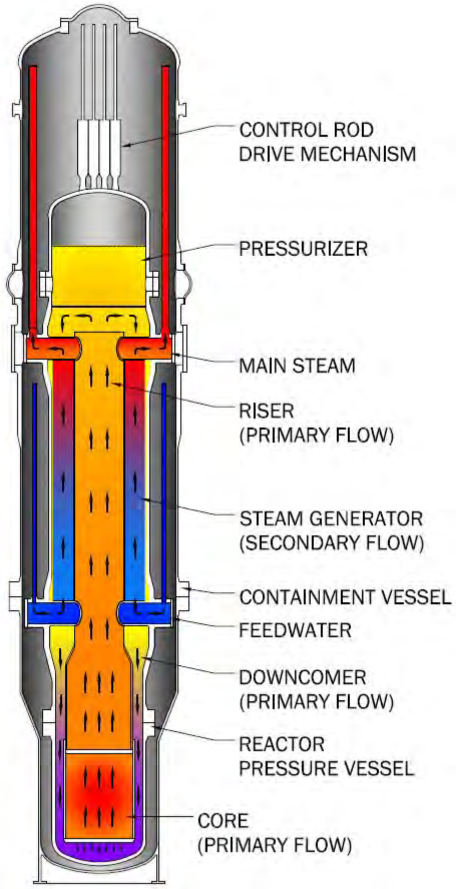
Theoretical and First Principles

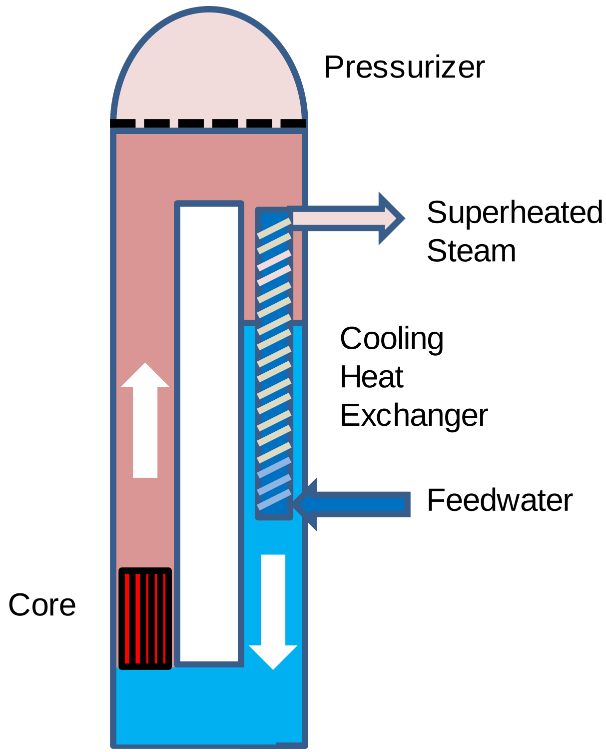
- A system with feedback processes may undergo oscillatory instability if the feedback is:
  - Negative (positive feedback is unconditionally unstable)
  - Delayed
  - Sufficiently strong
- NuScale natural circulation mode is examined
  - Feedback is negative. A perturbation increasing core flow decreases exit temperature thus decreases riser density head
  - Feedback is delayed. Transport delay for core exit condition to fill the riser and reach maximum density head effect.
  - Feedback strength is related to liquid thermal expansion and possibility of phase change, riser length, SG characteristics, reactivity feedback… Requires detailed modeling

Main Stability Analysis Tool: PIM

- Transient 1-D 2-phase non-equilibrium primary loop flow







Model Equations of the PIM code

- Thermalhydraulic conservation equations
  - Liquid and vapor mass balance

$$\frac{dM_{l,n}}{dt} = \dot{m}_{l,n-1} - \dot{m}_{l,n} - \Gamma_n$$

$$\frac{dM_{g,n}}{dt} = \dot{m}_{g,n-1} - \dot{m}_{g,n} + \Gamma_n$$

  - Mixture momentum conservation with drift flux (integrated momentum)

$$\frac{dI}{dt} = \Delta P_{grav} - \Delta P_{friction} - \Delta P_{local} + \Delta P_{resid}$$

  - Energy conservation (assume saturated vapor)

$$\frac{d}{dt}\left(M_{l,n} h_{l,n}\right) = \dot{m}_{l,n-1} h_{l,n-1} - \dot{m}_{l,n} h_{l,n} - \Gamma_n h_{fg} + \dot{Q}_n$$

t	time
M_l	liquid mass
M_g	vapor mass
$\dot{m}_l$	liquid mass flow rate
$\dot{m}_g$	vapor mass flow rate
Γ	rate of evaporation
I	integrated momentum
ΔP_{grav}	gravitational press. drop
$\Delta P_{friction}$	friction pressure drop
ΔP_{local}	local pressure drop
ΔP_{resid}	residual pressure drop
h_l	liquid enthalpy
h_{fg}	latent heat
Q	power
n	control volume index
$n-1$	upstream index

Model Equations of the PIM code

- Point Nuclear Kinetics

$$\Lambda \frac{d\Phi}{dt} = \beta(\rho - 1)\Phi + \lambda C$$

$$\frac{dC}{dt} = \beta\Phi - \lambda C$$

C	Concentration of the delayed neutron precursors
λ	Decay constant of the delayed neutron precursors
Φ	Neutron flux amplitude
β	Delayed neutron fraction
Λ	Prompt neutron lifetime
ρ	Reactivity

  - Thermalhydraulic model provides reactivity input
    - Moderator density reactivity feedback model (equivalent to moderator temperature reactivity under single-phase flow)
    - Doppler fuel temperature reactivity feedback
  - Heat source from neutron kinetics feeds back to thermalhydraulics
    - Energy deposited in fuel pellets (proportional to neutron flux)
    - Fraction of fission energy deposited directly in coolant
    - Decay heat: input by the user as fraction of initial power

Model Equations of the PIM code

- Heat conduction in fuel rods
  - Pellet conductivity is function of temperature and burnup
  - Driven by energy deposited in fuel pellets
  - Heat flux at outer rod surface as power source to coolant
  - Pellet temperature needed for Doppler reactivity
- Heat transfer models for heat sink (steam generator)
  - Secondary side flow is driven by user-provided inlet forcing function
  - Secondary flow is subcooled, 2-phase equilibrium, and superheated
  - Primary flow parameters calculated from transient conservation equations
  - Heat transfer between primary and secondary flow
    - Heat transfer correlations
    - Transient heat conduction in tube walls

Model Equations of the PIM code

- Closing Relations and Correlations
  - Frictional pressure drop (single- and two-phase friction and local losses)
  - Drift flux parameters
  - Non-equilibrium evaporation and condensation model
  - Thermodynamic properties for water
  - Physical material properties (pellets, cladding, SG tubes)
  - Pellet-clad gap conductance
  - Reactivity coefficients as functions of exposure and moderator density
- What is not modeled
  - Pressurizer; pressure is input provided constant or forcing function
  - Heat transfer through riser wall, adiabatic riser is default option
  - Heat capacity of structures; only ambient heat losses through vessel

Flow Stability in Steam Generator Tubes

- Secondary side flow changes from single-phase liquid, to two-phase mixture, ultimately to superheated steam
- Flow in the tubes is subject to density wave instability
- Experiments demonstrate flow oscillations under certain conditions (generally low flow)
  - Oscillations in different tubes are not phase-locked
  - Effect on primary flow cancels out, confirming first-principle finding
- No impact on the primary flow and core stability
  - No feedback loop between primary and secondary oscillations

PIM Results of Perturbing Steady State

- Purpose is to calculate stability parameters of decay ratio and period at different conditions of power and exposure
  - Following a user-applied small perturbation flow will oscillate
  - Oscillations will grow with time if system is unstable
  - Oscillations will decay eventually returning to the pre-perturbation state if the system is stable
- Stability parameters, decay ratio and period, are extracted from the transient output. Observations:
  - Unconditional stability in the entire operational range
  - DR decreases with power and exposure
  - Period also decreases with power
  - Observations agree with the independent Reduced Order Model

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PIM Application Methodology

- For perturbations of steady state to get DR
  - Vary power within 5-100% of rated
  - BOC and EOC, and any point in between if warranted
  - Conservative assumptions for MTC and decay heat fraction
- For a depressurization transient (scram not credited)
  - Verify that unstable oscillations limit cycle without CHFR decrease
- Stability conclusion is generic, but confirmation is needed
  - For plant upgrades such as power uprates
  - Plant operation changes such as operating temperatures and maximum boron concentration
  - Changes in fuel design that would change natural circulation flow

Long Term Stability Solution

- Region Exclusion for NuScale
  - Unstable region defined by a single parameter (core exit subcooling)
  - Monitor and protect margin to riser exit subcooling (with temperature margin below saturation point at pressurizer pressure)
  - Operator alarm when subcooling margin is approached
  - Riser exit subcooling will be controlled by the reactor protection system as part of normal operating limits – not only for preventing instabilities
  - Generic solution: there are no fuel or cycle design elements

Summary and Conclusions

- Stability of the NuScale module was evaluated using a dedicated code (PIM) and supported by first principles analysis and experimental data benchmarking
- The module was found unconditionally stable within normal operation domain using conservative criterion
- Stability boundary identified as associated with riser voiding (loss of riser inlet subcooling)
- Stability protection methodology protects riser inlet subcooling with a margin to define the exclusion region enforced by the module protection system with scram

Portland Office
6650 SW Redwood Lane,
Suite 210
Portland, OR 97224
971.371.1592

Corvallis Office
1100 NE Circle Blvd., Suite 200
Corvallis, OR 97330
541.360.0500

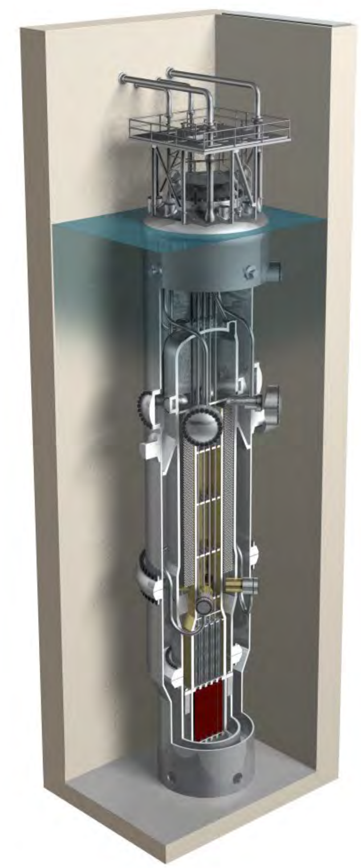
Rockville Office
11333 Woodglen Ave., Suite 205
Rockville, MD 20852
301.770.0472

Charlotte Office
2815 Coliseum Centre Drive,
Suite 230
Charlotte, NC 28217
980.349.4804

Richland Office
1933 Jadwin Ave., Suite 130
Richland, WA 99354
541.360.0500

Arlington Office
2300 Clarendon Blvd., Suite 1110
Arlington, VA 22201

London Office
1st Floor Portland House
Bressenden Place
London SW1E 5BH
United Kingdom
+44 (0) 2079 321700



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Presentation to the ACRS Full Committee

Staff Review of NuScale Topical Report

TR-0516-49417-P, REVISION 0
"EVALUATION METHODOLOGY FOR STABILITY ANALYSIS OF THE NUSCALE POWER MODULE"

Presenters:

Peter Yarsky, Ph.D.- Senior Reactor Systems Engineer, RES

Ray Skarda, Ph.D.- Reactor Systems Engineer, RES

Bruce Bavol - Project Manager, Office of New Reactors

July 10, 2019

(Open Session)

NRC Technical Review Areas/Contributors

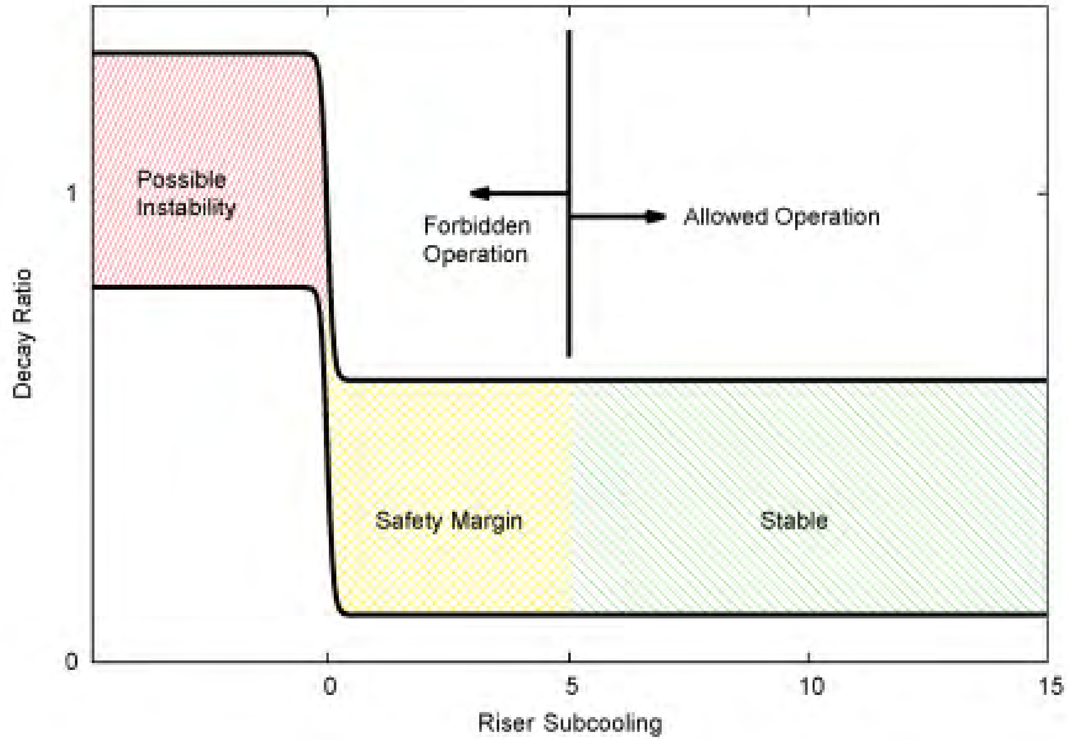
- **Ray Skarda** - RES/Division of Systems Analysis (DSA)/Code and Reactor Analysis Branch (CRAB)
- **Peter Yarsky** - RES/DSA/CRAB
- **Rebecca Karas** (BC) - NRO/Division of Engineering, Safety Systems and Risk Assessment (DESR)/Reactor Systems, Nuclear Performance, and Code Review Branch (SRSB)

Outline
Primary Review Areas

- Regulatory Criteria
- Long Term Stability Solution
- Instability Modes and Phenomenology
- PIM Evaluation Model
- Impact of Secondary Side Instabilities on Primary Side Stability
- Other Secondary Side Stability Concerns
- Uncertainty and Acceptance Criteria
- Stability with Worst-Rod-Stuck-Out (WRSO)
- Stability Topical Report Conclusions
- Design Certification Document (DCD) 15.9 Stability

Long Term Stability Solution (LTSS)
Exclusion Region Based Solution

- The NuScale LTSS is based on an exclusion region principle. GDC 12 and GDC 10 are met by preventing instabilities that could challenge specified acceptable fuel design limits (SAFDLs).
- The Module Protection System (MPS) precludes instability by enforcing riser subcooling margin and tripping the reactor. GDC 13, GDC 20, and GDC 29 are met by operation of the MPS to sense adverse conditions and trip the reactor.



PIM Evaluation Model

The PIM Evaluation Model is Simple but Acceptable

- The PIM evaluation model includes simple models for thermal-hydraulics, reactor kinetics, fuel thermal-mechanical response, and steam generator tube heat conduction and heat transfer.
- Integral validation provided against NIST-1 stability tests.

Decay Ratio (DR) Acceptance Criterion

- DR is insensitive to variations in most of the important phenomena over the PIM application range.
- DR Acceptance Criterion affords sufficient margin to account for bias and uncertainty.
- Numerical effects were considered as part of the DR bias.

ACRS SC Follow-up on SG Secondary Side Instability

Secondary side oscillations are not safety significant with respect to reactor (primary side) stability

- Staff finds that secondary side flow oscillations will not challenge the thermal margins.
- Applicant’s analysis demonstrates that severe secondary flow oscillations would result in a MPS protective trip on reactor power or superheat.
- Analyses performed by the applicant demonstrate flow oscillations that are not safety significant.
  - Chapter 3 technical staff will discuss SG thermal fatigue concerns as part of the DCD Chapter 3 review.

Stability Topical Report Conclusions
PIM-based Stability Analysis Method is Acceptable

- PIM is a simple model, but its models are anchored to upstream, high-fidelity models to improve accuracy.
- The DR is highly insensitive to variations in important phenomena and their models, leading to relatively small uncertainty in the DR.
- PIM predictions in steady-state and transients have been confirmed by the staff with independent TRACE confirmatory calculations.
- PIM is acceptable for performing stability analysis for the NuScale power module.

Stability Topical Report Conclusions
LTS Solution is Acceptable

- Primary instability mechanism properly identified by the applicant and confirmed by independent staff TRACE analysis.
- During normal, at power operation, the NuScale power module is very stable.
- The exclusion region based LTS solution is effective in preventing the reactor from becoming unstable during normal operation including the effects of AOOs.
- Potential instability during return to power with WRSO is not a safety concern.
- GDCs 10, 12, 13, 20, and 29 are met.

Questions?

Backup Slides

Staff Review Timeline

- NuScale submitted the Topical Report (TR) TR-0516-49417-P, "Evaluation Methodology for Stability Analysis of the NuScale Power Module," on July 31, 2016, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16250A851). Applicant provided supplemental information by letter dated December 3, 2016 (ADAMS Accession No. ML16340A756).
- Staff issued 62 requests for additional information (RAIs) with NuScale providing responses – all responses were resolved/closed.
- Staff plans to issue its final SER in late August 2019.
- Staff plans to publish the "-A" (approved) version of the TR in late November 2019.

Regulatory Criteria

General Design Criteria (GDCs) from Design-specific Review Standard 15.9.A

- GDC 10, “Reactor Design,” requires that specified acceptable fuel design limits (SAFDL) not be exceeded during any condition of normal operation, including conditions that result in unstable power oscillations with the reactor trip system available.
- GDC 12, “Suppression of Reactor Power Oscillations,” requires that oscillations be either not possible or reliably and readily detected and suppressed.
- GDC 13, “Instrumentation and Control,” includes requirements for the hardware implementation of long term stability (LTS) solution.
- GDC 20, “Protection System Functions,” requires the reactor protection system to initiate automatic action so unstable power oscillations are avoided.
- GDC 29, “Protection against Anticipated Operational Occurrences,” requires stability LTS solution design for an extremely high probability of accomplishing safety functions.

Non-Proprietary

Instability Modes

- Dynamic and static instability modes were considered.
- Applicant identified and evaluated many modes.
- The applicant’s findings in terms of modes are consistent with staff findings from an independent Phenomena Identification and Ranking Table development process.

Stability with WRSO

NuScale is stable at intermediate pressures

- Applicant conservatively analyzed stability for intermediate pressures (i.e, before emergency-core-cooling-system (ECCS) actuation).
- Strong moderator feedback increases likelihood of recriticality with WRSO, but strong moderator feedback is stabilizing.
- Applicant's analysis demonstrates stability margin at intermediate pressures.

Stability with WRSO

NuScale will experience mild flow oscillation at low pressure

- After ECCS actuation, level drops below the riser, natural circulation flow pattern is broken and flow oscillations occur where core flow is driven by density head differences provided by void formation in the core region.
- Analyses performed by the applicant demonstrate flow oscillations that are not safety significant.

DCD 15.9 Stability Review

Stability Performance during Steady State Conditions is Acceptable

- Stable under steady-state conditions
  – Analyses demonstrate at all power levels > 5 percent of rated that the DR remains well below the acceptance criterion.
- Transients analyzed
  – Certain events result in new stable, steady state conditions.
  – Certain events result in reactor trip due to MPS enforcement of the exclusion region prior to the onset of instability.

DCD 15.9 Stability Review

Stability Performance during Transients is Acceptable

- All AOO classes considered in the applicant's analysis.
  - Increase in heat removal by the secondary system
  - Decrease in heat removal by the secondary system
  - Decrease in reactor coolant system flow rate
  - Increase in reactor coolant inventory
  - Reactivity and power distribution anomalies
  - Decrease in reactor coolant inventory
- LTS Solution is effective in preventing the occurrence of instability.
- Therefore GDCs 10, 12, 13, 20, and 29 are met.

DCD 15.9 Stability Review

Increase in Heat Removal by the Secondary System

- Analysis consistent with the stability analysis methodology topical report.
- Applicant analyzed maximum feed flow increase that does not produce an automatic, prompt MPS trip.
- PIM calculations confirm that the reactor remains stable

DCD 15.9 Stability Review

Decrease in Heat Removal by the Secondary System

- The staff reviewed the feedwater (FW) flow reduction event analyzed in the DCA.
- The DCA demonstrates that even mild FW flow events will progress in similar manners, eventually leading to a MPS trip based on hot-leg temperature (i.e., the LTSS).
- Therefore, the staff finds that the LTSS is effective in preventing the reactor from reaching an unstable condition.
- The staff review of the DCA analysis does not impact the staff review of the Stability TR.

DCD 15.9 Stability Review

Decrease in Reactor Coolant System Flow Rate

- CVCS pump over-speed would be considered an AOO.
- The staff found that a CVCS flow reduction or a reduction in secondary side heat removal event would bound this class of events.
- Analysis of CVCS pump over-speed is not required to demonstrate compliance with GDC 12.

DCD 15.9 Stability Review
Increase in Reactor Coolant Inventory

- Events that increase inventory but also increase reactor coolant system pressure are non-limiting because higher pressure increases stability margin.
- Staff considered events such as CVCS or spray malfunction that could increase inventory but maintain pressure, and these would be bounded by events that increase or decrease secondary side heat removal.
- Analysis of this class of event is not required to demonstrate compliance with GDC 12.

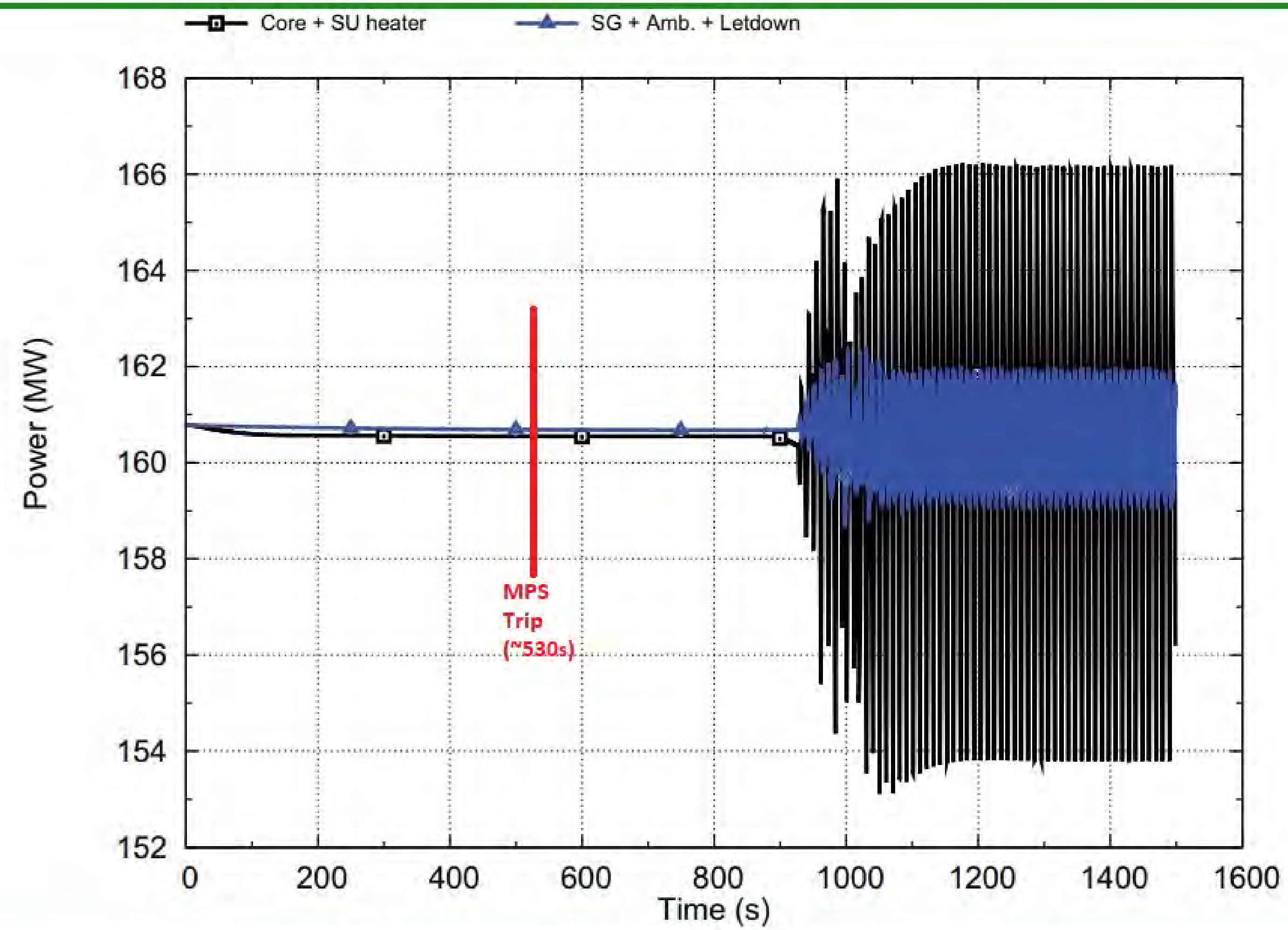
DCD 15.9 Stability Review
Reactivity and Power Distribution Anomalies

- A bounding event is analyzed based on a maximum control rod withdrawal that does not produce an automatic, prompt MPS trip based on high flux or high flux rate.
- PIM calculations confirm that the reactor remains stable.

DCD 15.9 Stability Review
Decrease in Reactor Coolant Inventory

- The limiting event is a slow depressurization where the low pressure trip is not credited.
- PIM calculations show that, eventually, the reduction in pressure leads to low riser subcooling, which initiated the LTS solution MPS trip to enforce the exclusion region.
- PIM calculations demonstrate that the onset of instability would occur well after the reactor is shutdown by control rods.

DCD 15.9 Stability Review
LTS Solution MPS Trip Timing



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July 9, 2019 Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Titled "ACRS NuScale Full Committee Presentation: NuScale FSAR Chapter 3, Design of Structures, Systems, Components and Equipment," PM-0719-66231, 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) NuScale Full Committee Meeting open session on July 10, 2019. The materials support NuScale's presentation of Chapter 3, "Design of Structures, Systems, Components and Equipment," of the NuScale Design Certification Application.

The enclosure to this letter is the nonproprietary version of the presentation titled "ACRS NuScale Full Committee Presentation: NuScale FSAR Chapter 3, Design of Structures, Systems, Components and Equipment," PM-0719-66231, 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

Distribution: Robert Taylor, NRC, OWFN-8H12
Michael Snodderly, NRC, OWFN-8H12
Samuel Lee, NRC, OWFN-8H12
Gregory Cranston, NRC, OWFN-8H12
Marieliz Vera, NRC, OWFN-8H12

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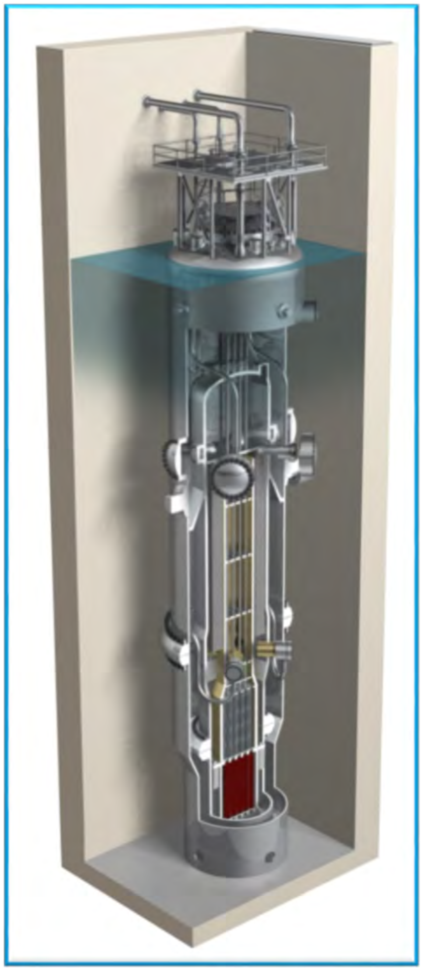
ACRS Full Committee Presentation



NuScale FSAR

Chapter 3

Design of Structures, Systems, Components and Equipment

July 10, 2019

PM-0719-66231
Revision: 0

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Template #: 0000-21727-F01 R5

Presenters

Marty Bryan

Licensing Project Manager

Amber Berger

Civil/Structural Engineer

Olivia Hand

Mechanical Engineer

Hannah Rooks

Supervisor, NSSS Systems Analysis

Turbine Missiles

ACRS Full Meeting on Turbine Missiles

- Illustrations of turbine missile trajectory and barriers
- Turbine missile barrier analyses details
  - Local analyses
  - Global analysis
- Turbine missile barrier results
- Conservatisms in NuScale turbine missile barrier approach

CVAP and TF-3

CVAP and TF-3 Discussion

- CVAP with particular emphasis on the purpose of TF-3 testing, including any possible feedback from recent testing:
  - TF-3 is a 5-column, full height, prototypic support test specimen. Modal testing (both in-air and in-water) and flow testing (unheated, primary side flow only) is planned.
  - The purpose of TF-3 testing is to validate the frequencies, mode shapes, and damping values used in the design analyses. The flow testing will provide the critical velocities and RMS vibration amplitudes of the SG tube bundle due to primary coolant flow. The testing will validate the safety margins for vortex shedding, fluid elastic instability, turbulence
  - Preliminary modal testing has shown good agreement with design analysis frequencies and damping.

SG DWO

SG DWO – FSAR Design Work

- Design objective for SG is to withstand all loadings for design life of the component.
- NRELAP5 SG model compared to TF-1 and TF-2 test data for predicting onset of density wave oscillation.
- DWO frequency much lower than natural frequencies of SG components. No resonance/FIV concerns.
- SG stability analysis shows limited flow oscillations due to DWO at low power operating conditions.
- Design objective for SG inlet flow restrictor (IFR) is to produce necessary pressure drop in the tube subcooled region to limit oscillation magnitudes to acceptable limits.
- Revision to FSAR Section 5.4.1.2 to clarify terminology; the design intent of the IFR is not to fully preclude DWO, but to ensure oscillations are within acceptable limits.
- DWO stability map based on TF-1/TF-2 data is under development for comparison with stability map developed by NRC.

SG DWO – ITAAC Closure Activities

- The SG tubes, tube-to-tubesheet welds, and the tubesheet are ASME Class 1 components evaluated to the rules of Subsection NB. These components will be included in an ASME Design Report that will be reviewed by the NRC. These actions are being tracked as part of ITAAC closure.
- SG FW plenum analysis is under NRC review.
  - The tubes and tube-to-tubesheet welds were not included and DWO loading was not considered, however maximum resultant usage in the vicinity of the tube-to-tubesheet welds is very low.
- SG tube and tube-to-tubesheet weld fatigue analyses will be completed; will consider all transient and DWO loading, and will confirm if IFR sizing is adequate.

Acronyms

AR	Acoustic Resonance	MCS	Module Control System
BC	Boundary Condition	NDRC	National Defense Research Council
CNT	Containment	RMS	Root Mean Square
CRB	Control Building	RXB	Reactor Building
CVAP	Comprehensive Vibration Assessment Program	SG	Steam Generator
DCR	Demand to Capacity Ratio	SSC	Structures, Systems, and Components
DWO	Density Wave Oscillation	TGB	Turbine Generator Building
FEI	Fluid Elastic Instability	VS	Vortex Shedding
FIV	Flow Induced Vibration		
FW	Feedwater		
FEM	Finite Element Model		
IFR	Inlet Flow Restrictor		
ITAAC	Inspections, Tests, Analysis, and Acceptance Criteria		

Portland Office
6650 SW Redwood Lane,
Suite 210
Portland, OR 97224
_971.371.1592_

Corvallis Office
1100 NE Circle Blvd., Suite 200
Corvallis, OR 97330
_541.360.0500_

Rockville Office
11333 Woodglen Ave., Suite 205
Rockville, MD 20852
_301.770.0472_

Charlotte Office
2815 Coliseum Centre Drive,
Suite 230
Charlotte, NC 28217
_980.349.4804_

Richland Office
1933 Jadwin Ave., Suite 130
Richland, WA 99354
_541.360.0500_

Arlington Office
2300 Clarendon Blvd., Suite 1110
Arlington, VA 22201

London Office
1st Floor Portland House
Bressenden Place
London SW1E 5BH
United Kingdom
_+44 (0) 2079 321700_

http://www.nuscalepower.com
_Twitter: @NuScale_Power_

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ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Full Committee Presentation: NuScale FSAR Chapter 15 – Transient and Accident Analyses," PM-0719-66249, Revision 0

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If you have any questions, please contact Matthew Presson at 541-452-7531 or at mpresson@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Robert Taylor, NRC, OWFN-7H4
Michael Snodderly, NRC, TWFN-2E26
Samuel Lee, NRC, OWFN-8H12
Gregory Cranston, NRC, OWFN-8H12
Rani Franovich, NRC, OWFN-8H12

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ACRS Full Committee Presentation

NuScale FSAR

Chapter 15

Transient and Accident Analyses

July 10, 2019

PM-0719-66249
Revision: 0

Presentation Team

Dr. Pravin Sawant (*)	- Supervisor, Code Validation and Methods
Dr. Brian Wolf (*)	- Supervisor, Code Development
Dr. Selim Kuran (*)	- Thermal Hydraulic Software Validation
Meghan McCloskey (*)	- Supervisor, System Thermal Hydraulics
Mark Shaver (*)	- Supervisor, Radiological Engineering
Matthew Presson	- Licensing Project Manager

(*) On the phone

Scope of Chapter 15

Evaluation of the safety of a nuclear power plant requires analyses of the plant's responses to postulated equipment failures or malfunctions

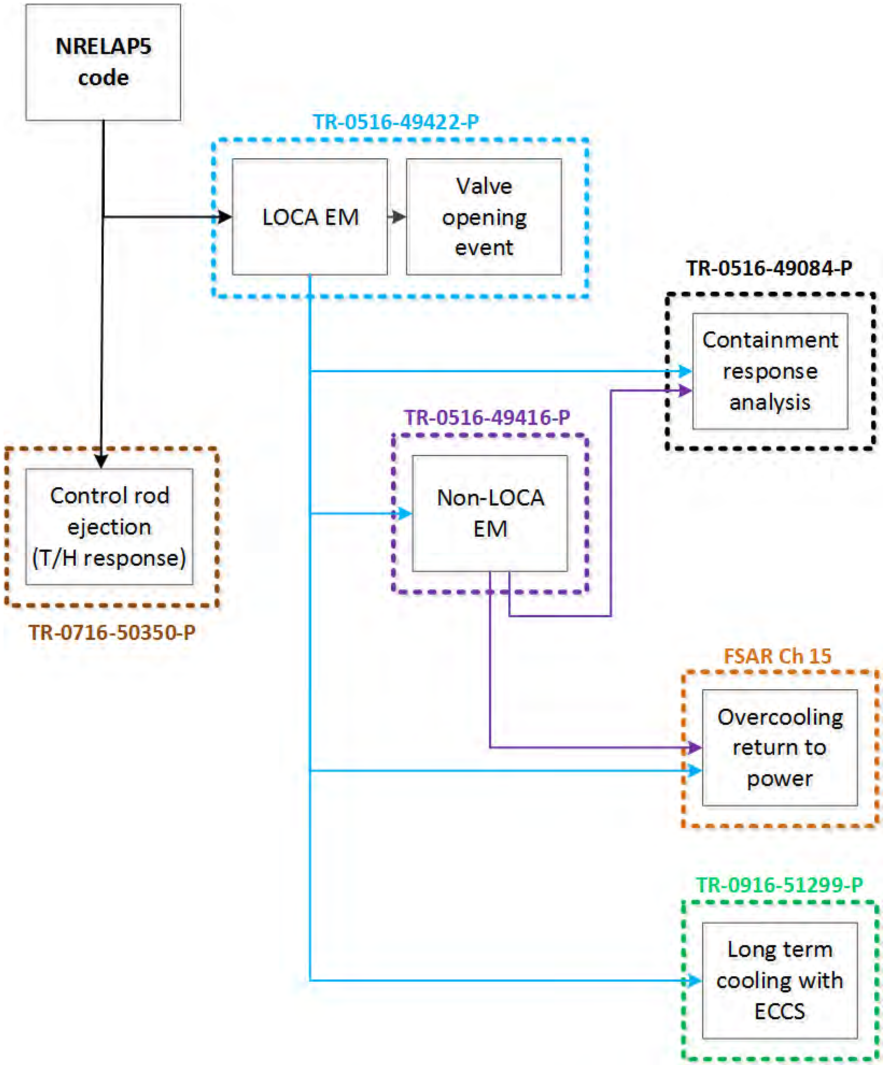
- FSAR Ch 15 addresses deterministic design basis safety analyses
- FSAR Ch 15 summarizes results of the NuScale design basis events identified, event classification, methodology for analysis, event results and margin to acceptance criteria
- FSAR Ch 15 provides results demonstrating radiological dose consequences remain below acceptance criteria

Chapter 15 Overview Agenda – July 10th

- NuScale design overview
- Ch 15 overview and analytical assumptions
- Limiting case results
- Radiological analysis
- Ch 6.2.1 Containment response analysis limiting results
- Long term cooling

System T/H Analysis Basis

- NRELAP5 code developed from RELAP5-3D
  - Modified to address NuScale-specific phenomena/systems
- LOCA Evaluation Model (EM) developed following RG 1.203 EMDAP
  - LOCA EM extended to derive EMs for other events as shown in this figure.
- Additional supporting EMs include
  - Nuclear Analysis Codes – TR-0716-50350-P-A
  - Critical Heat Flux – TR-0116-21012-P-A
  - Subchannel Analysis – TR-0915-17564-P-A
- Additionally, the NuScale design implements an exemption to GDC 27 to account for its passive assumption - No operator actions are required to achieve safety functions for 72 hours under Chapter 15 postulated conditions.
  - In implementing this exemption, the NuScale DCA assures fuel cladding integrity for all design basis events, including under postulated accident conditions.



Chapter 15 Radiological Dose Consequences

- Doses remain below applicable limits for all events
- Methods for determining dose consequences consistent with RG 1.183
- Design Basis Accident Radiological Consequences – TR-0915-17565
- Core Damage Event (CDE) Radiological Consequences – TR-0915-17565

EAB: Exclusion Area Boundary

LPZ: Low Population Zone

CR: Control Room

Event	Location	Acceptance Criteria (rem TEDE)	Dose (rem TEDE)
Iodine Spike Design Basis Source Term(1) (pre-incident iodine spike)	EAB	25.0	<0.01
	LPZ	25.0	<0.01
	CR	5.0	<0.01
Iodine Spike Design-Basis Source Term(1) (coincident iodine spike)	EAB	25.0	<0.01
	LPZ	25.0	<0.01
	CR	5.0	<0.01
Core Damage Event(2)	EAB	25.0	**0.63**
	LPZ	25.0	**1.37**
	CR	5.0	**2.14**
Main Steam Line Break (pre-incident iodine spike)	EAB	25.0	<0.01
	LPZ	25.0	<0.01
	CR	5.0	0.01
Main Steam Line Break (coincident iodine spike)	EAB	2.5	<0.01
	LPZ	2.5	<0.01
	CR	5.0	<0.01
Steam generator tube failure (pre-incident iodine spike)	EAB	25.0	0.08
	LPZ	25.0	0.08
	CR	5.0	0.20
Steam generator tube failure (coincident iodine spike)	EAB	2.5	<0.01
	LPZ	2.5	<0.01
	CR	5.0	<0.01
Primary coolant line break	EAB	6.3	0.02
	LPZ	6.3	0.04
	CR	5.0	0.08
Fuel handling accident	EAB	6.3	**0.55**
	LPZ	6.3	**0.55**
	CR	5.0	**0.89**

(1) The iodine spike DBST is not an event, rather it serves as a bounding surrogate for design-basis loss of primary coolant into containment events.

(2) The CDE is a beyond-design-basis special event.

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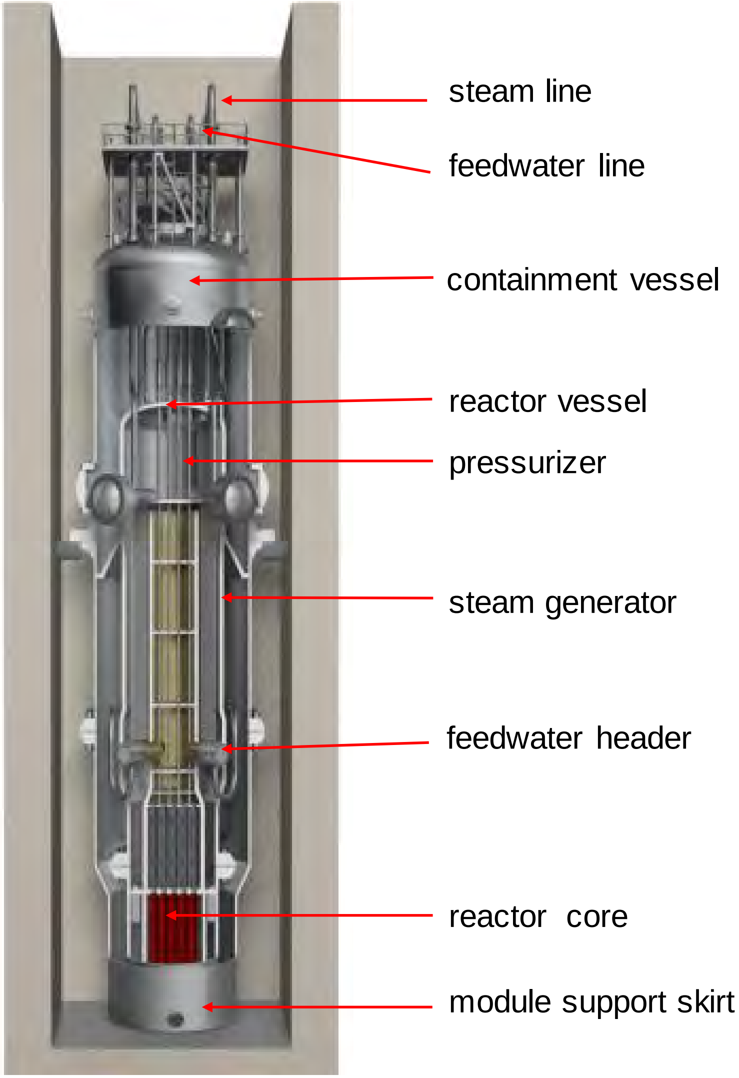
Power Module Overview

Integral Pressurized Water Reactor

- Core, steam generator and pressurize in one vessel
- Integrated reactor design, no large-break loss-of-coolant accidents
- Reactor coolant system operated in single phase (liquid) density driven flow
- Safety decay heat removal systems are passive and fail safe
- Module protection system designed to automate event mitigation actuations (no operator actions)



Module Protection System Functions

- Reactivity Control
  – Reactor trip
  – CVCS/Demineralized Water Isolation
- RCS and Secondary Inventory Control
  – Containment Isolation
  – Secondary Isolation
- Heat Removal
  – DHRS Actuation
  – ECCS Actuation
- Subcooling
  – Reactor trip

Event Mitigation

Increase in heat removal transients

- Reactor trip
- Secondary Isolation

Decrease in heat removal transients

- Reactor trip
- DHRS Actuation

Reactivity and power transients

- Reactor trip
- Demineralized water isolation

Increase in RCS inventory transients

- Reactor trip
- CVCS Isolation

Decrease in RCS inventory transients

- Reactor trip
- CNV Isolation
- ECCS actuation

Stability

- Reactor trip

NPM Design Basis Event Identification

- Ch 15 initiating events considered for internal events in a single NPM while at power
- Systematic process used to evaluate NPM systems to assure appropriate scope of design basis events identified
- Design basis events categories consistent with operating LWRs
  - Increase in heat removal, decrease in heat removal, reactivity and power distribution anomalies, increase in RCS inventory, decrease in RCS inventory, radioactive release from a subsystem or component
- NuScale-specific phenomena/event progression:
  - PWR stability
  - Return to power

NuScale Unique Ch 15 Events

Sec.	Event	Classification	System T/H (code)	Subchannel (code)	Radiological
15.1.6	Loss of containment vacuum/containment flooding	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a
15.2.9	Inadvertent operation of the decay heat removal system	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a
15.6.6	Inadvertent operation of emergency core cooling system	AOO	Valve opening event (NRELAP5)	n/a	Yes (1)

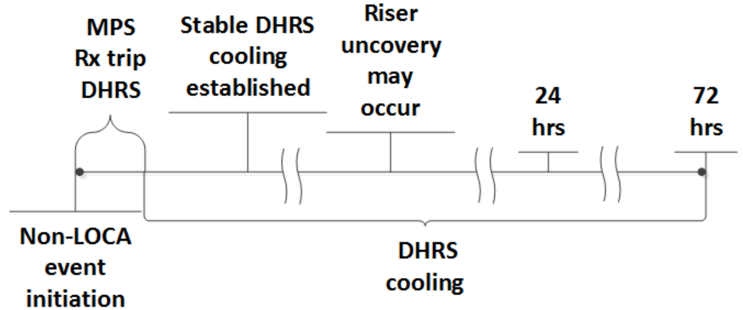
(1) See FSAR Ch 12

Analytical Assumptions for Ch 15 Analysis

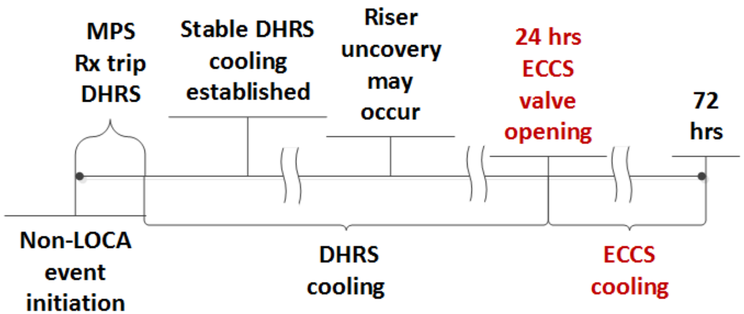
- **Operator action**
  - No operator actions required to achieve safety functions for 72 hours after an initiating event occurs
- **Single failure**
  - Worst single failure of safety-related component assumed
- **Loss of power**
  - Chapter 15 event analyses consider availability of AC power and DC power
- **Scope of event progression**
  - Safety analyses of design basis events are performed from event initiation until a safe, stabilized condition is reached
  - After safe, stabilized condition is reached:
    - ECCS long term decay and residual heat removal
    - Return to power
    - Extended DHRS operation

Loss of Power – Non-LOCA Event

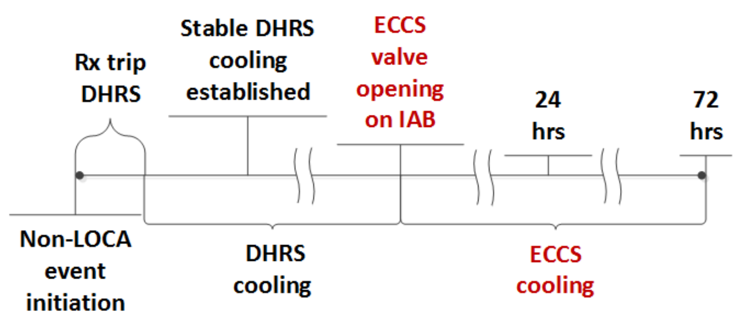
AC power available, DC power available



AC power unavailable, DC power available

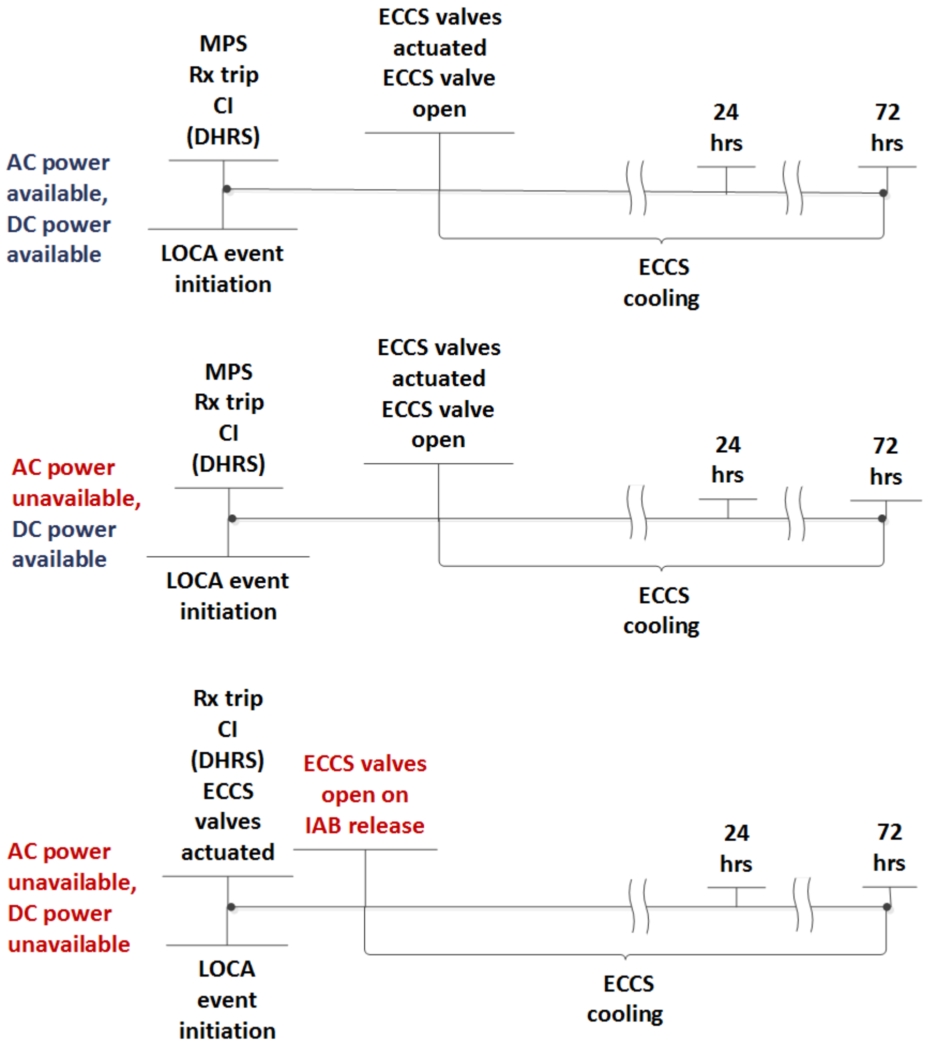


AC power unavailable, DC power unavailable



- Availability of AC, DC power affects whether ECCS valves actuate, and what time they open

Loss of Power –LOCA Event



- Availability of DC power affects whether ECCS valves open on level actuation or on IAB release
- Break size and credit for DHRS operation also affect whether ECCS valves open on level or on IAB release

PM-0719-66249
Revision: 0

Limiting Transient Results

Parameter	Event	Acceptance Criterion	Limiting Result
Maximum RCS Pressure	Several	< 2315 psia (110% P_{design}) or < 2520 psia (120% P_{design})	~ 2170 psia
Maximum SG Pressure	Inadvertent Operation DHRS	< 2315 psia (110% P_{design})	1582 psia
	SG tube failure	< 2520 psia (120% P_{design})	1806 psia
MCHFR	Single rod withdrawal	> 1.284	1.614
	Inadvertent Opening RRV	> 1.13	1.41
	LOCA	> 1.29	1.796
Level above top of core	LOCA	> 0 ft	1.5 ft

Chapter 6.2.1
Containment Response Analysis

July 10, 2019

PM-0719-66249
Revision: 0

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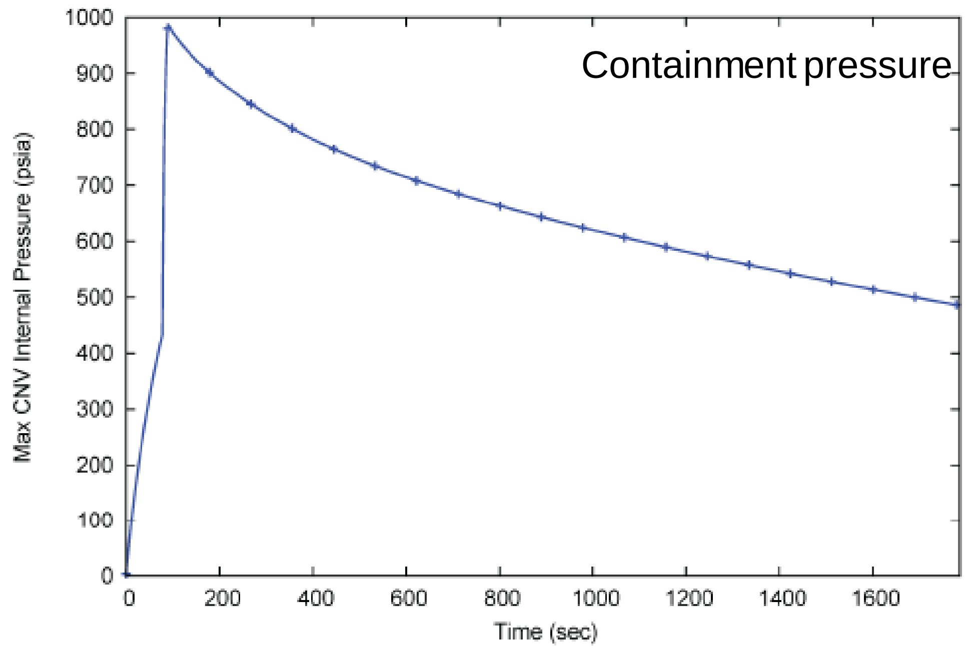
6.2.1 CNV Response Analysis

- CNV designed to withstand full spectrum of primary and secondary system M&E releases with worst case single active failure and loss of power
- CNV response analysis methodology based on NRELAP5 described in TR-0516-49084 Rev 0
  - NRELAP5 used to model integrated response of the RPV blowdown and CNV pressurization/heat removal
  - Containment response analysis based on models from LOCA EM or non-LOCA EM, with initial and boundary condition changes necessary to conservatively bias the mass and energy release and maximize the CNV pressure and temperature response
- Limiting results:
  - Peak pressure: 986 psia < 1050 psia design pressure
  - Peak wall temperature: 526 °F < 550 °F design temperature

Containment Response Limiting Pressure Case

Time (sec)	Event
0	Inadvertent RRV opening Loss of normal AC and DC power FW/MS isolation Reactor trip
0.4	High CNV pressure – CNV isolation
74	ECCS valve opening on IAB release - *Single failure of 1 RRV to open*
91	Peak CNV pressure **986 psia < 1050 psia design pressure**
~1800	CNV pressure < 50% peak pressure



Chapter 15
Long Term Cooling

July 10, 2019

PM-0719-66249
Revision: 0

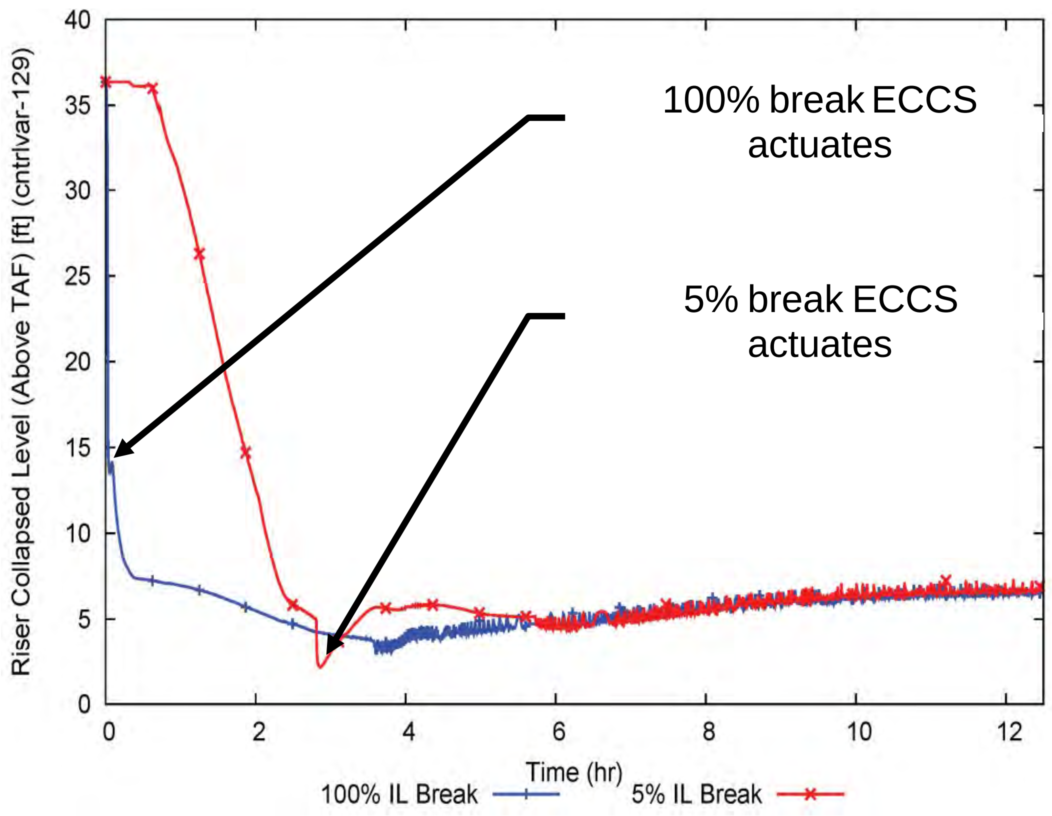
Template #: 0000-21727-F01 R5

Long Term Cooling Analysis

- Events considered:
  - LOCA events transition to ECCS cooling; full break spectrum evaluated
  - Non-LOCA events transition from DHRS to ECCS cooling
    - Events considered are SGTF, loss of FW flow, and a generic evaluation of DHRS cooldown
- Biasing approach and results:
  - **Minimum level:** All minimum level cases showed the core remained covered at all times during the LTC phase, and that boron precipitation is precluded during the time of minimum level when boron concentration is maximized
  - **Minimum temperature:** All minimum temperature cases showed margin to boron precipitation
  - **Maximum temperature:** All maximum temperature cases showed decreasing cladding temperature, with final clad temperature remaining well below operating temperature at full power

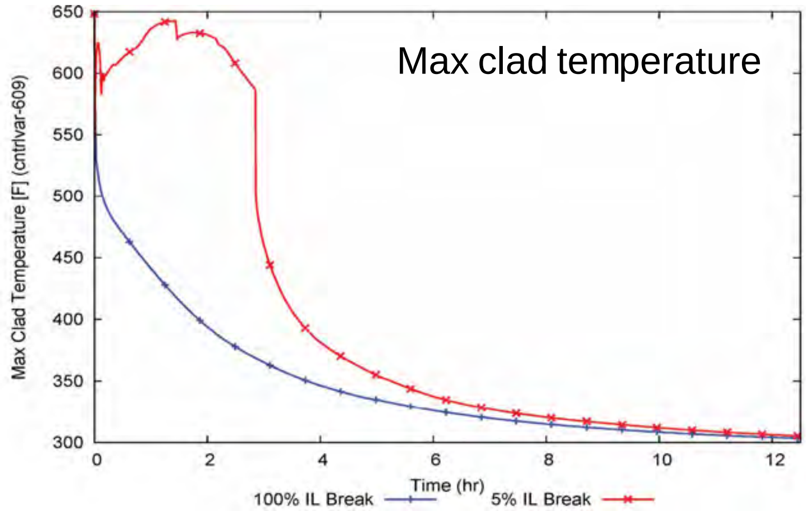
LTC Results – Injection Line Min Level

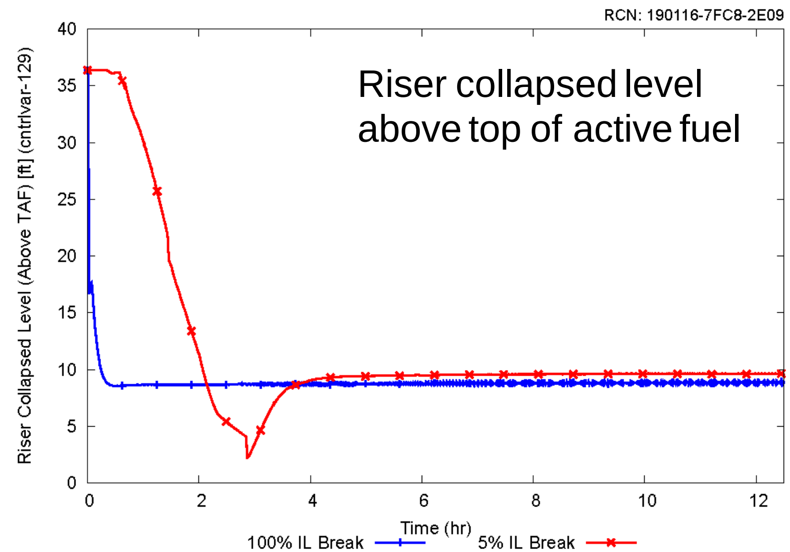
- Minimum level case maximizes RPV energy and minimizes CNV energy
- Higher pressure difference between the RPV and CNV increases coolant accumulation in the CNV which reduces RCS level
- Minimum level during LTC phase occurs 3-5 hours after ECCS actuation
- Long term, level recovers towards equilibrium as the pressure difference between the RPV and CNV is reduced with decreasing decay heat



LTC Results – Injection Line Max Temp

- Maximum temperature case maximizes initial RCS energy and decay heat while minimizing heat transfer to the reactor pool
- Results demonstrate clad temperature follows a decreasing long term trend
- Long-term level remains at equilibrium condition





Chapter 15.0.6
Return to Power

July 10, 2019

PM-0719-66249
Revision: 0

Principle Design Criteria 27

- Principle Design Criteria 27→ Passive Reactor GDC-27 equivalent

  The reactivity control systems shall be designed to have a combined capability of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

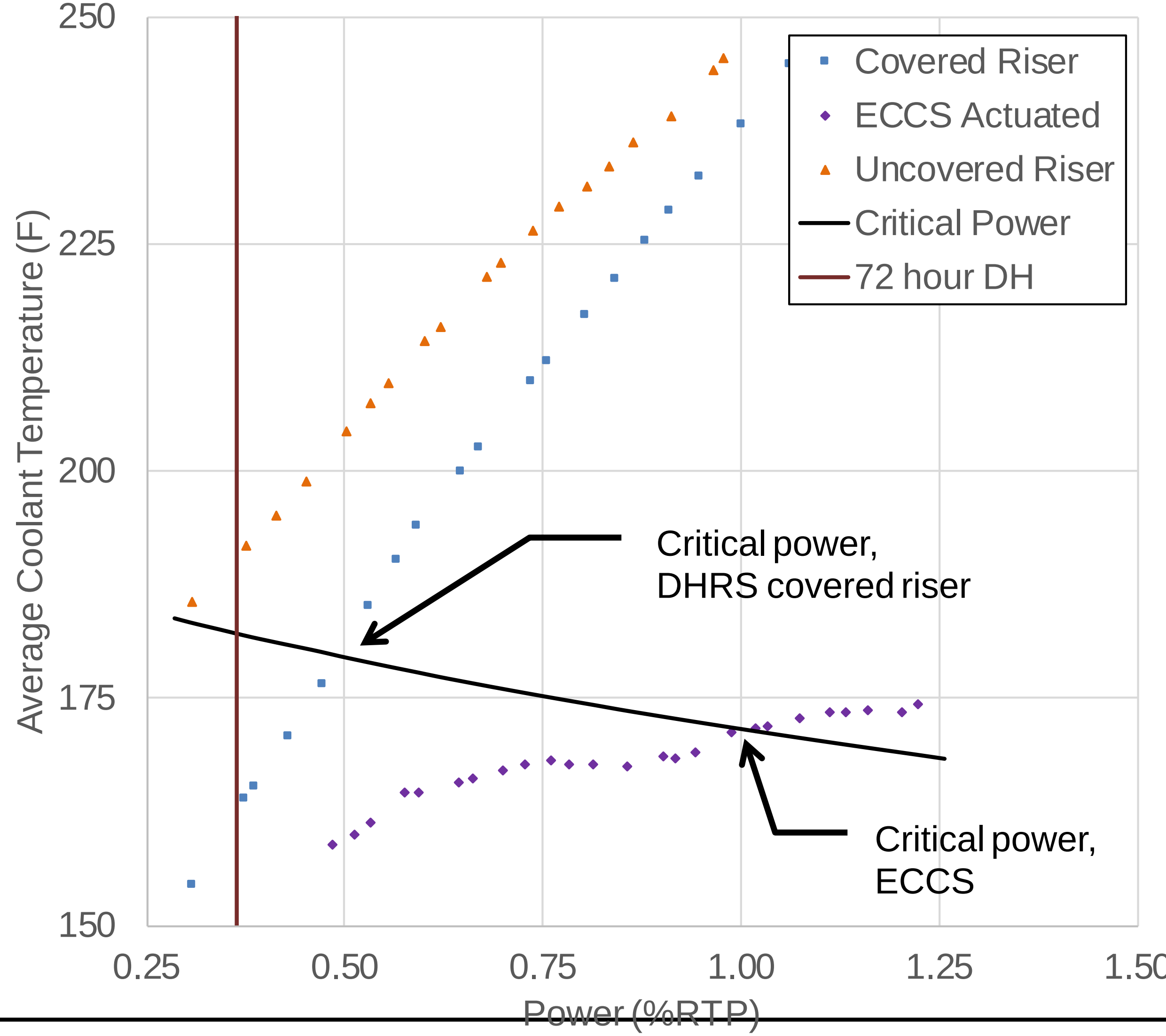
  Following a postulated accident, the control rods shall be capable of holding the reactor core subcritical under cold conditions, without margin for stuck rods.

- Compliance with PDC-27:
  - Immediate shutdown margin is sufficient to protect RCPB and SAFDLs with appropriate margin for the worst rod stuck out of the core
  - Cold shutdown is achieved with all control rods fully inserted
  - Loss of shutdown margin consequences
    - Worst case OCRP not challenging to SAFDLs
    - Critical power level does not challenge DHRS or ECCS heat removal capabilities
  - Probability of the combination of conditions that results in a loss of shutdown return to power with a single rod stuck out of the core is small

Return to Power Characterization

- Characterize the steady state RCS temperature a function of available decay heat for both DHRS and ECCS heat removal.
- Calculate the worst rod stuck out critical power level as a function of RCS temperature.
- Conclusions:
  – Loss of shutdown margin may occur around 40hrs with zero Boron in RCS after Xenon decay
  – RCS temperature low (< 200F based on best estimate SIMULATE results)
  – ECCS cooling is more limiting
  – Higher pool temperature can preclude worst rod stuck out criticality (140F based on best estimate SIMULATE results)
  – Simple pool boiling CHF analysis demonstrates large margin

Return to Power Characterization



Acronyms

- AOO – Anticipated Operational Occurrences
- ASME – American Society of Mechanical Engineers
- ASTM – American Society for Testing and Materials
- BPVC – Boiler Pressure Vessel Code
- CES – Containment Evacuation System
- Ch – Chapter
- CIV – Containment Isolation Valve
- CNV – Containment Vessel
- COL – Combined License
- CRDM – Control Rod Drive Mechanism
- CVCS – Chemical and Volume Control System
- DBST – Design Basis Source Term
- DHRS – Decay Heat Removal System
- ECCS – Emergency Core Cooling System
- EFPY – Effective Full Power Years+D51
- EM – Evaluation Methodology
- EPRI – Electric Power Research Institute
- FIV – Flow Induced Vibration
- FSAR – Final Safety Analysis Report
- ft – feet
- FW – Feedwater
- FWIV – Feedwater Isolation Valve
- GDC – General Design Criteria
- HZP – Hot Zero Power
- IAB – Inavertant Actuation Block
- ISI – Inservice Inspection
- LOCA – Loss of Coolant Accident
- LTC – Long Term Cooling

Acronyms

- **LTOP – Low Temperature Overpressure Protection**
- **LWR – Light Water Reactor**
- **M&E – Mass and Energy**
- **MCHFR – Minimum Critical Heat Flux Ratio**
- **MPS – Module Protection System**
- **MSIV – Main Steam Isolation Valve**
- **MSS – Main Steam System**
- **MWt – Megawatts thermal**
- **NDE – Non–destructive Examination**
- **NEI – Nuclear Energy Institute**
- **NPM – NuScale Power Module**
- **NPS – Nominal Pipe Size**
- **NRELAP – System Thermal Hydraulic Code**
- **OCRP – Overcooling Return to Power**
- **OD – Outside Diameter**
- **OE – Operations Experience**
- **oF – degrees Fahrenheit**
- **PDC – Priciple Design Criteria**
- **psia – pounds per square inch absolute**
- **P-T – Pressure and Temperature**
- **PTS – Pressurized Thermal Shock**
- **PWR – Pressurized Water Reactor**
- **PWSCC – Primary Water Stress–Corrosion Cracking**
- **PZR – Pressurizer**
- **RCCWS – Reactor Component Cooling Water System**
- **RCPB – Reactor Coolant Pressure Boundary**
- **RCS – Reactor Coolant System**

Acronyms

- RG – Regulatory Guide
- RPV – Reactor Pressure Vessel
- RRV – Reactor Recirculation Valve
- RSV – Reactor Safety Valve
- RTNDT – Reference Temperature for Nil–ductility Transition
- RVV – Reactor Vent Valve
- Rx – Reactor
- SAFDLs – Specified Acceptable Fuel Design Limits
- SDM – Shutdown Margin
- SG – Steam Generator
- SGTF – Steam Generator Tube Failure
- T/H – Thermal Hydraulic
- TRV – Thermal Relief Valve
- TS – Technical Specifications
- TT – Thermally Treated
- UHS – Ultimate Heat Sink
- USE – Upper Shelf Energy
- VIPER – Subchannel Code

LO-0719-66254

July 9, 2019 Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Full Committee Presentation: NuScale FSAR Chapter 20, Mitigation of Beyond-Design-Basis Events," PM-0719-66253, 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) NuScale Full Committee Meeting on July 10, 2019. The materials support NuScale's presentation of Chapter 20, "Mitigation of Beyond-Design-Basis Events," of the NuScale Design Certification Application.

The enclosure to this letter is the nonproprietary version of the presentation entitled "ACRS Full Committee Presentation: NuScale FSAR Chapter 20, Mitigation of Beyond-Design-Basis Events," PM-0719-66253, Revision 0.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Nadja Joergensen at 541-452-7338 or at njoergensen@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Robert Taylor, NRC, OWFN-7H4
Michael Snodderly, NRC, TWFN-2E26
Samuel Lee, NRC, OWFN-8H12
Gregory Cranston, NRC, OWFN-8H12
Omid Tabatabai, NRC, OWFN-8H12

Enclosure: ACRS Full Committee Presentation: NuScale FSAR Chapter 20, Mitigation of Beyond-Design-Basis Events, PM-0719-66253, Revision 0

Enclosure:

ACRS Full Committee Presentation: NuScale FSAR Chapter 20, Mitigation of Beyond-Design-Basis Events, PM-0719-66253, Revision 0

ACRS Full Committee Presentation

NuScale FSAR

Chapter 20

Mitigation of Beyond-Design-Basis Events

July 10, 2019

PM-0719-66253
Revision: 0

Presenters

Chris Maxwell
Senior Reactor Operator

Zackary Rad
Director, Regulatory Affairs

Nadja Joergensen
Licensing Specialist

Mitigation Strategies for Beyond-Design-Basis External Events 10 CFR 50.155(b)(1)

Indefinite Coping

- NuScale considers a minimum coping period of 14 days using only installed plant equipment to be sufficient time to establish “alternate means of removing heat.”
  - In the Fukushima Daiichi accident, without pre-planning or a hardened pool makeup connection, and with limited access to off-site resources, personnel began
    - adding water to the Unit 4 SFP with fire and concrete pump trucks after 9 days; and
    - injecting water via the fuel pool cooling system at 14 days.
- Beyond minimum installed equipment coping period, the continued use of installed plant equipment, ad hoc resources, and equipment repairs can be used to continue coping indefinitely.

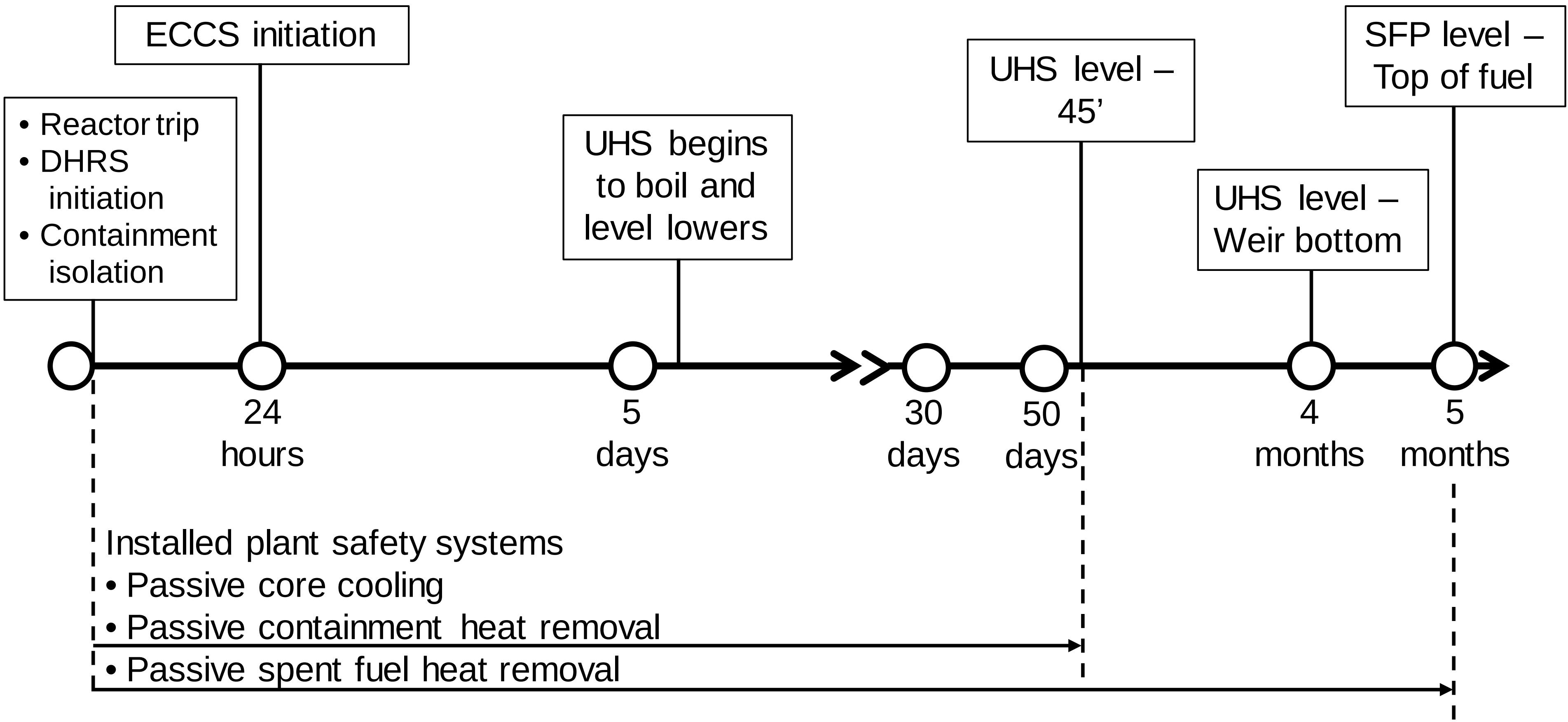
NuScale Coping Capability

NuScale Power Plant Design

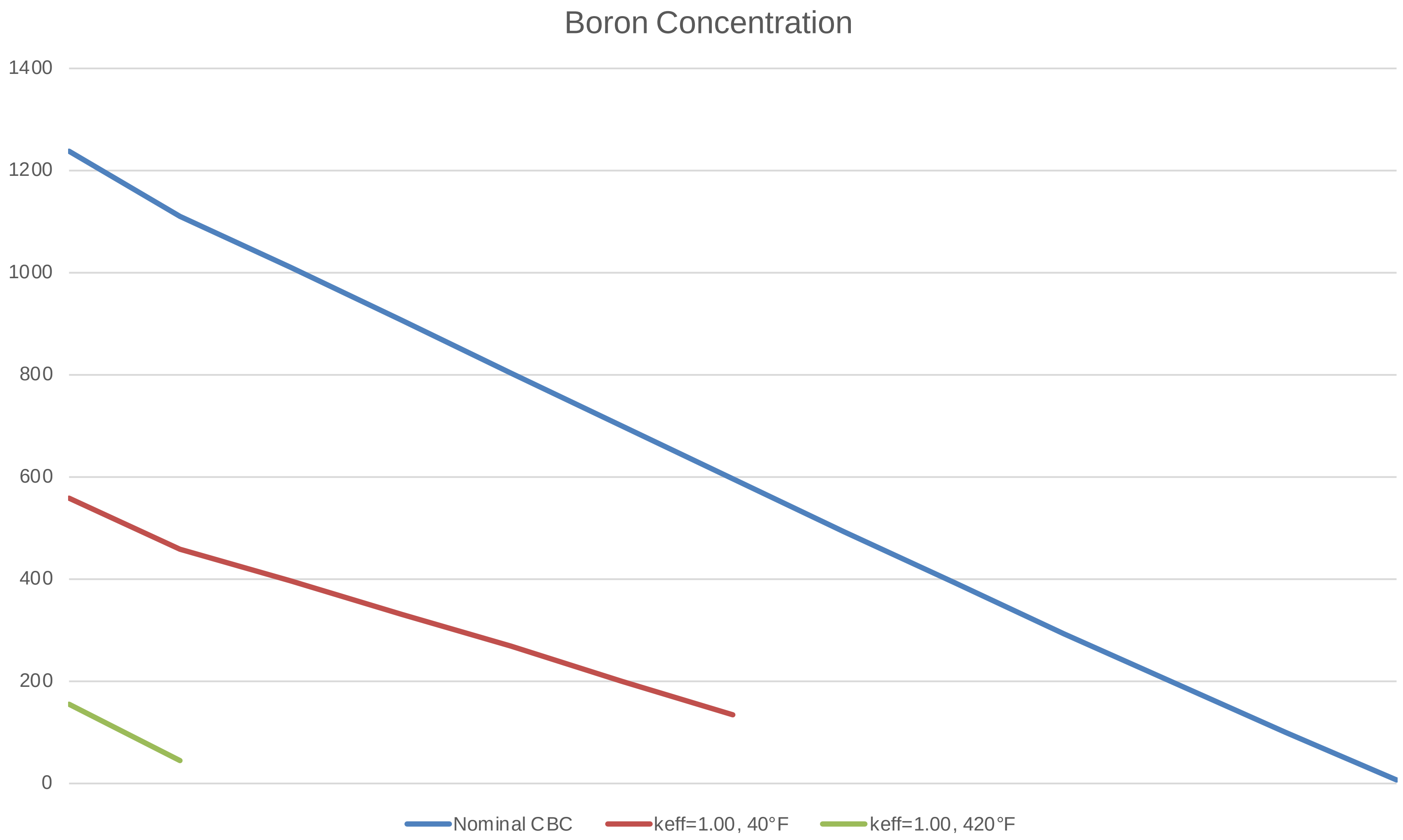
- The NuScale Power Plant was designed to provide coping during an ELAP concurrent with a LUHS <u>without</u>:
  - AC or DC electrical power
  - Inventory addition
  - Supplemental equipment
  - Off-site resources
  - Operator action (monitoring)
- Extended coping duration is provided by the automatic response of installed plant equipment alone.
- This strategy permits plant staff to focus on addressing the initiating event and restoring normal functional capabilities.

NuScale Power Plant Response

Plant response without operator action or off-site resources



Reactivity Control



Spent Fuel Pool Monitoring 10 CFR 50.155(e)

Spent Fuel Pool Level Indication

- The UHS system includes remote level indication for the following:
  - Reactor Pool
  - Refueling Pool
  - Spent Fuel Pool (2)
- Normally powered by the highly reliable DC power system via the plant protection system.
- Include a replaceable battery power source, independent from the plant AC and DC power systems, with a minimum capacity of 14 days.

Conclusion

- NuScale Power Plant Mitigation Strategy
  - Rely on the automatic response of permanently installed, safety-related plant equipment to establish and maintain the three key safety functions and provide extended coping capabilities of greater than 14 days.
- The NuScale Power Plant mitigation strategy does not require:
  - AC or DC electrical power
  - Inventory addition
  - Operator action
- NuScale SFPLI Strategy
  - Installed instrumentation with 14 day battery backup power supply.

Preliminary Safety Evaluation with Open Items: Chapter 20, “Mitigation of Beyond-Design-Basis Events”

NuScale Design Certification Application

ACRS Full Committee Meeting
July 10, 2019

Agenda

- Background and Overview
- Regulatory Framework for the Staff Review
- Staff Review of NuScale’s MBDBE strategy
- Phase 4 Review Plan
- Abbreviations

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Background and Overview

- 10/30/2018: NuScale submitted its DCA, Rev 2
- 01/24/2019: NRC approved Final MBDBE Rule (10 CFR 50.155)
- 03/28/2019: NuScale informed the staff that it was revising its Ch. 20
- 06/10/2019: NuScale submitted revised Ch. 20
- 06/14/2019: NuScale submitted Rev. 1 to ELAP Technical Report
- 06/26/2019: NRC issued Information SECY 19-0066[1]
- 07/12/2019: SECY 19-0066 will be publicly available (ML19148A443)
- Staff's (preliminary) Phase 2 SER for Chapter 20 is based on Rev. 2 of NuScale's DCA
- Based on the docketed information, the Staff's findings in Phase 2 SER are limited to the first 72 hours after initiation of a BDBE.

---

[1] *SECY 19-0066: "Staff Review of NuScale Power's Mitigation Strategy for Beyond-Design-Basis External Events"*

Regulatory Framework for the Staff Review

- The recently approved regulation, 10 CFR 50.155, for mitigation of beyond-design-basis events (MBDBE) does not apply to applicants for a design certification.
- NuScale is voluntarily seeking the NRC's approval of its proposal to use installed design features to mitigate beyond-design-basis external events.
- NuScale design incorporates several design features that provide enhanced capabilities for mitigating an extended loss of ac power compared to currently operating reactors.

Regulatory Framework (Cont'd)

TEXT OF 10 C.F.R. § 50.155(b), (c), and (e) APPROVED BY THE COMMISSION

(b) Strategies and guidelines. Each applicant or licensee shall develop, implement, and maintain:

(1) Mitigation strategies for beyond-design-basis external events. Strategies and guidelines to mitigate beyond-design-basis external events from natural phenomena that are developed assuming a loss of all ac power concurrent with either a loss of normal access to the ultimate heat sink or, for passive reactor designs, a loss of normal access to the normal heat sink. These strategies and guidelines must be capable of being implemented site-wide and must include the following:

(i) Maintaining or restoring core cooling, containment, and spent fuel pool cooling capabilities; and

(ii) The acquisition and use of offsite assistance and resources to support the functions required by paragraph (b)(1)(i) of this section indefinitely, or until sufficient site functional capabilities can be maintained without the need for the mitigation strategies.

. . . .

(c) Equipment. (1) The equipment relied on for the mitigation strategies and guidelines required by paragraph (b)(1) of this section must have sufficient capacity and capability to perform the functions required by paragraph (b)(1) of this section.

(2) The equipment relied on for the mitigation strategies and guidelines required by paragraph (b)(1) of this section must be reasonably protected from the effects of natural phenomena that are equivalent in magnitude to the phenomena assumed for developing the design basis of the facility.

. . . .

(e) Spent fuel pool monitoring. In order to support effective prioritization of event mitigation and recovery actions, each licensee shall provide reliable means to remotely monitor wide-range water level for each spent fuel pool at its site until 5 years have elapsed since all of the fuel within that spent fuel pool was last used in a reactor vessel for power generation.

SRM-M190124A, Enclosure 1, Federal Register Notice at 140-41.

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Staff Review of NuScale MBDBE

- Maintain consistency with scope of review performed for operating reactors and other design certifications.
  - Focus on the initial response coping period (first 72 hours) where the most critical and time-sensitive actions are projected to occur.
- Verify the design capabilities and capacities of the permanently installed SSCs satisfy the required safety functions, including the effects of credible transient phenomena, for 72 hours following initiating event.
  - Review will focus on design aspects of SSCs as described in the FSAR.
  - Operational aspects (e.g., procedures, training) deferred to COL stage.

Staff Review of NuScale MBDBE (Cont'd)

- COL applicant would need to describe how mitigating strategies (or sufficient site functional capabilities) are maintained for an indefinite time period.
  - SSC design aspects would only need to be addressed if there are credible transient phenomena (e.g., return to power) that could challenge core cooling, containment, or SFP cooling beyond 72 hours
  - Level of detail expected is commensurate with time available to implement actions.
- Staff plans to document that instrumentation is not relied on to support the mitigation strategies; however:
  - Instrumentation is expected to be available for the initial 72 hours,
  - Provides additional assurance that systems have responded as designed.

Phase 4 Review

- The staff will follow its plans, as described in SECY 19-0066, to complete its review of NuScale’s Ch. 20 in Phase 4 of the DCA review
- The Staff will evaluate NuScale’s DCA, Rev. 3, information for compliance with the requirements of 10 CFR 50.155
- NuScale DCA, Rev. 3, is expected to be submitted in late August 2019

Abbreviations

ACRS	Advisory Committee on Reactor Safeguards
CFR	Code of Federal Regulations
CNV	Containment Vessel
COL	Combined License
BDBA	Beyond-Design-Basis Accident
CNV	Containment Vessel
DCA	Design Certification Application
DHRS	Decay Heat Removal System
ECCS	Emergency Core Cooling System
EDSS	Highly Reliable Electrical System
ELAP	Extended Loss of AC Power
FSAR	Final Safety Analysis Report
JLD	NRC Japan Lessons Learned Directorate
MBDBE	Mitigation of Beyond-Design-Basis Events
MPS	Module Protection System
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	NRC Office of Nuclear Reactor Regulation
NSIR	NRC Office of Nuclear Security and Incident Response
NRO	NRC Office of New Reactors
SER	Safety Evaluation Report
SRM	Staff Requirement Memorandum
UHS	Ultimate Heat Sink

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Safety Evaluation with Open Items: Chapter 6, “Engineered Safety Features”

NuScale Design Certification Application

ACRS Full Committee Meeting
July 10, 2019

Agenda

- Overview of the NRC Staff’s Presentation
- Exemption Requests Contained in FSAR Section 6.2, “Containment Systems”
- Abbreviations

Overview of the NRC Staff’s Presentation

- The staff briefed the ACRS Subcommittee On June 18, 2019
- Members asked the staff to cover the following topics at Full Committee:

1. Section 6.3, “Emergency Core Cooling System,” with emphasis on ECCS valves and their performance out to 72 hours
   - ➢ *Staff will cover as part of its Ch 15. briefing*

2. ECCS valves description, test plan, and results
   - ➢ *Staff will cover as part of Ch. 3, and 15, briefings, as needed*

3. 6.2.1 and 6.2.2 with emphasis on results of limiting case
   - ➢ *NuScale will address as part of its presentation*

4. Exemption requests associated with Chapter 6
   - ➢ *Staff will address in the following slides*

5. Containment level measurement and the challenge that containment is dry during normal operation
   - ➢ *NuScale will address as part of its presentation*

NuScale Exemption Requests Regarding Containment Isolation Systems

(Section 6.2.4 of the SER)

Clint Ashley

Reactor Systems Engineer, NRO

Exemption Requests

- NuScale’s DCA (Part 7) contains exemption requests associated with the following containment isolation requirements:
  - GDC 55
  - GDC 56
  - GDC 57
  - 10 CFR 50.34(f)(2)(xiv)(E)

10 CFR 50.12 - Specific Exemptions

- Pursuant to 10 CFR 50.12 in part, the Commission may grant exemptions when special circumstances are present.
- According to 10 CFR 50.12(a)(2)(ii), special circumstances are present whenever, “application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.”

Underlying Purpose

- The underlying purpose of GDCs 55, 56, and 57 is to provide containment isolation capability that supports the safety function of containment to provide a barrier to the uncontrolled release of radioactivity to the environment.
- The underlying purpose of 50.34(f)(2)(xiv)(E) is to limit radiological releases by ensuring containment isolation for systems that provide paths to the environs.

Staff's Review

- For GDC 55, 56 and 57, the staff finds that the NuScale design accomplishes the containment safety function by providing a containment isolation capability comparable to that required by the GDCs and therefore, the underlying purpose of the GDCs is met.
- The staff find's NuScale's exemption request meets the requirements for an exemption as defined in 10 CFR 50.12.

Staff's Review

- For 50.34(f)(2)(xiv)(E), the staff finds that the NuScale design isolates systems that provide a path to the environs before core damage or degradation occurs, preventing significant releases from the containment, and therefore, the underlying purpose of the rule is met.
- The staff find's NuScale's exemption request meets the requirements for an exemption as defined in 10 CFR 50.12.

NuScale Exemption Request Regarding “Combustible Gas Control in Containment”

(SER Section 6.2.5)

Anne-Marie Grady

Reactor Systems Engineer, NRO

Regulatory Basis

- 10 CFR 50.44(c)(2) – requires that the plant accommodate hydrogen generation up to 100 percent fuel clad-coolant reaction while limiting containment hydrogen to less than 10 percent, and maintain containment structural integrity and other accident mitigation features.

- NuScale has requested an exemption from 50.44(c)(2), which would require the design provide a hydrogen control system to limit hydrogen concentrations below 10 percent.

- Pursuant to 10 CFR 50.12 in part, the Commission may grant exemptions when special circumstances are present.

➢ *Special circumstances are present (10 CFR 50.12(a)(2)(ii))* in that application of the regulation is not necessary to achieve the underlying purpose of the rule

➢ *Special circumstances are present (10 CFR 50.12(a)(2)(iii))* in that compliance would result in undue hardship or other costs that are significantly in excess of those incurred by others similarly situated.

Exemption Request

- Staff review focused on:
  - Hydrogen conditions in the CNV during the postulated accident
  - Demonstration of adequate containment mixing
  - Equipment survivability following postulated hydrogen combustion
- NuScale analyses demonstrated the containment would survive a bounding combustion event inside the containment at 72 hours
- Staff confirms the calculation for a limiting pressure pulse inside the containment resulting from a hydrogen combustion event at 72 hours for a very short duration, and containment integrity would be maintained.
- Providing a system to control the hydrogen concentration is not necessary to serve the underlying purpose of 10 CFR 50.44(c)(2), which is to prevent a loss of containment integrity
- Staff recommends granting the Exemption Request

NuScale Exemption Request Regarding "Containment Leakage Testing" (SER Section 6.2.6)

Anne-Marie Grady
Reactor Systems Engineer, NRO

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Exemption Request Regarding Containment Leakage Testing

- NuScale requested an exemption from the following regulations:
- 10 CFR 50, App. A, GDC 52, Capability for Containment Leakage Rate Testing, requires the capability to perform containment periodic integrated leakage rate testing (ILRT) (Type A) at containment design pressure.
- 10 CFR 50, App. J—Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, requires Type A tests preoperationally and periodically. thereafter.
- NuScale asserts that the CNV ASME design, analysis for leak tightness, 100% vessel inspectability, and pre service design pressure hydrostatic leakage test would satisfy the underlying goal of demonstrating CNV allowable leakage.

Exemption Request Regarding Containment Leakage Testing (Cont'd)

- NuScale CNV is an ASME Section III, Class 1 pressure vessel
- CNV leakage analysis, design specifications, capability for 100% vessel inspection, examination, and testing, provide assurance that the leakage integrity of the CNV is maintained.
- NuScale analyzed the CNV bolt design for the flanged openings, using ANSYS, based on the seal design and specification.
- NuScale calculated flange contact pressures and corresponding flange gaps, based on CNV internal accident pressure and temperatures.
- Staff reviewed these calculations during an audit.

Exemption Request Regarding Containment Leakage Testing (Cont'd)

- NuScale asserts that the CNV design combined with Types B and C test results are sufficiently representative of accident conditions to demonstrate that the TS leak rate, La, would not be exceeded.
- In addition to testing required by ASME, NuScale proposes a preservice design pressure test to confirm the expected performance of the CNV design. This would be verified by ITAAC.
- The CNV ASME design, analysis for leak tightness, 100% vessel inspectability, and pre service design pressure hydrostatic leakage test would satisfy the underlying goal of demonstrating CNV allowable leakage
- The staff recommends approving the exemption request to not require Type A testing, nor to require design capability for ILRT.

Abbreviations

ACRS	Advisory Committee on Reactor Safeguards
ASME	American Society of Mechanical Engineers
CES	Containment Evacuation System
CFDS	Containment Flooding and Drain System
CFR	Code of Federal Regulations
CNV	Containment Vessel
CRHS	Control Room Habitability System
CRVS	Control Room Ventilation System
COL	Combined License
DBA	Design-Basis Accident
DCA	Design Certification Application
DSRS	Design Specific Review Standard
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GL	Generic Letter
ILRT	Integrated Leakage Rate Testing
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
MSS	Main Steam System
NRC	Nuclear Regulatory Commission
NRO	NRC Office of New Reactors
PDC	Principal Design Criteria
RG	Regulatory Guide
SER	Safety Evaluation Report
SSC	Structures, Systems, and Components
TEDE	Total Effective Dose Equivalent

Presentation to the ACRS Full Committee

NuScale Design Certification Application Review

Safety Evaluation Report

Chapter 3: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Project Manager: Marieliz Vera

July 10, 2019

Staff Review Team

- **Technical Staff**
  - **Renee Li**, Senior Mechanical Engineer
  - **Tuan Le,** Mechanical Engineer
  - **Yiu Law,** Mechanical Engineer
  - **Alexander Tsirigotis,** Mechanical Engineering
  - **Yuken Wong,** Senior Mechanical Engineer
  - **Jason Huang,** Mechanical Engineer
  - **Nick Hansing,** Mechanical Engineer
  - **Michael Breach,** Mechanical Engineer
  - **Kaihwa Hsu,** Senior Mechanical Engineer
  - **Tom Scarbrough,** Senior Mechanical Engineer
  - **Jorge Cintron- Rivera,** Electrical Engineer
  - **Sheila Ray,** Senior Electrical Engineer
  - **Vaughn Thomas**, Structural Engineer
  - **Robert Roche-Rivera,** Structural Engineer
  - **Sunwoo Park,** Structural Engineer
  - **Pravin Patel,** Structural Engineer
  - **Ata Istar,** Structural Engineer
  - **George Wang,** Structural Engineer
  - **Maryam Khan,** Structural Engineer
  - **Manas Chakravorty,** Senior Structural Engineer
  - **Bhagwat Jain,** Senior Structural Engineer
  - **John Ma,** Senior Structural Engineer
  - **Edward Stutzcage,** Health Physicist
  - **Raul Hernandez,** Reactor System Engineer

Staff Review Team

- **Technical Staff**
  - **Nicholas Mcmurray,** Materials Engineer
  - **John Honcharik,** Senior Materials Engineer
  - **Eric Reichelt**, Senior Materials Engineer
  - **Seshagiri Tammara,** Physical Scientist
  - **Vladimir Graizer,** Seismologist
  - **John Wu**, Mechanical Engineer
  - **Jim Strnisha,** Mechanical Engineer

- **Project Managers**
  - **Gregory Cranston,** Lead Project Manager
  - **Marieliz Vera**, Chapter Project Manager

Sections with no Open Items

- 3.2.1 – "Seismic Classification"
- 3.2.2 – "System Quality Group Classification"
- 3.3.1 – "Severe Wind Loads"
- 3.3.2 – "Extreme Wind Loads (Tornado and Hurricane Loads)"
- 3.4.1 – "Internal Flood Protection for Onsite Equipment Failure"
- 3.4.2 – "Analysis Procedures"
- 3.5.1.1 & 3.5.1.2 – "Internally Generated Missiles (Outside and Inside Containment)"
- 3.5.1.4 – "Missiles Generated by Tornadoes and Extreme Winds"
- 3.5.1.5 – "Site Proximity Missiles (Except Aircraft)"
- 3.5.1.6 – "Aircraft Hazards"
- 3.5.2 – "Structures, Systems, and Components to be Protected from Externally Generated Missiles"

Sections with no Open Items (Contd)

- 3.6.1 – “Plant Design for Protection against Postulated Piping Failures in Fluid Systems Outside Containment”
- 3.7.1 – “Seismic Design Parameters”
- 3.7.4 – “Seismic Instrumentation”
- 3.8.5 – “Foundations”
- 3.9.1 – “Special Topics for Mechanical Components”
- 3.10 – “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment”
- 3.12 – “ASME BPV Code Class 1, 2, and 3 Piping Systems and Associated Support Design
- 3.13 – “Threaded Fasteners—ASME BPV Code Class 1, 2, and 3”

Summary of open items

- Chapter 3 (without 3.9.2 and 3.11) has 18 Open items on the P2 SE
- 9 items have been resolved
- Items that are still open
  - 03.05.01.03-1 – turbine missile
  - 03.05.03-1 – turbine missile barrier
  - 03.06.02-1 – RVV and RRV bolted connection break exclusion justification
  - 03.07.03-1 – Bioshield
  - 03.08.02-1 – Updated CNV stress due to new seismic loads
  - 03.08.02-2 – Stress analysis of CNV RPV support
  - 03.08.03-3 – CNV fatigue evaluation
  - 03.09.03-1 – stress and fatigue analysis Reactor vessel and reactor vessel internals
  - 03.09.06-1 – ECCS Valve Design Demonstration Testing

Section 3.5.1.3 – Turbine Missile

John Honcharik

3.5.1.3 Turbine Missiles

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 essential SSCs that meet the acceptance criteria in DSRS Section 3.5.3.

Determining Turbine Missile Parameter on Barriers

- Open Item 03.05.01.03-1 : The staff determined that NuScale had not used the full spectrum of turbine missiles (size, weight and speed) which includes up to half of the last stage of the rotor in the barriers analysis.
- NuScale provided additional information in June which responded to staff RAIs and is currently under review.

3.5.1.3 Turbine Missiles

Current Staff Review of NuScale RAI Responses

- Need basis for destructive overspeed of 160% when other sources, including the EPRI Report 1006451, states 180 to 190%.
  - This will be the destructive overspeed used in barriers analysis as the bounding speed.

- Need methodology of determining speed of half of rotor based on NRC RAI No. 9596, Question 03.05.03-4 with the blades attached.

- Need to update previous reports in electronic reading room (Report ER-F010-6250, Revision 0 and Report ER-F010-6488, Revision 0) to include the analysis with the half of the rotor and other changes made based on audit and RAI responses. These reports are basis for turbine missiles and therefore need to be submitted to the NRC and documented in the DCD.

3.5.3 – Barrier Design Procedures

BP Jain

Use of Barriers

- To meet the regulatory requirements of 10 CFR Part 50, Appendix A, GDC 4, NuScale proposes:
  - to use the existing structures as barrier for protection of essential SSC against turbine missiles
  - the structures meet the acceptance criteria in SRP Section 3.5.3.

Turbine Missile Barrier

NuScale considered the following information in evaluating the barriers against potential turbine missile impact.

- NuScale considered the following missiles for evaluating barriers:
  (A) Turbine Blade : Weight: 32.6 lbs.; Velocity 784 mph ; 1.41" equivalent diameter ; (at 120% overspeed)
  (B) Turbine Blade with rotor fragment : Weight 52.6 lbs.; velocity 996 mph; rotor fragment width, 4.5"; (at 160% overspeed)
  (C) Half of last stage rotor: weight 3079 lbs.; velocity 350 mph; 12" wide by 48" in diameter; (at 160% overspeed)
- Acceptance Criteria – Concrete barriers should be thick enough to prevent:
  Back face scabbing
- Acceptable Impact analysis methods:
  - Empirical equations based on test results (SRP 3.5.3)
  - Finite Element analysis with validation

Turbine Missile Barrier

Staff Review of applicant information Summarized

- Using Empirical relations:
  - NDRC
    (Based on Equivalent Dia.)
  - Penetration
  - Scabbing

  Other Formula:
  - Penetration
  - Scabbing
- Limitations of the empirical methods:
  - Range limitations (test data limitations)
  - Missile and target both considered as hard

- Finite Element Method:
  (Based on blade Geometry)
  - Penetration
  - Scabbing
  (Based on Empirical Formula)

Turbine Missile Barrier

Staff Review of applicant information Summarized

- FEM procedure is not previously reviewed by the staff for high speed turbine missile impact
  - Benchmarking of computer results
    - Reported computer results against relevant solid bullet test data
  - Expected results
    - Deformable missiles expected to provide less penetration depth
    - Larger Kinetic energy of impact expected to provide larger penetration depth

Section 3.5.3 - Turbine Missile Barrier

Open Item 03.05.03-1

- Global damage assessment
  - Staff review inconclusive
  - Need additional information regarding the missile model, analysis methodology and procedure
  - Assumption of an automobile impact as a surrogate to half of the last stage rotor missile impact (3079 lbs) is inappropriate due to significant dissimilarities in:
    - weight of the hard missile (auto engine weighs much less than 3000 lbs)
    - Rotor foot print is less than auto foot print
    - Missile impact velocity (350 MPH) is much higher than auto crash MPH

Conclusions

- Staff will discuss its findings and path forward with Nuscale in a public meeting on July 16, 2019.
- NuScale to provide additional summarized information to demonstrate the consistency of the results between empirical and FEM approaches.

TF-3 Testing Purpose and Feedback on TF-3 Demonstration Modal Testing

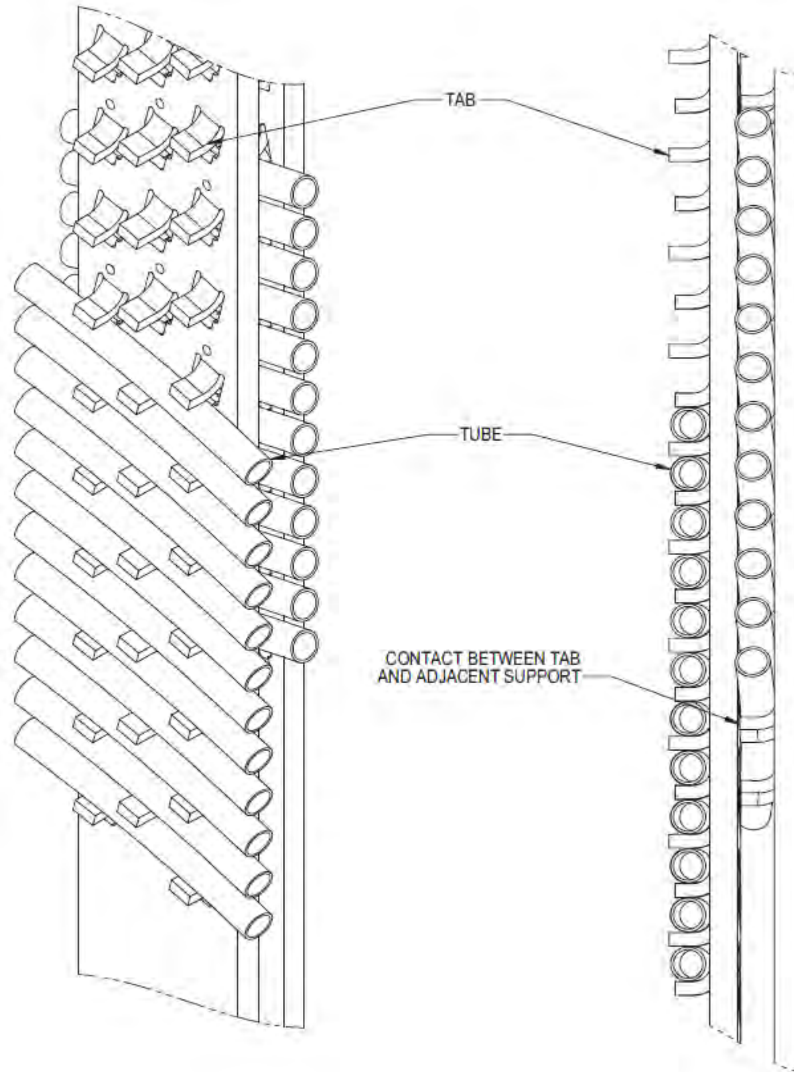
Yuken Wong

TF-3 Testing Purpose

- Obtain flow-induced vibration data for prototypical helical coil steam generator (HCSG) for validation of Fluid-elastic instability (FEI), vortex shedding (VS), and turbulence buffeting (TB) design analysis
- Specific objectives are:
  - Determine in-air and in-water natural frequencies and mode shapes of prototypic HCSG construction
  - Determine in-air and in-water damping values
  - Obtain data to characterize primary flow dynamic pressure fluctuations, SG tube and tube support vibration amplitudes
  - Obtain high flow rate vibration amplitudes to demonstrate margins to FEI and VS

TF-3 Demonstration Modal Testing - FEI

- Outer 2 (Columns 12 and 13) of 5 tube columns were installed
- Half of long tube spans appear tight (clamped/fixed) and half appear loose (pivoted)
- Short tube spans appear tight



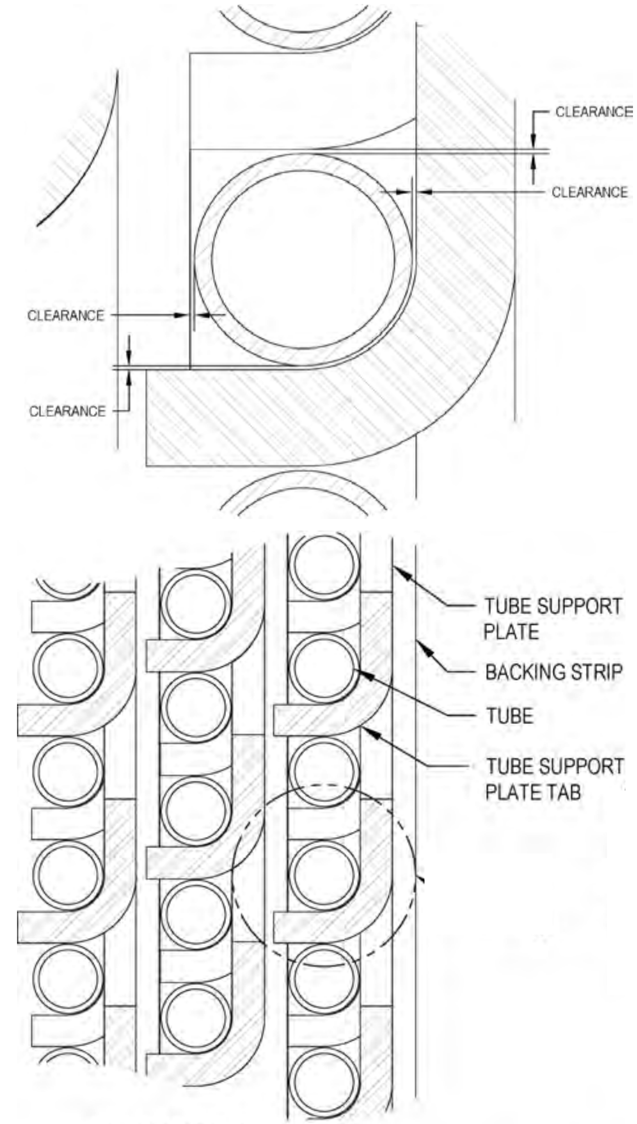
TF-3 Demonstration Modal Testing – FEI (cont'd)

- TF-3 was intended to replicate the thermal expansion of the tubes and support system and stiffer (clamped) boundary conditions at the supports

- However, TF-3 preloading system does not clamp all tubes into the supports due to clearances between tubes and supports

- Preloading system increases tube frequencies by less than 10%

- TF-3 will now test the looser (and therefore bounding) boundary condition without simulating the thermal expansion of tubes

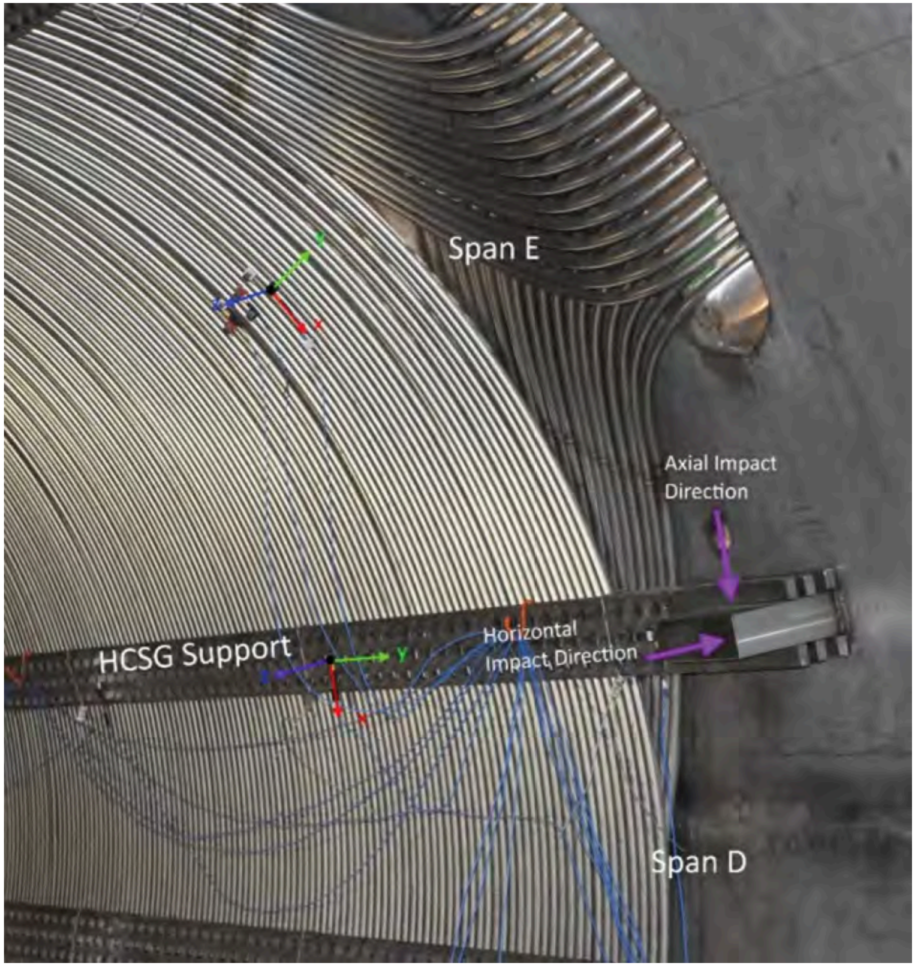


Analytical Prediction of Thermal Expansion - FEI

- NuScale presented thermal expansion calculations of selected tubes and supports
  - Thermal expansion will push the tubes down and radially out onto the tube supports
- Staff asked how other forces will affect the tube to support contact
  - Primary flow and gravity will also push tubes down
  - Secondary flow will push tubes radially out
- NuScale will quantify the effects of gravity and flow
- Staff will review the thermal expansion and gravity/flow effects

SG Tube Vortex Shedding

- Only lower tubes are susceptible to VS
- The feedwater transition bends have stiff geometry and support conditions
- Measured frequencies are much higher than previously calculated frequencies
- NuScale will soon submit an updated HCSG VS assessment which should show increased margin against VS



SG Tube Density Wave Oscillation and Thermal Fatigue

SG Tube DWO

- SG inlet flow restrictors (SGIFR) limit density wave oscillation (DWO) during normal operation
- TF-1 and TF-2 tests show DWO frequencies are very low compared to the SG tube resonance frequencies
- The DWO observed in TF-1 and TF-2 is not a concern for SG flow-induced vibration
- Minor differences in final SGIFR design and tested design will have insignificant effect on differential pressure

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SG Tube Thermal Fatigue

- SG tubes are ASME Class 1 and fatigue analysis is required
- Staff has not reviewed a specific SG tube fatigue analysis
  - Staff is reviewing the SG tube plenum fatigue analysis
  - NuScale has provided a ASME Fatigue Screening Report identifying most critical fatigue locations.
  - Those select locations are being reviewed as the analyses are prepared.
- ITAAC 2.1-4.2 exists to confirm that ASME data reports have been prepared for ASME Class 1 components in Table 2.1-2.
  - Table 2.1-2 includes RCS integral RPV/SG/Pressurizer

Section 3.9.6 – Functional Design and Qualification, and Preservice Testing and Inservice Testing of Pumps, Valves, and Dynamic Restraints

Thomas Scarbrough

ECCS Valve Design Review

- 3 Reactor Vent Valves (RVVs) and 2 Reactor Recirculation Valves (RRVs) allow natural circulation for emergency core cooling.
- Each RVV and RRV has first-of-a-kind (FOAK) design arrangement of a main valve, inadvertent actuation block (IAB) valve, solenoid trip valve, and solenoid reset valve connected by hydraulic tubing.
- NuScale conducting ECCS Valve Design Demonstration Testing at Target Rock to satisfy 10 CFR 50.43(e).
- Ongoing NRC audit with onsite test observation during June 3 week.

ECCS Valve Functional Design and Qualification, and IST

- ASME Standard QME-1-2007 for qualification of ECCS valves as accepted with conditions in RG 1.100 (Rev. 3).
- ITAAC acceptance criteria for functional qualification of safety-related valves require Qualification Report (specified in QME-1 standard).
- Code of record is ASME OM Code (2012 Edition) as incorporated by reference in 10 CFR 50.55a
- NuScale to apply Appendix IV to OM Code (2017 Edition) for preservice and inservice testing of ECCS valves as 50.55a alternative.

ECCS Valve Review Next Steps

- Complete NRC audit of ECCS Valve Design Demonstration Testing to close SER Open Item.
- Review Design Specification audit follow-up items to close SER Confirmatory Item.
- Update SER as necessary for Phase 4.

Abbreviations

- ACRS Advisory Committee on Reactor Safeguards
- AOV Air-Operated Valve
- ASME American Society of Mechanical Engineers
- BPV Boiler & Pressure Vessel
- BTP Branch Technical Position
- CIV Containment Isolation Valve
- COF Coefficient of Friction
- COL Combined License
- CNV Containment Vessel
- CRB Control Building
- DC Design Certification
- DSRS Design Structure Response Spectra
- DCA Design Certification Application
- DWO Density Wave Oscillation
- ECCS Emergency Core Cooling System
- FEM Finite Element Method
- FWS Feedwater Piping
- FEI Fluid-Elastic Instability
- FOAK First of a Kind
- FSAR Final Safety Analysis Report
- HCSG Helical Coil Steam Generator
- HOV Hydraulic-Operated Valve
- IAB Inadvertent Actuation Block
- IST Inservice Testing
- ITAAC Inspections, Tests, Analyses, and Acceptance Criteria
- LBB Leak Before Break
- MSS Main Steam Piping
- MOV Motor-Operated Valve
- NDRC National Defense Research Committee

Abbreviations

- NRO NRC Office of New Reactors
- OM Operation and Maintenance
- PST Preservice Testing
- QA Quality Assurance
- RAI Request for Additional Information
- RG Regulatory Guide
- RCS Reactor Coolant System
- RPV Reactor Pressure Vessel
- RRV Reactor Recirculation Valve
- RVV Reactor Vent Valve
- RXB Reactor Building
- SER Safety Evaluation Report
- SRP Standard Review Plan
- SSCs Structures, Systems, and Components
- SG Steam generator
- SGIFR Steam Generator Inlet Flow Restrictor
- VS Vortex Shedding
- TB Turbulent Buffeting

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Safety Evaluation with Open Items: Chapter 15, “Transient and Accident Analyses”; Containment Performance

NuScale Design Certification Application

ACRS Full Committee Meeting
July 10, 2019

Non-Proprietary

NRC Staff Review Team

- Technical Reviewers:
  - Antonio Barrett, NRO/SRSB
  - Andrew Bielen, RES/CRAB
  - Tim Drzewiecki, NRO/ARTB
  - Jim Gilmer, NRO/SRSB
  - Syed Haider, NRO/SRSB
  - Michelle Hart, NRO/RGRB
  - Andrew Ireland, RES/CRAB
  - Shanlai Lu, NRO/SRSB
  - Ryan Nolan, NRO/SRSB
  - Jeff Schmidt, NRO/ARTB
  - Alex Siwy, NRO/SRSB
  - Ray Skarda, RES/CRAB
  - Matt Thomas, NRO/SRSB
  - Jason Thompson, RES/CRAB
  - Boyce Travis, NRO/ARPB
  - Carl Thurston, NRO/SRSB
  - Chris Van Wert, NRO/ARTB
- Project Management
  - Rani Franovich, NRO/DLSE

Agenda

- Unclear Open Items (UOIs)
- Staff Requirements for SECY-19-0036
- Chapter 15 Limiting Cases
- Critical Heat Flux Ratio (CHFR) Variance
- Long Term Cooling
- Chapter 15 Exemptions
- Return to Power
- Containment Structure
- Containment Heat Removal

Unclear Open Items

- Items without a mutually understood and clearly defined path towards resolution
  - Includes items for which there may be an agreement between NuScale and NRC, but documentation and supporting information is pending
  - UOI list changes frequently as new material is submitted by NuScale
  - List of UOIs has been reduced since Chapter 15 SER drafted

Status of Current Chapter 15 Unclear Open Items

Return to Power Analysis

- Recriticality while natural circulation is interrupted (15.0.5-1): Awaiting calculation and revised RAI response from NuScale
- Excluding event scenario of ejected rod with margin to stuck rods (15.0.6-6): NuScale RAI response pending, expected to support decision that CRDM housings robust with enhanced features, precluding need to evaluate long term effects such as recriticality within design basis, but retaining short term analysis
- Boron redistribution during ECCS (15.0.6-5): Continuing dialogue regarding volatility correlation and mixing assumptions
- Nuclear analysis parameters in Return to Power analysis (15.0.6-3): Expected to be addressed in revised calculations
- Decay heat level precluding a return to power and addressing EOC ECCS return to power (15.0.6-4): NuScale analysis of EOC ECCS return to power pending

Long-Term Cooling

- Statements in LTC technical report that analysis demonstrates adequate cooling for 30 days (15.0.5-2): NuScale reevaluating analytic basis for statement and potential revisions to statement

Topical Report Methodologies (15.0.2-2, LOCA; 15.0.2-4, non-LOCA)

- NRELAP5 v1.4 and revised base model (LOCA TR): Staff recently received NRELAP5 v1.4 and revised base model and is evaluating
- SG heat transfer uncertainty (non-LOCA TR): Staff awaiting revised NuScale analyses expected to demonstrate results insensitive

Chapter 15 reanalysis from design and methodology changes

- ECCS logic changes (15.0.0.4-1, 15.6.5-1): Submittal of analysis revisions expected August 30
- DHRS logic changes (15.0.0.4-1, 15.6.5-1) Submittal of analysis revisions expected August 30
- NRELAP5 v1.4 (15.0.2-1): Submittal of analysis revisions expected August 30

Staff Requirements for SECY-19-0036

- The inadvertent actuation block (IAB) valve is part of the emergency core cooling system (ECCS) valve and is designed to prevent high-pressure RPV blowdown when the dc power supply (EDSS) is lost to the ECCS trip valve.
- Chapter 15 analyses don't assume single failure of the IAB to block
- The Commission determined:

  "The staff should review Chapter 15 of the NuScale Design Certification Application without assuming a single failure of the inadvertent actuation block valve to close."

Chapter 15 Limiting Cases

Figure of Merit	Limiting Event	Limiting Value	Acceptance Criteria	Margin
RCS Pressure	15.2.8, Feedwater System Pipe Breaks Inside and Outside Containment	2164 psia	AOO: 2310 psia PA: 2520 psia*	AOO: 6.32% PA: 14.1%
SG Pressure	15.6.3, Steam Generator Tube Failure	1806 psia	AOO: 2310 psia PA: 2520 psia*	AOO: 21.8% PA: 28.3%
MCHFR – VIPRE	15.4.3, Control Rod Misoperation (Single Rod Withdrawal)	1.614	1.284	25.7%
MCHFR – NRELAP5	15.6.6, Inadvertent Operation of ECCS	1.41	1.13	24.8%
Collapsed Liquid Level (CLL) Above Top of Active Fuel	15.6.5, LOCA	< 5 inches	0 inches	N/A

* Although these events are classified as postulated accidents (PAs), they meet the acceptance criteria for anticipated operational occurrences (AOOs).

Chapter 15 CHF Correlations

Non-LOCA Events (Non-Loss of RCS Coolant)

- NSP4 CHF Correlation
- VIPRE-01 Sub-Channel Analysis Code
- NRC Approved Methodology

LOCA and LOCA-Like Events (Loss of RCS Coolant)

- High-Flow CHF Correlation and Low-Flow CHF Correlation
- NRELAP5 Systems Code
- Methodology Under NRC Staff Review

Chapter 15 CHF Correlations

Critical Heat Flux Ratio Variance

<u>Correlation</u>

Critical Heat Flux

- NSP4 correlation for non-loss of RCS coolant analyses
- High-flow and low-flow correlation for loss of RCS coolant analyses

<u>Initial Conditions</u>

Analysis Input Condition Biasing

- RCS temperature, RCS Flow, Pressurizer Pressure, etc.

<u>Methodology</u>

Analysis Code

- VIPRE-01 sub-channel code calculates MCHFR during the period of interest only
- NRELAP5 systems code calculates MCHFR continuously throughout the event

Modeling

- Difference in peaking factor and heat flux assumptions

ECCS Long-Term Cooling Capability and Analysis

- Addresses residual heat removal for LOCA and non-LOCA analyses when on the ECCS
  - Focuses on reduced inventory events which would challenge ECCS cooling capability
  - Assumes shutdown
  - Evaluates out to 72 hours
- Figures of merit include:
  - Decreasing clad temperature
  - Minimum RCS temperature to prevent boron precipitation
  - CLL remains above the top of active fuel
- Staff review focused on
  - Validation to NIST test data
  - NRELAP5 model
  - Assumed analysis conditions
- In-vessel downstream effects evaluation performed in DCA Section 6.3
  - Staff found NuScale evaluation acceptable

LOCA Exemption

<u>Appendix K – LOCA Exemptions</u>

- ANS-71 replaced with ANS-73 with RELAP5 standard precursors
- Post LOCA – PWR reflood and refill (phenomena are not to be encountered)
  - post-CHF regime heat transfer correlations
  - Baker-Just Metal Water Reactions
  - Clad swelling and rupture

GDC 27 Exemption

- General Design Criterion (GDC) 27 states:
  - The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.
- Staff took the position in the pre-application Gap 27 letter (ML16116A083) that “reliably controlling reactivity” in GDC 27 means shutdown as the final state when considering the totality of NRC regulations regarding reactivity control
- Following an initial shutdown, the NuScale reactor can return and maintain criticality during a cool down on the safety-related, passive heat removal systems (DHRS and ECCS) under certain conditions
- NuScale submitted an exemption to GDC 27 and requested approval of a principal design criterion (PDC), PDC 27
  - Exemption evaluation includes the lack of ECCS injection

SECY-18-0099

- SECY-18-0099 (ML18065A540) used the following three criteria used to evaluate the exemption:
  - The design of the reactor must provide sufficient thermal margin such that a return to power does not result in the failure of the fuel cladding fission product barrier, as demonstrated by not exceeding specified acceptable fuel design limits (SAFDL) for the analyzed events.
  - The combination of circumstances and conditions leading to an actual post reactor trip return to criticality is not expected to occur during the lifetime of a module.
  - The incremental risk to public health and safety from the hypothesized return to criticality at a NuScale facility with multiple reactor modules does not adversely erode the margin between the Commission's goals for new reactor designs related to estimated frequencies of core damage or large releases and those calculated for the NuScale design.
- ACRS supported the proposed staff criteria with the addition of evaluating the overall facility risk, which is reflected in the third criterion above (ML18052A532)
- Satisfying the three criteria in SECY-18-0099 would ensure no undue risk to public health and safety

Return to Power Scenarios

- Three scenarios can potentially lead to a return to power
  - DHRS cooldown with dc power (EDSS)
  - DHRS cooldown without dc power (EDSS)
    - ECCS actuation at IAB setpoint
  - ECCS cooldown
- Can occur as a result of most Chapter 15 events
- Key assumptions assumed in the return to power scenarios
  - No operator action
  - Only safety-related equipment is used to mitigate the event
  - The worst stuck rod is assumed out of the core consistent with current GDC

Return to Power Analyses

- DCA Part 2, Tier 2, Section 15.0.6, Revision 2, presents a DHRS cool down, retaining single phase natural circulation with ECCS valves opening at the maximum return to power, which is thought to bound all three cooldown scenarios
  - Maximum, Core average return to power 10% rated thermal power (RTP), equilibrium return to power of approximately 2.5% RTP
  - MCFHR limit met
- A potential DHRS return to power when water level drops below the riser (UOI 15.0.5-1) still needs to be evaluated to ensure SAFDLs are met
- EOC, ECCS return to power case (UOI 15.0.6-4) still needs to be evaluated to ensure SAFDLs are met
- Boron redistribution and a potential return to power at times than EOC still needs to be resolved (UOI 15.0.6-5)
- Completion of staff's confirmatory analyses

Containment Performance

- Containment Structure
  - Peak Containment Pressure/Temperature Review
- Containment Heat Removal

6.2.1.1: Containment Structure

<u>Scope of the Staff Review</u>

- NuScale FSAR Section 6.2.1.1 (DCA Part 2, Tier 2)
  - Containment Response Analysis Methodology (CRAM) Technical Report (TeR)

- Key regulatory requirements
  - Sufficient margin in containment design pressure/temperature (GDCs 16 & 50)
  - Pressure reduction by 50% from its peak value within 24 hours (GDC 38)

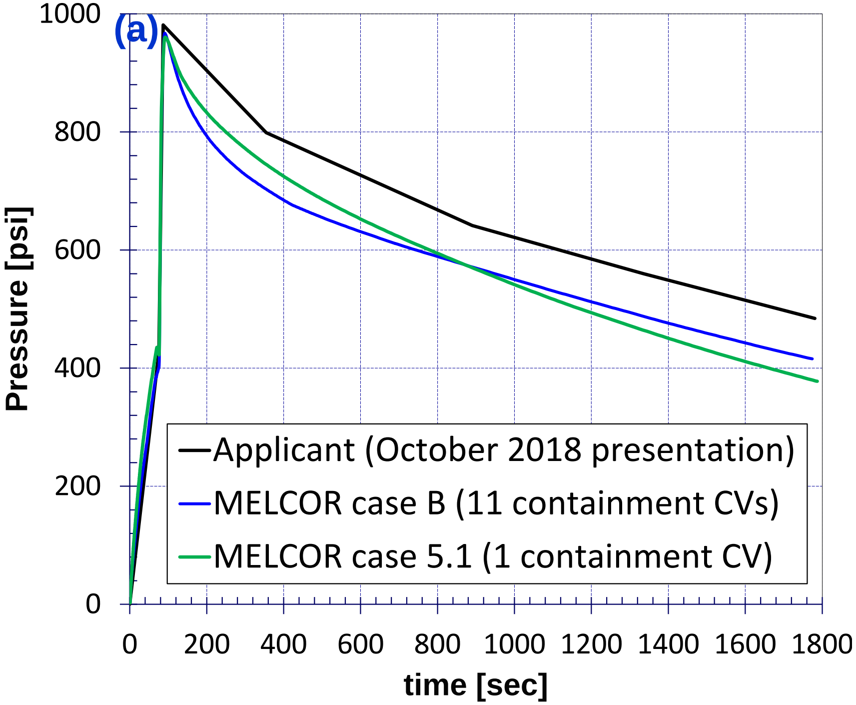
- Qualification of the NRELAP5 code to predict the NPM containment response
  - LOCA TR for primary side pipe breaks and reactor valve opening events
  - Non-LOCA TR for MSLB and FWLB events

- No significant differences in physical phenomena b/w the LOCA and valve opening events. LOCA and non-LOCA PIRTs also applicable to the valve opening events and CRAM
  - CRAM is an extension of the NuScale LOCA, valve opening event, and non-LOCA methodologies.
  - CRAM TeR references these methodologies and justifies differences for the CNV response analysis.

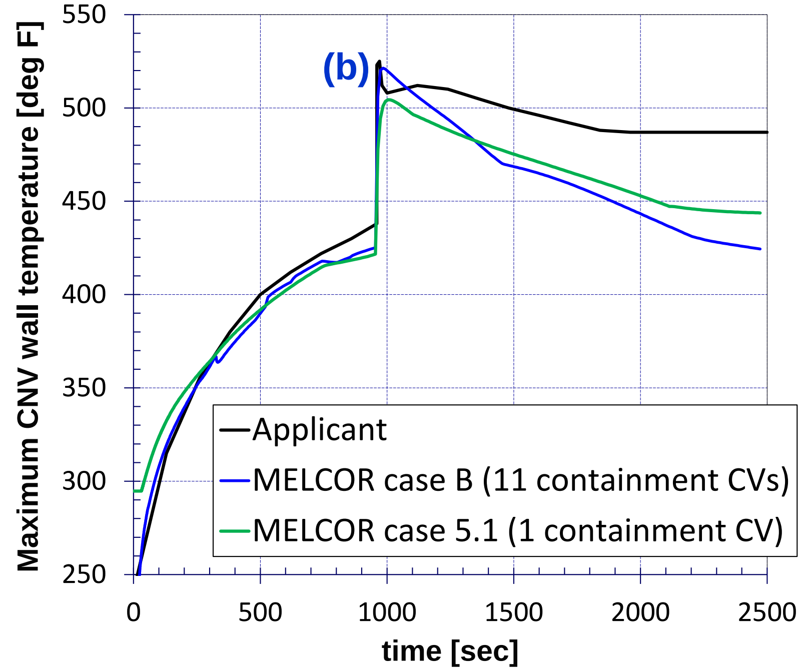
- 9 FSAR Chapter 6.2.1.1 RAIs issued with 21 questions in addition to related LOCA TR RAIs

6.2.1.1: Containment Structure

Staff & Applicant's Results for the Limiting Pressure & Temperature DBA Scenarios



(a) Pr. Limiting: Inadvertent RRV Opening



(b) Temp. Limiting: RCS Injection Line Break

Applicant Peak Values by NRELAP5	**986 psia**	**526 °F**
Staff Confirmatory Peak Values by MELCOR	**967 psia (a)**	**521 °F (b)**
FSAR CNV Design Values	1050 psia	550 °F

Section 6.2.2 – Containment Heat Removal

NuScale Approach for Long-term Cooling

- Limit containment debris to latent only
- No strainers subject to clogging
- Low fiber limit precludes a filtering bed at the fuel inlet
- Exclude material-chemistry combinations expected to cause chemical effects

Finding

- Using DCA specified debris limits, staff determined there is reasonable assurance that consequential debris will not impair long-term core cooling functionality.
- The type and amount of assumed chemical effects are acceptable for fuel inlet blockage and deposition on the fuel based on: staff's approval of WCAP-16793-NP-A for testing and conservative debris limit

GDC 40 Exemption

- Periodic inspections of the containment heat removal surfaces (as required by GDC 39, in accordance with ASME BPVC) will identify surface fouling or degradation that could potentially impede heat transfer from the containment
- The other systems that act to remove heat from the containment, such as ECCS are also inspected and tested
- The underlying purpose of the rule, to verify that the performance characteristics of the CHRS remain with acceptable parameters and to ensure operability, will still be accomplished

Open Item

The limits imposed on the cleanliness program were not clearly identified in the FSAR, and staff could not make a finding regarding a COL item; as such, staff is tracking Open Item 6.2.2-1, pending a submittal from NuScale to address the lack of clarity in the FSAR to explicitly identify the debris limits in Tier 2.

Acronyms

- AC alternating current
- ACRS Advisory Committee on Reactor Safeguards
- BOC beginning of cycle
- CFR *Code of Federal Regulations*
- CHF critical heat flux
- CHFR critical heat flux ratio
- CLL collapsed liquid level
- CNV containment vessel
- COL combined license
- CVCS chemical and volume control system
- DCA design certification application
- DHRS decay heat removal system
- DSRS design-specific review standard
- ECCS emergency core cooling system
- EDSS highly reliable dc power system
- ELVS low-voltage ac power distribution system
- EM evaluation model
- EOC end of cycle
- FSAR final safety analysis report
- FWLB feedwater line break
- GDC general design criterion/criteria
- HZP hot zero power
- IAB inadvertent actuation block
- LOCA loss-of-coolant accident
- MCHFR minimum critical heat flux ratio
- MPS module protection system
- MSIV main steam isolation valve
- MSLB main steamline break
- MTC moderator temperature coefficient
- NPM NuScale Power Module
- PDC principal design criterion/criteria
- PIRT phenomena identification and ranking table
- RAI request for additional information
- RCS reactor coolant system
- RPV reactor pressure vessel
- RRV reactor recirculation valve
- RTP rated thermal power
- SAFDL specified acceptable fuel design limits
- SER safety evaluation report
- SFC single failure criterion
- SG steam generator
- SGTF steam generator tube failure
- SRP Standard Review Plan
- TR topical report

Questions/comments from members of the public before the closed session starts?

Backup Slides

Changes to DHRS/ECCS logic

- NuScale submitted a plan to revise the module protection system (MPS) logic for operational considerations:
  - Existing DHRS actuation signal will be split into two signals:
    - A secondary side isolation (steam and feedwater isolation)
    - DHRS actuation (opening the DHRS actuation valves)
  - NuScale further plans to remove the ECCS actuation signal on low reactor pressure vessel level, which the applicant states is not credited

- Staff has received the proposed changes, with the exception of planned changes to Chapters 15 and 19

- Staff review of these changes is ongoing, and will not be complete until after the planned re-analysis of Chapter 15 transients, which will incorporate the logic changes above