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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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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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NUSCALE DESIGN-CENTERED SUBCOMMITTEE

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TUESDAY

FEBRUARY 18, 2025

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The Subcommittee met via Teleconference, at 8:30 a.m. EST, Walter L. Kirchner, Chair, presiding.

COMMITTEE MEMBERS:

WALTER L. KIRCHNER, Chair
RONALD G. BALLINGER, Member
VESNA B. DIMITRIJEVIC, Member
CRAIG A. HARRINGTON, Member
GREGORY H. HALNON, Member
ROBERT P. MARTIN, Member
SCOTT P. PALMTAG, Member
THOMAS E. ROBERTS, Member
MATTHEW W. SUNSERI, Member

ACRS CONSULTANTS:

DENNIS BLEY

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

MICHAEL SNODDERLY

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

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

8:31 a.m.

CHAIR KIRCHNER: Okay. Good morning. The meeting will now come to order. This is a meeting of the NuScale Design-Centered Review Subcommittee of the Advisory Committee on Reactor Safeguards.

I'm Walt Kirchner, Chairman of today's subcommittee meeting. ACRS members in attendance are Ron Ballinger, Craig Harrington, Robert Martin, and Thomas Roberts. ACRS members in attendance virtually via Teams are Vesna Dimitrijevic, Greg Halnon, Scott Palmtag, Matt Sunseri, and myself.

We have one of our consultants participating virtually via Teams, Dennis Bley. If I've missing anyone, either members or consultants, please speak up now. Michael --

DR. SCHULTZ: Walt --

CHAIR KIRCHNER: -- Snodderly -- yes.

DR. SCHULTZ: Walt, Steve Schultz is here.

CHAIR KIRCHNER: Oh, thank you. Our consultant, Steve Schultz, is also with us. Thank you, Steve. Michael Snodderly of the ACRS staff is the Designated Federal Officer for this meeting.

No member conflicts of interest were identified for today's meeting. And I know we have a quorum as well. During today's meeting, the subcommittee will receive a briefing on the staff's evaluation of NuScale Power, LLC's US460 standard design approval application, Chapter 6, engineer safety features, Section 17.4 of Chapter 17, reliability assurance program, and Chapter 19, probabilistic risk assessment and severe accident evaluation.

We previously reviewed the certified NuScale US600 design as documented in our July 29, 2020 letter report on the safety aspects of the NuScale small modular reactor. Like the staff, we are performing a delta review between the two designs, including a power uprate from 50 to 77 megawatts electric per module. We are reviewing these chapters as part of our statutory obligation under Title 10 of the Code of Federal Regulations, Part 52, Subpart E, Section 14.1, referral to the Advisory Committee on Reactor Safeguards to report on those portions of the application which concern safety.

The ACRS was established by statute and is governed by the Federal Advisory Committee Act or FACA. The NRC implements FACA in accordance with our regulations. Per these regulations and the committee's bylaws, the ACRS speaks only through its published letter reports.

All member comments should be regarded as only the individual opinion of that member, not a committee position. All relevant information related to the ACRS activity such as letters, rules for meeting participation, and transcripts are located on the NRC public website and can be readily found by typing About Us ACRS in the search field on the NRC's home page. The ACRS, consistent with the agency's value of public transparency and regulation of nuclear facilities, provides opportunity for public input and comment during our proceedings.

We have received no written statements or requests to make an oral statement from the public. However, we have set aside time at the end of the meeting for any public comment should there be any. Portions of this meeting may be closed to protect sensitive information as required by FACA and the government in the Sunshine Act.

Attendance during the closed portion of the meeting will be limited to NRC staff and its consultants, applicants, and those individuals and organizations who have entered into an appropriate confidentiality agreement. We will confirm that only eligible individuals are in the closed portion of the meeting. The ACRS will gather information, analyze relevant issues and facts, and formulate proposed conclusions and recommendations as appropriate for deliberation by the full committee.

A transcript of the meeting is being kept and will be posted on our website. When addressing the subcommittee, the participants should first identify themselves and speak with sufficient clarity and volume so that they may be readily heard. If you're not speaking, please mute your computer on Teams or by pressing *6 if you are on your phone.

Please do not use the Teams chat feature to conduct sidebar discussions related to the presentations. Rather limit use of the meeting chat function to report IT problems. For everyone in the room, please put all your electronic devices in silent mode and mute your laptop microphone and speakers.

In addition, please keep sidebar discussions in the room to a minimum since the ceiling microphones are live. For the presenters, your table microphones are unidirectional and you'll need to speak into the front of the microphone to be heard. Finally, if you have any feedback for the ACRS about today's meeting, we encourage you to fill out the public meeting feedback form on the NRC's website.

And with that, we'll now proceed with the meeting. And I think, Mike, it's best for me to turn to Bob Martin and let him run the meeting from there. He'll be able to better coordinate than myself. So with that, Bob, I think our next step is to turn to the NRC project management team for NuScale.

MEMBER MARTIN: That's right. I think it's MJ.

MR. Jardaneh: Yes, thank you. Good morning, Chair. And good morning to ACRS subcommittee members, NuScale participants, NRC staff, and members of the public.

My name is Mahmoud Jardaneh. I serve as the branch chief, the new reactor licensing branch responsible for licensing of the NuScale US460 design in addition new and renewed licenses at NRR. Thank you for the opportunity today, for the staff and their review of select NuScale US460 standard design approval application or SDAA chapters and topical reports.

As you are aware, the staff is reviewing is reviewing all chapters of the SDAA concurrently with standard completion dates based on the complexity of the chapter and the extent of the changes from the certified NuScale US600 design. Today, the staff will be presenting their review. That's six SDAA chapters, including Section -- including Chapter 6, engineered safety features, the remaining section of Chapter 17, quality assurance and reliability assurance, and Chapter 19, probabilistic risk assessment and severe accident analysis.

Previously, the staff presented to this subcommittee on Chapters 2, 3, 5, 7, 8, 9, 10, 11, 12, 13, 14, 16, portions of Chapter 17, and Chapter 18. The staff also presented on the loss of coolant accident evaluation model topical report, one of the three topical reports associated with this application. The staff is finalizing the review of the remaining SDAA chapters of topical reports, and we will inform the ACRS from the safety evaluation of the remaining chapters where topical reports are available to the ACRS.

Today's meeting, the staff will focus on the delta from the design certification that the NRC has and the subcommittee reviewed in the test. Once again, thank you for the opportunity. And we look forward to a good discussion today.

MEMBER MARTIN: Thanks, MJ. I assume we'll move to NuScale. Tom?

MR. GRIFFITH: Good morning, ACRS

subcommittee members, NRC staff, NuScale staff on the line, and the public. This is Thomas Griffith, Licensing Manager for NuScale's US460 Standard Design Approval Application. We are looking forward to the opportunity today to present Chapter 6, 17.4, and Chapter 19. I look forward to the discussion that we're going to have today. And with that, I will turn over to Tyler Beck to start the presentation on Chapter 6.

MR. BECK: Hello. My name is Tyler Beck. I'm a licensing engineer with NuScale, and I'm the licensing engineer for Chapter 6 amongst some other chapters.

Part of my time at NuScale, I was a reactor systems engineer at the -- with NRC staff. And part of my time with NRC, I got my bachelors of science in nuclear engineering from University of Tennessee. Next slide. We'd like to acknowledge that this work, we have DOE support from. Next slide.

This is an overview of Chapter 6, and it lists this section that we covered today. I'd like to note that this is the design of engineered safety features as discussed in the FSAR. Chapter 6 includes a breadth of components and systems.

This presentation is not specific to the accident sequences or evaluations such as Chapter 15 or the PRA which will be discussed later today. As previously noted, there will be -- this presentation will be a delta review from the US600 design certification application to the US460 standard design approval application. Next slide. Section 6.1 is engineered safety feature materials.

And for noteworthy changes from the design certification application, the containment vessel upper portions of the vessel materials have changed. So previously in the DCA, it was SA-508, low-alloy steel, and the SDA design proportions are F6NM martensitic stainless steel. Along with that change in the SR Section 6.1, we're added a new table for dissimilar metal welds.

It describes dissimilar metal welds. And we've implemented additional welding controls such as post weld heat treatment controls and in regard to the staff audit and NRR review base. Next slide. Mentioned the material change is the significant change here.

And here we have a couple of figures from the application. On the right, you can see the containment system as a whole. On the left, you can see the lower containment vessel and you can see where the material change is from FXM-19 to F6NM.

And along these lines, I believe there was an outstanding question from the last meeting on FXM-19. And there was a mention of a paper. And we just wanted to clarify that we had reviewed the paper.

And I believe we determined it wasn't applicable in certain respects to our design. And we have a couple of subject matter experts on the phone. And I think, Steve, you wanted to say something in this regard.

MR. WOLBERT: Sure. Yeah, good morning. This is Steven Wolbert, the manufacturing engineer with NuScale. Yeah, we did review the paper and I have seen this paper before among others.

Some of the conclusions drawn from the paper, I guess, start off the boundary conditions of the paper studied. This paper primarily looked at case hardening via nitride treatment on XM-19. I guess just a noteworthy comment there is we don't employ any case hardening on XM-19.

It's kind of a more severe condition tested there. And then additionally, the paper also studied XM-19 tubing with 25 to 35 percent cold work and case hardening. But of these conditions are much more extreme than what NuScale permits. And then in addition, the study concluded that the XM-19, even in those extreme conditions, actually outperformed the control specimens of a standard F-304 material. So we're confident that we've covered most of the concerns raised by the paper, the conditions raised by the paper and welcome any other questions in this regard.

MEMBER MARTIN: This is Bob Martin. And we'd like to have some questions. Regarding -- and so XM-19 is relatively recent addition to the code case. Is that correct? I'll look over to Ron.

MEMBER BALLINGER: You said with respect to --

(Simultaneous speaking.)

MEMBER BALLINGER: It's been around forever.

MEMBER MARTIN: Well, the material itself --

MEMBER BALLINGER: Yeah.

MEMBER MARTIN: -- right? So I was just going to ask the question. So say, 50 years from now you have this containment sitting in water. Obviously, you have to have the standard inspections and what have you.

But you find that there's a problem. Is

this a containment that can be replaced? So if you wanted to go, say, 100 years. Or what would happen at that juncture ran into a problem and everything else is working just fine?

MR. WOLBERT: Yeah, this is Steve Wolbert again. Prior to NuScale, my career was component repair and replacement with another company where we do steam generator replacements, head replacements, those types of things. Obviously, an effort like that would be pretty extensive.

The first option if you ran into some kind of problem would be to a field repair. And so we have looked into that type of methods that one would employ. Really a lot of that starts to fall into the Section 11 -- ASME Code, Section 11, rules and requirements.

So we do have NuScale members on those committees in ASME that are looking into those options and additionally with our supplier partners. Now this is something that would really be on the plant owner to control but is something that we understand the question and welcome it and want to have those dialogues with our customers as well. So we do have some options that we're looking into, including some additional -- we've done extensive weld testing with our suppliers. And we do believe there are some relevant field repair methods that would be applicable.

MEMBER MARTIN: Right. Appreciate that. Obviously, it's just kind of a sidebar kind of question. But certainly the uniqueness of the design gives you certain opportunities that maybe other plants don't have and put some resilience into your design where another alternative would not necessarily have that. So I appreciate that answer. Go ahead, continue.

MR. BECK: Next slide, please. Section 6.2 is containment systems. For the significant changes from the DCA, the last slide, we mentioned the material changes for the containment vessel. There are a number of containment vessel penetrations from the DCA to the SDA.

The design ratings have been increased. So the design pressure ratings have been increased 1,200 psi. Design pressure rating has been increased to 600 degrees Fahrenheit.

And then otherwise for containment vessel penetrations, the CVCS injection and discharge line penetrations include venturis that are integral to the penetration. And that's to mitigate potential breaks that occur outside of the containment vessel. On the topic of combustible gas control, we have differing strategy in the SDA.

So in this design, we include a safety-related passive autocatalytic recombiner or PAR. And that maintains an inert containment atmosphere. With that change, it removes potential combustion loads because flammability precluded in an inert environment is maintained. And it also coincides with the exemption we have for combustible gas monitoring requirements. And so there are no combustible gas monitoring provisions strictly in the containment vessel.

MEMBER ROBERTS: Tyler, this Tom Roberts. The removal of combustion loads will maintain an inert environment, reading through the staff's SE, they seem to indicate that you have maintained the combustion load. And there's an RAI where you discuss that you did the analysis to show that you could still withstand a combustion load if it were to occur. So I'm a little confused as to what's the intent of that. Is the intent to -- or I would say, what is the intent?

MR. BECK: I think you're referencing Chapter 19, adiabatic, isochoric, complete combustion analysis.

MEMBER ROBERTS: It's the Chapter 19 RAI that has a change in Chapter 6 in it. And the staff's Chapter 6 safety evaluation basically talked about that. It says part of the basis for accepting the pressure table of the PAR is you still have the analysis that you can withstand the combustion load even if the PAR is there to inert the environment. You've still got the analysis and if you had the detonation, it would still be covered. I was trying to understand what would that bullet mean.

MR. MULLIN: Yeah, this is Etienne Mullin from NuScale PRA. That analysis was part of several analyses that we prepared and shared with the staff to demonstrate that we don't need the PAR for the success criteria of the PRA. We don't need the PAR to prevent a core damage event.

We don't need the PAR to prevent a large release event or core damage event. And so we prepared several analyses to demonstrate that the PAR wasn't necessary. And for that reason, it's not included in our containment event trees.

MEMBER ROBERTS: So I'm confused what this bullet means. It seems like you've done the analysis with a convection mode in support of the PRA. Is that a nondesign analysis so it's got less QA? I'm just trying to understand what the distinction is.

MR. MULLIN: I think that's a fair characterization. The combustion analysis that was performed is for a beyond design basis event prepared with different analysis assumptions.

MR. BECK: And I believe if you include detonation loads, for example, those are included in the individual design specifications for the components. And that is not specifically included in the US460 standard design because we do preclude those loads.

MEMBER ROBERTS: Yeah, okay. It seemed to me the most important point is that the containment itself is essentially -- I think it's uprated, right, from the US600 containment. So there's no reason to believe that you've got more vulnerability to detonation. Plus you have -- you might call it defense in depth multiplier, though you have a much lower likelihood of having a combustion event. Is that a fair characterization?

MR. BECK: Yeah, I would say that's a fair characterization except I wouldn't say much more lower likelihood with the PAR that the loads are precluded and detonation is precluded entirely.

MEMBER ROBERTS: Okay, thank you.

MEMBER MARTIN: I wanted to get a clarification on your last bullet regarding exemption from monitoring requirements. You obviously are going to -- I mean, it's an essential vacuum, right? You will be monitoring pressure which would be basically zero all the time, correct?

What do you mean specifically by that? Because you have to have some safety-related monitoring because it might -- and if you had air in there, again, that's a source term of the combustion event. That would factor back into, say, a design basis analysis because air would affect, say, condensation rates and such like that. I want to give you an opportunity to clarify what you mean by monitoring requirements.

MR. BECK: There are not specific provisions within containment to monitor hydrogen gas and oxygen gas concentrations.

MEMBER MARTIN: But at least pressure?

MR. BECK: Yes.

MEMBER MARTIN: So if pressure was elevated from at least the target, you would otherwise expect that more than likely you had some kind of air ingress. And then you could act on that. So it's not like you have no monitoring.

MR. BECK: Yes.

MEMBER HARRINGTON: This is Craig Harrington. The PAR, do you have to do anything in the outage? Just continue happily combining oxygen with any hydrogen it finds? Or what happens?

MR. BECK: The PAR is included in tech specs. And there are inspection and testing requirements to test some sample of the catalytic plates to make sure they're recombining the right amount of hydrogen and oxygen.

MEMBER HARRINGTON: Okay. That continues -- when you open the system up to atmosphere, it just keeps doing the same thing?

MR. BECK: Yes.

MEMBER HARRINGTON: Okay. And a couple things unrelated probably to the delta between the 600 and this design, I wasn't around for that. There was wording in the FSAR chapter that says it will be fueled in a partially flooded condition. NPM is moved loosely from the reactor building frame to the refueling area without loss of reactor coolant inventory and refueled in a partially flooded condition, precluding operation with reduced inventory. What does that mean?

MR. BECK: That was in Chapter 6.

MEMBER HARRINGTON: Yes, Section 6.2.1.1.2.

MR. BECK: I'm not sure on the intent of the statement originally.

MR. MULLIN: Yeah, I'm not familiar with the statement you're referring to. But the NPM is filled with water before being moved. It's not filled to the top. It's partially filled up to approximately the pressurizer baffle plate is being referred to.

MEMBER HARRINGTON: I mean, it seemed to be after. You pick it up. You move it over. You separate the flanges, lift the top off.

MR. MULLIN: The core is certainly submerged by the depth.

MEMBER HARRINGTON: I mean, it seemed obvious. I just didn't make any sense of the words. And there's also discussion of four instruments that measure and monitor containment water level. This is during operation during an accident phase. They're at the reactor pressure boundary interface, four independent channels of CNV water level instrumentation. What kind of instruments do you use?

MR. BECK: Do we have anyone from I&C on the call to discuss the containment vessel water level instrumentation?

MR. MEYER: Yes, Rob Meyer, NuScale, I&C. What would you like explained?

MEMBER HARRINGTON: What kind of instrumentation are you using?

MR. MEYER: Oh, it's a thermal dispersion sensor.

MEMBER HARRINGTON: Okay, okay. So you just get elevation readings at discrete points? Okay.

MR. MEYER: That's correct.

MEMBER HARRINGTON: Okay.

MEMBER PALMTAG: This is Scott Palmtag. Oh, sorry. I'm sorry, Craig. Go ahead.

MEMBER HARRINGTON: I'm done, Scott.

MEMBER PALMTAG: Okay. This is Scott Palmtag. Slightly off topic, but kind of since we're talking about the containment and instrumentation, I have some questions about the valves that are used inside. I think it's RVV and RPV. Can you explain that?

MR. BECK: You're asking about the containment isolation valves?

MEMBER PALMTAG: No, the -- sorry. It's later in the slides.

MR. BECK: Oh, you're talking about the

ECCS valves?

MEMBER PALMTAG: Yes.

MR. BECK: There will be a slide --

MEMBER PALMTAG: Those are inside the --

MR. BECK: Yes, those will be discussed in a few slides.

MEMBER PALMTAG: Okay. Thank you.

MR. BECK: Next slide.

CHAIR KIRCHNER: This is Walt Kirchner. Going back to Craig's question about the PARs. So when you do a refueling operation, the containment isn't entirely flooded. You keep the PAR -- the PAR location is high in the containment and it is not immersed in water?

MR. BECK: That's correct.

CHAIR KIRCHNER: Okay. And then Craig, what I remember from the DCA was that they were considering for the level measurements a radar kind of based system rather than this -- I think they call it a dispersion type sensor now. So that was a significant change in the SDA design to my knowledge.

MEMBER HARRINGTON: Thanks, Walt.

MR. BECK: Next slide. Continuing on containment changes for containment isolation. A significant change from the DCA is the addition of a containment isolation test fixture valve. It's between the CNV nozzle safe-end and the containment isolation valve body.

This is an enhancement to improve the ability to perform Appendix J testing whereas the DC design included first of a kind testing features that are integrated into the CIV body. The containment isolation valves are welded directly to that CITF. And the CITF is welded directly to the nozzle safe-end.

And you can see that depicted on a picture in the right side where CITF is. There also was a change in the closure time of CIVs which was changed from 7 seconds to 10 seconds now in the SDA.

MEMBER HARRINGTON: So this is Craig Harrington again. On the CCW line isolation states that two CIVs instead of just the one that would be required for GDC 56, the particular issue that drove the decision to go conservative?

MR. BECK: I'm not sure of that. If anyone is on the call that's aware of the RCCW CIVs. However, I do believe that kind of standard CIVs across the penetrations where we can. And so that might be the reason why.

MR. LASSITER: Tyler, this is Dan

Lassiter, NuScale, design engineering. The impetus to have the dual isolation values on the closed lines is so that we don't have to design the piping and components of that line to -- as a containment boundary particular ASME Class 2. So we design the piping and components inside containment with lower pressure boundary integrity requirements and just rely, excuse me, on the containment isolation valves themselves.

MEMBER HARRINGTON: Just standard design tradeoffs. Okay. And just one thing that I noticed, it just seemed like a big of an inconsistency between -- this is in Section 6.2.4.2.2, component design. And it talked about the SSCIVs.

The tech says it allows for maintenance repair and replacement. Those same words aren't there for the PSCIVs. I'm just curious if that was just an oversight, just an inconsistency in words, or if there was some other issue.

MR. BECK: I would imagine that is just an oversight, an inconsistency.

MEMBER HARRINGTON: Okay, thanks.

MR. BECK: Next slide. The last slide for Section 6.2. For the containment response analysis, it was previously presented as part of the methodologies previously presented as part of the LOCA evaluation model topical report. So the discussion today is just really going to talk about the implementation.

For initial conditions, those have been changed from the DCA to align with the new design. An example of that would be ultimate heat sink pool level exchanged. However, it's not really that.

There is a similar amount of stored energy compared to the US600. Because of our operating containment design, there is significantly more design margin, particularly with pressure. And you can see on the right side peak cases.

And so the primary events peak pressure is similar between the two designs. But because of the operating containment vessel design rating, there's more margin. And you can see that peak temperature is also comparable.

MEMBER MARTIN: Question, this is Bob. So what drove the increase in design in pressure? Anything -- 10 percent is, I guess, template guidance. And before, I guess you were kind of just not quite 10 percent there.

But you definitely are. And then you were 10 percent even with respect to the old design pressure. Now you've gone to 1,200. What drove that change?

MR. BECK: More margin was the goal certainly. And we're over 20 percent now. So a little above the SRP guidance of 10 percent, and I think that was a significant part of the decision.

MEMBER MARTIN: Okay. What margin? The cost take a little bit more. But you decided that the safety margin was more valuable to you than, say, the cost of the vessel itself. Another thing that's obvious here is your secondary event peak pressure where all the other ones kind of look more or less what you'd expect. That one is doubled. So that has implied that the event has changed or the design. What drove that?

MR. BECK: And we'll get to the --

MEMBER MARTIN: To the extent that you can talk about it.

MR. BECK: And we'll get to the ECCS changes in the next slide, I think, maybe. But it's the removal of the IABs. And so you've have a main steam line break and a coincident ECCS actuation. And so whereas in the DCA design, the valves had the IABs. And so now you have that value actuation with the main steam line break and thus form --

MEMBER MARTIN: How to get it twice.

MR. BECK: And then the last bullet points on this slide, so there's previous COL item in the DCA for containment leakage rate testing. That COL item has been removed simply because those requirements are already specified the requirement, that's already required and specified. So that was removed.

And we've also removed a COL item that related to containment vessel volume. And that is because there is now an ITAAC that confirms that parameter. And for Section 6.2, there's extensive audit. And we had 17 audit items and 4 RAIs resolved in this section. Next slide.

Section 6.3 is the emergency core cooling system. For ECCS changes versus the ECCS valve changes, and several changes here related to safety analysis optimization, some of which you have already or you have already heard about. So there are two vent valves from three in the DCA.

And that change is made coincident with the ultimate heat sink pool level change. The vent values do not include inadvertent actuation block valves. And so now the vent valves open upon ECCS actuation.

The IABs are still on the recirculation

valves. However, the threshold and release pressures are lower than the SDA design. And there are integral venturis to the reactor recirculation vales and reactor vent valves that limit flow during high differential pressure conditions.

And that entry change is made to decouple the function -- the flow limiting function of the valve internals. And now the venturi performs that function. For other ECCS changes, there previously was one trip solenoid valve per ECCS main valve in the DCA design.

And now in the SDA, there are two in series trip solenoid valves per ECCS main valve. And then the last bullet on the screen is related to actuation signals. So there was -- the DCA, there was a high CNV level and low RCS pressure ECCS actuation signals.

And those have been removed. And instead, now there is a low and low-low RPV riser level actuation signal. Additionally, there are now high-high RCS pressure and high-high RCS Tave ECCS actuation setpoints or beyond design basis events.

MEMBER HARRINGTON: This is Craig again. Knowing in the review, the ACRS review of the US600 design, there were -- I guess there was ongoing testing of these valves at the time. I guess it's completed now.

But one of the concerns was just the complexity of the valve system and making sure that it would be reliable in adding another trip solenoid valve maybe helps with inadvertent actuation. But makes it more complex again. Has that all been thought through? I'm sure it has been thought through.

MR. BECK: For our Chapter 15 analysis of that we do to the periodicity of failures and that sort of thing. And that is included in our design and safety analysis.

MEMBER HARRINGTON: So return to that --

MR. LASSITER: This is Dan Lassiter, NuScale, design engineering. Just to comment briefly on the test programs, there was a test program executed specifically for the purpose of DCA review. And that use representative of components and demonstrated that the valve performed all safety functions with the representative arrangement of parts as you said as a valve system.

Between DCA and SDA, we've also executed a fully prototypic valve test program, NTS, Huntsville. I think it changed names now again. But that covered really the whole scope of what we intend to qualify the valve for.

There will be some additional testing in the future or qualification testing to meet all ASME QME-1 qualification requirements. But the valve performed all its safety functions up to full pressure/temperature conditions. So we have high confidence in the ability of the ECCS valve to perform its safety function.

MEMBER HARRINGTON: Okay. Thanks. I appreciate that.

MEMBER ROBERTS: This is Tom Roberts. I had two questions on inadvertent actuation of these ECCS valves. One is there was a 2019 SECY document that talked about the potential IAB valve to not swing shut during the accident.

And as I understood from that SECY was inadvertent actuation of a valve during the event. With your change now, two of the valves -- the vent valves don't have IABs at all. Is the concern identified in that document still in effect?

Because that seemed like a reliant with the IABs to prevent basically anything if they don't exist in those valves. So there's a whole debate about single failure criteria and how they would apply. And is any of that discussion relevant? Or is it all covered, I imagine, with the design change and the IABs?

MR. BECK: Do we have the analysis from hydraulics group?

MR. CUMMINGS: I can address that. So this is Kris Cummings, NuScale. I've been with NuScale for about 5 years, 25 years of experience in the industry working on fuel safety analysis and spent fuel issues.

So in particular, we still have the IAB on the RRV. So that's still applicable. And that SECY was germane to whether that component was essentially single failure criterion needed to apply to that.

So the Commission decided that was not the case. So we still apply that aspect of it to the safety analysis. But we don't apply single failure. So it's basically determined that it's very similar to a check valve. So from that perspective of the SECY-19-0036, that part still applies.

MEMBER ROBERTS: But what's the scenario that currently exists with the new design where you would need to take account for that exception? Is there a scenario where it matters?

MR. CUMMINGS: Yeah, I think I'd have to

ask Meghan to chime in on the safety analysis side of that or Devon, if you're on.

MS. McCLOSKEY: This is Meghan McCloskey from NuScale safety analysis. The IAB single failure itself has a -- during the scope of events that we analyze in Chapter 15, the IAB single failure has relatively little impact on our progression now with the exception of what scope of initiating events we need to consider. In the DCA design, we evaluated the inadvertent opening of one ECCS valve, either one vent valve or one recirculation valve as the inadvertent -- as the initiating event.

And then we apply the deterministic Chapter 15 criteria of single failures and loss of power scenarios. And so what the IAB SECY meant to us in the DCA space was that it was not necessary to evaluate the simultaneous opening of two valves because one was the initiating event. We assumed a loss of DC power supply, again, a very deterministic assumption.

And then an IAB single failure, we only needed to cover the single event and the loss of power supply. In the SDA design, we evaluate the scope of events that are appropriate for this ECCS valve system. So we cover inadvertent opening of a single value.

We also address an inadvertent ECCS actuation signal that results in two vent valves opening simultaneously. And we cover those scenarios with and without power available. So one of our limiting cases is evaluating an inadvertent recirc valve opening event with loss of DC power. And that results in three valves opening simultaneously. The IAB SECY continues to apply in that it's not necessary to evaluate all four valves opening simultaneously in that scenario.

MEMBER ROBERTS: Okay. Thank you. I think I understand. Would it make a difference? Three valves seems like you've got most of the system already in actuation. Is there a benefit to having the fourth valve assumed to not open?

MS. McCLOSKEY: There's still a bit of a benefit in terms of how much of a flow in -- a core flow in reduction. It has an MCH-4 margin that we're evaluating.

MEMBER ROBERTS: Yeah, okay. Thank you. That's helpful. The other question is related. It has to do with the inadvertent opening of a reactor vent valve.

We had a discussion last month in a closed session about how your safety analysis doesn't assume a complete loss of all the DC power coincident with an unrelated event. And the reasoning was the closed head. But I think the fact that you have that assumption is certainly not closed.

That was just specific to the loss of DC power which would cause two RVVs to open spuriously. But that's not the only way you get a spurious opening of an RVV, right? You have other ways you could do it. Is spurious opening of one RVV a problem or is it just two that's a problem in the analysis?

MS. McCLOSKEY: Spurious opening of one vent valve is part of the design basis initiating event scope. And we take the -- we evaluate the loss of DC power coincident with the initiating event. And as we -- I think as we discussed in the last meeting or a couple before that, in the Chapter 15 design basis space, we don't stack initiating events on top of each other, so to speak.

And so we take the valve opening as an initiating event or we evaluate an inadvertent ECCS actuation. But we -- it's not necessary to assume that happens randomly in the middle of some other event like a reactivity insertion event or a cool down event. That's beyond the scope of the design basis event.

MEMBER ROBERTS: Right, I understood that. That was specific to the loss of DC power. I'm thinking about other ways to get an actuation of a reactor vent valve which would potentially be a single failure. You would need to assume it in conjunction with this unrelated reactor condition event.

One scenario that comes to mind is if you've got the two trip valves, right, for each RVV. But I think in a failure in one trip valve as was discussed in the Chapter 16 review, there's no requirement in tech specs to limit operation. So if you had one trip valve that was open, you could still continue to operate the plant which now you'd be portable to a single failure of the other trip valve causing inadvertent actuation of that RVV. Is that a scenario that would be of concern?

And if it would be a concern, is it something you would need to consider as a single failure? Or is that wrapped up in the 2019 SECY, very unlikely single failure basis? That's something I didn't see discussed is other ways to get a single RVV to trip spuriously. And that's one that occurred to me. There may be others. I don't know.

MS. McCLOSKEY: Right, right, sure. But

that would be a random failure that's not impacted by the initiating event or the event progression. And so when we consider the single failures, we're considering for the active system components, it's failure to -- it's really related to failure to actuate upon demand.

The electrical system components can have latent failures. And that's why the module protection systems are designed to accommodate that with the logic. But in the case of a solenoid for the ECCS valve, if the ECCS valves are not being demanded by the initiating event, it would have to be a random failure that occurs. And that's outside the design basis event progression as well.

MEMBER ROBERTS: I would've thought that a single failure criteria would have you assume that there's an unrelated failure occurring in the protection system, either active or passive. The passive failure exception for 10 CFR 50 is only for fluent systems or mechanical systems, not for electrical systems. So it would seem like you would need to include that as a potential single failure.

I guess I'm wondering why you wouldn't. Now it's the combination of the solenoid being pre-failed and this single failure of the control system is very unlikely. I see you can probably make a similar argument about likelihood.

I don't know if you were making that argument or the argument you're making is that this is not a valid unrelated signal failure you assume occurs coincident with the casualty of the event. I'm just trying to understand which it is. And the general question, I suppose, is, is there a requirement to look at the vulnerability to inadvertent trip of the ECCS valves given this linkage to unrelated transient events?

MR. GRIFFITH: So this is Thomas Griffith, licensing at NuScale. So let's make sure I understand what you're saying clearly is that for the RVVs, you have two solenoids that need to de-energize in order to cause an actuation. And when I hear the concern is if one of the solenoids is out of service and is in the open position meaning that you are one solenoid away from potentially having an inadvertent operation of a relief valve. Is that the setup scenario?

MEMBER ROBERTS: Yes.

MR. GRIFFITH: Okay. So I would expect that an operating plant evaluates and takes control of that situation using the correct batch in process as well as maintenance role. And the online risk program would have to evaluate the time that it's acceptable to stay in that position. And in my view, this is no different than if a half scram was to come in on a plant or a half actuation of a safety system. So obviously, there's some period of time that would be required for an operating plant to evaluate the condition, assess risk, and take appropriate corrective actions commensurate with the safety significance.

MEMBER ROBERTS: Yeah, I think the distinction -- you can tell me if I'm wrong. A half scram is in the safe direction. So you're closer to losing continuity of power which is obviously something the plant wants to avoid because you want to keep running.

But for inadvertent actuation of the ECCS valve, there's a potential safety implication that if you have this unrelated reactivity initiated event and you were single failure would be the actuation of the other solenoid. Then that would now compound the event to the extent that you would see your CHF limits. Now you discussed last month that's also -- there's margin in your CHF analysis. Then there's other arguments you can make. I was trying to understand just what the line was in terms of how you parse single failure and the unrelated event.

MR. GRIFFITH: I think Chapter 15 already analyzes the RRVs spuriously opening with a concurrent loss of DC power. In the event that the solenoid that was out of service was on the RRV, you have the IAB that still exists. And given that the RVVs don't have the IAB and you assume a loss of DC power, that event is analyzed in the design basis.

MEMBER ROBERTS: Yeah, the event I'm asking about is the unrelated reactivity addition. So that one, the discussion we had last month is you, I think justifiably, are assuming that you don't lose your reliable DC power system coincident with that event because that's a redundant system and there's reasons why it's reasonable to not assume loss. So I'm asking about other ways to inadvertently operate the RVV during this unrelated reactivity initiated event and whether you thought through what the requirement is to reasonably prevent them and whether that's a constraint that ought to be covered in Chapter 6.

MR. GRIFFITH: I think our discussion is that a loss of DC power de-energizes the solenoids and results in the RVV's opening. Whether or not an RVV solenoid is out of service or not, the failure that's being discussed there is a loss of DC power. So the total number of solenoids, if one is already in the fail safe position, there's less solenoids to have to move to the safe position. The safe position for ECCS is in the open position.

MEMBER ROBERTS: Right. But there's more scenarios once you got a solenoid out of service that causes inadvertent actuation of the RVV.

(Simultaneous speaking.)

MEMBER ROBERTS: Lots of things beyond loss of DC power that can cause the other solenoid to trip and then the RVV to open.

MR. GRIFFITH: So I think I agree with you that the likelihood of an event that results in that RVV to open because there's only one of two solenoids in service is an accurate statement because there's only one solenoid remaining. However, I would argue that is appropriately managed under the online risk program by the licensee.

MEMBER ROBERTS: Right. Probably the continuity of operation perspective.

MR. GRIFFITH: Correct.

MEMBER ROBERTS: Not from the -- you now have either an assumed or a stated assumption in the safety analysis that an event valve not opened coincident with a reactivity addition event. There is a distinction there, I think.

MR. GRIFFITH: Yeah, and I think that what Chapter 15 specifies specifically is that the IORV event needs to be less -- the inadvertent actuation of ECCS needs to have a frequency of less than once per module lifetime. And so you would have to evaluate an operability evaluation whether or not you're in conformance of your licensing basis and for how long that service could take place. But that is not -- that would be a more complex evaluation based upon whatever failure occurred.

MEMBER ROBERTS: Yeah, okay. I think I'll ask the staff for their view when they come up. Again, I don't know that I'm concerned about the combined likelihood of these because I think they're extremely low. I was trying to, again, understand what your threshold was for what single failures you would still assume in your safety analysis for these things like the reactivity initiated events where the loss of DC power I think we've discussed at length and it's reasonable.

The system is very reliable and it's redundant. And it's nonsafety which is almost a term as opposed to a real distinction for the reliability of that system. But for these other scenarios that cause inadvertent actuation, I'm not sure that same argument applies. And that's what I'm trying to understand.

MR. GRIFFITH: Yeah, so I think I'd like to point out that failure of ECCS to actuate properly contributes significantly to the overall CDF of the plant. And I think that that'll be discussed as part of Chapter 19 is that over -- and Jim, you can correct me here. But roughly 90 percent of the core damage events is due to the ECCS not actuating properly. And I think, Kevin, I think you -- Kevin Lynn, if you're on the line, you had something you want to join in here?

MR. LYNN: Yeah, this is Kevin Lynn of licensing. I'd just like to add I think one of the things is when it comes to our design to keep in mind is that actuation of ECCS is a safe -- is the safe position. So it's similar to Tom's analogy with putting the reactor at an operating plant, putting it in half trip.

So you're essentially putting ECCS in a half trip situation here in your postulating scenario where on solenoid valve is out of service. So you're halfway to ECCS actuation. And in our plant, ECCS actuation puts us in a safe state.

So there's not a concern about going to a safe state. The issue is as you've raised it is about how do you interpret that in terms of the Chapter 15 assumptions. So NuScale's position is when we talk about single failures to apply, if you look at the history of the discussion, it's always applied in terms of applying the single failures to things that mitigate the event.

So for example, at an operating plant, if you need diesel to start to mitigate that event while you single fail one of the diesels because that hurts you. In our case if we're talking about a reactivity insertion event, you don't need ECCS to actuate. So applying a single failure to ECCS doesn't make sense.

ECCS is a separate system not being relied upon for that particular event. So what we're doing is we're saying we're not going to take a single failure that initiates a different event during an unrelated event when there's no reason to assume so. So I think that's the key is when you apply the single failure, you don't apply it to unrelated systems. You apply it to mitigated systems. And for these events, like a reactivity assertion event, ECCS is not a mitigated system.

MEMBER ROBERTS: Okay. Yeah, I understand the argument. I'm kind of puzzled about the active versus passive failure aspect of it. But I'll think about that. So thank you.

MEMBER HARRINGTON: One other question on the IAB. Clear the block at 450 psid differential pressure. And at that point, I don't remember there being a discussion on this. But it seems like obviously you would then be flowing water out the RRV instead of flowing back in. Is that a problem or why pick 450 and not some lower differential?

MR. BECK: I think there are manufacturing of procurement reasons for the thresholds picked for the valve. But we do analyze the 450 psid. And that released pressure on the recirculation valves.

MEMBER HARRINGTON: From an event progression standpoint, losing the water out the RRV at that stage doesn't --

MR. BECK: In the DCA, I would say that the release pressure was 900. So it is reduced in this design by a significant amount.

MEMBER HARRINGTON: Okay, okay.

MR. BECK: Next slide. For the ECCS changes, there is now an ECCS supplemental boron or ESB feature. That includes boron hoppers, condensate channels, dissolvers, and mixing tubes. And there's a schematic FSAR on the right side.

You can see the hoppers, dissolvers, and mixing tubes. And on the next slide, there's another picture that's a little bit more detailed of the FSAR. It shows more detail to the ESB system. I'll show that in a second.

But for the last two points for Section 6.3, there's also an added 8-hour ECCS actuation timer following reactor trip in the SDA design. That timer did not exist in the DCA design. And there was a significant audit during the test review. There's 14 audit items and 5 RAIs.

MEMBER HARRINGTON: It's Craig again. I think toward the end of the chapter found discussions somewhere about some event that might occur during operation caused condensation and impacts on the dissolver, the contents of the dissolver, the boron oxide. I guess those kinds of events during operation really force you into a situation of having to get back into the module, reassess the status of the pellets and the dissolver, clean all that up before you can go back into operation. Okay?

MR. BECK: Yeah, that's correct. If the pellets are wetted, we'll have to probably replace the pellets and certainly evaluate the pellets.

CHAIR KIRCHNER: So Tyler, this is Walt Kirchner. Following up on Craig, so the hoppers in a refueling operation wouldn't be immersed.

MR. BECK: That's correct.

CHAIR KIRCHNER: Yeah. But they would be in a moisture environment. So the water level basically in the revised design for the SDAA, where would the water level be under normal operations in the reactor building, about the hopper level, below it? I think it would be below, right?

MR. BECK: Yes. And I don't remember off the top of my head what specific elevation the hopper is at.

CHAIR KIRCHNER: Just a couple of questions then.

MR. BECK: But it is below.

CHAIR KIRCHNER: Yeah. So that hopper is going to see there's quite a temperature differential inside the containment vessel above the water level line and below. Have you looked at the environmental qualification of the hopper? It's probably going to see pretty -- well, it's going to be likely seeing, pardon me, something close to the steam temperature because it's in a vacuum during normal operation.

So it's probably seeing something like that. So is there any concern that this whole mass in the hopper would solidify in some way that it wouldn't dissolve once you -- on demand when you had the actual scenario of a steam environment that you're relying on to essentially release that boron into your dissolver and then further down? Have you looked at the environmental qualification of the system?

MR. BECK: Yes. But first I'll say so the hopper is loaded during refueling. And it's above the pool level. And so that's -- you load it with the pellets initially. And then once you're starting up and you drained out a containment vessel, the pellets are released and they actually fall into the dissolver baskets. And so that's where the pellets are during operation, the dissolver baskets. For --

CHAIR KIRCHNER: Okay.

MR. BECK: -- the question on temperature and the conditions, yes, the pellets and ESB are included in the environmental qualification program. They are qualified for that environment.

CHAIR KIRCHNER: Thank you.

MR. BECK: Next slide. And this is tough to see on the slide format. But this is from the FSAR Figure 6.3-5. And it just shows a bit more detail of the ESB feature.

And so on the left, you can see the dissolver, the dissolver baskets. And on the right, you can see the system as a whole. So that includes the hoppers and then the pellets falling into the dissolvers.

You can see the associated main and auxiliary condensate panels. They allow condensation while the pellets do dissolve accordingly and then the lower containment mixing tubes. That was the last slide for Section 6.3.

This is Section 6.4, control room habitability. For changes for the DCA, relatively minor changes. The first is that there's a ten-minute delay that is added. So when you have loss of battery chargers, CRHS actuates.

However, now in the SDA design, there's a ten-minute delay. That just allows the operators time to try and figure out what's going on. There was also previously toxic gas detection that's included directly in the scope of the design.

Now it's in the scope of COL Item 6.4-1. There was a COL item that required testing and inspection requirements for CRHS to be specified. That was removed simply because those testing and inspection requirements are already specified elsewhere throughout the FSAR.

And then on the note of toxic gas detection, during the audit and RAI days, there was a clarification. And we revised one of our initial test program tests and that COL item. It clarified the scope of toxic gas detection and control room habitability. Next slide.

Section 6.5 is fission product removal and control systems. And it is essentially unchanged from the DCA. Next slide. Section 6.6 is the last section we'll discuss today.

And it says inservice inspection and testing of Class 2 and 3 components. There aren't any significant changes from DCA. So the design still satisfies the relevant 50.55a requirements and allows for the optional Reg Guide 1.147 code cases.

We did remove a COL items that required specifying -- I think it's related to inservice testing for Class 2 and 3 components. And that's because inservice testing program is described in Section 3.9.6. And that's it for Chapter 6.

MEMBER MARTIN: Okay. Members, any other questions before we transfer to the staff?

MEMBER HARRINGTON: Just one other quick

question. This is Craig. And I don't know if you want to respond now or in closed session. But the conditioning of the studs, reactor vessel studs, containment studs is nontrivial activity. And this design has to be done remotely. Describe a little bit about how that's going to occur.

MR. BECK: Do we have -- I don't know that we have for fueling handling on the call right now or anyone in the group can address it. If not, we may want to defer this question in closed session.

MEMBER HARRINGTON: Okay.

MR. CUMMINGS: I mean, I'll just say -- Kris Cummings again, NuScale. Yeah, you're right at a high level. We do have some details to that. But that's not a level of scope that we include in the SDA because it's not a safety-related activity, right?

I mean, you do fuel handling and things like that and that sort of stuff is covered in the SDA. We certainly have considered that. But that's just not content that's included in the SDA.

MEMBER HARRINGTON: Okay.

MEMBER PALMTAG: Yeah, this is Scott Palmtag again. Can you go back to slide 10? I do have some questions about the reactor recirculation valves. I just have some -- can you just first -- and you may want to defer this until the closed session. But can you tell me how far the reactor recirculation valve is above the core?

MR. BECK: I'm not sure on the specific change, but it's on top of the reactor pressure vessel. Oh, you said the recirc valve. Sorry.

MEMBER PALMTAG: Yeah.

MR. BECK: We probably would have to get that in the closed session.

MEMBER PALMTAG: Okay. And I assume these are electrically actuated solenoids?

MR. BECK: Yes.

MEMBER PALMTAG: And how do you run the instrumentation? Where does that go? I mean, is that in a pipe --

MR. BECK: The solenoids --

MEMBER PALMTAG: -- that runs up the side?

MR. BECK: The solenoids are actually technically outside of the containment vessel. And there's hydraulic lines between the main valves, so, for example, the vent valves and those solenoid valves. And so there's no associated electronics in the containment vessel.

MEMBER PALMTAG: Okay. That makes more sense. It doesn't show up on the diagram. So all the solenoids --

MR. BECK: There's a --

MEMBER PALMTAG: -- are outside of the containment vessel?

MR. BECK: Yeah, and there's a figure in 6.3 that shows a schematic of the trip to reset valve assembly, I think. And that assembly is a containment penetration. So the valves are located technically outside of the vessel.

MEMBER PALMTAG: Okay. Thank you.

MEMBER MARTIN: All right. Any other questions, members online? Not hearing any, let's make a quick switch. I mean, just we're going to pause here for a second and then move right into with the staff's presentation. Thank you.

(Pause.)

MR. SNODDERLY: Bob?

MEMBER MARTIN: Yes.

MR. SNODDERLY: Just for the interested members of the audience and participants, even though it looks like on the schedule we're following the plan on time, we need to keep in mind right now we do not have the presentation from either the staff or NuScale, who will be there to answer questions. It's okay to take more time in open session, but we're covering some material we're covering. You have the power to stop for a break. I think that's the only way.

(Pause.)

MEMBER MARTIN: Okay. Everyone is seated. Who's going to kick us off?

MR. TESFAYE: Yes, thank you. Good morning. Let me just get the slides.

(Pause.)

MR. TESFAYE: Okay. Good morning. Again, my name is Getachew Tesfaye. I'm the lead project -- oh, can you hear me now? Good morning. My name is Getachew Tesfaye. I'm the lead project manager for the NuScale U.S. standard design approval, US460.

I work for the Chapter 6 PM. We start our presentation with Chapter 6. A quick overview, as we've been saying for a while, NuScale submitted Chapter 6, engineered safety feature, Revision 0 of the SDAA FSAR on December 31st, 2022 and Revision 1 on October 31, 2023. And the safety evaluation is based on Revision 1.

NRC developed audit of Chapter 6 was performed from March of 2023 through August of 2023, generating 46 audit issues. Questions raised during the audit were resolved within the audit. Six RAIs were issued and the response were all acceptable. And the response are also documented.

Staff completed Chapter 6 review and issued an advanced safety evaluation to support today's ACRS subcommittee meeting. The first draft sent to you was on January 18 and then the final draft -- not the final but the final draft was submitted last week officially in a memo. Several NRR staff participated in reviews.

And today we're going to concentrate on those chapters or sections that have a significant change from the DCA. There are seven sections. 6.7 doesn't apply for NuScale's PWR section. But the rest of them are in our safety evaluation.

The report and most of the change that we're going to be discussing here today will be in 6.1, 6.2, and 6.3. With that, I'll pass the mic over to Robert Davis, Bob Davis who's online to present significant changes, 6.1.1, engineered safety feature materials. Bob, are you ready?

MR. DAVIS: Yes, I'm here. Can you hear me?

MR. TESFAYE: Yes.

MR. DAVIS: Okay. So the biggest difference in 6.1.1 is the change of the containment vessel to -- or the use of code case N774 which allows the use of F6NM, martensitic stainless steel for the containment vessel. And this is allowed via code case N774 which is listed in Reg Guide 1.84 Revision 39 as permitted for use without conditions. And this applies to the upper containment vessel and a portion of the lower containment vessel below the upper/lower vessel flange. Next slide.

Okay. So this material is very different from typical materials that we use in PWRs. The applicant has considered the effective welding procedures, one, the martensite start temperature, the martensite finish temperature. Like I said, this is very different from typical materials that we deal with.

The applicant will not follow recommended preheat temperatures listed in the nonmandatory Appendix D of Section 3. And the applicant is employing an extensive testing program to determine the appropriate pre-temperature to prevent hydrogen cracking while at the same time promoting martensite formation during welding. Next slide, please. Okay. So welding F6NM requires special considerations in addition to ASME code requirements.

And welding processes that employ flux may require post weld heat treatment times greater than those specified in ASME code due to the pickup of oxygen from flux welding processes which may require post weld heat treat times greater than those specified in the code. So typically, when welding procedures are developed for standard vessel materials, they're post weld heat treated for a very, very long time because to account for repairs and things like that. And so you're worried about the length of the post weld heat treatment whereas in this case for -- and so the standard times listed in the code are more than adequate to get the appropriate toughness.

You're worried about post weld heat treating something too long to where you can decrease the tensile properties below what's required by code. However, with this material, using flux welding -- flux processes, we're worried that if you qualify a welding procedure for, say, 20 hours and then you weld something that the code requires, say, a 3-hour post weld heat treatment that the impact properties may not be adequate if you post weld heat treat it for 3 or 4 hours. You may need much longer times.

So the applicant has addressed this by -- has addressed this in their application. I guess part of the presentation that we'll give later on today as to how they do that. Next slide, please. ASME code specifies that post weld heat treat temperatures for F6NM is 1050 to 1150.

However, the lower critical temperature of a 410 nickel-moly type weld filler metals which is what the applicant uses and F6NM-based materials can be as low as 1150. So if you have variances in your post weld heat treatment which it's impossible to get the exact temperature, you could actually if you were post weld heat treating at the higher end of what's required by code, you could actually be going into -- going beyond the lower critical temperature which, of course, would cause the formation of martensite and not the tempering of martensite. However, the applicant has agreed to modify the application to state that their temperature will be 1075 plus or minus 25 degrees.

And of course, this provides a margin to ensure that they do not reach the AC-1 temperature. And so the staff has determined that the additional control considerations placed on the fabrication of the F6NM are adequate. And our ultimately conclusion for the 6.1.1 did not change from the last design. And I think that's my last slide unless there's another one.

MR. TESFAYE: If there are no questions for Bob, we'll go to Syed.

MR. HAIDER: Thank you. Good morning. My name is Syed Haider. I'm from NRR Division of Safety Systems, Nuclear Systems Performance Branch. Today, I present a high-level summary of the design changes in NuScale SDAA FSAR Sections 6.2.1 and 6.2.2, and they are mainly related to evaluating the NPM-20 containment design application for the NuScale SDAA for the peak contaminant pressure and temperature during a design basis event involving mass energy release from the reactor pressure vessel into the containment during a primary or secondary systems pipe break or an anticipated operational occurrence, or AOO.

This slide has the most significant design changes on the NPM-160, for the DCA NPM-20 of the SDAA FSAR Section 6.2.1 on containment functional design and Section 6.2.2 on containment heat removal systems.

The staff review established the consistency and conservatism of the modified design parameters with the SDAA Technical Specifications and also verified that all design changes are properly implemented in the Applicant's engineering applied model for the containment response analysis through various initial and voluntary action.

And now I'll go over some of the major design changes NuScale made in the SDAA with respect to the containment thermal hydraulics. In the NPM-20 module for the LTAA, the reactor thermal power has been increased by about 56 percent compared to that of NPM-160. And the containment upper vessel material has been changed from SA-508 to SA-336, while the lower containment vessel material is still the same as SA-965. This reduces the thermal conductivity of the upper part of the containment by about 35 percent, while wall thicknesses have somewhat changed.

The initial reactor pool water temperature has been lowered from 65 feet for NPM-160 to 32 feet in the NPM-20 Tech Specs. The staff found the change to be conservative, as it would reduce the heat transfer from the containment to the reactor pool, and thereby, leading to a higher peak containment pressure and temperature.

It's worth mentioning that containment analysis credits only the pool water inventory available in a single day around the NPM for the ultimate heat sink for the analysis, but not the entire pool. And this is conservative.

The initial reactor pool temperature has been increased from 110 degrees Fahrenheit that was used in the DCA analysis to 140 degrees Fahrenheit in the SDAA analysis, even though the Tech Spec values for the pool temperature is increased to 120 degrees Fahrenheit, which is conservative as 150 degrees Fahrenheit would further suppress the containment heat removal to the pool.

The 140 degrees Fahrenheit initial pool temperature leads to around 150 degrees Fahrenheit initial containment wall, such as temperature below the pool level under normal operation, the steady-state operation. However, the initial containment water temperature above the pool has been significantly increased from 240 degrees Fahrenheit from the DCA to 500 degrees Fahrenheit in the SDAA, based on the results of a 3D FEM analysis for NPM-20.

The containment analyses also assume the outer surface of containment head and the wall above the pool level as adiabatic, which the staff found to be conservative.

The number of RRVs, or reactor regulation valves, located on the top of the reactor pressure vessel has been reduced from three to two in the NPM-20. In NPM-160, the inadvertent actuation blocks, or

IABs, were used on RRVs, as well as RVVs, while in the NPM-20 design, IABs are used only with RRVs and open in the IAB design criteria based on the differential release pressure. As RVVs do not have IABs anymore, they can openly indicate the loss of AC and DC power. In the SDAA, IAB release pressure has also been reduced from 950 psid nominal to 450 psid nominal. That will typically delay the activation of IAB.

Now, NPM-20 design uses venturi nozzles on all the RVV and RRV lines, while the NPM-160 design does not have any venturi nozzles; it, rather, had orifices. In the NPM-160 containment safety analysis, DHRS heat exchanger operation was not credited to the containment design basis, even during mitigation, but in NPM-20 it is credited. Even though there are two single failure-proof safety-related DHRS cranes, the staff has mandated a 50 percent NRELAP5 fouling factor penalty to both sides of the DHRS heat exchanger tubes in the DHRS model for peak containment pressure and temperature calculations, as an indication and condition for using the NPM-20 containment response analysis methodology. The limitation and condition is documented in the LOCA Topical Report SER.

And all the containment internal design pressure for the NPM-20 has increased from 1050 psia to 1200 psia, and the containment design pressure has increased from 550 psia to 600 psia. These increases have had even higher containment design modules for the SDAA.

Next slide, please. This slide summarizes some additional important changes from the DCA to the SDAA applicable to Section 6.2.1 and 6.2.2 that are worth underscoring.

First off, the containment response analysis methodology, or CRAM, for the DCA was documented in a standalone Technical Report that was incorporated by reference in the DCA. However, the CRAM methodology, as modified for the SDAA containment design for NPM-20 is now included in the LOCA Topical Report that has been presented to the ACRS Subcommittee meeting on January 15 as being acceptable to the staff.

Another significant change is the inclusion of a one-time containment free volume ITAAC in the SDAA to verify that the as-built containment free volume bounds the minimum value of 6,000 cubic feet used in the Chapter 6 containment design basis analysis and its validation to the ITAAC did provide an overall indication that the containment has been built as designed.

Containment free volume is a key important parameter to be verified by the ITAAC because it needs the various key parameters introduced and underscored in several 14.3 SRP sections.

With NPM-20 being a standard design module, and free volume being a best feature not subject to significant changes from module to module, the staff found it acceptable that this ITAAC will be performed for the first module ever built and not for subsequent adopters of the SDAA. And the specified design control process will, rather, be used to maintain the containment free volume in accordance with the design.

It is worth emphasizing here that, unchanged from the DCA, the SDAA also includes a separate, but related ITAAC to verify the passive heat sink parameters for the as-built NPM-20 containment vessel structure that includes the containment walls and linings by evaluating the heat sink materials of this area, thicknesses, and properties that have been relied upon in the containment safety analysis. So these two ITAACs are closely related.

As previously mentioned, the DHRS is not credited to the containment design basis event mitigation for the SDAA. While it was not credited to the DCA containment DBEs, now with a 56 percent higher decay heat for NPM-20, and, apparently, insufficient reactor coolant pool normalization around DHRS and containment, the staff looked closely into the reactor coolant pool heatup and thermal stratification due to their potential for DHRS and containment heat removal performance degradation and the resulting impact on the containment LOCA response.

MEMBER MARTIN: Syed, for some of us that haven't been on the Committee so long, could you give a little bit of the backstory on why maybe the DHRS was not credited previously. It's a passive system, right?

MR. HAIDER: It's a passive system, yes.

MEMBER MARTIN: Right. In the DCA, was that the way NuScale came in, basically, not crediting? Because they didn't need to --

MR. HAIDER: Yes.

MEMBER MARTIN: -- or that was their position? It was kind of a defense-in-depth-type system, and now, of course, with their passive system, there's no reason not to? Can you fill in the backstory?

MR. HAIDER: Yes, that's a fair and correct characterization. I mean, the simple truth is that they did not need crediting DHRS.

MEMBER MARTIN: Okay.

MR. HAIDER: Even the limiting containment design basis accidents, either in the small-break LOCA --

MEMBER MARTIN: Right, right, right.

MR. HAIDER: -- large-break LOCA regime. But now, being an eastern margin, and also the core, the entire spectrum from large-break LOCA to small-break LOCA, they had to credit.

MEMBER MARTIN: Okay. Thank you.

MR. HAIDER: And in the same vein, the staff also deep dived into the sensitivity of the containment response break size and ECCS actuation, as with the uncertainty in modeling natural convection heat transfer.

NuScale provided additional LOCA spectrum analysis results -- coming to your point -- results going down from 100 percent large-break LOCA to 2 percent small-break LOCA regime for the discharge line, as well as high point vent line breaks to cover both the liquid -- break LOCA and also the reference break LOCA.

The submitted results showed that the peak containment pressure and temperature are not very sensitive to the DHRS performance degradation caused by the pool heatup. RAI-10359, the response also provided justifications for modeling the pool heatup around the containment, thermal stratification, and natural conduction heat transfer modeling. Now, there is no open item outstanding in the Chapter 6 FSAR Sections 6.2.1 and 6.2.2.

Now, the table at the bottom of this slide captures the changes in the limiting containment pressure and temperature design basis event from the DCA to the SDAA, as well as the corresponding results.

In the DCA, an inadvertent RRV opening was the containment pressure design basis event that had led to a peak calculated containment pressure of 994 psia that had about a 5-percent margin with respect to the containment design basis pressure of 1050 psia for the SDAA.

While in the SDAA an RCS design line break LOCA is different in the pressure design basis event that led to a peak calculated containment pressure of 957 psia. That has about a 32 percent margin with respect to the modified containment design pressure of 1200 psia.

The same RCS discharge line break LOCA also happens to be the containment temperature design basis event for the SDAA that led to a maximum containment temperature of 533 degrees Fahrenheit, which is 67 degrees Fahrenheit lower than the containment design temperature of 600 degrees Fahrenheit for the SDAA.

Previously, an RCS injection line break LOCA was the containment temperature design basis event for the DCA that led to a maximum containment temperature of 526 degrees Fahrenheit, which was 44 degrees Fahrenheit lower than the containment design temperature of 550 degrees Fahrenheit for the SDAA.

Anyway, in summary, both the peak calculated pressure and temperature have not changed much from the DCA in the SDAA, but significant increases in the containment design pressure and design temperature have led to higher containment pressure and temperature margins.

Next slide, please. So I have entered this slide to show the comparison between the staff confirmatory analysis and also the Applicant's analysis. So basically this slide is showing that the staff's -- that the Applicant's analysis is conservative.

MEMBER MARTIN: I've got to jump on this one.

MR. HAIDER: Sure.

MEMBER MARTIN: The phenomena that we're looking at here is pretty straightforward, right? You have two bottles, concentric. Nothing opens up. You know, a pathway opens up between the two and you're moving energy from one to the other.

I would not expect a 100 degree psi difference in the plot you're showing here on the left. Have you investigated that? I mean, are you using a best-estimate-type approach? Or what are the differences that result in that 100 degree -- 100 psi, I'm sorry?

MR. HAIDER: Yes, you are right. I mean, we deeply investigated this. This is -- the green curve, the 20-year for our confirmatory analysis, and the blue curve is from MELCOR, while the green curve is from NRELAP, and, yes, there's about 100 psi difference. And we spent a lot of time reconciling the geometry and going over the differences and made sure that there is no sensitivity that we could conduct, and we did not conduct, to identify exactly where the differences were coming from.

And that's why we've also have done a case confirmatory analysis. As you see, the peak containment pressure here, on the left, is about 957 psia, while MELCOR is predicting about 812. And we conducted the same exercise using TRACE, aligning the conservatism, the models, the initial conditions, the boundary conditions, everything in TRACE, and the TRACE was around 870 psia.

MEMBER MARTIN: So we're in the middle?

MR. HAIDER: Yes, so TRACE was somewhere in the middle. So from these results, we can conclude clearly that the Applicant's analysis is very conservative. And we have gone through evaluating all the phenomenologies like the effect of non-condensable on condensation heat transfer and the decay heat, and also the critical flow models. But we were not able to pinpoint where exactly the differences are coming from.

But the TRACE was also about 50 pounds below where the Applicant is. So considering that we were getting the evaluation of the results, the validation of the conservatism of the Applicant with our two different independent models, and also, considering the time, we did as much investigation as we could.

MEMBER MARTIN: I guess my expectation would be, since MELCOR, you know, has been more of a severe accident containment code, going back to at least the containment models, going back to the CONTAIN code once upon a time, maybe it's the heat transfer package related to containment heat transfer is maybe a little more sophisticated or more accurate. Whereas, codes like RELAP5-3D was not developed with that in mind, and not knowing what's actually in TRACE, although I know there's some similarity, a lot of similarities between TRACE and RELAP5, it's likely that the containment heat removal heat transfer package, whether it's condensation or whatever, is maybe just not as vetted.

But what I think that this doesn't say is that, more than likely, I think I would have more confidence in the MELCOR prediction of containment response, given its history and validation. And then you're certainly confirming significant margin, whether NuScale has quantified it or not, but it gives us a lot of confidence.

MR. HAIDER: But I would like to also add one more piece of information that I believe is relevant in this context. In the DCA, the peak containment pressure was predicted by RELAP at about 994. And we literally used a very similar containment volume model using MELCOR in the DCA stage. And it predicted around 986.

MEMBER MARTIN: Oh, so you were much closer?

MR. HAIDER: So we were very close. And the same model was modified in RELAP5 for SDAA, using the modified initial conditions, and literally, they are using the same condensation model, the same models for critical flow and the same model for decay heat. But, yes, the pressure came out far below. But TRACE is also --

MEMBER MARTIN: In the ballpark?

MR. HAIDER: It is in the ballpark.

MEMBER MARTIN: That's interesting to me, and maybe the more significant thing is just the level of the reactor pool, a larger condensation area. I don't know. That's interesting that they would be so different.

But, anyway, I won't belabor that one. Thank you.

MR. TESFAYE: So now, I think you can (audio interference) forward. So this slide essentially summarizes the SER conclusions for all subsections of Sections 6.2.1 and 6.2.2. The staff concludes that the containment safety analyses have appeared to be moderate.

All relevant physical phenomena in the

NPM-20 containment response, that includes condensation heat transfer, the degrading impact of non-condensable gas on condensing heat transfer, decay heat, choked flow, DHRS and ECCS sensitivities, and containment taking more of the area of the pool.

The staff review of NuScale's SDAA FSAR Chapter 6 has shown that the NuScale containment design incorporates sufficient conservatism in the NPM-20 containment model through initial and bounding conditions and appropriate constitutive models.

The staff also concludes that the SDAA FSAR has provided sufficient description of the spectrum of primary and secondary design basis events and acceptable results for the limiting mass energy released into the containment and the resulting containment pressure and temperature responses.

In summary, the NuScale containment design for the SDAA meets all regulatory requirements and acceptable criteria for the containment safety design. This concludes my presentation. Thanks for the time for presenting the staff's review. I would like to know if the Committee would have any other questions about the staff's review of SDAA Sections 6.2.1 and 6.2.2.

MEMBER MARTIN: Anyone in the room or

online?

I'm not hearing any. Thank you.

MR. TESFAYE: Anne-Marie?

MS. GRADY: Good morning.

MEMBER MARTIN: You might want to come a little closer to the microphone. You're kind of soft-spoken. Pull the microphone closer to you, please.

MS. GRADY: I've never been accused of that before.

One more time. Good morning. My name is Anne-Marie Grady, and I'm a severe accident analyst, and also I reviewed the design of combustible gas control, actually both for DCA and SDAA. And there are some changes in combustible gas control which are summarized on the slide in front of you.

The first one is the applicable regulation. The DCA applied 10 CFR 50.44(c), which is for new reactors. The SDAA decided that the appropriate applicability was 10 CFR 50.44(d), which is for reactors of new design that hadn't been envisioned when the combustible gas control regulation was issued.

(C) is a much more prescriptive regulation. SDAA, by its very nature, is less so. The guidance that's applicable for DCA is SRP 625, which is combustible gas control, and 19.0, which is severe accident. The guidance that's applicable for the SDAA, however, is a little bit different. It's Reg Guide 1.7, combustible gas control, and again, SRP 19.0.

The combustible gas control design is based on combustion analysis, so the -- I'm sorry. The DCA design was based on combustion analysis. Now, NuScale analyzed combustion in containment. They analyzed the transition to detonation, DDT. They also analyzed detonation and they proved that the containment integrity was protected via the analysis. There was no PAR in that design.

The SDAA, however, has changed their approach to showing that the containment would retain its integrity. And they show that by adding a PAR to the design and showing that it maintains that the containment atmosphere is always inert. In some instances, it's natural inert. For example, during normal operation, it's almost a complete vacuum.

There are no combustible conditions in the containment then, but there are other design basis accidents to consider. There are severe accidents to consider and there's also long-term radiolysis. Various stages could be considered in evaluating the combustible gas control. So now, we have a PAR, a single one, safety-related.

Okay. The safety category is --

MEMBER ROBERTS: Hey, Anne-Marie, yes, Tom Roberts. May I ask a question now?

The safety evaluation addressed at least took partial credit for the fact that it continued to do the combustible analysis. And we heard from the Applicant that they did that as part of the PRA, and that that may not have been to the same level of quality as a design analysis.

Can you comment on the role of the safety analysis of the combustible gas, please?

MS. GRADY: Well, first of all, I could say that the statement that's in the SER, in 6.25, was in the section that was talking about PDC-41. That sentence, while it's correct, is appropriately addressed in Chapter 19 for the severe accident, and it doesn't support the discussion on the PDC-41.

And I heard NuScale's description of why they did that combustible analysis and it was for severe accident analysis and the PRA.

MEMBER ROBERTS: In looking at the graph, I see there's a section called Structural Analysis Containment Integrity. And it goes on to talk about the analysis that they did with the bounding hydrogen and oxygen mix, similar to the NPM-160.

MS. GRADY: Are you talking about 19.2428?

MEMBER ROBERTS: It's hard to find the section number here.

MS. GRADY: In Chapter 19.

MEMBER ROBERTS: Yes. The top of page 101 in Chapter 6. It says during the regulatory audit, the staff reviewed the NuScale evaluation and agreed with the conclusions, which is about the structural capability of the containment, assuming a combustion event. And it says the staff agrees with this and concludes that the NPM design meets the required criteria in Reg Guide 1.7, Section C.(5).

That paragraph kind of confused me because the argument seemed to be that the PAR maintained the environment inert, but it seemed like the staff acceptable was at least partially based on the containment calculation, assuming the combustion happened.

MS. GRADY: The AICC analysis was done of Chapter 19 for severe accident to show that it was not, that combustion was not going to threaten the containment integrity. It doesn't belong in Chapter 6, SER. It was in there inadvertently with PDC-41, and the sentence, since you have read it, has been taken out of Chapter 6.25. It's still appropriate in Chapter 19.2.

MEMBER ROBERTS: Okay. Thank you. So what I just read will be removed from the draft?

MS. GRADY: That sentence that you -- yes, absolutely.

MEMBER ROBERTS: Okay. Thank you.

MS. GRADY: You're welcome.

MEMBER MARTIN: Anne-Marie, just a question. It's been a while since I've analyzed, done analysis related to PARs. You used the word safety-related PARs. I would say, 15 years ago, that wasn't a thing, right? Well, at least in my experience. Is there something different about design of PARs today that distinguishes them as safety-related versus a non-safety-related PAR?

MS. GRADY: I can't speak about PAR manufacturers marketing a product. What I can say is, when NuScale agreed that it would be a safety-related PAR, they also agreed that there would be significant more testing in the design specification and it would be a specific design specification; that they would have more inspection. It would be an ITAAC and there would be a Tech Spec on the PAR. So it really gave us a much fuller confidence in how it was going to be designed; how it was going to be analyzed, and how it was going to be installed and operated.

MEMBER MARTIN: Oh.

MS. GRADY: So that's almost a --

MEMBER MARTIN: So it might be the same product, but they might have been available when I last looked at them. But it's the testing and the monitoring Tech Spec; it's all the other layers that control --

MS. GRADY: It's the design specification --

MEMBER MARTIN: Okay.

MS. GRADY: -- does meet the conditions that we were concerned about, yes.

MEMBER MARTIN: Okay.

MS. GRADY: It probably is indistinguishable from off-the-shelf --

MEMBER MARTIN: Right, right. Okay. Thanks.

MS. GRADY: As we just said, the safety category, there was no PAR in the DCA and there is now a single safety-related PAR in the SDAA.

There is now an ITAAC in the SDAA; whereas, there wasn't one in the DCA. Actually, there are three ITAACs. There was one that verifies the physical arrangement and the installation. There is one that specifies the analysis and the testing, and the test of the recombination rate of the PAR, because there is a minimum recombination rate. And there's an ITAAC on the fact that the PAR would be part of the cube.

MEMBER MARTIN: Okay.

MS. GRADY: There were no Tech Specs because there was a PAR in the DCA, but there is a Tech Spec on PAR operability now. And that really involves that the PAR would be inspected during every refueling, and physically inspected. The PAR would be tested in general and in a sampling process to make sure that the recombination rates are still being maintained from refueling to refueling. And the PAR will be reinstalled if it has to be moved, and I don't believe it has to be moved, but if it does, back in the same location that it was always intended to be. So there are Tech Specs and there are ITAACs.

MEMBER HARRINGTON: So this is Craig Harrington. I'm confused. In 6.25.1 of the FSAR version that I looked at, it says, the design includes a passive autocatalytic recombiner PAR that is non-safety-related, Seismic Class II, of the new requirements.

MS. GRADY: That's Rev 1. Our Rev 2 would say that it's safety-related.

MEMBER HARRINGTON: Okay. Okay.

MS. GRADY: Yes, it absolutely is safety-related and that has been a change.

MEMBER HARRINGTON: Okay. All right. Thanks. That clears the confusion.

MS. GRADY: A combustible --

MEMBER DIMITRIJEVIC: Sorry, this is Vesna, Vesna Dimitrijevic. But the PAR is supported with the augmented DC system, right?

MS. GRADY: I'm sorry?

MEMBER DIMITRIJEVIC: Which is not safety-related, right?

MS. GRADY: I'm sorry, Vesna, the PAR is safety-related. So what was the first part of your statement again?

MEMBER DIMITRIJEVIC: Okay. My question is, does it -- it requires DC, an EDAS system, right, for operation?

MS. GRADY: No, no, no. No, Vesna, it's passive.

MEMBER DIMITRIJEVIC: Oh, okay.

MS. GRADY: It's essentially an open

chimney with some catalytic plates at the bottom, and the combustible gases come up past the plates, recombine, and express steam out the top. It doesn't require any electrical signal or any electrical supply or any other supporting systems. It's passive.

MEMBER DIMITRIJEVIC: All right. I thought it requires a signal?

MS. GRADY: No.

MEMBER DIMITRIJEVIC: I mean, that's my misunderstanding. All right. Okay.

MR. BECK: This is Tyler Beck with NuScale.

I'll just clarify that Anne-Marie is correct; it is a fully passive component. It's a passive catalyst that serves for the recombination reaction of hydrogen and oxygen.

MEMBER MARTIN: I think one thing that's maybe unique -- again, because NuScale's containment design or whole design is unique -- is that that PAR is going to be exposed to rather high temperatures. And, of course, in an earlier slide, or your slide, but earlier in the presentation here, it noted boundary conditions, assumed analyses, and one of them being a containment surface above the water level is like 500 degree F. That would be significantly higher than, say, what PARs have probably been considered in the past, right? Because in large containments the environment is typically that it would be below 120.

So there would be some unique EQ-type work that would need to be done in this particular case. So that, I guess, would go to the extra work NuScale would be otherwise expected to do in their testing. And has that work happened or is it ongoing? Whether they're working with a manufacturer or fabricator, or whatever we call them today, to move in that direction; to have that all complete, say, by the time we get approval? Or is that just ITAAC?

MS. GRADY: The design specification would indicate the conditions in the containment the PAR would see under all the different conditions. I can't speak specifically to a temperature, but I know NuScale has the intent of having in the design specification a maximum temperature.

But even more interesting, as far as I'm concerned, with respect to the PAR, is the fact that it's inside containment. It's inside a very small containment. It's relatively close to the reactor vessel, and it's going to see high neutron irradiation during normal operation, and that's something that the PARs off the shelf today don't necessarily -- don't provide.

MEMBER MARTIN: Well, certainly --

MS. GRADY: They're not exposed to such conditions. But that is in the NuScale design specification.

MEMBER MARTIN: Right. But will the testing and all that be resolved before, say, approval? That would be just pushed to an ITAAC? Is that the intent of NuScale?

MS. GRADY: I think NuScale would have to answer that.

MEMBER MARTIN: Yes.

MR. BECK: Tyler Beck with NuScale again. So one thing is we've specified environmental qualification as a requirement for the PAR, and there's an associated ITAAC with that. And so that would -- it would need to be qualified prior to completing ITAAC and the 52.103g finding.

MEMBER MARTIN: Okay. Has that been initiated, or is that just kind of --

MR. BECK: I can't speak right now for the engineering procurement process on that.

MEMBER MARTIN: All right. Thank you.

MS. GRADY: All right. There was a Technical Report which is, basically, a combustion and containment report for the DCA. There is no such report comparable to combustion for the SDAA. However, there's a limiting -- sorry. There are several calculations, several analyses on different aspects of design basis accidents, severe accidents, long-term radiolysis, that are in the electronic reading room and they have been proprietary, but they are there. And they have been reviewed by us, meaning me and others.

There is an exemption request also. There was in the DCA and it really resulted in -- it was for hydrogen and oxygen monitoring in the containment during an accident to be able to inform the operators of whether or not a severe accident had taken place and how it was progressing. How much hydrogen had been generated in the containment would be a measurement of how much core damage would be there. And there was a requirement, there is a requirement to have that kind of monitoring.

In the DCA, NuScale came up with a, more or less, uncertain means of post-accident monitoring of hydrogen and oxygen. It was a conceptual design, and that's all I can say about that.

Now, in the SDAA, the exemption request wants to have no hydrogen and oxygen monitoring in the containment post-accident. What they want to rely on, instead, is, first of all, they have a PAR. The PAR can take and operate and can recombine and prevent combustible mixture from occurring, whether it's the DBA, whether it's the severe accident, whether it's long-term, days and weeks down the road, and from long-term radiolysis. So the PAR is maintaining the containment indirect under all the different circumstances it's likely to see. So the PAR is doing that.

In addition to that, the operators will be able to rely on radiation monitors under the bioshield, and also exothermocouples to give them some indications of the severity of the accident in containment. So the exemption request is to have no monitoring of hydrogen and oxygen in this design, and we've recommended that.

Next slide, please. Okay. The acceptability of applying 50.44(d) as the applicable regulation for combustible gas control in the SDAA, we reviewed that also, because that was a change in the application.

The CNV is not inert. However, the CNV is not inert in the presence of hydrogen of -- less than 4 percent oxygen in the presence of hydrogen during a design basis accident in the first 24 hours of a non-core-damaged AOO. So in other words, when there is a specific design basis accident in the containment, the CNV is not inert.

And CFR 50.44(c) applies mainly to severe accidents. 10 CFR 50.44(d)(2) applies to the safety impacts of combustible gases during design basis and significant beyond-design-basis accidents. And for those reasons, we believe 50.44(d) is applicable and we agreed with the change.

Do you have any other questions? All right.

Combustible gas control conclusion. During a core-damaged DBA, the PAR is credited to maintain an inert containment. Post-accident, post-severe-accident, the CNV remains inert without crediting the PAR. During long-term radiolysis, PAR is credited to maintain an inert CNV.

In the exemption request, the post-accident monitoring of hydrogen and oxygen are not required to assess core damage. The assessment is going to be accomplished, as I've just said, by the core reg's thermocouples and the radiation monitors beneath the bioshield.

As far as combustible gas control is

concerned, are there any other questions?

MEMBER HARRINGTON: Well, this is Craig. I guess that non-LOCA event where in the short term maybe you're not quite inert, it kind of feels like a technicality there why that might be okay, but is it just happening too fast at that point for the PAR to keep up or?

MS. GRADY: Initially, during normal operation, there's a vacuum. If you have that non-core-damaged DBA LOCA, in other words, when the ECCS timer opens the relief valve. There is almost immediately, because of the materials that are released from the RCS, almost immediately a combustible mixture in the containment. You haven't had core damage, but you have hydrogen and oxygen that will support combustion. NuScale's analysis shows that and they show that they need to address that as a design basis accident, and our confirmatory calcs confirm that as well.

A PAR is needed for that very specific, but non-core-damaged LOCA; whereas, if you had a core-damaged LOCA, there would be so much more hydrogen going in there, it would suppress the oxygen and there wouldn't be -- it's almost better from that aspect, anyway.

So the DBA requires a PAR. That's the non-core-damaged DBA. However, if you didn't have that, we can talk about that separately, and these three bullets have separate analyses that NuScale has done and that we have confirmed in ours.

In the severe accident, because there now has been core damage, in fact, significant core damage, now you have sufficient hydrogen certainly, but you also have oxygen, but you don't have enough to ever exceed 4 percent oxygen, because there's so much hydrogen in there. So the hydrogen is almost keeping the containment indirect after a severe accident.

Long-term radiolysis, there's no more hydrogen generated from the core damage. However, the PAR is credited because there's long-term radiolysis taking place. NuScale has done a calculation and looked at what happens long term. And around 37 days, there could be a combustible mixture again, but the PAR is in there. It's always in there. It's always -- I can't say operational; that's odd -- but it will do its job.

So that's why there are separate bullets here.

MEMBER HARRINGTON: So on the previous slide, the top bullet that says it's not inert, that's not credited PAR?

MS. GRADY: If you don't credit the -- yes. I'm sorry.

MEMBER HARRINGTON: Okay. Okay.

MS. GRADY: Yes, absolutely.

MEMBER HARRINGTON: All right. Fine.

MS. GRADY: That's one of the reasons the PAR is in there.

MEMBER HARRINGTON: Yes. Thank you.

MEMBER MARTIN: Anne-Marie, you made a point about the advantage of this small containment; it's the proximity of the PAR to the vessel, and that, of course, would probably improve its performance or at least your uncertainties related to performance, because everything is really tight in there.

One possible failure mode, not knowing anything else, is that that proximity -- is there a possibility that there's a jet impingement scenario where, okay, the opening of the RVVs in some way directs the coolant towards the PAR? And that has -- okay, you're nodding your head. So they have obviously thought about it and maybe mitigated that possibility?

MS. GRADY: Yes, there are two points to make about that.

There's an ASME AG Code that requires that, if you comply with that Code -- and NuScale intends to comply with the Code -- that you would have to take into account the jet impingement loads on the PAR as part of the qualification of the PAR. That's No. 1.

And No. 2, they have that condition, jet impingement, as one of the conditions in their design spec.

MEMBER MARTIN: Okay. I mean, one solution would be just to kind of direct the flow away from the PAR. Do they --

MS. GRADY: I don't know about alternate solutions. I just know they're going to address it, so that the PAR is designed for that.

MEMBER MARTIN: Okay. Is that considered an ITAAC?

MR. BECK: Yes, so the environmental qualification is a piece of that. It is that associated ITAAC.

But we have looked at it for the closest possible position of the PAR on the vessel to the vent valves and RSVs. The jet loads are not significant.

MEMBER MARTIN: All right. Thank you. I assume there's no more questions in the room. Is there anyone online, any member or consultant online with a followup question before we move on?

I'm not hearing any. Move on.

MR. WIDREVITZ: Hello. This is Dan Widrevitz. I can quickly address Section 6.2.7 for actual prevention of the containment vessel.

Here we have a significant difference between the NuScale DCA and the NuScale SDAA FSAR, which is primarily that they're using F6NM to replace SA-508, Grade 3, Class 2, from previous designs, with the upper CNV and a portion of the lower CNV below the upper lower vessel flange.

This is, of course, interesting because, when you have heard the word Martensite, and you think of the word pressure toughness, they don't usually go together, but this is a pretty tough Martensite.

The staff verified that the material change would not result impacts to the fracture toughness management for the CNV, particularly if you have to meet ASME Code fracture toughness requirements, which you are quite capable of doing with this material. And therefore, the staff conclusion did not change from the DCA.

Any questions?

MEMBER BALLINGER: The proper word is

tempered Martensite.

MR. WIDREVITZ: Yes.

MEMBER MARTIN: I guess if there are no questions, go on to the next slide.

MR. NOLAN: This is Ryan Nolan, like the baseball player, but backwards.

I'm in the Nuclear Methods and Systems for New Reactors Branch, and I was one of the reviewers for Section 6.3. NuScale covered most of these changes, so I'll go through fairly quickly.

So one significant change is that they added the supplemental boron feature. If you recall the DCA, they did have an exemption to GDC-27. So one condition of the system, they are now complying with GDC-27.

The staff's evaluation to that particular criterion is performed as part of Chapter 4, which I believe you'll see in April.

There is an Extended Passive Cooling Topical Report which provides the methodology for this system. And so I believe that will be presented next month, and then the evaluation of the system is performed as part of Chapter 15. Again, it will be presented in April.

One thing the staff did ensure was that

this system is tested as part of the initial test program or ITAAC. This is a new system. And so we did verify that a first-of-a-kind test does exist to test the system in an integrated fashion to verify that they're getting the dissolution rates in the mixing as expected in the analysis.

Another change was the removal of the IAD on the vent valves, as well as the reduction of number of vent valves from three down to two.

In order to sort of compensate for this, as well as other design changes, NuScale had added flow-restricting venturis into the RVVs and the RRVs. This raised an interesting question as to whether the design could mitigate a break at the flange versus just an inadvertent opening of a valve, which was the main focus of the DCA and the staff's review of that particular design. This particular question will be addressed as part of Chapter 15 and was the subject of a high-impact technical issue, and we'll certainly discuss that in more detail in April.

They did change the ECCS actuation signals. In the DCA, it was containment parameter-based, and for the SDAA, they went to more direct measurement of a mixture level. This was discussed briefly when the staff presented the LOCA Topical

Report.

And then lastly, they did add an eight-hour ECCS timer, which will actuate ECCS eight hours after an automatic or manual reactor trip. This does two things. One is it ensures that the ESB is utilized when needed to maintain subcriticality, and as well, vent any combustible gases due to radiolysis.

And so Anne-Marie had already presented on the analysis we looked at. But, basically, when we looked at the long-term radiolysis development, we ensured that NuScale's calculations showed that any combustible mixture within the RCS does not occur within the eight-hour timeframe. And so you hit the eight-hour timer. Everything vents into containment.

We also did our own confirmatory calculation and we had results in the same ballpark or magnitude as NuScale.

MEMBER ROBERTS: Hey, Ryan, I want to ask you about the single failure assessment of inadvertent actuation of the RVV during an unrelated event. There is some discussion in the SE about the loss of EDAS, the DC power system, which we have talked about in previous meetings.

But we talked earlier this morning about the single failure criterion for cases other than loss of EDAS that would lead to an inadvertent actuation of an RVV during an unrelated event.

The purpose of the single failure criterion from NuScale is that you wouldn't have to assume that. I'm just wondering what your view is. Is that something that you would not assume because it's not directly part of the protected action in response to the reactivity addition event? Or is there concern that it is a single failure that would cause something that impedes the ability to show protection for that event?

MR. NOLAN: Yes, so when it comes to interpreting the single failure criteria, a lot of our guidance lives in policy space. And so one of the best sources of information is SECY-77-0439. That was the agency's first attempt at distinguishing various single failures.

And so if you look at that SECY paper, you know, we sort of break it out into mechanical and electrical components. So in this case, in 6.3, we're focused on the valves as a mechanical component. It's an active component, right? It requires movement to perform its function. And so it's a single active -- or it's an active component, subject to the single failure criterion.

And so when we look at how we apply single failures to mechanical systems, we typically look at it at two times, either at event initiation, you know, is it subject to single failure, as well as on-demand. And so I think if we're talking about two unrelated events, if the valve inadvertently opens at time zero, well, you've terminated the event and you have an IO or EE analysis, or in Chapter 15, you have the results for that.

If you're looking at having to use the reactivity insertion scenario, ECCS isn't demanded at any time during that until after the reactor is tripped and the event is terminated. And so we don't necessarily see a single failure consideration for that particular scenario that wasn't already addressed in our Chapter 15.

MEMBER ROBERTS: But the loss of EDAS was considered? And then that was thrown out because of the redundancy in the EDAS system. But that was considered? Even though you could make the same argument for loss of EDAS, there's nothing -- you know, loss of EDAS is a safe action for the reactivity addition event, because it causes the scram independent of the rest of the system. And yet, there was still the consideration.

MR. NOLAN: Yes, and --

MEMBER ROBERTS: So, I mean, it's kind of an interesting question on the single failure criterion that really hadn't occurred to me until the point from NuScale this morning, that you would actually parse the single failures that you would consider based on whether or not they're part of the system that you, in effect, actuate, as opposed to they are systems, and if they were to cause -- if the single failure were to cause an actuation, they would take away the ability of the system to protect the reactor.

It seems like they're the same thing. In my mind, they would get the same concern, which is, if there's some likelihood of a single failure in systems, and if they either prevent or protect the system from actuating at all, or if they cause the plant conditions to change, such as the protective system can't protect, it seems like in either case you would need to consider that single failure scenario. But I'm just wondering if you've got any thoughts on that.

MR. NOLAN: Yes, and I think we're going to -- it's probably unsatisfying to say, but a lot of this discussion will probably occur during Chapter 15, 15.0. The staff is still working to finalize the engineering evaluation and controls associated with the EDAS. And so we'll have all of our documentation prepared for April for that.

But when we're talking about EDAS and the HITI that was raised, it was more of a classification issue the staff had. It wasn't really a single failure issue. Because when we look at how you apply single failures, we apply single failures to safety-related systems, right? The safety-related systems are those systems that are mitigating Chapter 15 events.

MR. BARRETT: Right. This is Antonio of the staff. Yes, so those two trip valves are in series. They're both safety-related.

MEMBER ROBERTS: Speak up.

MR. BARRETT: Yes, my name is Antonio Barrett of the NRC staff.

So, yes, those two valves are in series. They've both safety-related. So if one was to fail, you would still have the other one. So that's how we are thinking about that and it has all the protections --

MEMBER ROBERTS: Right, and I agree with that. But the question came up during the Tech Spec discussion that there's no plant restrictions if one of those were to fail, and then be open for, presumably, as long as the plant would be willing to live with (audio interference) away from ECCS --

(Simultaneous speaking.)

MR. BARRETT: Sure. Correct.

MEMBER ROBERTS: Which I agree. We heard this morning there are some concerns there. You need to have a low probability of an actuation. So the plant would be acting on that --

MR. BARRETT: Yes.

MEMBER ROBERTS: -- but the Tech Spec would not preclude that, which would then put you in the single failure space, I would think, because that's now -- a lot of it is allowable.

MR. BARRETT: You're 100 percent correct. We will probably address that later on when we get to the Chapter 15 section. But you're 100 percent correct.

But, generally, if you have two safety-related pieces of equipment and one of them fails, that's your single failure. You have those other considerations which are 100 percent accurate. So we'll be talking about that. Okay?

MEMBER ROBERTS: Great. Yes, I'm willing

to have all the questions in Chapter 15. So thank you very much.

MR. NOLAN: Yes. Yes, there's certainly a symbiotic relationship between the Chapter 15 safety analysis and the Technical Specifications, right?

We would, typically, perform the Chapter 15 analysis. We would use the operability requirements to determine what systems are there, right, that they have operability requirements to mitigate in the system. So those two definitely play a large role together.

MEMBER MARTIN: Okay. That's your last slide, correct?

MR. NOLAN: This is the last slide. If there are no questions, I'll turn it back to Getachew.

MR. TESFAYE: Yes, that concludes -- excuse me. This is Getachew Tesfaye again. That concludes the Chapter 6 presentation.

MEMBER MARTIN: Okay. Any last questions related to Chapter 6 in the room or online?

I'm not hearing any. I think it's time for a break. So I'll say maybe a 20-minute break?

MR. SNODDERLY: To 11:15?

MEMBER MARTIN: 11:15? I don't think we're going to get through that part of the open before lunch. It would be appropriate. But that's okay.

MR. SNODDERLY: Yes, as I said, I'm not --

MEMBER MARTIN: We can continue after lunch with some open, and then --

MR. SNODDERLY: That's right.

MEMBER MARTIN: Okay. Then that's what we'll do. So we will recess until 11:15.

(Whereupon, the above-entitled matter went off the record at 10:54 a.m. and resumed at 11:15 a.m.)

CHAIR KIRCHNER: We're back with now NuScale, who has more to discuss, I guess with just one slide, on Section 17.4, and then the bulk of it will relate to Chapter 19. So, Sarah, you're nodding the most, so who's going to lead us off?

MS. BRISTOL: Ultimately Pete Shaw will be the presenter and he will be the presenter, and he'll be online, so --

CHAIR KIRCHNER: Okay.

MS. BRISTOL: -- he'll start on 17.4. Pete?

MR. SHAW: Hi, good morning. My name is Peter Shaw. I just want to double-check that my mic's coming through?

MEMBER MARTIN: It is.

MR. SHAW: Okay. So once again, my name is Peter Shaw. I'm a NuScale licensing engineer. I've been in the industry for over 15 years now. And prior to my tenure at NuScale here I worked for 10 years at the Vogtle 3 and 4 construction project.

I'm going to be starting this next run of presentations with, as said, the slide for Section 17.4.

Next slide, please? So 17.4 is the Reliability Assurance Program. As in the DCA, the Design Reliability Assurance Program reviews and approves safety and risk classification for the NuScale SSC. For the US460 the evaluations were completed.

The D-RAP panel expert insights resulted to changes in some methodology for the panel insights, but without design changes. These include the steam generator tubes as safety-related, not risk-significant components, as well as the control rod drive mechanisms, safety-related, not risk-significant.

There were 10 audit items that were resolved. These resulted in updates to Section 8.2 and the Figure 17.4-1 to clarify the SSC classification process. And there was a single RAI. This was a clarification -- this resulted in a clarification for the section that the process does not assume a risk significance based on safety-related classification. And it also resulted in clarifications in FSAR 17.4.3.2 and the role of the backup diesel generators in Table 19.1-56.

Another note of a Revision 2 change that -- from Revision 1 to Revision 2 is the top support structure for the containment vessel was added as a risk-significant component given that it is the connection between the containment and the crane, both of which are risk-significant components. And also the secondary side for the CVCS valves and the pressurizer spray valve were removed as risk-significant components.

MEMBER MARTIN: Peter, this is Bob Martin. Earlier there was a number of questions related to what would be the safety-related PAR. Did you explicitly address that in the D-RAP?

MR. SHAW: Yes, the D-RAP process reviewed the classification of the PAR as it was presented to them by the responsible system engineers. And in review of the PAR, as stated before, given the significance between both design-basis and beyond-design-basis it is classified as a safety-related non-risk-significant component.

MEMBER MARTIN: Okay. That's was what I was looking for. Thanks.

MR. SHAW: Yes. Okay. Without further questions, I will turn it over to Jim Schneider.

MR. SCHNEIDER: Thank you, Pete.

Next slide, please?

Good morning. My name is Jim Schneider and along with Peter I'll be presenting Chapter 19, the application. I've been with NuScale licensing for three years, and prior to that I spent 20 years in operations at Braidwood Station where I was licensed as both a reactor operator and (Audio interference.)

Next slide, please? So Chapter 19 covers the PRA and severe accident evaluation. You see the different sections there up on the slide. During the two years of the staff's review in Chapter 19 there were 156 audit issues resolved in the audit including 84 document requests. The majority of those audit issues and document requests were in 19.1, 19.2 related to the PRA. There were many discussions on crosscutting issues in Chapter 19. I think it made for a risk-informed review of the application. And then after the audit phase we have 15 RAI questions resolved in Chapter 19.

I just want to note in the Chapter 19 presentation if you see an asterisk, that represents information that was added to Rev. 2 of the SDAA. So the committee hasn't had an opportunity to see that yet, but we wanted to point that out. It might be a change what you've read.

Next slide?

MEMBER MARTIN: I'm just curious, 84 documents is an awful lot of documents.

MR. SCHNEIDER: It is.

MEMBER MARTIN: Are these calculations? Are these --

MR. SCHNEIDER: There were a lot of reports, PRA notebooks. There were calculations. I'm not sure how else to --

MS. BRISTOL: We'll say the underlying technical basis of --

MR. SCHNEIDER: Yes.

MEMBER MARTIN: Okay.

MS. BRISTOL: Yes, like all of the various notebooks.

MEMBER MARTIN: Thanks.

MR. SCHNEIDER: Okay. I'm going to start with 19.1, the PRA. And I wanted to start with just an overview of the PRA, why we have one in our application and just to give some context for when we talk about NuScale's PRA numbers.

So we have a PRA In the application because it's required by regulation. And as the design progresses from the time it's just an idea on paper to when it's built and producing power for customers there is a PRA required at each phase of a plant's development. And that is a living PRA. It evolves with the plant design. And so the PRA in the US460 SDAA has evolved as the design has evolved from the US600 DCA. And then in the SDAA we do have COL items that ensure that an applicant will have the proper PRA in each of those phases as it moves towards construction and operation.

At this phase of the design, the design phase, the purposes of the PRA in general include to evaluate the overall safety of the plant design and provide insights into that -- the design for improvements of the design. And as a reminder, the Commission's safety goals for all nuclear plants are a core damage frequency of less than 1.0E-4 each reactor year and a large release frequency of less than 1.0E-6 each reactor year, which leads into the next slide, please?

DR. SCHULTZ: Jim, before you go to the next slide, this is Steve Schultz --

MR. SCHNEIDER: Yes?

DR. SCHULTZ: -- just a comment or a question associated with the general overview you've just provided. As you've indicated each licensee is going to have the requirement to have a PRA and will be using it throughout operation. Are you expecting that each licensee is going to develop their own PRA? Is there going to be a common approach taken by NuScale licensees associated with PRA? What do you envision?

MR. SCHNEIDER: Each licensee will be responsible for their own PRA. I'm not sure if there's any -- no, right now, I mean, we don't have any plans we can share with -- I think you're talking about sort of the owner's group, I think.

DR. SCHULTZ: Yes, I am.

MR. SCHNEIDER: As far as I know unless anyone wants to chime in, that's not in the works for now. I think we're too early in the development.

DR. SCHULTZ: It seems like it would be both prudent and also extremely efficient and useful given the new design and the potential applications to many licensees, but just perhaps a comment for now.

Thank you.

MR. SCHNEIDER: Okay. Thank you.

MEMBER HARRINGTON: And this is Craig to follow up with that. I assume the PRA would equally apply to all six modules in the facility?

MR. SCHNEIDER: From what I understand -- and, Sarah, you may know more -- each module will have its own PRA. Is that correct?

MS. BRISTOL: This is Sarah Bristol, manager of the PRA Team. NuScale potentially has services opportunities currently for the SDAA. We've got a single module PRA. And so it is pretty -- it's equivalent to Module 1 or Module 2. It's indifferent of module, but it a single module PRA. And then we'll also take into account multi-module effects. And so we also do expand that into -- and as you'll see on the next slide just insights from multi-module potential, but it truly is single module PRA.

MEMBER HARRINGTON: And then the site operator would just have to either have separate PRAs for each module or somehow manage any differences that might develop during (Audio interference.)

MS. BRISTOL: Yes, that is true. They would have to take that -- but as of now all of the modules are the same, consistent, and so there's no need for modifications at this point (Audio interference.) Yes, thank you.

MR. SCHNEIDER: Okay. Next slide, please?

CHAIR KIRCHNER: (Audio interference.) address that previous slide. Since one of the purposes is to identify potential design improvements, can you -- you've actually -- from your DCA application you've been working on your PRA for this plant for quite some time. Can you point to any specific areas where the PRA insights led to specific design improvements from the DCA submittal to the SDAA submittal?

MR. SCHNEIDER: One improvement that was informed by the PRA was in the ECCS design. We added venturis at the containment isolation valves to limit the inventory loss in the case of a failure of the containment isolation valves to (Audio interference.)

CHAIR KIRCHNER: Okay. Thank you.

MR. SCHNEIDER: Okay. Next slide, please? So here is a comparison of the results of the PRA from the US600 and the US460, and you'll see that core damage frequency and the large release frequency for the different hazards. We aren't going to go over all the differences in the numbers. They all changed as you can see and that's a reflection of both the overall design changes as well as some changes in outside inputs. There are some changes in the generic data, which is an input to the PRA. But it's all reflective of a living PRA. One thing that hasn't changed --

MEMBER DIMITRIJEVIC: This is Vesna. So I have a question about this because this was one of my questions. These changes shows little -- the core damage frequency getting little worse. And I assume that this is because of ECCS changes, because ECCS performance has a little degradation because removal of the valve or adding the SOVs.

But the thing is which is really shocking to me is this improvement in large release frequency where the previously condition of failure -- of containment failure probability was in order or 0.1, which is requested in -- or expected the safety goal. And it suddenly improve 1,000, like three order or two order of magnitude. So that's a really big change in the results.

So, okay. Here's my question: So I assume that all the design changes are reflected in the PRA. And you said also there was some change in generic data and some outside inputs. So what would the other changes than design changes reflected in differences between 460 and 600?

PARTICIPANT: (Audio interference.)

MR. SCHNEIDER: Okay. Go on.

MEMBER MARTIN: Your question mostly focused on the large release frequency, Vesna?

MEMBER DIMITRIJEVIC: No, my question -- first I want to understand before we start discussing because I see that we will come to discuss all of those. So I just pointed out that there's a huge difference in the large release frequencies. And I mean, that could be from -- due to these venturis in the flow restrictions in containment isolation valves. I don't know why it is, but that's a really big difference. And it's really -- I expect to see that through discussion.

My question at this moment before we go to the specific discussion, are those differences mostly because of all design changes or there was some other, because you said there was a difference in the outside inputs or in the data? Or will these changes in the data or other inputs impact changes we see here in the results? What about the changes, the design changes consider when this PRA was made specific for 460?

MR. SCHNEIDER: I can't speak on which were larger influences in the actual numbers.

Sarah, do you have an idea (Audio interference.)

MS. BRISTOL: Etienne.

MR. MULLIN: This is Etienne Mullin, PRA with NuScale. The change in our results is a combination of changes to the design, changes to the generic data, the input to our models. I think on the upcoming slides we'll talk a bit about why the core damage frequency is changed and then we'll focus quite a bit on the large release frequency changes.

MEMBER MARTIN: Those insights that you gain impact in the approved US600 PRA? These are always living documents, but did you get insights? Say as time goes on you always get more information on changes in that sense. Is there kind of a go-back as you learn more about this design, even though of course it is different, that plays back into the US600? And is that being updated?

MR. SCHNEIDER: I can't speak to updates that we may or may not be making to the US600 design, but we have evidently learned a lot through the years of maintaining our PRA and applied some of those lessons to the design of the US460 design.

MEMBER MARTIN: Okay. I believe Walt had his hand up.

CHAIR KIRCHNER: Yes, I wanted to add onto Vesna's line of questioning. So I look at this chart and I see -- let's just start with internal events. You've got a CDF for the DCA of 3 times 10 to the minus 10th. You put a conditional release on the containment, which is 0.1. And you get a large release frequency of 2.3E-11th. So one order of magnitude difference between the two. Then you go over to the new SDAA design. You've got a higher CDF and on the order of three, four order of magnitude difference in the large release frequency.

The venturis obviously help you on things like CDF, but they don't isolate containment. So I don't see the marked improvement in the containment design, notwithstanding all the higher pressure rating and such that would give you from an engineering design standpoint four order of magnitude difference.

So could you elaborate? That is an enormous spread in PRA space. Take the absolute numbers off and talk about orders of magnitude. That is really significant.

MR. SCHNEIDER: Yes, well, we're going to address that in a couple of slides, but just at a quick high level it's a consequence of some changes we made to the ECCS actuation criteria, the removal of the IABs, the addition of the venturis. All of these design changes have allowed us to mitigate breaks outside of containment with a failure of isolation without requiring the operator action to add coolant. And that's primarily responsible for the dramatic reduction in the large release frequency.

CHAIR KIRCHNER: Well, the ECCS changes; I get that, mainly impact your CDF. I don't see how they impact the containment integrity.

MR. SCHNEIDER: That's correct. The ability to mitigate these un-isolated breaks outside of containment is due to the ability to actuate ECCS early and depressurize the system to atmospheric pressure such that we are no longer losing coolant and you can keep the core covered without having to add water.

CHAIR KIRCHNER: But your CDF has gone up.

MR. SCHNEIDER: That's right, but I think that's largely unrelated to the reduction in large release frequency.

MEMBER DIMITRIJEVIC: Well, basically your reduction in large release frequency comes from the definition of large, right? By introducing -- by depressurizing and restricting releases you -- what you define as large release has significantly reduced.

Is that what's happening? And we will see this when we start discussing specifically the LOCAs outside containment of your steam generator tube ruptures and things like that which become totally insignificant contributors to the large release.

MR. SCHNEIDER: The reduction in large release frequency is not -- has nothing to do with our definition of what a large release is. It reflects the fact that events that previously were core damage and large release are no longer a core damage event and therefore are not a large release event.

MEMBER DIMITRIJEVIC: I see. All right. Okay. My original question was that -- what I was concerned is that by the changing data also this contributes to the significance, because you said there was some change in the data. So I just want to make sure that we will understand what are changes due to design changes and what are changes due to the different inputs.

Okay. Well, once discuss we will go to the specific right changes and discuss them as we go, right?

MR. SCHNEIDER: Yes, we can do that.

MEMBER HARRINGTON: And this is Craig Harrington real quick. I assume that the changes in high winds categories were all input data?

MR. SCHNEIDER: The last thing I wanted to say about this slide was the one thing that hasn't changed from the DCA is that these CDFs and LRFs are still many orders of magnitude below the Commission's goals. So we still have a very safe plant to offer.

Okay. Next slide, Wendy? So I think we --

MEMBER DIMITRIJEVIC: Well, that's true what you said, but that could reflect the degree of uncertainty. We don't have a concern about that you safety goals or not. The questions is the -- with determining significant agreement and with determining uncertainty. And this is what changes also. That's why. We are not going to drill you on the -- we understand that this is a safe plant. It's just the question what degree of uncertainties we see in these results and where the PRA provides input how -- why the goals are (Audio interference.)

MEMBER MARTIN: Thank you, Vesna. As I've listened to Vesna's question and looking at your slide here, the conditional containment failure probability, of course you say less than 0.1, which is -- I guess comes from the SECY, I don't know, 8387 or whatever. Was there much change between US600 and US460, more or less the same magnitude?

MS. BRISTOL: This is Sarah Bristol. The magnitude could be calculated with the LRF over CDF. And so it is different. It is orders of magnitude different, yes.

MEMBER MARTIN: Okay.

MS. BRISTOL: Ultimately as you know that's less than safety goal.

MEMBER MARTIN: Yes.

MS. BRISTOL: Yes.

MR. SCHNEIDER: We pretty much already discussed the contents of this slide about how internal event CDF increased due to in part ECCS changes. And internal events large release frequency decreased. And that's primarily also due to changes to ECCS. But as Etienne said, for a large release those changes are to allow breaks outside of containment to be mitigated without the need for inventory makeup.

MEMBER DIMITRIJEVIC: So basically ECCS become less reliable with the current changes, right? Because the level of the vessel is reduced and this -- actually this -- the trip valves now dominate this, right, because fail of tube will fail the system. So ECCS become less reliable. That change is understandable.

Now that's what causes the changes in CDF and also in the contribution from the external events like the winds and tornadoes, right? And I assume that these -- right now we're going to discuss these breaks outside of containment for LRF.

MS. BRISTOL: That is correct.

MR. SCHNEIDER: And sorry, I think I would clarify, the ECCS valves are not less reliable, but we anticipate more actuations which provides an opportunity for an incomplete actuation which is a contributor to the core damage frequency. So more frequent ECCS actuations result in more frequent incomplete ECCS actuations which results in more frequent core damage frequency.

MS. BRISTOL: To clarify also, the developed reliability as you mentioned, Vesna, is less reliable. And so that's one thing that you're seeing here that isn't an apples-to-apples comparison in the cut sets from DCA to SDAA.

MEMBER DIMITRIJEVIC: Right.

MS. BRISTOL: And so the main valves' reliability reduce -- or increases an order of magnitude. And that's significant. That can be seen in the cut sets. And so while the NuScale design hasn't changed, we believe the valves will be reliable.

The generic data has changed for what we're using in the PRA for both the main valves as well as those trip valves. So that reliability decreased as well in that generic data that we're using.

And as you mentioned, Vesna, there are two trip valves now also in the common cause of those. And so a lot of the potential failures that we consider in the PRA are shown in the increase in CDF. And that's why you see that increase in CDF in all the external events. ECCS can mitigate all those that increased as the ECCS reliability data and the generic data.

And so again, there are a lot of ECCS changes that are mitigating these breaks outside containment and that's where you see that -- the increase in the -- or decrease in the LRF frequency, but the CDF is increased because of the ECCS generic data reliability pretty significantly.

MEMBER DIMITRIJEVIC: But because of the -- yes, SOV failure rates for three valves dominates now the ECCS. That's correct.

Okay. And now we will see on this

frequency. This is also interesting. All right. Okay.

MR. SCHNEIDER: Okay. So this slide we're going to talk about the mitigation of those un-isolated breaks outside of containment. So early ECCS actuation limits inventory loss through the break by reducing the systems to atmospheric pressure.

And the relevant ECCS design changes, which we've discussed all of these in the earlier session. The removal of the IABs on reactor vent valves, the addition of a low reactor pressure vessel riser level, ECCS actuation signal, and then those venturi flow restrictors on the CVCS lines.

That limits the break flow before you get pressure released to atmospheric pressure.

During the review we added an uncertainty to our table of uncertainties in the application addressing the likelihood of weld failures between the containment vessel and the containment isolation valves. So it's a very unlikely weld failure and there are means for a plant to identify a possible weld leak before it gets to the weld break stage. And so those factors combined we get an event that we don't specifically analyze, but we wanted -- we included it as an uncertainty (Audio interference.)

MEMBER DIMITRIJEVIC: Okay. So basically you don't consider isolated breaks, right, because of the low likely of failure? So that's all right. How about -- okay. So what you're said previously is that the early actuation and reduction in the pressure actually you -- the loss of the coolant outside of containment doesn't require any makeup in the -- and then that's a main difference between the previous -- between 600 and 460?

MR. SCHNEIDER: That's correct.

MEMBER DIMITRIJEVIC: All right. So now I see what you mean. So the core damage frequency never occurred because you didn't really -- you didn't need any makeup for those losses?

MR. SCHNEIDER: Correct.

MEMBER DIMITRIJEVIC: And you guys done success criteria? And that's not a shock?

MR. SCHNEIDER: Yes. Yes, that's correct. Yes.

MEMBER DIMITRIJEVIC: I see. And same thing for steam generator tube ruptures?

MR. MULLIN: This is Etienne again. The question is have we preformed simulations -- success criteria simulations for steam generator tube ruptures. And the answer is yes.

MEMBER DIMITRIJEVIC: All right. And was there anything else significant that would -- did you need the DHRS for the initial pressure reduction or the -- I'm just trying to think that this is a big difference basically. You don't really have LOCAs outside of containment anymore. You don't have containment bypass events which dominated previous LRF, yes.

MR. MULLIN: So the -- I -- maybe call it a system success criteria for this event is unique to have success without adding coolant to the NPM. We need all of the ECCS valves to open, not just one vent valve and one recirc valve. And we also need DHRS to (Audio interference.) So we need our passive systems to work effectively to be able to show success without adding coolant.

MEMBER DIMITRIJEVIC: And what happen if that fails?

MR. MULLIN: What happens if that fails is --

MEMBER DIMITRIJEVIC: What happen if you fail like for example -- I'm just -- I'm sorry. I'm just opening your event trees. So what happen if you fail the -- you need the operator to bypass -- okay, here is un-isolated. Okay. So you need to open all four DHRS. You have to open all four ECCS valves, right? And --

MR. MULLIN: That's right.

MEMBER DIMITRIJEVIC: -- then if you don't open those all four, then you go for makeup, right?

MR. MULLIN: Yes, but the operator is going to --

MEMBER DIMITRIJEVIC: You need to open at least two and then makeup. Is that the true statement?

MR. MULLIN: (No audible response.)

MEMBER DIMITRIJEVIC: Okay. All right. Sorry. Sorry. Sorry I interrupted you. So if you don't open four, then your success criteria open two and other makeup, right?

MR. MULLIN: That's correct.

MEMBER DIMITRIJEVIC: I see. And does that sequence lead to the large release?

MR. MULLIN: If operators fail to add coolant, that's correct.

MEMBER DIMITRIJEVIC: So if you fail to open all four ECCS valves, those sequences will lead to the large release frequency?

MR. MULLIN: Yes, but again operators would have to also fail to add coolant.

MEMBER DIMITRIJEVIC: All right. Thanks. You can continue. I just have to think about that. I have to check in the LRF cut set will I see this. Okay. All right. Continue.

MR. SCHNEIDER: Next slide, please?

MEMBER ROBERTS: Yes, this is Tom Roberts. Just wondering, if you're reducing the system's atmospheric pressure is there a potential for air leakage back into containment?

MR. SCHNEIDER: Yes, we would expect that.

MEMBER ROBERTS: Then that would seem to be a lot more oxygen than the assumption of just radiological decomposition that -- if you start sucking in air, it seems like you then are more relying on the power or some other means to maintain the inert environment? That right? I thought since there's no concern on certain reactions on hydrogen combustion. It sounds like if you start sucking air back into containment, then you would have to provide more on the PAR.

MR. SCHNEIDER: So this is not a severe accident. Haven't experienced core damage.

MEMBER ROBERTS: Okay. So there's no scenario like this where you reduce system pressure where you do get core damage?

MR. SCHNEIDER: So in events with core damage and a failure of the containment boundary we already consider that a large release. So let's say all four ECCS valves did not open, operators failed to add coolant, core damage occurs with an open CVCS line. And there will be a period of time where you're generating hydrogen and maybe your system pressure will actually increase above atmospheric, but you could get to a case where come back down to atmosphere you're pulling air in. The assessment on combustibility within the containment is largely irrelevant. We've already had a large release (Audio interference.)

MEMBER ROBERTS: Okay. So the compounding effect of the hydrogen issues and consider the analysis because you're already -- basically you have your release?

MR. SCHNEIDER: Yes, we don't evaluate it beyond the point of it being a large release.

MEMBER ROBERTS: Okay. Thanks.

MEMBER MARTIN: And this is Bob. You all run like NRELAP5 analyses of these scenarios? I would not expect a whole lot of air typically to blow down. And there's a brief period where you might draw from the atmosphere, but it's hot and it's going to boil. Then you're going to push that out. I would expect that you've done analysis on what you've see.

MR. SCHNEIDER: So, yes, I might be going a little bit out on a limb, but if you were to have an open penetration to the environment, you would depressurize to below atmospheric pressure. I don't know the figure right -- but several psi, certainly many psi below atmospheric pressure. And so that delta between what you would depressurize to and atmosphere that's how much air you're going to be holding.

MEMBER MARTIN: Right. Then once that little -- it's really a brief period of time where you drop below and then you will -- because these things are at the top, right, this basically events at this point?

MR. SCHNEIDER: Yes.

MEMBER MARTIN: So it should be a relatively small amount of oxygen that (Audio interference.) You're still going to be -- you're still hot, you know, boiling, and it's going to continue post-critical flow to release steam.

MR. SCHNEIDER: Yes.

MEMBER MARTIN: So you would not expect

that. My question was really that the analysis was NRELAP5, right? Or something.

MR. SCHNEIDER: I primarily use NRELAP5 for these success criteria analyses to demonstrate that core damage doesn't occur. We also use MELCOR for following severe accidents, core melts scenarios. But there's some overlap where it will benchmark the codes.

MEMBER MARTIN: Okay. Thanks.

MR. SCHNEIDER: So our next topic is regarding how NuScale determines --

MEMBER MARTIN: (Audio interference.) your microphone, please.

MR. SCHNEIDER: Thank you. So our next topic is how NuScale determines component candidates for risk-significance. We use both an absolute criterion and a sliding scale to determine the components. And the sliding scale is a change from the DCA.

So the sliding scale applies only to an importance factor. There is no change to the absolute conditional core damage frequency and conditional large release frequency thresholds. And you can see that in the top two rows of the table there, which is from the application. That's for a component and then system. And those criteria are unchanged from the DCA.

Now what's new. The next four rows, which is that sliding scale for this importance factor. And the reason that NuScale did that is to identify components that have -- that actually contribute to absolute risk in the PRA. And so, the sliding scale was chosen in a way that tries to equalize the absolute risk. As the core damage or large release frequency gets lower it tries to identify components that contribute the same absolute risk to that hazard with some allowance for -- from uncertainties in it.

MEMBER DIMITRIJEVIC: I have a lot of comments, yes. I don't want to go to this discussion on absolute relativity because I happen to disagree of this discussion, but that's another one. That's a philosophical question.

What is relevant for my discussion on this, if you look -- like let's say look in the large release frequencies. And your conditional LRF is 3 minus 7. That's mean that the -- and you know, and I'm much more interested in this measure than Fussell-Vesely. Fussell-Vesely reflects basically -- if you're going to make these component, which you evaluating perfect, how much you will improve CDF?

When it comes to D-RAP -- went through a lot of thing -- much more is interesting what will happen if this component is like to fail? In that case your conditional LRF, or what will be risk achievement? What is the condition of core damage frequency, and much more interesting, risk measure?

And what you are saying here in this table -- we just saw that your LRF is in the order of the E-13. So you're saying if the component fails and LRF is now 3 minus 7 -- if is -- your LRF is now less than 3 minus 7. So let's say is 2.0E-7. It's changed from minus 13 to 3 minus 7. That component is not important. That doesn't make any sense because this is a huge increase in LRF. And how can you say that that component would not be important?

You see what I'm saying, that when looks at risk achievement vault, if you're allowed in your conditional LRF given this component failure to be -- as long as it's less than 3 minus 7, that component is not important? But your actual LRF is -- I don't know, I mean, is dependent of the events. But let's say the total LRF is 10.E-11. You allowed four order of magnitude to increase in LRF if this component is fail and this component is still not considered risk-important. There has to be some breaks there. Who can say that component which was 1,000 time increase my large release is to important? That's a question. I would like to hear your thinking on it.

MS. BRISTOL: Thank you, Vesna. This is Sarah Bristol. I understand the question. And I think what we're looking at here, as you know, are potential risk -- significance criteria for consideration for candidates, just one part of the consideration, and they are thresholds that we proposed and reviewed in the topical.

And I guess I would say ultimately they're well below the safety goals for one. And so just because of that large delta it doesn't necessarily indicate a less-safe design or a less-safe system or component. And so we can -- is E-7 important? But as you mentioned, we're looking at overall frequencies and large release frequencies in this case in the order of E-13.

Based on the analysis we did our numbers for the system importance didn't really get up that high and so it might be a no-never-mind here, but ultimately these thresholds are well below the safety goals. And so I guess I would just stop there. And I understand that delta, how that could seem, quote/unquote, significant, but ultimately the overall safety of the plant is still confirmed by the design and the risk insights.

MEMBER DIMITRIJEVIC: And as I promised, we are not questioning safety of the plant. Even we can question if you liked all of these components not to be in tech specs. And it can be definitely out operation and things like that and combine all of those which are classify as not risk-significant. But one of the question is you're coming here as a plant which is much safer as currently operated plant. And that's probably true.

I just want to say though your risk-significance determination, it does not really show that. I mean, you have to -- this is where using -- actually you are using absolute risk measure, not your relative increase. You're just using what current plants are using and saying, okay, well, we are still much better than that. That doesn't mean when you combine all of those systems that you are going to declare to be not risk-significant that we don't know really what the risk profile is.

Also the other thing is here when you go -- like say for example, for D-RAP -- and I did not discuss that in 17.4 -- if you're going to -- if you're just going to give them these risk measures, there is nothing that which is going to be risk-important. You have a very low risk profile. And then you're using Fussell-Vesely's. It's no wonder that -- other than the main stuff. You're not going to find anything else that will be risk-important.

Maybe some additional inputs have to be provide so that those components which have a very high -- if they're left to run to failure, have a very high impact on risk should be identify. That's my point. Because that's basically showing that your defense-in-depth is significantly reduced.

So my point is we are just -- and I know that you have this TR approve as a part of your 600 application. And we are ready to -- we would like also to discuss these things with the staff. The thing is here is that some additional break should be put in so that defense-in-depth is not significantly reduce and that this profile is -- this profile remains low. So, okay. I just made this point.

I do know that you can -- you're not going to change things and things like that. But when comes to the D-RAP and things like that maybe some additional inputs should be identify. And maybe you should provide this high conditional core damage frequency and large release frequency as inputs.

MS. BRISTOL: Thank you, Vesna. Yes, we are a member of the D-RAP panel and we are able to take our insights and the actual calculations. So while maybe these thresholds seem high, they are what they are. But the PRA individual is able to go to the D-RAP panel and share those insights, share those deltas, the actual values. And the panel can then determine from that. So it's not necessarily just limited to these thresholds itself. The PRA individual does bring those values and those insights to the panel for discussion. The panel can even decide to make things -- classify them as risk-significant even if they don't meet this criteria.

So again, I wouldn't say we're limited to this table, but the panel itself can make their decisions with this consideration -- with these considerations and these inputs.

MEMBER DIMITRIJEVIC: Okay. Good. Thanks. This is good to hear because one of the examples -- you're relying on this very passive systems which there is not much operating experience and you are also going to evaluate this passive cooling criteria. But then all the backup systems to the makeup are coming as a known risk-significant. So it's just like the operator action or things necessary. So this is one of my concerns.

MR. SCHNEIDER: And I'll just add on, Sarah, to what you said about the panel. In his discussion of 17.4 Peter Shaw mentioned that we may be made the top structure risk-significant. And that was based on the judgment of the expert panel, not because of input from the PRA.

Okay. I believe that ends --

CHAIR KIRCHNER: Well, since you threw that out, I can't pass up -- this is Walt Kirchner. Well, first of all, I share Vesna's concerns because from the DCA we had -- when you did the D-RAP the CVCS system was not considered important, yet that was the only means really, that and the containment drain and fill system for actually restoring any lost coolant. So that makes one, pardon my saying it like this, scratch my head and say is this a mathematical exercise or is this an engineering exercise?

And so I share Vesna's concern that this may be consistent with the Reg Guides and the PRA standards, but from an engineering standpoint it begs the question about defense-in-depth.

As far as the upper structures, of course they would be important in seismic analysis because if those pipes aren't properly supported like the CVCS inlet and outlet lines, then the possibility of those being fractured during a seismic event become much higher. So it's just good engineering to make that logical conclusion even if the PRA hadn't specifically gone to that level of engineering detail and analysis. So I meant that more as an observation, not as a question.

MR. SCHNEIDER: Okay. Thank you. I believe that ends our presentation on 19.1.

So next slide, please? We'll move onto 19.2, which is severe accident evaluation. There's one change to 19.2 that we wanted to present to the Committee and there's a new COL --

MEMBER DIMITRIJEVIC: Excuse me. Excuse me. Can we just go back? I don't want to leave 19.1 before discussing steam generator tube rupture with the NuScale because the steam generator tubes were evaluated as not risk-significant. And in our discussion with -- about the DW, that was -- one of the argument was the steam generator tube rupture was found not to be risk-significant from the PRA.

So what we saw previously in the 600 is that the steam generator contributed 1 percent to the LRF. It was much higher in that time. And now contribute less than -- I don't think 0.1 percent to the LRF, which is much lower. That's mean actually steam generator tube rupture contribution to the LRF was totally eliminated. It's down to 10 to minus 14 or something less than that, or 10 to minus 15.

So how that happen? What's the difference? What is the difference in design that contribute to the steam generator tube rupture is not important? That's one of my questions. I wasn't sure, should I ask that when staff discuss it or with you. But I would like to hear NuScale argument on that.

And the second thing is when the sensitivity runs around for the multiple steam generator tube ruptures and the tube ruptures in the two different steam generators, they both show as not risk-significant. So can we just have a discussion on it?

MR. SCHNEIDER: Yes, certainly.

Etienne, do you want to address those issues?

MR. MULLIN: So to your first question, steam generators can contribute -- steam generator tube failures can contribute to a large release or can I guess result in a large release in the same way that a CVCS injection line break outside of containment can lead to a large release. You need failure of isolation. And there's the ability to reach atmospheric pressure without coolant addition. So from going from US600 to US460 the same design changes that have reduced the large release frequency across the board also apply to steam generator tube breaks.

As for the relative contribution of steam generator tube breaks to the large release frequency compared to -- injection lines breaks rather contributes to a large release frequency, I can't speak to that directly at this moment.

As for I think your second question, we perform sensitivities on steam generator tube breaks specifically looking at multiple tubes failing instead of just one tube. And we demonstrated that that has no meaningful impact on the event progression. You'll just reach a low RPV level faster. And the normal progression is you'll isolate the line sooner.

We also looked at steam generator tubes failing in both trains of steam generators simultaneously, or both steam generators I should say. And the impact of that has is both trains of DHRS become inoperable in effect, or ineffective. And we had sensitivities already for the PRA where we assumed that DHRS always fails and could be demonstrated through those sensitivities that the DHRS is not risk-significant. So, yes, I think that's how I'd answer your questions.

MEMBER DIMITRIJEVIC: Well, when I look at the steam generator response, the three, I don't see any -- if an insulated steam generator tube rupture, just require ECCS actuation. And it's not the makeup to the CVS makeup. I guess you don't really -- CVS is variable here. You're not considering the flood -- containment flood system. But the thing is, so what is the difference? Why was the large release much higher in the -- I don't see any -- there is no equalizing here pressure, the -- all four valves opening, things like that. What is the difference between the steam generator tube rupture in the 600 and here?

MR. SCHNEIDER: And so, again, it's similar to the impact on injection line breaks with the US460 design. If ECCS actuates successfully, it will reach atmospheric pressure and the core will remain submerged without requiring coolant addition. So, that dramatically reduces the contribution to large release frequency.

Steam generator tube breaks generally are less challenging than the injection line break case.

So as in the case of the injection line break, to reach atmospheric pressure with the core covered without having to add coolant, you need all of the ECCS valves to open. You need DHRS to work.

But for the similar scenario with this unisolated steam generator tube break, you actually only need one train of ECCS to succeed, I believe, or one vent valve and one recirc valve.

And you don't need any DHRS. And that's simply because it's a less challenging event with a smaller flow area, more pressure drop along the steam generator tube path. And I believe that the minimum elevation of the break is higher than the opening of the injection line.

So, for all of these reasons, steam generator tube breaks are less challenging than other unisolated breaks, outside of containment.

MEMBER DIMITRIJEVIC: Okay, well, so let me just ask you. So, here there was no new design change to contribute to that, it's just that you have different success criteria?

That's the first question. The second question is, would then multiple tubes make difference? I mean, obviously, we are just discussing the size of the LOCA here. But the thing is that if it is less challenging because there's a smaller LOCA size, then I mean, then multiple tube ruptures will make a difference, and you said that that doesn't matter.

I mean, so -- and okay, so I don't want to mix multiple questions. Let's just start with the first. Here, it's totally the same assumptions as was in the 600. It's a totally same sequence. And in the 600 it was assumed to lead to the large release and here, assume not to lead to large release so that you just have different success criteria.

MR. SCHNEIDER: No, the event progression is different because of changes to the ECCS scheme, that is a removal of the IABs, the use of the low RPV level signal.

Those are changes we implemented for US460 in order to mitigate containment bypassing breaks.

MEMBER DIMITRIJEVIC: Okay, I get that. So it's just earlier ECCS actuation made the difference. That's what you're saying?

MR. SCHNEIDER: That's right.

MEMBER DIMITRIJEVIC: So, earlier ECCS actuation actually prevents the --

MR. SCHNEIDER: It prevents core damage for the event.

MEMBER DIMITRIJEVIC: But always, it always prevent the core damage and so now I'm just like, I mean didn't ECCS actuation always prevent the core damage?

I mean, I just like I cannot see what is different.

MR. SCHNEIDER: What's different is we no longer require containment isolation to be successful. And I'm talking about the difference between US600 and US460.

MEMBER DIMITRIJEVIC: All right, I have to think about that.

All right, thanks.

MS. BRISTOL: And also to add on to that Vesna, if you looked at the event tree for tube failures from DCA, we needed inventory addition for success.

And so, here again with that earlier ECCS actuation similar to breaks outside containment, you don't need that addition, inventory addition.

And so, for tube failures in the DCA design, we had taken credit for RCS injection, as well as containment flooding.

MEMBER DIMITRIJEVIC: Okay, all right. That's interesting. I mean, that's really -- okay, thank you.

MR. SCHNEIDER: In 19.2, we added a COL item related to survivability in our design. There are several components that will have a severe accident dose that is actually larger than the EQ dose, environmental qualification dose.

So, to ensure that a licensee captures that in the design specs for that equipment, we have a COL item for them to identify those components.

And, that is a change to rev 2, so you won't see that.

MEMBER HARRINGTON: But wouldn't those, this is Craig. Wouldn't those components be part of the MPM supplied by NuScale?

MR. SCHNEIDER: I don't know. I'm not familiar with the supply chain.

MEMBER HARRINGTON: Yes, but it seems --

(Simultaneous speaking.)

MR. SCHNEIDER: So, yes --

MEMBER HARRINGTON: -- that you would have something for an applicant to fill out, that's going to be a part that's physically in the module when it arrives on their site.

MR. SCHNEIDER: It's just I guess an extra insurance that the dose requirements are, yes.

CHAIR KIRCHNER: Why, yes, just to add, this is Walt Kirchner, to add on to Craig's comment. This goes somewhat related to our discussion in earlier meetings about what really a standard design means.

If they're not captured by the licensee, then that's a major equipment modification, i.e., if the severe accident dose is greater than the environmental qualification dose.

So, this seems to pose a rather difficult set of requirements on a COL applicant to, I hate to use the word backfit the design, to meet the severe accident dose requirements.

MR. SCHNEIDER: It's not a backfit. They will, I mean they're going to address it up front. And, that information is already in the application in Section 19.2, those components are identified.

CHAIR KIRCHNER: So why would you not design for this requirement in the SDAA, and not wait for the COL?

MR. SCHNEIDER: It's a standard design. We don't have, I mean on a lot of components we don't have just the -- I don't want to say this wrong.

They are designed for this requirement.

CHAIR KIRCHNER: Well, that's the logical answer I would like to hear. So why is it a COL item?

MR. SCHNEIDER: There was a concern that, I think that the requirements could get overlooked because they are in chapter 19. I think that was the concern that was brought to NuScale.

It's unusual for components to have a severe accident dose greater than the EQ dose. And so, there was a concern that an applicant might just go off of the EQ specs in chapter 56, but --

MEMBER MARTIN: Well, you have, this is Bob. You've done the severe accident evaluations. You made assumptions based on your design of where everything is.

I don't think there could be too much that a specific plant could do to change, change a design and where things are located, right?

You don't expect a big difference, but you're saying that this is a way to elevate the importance? Kind of to --

MR. SCHNEIDER: Yes.

MEMBER MARTIN: -- just not let this slip through the cracks?

MR. SCHNEIDER: Correct. I think that's --

(Simultaneous speaking.)

MEMBER MARTIN: It shouldn't be necessary.

MR. SCHNEIDER: Correct.

MEMBER MARTIN: But this kind of shines the light on it.

MR. SCHNEIDER: Yes. All the information was there before the COL item was added.

MEMBER HARRINGTON: But it almost feels like different chapters aren't talking to each other. Find this over here in chapter 19. You go write it down in the other chapter that has the EQ requirements, then you don't need the COL item.

MEMBER MARTIN: Sometimes when you're trying to be extra careful, it just raises more questions, so. I'm sympathetic, I've been on your side of the table.

Any more questions on Section 19.2?

MR. SCHNEIDER: Let's move on to 19.3.

19.3 is Regulatory Treatment of Non-Safety Systems. There was no change from the DCA in terms of methodology or the results, and no SSC were identified as needing regulatory treatment for non-safety systems.

Next slide, please.

19.4 is Strategies and Guidance to Address Mitigation of Beyond Design Basis Events. And, for the SDAA, the applicant has the responsibility of addressing that issue.

And, we expect the applicant to use NuScale's topical report for that, which has been presented to the NCRS.

And, that concludes my presentation in chapter 19. I will pass it off to Peter Shaw, to present 19.5.

MR. SHAW: Hello, this is Peter Shaw again. As Jim said, I will be presenting a couple of slides here on the aircraft impact assessment for the delta between the SDA, and the DCA.

Most significant obviously, is the 6-module design versus the 12-module. So, the building footprint changed.

The other significant change would have been the fuel plate composite walls, along with the reinforced concrete members. That was also a change between the two.

There are some additional differences between the SDA and the DCA. No other buildings are credited as intervening structures.

The DCA credited the rad waste building. FSAR Section 19.5.1 updates how the assessment was performed.

This includes models for concrete and steel, as discussed before. That was part of the fuel composite design change.

19.5.4.1 had some updates for the physical damage. These include key design features.

The reactor building equipment fissile work door design changed. This also includes an SC construction, and these were included as key design features.

And then last was the emergency core cooling system was identified as a key design feature, as well.

Next slide, please.

So, for the review, there were 12 audit questions. Four were resolved with no changes to the SDA; eight were transitioned to RAIs.

And for the RAIs, some additional changes were included in the FSAR. These were clarifications on the basis of the steel composite wall efficacy.

Some details were clarified for key structural features. The reactor building equipment door was discussed with some equivalents for SC walls.

And then, some other key design features were added in accordance with the NEI 07-13 guidance.

And, there were some changes that will be upcoming from the SDA revision 1 to revision 2. There's also supporting updates in Figures 1.2, but the overall conclusions remain the same.

It is still consistent with the NEI 07-13 Revision 8 Guidance without exception. And, it meets 10 CFR 50.150(a) with containment, core cooling capability, and spent fuel integrity.

And, this is largely owed to the fact that our safety related features and the 51.50 components are consolidated to the modules themselves.

So, so long as the bays inside of the reactor building are intact, then we are also assured that those three requirements are also met.

MEMBER MARTIN: I believe that's your last slide, correct?

MR. SHAW: Yes.

MEMBER MARTIN: Any further questions, whether in the room or online? Members and consultants.

Not hearing any, it's 12:31 and that was according to our schedule, we're going to have a lunch break.

So, but we're not done with the open session. We will come back. We will recess for an hour for lunch, and then we'll hear from the staff on our return at 1:30.

So, recess, come back at 1:30.

(Whereupon, the above-entitled matter went off the record at 12:32 p.m. and resumed at 1:30 p.m.)

MEMBER MARTIN: Okay, it is 1:30. This is NuScale subcommittee. We've been discussing chapter 6, chapter 17, or Section 17.4 and chapter 19 this morning.

We're reconvening with the staff's review of Section 17.4 and chapter 19.

Who will get us started?

MR. CHOWDHURY: Yes, yes, good afternoon. Alina, would you please go back to the previous slide? Thank you.

So, good afternoon. My name is Prosanta Chowdhury. I am a senior project manager at the NRC's Office of Nuclear Reactor Regulation, Division of New and Renewed Licenses.

I have been with the NRC for very close to 20 years now, and 17 of which I have been a project manager.

Staff will present to the ACR subcommittee their review of NuScale SDAA FSAR, Revision 1, Chapter 17, Quality Assurance and Reliability Assurance. Specifically, Section 17.4, Reliability Assurance

Program.

I would like to mention here also that it is not only the revision 1 itself that came in October of 2023, but subsequently, the docketed responses to audit questions, and docketed response to request for additional information.

So, all other sections of chapter 17 were presented to the ACRS subcommittee on March 19, 2024.

Next slide, please.

So, this is an overview slide. NuScale submitted chapter 17, revision 0 on December 28, 2022, and then revision 1 on October 31, 2023.

The NRC staff performed a regulatory audit as part of its review of chapter 17, Section 17.4, in this case from March 2023 to June 2024.

Questions raised in the audit were resolved within the audit, one RAI was issued and the response was acceptable.

NuScale already showed that RAI and what -- its impact on certain section of the FSAR.

Staff completed the review of this section and issued an advanced safety evaluation report, to support the ACRS subcommittee meeting today.

But there are no significant changes between the draft SE that the staff provided to ACRS on 18 of January, and the one we recently provided 12 of February 2025.

Next slide, please.

The contributors to this section, technical reviewers are Alissa Neuhausen, Steven Alferink, and Keith Tetter.

Again, I am the project manager and Getachew Tesfaye is the lead project manager.

Next slide, please.

So, this slide and the next slide, there are two slides the NRC staff Steven Alferink will present. And, I'll turn it over to Steve.

Steve, take it away, please. Thank you.

MR. ALFERINK: Thank you, Prosanta.

As Prosanta said, my name is Steven Alferink. I'm a reliability and risk analyst in the Division of Risk Assessment.

I was one of the reviewers and I'll be presenting the staff's review of FSAR Section 17.4.

During our review, the staff focused on four areas where there were significant changes from the DCA to the SDAA.

The first two related to changes in the plant design, and the last two related to changes in D-RAP classification.

The first was related to the augmented DC power system EDAS, which holds the reactor vent and valves closed, and maintains the reactor coolant pressure boundary during normal operations.

EDAS contributes to defense in depth in the design because the reactor vent valves do not include an inadvertent actuation block valve that was present with DCA, as we discussed earlier.

The second was related to the safety related-PAR, which maintains a containment atmosphere inert during design basis and significant beyond design basis events.

The third was related to the safety related steam generator system, or SGS.

And, the fourth is related to the safety related components to the control rod drive system, or CRDS, which were not identified as risk-significant in the SDAA, but were identified as risk-significant in the DCA.

Next slide, please.

Based on its review, the staff made the following findings. For EDAS, the staff finds that the augmented design requirements are comparable with the design requirements for D-RAP SSCs.

For the PAR, the staff finds that the

safety classification of the PAR is acceptable.

For SGS and the CRDS, the staff finds that the SGS and applicable CRDS components are safety related, and subject to the requirements of the quality assurance program description, QAPD.

So in summary, the staff finds that the design and quality requirements for EDAS, the PAR, SGS, and the safety related components meet the intent -- sorry, safety related components, CRDS -- meet the intent of the Commission policy stated in item E of SECY-95-132.

And, that the design and quality requirements resulting from the classification of SSCs is consistent with the guidance in SRP section 17.4.

So that is the end of the staff's presentation on Section 17.4.

MEMBER MARTIN: Okay, if there are no questions, I assume next it's chapter 19, correct?

MR. ALFERINK: Yes.

MEMBER MARTIN: Okay.

PARTICIPANT: Speak up, please.

MEMBER MARTIN: No, you're still coming through pretty quiet.

MS. SCHILLER: I would like to thank the ACRS subcommittee, NuScale Power, and the general public for staff's opportunity to present the significant changes from the DCA and SDAA for chapter 19 for probabilistic risk assessment and severe accident evaluation.

NuScale submitted chapter 19 version zero SDA safety evaluation analysis report in December 2022, and revision 1, October 2023.

From March 2023 through August 2023, the NRC conducted a regulatory audit on chapter 19, which generated 117 issues.

Issues raised during the audit, were resolved within the audit.

Six requests for additional information were issued, and all of those were accepted in this document.

The staff completed chapter 19 review and issued an advanced safety evaluation to support today's ACRS subcommittee meeting.

Since providing the draft safety evaluation to ACRS in January, on Table 19.1-4 was updated to include two COL items which were inadvertently missed from the draft.

The contributors were Alissa Neuhausen, Marie Pohida, Sunwoo Park, Keith Tetter, Michael Swim, Anne-Marie Grady, Steven Alferink, George Wang, Thinh

Dinh, and Ryan Nolan.

I'm the chapter team lead here and Getachew Tesfaye is the SDAA lead here.

Today's presenters are Marie Pohida, Anne-Marie Grady, Mike Swim, and George Wang.

The slide lists the five sections and now I'm turning over to the first presenter, Marie Pohida.

MS. POHIDA: Thank you very much and good afternoon. I'm a senior reliability and risk analyst in the Division of Risk Assessment.

Okay, I'd like to start on slide 28. Thank you very much.

All right, what we have here on this slide is a list of the significant changes to the risk profile between the DCA and the SDA. And, that's based on design changes.

First, the core damage frequency. The CDF increased due to more frequent actuations of ECCS valves.

The dominant contributors to CDF include high winds, module drop, external floods, internal events, and internal fires, which is a complete difference from the DCA where module drop comprised over 90 percent of the core damage risk.

So, the risk profile changed quite a

little bit.

Regarding the large release frequency, the LRF, large release frequency decreased to earlier actuation of ECCS valves.

And, the contribution to LRF from breaks outside of containment decreased. And I'll be discussing that in a future slide.

In this design, there's the addition of the digital reactor building crane control system. And, that minimizes operator error.

Particularly, operator errors of commission. And that will also be discussed in a future slide.

MEMBER HARRINGTON: This is Craig. Can you speak to why high winds are a contributor?

MS. POHIDA: Yes.

In the DCA, the risk profile was dominated by module drop. And, those failures were driven by failures of limit switches and operator errors of commission, okay.

With the addition of this digital control system, that portion of risk decreased. And, what increased were other initiating events like high winds, external floods, internal events.

And for example, high winds if you were to have a sustained loss of power, of offsite power, and if you would not, if the diesel generators failed to actuate, then that would require an ECCS actuation.

So, this is going to be a theme that common cause failure of the ECCS to actuate is like 90 percent of the core damage frequency.

Does that help your question?

MEMBER HARRINGTON: So, the high wind issue is more about power lines and things like that, than it is any other impact of high winds?

Everything's inside a concrete building so it seems like high wind would not be a big issue. But if it's tied to offsite power, I can see that.

MS. POHIDA: There is a 24 ECCS timer that actuate if offsite power is not restored within 24 hours.

Okay, so for example, if you were to have a loss of offsite power, you would have a loss of, you were to have a high wind event, okay?

There's two backup diesels. If they're not able to provide power, successful actuation of DHRS will not prevent the ECCS 24-hour timer from actuating, and requiring an ECCS demand.

Does that help?

MEMBER HARRINGTON: Yes, I think so.

MS. POHIDA: Okay.

Anyway, please don't hesitate to ask me any questions.

All right, I believe I was done with that slide, so if I may, I'll continue to slide 29, but thank you.

All right, so this is a list of the focus areas for our PRA, severe accident review. And, I will discuss the specific impacts in future slides.

I want to discuss the impact of change ECCS actuation set points, the PRA modeling of the EDAS system, CVCS line breaks outside of containment.

Unisolable CVCS line breaks outside of containment. And, I'm talking about weld failures at the containment isolation valves.

My colleague Mike Swim is going to be discussing density wave oscillation impacts on steam generator tube rupture failure.

I'll be continuing with discussion of the reactor building crane digital control system, be talking about the top support structure, and that connection to the nuclear power module.

And where that's relevant, that's relevant to the drop of a module that's being moved for refueling, if it were to be dropped and impact an operating module.

And, my colleague Anne-Marie will be discussing the addition of the PAR.

And I'll stop here if there's any questions?

Go to slide 30.

Okay, the impact of the ECCS actuation changes on CDF and LRF. About 90 percent of the core damage scenarios involved incomplete ECCS actuation.

Also, the ECCS design was changed. There's now actuation signals on the low RCS level that's at the top of the riser, and at the low load RCS level that's mid-riser, that result in earlier ECCS actuation.

There's also an 8-hour ECCS timer that was added. And, operators may bypass the timer after checking for shutdown margin and hydrogen concentration.

This operator action was not found to be risk significant. This operator action to check for shutdown margin and hydrogen concentration is performed after every reactor trip with successful decay heat removal system actuation.

And the human error probabilities in the FSAR reflect this.

There I'll stop and see if anybody has questions.

MEMBER ROBERTS: Yes, a real quick one. This is Tom Roberts.

MS. POHIDA: Thank you.

MEMBER ROBERTS: Can you clarify what incomplete ECCS actuation is?

MS. POHIDA: Let's say the ECCS is demanded and the RVV valves open, and the RRV valves do not open.

MEMBER ROBERTS: Okay, thanks. So it's not unnecessary ECCS actuation?

MS. POHIDA: No.

MEMBER MARTIN: So what is it about the partial actuation that makes things worse than a complete actuation?

MS. POHIDA: There's many scenarios that demand ECCS actuation. It's high winds -- it's external events like high winds. External events like external flooding, and it's internal events.

It's LOCAs within the containment. It's LOCAs that are unisolable outside the containment.

MEMBER ROBERTS: Okay, I see. So these are cases where ECCS is intended to operate --

(Simultaneous speaking.)

MS. POHIDA: Absolutely.

MEMBER ROBERTS: -- but fails. Okay, thank you.

MEMBER MARTIN: Marie, a particular example you gave, that's got to be the domino one, right?

Of all the incomplete ECCS actuations, a scenario where you basically lose inventory but you can't get it back in, correct?

Or is there another example you can mention that is equally --

MS. POHIDA: Not on the tip of my tongue. As I understand the question is what's most likely --

(Simultaneous speaking.)

MEMBER MARTIN: The answer you gave, that's the, that to me was the obvious one. I just wondered if there's anything else in there, in the PRA, that would come close to being as significant as that particular scenario that you used as an example.

MS. POHIDA: Depending on the scenario, and I have to be careful here so I don't stumble.

For many scenarios, the lifting of one reactor vent valve, and the opening at the appropriate time of one reactor recirculation valve, is sufficient.

But there's scenarios, right now we're in beyond design basis.

MEMBER MARTIN: Sure.

MS. POHIDA: So, there's going to be scenarios that, that is not sufficient to preclude pumped injection.

But you're asking me what the most likely one is and I would defer to my lifelines in the --

MEMBER MARTIN: Well, I guess what I was asking is that one is just most obvious.

MS. POHIDA: Yes.

MEMBER MARTIN: Right? And I'm trying to think what could be kind of second on that list that might be close.

MS. POHIDA: Yes.

MEMBER MARTIN: As far as maybe not likelihood, but certainly, maybe likelihood. It just seems hard to damage the core without that one particular scenario that you mentioned.

Because it's all about keeping inventory, and the only mechanism I would see you couldn't keep inventory is if you couldn't otherwise get water back in.

I mean, you can get more and more incredible, less likely, come up with scenarios, but as far as the one that's most likely, that would seem to be the one that's obvious.

I just can't think of another one that would, that would really rise very high on the list --

MS. POHIDA: Yes.

MEMBER MARTIN: -- than the example you gave.

MS. POHIDA: May I take that back and think about that?

MEMBER MARTIN: All right.

MS. POHIDA: And go look at the numbers.

MS. NEUHAUSEN: Can I add one more? And that’s just if both (Audio interference.) fail and the -- both (Audio interference.) open and (Audio interference.) stay closed?

MEMBER MARTIN: That’s the example she gave, right?

MS. NEUHAUSEN: It’s either the two on the top or the two on the bottom. The difference would be liquids base or vapors base.

MEMBER MARTIN: Right, right, but if the ones at the bottom open up, yeah, you would lose inventory. But it would fill up, and you could, you know, more or less maintain probably some circulation in that scenario. But anyway. And maybe if you only had one of those. All right.

MS. NEUHAUSEN: Thank you.

MEMBER MARTIN: These are fun things to ask.

MS. POHIDA: They're fun things to think about. My next slide, I'll continue to -- on the next slide on EDAS modeling. In this SDA design, it's different than the DCA. The ECCS reactor vent valves are held closed by EDAS, okay.

So failure of any two of the four channels of EDAS, and I'm talking about the module-specific EDAS as opposed to EDAS-C, which is shared among modules, would cause a reactor trip in ECCS actuation. So the PRA group looked into the PRA modeling of EDAS, and particularly how it was modeled in the PRA and the failure data. So we spent a significant amount of time doing that.

I'm going to go back to the second bullet. EDAS was not identified as risk-significant from, you know, PRA importance measures. It is a single failure-proof system. And there is physical separation between the divisions. And I'm talking about Division 1, which is Channel A and C, and Division 2, which is B and D.

But as I mentioned earlier, failure of two channels of the module-specific, that's EDAS-MS, results in reactor-trip and ECCS actuation.

We did note, when we were reviewing the fall trees, that common cause failure of the EDAS electrical busses are -- the common cause failure was not modeled in separate compartments. Common cause failure was modeled between one division or the other division, but not among, you know, both -- not among both divisions that are physically separated.

And it's also important to note that the data for EDAS common cause failures modeled in the PRA is derived from operating plant data, and where DC power is safety-related.

So, the FSAR states, and this is in Section 8.3 of the FSAR, it states that the EDAS will be included in the owner's controlled requirements manual. And that's a COL item under Chapter 16, that's a COL action item, and the maintenance role.

And so specifically, you know, it states in FSAR Section 8.3, excuse me, that the goal is to ensure during operation that common cause failure does remain as the dominant failure mode, and that the reliability of EDAS is equivalent to a Class 20 system.

EDAS did not meet the RTNSS criterion. It did not meet the RTNSS criterion for being a significant contributor to CDF and LRF. And also in 8.3, it also stipulates that EDAS will be included in the maintenance role. So with here, I'll just stop and see if anybody has any questions.

MEMBER ROBERTS: Maybe I missed the conclusion, but the conclusion is that, do you agree that it's not risk-inevident, is that the conclusion?

MS. POHIDA: Alissa, may I defer that to you?

MS. NEUHAUSEN: Yeah, the EDAS includes augmented volume requirements that are similar to those for (Audio interference.) program.

MEMBER ROBERTS: Thank you. I was trying to -- you'd agree what this slide. I think what this slide means is that you agree with the licensee, or the applicant, rather, that the EDAS as designed and as they plan to manage it is equivalent essentially to what you already modeled -- what you already modeled under the PRA.

It was a common cause failure from safety-related electrical system, that type of thing. You were in agreement that that's reasonable?

MS. NEUHAUSEN: Yeah, we agree that the modeling is consistent with the design.

MEMBER ROBERTS: Okay, thank you.

MS. POHIDA: I'm going to go to slide 32. These are CVCS line breaks outside of containment. So as NuScale mentioned and as was mentioned this morning, there are flow-restricting venturis in the injection and discharge lines, so that, you know, controls the inventory loss and aids to reduce the large release frequency from CVCS line breaks outside of containment.

If at least one train of decay heat removal system is available and all the ECCS valves are open, that means the two reactor vent valves and the two reactor recirculation valves are open, the PRA success criteria is met. And that pumped injection is via the non-safety-related cavity flood and drain system, and CVCS is not needed to prevent core damage.

And that's -- that is a risk-significant design enhancement from the DCA where unisolated breaks outside of containment, which were not isolated. And you know, failure of pumped injection, you know, comprised a majority of the large release frequency. So this was a significant design enhancement.

MEMBER DIMITRIJEVIC: Hi, is this is Vesna. Well, I had forgotten to ask that while the

NuScale was presenting. When they decided their importance measures, were those importance measures based on the total CDF over LRF, or they are a base for every, you know, hazard separately shutdown?

So if something was important in internal events, it's considered important? Or just it has to be important in the total CDF and LRF? Are importance measures means -- everything which is important, for example, for external flaps is considered important.

MS. POHIDA: Okay.

MEMBER DIMITRIJEVIC: You know what I mean? There is importance measures for every CDF and LRF, and we have a different CDF and LRFs, and we never discussed total CDF and total LRF. So I assume the importance measures are based on the -- you know, on the 123s 810-plus LRFs of 20 different factors.

MS. POHIDA: Okay, I'm kind of at a loss on how to answer this question. Are you talking about the difference of importance about CVCS line breaks that weren't isolated, you know, outside of containment? Are you talking about the difference in importance between the DCA and the SDA?

MEMBER DIMITRIJEVIC: Well, I'm sort of like -- why I'm asking this question because I suddenly I got curious in importance. And I'm actually very curious in importance of DHRS. Because of so many times it comes up in discussion. So if we have importance measures which are connected with large release frequency for internal events, obviously the DHRS would be important because it prevents.

If it's not available, it will not prevent the loss of inventory. So that's what I want -- that's what I was trying to bring up. So I was sort of curious, like was that for -- is it LRF for internal events one source of importance measure? That's my question, and then I will bring this DHRS discussion, so.

MS. POHIDA: Okay, well, you know, please forgive my slowness. So the concern is about the importance of DHRS --

MEMBER DIMITRIJEVIC: Right.

MS. POHIDA: -- as a system in the SDA.

MEMBER DIMITRIJEVIC: Yes, as a system in SDA. Yes.

MS. POHIDA: Okay. Well, and not just for this scenario but for -- in the PRA as a whole.

MEMBER DIMITRIJEVIC: Well, that's why I'm asking you, are importance measures based -- and what are the important measures based on? Is there importance measures which are related to large release for internal events where that was part of the CVCS line breaks outside of containment used to dominate in DCA?

MS. POHIDA: Yes, they did.

MEMBER DIMITRIJEVIC: So now if we don't have a DHRS, they will still dominate. I mean, I assume, because I would not see any other difference, you know, because DHRS was important to credit this ability to prevent lost inventory.

So that's why I'm asking, are importance measure only measure to what is the total LRF for all the events and, you know, hazards and shutdown and blah, blah, blah? Or the importance measures are the ten different categories?

MS. POHIDA: Okay, for this design, regarding the importance of DHRS, it is safety-related in this design.

MS. NEUHAUSEN: This is Alissa Neuhausen again. So, Vesna, if I understand the question correctly, I think the absolute risk metrics are based on the aggregated hazard, and the Fussell-Vesely are based on individual hazards.

MEMBER DIMITRIJEVIC: So what was the first thing you said, Fussell-Vesely were based on the individual, and what is the CDF based on, the risk-achievement or your conditional, what is that based on?

MS. NEUHAUSEN: On the aggregated hazard.

MEMBER DIMITRIJEVIC: On what?

MS. NEUHAUSEN: On all of the hazards, and I believe NuScale is also got --

MS. BRISTOL: Thanks, Alissa. This is Sarah. So we, for that Fussell-Vesely and the risk-informed criteria we provided earlier, we look at that for every hazard. And we'll go internal events and all of the external hazards, low power shutdown. And we'll look at those criteria for each hazard for that power (Audio interference.)

MEMBER DIMITRIJEVIC: Okay, all right. So starting now, we understand what I'm getting in. This is based on Fussell-Vesely even, because this -- this type of events used to dominate all large release, right? It was only thanks to --

MS. BRISTOL: Correct.

MEMBER DIMITRIJEVIC: -- this DHRS process of opening all the valves that you were able to eliminate that, right?

MS. BRISTOL: Correct, but as you know, that's one portion of events that make up core damage. And so we look at that as --

MEMBER DIMITRIJEVIC: I'm talking large release here, just large release. All right, so.

MS. BRISTOL: Understood. But ECCS is going to mitigate that event and prevent a core damage before DHRS would need to be important to mitigate a large release. You wouldn't get there if ECCS is successful, even with --

MEMBER DIMITRIJEVIC: And I understand this all. I'm just talking is about what will happen is the current LRF will change to the value of previous LRF if you don't have a DHRS. Is that a true statement? Your currently LRF frequency would be the same as it was in that DCA.

MS. BRISTOL: I understand what you're saying and I believe it would increase. I can't say to what value it would increase to, but that does make sense that it would increase if DHRS ––

(Simultaneous speaking.)

MEMBER DIMITRIJEVIC: Yes. So therefore it would increase to the, you know, whatever value was the -- we said it was like from that 13 to the -11 or something. So this is what I was sort of trying to say. Wouldn't that really indicate the importance of DHRS?

But that's all right, I mean, I'm -- you know, I don't really know your numbers, so I mean, it just looks to me that if you without DHRS would go to much higher LRF, that would be -- yeah, but not have you just looking in Fussell-Vesely and you know, CBDP will be all right. Okay.

MS. POHIDA: Vesna?

MEMBER DIMITRIJEVIC: So that will definitely -- yeah.

MS. POHIDA: I would just like to add that the internal events portion of the PRA has multiple scenarios in it. I mean, this is just one of them. This is CVCS line breaks outside of containment. We have LOCAs, you know, inside containment.

MEMBER DIMITRIJEVIC: Yeah, I understand but -- yes, but this particular event dominated your LRF in the huge project, right? It was almost all your LRF come from those -- the CVCS breaks outside containment.

MS. POHIDA: In the DCA.

MEMBER DIMITRIJEVIC: In that DCA, yes. That's why I'm sort of concerned. That's what I thought. I mean, I can go and look in percentage, but that -- that was -- I remember that this dominated all LRF and then I was wondering. All right, okay, well let's continue with the discussion.

MS. POHIDA: Thank you. I’ll go to slide 33. Okay, now we’re talking about, in this slide, unisolable CVCS breaks outside of containment. And so what the PRA staff evaluated, we looked at the likelihood of weld failures at the junction between the containment vessel and the CVCS containment isolation valves. We looked at the likelihood of these weld failures because they’re not modeled in the PRA.

But it’s important to note that the plant behavior and the consequences of an unisolable CVCS LOCA outside of containment are modeled. And how that is modeled is you’re modeling a CVCS break downstream of the containment isolation valves with failure of containment isolation valves.

So the weld failure was not numerically included, but the plant behavior and the consequences of this break are modeled in the PRA. So this weld failure, weld failure frequency, there’s uncertainty on this weld failure frequency. It’s identified as a key source of level two uncertainty in the upcoming revision of the FSAR, Revision 2.

And the impact of this weld failure frequency is minimized by leak detection and operator response. And as two examples, it would be Tech Spec

345 on RCS operational leakage, and temperature monitoring under the bioshield.

And with that, I'll just break here and see if anybody has any questions.

MEMBER DIMITRIJEVIC: The interesting -- how was the frequency of isolable CVS the LOCA calculated? What was the used, the weld number, pipe lengths, or?

MS. POHIDA: I beg your pardon, for the weld failure frequency?

MEMBER DIMITRIJEVIC: Well, how -- okay, my question is how was isolable CVCS break, you know, outside of containment calculated? How was that frequency calculated?

MS. POHIDA: I may have to take that back and get back to you, how was it calculated.

MEMBER DIMITRIJEVIC: My question was it based on the weld estimate, weld number? So was it based on pipe length, or what was it based on?

MS. POHIDA: You know, I'm going to have to take that question. I don't recall a change in modeling of the CVCS break frequency outside containment between DCA and SDA. But since Sarah's here, I defer.

MS. BRISTOL: There was no change.

MS. POHIDA: I beg your pardon?

MS. BRISTOL: There was no change between DCA --

MS. POHIDA: I didn't think there was.

MS. BRISTOL: -- and SDAA. And so, well for these line breaks, we calculate, we measure the length of pipe between the various distances that are designated. So pipe breaks inside containment would go up to the containment isolation valve, and so that's a distance.

And so the isolable IE frequency is then calculated from the pipe length inside containment of those lines designated up to the CIV.

Does that help, Vesna?

MEMBER DIMITRIJEVIC: Well, no, I was wondering, did you use the weld number of the pipeline? Because you can use either based on EPRI done on flat frequency. So I mean, the thing is like I was wondering did you have a data on this weld failure frequency? If you worked with weld frequency.

MS. BRISTOL: Generic date, yup. And so we used the pipe failure for large and small breaks for the length, and then we used just the generic HOV CIV and reliability data.

MEMBER DIMITRIJEVIC: All right, so you

did not use weld frequency. That was, I was just curious about that. Okay.

MS. POHIDA: May I continue?

MEMBER DIMITRIJEVIC: Sure, thanks.

MS. POHIDA: And with that, I'm going to turn it over to Mike, who's going to speak on DWO. But thank you.

MR. SWIM: Yeah, thanks, Marie. And good afternoon, everyone, my name's Mike Swim, I'm a Reliability and Risk Analyst. Been with the agency about one year. I was out in industry for 13 years before that as a licensed SRO and diesel generator engineer.

And for my role in this review, I was assigned a disposition with DWO for Chapter 19, and Chapter 19 specifically. So the PRA did not explicitly model DWO impacts to the steam generator to failure-initiating event frequency. It considers things like high cycle fatigue, fretting wear from normal operating conditions.

And so why was this okay from a Chapter 19 perspective? And what I came down to between interactions with the staff was there's no -- even if it were to be modeled, there were no significant impacts to the results or insights of the PRA. And so

I'll go through in a little more detail here how that works.

So how can DWO impact the normal event progression for a steam generator tube failure? And some of the things that we considered as a staff was the increase in the steam generator tube failure initiating event frequency. And also potentially worst case consequences and what are required in the Chapter 15 analysis of the single tube failure.

So looking at it from a perspective of maybe multiple steam generator tube failures. And as a result of maybe the loss of -- or a failure of steam generator tubes and both steam generators, which would result in a loss of both trains of the decay heat removal system.

I do want to note, you know, these are things that we looked at to make a safety evaluation from a Chapter 19 perspective. So you know, some of the steam generator tube failures, or multiple steam generator tube failure, for instance or this appearing in both steam generators.

I'm not making a declaration that this is likely to occur from DWO conditions. This is just something that we looked at to bound the potential risk of this condition in the operating plan.

So for the event progression for multiple steam generator tube failures, including in both the steam generators, there's -- based on interactions with NuScale, there's no discernable difference, with a couple of exceptions. The expected response would happen faster, so you'd reach various actuation setpoints quicker due to a larger break size.

And of course what I've already mentioned with the, if this were to occur on both steam generators, both trains of the decay heat removal system would be lost simultaneously in both steam generators.

So there is some sensitivity analysis performed by NuScale. One that's referenced in their table 19.1-22 was they increased the initiating event frequency by more than a order of magnitude, with no change in CDF or LRF, and a sensitivity to a loss -- study of a loss of both trains of decay removal system, which resulted. And still not identifying the decay heat removal system as a candidate for risk significance.

So a combination of those in and of itself was enough for us to find out that NuScale PRA was still technically adequate and consistent with the Chapter 19 standard review plan, even without further explicit consideration of DWO as it impacts on the initiating event frequency.

We did do some further analysis in house, just to build a stronger comfort level with this stance. So we did some informal validation. The NRC has a SAPHIRE SPAR model for the NuScale design, and so we did some further worst-case hypotheticals manipulating the SPAR model internally.

Looked at things like assumed an initiating event frequency of times 100. No significant changes to CDF and LRF. And developed a new, a venturi, even to include common cause failure of the steam generator tube failures as a result of DWO. But no significant changes to CDF and LRF.

That's all I have for slide 34. Are there any questions?

MEMBER DIMITRIJEVIC: Did you do this simultaneously? Did you increase frequency the number of the tubes and put them in both steam generators? I mean, was that trial performed? Did you assume multiple tubes of the both -- I mean, you know, distributed between two steam generators and you know, and then see what is totally impacted. Then look how sensitive it is to the frequency.

MR. SWIM: Vesna, just to make sure I

understand your question, you're asking with respect to internally with our SPAR model, or what the direct question asked of the, of NuScale?

MEMBER DIMITRIJEVIC: Yeah. I mean, sensitivity is the -- I don't know could you do that on the SPAR model because you had to run success criteria, you know, to see what was the --. If you have a larger sized LOCA and do you need DHRS to, you know, to prevent inventory loss. I mean, things like that. So you couldn't run that.

So I was sort of wondering if they run multiple tubes of the different steam generators and then see how sensitive to the frequency. Because I have to put that this frequency is, you know, based on expert opinion and is much lower than what we see in the current industry, so. So that's what I was sort of wondering was that combination of those factors, what is the sensitivity run by NuScale.

MR. SWIM: So, that, I'd say that consideration was embedded in our question to NuScale for consideration. I don't have specifics on how large a break they did do or analyzed. I will say with respect to the system response and how the event would progress, at a certain point, the things like the secondary system isolation and ECCS do actuate on

RCS-level-type setpoints.

So you would be reaching those setpoints faster. Does that answer your question, Vesna?

MEMBER DIMITRIJEVIC: No, I mean, I'm aware of this. I'm just really, you know, my question was that, you know, the -- you know, preventing inventory loss, I mean. So that's what my question was, similar to the, you know, charging outside of the containment when, you know. We have a situation where you need DHRS and opening all ECCS valves.

I mean, I was wondering, and we have a similar situation with steam generator tube ruptures where the DHRS may not be available. So that was my concerns.

MR. SWIM: Okay. Was there anything additionally you'd like from me? I guess I didn't hear another question in there. Was there --

MEMBER DIMITRIJEVIC: No, that's okay, I mean, I already discussed that with the NuScale. I mean, I don't have really have my answer, but you know, I will look more into that.

MR. SWIM: Thank you, Vesna. Was there any other questions? All right, well, I'll --

MEMBER DIMITRIJEVIC: What I wanted to say, it's difficult for me to see, and I'm not really concerned the thought of risk, as I said many times. I'm just concerned about what the PRA identify as important for the other consideration. And for me to see the steam generator tubes are not important, it's difficult thing to, you know, to fathom. So I just, that's why I'm sort of questioning. And then you know.

The assumption is that this is much better frequency, that we're only going to have a single failure, that blah blah blah. So you know, I'm not too comfortable with it.

MR. SWIM: And I understand your perspective. I think that was part of why we dug a little deeper with the SAPHIRE SPAR model. I mean, my time with the operating plants, you know, E-3 was one of the procedures you really didn't want to get into. I didn't look forward to getting into because of the all the time-critical operator actions, cooldown to pressurization.

But one thing with the NuScale design that is unique to operating plants is there's no relief valves between the containment isolation and the containment boundary itself. So any kind of -- once you get into that steam generator tube failure scenario, their response becomes a lot simpler.

MEMBER DIMITRIJEVIC: Right, but we are just discussing unisolated steam generator tube.

MR. SWIM: With that, I'll pass the presentation back to Marie.

MS. POHIDA: Thank you very much. Okay, if I may, I'd like to go to slide 35. And thank you, Anne Marie.

I'd like to go back to the presentation on slide 35 on the reactor building crane digital control system. So with the SDA there was the addition on the reactor building crane digital control system, which significantly decreases the contribution of operator errors of commission. And I'm thinking of in the past DCA design, examples are would be overspeed, overtravel, overload.

This control system was designed to prevent such errors. In fact, the reliability of this reactor building crane control system is such that now the dominant contributors to module drop are actually redundant load path failures like, you know, catastrophic gearbox failure and wire rope failures. The contribution of the operator to module drop in this SDA design is very, very small.

This digital control system is classified as non-safety-related. However, it is risk-significant. And the programmable logic controller is validated and verified under software integrity level 3. There's more information about this control system in FSAR Section 9.1.5.5.

And you know, to add some more information, this single programmable logic controller, it controls operation, it controls monitoring. And their software interlocks to prevent, you know, collisions with other SSCs and to prevent the operator from operating the crane outside its equipment design capabilities.

And I'll go to slide --

MEMBER ROBERTS: And Marie, just out of curiosity, does level 3 define how you model the failure rate to say a software error? Or if not, what do you use to estimate the software failure rate?

MS. POHIDA: I'm going to have to get back to you on that. What we did is when we reviewed the PRA, we worked with expertise from the I&C branch regarding making sure we understand, you know, how this programmable logic controller's going to work and what requirements were on that. And I'd like to take that back and get back to you. It's a probability.

MEMBER ROBERTS: Okay, thank you.

MS. POHIDA: Thank you for your question.

Excuse me, just jotting down some notes.

Okay, slide 36. All right, top support structure, excuse me, the connection to the nuclear power module. Okay, in the NuScale design, you have a lower block assembly and that's at the bottom of the main hoist. And that connects with the lifting logs to the top support structure that sits on top of the nuclear power module.

So once again, this is in context of a postulated module drop on top of an operating module. If a dropped module strikes an operating module, you know, piping including the pressurizer spray piping, the DHRS piping at the front of nuclear power module have the potential to be impacted.

The three -- excuse me, not the three. The safety-related CVCS containment isolation valves are located under the top support structure. They protect these -- the CIVs from impacts from a postulated module drop. And this top supports structure is classified as non-safety related and it is risk-significant in FSAR table 17.4-1.

So, if there's a postulated module drop that impacts an operating module, the expectation is the containment isolation valves were closed. But since -- but both trains of DHRS could be unavailable.

They could be impacted by the drop.

If that occurs and one reactor safety valve successfully cycles open and closes, the RCS will be pressurized and then ECCS will be demanded. And even if the RSVs, the reactor safety valves, fail to open, the ECCS functioning remains a success path to keep -- to prevent core damage.

MEMBER DIMITRIJEVIC: But that scenario is not analyzing PRA. There is no -- yeah, go ahead.

MS. POHIDA: Thank you, thank you, Vesna. Yes, multi-module events, they are analyzed in the FSAR. Multi-module internal events are quantified. Multi-module external events are qualitatively evaluated.

So this is, you know, postulated drop of a module being moved for refueling on top of an operating is qualitatively evaluated.

MEMBER DIMITRIJEVIC: All right. I mean, I'm just -- no, I saw this qualitative evaluation. I was just wondering how much. Because this is now operating modules. So you know, you analyze this as a shutdown risk, but this would be operating risk, you know, so. All right.

MS. POHIDA: Thank you. Okay, may I turn it over to Anne-Marie for a discussion on PAR. Thank you.

MS. GRADY: Final discussion. The PAR was added, as we discussed earlier. Yes, thank you.

A single, safety-related PAR was added to the design. It’s not modeled in the PRA. Equipment survivability dose for the PAR post-severe accident, the two functions must be maintained at a containment integrity in post-accident monitoring.

The PAR has been added to Table 19.2-8 for equipment survivability list. A new COL item, which was discussed this morning, shows that the applicant will identify from the list of equipment on the equipment survivability list the components in the severe accident doses for cases which the severe accident dose is greater than an EQ, as described in COL item 19.2-4. Next slide, please.

Conclusion. The staff reviewed the US460 design-specific PRA. Oh, sorry, this is not my slide.

MS. POHIDA: That’s okay.

MS. GRADY: Okay, and other PRA-related information in FSAR 19.1, in accordance with SRP 19.0, DC COL ISG-28 for applicable modes and hazards. The applicant addressed the full scope of the internal and external initiating events for both full power and low power shutdown conditions.

The applicant developed quantitative risk insights from multi-module internal events and qualitative risks insights for multi-module shutdown and external events. The PRA is of sufficient technical adequacy to support the SDA. The staff's review concludes that the Commission's CDF and LRF goals have been met with margin.

Any questions on that? Next slide, please.

Okay, I think they're out of order. Still talking about the PAR and combustible gas control in containment. Because it's also described, part of the design is described in Chapter 19. Certainly equipment survivability, but also an evaluation of why the PAR isn't a PRA.

So, to refresh our memory from this morning, hydrogen combustion in the CNV. The DCA addressed a potential combustion event in the CNV analytically and demonstrated the CNV design pressure was not exceeded. The SDA, in contrast, had at a PAR, which precludes combustion events from occurring during DBAs and SAs.

A containment performance with no combustion, but the SDAA table at 19.2-1, core damage simulations for severe accident evaluation, identifies a spectrum of severe accidents that could challenge containment integrity.

The subsequent tables, 19.2-2 through 7, document the CNV design pressures, include those that generated hydrogen, the presence of hydrogen in the containments, are not exceeded. That's the demonstration, frankly that, well, that containment integrity is maintained.

The conclusion. The applicant addressed severe accidents consistent with Commission policy. The SDAA design of containment performance meets the containment structural integrity criteria of Reg Guide 1.7 and the containment leak tight criteria of SECY-93087. Next slide, please.

MS. POHIDA: Thank you, Anne-Marie.

I'd like to go over our Chapter 19 review as it relates to RTNSS, that's the regulatory treatment of non-safety systems. We had one RAI on this topic, and it had to do with the backup diesel generators, that they're not scoped into RTNSS.

We concluded that the backup diesel generators do not prevent the occurrence of an initiating event. Specifically, that would initiate the actuation of a passive system. They're not needed for long-term post-accident capabilities. They're not needed to support defense-in-depth systems.

And all the components of the backup power supply system, including the backup diesel generator and closures, are seismic category 3. And the backup diesel generator enclosure is rated for wind speeds in excess of the weather-related events included in the loop initiating event in the PRA.

So with that, I'll proceed to slide 41. So regarding RTNSS, the staff has reviewed the NuScale 460 evaluation of RTNSS in accordance with SRP 19.3. NuScale did not find any SSCs in the scope of RTNSS. And the staff, we concluded that we didn't find any SSCs that met the criterion for requiring additional regulatory treatment. But thank you.

And with that, I will turn it over to our discussion on aircraft impact analysis.

MR. WANG: Good afternoon, my name is George Wang. I'm a Structural Engineer in the Structural Stability Technical Engineering Branch. I'm from the Office of the NRR.

I'm a technical reviewer for FSAR Chapter 1925 (Audio interference.) adequacy of the design features and functional capabilities identified at this (Audio interference.) for withstanding impacts.

So next I want to talk about two safety

changes between DCA and SDA from a structural perspective. First, secondary change as SDAA use at the steel-plate composite walls.

So because it’s a unique design, so NuScale performed design-specific aircraft impact assessment both global and local assessments, experimental data to benchmark the methodology and results. NuScale followed NEI 07-13, Revision 8, without exceptions.

Due to a unique design of the SC wall, so NuScale identified some additional key design features. Base strength, then a SC wall to reinforce concrete slab connections, local detailing with tie rods in SC walls to wall connection. Also the structural steel beam seat connections. So that’s key design feature identified for SC walls.

And second significant change is for ECA (Audio interference.) with buildings at the main structures through limited potential structural changes for west side within the reactor buildings. But SDAA had not prepped the (Audio interference.)

That means that the west end of the reactor building is sufficient aircraft strike. So that’s two major difference at the (Audio interference.) from DCA to SDAA.

And this concludes my presentation. Thanks for your time.

MEMBER MARTIN: Okay, I guess your last slide here is kind of back material, correct. Something coded in a review approach. If there's no other comment --

MR. SWIM: We can go ahead and have it written out. I believe the intent was to get it written.

MS. NEUHAUSEN: Yeah, we were -- this is Alissa Neuhausen. We were asked to cover it. In, you know, in the DCA review I think we covered a lot of our review process up front. And so this is a newer tool that we've (Audio interference.) SDA we were asked to just bring.

MEMBER MARTIN: Feel free.

MR. SWIM: This is Mike Swim again. And we just wanted to highlight that for the NuScale, staff emphasized the use of the integrated risk-informed decisionmaking to bring all the technical disciplines and decisionmakers together. This slide, as we already mentioned, is a visual representation of that communication tool that was used to support the integrated review approach.

Staff would evaluate all five principles of risk-informed decisionmaking, as found in Reg. Guide 1174 across the disciplines, and used color-coding to quickly indicate status towards meeting each principle.

Now, this was an effective way to get everyone on the same and to gain alignment and communicate what the applicant still needed to provide. And part of why I'm also bringing this us is this framework is referenced from time to time as you may see as a part of the review, the SER, including explicitly in Chapter 5 of the SER for (Audio interference.)

That's all. Should be it for everything.

MEMBER MARTIN: Appreciate that. Are there any questions from the members here in the room or online?

MEMBER DIMITRIJEVIC: Yeah, I have one question about uncertainty analysis. So you reviewed uncertainty analysis and noticed there was a couple comments in the SER about that. So did you find uncertainty analysis satisfactory or not important? I mean, what was the conclusion on uncertainty analysis associated with the SDA?

MS. POHIDA: Vesna, I think I need a clarification. As I understand the question is were we -- was the multi-module assessment technically acceptable.

MEMBER DIMITRIJEVIC: No, uncertainty. I don't know if my microphone maybe not working perfectly. Uncertainty analysis, it's nothing to do with multi-module. I mean, uncertainty analysis --

MS. POHIDA: Oh, I'm sorry.

MEMBER DIMITRIJEVIC: -- is done for the multi-modules too. So you guys review uncertainty analysis I know because I saw the couple questions you also issue on the uncertainty distribution.

What was your conclusion on the review of the summary of results with uncertainty, the, you know, levels identified? Did you find this acceptable, or in this moment you think it's irrelevant because the safety goals are met with this margin?

Okay, my question is did you find the uncertainty analysis adequate in the SDA?

MS. POHIDA: I'll take a stab at this. We reviewed the sensitivity studies, the various sensitivity studies that were documented in the FSAR, and I believe it's Table 19.1-22. You know, I'm saying from my review of the passive safety system reliability analysis, you know, I spent a significant amount of time looking at data inputs to RELAP and uncertainty distributions.

So for, based on my review, yes, I found -- I found the sensitivity studies that were done and with respect to uncertainties, I found that technically acceptable.

Does that answer your question?

MEMBER DIMITRIJEVIC: Well, I was wondering about that. I mean, I had the, you know, on the sensitivity studies, there was sort of like a, you know, the very, how would I say? It's, they're not really finely defined. Like because they've saying they set all the common cause fire to 95%, then increases in 100.

Well, surprise, surprise, what does it mean, MDCCF? I mean, you know, like you know, we can just concentrate on ECCS failure. Then when you talk about the, you know, positive heat transfer, I mean, I don't know how does it reflect in the RELAP.

But here there is a number assumed that even if you have a successful actuation of ECCS, that mean, you know, positive heat transfer to reactor pool is estimated to be 1 in -7, and that's something which needs to be confirmed in the future, the test.

Well, you know, this is a very, you know, the -- what we're saying, that this is a passive plant. There is a lot of things which we are not familiar with. And this are the things which have to come with the bigger uncertainty. But this plant, uncertainty analysis shows incredibly uncertainty.

And also it's not really clear where those uncertainties, you know, where -- I'm not sure is this passive heat transfer failure at all involved because it's small compared to the, you know, valves opening or something. But the uncertainty associated that is probably high, and so is the valves operation.

This is not reflected in sensitivity if I don't say this ECCS value 95 percentile and then we see increases higher than 100. What does it mean? I mean, you know, or HEP, same thing, sector 95.

So my question is did you guys have a discussion about that, then what was the -- when you saw the narrow range of the distributions for all those things, were you concerned about that? Have you been concerned that mean value in point estimates are the same? And the state of knowledge that wasn't reflected in these things when the lot of things are common valve failures.

So I mean, I was just, you know, wondering about did you guys have a discussion about this.

MS. POHIDA: Alissa, I defer to you. I can speak specifically about the passive safety system reliability analysis, if that would be helpful. But as to overall discussions, I'm going to defer to Alissa.

MR. VASAVADA: This is Shilp Vasavada from the NRC staff. So if I understand the concern correctly, you're saying that there's not enough data, there's uncertainty.

One thing that we did look at in the review is the list of key assumptions that includes uncertainty are appropriate and capture as you can say a kind of state of knowledge, items that need to be revisited and confirmed during a COL and also later on during operation.

That was when we are dealing with the, we can call it the uncertainty variables that you were talking about, if that helps.

MEMBER DIMITRIJEVIC: That will help if it's identified somewhere, you know. It's really I have a -- I mean, this is -- this is a SDAA PRA, and it's probably best we will ever see because they have already done DCA and this PRA has a lot of details and you know, as a PRA, it's a great PRA.

Now, the question is best PRA is great as a PRA. Does this PRA realistically identify important thing. That's what I am not 100% sure. Does this PRA realistically show uncertainties, that I'm not 100% sure. This is the question, you know, I ask myself when I look in there.

Is this good PRA? Yes, it's an excellent PRA. But is a PRA good if it doesn't identify well, you know, the important contributions. And does -- is it reflecting on the uncertainties associated with this new design with the passive features and things like that.

And I question its importance for some things, but for the uncertainty analysis I totally question that, that that's uncertainty present here it's totally unrealistic, so. You know, if you say the passive failure, the passive cooling, you know, it's one E minus seven what is uncertainty distribution on this.

So I like what you said. Maybe these things should be identified in the COL, but that should be somewhere identified as something which will be look in the more details.

That's my speech then, so. And I like your presentation very much. It was very helpful, thank you.

MS. POHIDA: Thank you, Vesna. I appreciate your question. There are tables in the FSAR of key sources of uncertainty. You know, we recognize that this is a paper plant with unique design features and there's uncertainty.

But those -- but in the FSAR there are listed key sources of -- key sources of uncertainty that, you know, if someone were to come in with a risk-informed application, those would need to be reviewed and evaluated. But thank you.

MEMBER DIMITRIJEVIC: Okay.

MEMBER MARTIN: Are there any further questions? From members in the call? Not hearing any, it's time to go to public comment. Oh, feel free. Introduce yourself.

MR. OSBORN: Yeah, my name's Jim Osborn, NuScale Licensing. I just want to be very clear that, because this could be taken out of context, out of abundance of caution here, that this slide is a example, is not necessarily a particular reflection on NuScale design or NuScale PRA.

So I just wanted that on the record, that that's an example slide and not necessarily --

MEMBER MARTIN: It's a lot of red.

Okay, I think at this time we'll move to public comment. So if you're a member of the public and wish to express yourself, please maybe raise your hand using the MS Teams hand. And we'll identify you and at that time we'll ask you to identify yourself and your affiliation and your comment.

All right, going once, going twice? All right, not hearing any public comment, I think we can move to adjourn the open session. I'm looking for a nod. Okay. Okay, all right.

So this concludes the open session on Chapter 6, Section 17.4 and Chapter 19. There is a closed session scheduled. We don't have any presentations for that. So I don't know if, will we actually enter closed session?

MR. SNODDERLY: So we can have a discussion right now. I mean, I think the question is --

MEMBER MARTIN: Is it –

(Simultaneous speaking.)

MR. SNODDERLY: Yeah.

MEMBER MARTIN: Okay.

MR. SNODDERLY: How about this --

MR. TESFAYE: This is Getachew Tesfaye again. I have additional information in Section 19.5 in the closed session.

MEMBER MARTIN: So you're basically saying yes. So we will have this closed session. So then for the time being, I'm going to adjourn this open session, and then we will clean everything up and check out who can and cannot be in the room. And then we'll reconvene, it will be in about ten minutes. Let's just --

MR. SNODDERLY: I think we can do it in -- it's up to you, Bob. Do you want to break for 15 or 10? I can be ready in 10.

MEMBER MARTIN: Okay, let's do it, let's do it, well, 10's basically 3:00 o'clock. So let's reconvene at 3:00 o'clock, so we'll split the difference and reconvene with the closed session.

But this otherwise adjourns the public session.

(Whereupon, the above-entitled matter went off the record at 2:48 p.m.)

February 12, 2025 Docket No. 052-050

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 Material Entitled "ACRS Subcommittee Meeting (Open Session) Chapter 6, Section 17.4 and Chapter 19," PM-179462, Revision 0

The purpose of this submittal is to provide presentation materials for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) NuScale Subcommittee Meeting on February 18th, 2025. The materials support NuScale's presentation of the subject chapters and section for the US460 Standard Design Approval Application.

The enclosure to this letter is the nonproprietary presentation entitled "ACRS Subcommittee Meeting (Open Session) Chapter 6, Section 17.4 and Chapter 19," PM-179462, Revision 0.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Jim Osborn at 541-360-0693 or at josborn@nuscalepower.com.

Sincerely,

Thomas Griffith
Director, Regulatory Affairs
NuScale Power, LLC

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Prosanta Chowdhury, Senior Project Manager, NRC

Enclosure 1: ACRS Subcommittee Meeting (Open Session) Chapter 6, Section 17.4 and Chapter 19, PM-179462, Revision 0, Nonproprietary

Enclosure 1:

ACRS Subcommittee Meeting (Open Session) Chapter 6, Section 17.4 and Chapter 19, PM-179462, Revision 0, Nonproprietary

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ACRS Subcommittee Meeting
(Open Session)

February 18, 2025

Chapter 6
Engineered Safety Features

Presenter: Tyler Beck

Acknowledgement and Disclaimer

This material is based upon work supported by the Department of Energy under Award Number DE-NE0008928.

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Chapter 6 Overview

- Section 6.1, Engineered Safety Feature Materials
- Section 6.2, Containment Systems
- Section 6.3, Emergency Core Cooling System
- Section 6.4, Control Room Habitability
- Section 6.5, Fission Product Removal and Control Systems
- Section 6.6, Inservice Inspection and Testing of Class 2 and 3 Systems and Components
- Note: The Chapter 6 presentation covers design of engineered safety features as discussed in FSAR Chapter 6
  - The presentation does not cover specifics of accident sequences or evaluations (Ch. 15), Probabilistic Risk Assessment (Ch. 19), etc.
  - The presentation focuses on differences from the US600 DCA to the US460 SDAA

Section 6.1, Engineered Safety Feature Materials

- Containment vessel (CNV) material changes:
  - US600: CNV composed of FXM-19 (austenitic stainless steel) and SA-508 (low-alloy steel)
  - US460: CNV composed of FXM-19 and F6NM (martensitic stainless steel)
  - Addition of new Table 6.1-1, Dissimilar Metal Welds
    - Addition of weld metals due to CNV materials changes
    - Provisions for welding dissimilar metals
- Implemented additional welding controls in response to NRC staff audits (e.g., post weld heat treatment)

Containment Vessel

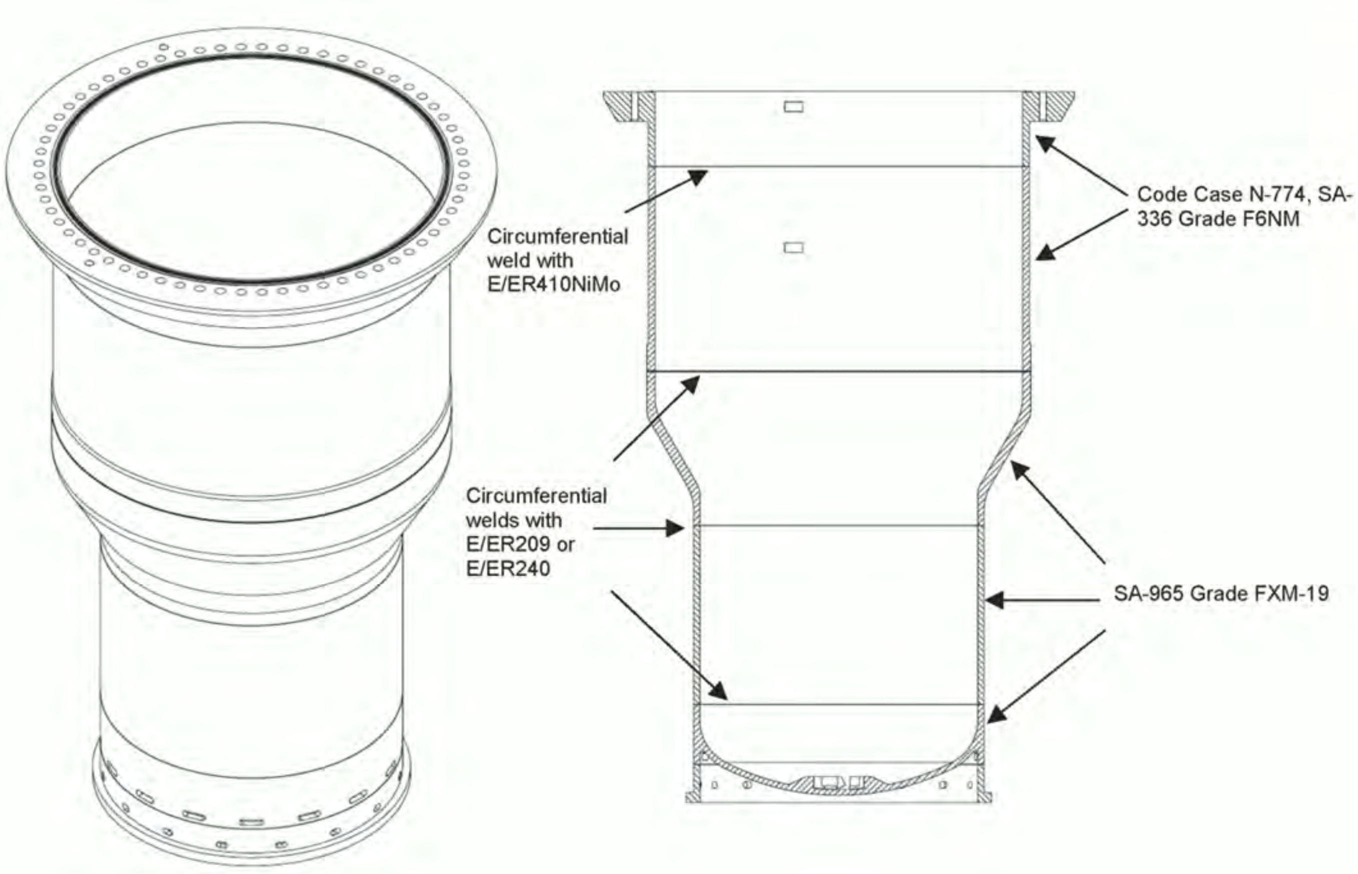
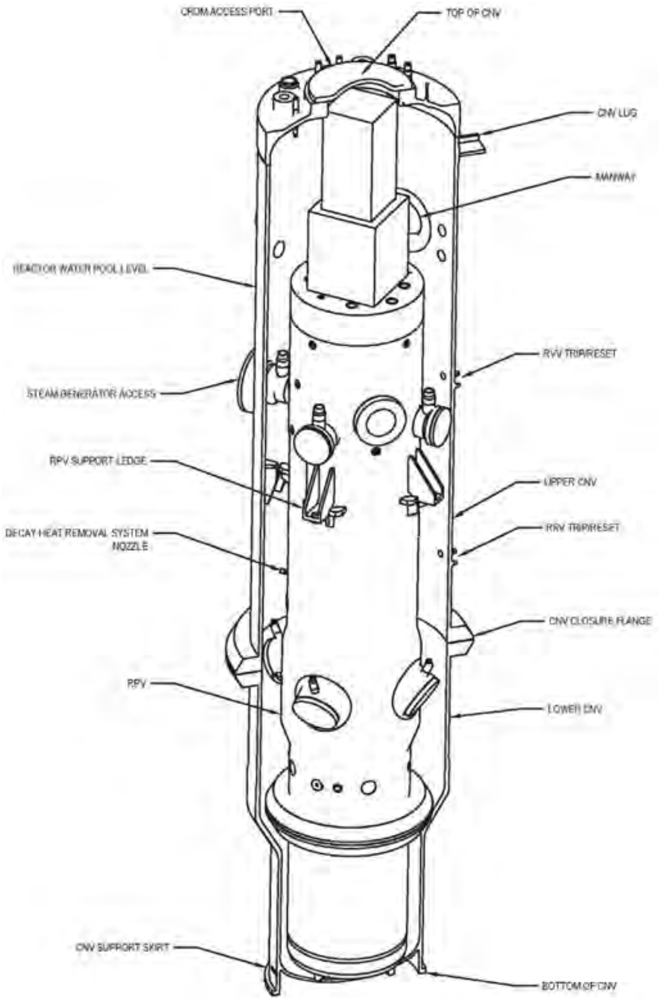


Figure: Lower Containment Vessel



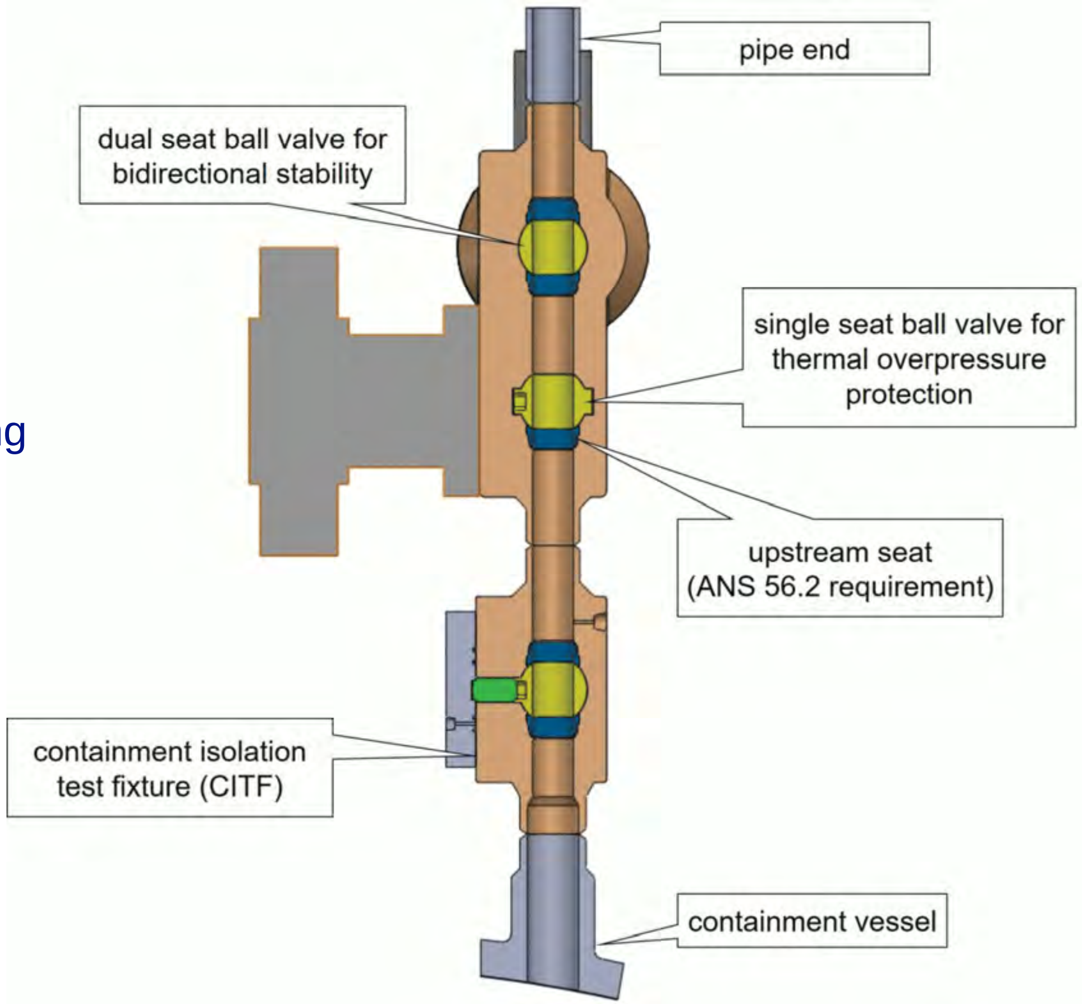
FSAR Figure 6.2-1: Containment System

Section 6.2, Containment Systems

- Containment system (CNTS) changes:
  - Containment vessel:
    - Material changes (discussed in Section 6.1)
    - Number of CNV penetrations changed from 42 penetrations to 45 penetrations
    - Design pressure rating changed from 1050 psia to 1200 psia
    - Design temperature rating changed from 550°F to 600°F
    - CVCS injection and discharge line include venturis integral to the CNV penetration
      - Mitigates line breaks outside the CNV
  - Combustible gas control:
    - Addition of safety-related passive autocatalytic recombiner (PAR) to maintain inert containment atmosphere
      - Removal of combustion loads as a result of maintaining an inert environment
    - Removal of combustible gas monitoring and an exemption from monitoring requirements

Section 6.2, Containment Systems (cont.)

- CNTS changes:
  - Containment isolation:
    - Addition of containment isolation test fixture (CITF) valve between the CNV nozzle and the containment isolation valve (CIV)
      - Improves ability to perform Appendix J testing
      - DCA design included first of a kind leak testing features integrated into the CIV assembly
    - CIVs are welded directly to CITF, which are welded directly to the CNV nozzle safe-end
    - CIV closure time changed from 7 to 10 seconds



FSAR Figure 6.2-4: Primary System Containment Isolation Valves Dual Vale, Single Body Design

Section 6.2, Containment Systems (cont.)

- CNTS changes:
  - Containment response analysis:
    - Initial conditions align with US460 standard design
    - Similar stored energy to US600
    - US460 includes more design margin
    - Methodology included in the LOCA topical report
  - Removal of COL item related to containment leakage rate testing program
  - Addition of ITAAC verifying CNV free volume (and removal of previous COL item)
- 17 audit items and 4 RAIs resolved

	US600 DCA	US460 SDAA
Primary Events Peak Pressure (psia)	994 (IORV)	937 (DL break)
Primary Events Peak Temperature (°F)	526 (IL break)	533 (DL break)
Secondary Events Peak Pressure (psia)	449 (MSLB)	900 (MSLB)
Secondary Events Peak Temperature (°F)	433 (MSLB)	530 (MSLB)

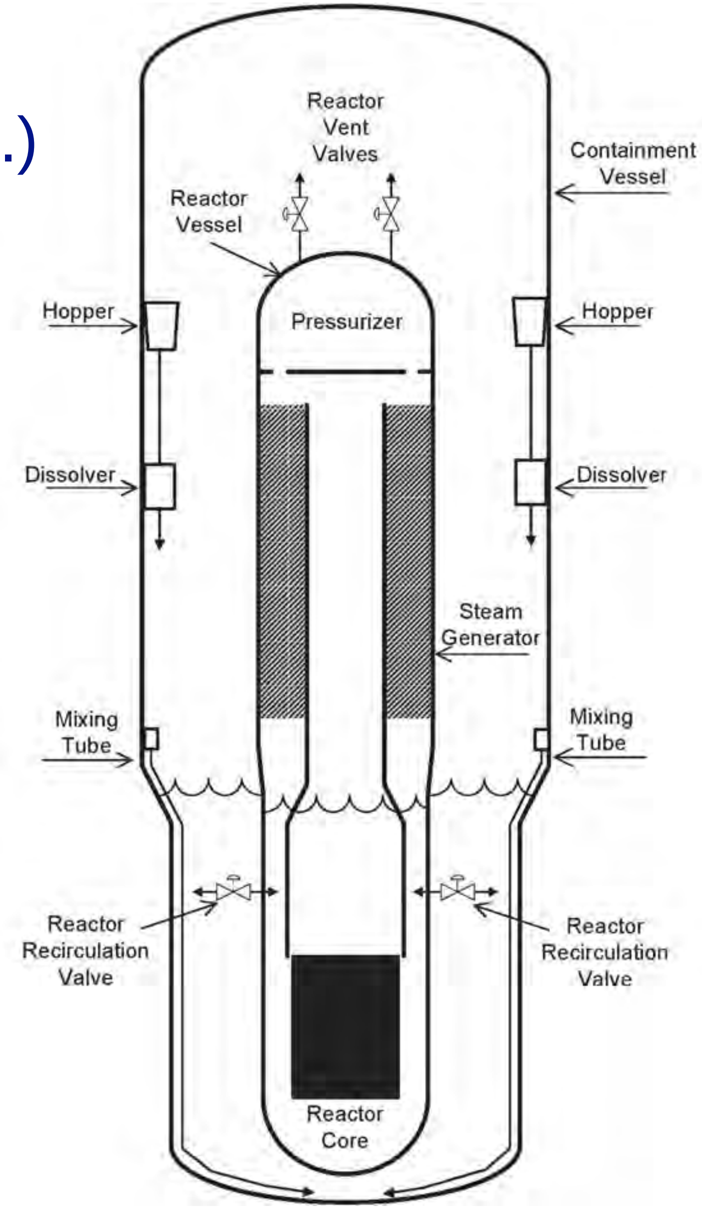
Section 6.3, Emergency Core Cooling System

- ECCS changes:
  - ECCS valves:
    - Changes related to safety analysis optimization:
      - ECCS includes two reactor vent valves (RVVs) from three in the DCA (change coincident with UHS pool level change)
      - RVVs do not include inadvertent actuation block (IAB) valve: RVVs open upon ECCS actuation
      - RRV IABs modified to 900 psid threshold (block) pressure and 450 psid release pressure
      - Addition of integral venturi to RRVs/RVVs to limit flow during high differential pressure conditions
        - Decouples flow limiting function of valve internals
    - Other operational enhancements:
      - Two in-series trip solenoid valves per RRV/RVV from a single trip solenoid valve per RRV/RVV in the DCA
  - ECCS actuation:
    - Removal of high CNV level and low RCS pressure ECCS actuation signals
    - Addition of low and low-low RPV riser level actuation signal
    - Addition of high-high RCS pressure and high-high RCS T_{ave} ECCS actuation setpoints for BDBEs

Section 6.3, Emergency Core Cooling System (cont.)

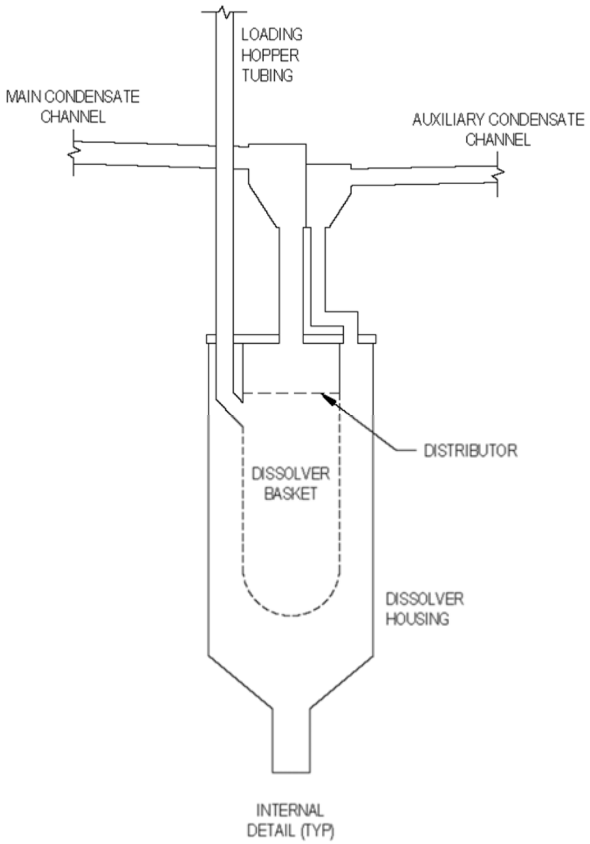
- ECCS changes:
  - ECCS includes an ECCS supplemental boron (ESB) feature:
    - Boron hoppers, condensate channels, dissolvers, mixing tubes
  - Addition of 8-hour ECCS actuation timer following reactor trip

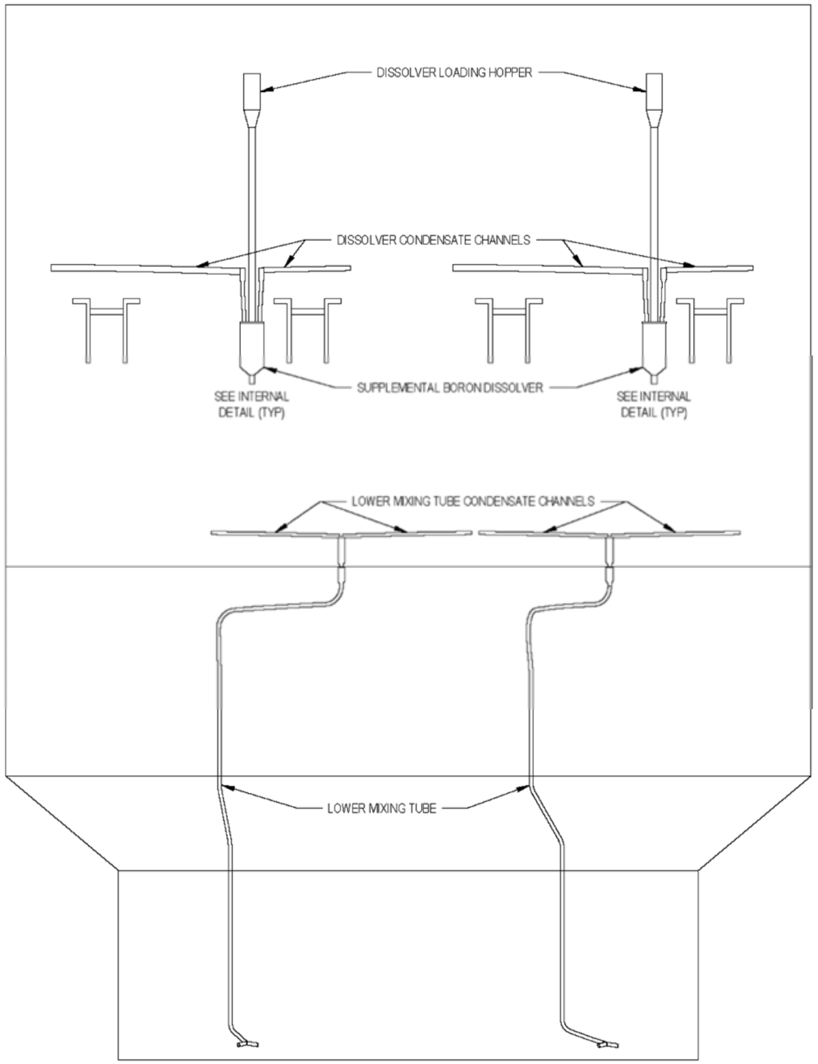
- 14 audit items and 5 RAIs resolved



FSAR Figure 6.3-2: Emergency Core Cooling System Operation

ECCS Supplemental Boron





Detail from FSAR Figure 6.3-5: ECCS
Emergency Supplemental Boron Feature Details

Section 6.4, Control Room Habitability

- CRHS changes:
  - Ten minute delay added to actuation due to a loss of power to battery chargers
  - Toxic gas detection is within the scope of COL Item 6.4-1

- Removed previous COL Item 6.4-5 that required testing and inspection requirements be specified for CRHS

- Audit and RAI Results
  - One audit item concerning test method for test 16.02.03 (FSAR Table 14.2-16) and COL Item 6.4-1, resolved successfully

Section 6.5, Fission Product Removal and Control Systems

- Unchanged from DCA

Section 6.6, Inservice Inspection and Testing of Class 2 and 3 Components

- No significant changes from DCA
    - Inservice Inspection of Class 2 and 3 components satisfies relevant 50.55a requirements and allows optional RG 1.147 code cases

- Removed previous COL Item 6.6-1
    - Inservice testing program is described in Section 3.9.6

ACRS Subcommittee Meeting
(Open Session)

February 18, 2025

Section 17.4
Reliability Assurance Program

Presenter: Peter Shaw

Section 17.4: Reliability Assurance Program

- As in the DCA, the Design Reliability Assurance Program (D-RAP) reviews and approves safety and risk classification
- NuScale re-evaluated the structures, systems, and components (SSC) classifications for the US460 standard plant design
- D-RAP expert panel insights resulted in changes to methodology for panel insights, without design changes
    - Steam generator tubes are safety-related, not risk-significant
    - Control rod drive mechanisms are safety-related, not risk-significant

- Audit Results
    - 10 items resolved in audit and resulted in updates to FSAR Section 8.2 and Figure 17.4-1 to clarify the SSC classification process and corresponding section references.

- RAI Results
    - RAI 10199, Question 17.4-11 Resolved
        - Clarified the process does not assume risk significance based on safety-related classification
        - Resulted in clarifications to the default classification in FSAR Section 17.4.3.2 and role of backup diesel generators in Table 19.1-56 (Revision 2)

ACRS Subcommittee Meeting

(Open Session)

February 18, 2025

Chapter 19

Probabilistic Risk Assessment and Severe Accident Evaluation

Presenters: Jim Schneider and Peter Shaw

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Chapter 19 Probabilistic Risk Assessment and Severe Accident Evaluation

- 19.1 Probabilistic Risk Assessment
- 19.2 Severe Accident Evaluation
- 19.3 Regulatory Treatment of Nonsafety Systems
- 19.4 Strategies and Guidance to Address Mitigation of Beyond-Design-Basis Events
- 19.5 Adequacy of Design Features and Functional Capabilities Identified and Described for Withstanding Aircraft Impacts

Application review summary:

- 156 audit issues resolved in the audit, including 84 document requests
- 15 RAI questions resolved

Note: an asterisk (*) indicates information that was added to Revision 2 of the SDAA

Section 19.1: Probabilistic Risk Assessment

General Overview

- 10 CFR 52.137(a)(25) requires a standard design applicant to develop a design-specific PRA.
- When a site is chosen and a plant built, a licensee will develop and maintain a plant-specific PRA for the life of the plant (that is, each plant shall have a *living* PRA).
  - The SDAA includes COL items that ensure the applicant has a PRA in the combined license, construction, and operational phases.
- The purposes of the PRA at the design phase include:
  - evaluate the overall safety of the plant design
  - provide insights for potential design improvements
- The safety goals of the Commission are a core damage frequency (CDF) of less than **1.0E-4** each reactor year, and a large release frequency (LRF) of less than **1.0E-6** each reactor year.

Comparison of PRA Results (mean values)

US600 DCA

Full Power		
Hazard	CDF (per mcyr)	LRF (per mcyr)
Internal Events	3.0E-10	2.3E-11
Internal Fires	9.7E-10	4.3E-11
Internal Floods	6.1E-11	<1E-15
External Floods	8.7E-10	7.9E-14
High Winds (Tornado)	9.9E-11	<1E-15
High Winds (Hurricane)	7.2E-10	6.4E-14
Seismic (SMA)	0.88g	
Low Power and Shutdown		
Hazard	CDF (/mcyr)	LRF (/mcyr)
Internal Events	4.9E-13	2.0E-14
Module Drop	8.8E-08	N/A
Multi-Module		
Hazard	Conditional Probability of Core Damage	Conditional Probability of Large Release
Multi-Module	0.13	0.01
Composite CCFP < 0.1		

US460 SDAA

Full Power		
Hazard	CDF (per mcyr)	LRF (per mcyr)
Internal Events	6.0E-09	6.6E-13
Internal Fires	4.6E-09	1.3E-11
Internal Floods	1.6E-10	3.4E-14
External Floods	9.5E-09	1.4E-12*
High Winds (Tornado)	2.6E-09	1.6E-13
High Winds (Hurricane)	1.9E-08	1.3E-12
Seismic (SMA)	0.92g	
Low Power and Shutdown		
Hazard	CDF (/mcyr)	LRF (/mcyr)
Internal Events	4.0E-11	3.5E-12
Module Drop	1.8E-08	N/A
Multi-Module		
Hazard	Conditional Probability of Core Damage	Conditional Probability of Large Release
Multi-Module	0.21	0.03
Composite CCFP < 0.1		

mcyr = module critical year
CCFP = conditional containment failure probability
SMA = seismic margin assessment

Section 19.1: Probabilistic Risk Assessment

Overview of PRA Results

- Internal events CDF increased, in part because of changes to ECCS, such as reducing the number of RVVs from three to two, the addition of an 8-hour actuation timer, and the addition of redundant trip valves on RRVs and RVVs.
  - from 3.0E-10 per module critical year (mcyr) to 6.0E-09 per mcyr
- Internal events LRF decreased, primarily because of changes to ECCS that allow breaks outside of containment with failed containment isolation to be mitigated *without the need for operator action or inventory makeup.*
  - from 2.3E-11 per mcyr to 6.6E-13 per mcyr

Section 19.1: Probabilistic Risk Assessment

Mitigation of unisolated breaks outside of containment

- Early ECCS actuation can limit coolant loss from the break by reducing system to atmospheric pressure.
  - core stays covered and core damage is avoided without requiring addition of coolant to the module
- Relevant design changes:
  - removal of inadvertent actuation blocks on the reactor vent valves
  - addition of low reactor pressure vessel riser level ECCS actuation signal
  - addition of venturi flow restrictors to CVCS injection and discharge lines to limit maximum break flow
- NuScale added an uncertainty to Table 19.1-28 addressing the low likelihood of weld failures between the CNV and the CIVs for CVCS*.
  - The low likelihood of this weld failure, combined with leak identification and response requirements, minimize the impact of this event on the LRF.

Section 19.1: Probabilistic Risk Assessment

Criteria for Risk Significance

- For determining component candidates for risk significance, NuScale uses both an absolute criterion and a sliding scale.
- The sliding scale only applies to relative FV threshold; there is no change to the absolute conditional core damage frequency (CCDF) and conditional large release frequency (CLRF) thresholds.
- At lower CDF and LRF, a higher Fussell-Vesely (FV) value is tolerated due to the low absolute risk.
- The criteria are listed in FSAR Table 19.1-19, Criteria for Risk Significance:

Parameter	Core Damage Criteria for Risk Significance[1]	Large Release Criteria for Risk Significance[1]
Component	CCDF ≥ 3E-06	CLRF ≥ 3E-07
System	CCDF ≥ 1E-05	CLRF ≥ 1E-06
Component[2]	Total FV = 0.005 if CDF > 1E-07	Total FV = 0.005 if LRF > 1E-08
Component	Total FV = 0.2 if (1E-07 ≥ CDF > 1E-08)	Total FV = 0.2 if (1E-08 ≥ LRF > 1E-09)
Component	Total FV = 0.5 if (1E-08 ≥ CDF > 1E-09)	Total FV = 0.5 if (1E-09 ≥ LRF > 1E-10)
Component	Total FV = 0.9 if (1E-09 ≥ CDF ≥ 1E-10)	Total FV = 0.9 if (1E-10 ≥ LRF ≥ 1E-11)

Notes:
1.Risk values are provided in units of per mcyr.
2.Risk values are based on Condition 4 of the SER, which requires CDF to be approximately 1E-07/year or less, along with the CCFP goal of 0.1.

Section 19.2: Severe Accident Evaluation

- New COL Item 19.2-4 related to survivability*:
  - “An applicant that references the NuScale Power Plant US460 standard design will identify from Table 19.2-8 (*Equipment Survivability List)* the components and their severe accident doses for cases where the severe accident dose is greater than the environmental qualification dose.”
  - This COL item ensures that severe accident dose requirements are captured by the licensee in equipment specifications.

Section 19.3: Regulatory Treatment of Nonsafety Systems

- No change in methodology or results from the DCA: no SSC satisfy the criteria for Regulatory Treatment of Nonsafety Systems.

Section 19.4: Strategies and Guidance to Address Mitigation of Beyond-Design-Basis Events

- An applicant that references the NuScale Power Plant US460 standard design has the responsibility of addressing mitigation of beyond-design basis events in accordance with 10 CFR 50.155.
- NuScale has presented its topical report on the NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events to the ACRS.

Section 19.5: Adequacy of Design Features and Functional Capabilities Identified and Described for Withstanding Aircraft Impacts

- High-level SDAA design changes reflected in the Aircraft Impact Analysis (AIA):
  - The SDAA Reactor Building (RXB) reflects 6 modules (12 modules in the DCA) with updated building and site layout configuration.
  - In the SDAA the RXB uses steel-plate composite (SC) walls along with reinforced concrete (RC) members.

- Additional AIA differences in the SDAA:
  - No other buildings are credited as intervening structures in the analysis (DCA credited the Radioactive Waste Building)
  - FSAR Section 19.5.1 updates how the assessment was performed, including models for concrete and steel
  - FSAR Section 19.5.4.1 Physical Damage updates reflect key design changes with the updated analysis for SC construction and site layout
  - Reactor Building equipment door design changed (with the SC construction) and details updated for the key design feature including reinforcement and connection details
  - Emergency core cooling system (ECCS) identified as a key design feature to ensure adequate core cooling

Section 19.5: Adequacy of Design Features and Functional Capabilities Identified and Described for Withstanding Aircraft Impacts

- Audit Responses
  - 12 audit questions, 4 resolved with no changes to the SDAA, 8 transitioned to RAI
- RAI Results
  - 8 RAIs: Resulted in additional design details additions in FSAR Section 19.5 to support the RAI responses
    - Clarification on the basis of steel composite wall efficacy for resisting aircraft impact
    - Clarified details of certain structural features credited as key design features for aircraft impact analysis
    - Reactor building equipment door details were discussed for equivalence to SC walls
    - Key design features added to the SDAA consistent with NEI 07-13 guidance
- SDAA Revision 2 updates to include AIA key design feature updates in FSAR Section 19.5 with supporting Figure 1.2 updates, conclusions remain the same:
  - Consistency with NEI 07-13 Revision 8
  - Meets 10 CFR 50.150(a) with containment intact, core cooling capability, and spent fuel pool integrity

Acronyms

ACRS	Advisory Committee on Reactor Safeguards
AIA	Aircraft Impact Analysis
BDBE	beyond-design-basis event
CCDF	conditional core damage frequency
CCFP	conditional containment failure probability
CDF	core damage frequency
CFR	Code of Federal Regulations
CITF	containment isolation test fixture
CIV	containment isolation valve
CLRF	conditional large release frequency
CNTS	containment system
CNV	containment vessel
COL	combined license
CRHS	control room habitability system
CVCS	chemical and volume control system
DCA	Design Certification Application
DL	discharge line
D-RAP	Design Reliability Assurance Program
ECCS	emergency core cooling system
ESB	ECCS supplemental boron
ESF	engineered safety feature
FSAR	Final Safety Analysis Report
FV	Fussell-Vesely
IAB	inadvertent actuation block
IL	injection line
IORV	inadvertent operation of a relief valve
ISI	inservice inspection
IST	Inservice Testing
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
LOCA	loss-of-coolant accident
LRF	large release frequency
mcyr	module critical year
MSLB	main steam line break
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PAR	passive autocatalytic recombiner
PRA	Probabilistic Risk Assessment
PZR	pressurizer
RAI	Request for Additional Information
RC	reinforced concrete
RCS	reactor coolant system
RG	Regulatory Guide
RRV	reactor recirculation valve
RVV	reactor vent valve
RXB	Reactor Building
SC	steel-plate composite
SDAA	Standard Design Approval Application
SER	Safety Evaluation Report
SG	steam generator
SMA	seismic margin assessment
SSC	structures, systems, and components

Presentation to the Advisory Committee on Reactor Safeguards Subcommittee

Staff Review of NuScale's US460 Standard Design Approval Application (SDAA) Final Safety Analysis Report (FSAR), Revision 1

Chapters 6 and 19, and Section 17.4

February 18th, 2025
(Open Session)

Non-Proprietary

United States Nuclear Regulatory Commission
Protecting People and the Environment

Presentation to the ACRS Subcommittee Staff Review of NuScale SDAA FSAR, Revision 1

Chapter 6, “Engineered Safety Features”

February 18th, 2025
(Open Session)

NuScale SDAA FSAR Chapter 6 Review

Overview

- NuScale submitted Chapter 6, “Engineered Safety Features” Revision 0 of the SDAA FSAR on December 31, 2022, and Revision 1 on October 31, 2023
- NRC regulatory audit of Chapter 6 was performed from March 2023 to August 2023, generating 46 audit issues
- Questions raised during the audit were resolved within the audit. Six RAIs were issued, and the responses were acceptable
- Staff completed Chapter 6 review and issued an advanced safety evaluation to support today’s ACRS Subcommittee meeting
- No significant changes between draft SE provided to ACRS on 1/18/25 and SE submitted on 2/12/25

NuScale SDAA FSAR Chapter 6 Review

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NuScale SDAA FSAR Chapter 6 Review

Sections

- Section 6.1 – Engineered Safety Feature Materials
- Section 6.2 – Containment Systems
- Section 6.3 – Emergency Core Cooling System
- Section 6.4 – Control Room Habitability
- Section 6.5 – Fission Product Removal and Control Systems
- Section 6.6 – Inservice Inspection and Testing of Class 2 and 3 Systems and Components
- Section 6.7 – Main Steamline Isolation Valve Leakage Control System (BWR)

NuScale SDAA FSAR Chapter 6 Review

Section 6.1.1 Engineered Safety Features Materials

- Significant differences between NuScale DCA FSAR and NuScale SDAA FSAR include:
  - The use of ASME Code Case N-774, “Use of 13Cr-4Ni (Alloy UNS S41500) Grade F6NM Forgings Weighing in Excess of 10,000 lb (4540 kg) and Otherwise Conforming to the Requirements of SA-336/SA-336M for Class 1, 2, 3 Construction Section III, Division 1.”
    - Code Case N-774 is listed in Regulatory Guide 1.84, Rev. 39, “Design, Fabrication, and Material Code Case Acceptability, ASME Section III, Division 1,” as permitted for use without conditions.
    - F6NM replaces SA-508, Grade 3, Class 2 from the previous design for the upper CNV and a portion of the lower CNV below the upper/lower vessel flange.

NuScale SDAA FSAR Chapter 6 Review

Section 6.1.1 Engineered Safety Features Materials

- Welding/fabrication when using F6NM requires special considerations in addition to ASME Code requirements:
  - The applicant has considered the effect of welding procedures on the Martensite start (M_s) and Martensite finish (M_f) temperatures
  - Applicant will not follow recommended preheat temperature listed in Section III, non-mandatory Appendix D regarding weld preheat temperatures
  - The applicant is employing an extensive testing program to determine the appropriate preheat temperature to prevent hydrogen cracking while at the same time promote martensite formation.

NuScale SDAA FSAR Chapter 6 Review

Section 6.1.1 Engineered Safety Features Materials

- Welding/fabrication when using F6NM requires special considerations in addition to ASME Code requirements (cont):
  - Welding processes that employ flux may require post weld heat treatment (PWHT) times than those specified in ASME Code.
    - Oxygen pickup from flux welding processes may require PWHT times greater than those specified in ASME Code to ensure adequate impact toughness.

NuScale SDAA FSAR Chapter 6 Review

Section 6.1.1 Engineered Safety Features Materials

- Welding/fabrication when using F6NM requires special considerations in addition to ASME Code requirements (cont):
  - ASME Code specifies that the PWHT temperature range, for F6NM welds, is 1050°F to 1150°F. The lower critical (Ac1) temperature for 410NiMo type weld metals and F6NM base material can be as low as 1150°F or slightly lower.
  - SDAA Section 6.1.1.1 will be modified to state, “Post weld heat treatment of SA-336 Gr F6NM for the CNV and supports shall be 1075°F +/- 25°F.”
    - Provides adequate margin to ensure that PWHT temperature does not exceed Ac1.
  - Staff determined that additional controls/considerations placed on the fabrication of F6NM are adequate.
  - Staff conclusion did not change from the DCA

NuScale SDAA FSAR Chapter 6.2.1/6.2.2 Review

Major Design Changes from DCA to SDAA

	NPM-160 for US600 (DCA)	NPM-20 for US460 (SDAA)
Rated thermal power	160 MWt	250 MWt
CNV upper vessel material	SA-508	SA-336 (F6NM)
Reactor pool level	65 ft	52 ft
Initial Reactor pool temperature	110 °F	140 °F (TS=120 °F)
Initial CNV wall temperature above pool level	240 °F	500 °F
Number of RVVs	3	2
IABs used on	RRVs & RVVs	RRVs
IAB release pressure range	900-1000 psid	400-500 psid
Venturis used on	None	RRVs & RVVs
DHRS operation for the DBE mitigation	Not credited	Credited
CNV design pressure	1050 psia	1200 psia
CNV design temperature	550 °F	600 °F

NuScale SDAA FSAR Chapter 6.2.1/6.2.2 Review

Additional Significant Changes from DCA to SDAA

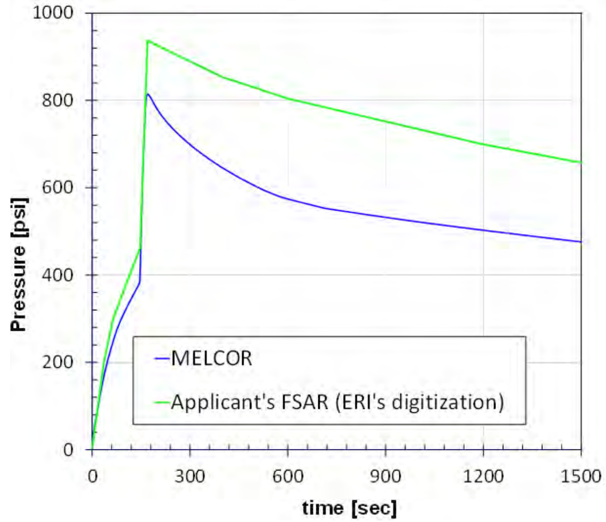
- Containment Response Analysis Methodology (CRAM) TeR was IBRed in the DCA. Modified CRAM for the SDAA CNV design for NPM-20 is a part of the LOCA EM TR-0516-49422.
- A CNV free volume ITAAC included in SDAA to ensure that the as-built CNV free volume bounds the minimum value of 6000 ft^3 used in the CNV safety analyses.
- DHRS credited to SDAA CNV DBEs: Reactor cooling pool heat-up and thermal stratification effects on DHRS and CNV heat removal performance degradation
  - Sensitivity of the CNV LOCA T/H response to break size & ECCS actuation
  - Justification for the natural convection heat transfer modeling
  - NuScale provided necessary analyses and justification through RAI 10359 response
- Containment P/T limiting design basis events have changed

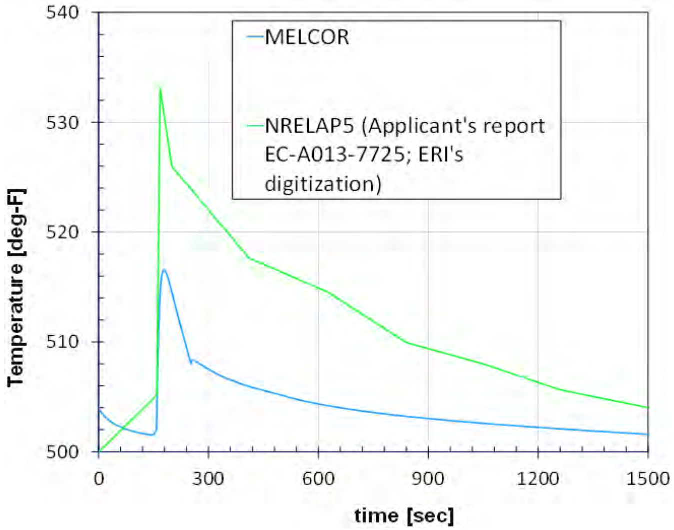
	DCA	SDAA
Peak CNV Pressure DBE	Inadvertant RRV opening	RCS discharge line break LOCA
Peak CNV Pressure	**994 psia**	**937 psia**
CNV Pressure Margin	~5% (vs. p_{design} = **1050 psia**)	~22% (vs. p_{design} = **1200 psia**)
Peak CNV Temperature DBE	RCS injection line break LOCA	RCS discharge line break LOCA
Peak CNV Wall Temperature	**526 °F**	**533 °F**
CNV Temperature Margin	24 °F (vs. T_{design} = **550 °F**)	67 °F (vs. T_{design} = **600 °F**)

NuScale SDAA FSAR Chapter 6.2.1/6.2.2 Review

Staff Confirmatory Analysis Results for the SDAA NPM-20 CNV

- Staff (MELCOR) & Applicant's (NRELAP5) Results for the Combined P/T Limiting DBA Case
  - LOCA caused by RCS (CVCS) discharge line break from the downcomer (limiting CRAM DBE) (DL) – A primary system's M&E release event





- NRELAP5 Results:
  - Peak containment pressure predicted is 937 psia (<1200 psia limit)
  - Maximum containment wall temperature predicted is 533 ºF (< 600 ºF limit)

NuScale SDAA FSAR Chapter 6.2.1/6.2.2 Review

Conclusions

- ❖ The containment safety analyses appropriately modeled the relevant phenomena in the NPM-20 CNV response including condensation heat transfer, non-condensable gas effect, decay heat, choked flow, DHRS/ECCS impact, and CNV heat removal to the reactor pool.
- ❖ NuScale CNV design incorporates sufficient conservatism in the NPM-20 CNV model ICs/BCs for the US460 design.
- ❖ NuScale SDAA FSAR Chapter 6 provides sufficient and acceptable information for analyzing the M&E release into the CNV for the spectrum of primary and secondary design basis events, and determining the limiting CNV pressure and temperature response.
- ❖ NuScale CNV design meets all regulatory requirements and acceptance criteria for the containment safety design.

NuScale SDAA FSAR Chapter 6.2.5 Review

Significant Changes from DCA to SDAA

Change	DCA	SDAA
Applicable Regulation	10 CFR 50.44(c)	10 CFR 50.44(d)
Guidance	SRP 6.2.5, 19.0	RG 1.7, SRP 19.0
Combustible Gas Control	CNV combustion analysis	PAR maintains inert CNV
Safety category	No PAR	Safety-related PAR
ITAAC	none	Physical arrangement and installation; analysis and test of recombination rate; part of EQ
Tech Specs	none	LCO 3.6.4 on PAR operability
CGC technical report	TR-0716-50424, rev 1	Several - prop, ECI
Exemption Request #2	Uncertain means of post-accident monitoring of H_2, O_2	No post-accident H_2, O_2 monitoring

NuScale SDAA FSAR Chapter 6.2.5 Review

Acceptability of 50.44(d) as Applicable Regulation for CGC in SDAA

- The CNV is not inert (<4% O_2 in presence of H_2) during a design basis accident (DBA) in the first 24 hours of a non-core damage AOO.
- 10 CFR 50.44(c) applies mainly to severe accidents
- 10 CFR 50.44(d)(2) applies to the "the safety impacts of combustible gases during design basis and significant beyond design basis accidents…"

NuScale SDAA FSAR Chapter 6.2.5 Review

Conclusion

- ❖ Combustible Gas Control conclusion:
  - ❑ During non-core damage DBA LOCA, PAR is credited to maintain an inert CNV
  - ❑ Post severe accident, CNV remains inert without crediting PAR
  - ❑ During long term radiolysis, PAR is credited to maintain an inert CNV
- ❖ Exemption request #2
  - ❑ Post accident monitoring of H_2 and O_2 not required to assess core damage. Assessment to be accomplished by core exit thermocouples and radiation monitors beneath the bioshield.

NuScale SDAA FSAR Chapter 6 Review

Section 6.2.7 Fracture Prevention Containment Vessel

- Significant differences between NuScale DCA FSAR and NuScale SDAA FSAR include:
  - F6NM replaces SA-508, Grade 3, Class 2 from the previous design for the upper CNV and a portion of the lower CNV below the upper/lower vessel flange.
  - Staff verified that material change would not result in significant impacts on fracture toughness management of CNV.
  - Staff conclusion did not change from DCD.

NuScale SDAA FSAR Chapter 6 Review

Section 6.3 Emergency Core Cooling System (ECCS)

Significant differences between NuScale DCA FSAR and NuScale SDAA FSAR include:

- Addition of passive Emergency Supplemental Boron (ESB) feature.
  - Chapter 14 includes first of a kind test
  - Extended Passive Cooling topical report and SDAA 15.0.5 contain boron transport methodology and analysis
- Removal of Inadvertent Actuation Block Valves (IABs) on Reactor Vent Valves (RVVs) – IABs retained for Reactor Recirculation Valves (RRVs).
- Inclusion of flow restricting venturis in RVVs and RRVs.
  - Exclusion of flange breaks from LOCA break spectrum evaluated in SER 15.6.5
- ECCS actuation signals changed to RPV riser level.
  - Chapter 15 review confirms modeling of the riser level sensor
- 8 hour timer actuates ECCS valves after an automatic or manual trip
  - Recirculates boron from ESB into core to maintain subcriticality
  - Vents accumulated combustible gas from radiolysis

Presentation to the ACRS Subcommittee
Staff Review of NuScale SDAA FSAR, Revision 1

Chapter 17, “Quality Assurance and Reliability Assurance,”
Section 17.4, “Reliability Assurance Program”

February 18, 2025

(Open Session)

NuScale SDAA FSAR Section 17.4 Review

Overview

- NuScale submitted Chapter 17, “Quality Assurance and Reliability Assurance,” Revision 0 of the NuScale SDAA FSAR on December 28, 2022, and Revision 1 on October 31, 2023.
- NRC performed a regulatory audit as part of its review of Chapter 17, Section 17.4, from March 2023 to June 2024.
- Questions raised during the audit were resolved within the audit. One RAI was issued, and the response was acceptable.
- Staff completed the review of Chapter 17, Section 17.4 and issued an advanced safety evaluation to support the ACRS Subcommittee meeting.
- No significant changes between draft SE provided to ACRS on 1/18/25 and SE provided on 2/12/25

NuScale SDAA FSAR Section 17.4 Review

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NuScale SDAA FSAR Section 17.4 Review

Significant Changes from DCA to SDAA

- EDAS provides power to maintain ECCS valves closed during normal operation and contributes to defense in depth in the design.
  - Reactor vent valves do not include an inadvertent actuation block valve.
- Safety-related PAR added to maintain the containment atmosphere inert during design-basis events and significant beyond-design-basis events.
- Safety-related steam generator system and safety-related components in the control rod drive system are not identified as risk-significant in FSAR Table 17.4-1
  - These SSCs perform the same system functions in the US600 design and were identified as risk significant in the DCA.

NuScale SDAA FSAR Section 17.4 Review

Conclusion

- Augmented design requirements for EDAS are comparable with the design requirements for D-RAP SSCs.
- SER Section 6.2.5 concludes that the safety classification of the PAR is acceptable.
- The SGS and CRDS components are safety-related and subject to the requirements of the QAPD TR described in FSAR Section 17.5.
- The staff finds that the design and quality requirements…
  - for EDAS, the PAR, SGS, and the safety-related CRDS components meet the intent of the Commission policy stated in item E of SECY-95-132.
  - resulting from the classification of SSCs is consistent with the intent of guidance in SRP Section 17.4.

United States Nuclear Regulatory Commission
Protecting People and the Environment

Presentation to the ACRS Subcommittee Staff Review of NuScale SDAA FSAR, Revision 1

Chapter 19, “Probabilistic Risk Assessment and Severe Accident Evaluation”

February 18, 2025

(Open Session)

NuScale SDAA FSAR Chapter 19 Review

Overview

- NuScale submitted Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation," Revision 0 of the NuScale SDAA FSAR on December 31, 2022, and Revision 1 on October 31, 2023
- NRC regulatory audit of Chapter 19 was performed from March 2023 to August 2023, generating 173 audit issues
- Issues raised during the audit were resolved within the audit. 6 RAIs (15 Questions) were issued, and the responses were acceptable
- Staff completed Chapter 19 review and issued an advanced safety evaluation to support today's ACRS Subcommittee meeting
- Since providing draft SE to ACRS on 1/18/25, Table 19.1-4 was updated to include COL Item Nos. 19.1-7 and 19.1-8, which were inadvertently missed from the draft SE

NuScale SDAA FSAR Chapter 19 Review

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NuScale SDAA FSAR Chapter 19 Review

Sections

- 19.1 Probabilistic Risk Assessment
- 19.2 Severe Accident Evaluation
- 19.3 Regulatory Treatment of Nonsafety Systems
- 19.4 Strategies and Guidance to Address Mitigation of Beyond-Design-Basis Events
- 19.5 Adequacy of Design Features and Functional Capabilities Identified and Described for Withstanding Aircraft Impacts

NuScale SDAA FSAR Chapter 19 Review

Significant Changes to Risk Profile Between DCA and SDAA

- Core Damage Frequency (CDF)
  - CDF increased due to more frequent actuation of ECCS valves.
  - Dominant contributors to CDF include high winds, module drop, external floods, internal events, and internal fires.
- Large Release Frequency (LRF)
  - LRF decreased due to earlier actuations of ECCS valves.
  - Contribution to LRF from breaks outside containment decreased.
  - Addition of digital reactor building crane control system minimizes operator error.

NuScale SDAA FSAR Chapter 19 Review

Focus Areas for PRA and SA Review

- Impact of changed ECCS actuation setpoints
- Augmented DC power system (EDAS) modeling
- CVCS line breaks outside containment
- Unisolable CVCS breaks outside containment
- Density wave oscillation (DWO) impact on Steam Generator Tube Failure (SGTF)
- Addition of reactor building crane (RBC) digital control system
- Top Support Structure (TSS) connection to RBC
- Addition of passive autocatalytic recombiner (PAR)

NuScale SDAA FSAR Chapter 19 Review

Impact of ECCS Actuation Changes on CDF and LRF

- Approximately 90 percent of core damage scenarios involve incomplete ECCS actuation.
- Low RCS level (top of the riser) and Low Low RCS level (mid-riser) result in earlier ECCS actuation.
- 8-hour ECCS timer added; Operator action to bypass timer after checking shutdown margin and hydrogen concentration found not to be a significant human action.

NuScale SDAA FSAR Chapter 19 Review

EDAS Modeling in PRA

ECCS reactor vent valves held closed by EDAS

- Not identified as risk significant from PRA importance measures.
- Single failure proof system.
- Physical separation between divisions.
- Failure of two channels of module-specific EDAS results in reactor trip and ECCS actuation.
  - CCFs not modeled between electrical buses in separate compartments
  - Data for EDAS CCF modeled in PRA is derived from operating plant data where DC power is safety-related
- FSAR states that EDAS will be included in the Owner Controlled Requirements Manual (OCRM) and the Maintenance Rule.

NuScale SDAA FSAR Chapter 19 Review

CVCS line breaks outside containment

- Flow restricting venturis in injection and discharge lines control inventory loss and reduce LRF from CVCS line breaks outside of containment.
- If at least one train of the DHRS is available and all ECCS valves are open, PRA success criteria are met.
  - Pumped injection via CFDS and CVCS is not needed for scenarios where all ECCS valves open in contrast to the DCA.

NuScale SDAA FSAR Chapter 19 Review

Unisolable CVCS breaks outside containment

- The likelihood of weld failures at the junction between the containment vessel and the CVCS containment isolation valves are not modeled in the PRA.
- The plant behavior and consequences of an unisolable CVCS LOCA outside of containment are modeled through the CVCS break downstream of containment isolation with failure of containment isolation.
- The low weld failure frequency is identified as a key source of Level 2 uncertainty.
- The impact on LRF is minimized by leak detection and operator response.

NuScale SDAA FSAR Chapter 19 Review

DWO Impact on SGTF Initiating Event and PRA Results

- PRA did not explicitly model impact of DWO on SGTF.
- Staff considered worst-case hypothetical impacts of DWO on PRA results.
  - Multiple SGTF
  - Loss of both trains of DHRS
- NuScale sensitivity analyses demonstrate that the PRA results and insights are insensitive to the SGTF initiating event frequency and a loss of both trains of DHRS.

Non-Proprietary

NuScale SDAA FSAR Chapter 19 Review

<u>RBC Control System Reduces Module Drop Contribution</u>

- The RBC digital control system significantly decreases the contribution of operator errors of commission.
- Dominant contributors to module drop are redundant load path failures (i.e., catastrophic gear box and wire rope failures)
- The RBC digital control system is classified as non-safety related, risk significant, and SIL3.

NuScale SDAA FSAR Chapter 19 Review

TSS Connection to Module Crane

- If a dropped module strikes an operating module, piping, including pressurizer spray piping and DHRS piping, at the front of the NPM has the potential to be impacted**.**
- The safety-related CVCS CIVs location under the TSS protects these CIVs from postulated dropped NPM impacts.
  - The TSS is classified as non-safety related and risk significant in FSAR Table 17.4-1.
- If the CIVs close but both trains of DHRS are unavailable, if one RSV successfully cycles open and closed, as needed, the RCS depressurizes, and the ECCS is demanded.
- If the RSVs fail to open, ECCS functioning remains a success path.

Non-Proprietary

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

NuScale SDAA FSAR Chapter 19 Review

Addition of PAR

- A single safety-related passive autocatalytic recombiner (PAR) was added to the design.
- The PAR is not modeled in the PRA.
- Equipment survivability dose for PAR:
  - Post severe accident, the two functions that must be maintained are containment integrity and post-accident monitoring.
  - The PAR has been added to Table 19.2-8, “Equipment Survivability List.”
  - A new COL Item 19.2-4 states that the COL applicant will identify from Table 19.2-8, “Equipment Survivability List,” the components and their severe accident doses for cases in which the severe accident dose is greater than the EQ dose, as described in COL Item 19.2-4

NuScale SDAA FSAR Chapter 19 Review

Conclusion

- Staff reviewed the NuScale US460 design-specific PRA and other PRA-related information in FSAR Section 19.1, in accordance with:
  - SRP Section 19.0.
  - DC/COL-ISG-028 for applicable modes and hazards
- The applicant addressed the full scope of internal and external initiating events for both full power and LPSD conditions.
- The applicant developed quantitative risk insights for multi-module internal events and qualitative risk insights for multi-module shutdown and external events.
- The PRA is of sufficient technical adequacy to support the SDA.
- The staff's review concludes that the Commission's CDF and LRF goals have been met with margin.

NuScale SDAA FSAR Chapter 19 Review

Conclusion

H_2 Combustion in the CNV

- ❖ The DCA addressed a potential combustion event in the CNV analytically and demonstrated that the CNV design pressure was not exceeded.
- ❖ SDAA added a PAR which precludes combustion events from occurring during DBAs and SAs.

Containment Performance (no combustion)

- ❖ SDAA Table 19.2-1, "Core Damage Simulations for SA Evaluation", identifies the spectrum of severe accidents that may challenge CNV integrity.
- ❖ SDAA Tables 19.2-2 – 19.2-7 document that CNV design pressures, including H2 generated, are not exceeded.

Conclusion

The applicant addressed severe accidents consistent with Commission policy.

SDAA design for containment performance meets:

- ❖ the containment structural integrity criteria of RG 1.7, rev 3, "Control of Combustible Gas Concentrations in Containment."
- ❖ the containment leak tight criteria of SECY-93-087.

NuScale SDAA FSAR Chapter 19 Review

BDG Evaluation for RTNSS

- BDGs not scoped into RTNSS
  1) Do not prevent the occurrence of an initiating event
  2) Not needed for long-term, post-accident plant capabilities
  3) Not needed to support defense-in-depth systems
- All components of the backup power supply system, including the BDG enclosures, are seismic Category III.
- The BDG enclosure is rated for wind speeds in excess of the weather-related events considered in the LOOP initiating event.

Criterion C: SSC functions relied to meet the Commission goals for CDF < 1×10-4/yr and LRF < 1×10-6 /yr and SSCs needed to maintain initiating event frequencies at the comprehensive baseline PRA levels (SECY-94-084)

NuScale SDAA FSAR Chapter 19 Review

Conclusion

- Staff has reviewed the NuScale US460 evaluation of RTNSS SSCs in FSAR Section 19.3, in accordance with:
  - SRP Section 19.3.
- NuScale did not identify any SSCs in the scope of RTNSS.
- Staff finds that no SSCs meet the criteria for requiring additional regulatory treatment.

NuScale SDAA FSAR Chapter 19 Review

Aircraft Impact Analysis

Adequacy of Design Features and Functional Capabilities Identified and Described for Withstanding Aircraft Impacts: Structural

- Steel-Plate Composite Walls (only applicable to SDAA)
  - Both global and local assessment use experimental data to benchmark the methodology and results
  - Followed NEI 07-13, Revision 8 with no exceptions

- Additional key design features (only applicable to SDAA)
  - Strengthen SC wall to RC slab connections
  - Local detailing with tie rods in SC wall-to-wall connection
  - Structural steel beam seat connections along RX-B and RX-D

- Credit RWB as Intervening Structure to limit potential strike locations to the west end of the RXB (only applicable to DCA)

Integrated Review Approach – Communication Tool

TOPIC – 5 Principles of Risk Informed Decisionmaking	
Principle 1: Meets current regulations or exemption requested	• Yellow indicates applicant/licensee has provided some information on the topic. Staff still needs information, but there's a clear path forward.
Principle 2: Consistent with the defense-in-depth philosophy	• Green indicates that all reviewers agree that applicant/licensee has provided sufficient information. • E.g., backup systems that are available to mitigate the event
Principle 3: Maintains sufficient safety margins	• Red indicates that there is broad agreement that applicant/licensee did not provide information to make a regulatory finding. There is no clear path forward.
Principle 4: Increase in risk is small and consistent with the intent of the Commission's Safety Goal Policy Statement	• Integrated review team is established among technical review branches and risk analysts to align on a decision considering all 5 principles of RIDM.
Principle 5: Performance measurement strategies available for monitoring	

Acronyms

BDG	Backup Diesel Generator
CCF	Common Cause Failure
CDF	Core Damage Frequency
CFDS	Containment Flood and Drain System
CIV	Containment Isolation Valve
COL	Combined License
CVCS	Chemical and Volume Control System
DCA	Design Certification Application
DHRS	Decay Heat Removal System
DWO	Density Wave Oscillations
ECCS	Emergency Core Cooling System
EDAS	Augmented DC Power System
EQ	Equipment Qualification
FSAR	Final Safety Analysis Report
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LRF	Large Release Frequency
NPM	Nuclear Power Module
OCRM	Owner Controlled Requirements Manual
PAR	Passive Autocatalytic Recombiner
PRA	Probabilistic Risk Assessment
RBC	Reactor Building Crane
RCS	Reactor Coolant System
RSV	Reactor Safety Valve
RTNSS	Regulatory Treatment for Non-Safety-Systems
SBO	Station Blackout
SDAA	Standard Design Approval Application
SGTF	Steam Generator Tube Failure
SRP	Standard Review Plan
TSS	Top Support Structure

Meeting Title	**Open Session NuScale Subcommittee on Staff's Evaluation of NuScale SDAA Chapters 6 and 19 and Section 17.4**

Attendee

Michael Snodderly	ACRS (DFO)
Getachew Tesfaye	NRR
Shandeth Walton	ACRS
Larry Burkhart	ACRS
Thomas Dashiell	ACRS
Wendy Reid	NuScale
Matt Sunseri	ACRS
Augi Cardillo	NuScale
Brian Kanen	NuScale
James Cordes	Court Reporter
David Nold	NRR
Milton Valentin	NRR
Robert Martin	ACRS
River Rohrman	NRR
Meghan McCloskey	NuScale
Steven Bloom	NRR
Ron Ballinger	ACRS
Steven Wolbert	NuScale
Kris Nerczuk	NuScale
Brandon Haley	NuScale
Cindy Williams	NuScale
Amanda Bode	NuScale
Alina Schiller	NRR
Stephen Schultz	ACRS
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Scott Palmtag	ACRS
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Elisa Fairbanks	NuScale
PJ Evans	NuScale
Joe Remic	NuScale
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Angelo Stubbs	NRR
Mahmoud -MJ- Jardaneh	NRR