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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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NuSCALE SUBCOMMITTEE

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OPEN SESSION

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TUESDAY

FEBRUARY 4, 2025

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

SUBCOMMITTEE MEMBERS:

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DENNIS BLEY

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

MIKE SNODDERLY

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P R O C E E D I N G S

8:30 a.m.

CHAIR KIRCHNER: Good morning. I'm Walt Kirchner, Chair of today's Subcommittee meeting.

ACRS members in attendance in person are Ron Ballinger, Vicki Bier, Greg Halnon, Craig Harrington, Bob Martin, Scott Palmtag -- Dave Petti will join us shortly -- and Thomas Roberts. ACRS members in attendance virtually via Teams are Vesna Dimitrijevic and Matt Sunseri.

We have one of our consultants participating in person, Steve Schultz, and one of our consultants participating virtually Via Teams. That's Dennis Bley. If I've missed anyone, either ACRS members or consultants, please speak up now.

Michael Snodderly of the ACRS staff is the Designated Federal Officer for this meeting. No member conflicts of interest were identified. We have a quorum as well for today's meeting.

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, Sections 3.7, 3.8, and 3.9.2, and Chapter 5, Reactor Coolant System and Connecting Systems, including the Committee's area of focus on the potential for density wave oscillations occurring in the steam generators.

We previously reviewed the certified NuScale US600 design, as documented in our July 29, 2020, letter reporting the safety aspects of the NuScale small modular reactor.

Like the staff, we are performing a delta review between the two designs, including the 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 141, 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, therefore, should be regarded as only the individual opinion of that member and not a Committee position.

All relevant information related to ACRS activities, such as letters, rules for meeting participation, and transcripts, are located on the NRC public website and can be 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 also set aside time at the end of this meeting for any public comments.

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 the 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 are not speaking, please mute your computer on Teams or by pressing *6 if you're on your phone.

Please do not use the Teams chat feature to conduct sidebar discussions related to presentations, but, 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 we have live ceiling microphones.

For the presenters, these table microphones are quite unidirectional. You'll need to speak directly into the front of the microphone, particularly so the court reporter can transcribe today's session.

Finally, if you have any feedback for the

ACRS about today's meeting, we encourage you to fill out our public meeting feedback form on the NRC's website.

And with that, we will now proceed with the meeting. I will turn to the NRC staff.

Please go ahead, M.J.

MR. JARDANEH: Good morning, Chair Kirchner, and good morning to the ACRS Subcommittee members, NuScale, the NRC staff, and members of the public.

My name is Mahmoud Jardaneh, or M.J. I serve as the Branch Chief for the New Reactor Licensing Branch, responsible for the licensing of the NuScale US460 design, in the Division of New and Renewed Licenses in NRR.

Okay. Today, the staff will be presenting their review of a group of the SDAA Chapters, including Sections 3.7, 3.8, and 3.9.2 of Chapter 3, Design of Structures, Systems, Components and Equipment, and Chapter 5, Reactor Coolant System and Connecting Systems.

Earlier this year, the staff presented to the Subcommittee on Chapters 2, portions of Chapter 3, Chapters 7, 8, 9, 10, 11, 12, 13, 14, 16, portions of Chapter 17 and Chapter 18. The staff also presented on the LOCA, on the Loss of Coolant Accident Evaluation Model Topical Report associated with the application.

The staff is finalizing their review of the remaining SDAA chapters and Topical Reports, and we will inform the ACRS on the safety evaluations on the remaining chapters and Topical Reports that are available today to the ACRS.

At today's meeting, the staff will focus on the deltas from the Design Certification that the NRC has already approved and that the Subcommittee reviewed in the past.

Once again, thank you for the opportunity and we look forward to begin the session. Thank you.

CHAIR KIRCHNER: Thank you, M.J.

And with that, I think we'll turn to Tom Griffith of NuScale. Okay?

MR. GRIFFITH: Thank you.

Good morning, ACRS Members. Good morning, NRC counterparts and members of the public, as well as our NuScale counterparts out on the West Coast.

I am Thomas Griffith, licensing manager for the NuScale US460 Standard Design Approval Application. I've been with NuScale for, roughly, three years. I have a background as a former senior reactor operator/I&C manager, and it's background in re-occurrence (audio interference) as well as safety analysis.

Today, we are very excited and look forward to the opportunity to present the remaining sections of Chapter 3, as well as Chapter 5, and then, in the closed session, we'll touch on additional portions of the density wave oscillation topic itself.

So, with that, I'd like to turn it back over to my counterparts here to start the presentation.

DR. KARAOGLU: Thank you.

Good morning. My name is Haydar Karaoglu. I'm a civil engineer with a PhD from Carnegie Mellon University. Over the past five years, I have been with NuScale specializing in seismic analysis and design of structures, as well as the seismic analysis of the NuScale power modules.

Today, we will delve into the differences between the Certified Design and the Standard Design Approval Application for Chapter 3, which covers design of structures, systems, components and equipment.

MR. GRIFFITH: This is Thomas Griffith.

We do appreciate the Department of

Energy's support for the NuScale US460 Standard Design Approval Application, appreciate their support and all the efforts that they made out to the system thus far.

DR. KARAOGLU: Thank you. Yes, thank you.

So, for Chapter 3, we will on Sections 3.7, 3.8, and 3.9.2, Seismic Design, Design of Category I Structures, and Mechanical Systems and Components. Note that some sections, such as concrete containment, are excluded because they are not applicable to the US460 NuScale Power Plant design.

Next slide, please.

This slide here is the summary of key design features and updates.

The Standard Design Approval Application, SDAA, is a derivative of the certified design, design certification, DC.

SDAA structures reflect six modules, in support to the 12 modules in the certified design. And the difference necessitated updated structural analyses.

For the SDAA, the reactor building uses steel-plate composite walls, along with reinforced concrete members.

And the site layout reflects the updated building designs.

Seismic analyses for the SDAA are performed for a double-building model, which features the reactor building and the rad waste building and a separate surface-based control building model. The Certified Design, on the other hand, used a triple-building model and individual building models for the seismic analyses.

Finally, this presentation will focus on the high-level design and methodology changes, and important audit questions and requests for additional information, RAIs.

Next slide, please.

We begin with Section 3.7, which is seismic design. Section 3.7.1 addresses seismic design parameters.

For the percentage of critical damping, the Certified Design used separate fully cracked and fully uncracked models, and all the reinforced concrete members had the same damping ratio of 7 percent for the design calculations.

The SDAA, on the other hand, employs hybrid models with both cracked and uncracked members. The damping in the structural members varies based on their cracking status, as well as the purpose of the calculation, whether for the in-structure response spectra calculations or for design purposes.

More details of the damping values for essential critical damping are available in the Topical Report titled Building Design and Analysis Methodology for Safety-Related Structures.

Regarding the supporting medium, in the Certified Design, we had four generic soil profiles, representing rock, firm soil/soft rock, hard rock, and soft soil profiles, named as Soil-7, -8, -9, and -11 respectively.

In the SDAA, the Soil-8 profile is removed and, based on the Safety Analysis, the soil-separation scenario with the Soil-7 is included in the design basis.

There were no audit questions or RAIs for this section.

Next slide, please.

Section 3.7.2 covers seismic system analysis.

In the Certified Design, soil-structure interaction, SSI, analyses were performed using the extended subtraction method with the software SASSI.

In the SDAA, the SSI analyses are performed using the soil library methodology, which is a robust approach equivalent to the direct methods of

SASSI. In this method, the soil libraries are built using SASSI and the simulations are performed with ANSYS, leveraging fluid-structure interaction technology of the software.

As they are presented in this figure, using this methodology, it could model all different soil structures by soil, building, and fluid together and simulate the soil-structure interaction and fluid-structure interaction simultaneously.

More details of the methodology are available in the Topical Report entitled Improvements in Frequency Domain Soil-Structure-Fluid Interaction Analysis.

Another difference between the Certified Design and the SDAA is in the combination of the responses to three components of the ground motion.

In the Certified Design, the maximum responses were calculated using the square-root-of the sum-of the squares method.

In the SDAA, the responses from the three, statistically independent components of the ground motion are algebraically added.

Next slide, please.

SSI Numerical models using the seismic system analysis, you've seen this figure, the double-building model, featuring the reactor building and the rad waste building, which are in dark gray here, and the engineered backfill surrounding them, in blue.

In the SDAA, the reactor pool is modeled with FLUID elements of ANSYS and using the fluid-structure interaction technology. And the six NuScale power modules, NPMs, are modeled in detail using advanced features of ANSYS.

In the Certified Design, the pool was modeled as distributed mass and the 12 NPMs were modeled as simplified beam models, made of mass, spring, and beam elements.

Thirty-three questions were resolved in audit for this section, resulting in updates in the Final Safety Analysis Report, FSAR. Updates cover modal analysis, double-building model dimensions, and pool sloshing.

There were no RAIs for this section.

Next slide, please.

Section 3.7.3 addresses seismic subsystem analysis.

The SDAA includes updates to major subsystems, including the bioshields, the reactor building crane, and the NPMs.

For the SDAA, we developed three different

NPM models.

A simplified NPM model, which is represented in the figure here, is used in the SSI analysis to calculate the seismic demand on the structural members.

A detailed NPM model is used in the SSI analyses to calculate the seismic response around the pool.

And another detailed NPM model, which was developed using superelement technology of ANSYS, was used in the nonlinear transient analysis of the NPMs.

A summary of the models and the methodology are available in Appendix 3A. Also more details are provided in the Topical Report titled US460 NuScale Power Module Seismic Analysis.

Next slide, please.

In the SDAA, the nonlinear NPM seismic analyses are performed using a comprehensive local model that includes the six NPMs, the pool, and the surrounding structural members.

The local model used in the SDAA is shown in this figure here.

In the Certified Design, the NPM seismic analyses were conducted using a local model which included only one NPM at a time, the pool, and a rigid plane under the NPM.

No details of the methodologies are available in the Topical Reports listed in this slide.

For this section, four questions were resolved in the audit, resulting in additional bioshield details in the FSAR.

And there were no RAIs.

Next slide, please.

Section 3.7.4 covers seismic instrumentation.

In the SDAA, the locations and descriptions of the seismic instrumentations are updated due to the new layout of the buildings.

There were no audit questions or RAIs for this section.

Next slide, please.

CHAIR KIRCHNER: Haydar, may I ask a few questions?

DR. KARAOGLU: Sure.

CHAIR KIRCHNER: Oh, it went off. Let's try again.

First, what did you see as the result of your analyses with a different level in the reactor building pool versus loads on the modules? Did you see any noticeable difference because of lower water level? In other words, is there any impact in terms of seismic forces put on the individual modules?

DR. KARAOGLU: It's difficult to comment on that computing Certified Design and the Standard Design Approval Application because the models are different significantly --

CHAIR KIRCHNER: Right.

DR. KARAOGLU: -- as well as ours with 12 modules and the other six. So, even if we saw differences, I think it's not really possible to point to the pool level individually to say that that is the source of the difference. But I am sure that the pool level had some impact on some results.

CHAIR KIRCHNER: So, overall, did you see higher stresses, seismic stresses, as a result, or lower? In other words, what was the net impact of the pool on the modules?

DR. KARAOGLU: I understand that. Yes, it's kind of difficult to -- just specifically focusing on the pool, of course, the fact that it was lower definitely reduces the hydrostatic forces that we used, that's for sure, on the structural members and on the NPMs as well.

CHAIR KIRCHNER: Right.

DR. KARAOGLU: However, in terms of the

dynamic aspect of it, as I said, there are multiple differences between the models. So, I don't think it's really possible to point to a certain difference and say that it's because of the pool.

CHAIR KIRCHNER: What about buoyancy? So, you have a module, essentially, a little ship inside the pool. What about forces, buoyancy forces, and such, stresses in the building?

DR. KARAOGLU: It's like for the --

CHAIR KIRCHNER: The building is holding more load with a lower level, right?

DR. KARAOGLU: That's true. Well, I mean, the pool, compared to the Certified Design, the pool volume is lower; hence, the mass is less --

CHAIR KIRCHNER: I see.

DR. KARAOGLU: -- than what we had before. But the building size is also different. It used to be much lower in one direction --

CHAIR KIRCHNER: Right.

DR. KARAOGLU: -- compared to what we have now.

But regarding the buoyancy, yes, because the pool level is lower, the buoyancy on the NPMs is reduced as well.

CHAIR KIRCHNER: It looks like the

compartments are tighter. Is there any sloshing impact that you see in a seismic event?

DR. KARAOGLU: We looked at sloshing and I don't remember the numbers right now exactly, but our conclusion was that the maximum sloshing that was to be calculated in accordance with the equations provided in the proposed standards, they were not significant.

CHAIR KIRCHNER: Then, could you just kind of summarize for us -- you went to much more sophisticated models; much more higher fidelity is maybe a better way to say it. Did you see any noticeable differences, for example, for forces? You did the square root of the sum of the squares, and then, the updated methodology. Now, you're going in actually three directions, adding -- how did you say it? -- geometric or algebraic --

DR. KARAOGLU: That was algebraic ground forces.

CHAIR KIRCHNER: Yes. Did you see any noticeable difference in the seismic impact on the modules?

DR. KARAOGLU: I would say that the differences, it's not really possible -- and again, I'm conflicted in myself, but it is really in the same way, I think. The methodology may not be the only source of the differences that we are observing because the structural members are significantly different.

CHAIR KIRCHNER: Okay.

DR. KARAOGLU: But regarding the comparison of the two methods, like the algebraic addition of the times versus the SRSS method, you know, one approach is about -- this is, basically, captured in the new behavior, the average behavior.

CHAIR KIRCHNER: Right.

DR. KARAOGLU: And both the methods are acceptable, according to the Regulatory Guides and standards.

CHAIR KIRCHNER: Thank you.

MEMBER HALNON: Haydar, while we're off-script, this is Greg Halnon.

On a previous slide, one of the differences beyond methodology, I guess, was that you included six NPMs in the SDAA and you did the DC one at a time. Can you tell me what the impact of that decision is relative to, in a DC, theoretically, I guess, with the seismic analysis, each NPM stands on its own, is that correct, because you did model one at a time. How does that translate into the six NPMs?

Can still one module stand on its own from an analysis perspective?

DR. KARAOGLU: Yes. Well, you know, no model is perfect. So, they're all approximations. But we believe that in this approach that we followed in the SDAA, by putting all six NPMs together in this local model, you could capture the interaction between the NPMs much more accurately; also, thanks to using the fluid and soil-structure interaction.

But in the earlier methodology as well, it's also a valid approach. Most of these approximations are based on engineering judgment. So, for example, using a single NPM, you would expect, maybe because of the pool size getting larger, that a single bay becoming more dominant in capturing the enveloping demand on a single NPM.

And also, it's simple to say that, you know, in that model, the pool model was represented as less distributed mass. So, that's also an approximation.

So, I don't know if that answers the question, though.

MEMBER HALNON: Yes, well, I mean, when we get further into this presentation today, in the 3.9 section that we talk about, we're going to talk about prototype versus non-prototype plants with the Reg Guide 1.20.

And the first NPM is going to be a prototype. My question is, is this contrary -- not contrary -- but it is conflicting a little bit that they have to have one plant stand on its own, both from vibration and seismic, and everything else, in order to say that the rest of it is okay? So, you almost get a non-prototype.

I know we'll get to that in the future. But I was curious, in this DC, you did a single one, and then, you integrated the 12 together to show that all 12 would be fine. In the six NPMs, the SDAA, was that similar? You took all six; you modeled all six together, but you did still get the individual interactions on each module, adjacent modules, and that sort?

I'm trying to get a picture in my mind how that's going to work down the road. Maybe when we get to 3.9.2, we'll talk a little more about prototype/non-prototype and how those figure into that. I assume you'll assume be here, and if there's any questions, you can --

DR. KARAOGLU: Right. Again, yes, I believe that is something that I hope to discuss later on.

But just a brief response to your comment here, by having all six modules in the same model, we are kind of trying to represent that seismic waves traveling into the pool, from the ground to the pool, and all the interactions between the NPMs and their structural members around it, we tried to capture it as accurately as possible using advanced technology or enhanced --

MEMBER HALNON: Okay. So, it's a step forward -- I mean, from the standpoint of the DC, Walt said you use much more sophisticated modules and you're able to integrate it better.

DR. KARAOGLU: Right, right.

MEMBER HALNON: Okay. Thanks. We'll talk more about the prototype, and this question in my head may go ahead at that point, but we'll talk later with the staff this morning.

Thanks.

DR. KARAOGLU: Okay. Continuing with Section 3.8, which is design of Seismic Category I structures.

Section 3.8.2 addresses steel containment.

The differences of the SDAA from the Certified Design include the following:

An increase in design pressure and temperature for power uprate.

Material change from carbon steel with cladding to a combination of austenitic and martensitic stainless steels.

Pre-service and in-service inspections are changed from Class 1 to Class MC vessel with additional requirements in some areas.

Hydrogen detonation is removed from load combinations because of additional passive autocatalytic recombiners, the details of which are available in Chapters 6 and 15.

The majority of nozzles are changed from welded to integrally-forged.

Twelve audit questions were resolved.

And for this section, there were no RAIs.

Next slide, please.

Section 3.8.4 addresses other Seismic Category I structures.

In the SDAA:

The reactor building incorporates steel-plated composite walls which are designed according to AISC N690, 2018 version, using element- and panel-based approaches.

Reinforced concrete members are designed according to ACI 349, 2013 version, using the section-cut demands at critical locations.

The forces are calculated from numerical models with different cracked states associated which are associated with different load combinations.

And all the simulations are performed using ANSYS with the use of SASSI for soil library calculations.

A summary of the technology and results are reflected in Appendix 3B, and more details of the methodology are available in the Topical Report titled Building Design and Analysis Methodology for Safety-Related Structures.

In the Certified Design, the major structural members were of reinforced concrete type, and they were designed according to ACI 349, 2006 version, using an element-based approach. The simulations were performed using SASSI for the SSI analysis and SAP2000 for the other load combinations.

Fifteen questions were resolved in the audit, resulting in the updates in the FSAR. The updates cover: dynamic soil pressure, differential settlement analysis, definition of the supporting medium used for calculating the static load demands, and the design and analysis procedure.

There are no RAIs for this section.

Next slide, please.

Section 3.8.5 covers foundations.

In addition to the analysis and design of foundations, this section also covers the stability analysis of the structures.

In the SDAA, the nonlinear stability analysis is performed only for the Seismic Category I portion of the surface-based control building.

Also, the peak-bearing pressure values are calculated using a methodology tailored to the capabilities of the software utilized, which was ANSYS.

Twelve questions were resolved in the audit for this section.

And there were no RAIs.

Next slide, please.

Okay. I will turn it over to Emily Larsen now.

MS. LARSEN: Hi. I'm Emily Larsen, and I am a licensing engineer at NuScale. Previously, I was a system engineer at Braidwood Power Station, and then, I did design and analysis of hydraulic components. And I've been at NuScale about a year and a half.

Section 3.9.2 will finish out the presentations for 3.9. 3.9 was held from the rest of the presentations to allow time for the analyses for the DWO Safety Case to complete and allow TF-3 flow testing to complete and testing data to be analyzed.

Differences from the DC to the SDAA include:

Updating the comprehensive vibration assessment program, Regulatory Guide 1.20, to revision 4.

Updating the requirements for the ASME Operations and Maintenance Code to the 2017 edition.

The comprehensive vibration assessment program startup instruction changed from strain gauges and accelerometers to dynamic pressure sensors.

COL Item 3.9-14, the DC density wave oscillation carve-out, was removed.

Reactor vessel internals and flow-induced vibration analyses were updated for US460 loads, design changes, updated flow rates, and operating conditions, as appropriate.

An analysis case of both reactor vent valves actuating was added to the NuScale Power Module Short-Term Analysis Technical Report.

Next slide, please.

All audit questions were resolved and resulted in:

Updating the language for the NPM prototype classifications to match the NuScale Comprehensive Vibration Assessment Program Analysis Topical Report.

Providing a summary of flow testing results for review for TF-3.

And providing analyses to show the structural integrity of the steam generator during DWO.

There was one RAI, and we provided the preliminary Service Level D fatigue results for the reactor vessel internals and the steam generator components.

And this resulted in no changes to the SDAA.

Next slide, please.

Section 3.9.2 also supports the analyses pillar of the Safety Case for DWO.

Audit questions resolved on this topic were resolved and there were no RAIs.

The DWO Service Level A transient, along with the NPM lifetime limit for time in DWO is in Section 3.9.1.

And structural integrity of the steam generator during DWO supports Section 3.9.2.

Next slide.

And that ends the Chapter 3 presentations.

MEMBER HALNON: So, Emily, this is Greg Halnon.

You probably heard prototype versus non-prototype, the question that we were asking. Since the first module in operation will be the prototype, and every other module beyond that for Reg Guide 1.20 is going to be a non-prototype, can you explain how that's going to work with six modules being developed at the same time? And is it because of the huge amount of margin that you have that you're confident you can re-analyze any potential parameters that out of scope or out of range?

MS. LARSEN: So, the first module is a prototype. All other modules are going to be instrumented, so that, as they are prototyping, they won't be prototyped until the first goes through its final CVAP inspection program.

MEMBER HALNON: The entry of models to prototype, until you get at least one that's identical, and the rest of them can follow along, as long as they're instrumented and everything is within the bounds of the scope of the analysis, and they'll be non-prototype from here on out at that point. Is that the proposal, I guess?

MS. LARSEN: Yes. Unless there's some changes, and then, there may be a limited prototype.

MEMBER HALNON: Okay. Well, that makes sense.

And I had one other question for you. And I'm going to ask this question as a mechanical engineer and not civil engineer. So, the first word out of your mouth is going to be over my head.

The area's intensity, one of the measurements was outside of the 6-second range that you look for strong motion. There was very little justification why it was okay, but it seemed to be okay. Could you just give us a quick summary on why, when we're targeting, trying to get strong motion in that 5 to 75 percent range in the area's intensity, that this one is okay at 5.2? Are you familiar with what I'm talking about?

DR. KARAOGLU: Is this -- there's one that's less than 6 seconds --

MEMBER HALNON: Yes, the station is met in 1999. That one was 5.265 seconds.

DR. KARAOGLU: Right. Yes, it's the

easement. Yes, well, there, the justification is that the strong ground motion of the area's intensity is a way of quantified the strong ground motion. And those are some guidelines about how to calculate it.

However, if you look at the time history plotted and see that strong ground motion, how much oscillation, and what is that oscillation range? You can see that, independent of the area's intensity, you actually can see the strong ground motion starts earlier. It's because of, if, for example, we were to take much longer data for the time series, than we would see that area's intensity already catching up with that 5 percent limit way earlier.

MEMBER HALNON: Okay.

DR. KARAOGLU: So, that's why, in addition to the area's intensity, it's important to visually justify if that range is good for the strong ground motion.

MEMBER HALNON: Okay. Good job. I appreciate it.

DR. KARAOGLU: Thank you.

DR. SCHULTZ: Emily, with regard to the -- this is Steve Schultz -- with regard to the vibration assessment program and the change from the instrumentation to the dynamic pressure sensors, can you describe the advantage that provides for this design?

MS. LARSEN: It actually allows for the entire steam generator to be monitored at the same time, instead of just having a couple of tubes instrumented.

DR. SCHULTZ: So, that's the main advantage?

MS. LARSEN: Yes.

DR. SCHULTZ: Thank you.

MEMBER HALNON: So, Emily, this is Greg again.

Is that why you characterize this as extensive instrumentation as opposed to just -- anytime I see a word like extensive, it makes me wonder, well, what was it before, not extensive?

(Laughter.)

MS. LARSEN: I just want to add real quick, the dynamic pressure sensors also monitor the reactor vessel internals. And extensive is a word used in Regulatory Guide 1.20 -- and, yes.

(Laughter.)

MEMBER HALNON: Yes. So, I guess I was wondering how you met the term extensive. And I guess it's because you have this ability now to measure whole components and whole internals, what's going on.

Okay. Thanks.

DR. SCHULTZ: Steve Schultz again.

You described a large number of changes and advances in your seismic analysis for this design. And there are very few RAIs -- actually, no RAIs in the presentations that you made here were provided to the staff, but a lot of audit questions were answered. How does that relate to the changes that you've made? In other words, were you targeting that kind of performance as you made the changes? What do you credit for the resulting review by the staff? Very smooth, I would say, but, as you interacted with the staff, how would you describe that interaction?

DR. KARAOGLU: So, library methodology?

DR. SCHULTZ: Yes.

DR. KARAOGLU: Well, the methodology itself is actually, you know, it's relative, but it's, actually, rather straightforward. It's (audio inference) to the well-established direct methods of SASSI. The advantage is particularly in the computation. Initially, we paid a price for a demanding calculation for the soil library. However, later on, when we performed the harmonic analysis, they are smooth and quick. And also, we incorporated most features of ANSYS into the simulation.

So, because the underlying theory is straightforward, it's just a condensation of the impedance matrices. I believe that the evaluation of this and, also, in the Topical Report, we also showed how good it works through verification tests. So, I think that's why the procedure was smooth -- and to justify the use of this methodology, soil library methodology. Even though it's a significant difference, the advantages it brought were significant. However, the methodologies of the underlying theory is straightforward.

I don't know if that answers the question.

DR. SCHULTZ: That's very helpful.

With regard to the seismic forces and the database that was used to derive them, how would you characterize that with regard to, if you will -- I saw what you've chosen. How does that fare with regard to the seismic forces and systems that need to be evaluated, let's say, across the United States? Is it a bounding evaluation? I know, for COLs, the licensees are going to have to demonstrate that their site will either be enveloped or do additional calculations. Is your expectation that it will not be a problem for COLs, I'll just say, in the United

States, because we've got great experience associated with that?

DR. KARAOGLU: Right. Well, there are different aspects of it, of course, beginning with the design response spectra. So, our design response spectra, the Certified Design response spectra, it covered most of the sites in the U.S. And also, the rocky sites, because we had this high-frequency version of the response spectra as well. So, that's one aspect of it.

In regard to the response spectra, we are enveloping most sites in the U.S., but that's just one aspect. The other one, that soil-structure interaction is very much dependent on the soil profile itself. And by looking at very hard rock and very soft soil, we tried to address a wide range of soil properties. So, it's important to ensure that we see that they are calculated using the soil, local site properties, and make sure that their demands are enveloped with what we calculated.

However, you know, just making a general statement like that would be really difficult because there can be some special sites with very different profiles. For example, for most of the site, it might be enveloped by our soil profiles, but very close to the surface there might be some layers with, like, much softer layers, and then, they might amplify certain frequencies. So, it's still necessary to make sure that things are enveloped, the demands are enveloped with what we calculated.

DR. SCHULTZ: It's necessary to validate that?

DR. KARAOGLU: Right.

DR. SCHULTZ: And even what you've just described as a fairly straightforward approach to doing the evaluation, is that fair to say, that it will not be difficult for a COL applicant to perform that evaluation?

DR. KARAOGLU: Yes. Speaking for the SSI analysis --

DR. SCHULTZ: Yes.

DR. KARAOGLU: -- the methodology we are following is, because it's equivalent to the direct method of SASSI, that shouldn't be difficult.

DR. SCHULTZ: Thank you.

MEMBER PALMTAG: This is Scott Palmtag.

I just had a question about the seismic analysis. In the NuScale design, there's a lot of things moving around compared to a standard reactor, where things are pretty much stationary.

You're going to have one, two, three, four, five, six modules. In addition, you're going to be moving modules around. They might be in the refueling bay. They might be under a crane. How does that go into the seismic analysis? Are those relatively small changes that you can bound? Or do you actually have to do a seismic analysis for all of these different configurations that can happen?

DR. KARAOGLU: In the sensitivity analysis, we look into modularity. And our analysis shows that it's not a significant difference. It doesn't make a significant difference on the demand that will be within the structural members.

Also, it might be worth to point at the NPMs. A single NPM's mass is significantly small compared to the whole mass of the reactor building. So, you know, all these analyses, it doesn't really require a highly detailed model to be used to address wherever the NPM is located at.

MEMBER PALMTAG: Okay. So, all the different configurations are relatively small compared to the ability to --

DR. KARAOGLU: Yes.

MEMBER PALMTAG: Thank you.

CHAIR KIRCHNER: Haydar, I had another

question, just in terms of bracketing things.

So, with the steel-plate/concrete composite structure for the reactor building, you mentioned cracked/uncracked. Could you just give us an assessment? What's the impact if you have significant cracking in terms of the structural integrity, the seismic response of the building --

DR. KARAOGLU: Sure.

CHAIR KIRCHNER: -- cracked versus uncracked? Or what are you looking for when you do that analysis and what does that tell you?

DR. KARAOGLU: Sure. So, it's very good material for compression, but retention is weak. So, under seismic load, it cracks. Once it's cracked, what happens is that its thickness increases and, also, density increases. So, it's, basically, absorbing more of that seismic energy.

So, as a result of how widespread that cracking is, the behaviors of the structure change.

CHAIR KIRCHNER: Right.

DR. KARAOGLU: It gets, for example, lower frequencies.

CHAIR KIRCHNER: Right.

DR. KARAOGLU: Its natural frequency decreases.

So, by addressing the cracking using hybrid models, we tried to capture the dynamic characteristics of the building, the variation in the dynamic characteristics of the building under seismic conditions.

CHAIR KIRCHNER: And if it were fully cracked, and hence, as you just described, do you see any amplification in the seismic loads for the module, transferred to the module or any of the equipment, including above? Does the dynamics of the reactor building response materially impact any of the questions that would arise regarding the safety of the modules?

DR. KARAOGLU: The only time we see a cracked scenario is in the SDAA. So, I cannot really say much about it. But I should state that --

CHAIR KIRCHNER: How do you bound that then? You know, what spectrum of cracking do you look at?

DR. KARAOGLU: For the cracking, the way we decide on that is, you know, we calculate the -- after SSI analysis, we look at the demands on the structural members and, you know, compare the stresses with the cracking stress, obviously, on the (audio interference) walls, and then, we assign change in the material properties. Seeing how widespread it is in the model, we make it either cracked or uncracked.

Now, that definitely changes the forces --

CHAIR KIRCHNER: Right.

DR. KARAOGLU: -- being transferred. The load, you know, it changes. But how do we envelope that? Well, previously, by design, an uncracked model and a fully cracked model we'll use to kind of look at the two extremes, right?

CHAIR KIRCHNER: Extremes, right.

DR. KARAOGLU: But by looking at the hybrid model, we are actually kind of following each of the (audio interference). Because we actually start with an uncracked model and we perform this assignment, and seeing how widespread it is, we change the properties, and then, we run it again.

CHAIR KIRCHNER: Right.

DR. KARAOGLU: So, that way, we are kind of trying to follow the variation in dynamic cracking in the building. So, it's, either way, enveloping that variation.

I don't know if that answers the question.

CHAIR KIRCHNER: Well, I'm thinking through it. If you have a substantially cracked -- I'm not sure how to phrase it. If you have a significant amount of cracking in the steel-plate composite building, my sense is that, because it's steel-plate composite, you still retain the structural integrity, much more so than a reinforced concrete reactor building with a liner. In this case, we're presuming that the inner surfaces of the steel-plate composite is, indeed, also the pool liner. Or is there an additional liner?

DR. KARAOGLU: One of those, the composite is just the --

CHAIR KIRCHNER: It's just the inside surface? Right.

DR. KARAOGLU: Part of the surface, yes. Right.

Yes, but in regard to that, maybe I should point to AC 416 or 43, that it's basically, even under the cracked case, you know, we are modeling the whole structure as inelastic. So, even in that phase, you know, we are not assuming any significant damage to the building. You know, everything is intact. The reinforced concrete is also intact as well.

CHAIR KIRCHNER: So, to follow up on that earlier question by Scott, what about the building crane and having a module in transit, or something? Yes, in the overall picture, the mass of a module isn't that much. But once you've picked it up, it's out of order, partially out of order, whatever you're doing with it, and it's on the crane, the crane support structure, is that impacted in any material way by significant cracking in the steel-plate composite structure? Or is the steel-plate composite structure sufficient without the concrete bearing -- as you said, the concrete is creating compression. So, holding up the crane and everything. But is the steel-plate structure sufficient on its own to support those loads?

DR. KARAOGLU: Yes. Based on our calculations for the demand on the steel-plate composite walls, they are sufficient the way we designed them to resist those forces. But I should maybe point out that, compared to the seismic demand created by the seismic excitation, the reactor building crane and the impedance on the structural members, the effects are mostly local.

CHAIR KIRCHNER: Thank you.

MEMBER ROBERTS: Yes, Tom Roberts.

I'm looking for a little perspective on the removal of the detonation loads from the scope containment. I know that it's out of scope of this discussion. I'm sure we'll get into it with Chapters

6 and 15, the uncertainties and the ability of the PARs to remove hydrogen, some of the uncertainties in that methodology or that phenomenology.

So, by removing it from the analysis, do you have any sense of what the withstandability to hydrogen detonation remains? Is that degraded by some design change or is it just your (audio interference)?

DR. KARAOGLU: We'll need to give that to my colleagues on the call to answer that question. If this is something that we can take on at this point?

MR. GRIFFITH: Yes, Thomas Griffith, Licensing Manager.

So, we do expand the discussion in Chapter 6 on that. I would say that our position is that the PAR provides sufficient protection against the events that are postulated there. And I think that our analysis shows the PAR is safety-related. It performs sufficiently as sufficient design margin.

As far as your question on loads, I would have to confer with the Chapter 6 LEs on that specific question. But I don't think it's -- you know, what we were able to demonstrate in the review, I think, is that the event is not going to happen. The PAR is well designed for that.

MEMBER ROBERTS: Okay. Thanks. We could

sort of come back to that in Chapters 6 or 15. And if we need to come back to the withstandability of the structures to deal with it, we can come back to then subsequently.

Thanks.

CHAIR KIRCHNER: Since you're showing acronyms, I'm presuming we're at the end of your presentations. Is that correct, Tom? I don't have the slides in front of me.

MR. GRIFFITH: That is correct.

CHAIR KIRCHNER: Okay. So, Members, any further questions of NuScale in these sections of Chapter 3? No?

Okay. Then, we'll just take a momentary pause here and change out and ask the staff to come forward.

Thank you. Thank you.

MR. SNODDERLY: Chair Kirchner, I appreciate the great interaction between you and the Applicant. Just so that you know, we're about a half-hour behind.

CHAIR KIRCHNER: Yes. Thank you very much.

DR. CHOWDHURY: Good morning.

This is Prosanta Chowdhury.

If I missed it, are we at a break right now?

MR. TESFAYE: Yes. This is Getachew Tesfaye.

Our lead project managers for Chapter 3 and Chapter 5 are not able to join us in person. So, that they are leading the meeting virtually.

CHAIR KIRCHNER: Yes.

MR. TESFAYE: Thank you.

CHAIR KIRCHNER: Getachew, who's up first?

MR. TESFAYE: For Chapter 3, Prosanta.

CHAIR KIRCHNER: Are you first?

DR. CHOWDHURY: Yes.

MR. TESFAYE: Prosanta.

CHAIR KIRCHNER: Oh, Prosanta?

DR. CHOWDHURY: Yes.

CHAIR KIRCHNER: Okay.

DR. CHOWDHURY: Yes. Good morning.

CHAIR KIRCHNER: Okay, Prosanta, go ahead, go ahead, thank you.

DR. CHOWDHURY: Okay, good morning, thank you. Good morning, ACRS members, NuScale counterparts, NRC colleagues, and members of the public. My name is Prosanta Chowdhury. I am a senior project manager in the branch of New Reactor Licensing under Division of New and Renewed Licenses at the NRC.

I have been with the NRC for about 20 years, 17 of which I have been a project manager. My background is I have a master's in nuclear engineering and also a master's in electrical engineering. I have previously worked for the State of Louisiana in the radiation protection field for 18 years.

So I am the project manager for Chapter 3, and I will present the slides, and at the appropriate times I will hand the microphone over to the presenters.

So with that, please let me know if you can see the slides. Okay.

CHAIR KIRCHNER: Yes, we have the slides up.

DR. CHOWDHURY: Thank you, thank you. So this is the presentation to the Advisory Committee on Reactor Safeguards Subcommittee. A staff review of NuScale's US460 standard design approval application final safety analysis report, Revision 1. And these are sections are Chapter 3, Sections 3.7, 3.8, and 3.9.2.

This slide shows the technical reviewers that contributed to these sections of the FSAR review, Sunwoo Park, Scott Stovall, Ata Istar, Zuhan Xi, Luisette Candelario-Quintana, Yuken Wong, and Stephen Hambric is the consultant. The lead project manager is Getachew Tesfaye, and I am Prosanta Chowdhury again.

An overview of these sections of staff review. So NuScale submitted Chapter 3 of Revision 1 on October 31, 2023. NRC performed a regulatory audit as part of its review of Chapter 3 from March 2023 to June 2024. Questions raised during the audit are resolved within the audit. All RAI responses were acceptable.

So this is a blanket statement because we are not listing the number of RAIs in all these questions. They have been reflected in the appropriate sections of the safety evaluation, which was released to the public on January 30 this year.

Staff completed the review of these sections of Chapter 3 and issued an Advanced Safety Evaluation Report to support the ACRS meeting. Now, on January 4, staff submitted a draft Safety Evaluation to ACRS for a preliminary review, and there have been some changes, some updates.

Section 3.7 was updated regarding acceptability of strong motional time history being less than 6 seconds. Staff will elaborate that later if that question comes up.

Section 3.8 was updated regarding demand over capacity ratio values for reactor building, calculated and assessed by both element-based and panel section-based approaches.

NuScale SDAA Chapter, FSAR Chapter 3, these sections have some subsections. As listed here 3.7 has four subsections, 3.8 has five. And then 3.9.2 we have several topics that will be covered later.

So with that, we start with Section 3.7.1, and I'd like to turn the microphone over to Dr. Sunwoo Park.

Sunwoo, please go ahead.

MR. PARK: Thank you, Prosanta.

Good morning, I am Sunwoo Park, data and risk analyst (phonetic) at the NRR Division of Risk Assessment. I have been with the agency for 17 years now, previously serving as a structural engineer in the NRR Division of Engineering.

Although my current role focuses on seismic PRA, I was requested to support the review of NuScale SDAA seismic design because I reviewed also the TGA seismic design when I was a structural engineer. It was part of the inter-organizational collaboration efforts.

Yeah, before moving into the slide that I prepared, if I may briefly comment on the issue that was discussed concerning slow motion duration. Initially the staff did not explicitly evaluate that because staff thought the -- anticipated the impact would be minimal, but in the updated SER, staff reviewed it, and then they provided a steady variation in the SER, which was completed as unacceptable. The staff specifically reviewed the areas of intensity curve and noted that there was a quite steep slope on the curve and around 7 -- 5% and 75% time mark, which indicates quite strong shaking under that region, 5% and 75%.

So, effectively that indicates the slow motion invasion practically is graded at 6 second. So it is acceptable.

MEMBER HALNON: Something like that, maybe it’s just the only thing I could find that was really out of the norm. You mentioned that you didn’t have it in the -- I didn’t see the revised SER you had, so I’ll take a look at that.

The original SER you felt like, just from your experience, that that was minimal impact, so you just didn’t mention it at all basically. Is that the reason it wasn't in the original SERs from the minimal perspective?

MR. PARK: Yes, but you know, it was 5.3 seconds at the range of 60 second, the threshold. And also because there are multiple time histories are considered by the time histories, which are all accounted for in developing design basis. So I thought that the impact would be minimum.

MEMBER HALNON: Okay, and you completely agree with the NuScale rationale behind that?

MR. PARK: Yes, I reviewed the, this case the assertion in the FSAR, and confirmed that is acceptable.

MEMBER HALNON: Okay, thank you for adding that, I appreciate it.

MR. PARK: In Section 3.7.1, seismic design parameters, there are significant differences between DCA and SDAA, including structural damping values using seismic measures. In DCA, the reinforced concrete was used for safety-related structures and applied a uniform 2% damping value for both cracked and uncracked reinforced concrete members to generate in-structure response spectra.

Then SDAA, the two different types of structural material are used, including reinforced concrete and steel plate combined. And NuScale utilized the hybrid damping scheme to generate ISRS, including 7, 5, and 4 and 3 percent, depending on the cracking condition, whether cracked or uncracked, and also on the material type, whether reinforced concrete or steel composite.

In both cases, cracked and uncracked ISRS is the envelope enough to establish design basis. And the staff concluded that SDAA damping values are acceptable because they are consistent with the Reg Guide 1.6.1, the latest update in Revision 2, published in 2023, yeah. Just stop me if you have a question. Next slide please.

Another interest was in supporting media for seismic Category I structures. DCA, as mentioned earlier by NuScale, considered four different supporting media, including soft soil, firm soil/soft rock and rock and hard rock. By contrast, SDAA utilized just three supporting media: soft soil, rock, and hard rock.

In both cases, seismic response from each soil type enveloped to generate the design basis. And staff found the supporting media for SDAA are still acceptable because they reasonably represented a range of expected site soil conditions. Next slide, please.

In Section 3.7.2 on seismic system analysis, significant differences include the different methodologies for seismic soil-structure-fluid interaction analysis. DCA employed a two-step methodology to address the SSFI effects involving separate soil structure fraction and fluid structure interaction analysis, which involved a certain level of basically simplifications and approximations.

Whereas in SDAA, the single integrated methodology was evaluated to evaluated the defense. And the -- that new methodology used for SDAA was -- it is based on the topical report which was reviewed by the staff and approved in 2022. And, yeah, 2022.

And because the methodology was already approved in the topical reports. And also because staff verified that the analysis was performed in compliance with the applicable limitations and conditions specified in the topical report. The methodology is acceptable. Next slide, please.

The differences also included the different analysis models associated with design changes, which includes -- including six NPMs and updated NPM models, and the resized ultimate heat sink with a reduced water volume, water depth. And the relocated CRB and the new steel composite walls.

And also, DCA employed a Triple Building Model, which includes reactor building, control building and rad waste building for design basis seismic demand calculations. To whereas SDAA used a Double Building Model, including reactor building and rad waste building and also considering control building independently.

And the updated models were acceptable because they adhere to applicable industry standards and DSRS specific review standard acceptance criteria. Next slide, please.

The differences are also identified in different -- in the approaches to addressing the results of parameter sensitivity studies. Both DCA and SDAA conducted in-structure response spectra -- spectrum sensitivity analysis to evaluate the parameter variations, including structure-soil separation, empty dry dock, and the modularity.

In those cases, in both DCA and SDAA, the soil-separation scenario resulted in a noticeable exceedance of the design-basis ISRS. And there are different approaches to the addressing this exceedance between DCA and SDAA.

DCA addressed the exceedance by including a COL Item, referring to COL applicant to make sure that the site-specific ISRS soil-separation condition is bounded by DCA design basis ISRS.

Whereas in SDAA, the NuScale incorporated the soil-separation scenario into the design basis analysis cases, separate at the analysis cases, which is a preferred approach here from staff’s point of view and it is acceptable -- yeah, it was acceptable. It is now, it makes the, one of the design basis case. Next slide, please.

In Section 3.7.3, on seismic soil system analysis, the differences were -- was identified in seismic analysis of a building is varied. Seismic Category I piping, conduits, and tunnels.

These data do not include varied piping or conduits. But at least have, it included tunnel connecting reactor building and control building. And the tunnel was analyzed as part of the control building.

In SDAA, there was -- there is a underground pipe long underground in first counted duct bank containing conduits that connect -- that connect to reactor building and to control building. And the staff, they confirmed that the analysis was conducted in accordance with applicable industry standards and DSRS acceptance criteria.

The Section 3.4 -- 3.7.4 on seismic instrumentation was that reviewed by my colleague Scott Stovall. And he identified no significant differences between DCA and SDAA. So we're going to skip that and move on to Section 3A, presented by Ata, unless there is any question on 3.7.

CHAIR KIRCHNER: Sunwoo, can I ask a question? Since you highlighted this underground duct to connect the control, main control room with the reactor buildings, what's your figure of merit for success?

The implication I'm reading here is just that it meets codes and standards, but did you analyze whether there was displacement? Or did you look at their analysis to see displacement? Was there another figure of merit in terms of survivability of cabling and so on as that's contained in that structure?

MR. PARK: The detailed calculations on the underground -- the conduit, was not provided in the FSAR, rather there was a qualitative description how the analysis and the design were connected and stated that NuScale followed the guidance provided in the ASTE (phonetic).

CHAIR KIRCHNER: So from the civil structural standpoint, it meets the applicable standards and acceptance criteria?

MR. PARK: Yes.

CHAIR KIRCHNER: Okay. Thank you.

MR. ISTAR: Good morning. My name's Ata Istar, I'm a structural engineer at NRR, and I've been with the NRC over, well, almost 20 years. Prior to NRC I worked in, I started in the nuclear industry in 1979 working at the firms Entergy and Constellation and NRC.

And I'll be covering Section 3.8. And as Haydar described earlier, we had 12 audit questions for Section 3.8.2, which is the steel containment, and 15 audit questions 3.8.4 for other structural and Category I. So other-sized in Category I structures.

And each audit question had multiple requests under it. In Section 3.8.2, seismic containment, the design parameters are slightly different. And of course the material, as Haydar earlier mentioned. And what we realized in the, during the discussions, they also changed the, reconfigured the boundary conditions between the bottom of, head of the CNV and the RPV.

And in the DCA space, there was a pin connection, one single pin connection at the bottom heads connecting from OD of RPV to ID of CNV. With shims and -- shims located at each phase.

And the new configuration has three logs on the OD of RPV and clevis screws on the inside diameter of CNV. So I think this is a better configuration than a pin connection concentrated at the bottom. So staff cited that's a better improvement supporting the reactor vessel and the containment connection.

CHAIR KIRCHNER: But the other design was hardly a pin. It was significant structure.

MR. ISTAR: It was insert, some kind of insert.

CHAIR KIRCHNER: Okay.

MR. ISTAR: So basically design parameter is slightly higher, as I listed in the presentation. But of course the steel containment was designed for those conditions, and we had no issue with that. Next slide, please, Prosanta.

The other seismic Category I structures, and we had -- I would think this, we have sections into this. There's a methodology, which was presented for the development of this approach, which is SC walls. And there was a topical report that was provided to us. We reviewed it in detail, accept it, and it was accepted by the ACRS as well.

So that was a major change from the reinforced concrete design to SC wall, applicational SC walls with combination of reinforced concrete substructures with it.

I think this is a big improvement compared to the DCA design from numerous aspects. One, it's faster constructability of the reactor building. And much better seismic capability, as well as its aircraft impact assessment. So those are important elements that improve this SDAA design using the SC.

Again, the one thing we found during the review, of course the NuScale did both -- I should credit that, element-based and panel-based sections, which is an important thing.

And panel-based, section-based, panel section-based approach is provided in the N690, in SEN690, which is one times the thickness of the sea wall at the edges, at the corners, and two times of the thickness of the SC walls in the middle sections, which is accepted by these both.

And as we were reviewing the demand over capacity ratios, certain very, very localized areas were higher than 1.0. And I'm not sure we should elaborate this at this point or maybe in the closed section.

It's up to you, sir. And just, or NuScale, whether we should discuss that and how we concluded those exceedances at three locations in the reactor building are acceptable. It's up to you, I can elaborate this. As a matter of fact, I brought this big picture here so I can locate those.

CHAIR KIRCHNER: Well, let's, if I may, go back to 2022, don't hold me to the date exactly, when we did review the steel plate composite topical report from NuScale, and you approved that. And we thought you should issue that.

One of the areas that we highlighted, and perhaps you could take up, if not here, in the open session and the closed session, is connections. And I think you're hinting at that. And I think one would ask about leak-tight integrity of the structure as you -- with fasteners and connectors and from the base mat to the side walls.

Can you address that either now or in the closed session?

MR. ISTAR: Based on the design, and I mean, we're looking at the structural integrity of the, all of the building.

CHAIR KIRCHNER: Right.

MR. ISTAR: Not the leak-tightness here.

The leak-tightness comes in the construction capability of this at the -- at the, you know, steel plate, SC plate, goes into the reinforced concrete. It’s buried into it. And there, in the structure and the plate, bottom plate, there is another bottom plate on the foundation connecting to the vertical SC wall.

And that weld should provide, that joint weld should provide the leak-tightness at that location. And that is a critical weld for leak-tightness perspective. But we, in this section, 3.8.4 section, we’re looking the integrity of the structure. Leak-tightness is not the element that we will look at.

We’ll assume that weld is appropriately done and provides structural integrity of the walls as well as the foundation.

CHAIR KIRCHNER: But in your review, then, you would look at the -- that weld, that series of welds, actually, that comprise the, in effect, the liner equivalent that was in the DC design where you had a reinforced concrete building with a liner. So at some point, where do you look at the integrity of that weld in terms of a massive leak from the reactor pool?

MR. ISTAR: I just got a question. That

particular weld does not provide any integrity of the overall structure. The weld that has --

CHAIR KIRCHNER: No, understood.

MR. ISTAR: Okay, that does not provide anything except the leak-tightness, right, at that location. And in the, I recall in the design certification, there is, you know, that's a reinforced concrete structure that it's a huge, I think five-foot thick walls with pilasters. And you know, it's a major -- the cost of that is huge compared to this one.

But that location is, I don't recall exactly how it's -- probably it's similar to the SDAA configuration. But the liner, the liner comes in and butts into the -- butts into the SC wall face steel. And it should, there is a weld there at the -- I think that's all I can tell.

And whether there -- you know, I think, as I understand from your question whether under any seismic event or something that weld has some --

CHAIR KIRCHNER: Say you had a major fracture --

MR. ISTAR: Fracture at that location.

CHAIR KIRCHNER: -- along the length of the weld.

MR. ISTAR: Well, if you’re looking at the design over capacity ratios, and this is deeply embedded structure, deeply embedded structure, and we are -- if your design over the capacity ratios at that location is low. And remember, this is a linear elastic regime. It’s, we don’t have any plastic deformations at that location.

So as long as we are within the linear elastic area, we should not have any cracks at that location. It’s below the -- I think it’s over 80 foot below the ground level. And you are kind of confined into this space.

And they actually, which we’ll hopefully discuss that later, design over capacity ratios which are over one, they are upper sections, in the upper sections. Not at that location. They all meet the -- they are well below the --

CHAIR KIRCHNER: The N690 or?

MR. ISTAR: They’re, I don’t think there’s going to be any fracture under any external load conditions.

CHAIR KIRCHNER: Good, that’s what I wanted you to answer in the public session. Okay, thank you.

MR. ISTAR: And I think -- should we

discuss the design over -- demand over capacity ratio discussions?

CHAIR KIRCHNER: Can you just summarize?

MR. ISTAR: Yeah, I can summarize. There, you know, there are three locations. NuScale staff who are members, engineers, identified there -- demand over capacity ratios are larger than 1. Which is, the highest one is 1.05. That will be -- the other one is 1.04. The other one is 1.02.

And we closely look at those locations. And I can maybe -- because it's hard to explain it. I pull up, this is the DCA design.

DR. CHOWDHURY: This is Prosanta, Ata. I apologize for interruption, but I assume it's okay, but please make sure that we are not bringing up any proprietary information in this section. Thank you.

MR. ISTAR: Thank you. This is a old design, but I just, the reason I'm pointing this out, the shear wall numbers are similar. So this wall is RX1, where the main entrance is. And as you can see, the structure's deeply embedded. And I think this is 83 feet underground.

And the one that is the highest demand over capacity ratio is at this little location. And the second one is at this location on the opposite side, not on this side. On the opposite side. And it was in the application, revised application, I should say, this was described as due to the geometric discontinuities, that was developed.

And in the finite element analysis, there are always glitches when you have geometric discontinuities. You have high peak stresses in areas that are not -- they're very cornered --

CHAIR KIRCHNER: Corners, fasteners, yeah.

MR. ISTAR: -- openings and that kind of thing. And please note that these, the results are based on the high seismic conditions that, you know, conservative loading combinations. And with the all soil types, that was all soil types. So these are the maximum worst conditions, you could see it.

You can see it, it's in the upper sections, in the higher elevations. And very, very concentrated area.

They're, you know, from a structural perspective, if you want to, you know, if they want to make those numbers lower, there are two things that can be done. As I told your earlier panel, this is the result for a panel section-based results.

CHAIR KIRCHNER: Right.

MR. ISTAR: And panel section-based

design, it can be merged with adjacent members, which will reduce the stresses. Or if that area, I don't believe it's critical to the overall structural behavior, the face plates could be enlarged and make that area reduced.

I don't think any of these things are necessary because these are developed due to the mathematical calculations in the finite development analysis. And these are very, very localized. And as you can see it, it doesn't affect the big overall structure. It's a very localized location.

So that's all -- I can elaborate this --

CHAIR KIRCHNER: In the closed session, thank you.

MR. ISTAR: -- more if you like. I think next section is 3.8.5 foundation. I would like to ask Zuhan contribute to this section.

Thank you.

MR. XI: Hi, my name is Zuhan Xi. I have been with the agency for 18 plus years. I am currently a geotechnical engineer --

CHAIR KIRCHNER: Pull that microphone closer to you.

MR. XI: Okay, I'm sorry.

CHAIR KIRCHNER: Yeah, just reintroduce

yourself so the court reporter gets it.

MR. XI: Yeah. I start over again. My name is Zuhan Xi. I'm with agency for 18-plus years. And I'm a geotechnical engineer. And previously I was a structural engineer. Prior to joining the NRC, I was a contractor at The Fairbank Highway Research Center. I was a research structural engineer.

I reviewed 3.8.5 and which is the foundation. Yeah, the big difference is, you know, I noticed for the embedment of the control room building, which is in the SDAA. The CRB is modeled as a surface-founded structure, conservatively, ignoring the five-feet embedment of the foundation of its stability analysis.

In the DCA, the CRB with an embedment that's of 55 feet is modeled is as embedded structure with the backfill. So that's the major, you know, difference between the SDAA and a DCA. So SDAA, SC conclusion is the same as DCA SC type conclusion.

CHAIR KIRCHNER: Just for the record, could you state what that conclusion is?

MR. XI: The conclusion is the safety integrity is with the limits.

CHAIR KIRCHNER: Thank you.

MR. WONG: My name is Yuken Wong, I'm a

senior mechanical engineer in the Mechanical Engineering and In-Service Testing Branch. I have been with the NRC for 18 years. And prior to that, I've worked for Westinghouse for 15 years.

The review of Section 3.9.2, dynamic testing and analysis, involves three main areas. First is the piping vibration and thermal expansion testing. Second is the comprehensive vibration assessment program, or CVAP, of the reactor vessel internals, which include steam generators.

The staff reviewed two technical reports relating to flow-induced vibration analysis and testing and inspection of reactor vessel internals.

Third area is the analysis of reactor vessel internals under ASME Service Level D conditions. Those are the earthquake events and loss of coolant accidents.

The staff reviewed two technical reports that provide the seismic loads and the short-term transient blowdowns. The staff also reviewed the stress and deflection analysis. Next slide, please.

For the DCA, there were deferred or unresolved issues. The qualification of steam generator components due to the DWO was a carve-out. The validation testing to demonstrate the steam generators is not at risk to FIV. It was deferred to the COL applicant.

The Service Level D evaluations did not include the hard rock high frequency seismic loads. And those were deferred to the COL applicant. Next slide, please.

For the SDA CVAP, there were significant changes. Higher flow velocity lead to stronger FIV loads. NuScale introduced a temperature approach method in the later part of the SDAA review to limit operating conditions that, where COL may occur and produce the DWO loads.

The steam generator inlet flow restrictors were redesigned and no longer a risk to increased flow instability.

The steam-generated tube supports are changed to provide more surface area, more compact surface area and provide improved dimensional variability. The secondary flow piping branches are changed and improved, minimize the risk to acoustic resonance.

A qualification of steam generator due to the DW load is no longer a carve-out. NuScale performed the steam generator validation testing, which confirmed there's a minimum risk to FIV. Next slide, please.

In the DCA and early part of the SDA reveals --

MEMBER PETTI: I'm sorry, go back to the previous slide.

MR. WONG: Okay.

MEMBER PETTI: Last bullet suggests there's not a big risk of flow-induced vibration. The previous slide basically said that it, they hadn't demonstrated that significant flow-induced vibration in the steam generator tubes. Those seem contradictory.

MR. WONG: That's correct. This slide refers to the SDA, what's current now for the SDA. And the previous slide was highlighting --

MEMBER PETTI: Oh, the DCA --

(Simultaneous speaking.)

MR. WONG: -- in the DCA.

In the DCA or early part of the SDA review, there were concerns that during high amplitude, reverse DWO flow to phase region in the steam generator tube may approach the inlet, leading to a cavitation and condensation-induced water hammer.

There was no limit on the number of DWO cycles during the life of plant, so significant numbers of cycles were made cumulatively. Next slide, please.

For the review of the steam generator due to DWO loads, it's based on three-tiered approach. First, boiling boundaries are highly unlikely to approach the steam generator inlets, even during DWO. So cavitation and condensation-induced water hammer, highly likely would not occur.

The Chapter 5 review confirms that NuScale's analysis provides reasonable prediction of the boiling heights. The NRC Office of Research performed the independent analysis using the computer code TRACE. The results show condensation-induced water hammer is highly unlikely.

In the unlikely event this mechanisms occur, NuScale calculated the steam generator tube and inlet flow restrictor erosion due to cavitation and the tube stress due to liquid slack (phonetic) flow. Results show damage to the components is not like.

Finally, the steam generator program inspection would detect any unexpected wear for modules. One hundred percent of the tubes will be inspected during the first refueling outage, and after that, at least 72 effective full power months.

I'm going to turn over for the review of the remaining of Section 3.9.2 to Dr. Hambric.

DR. HAMBRIC: Hi, everyone, I'm Dr. Steve Hambric. I've been a consultant for the NRC for about 20 years now, going back to boiling water reactors steam fire failures. And I've worked in flow-induced vibration and noise for over 35 years, primarily for the U.S. Navy, but also U.S. industry, both in managements and simulations.

I will point out that we do have a bunch of backup slides if you guys want to dig deeper into the DWO stuff or anything I'm about to tell you here. We can do that in the closed session.

A lot of work on steam generators in the SDAA. So the next topic is making sure they were not subject to significant FIV due to vortex shedding and fluid-elastic instability. Those are mechanisms that can make these tubes shake around a lot and potentially fail over time.

NuScale had built, actually several years ago, a pretty nice scale model facilities, actually full scale, but it's not as -- it's not all the tubes, in Piacenza, Italy, at the SIET facility. And had not tested it at the end of the DCA.

But they did test it that past summer, and we actually went on site and looked at the facility, made sure it looked good to us. Had some preliminary flow results. They look good as well.

And the two big things we were after were, number one, is it a valid facility. And so they were able to prove to us by looking at vibration measurements that they had a nice, tightly fitting setup. All the tubes were really well connected to the support system.

That's important to us because if we had a sloppy connection, that would induce a bunch of damping that would essentially invalidate the entire test. You'd never be able to get a flow-induced vibration instability going if you had a loose, sloppy system. But they got a nice, tight system, looks good.

The other thing we were concerned about is when they built this, it was an old design with a support system. New design is a little bit different, it's better. Wanted to make sure that the support system wasn't going to somehow invalidate the test, and it will not. It's a good, tight facility, and we're quite happy with it.

The neat thing they were able to do in this facility that they couldn't do in the real actual NPM is they could crank up the power to 250% and higher, really get the flow moving. And did not see any evidence of anything untoward in any of their tubes. So we're very happy about that, no vortex shedding, no FEI.

We've got a bunch of detailed stuff we can show you in the closed session if you like, but it looks like nice, linear response all throughout. So we're quite confident that these steam generators should not experience significant vortex shedding or FEI in service.

MEMBER HALNON: So Steve, this is Greg, so all the clearances, everything for a leakage flow type, you looked at all those and they were, even though it had a lot fewer tubes, it was prototypical enough to be able to see the, what is it, like six or seven different flow-induced vibration type phenomena?

DR. HAMBRIC: Yeah, the only two we're worried about here are vortex shedding, which is individual tubes and the vortices behind them shaking the tube up and down and locking in, and fluid-elastic instability, where multiple tubes can kind of grab onto each other and start moving significantly.

There's no really concern about leakage flow instability in the vortex -- in the steam generator. We did evaluate that phenomenon in the rest of the plant. In particular, there’s a hose in the support place for the CRDS tubes to go through and the ICIGs to go through. So we looked at leakage flow there.

MEMBER HALNON: And the reason I ask --

DR. HAMBRIC: But not in the steam generator.

MEMBER HALNON: Okay, the reason I ask in the steam generator is because the flow issues will be on the outside of the tubes. And to inspect that, I haven’t got a clear view in my head how that’s going to be inspected after a certain amount of operation. And it’s going to be done visually, I guess.

So that’s kind of the reason I asked. The structures around the outside of the tubes and the clearances and whatnot that’s holding in place, whether or not there would be any problems. So it’s sounds like you’ve --

DR. HAMBRIC: If there were clearances, we would have seen that in the flow-induced vibration. There would have been kind of a lot of sloppiness in the vibrational elements we were seeing in the spectrum. We didn’t see any of that.

MEMBER HALNON: Maybe a little sloppiness, not a lot of sloppiness. All right, just trying to put it in context.

DR. HAMBRIC: Yeah, we saw almost none. The damping was tidy. It looks like a nice, tight fit. When you turn the thing on and you've got the fluid in there and it's pushing everything together, everything's tight, so.

MEMBER HALNON: Good, thanks.

DR. HAMBRIC: Once again, we've got more details. They've put together a really nice report summarizing their results. And it just looks as good as we could have hoped it to be. Next.

One other TF-3 related test that they did for us is, if you remember when NuScale presenting, they did change pretty significantly the initial startup testing instrumentation. In the DCA, they were going to individually instrument several tubes with accelerometers, strain gauges to directly measure the vibration during startup.

And in the SDAA, they said no, we're going to switch to dynamic pressure sensors scattered throughout the plant. And a couple of good reasons for that.

As Emily pointed out, really able to hear anything. If it's an individual tube instrumented, you might not hear it if another tube is vibrating.

But if you have the pressure sensors, you can pretty much hear anything crazy happening throughout the plant.

The other nice thing about it is you don't have wires and other stuff feeding through the flow and potentially causing some problems. Instrumentation is falling off. It's a much cleaner approach having the external for the -- or dynamic pressure sensors.

But the one thing they really didn't show us until recently was that those were going to be sensitive enough to hear if anything bad was happening inside the plant. That's all external stuff. I mean, it's close to the internal -- or the internal components, but not on the internal vibration components.

But the neat thing they did during the TF-3 testing is along with instrumenting tubes with actual accelerometers and strain gauges, they put those same pressure sensors in the TF-3 test facility and were able to show us that when a tube did start vibrating, not non-linearly, it was total linearly, they could actually hear that tube vibration in those pressure sensors.

And they could hear it quite clearly. And that proved to us that their approach for the NPM is sound, that those pressure sensors ought to hear if anything crazy is happening inside the plant. So that gave us a lot of confidence, feel good about that. But that is not something we had in DCA but we have now. Okay, next, please.

Let’s move on to the Service Level D stress analyses. And just a note, I did not perform the DCA review, David Ma (phonetic) did. But I worked closely with Yuken and David during that, so I have a pretty good feel for what they did.

There are a few differences between the DCA and the SDAA approach. It’s obviously completed for building. We’ve talked about that already. The seismic loads for the Service Level D calcs, the SDAA did include both soft soil and hard rock.

Under DCA it was I think only soft soil or something intermediate. But it was one condition, but in the SDAA that they expand everything.

And the reason that’s important is the hard rock shifts some of the peak loads up in frequency. And that ended up aligning with some of the low frequency resonances of the steam generator tubes itself, so that was something we looked pretty closely at.

They did make some other changes to the designs, not going to get into that here. But the one thing I want to highlight is that the modeling approach when you compare the DCA to SDAA, much, much improved. Much cleaner, simpler, more rigorous, more detailed. So it's a much simpler evaluation we were able to do.

Also their assessment of the overall stresses throughout the RVI, the steam generators. Comprehensive, quite thorough. We did not see any significant risk of damage to worry about.

It is preliminary, they will do an updated calculation before they actually build the thing. But we're pretty confident they've got a bounding evaluation and there shouldn't be anything to worry about.

Now, we've got some details we can get into if you like, but I'd like to skip the next couple of slides unless you want to ask some questions.

Oh, one final point. The transient loads are pretty significant here, like the blowdowns from inadvertent vent openings. It's pretty much the seismic that dominates everything by about an order of magnitude.

Okay, I think the next two are just kind of backup, we can go ahead. Next one, please. Next one, please. Okay, so that's -- we are concluded.

DR. CHOWDHURY: So thank you, thank you, Steve. This is Prosanta Chowdhury again, Project Manager, NRC. So this is an overall conclusion slide. As staff already described some of the differences between the DCA and SDAA. The staff found that the Applicant provided sufficient information to support the staff's safety findings.

And the staff found that all applicable regulatory requirements were adequately addressed. And that concludes Chapter 3, Sections 3.7, 3.8, and 3.9.2 formal presentations. Thank you.

CHAIR KIRCHNER: Thank you, Prosanta.

Members, any questions? We can also take up some of this in the closed session later.

Okay, with that, we're a little bit behind on the schedule, but we can catch up later. Let's take a break until 10:45 a.m. Eastern Time. And we'll pick up Chapter 5 and the NuScale presentation.

Thank you to all the presenters.

(Whereupon, the above-entitled matter went off the record at 10:28 a.m. and resumed at 10:45 a.m.)

CHAIR KIRCHNER: Okay. We're back in

session, and we're going to turn to NuScale and Chapter 5. And Wendy, are you up first? Go ahead, please.

MS. REID: Hello. My name is Wendy Reid. I'm a licensing engineer with NuScale and have been for three years now. Previous to NuScale, I was an engineer with Electric Boat. I specialized in taking first of a kind technologies through qualification and installing them on the submarine for proof of concept trials.

I will be introducing Chapter 5 here and presenting Sections 5.1 and 5.2. And then I will turn it to Erin who has Sections 5.3 and 5.4. Chapter 5 is the reactor coolant system and connecting systems.

It's where we described the reactor coolant pressure boundary and its components over pressure protection, the reactor vessel and its properties, and our system components in the subsystem design, so DHRS and steam generators, the pressurizer. I would like to note that Section 5.3 is where we incorporate the Pressure and Temperature Limits Methodology Technical Report which does have an SER, although it's a technical report. And we are including it in our presentation today.

And where changes were made in Revision 2 of the FSAR, we have been noting that (audio interference) as with Chapter 3. Our presentation focuses on the differences from DCA. In the staff's review of Chapter 5 and the PT Limits methodology report, we had 59 questions -- audit questions in Chapter 5.

We had 20 additional questions against the technical report. And we had one RAI in Chapter 5. All of these were successfully resolved.

In Section 5.1 is our summary description. We have tables describing the normal operating parameters. And they do show changes in both the primary and the secondary operating pressure, operating temperature, and the flow rates.

These are all a result of the power uprate. Also, the design pressure is the same for primary and secondary, so both sides of the steam generator tubes. Both those design pressures changed from 2,100 psi to 2,200 psi.

And we made a classification change to the upper steam generator support based on feedback from the manufacturer. The requirements for that support remain consistent with ASME code. Finally, there was a change to the RCS volume.

MEMBER HALNON: Wendy, what's --

MS. REID: Sure.

MEMBER HALNON: This is Greg. What's the footnote there that you get --

MS. REID: Oh, the footnote?

MEMBER HALNON: You got it on every single one of them.

MS. REID: Sure. The footnote was where Revision 2 of the FSAR has a markup from Revision 1.

MEMBER HALNON: Okay. So those are all adds on the original one.

MS. REID: Yeah, the last two bullets are. The first two bullets are consistent with Revision 1 of the FSAR.

MEMBER HALNON: Okay, thanks.

MS. REID: In Section 5.2, integrity of reactor coolant pressure boundary -- reactor coolant boundary. Section 5.2 is where we describe code compliance and it's where we adopt the 2017 additions of the boiler and pressure vessel in the operation maintenance codes. In 5.2, we also describe RCS leakage.

And there is a change to the requirement for sensitivity of detection in Chapter 5. But our containment evacuation system itself, that equipment, and its capabilities didn't change. And there was no change to the limits and tech specs for leakage.

Chapter 5 describes the reactor coolant pressure boundary. So we also describe the change from three to two reactor vent valves there as was previously discussed in the LOCA presentation. The setpoints and the design of the reactor safety valves had a change.

The setpoints increased with the design pressure. We also staggered those setpoints. Previously, they both had the same pressure.

We increased the minimum design capacity per valve. And for the design of the valve itself, we moved from pilot operated to spring operated. In SDAA, we added -- yes.

MEMBER HARRINGTON: This is Craig Harrington. For the reactor safety valves, the spring operated safety valves, have the designs of those, I guess, benefitted from the testing that was done after TMI, spring operated safety values? A lot of testing done.

Obviously, these were 30 years on, whatever. And the same valves aren't available, and these were going to be smaller than legacy plants. But has knowledge gained from that testing been --

MS. REID: I know we made the change to

spring operated because they had better OE. But Brian Kanen is the one who can provide better context on that. Brian, are you on the line?

MR. KANEN: Yeah, I am here. The primary reason for going to the spring operated is it simplified the design and made it also smaller. It was more complex than it needed to be with the pilot operated valve.

We are currently working with a couple suppliers. But we haven't gone into the details of all the matter, I guess, with the testing of OE. We haven't selected -- we haven't downselected this specific supplier yet. So I can't speak on that exactly.

MR. CARDILLO: This is Augi Cardillo from NuScale. We have considered that as part of the valve design. And as part of the test regime, that will happen post -- all the testing will get done in accordance with the OM code, et cetera, and the design of the valve itself as we go with the vendors. So we are looking at that and we'll include that in our testing regime for the -- like the industry continues already.

MEMBER HARRINGTON: Okay. Thank you.

MS. REID: All right, continuing. In

SDAA, we added the containment isolation test fixture. And we also introduced some augmented examinations above and beyond what ASME requires for the valve bodies and the welds on the four chemical and volume control system lines.

And then the low temperature overprotection -- overpressure protection setpoints changed due to the material change in the lower RPV. And then last for 5.2, Table 5.2.3 is where we show the materials for the reactor coolant pressure boundary components and support materials. That table does look substantially different from DCA.

But it is more of a change in how we report the information than actual design changes to the components themselves. The biggest design change is easily the lower RPV material change to austenitic which is discussed further in Section 5.3. But in how we report that information, we added permissible materials to that table when an alternate material was ASME approved and acceptable.

We included it in that table to add flexibility for the COL applicant. And then we had some changes for consistency in completeness and response to audit questions. And we also reconciled our naming conventions with internal design documents.

So the names of some components changed in that table.

MEMBER HARRINGTON: This is Craig. You speak to maybe this afternoon the decision to change the lower vessel material. What drove that?

MS. REID: In the next slide --

(Simultaneous speaking.)

MS. REID: -- we have a discussion about it, yeah.

MEMBER BIER: I have another question, Vicki Bier. This has come up in some past meetings also. So it's not unique to this presentation.

But when you talk about increasing flexibility for alternate materials, how does that fit with the goal of standardization? How big could the cost pressure or other performance pressure be to require alternate materials? And would there be any safety or analytic impact, or you think they're really all equivalent?

MS. REID: We see it primarily as avoiding a departure in the COL if it's already a licensed material and agreed to be acceptable by NuScale and the staff. Erin, do you want to add any context to that?

MS. WHITING: This is Erin Whiting from NuScale. I would say that in addition to that, everything is ASME approved. So we're not introducing any novel materials in this application.

So yeah, it's just to make sure that people have options. We have supply chain concerns, things like that. So that was really that and anticipating it.

MEMBER BIER: All right. Thank you.

MS. REID: Yeah, the next slide is Erin.

MS. WHITING: Hi, I'm Erin Whiting. I've been a licensee engineer at NuScale for about two years now. Prior to that, I had 15 years of analytical experience at Westinghouse.

Section 5.3 is the reactor pressure vessel. To Craig's point, we're going to discuss the material change for the lower RPV. We moved from ferritic steel to FXM-19 austenitic stainless steel mostly because it was a better material for fluence concerns.

This plays out into the PT limits report. The methodology we used is different because we don't have the beltline fluence concerns. We did for the PT limits report expand the COL Item 5.3-1 in response to audit.

And we took exemptions for 50.60 for fracture toughness, including Appendices G and H and also 10 CFR 50.61, pressurized thermal shock. There's superior ductility for the use of austenitic stainless steel. And it is less susceptible to the effects of neutron and thermal embrittlement which was really what informed that decision. And we do not have a material surveillance program requirement for Appendix H.

MEMBER BALLINGER: This is Ron Ballinger. I've mentioned this quite a few times. FXM-19, by the way, will crack in high temperature water. Temperature is much lower.

But all the documents I read justifying use -- and it's a code case too -- don't mention that at all. So I'll say it again. You ought to be a little bit cautious to make sure that you're not running yourself into trouble, especially with a weld.

MS. WHITING: Thank you. Are there any other comments on -- did I address your question? Okay. We also removed a COL item concerning onsite cleaning of the RPV during construction because that's covered under NQA-1. It was redundant.

We removed the flow diverter, and we changed the seismic restraint fixture which was a feature which was already discussed in Chapter 3. Next slide, please. In Section 5.4 which is RC component and subsystem design, they outline the performance requirements of the decay heat removal system. We changed the size less for its -- sorry, more for manufacturing concerns and less about the actual performance of the system.

We do credit DHRS and safety analysis for containment peak pressure response to a loss of coolant accident in SDAA which is a change from DCA. We do address this in Chapter 5 as a result of the audit. We added details on the emergency core cooling system venting to limit hydrogen accumulation in the reactor pressure vessel during containment isolation.

And the DHRS meets the intent of SECY 94-084 by achieving a passively cooled safe shutdown condition within 36 hours. We added off-nominal cases at staff request during the audit for worst case DHRS. And we added details about the actuation valve accumulator pressure.

We also expanded a description of the steam generator supports as Wendy mentioned earlier. And we added descriptions of flow paths between the riser and the downcomer as a result of the audit. We also changed the description of the steam generator to plugging criterion due to bracketing the two plug-in value and technical specifications.

MEMBER HALNON: So Erin, before you get into the DWO discussion, I was talking to my colleagues. And one of the things that we take great comfort in is there's a lot of margin in this overall design. But when you see things, like, going from three to two valves, higher pressures, and stuff like that, I just wanted to get it on the record for you to all say that all those things that you did actually either maintained or improved that margin or at least sufficient margin. Let's say the same margin. But can you make that statement in public that all these changes did not eat away any of the margin in any significant manner?

MS. WHITING: There's several --

MEMBER HALNON: It's a broad question.

MS. WHITING: Yeah, I was going to say it's a broad question and it's also -- it's hard to say that we didn't sacrifice any margin when we made changes to the design.

MEMBER HALNON: Sufficient margin.

MS. WHITING: Yes.

MEMBER HALNON: And we took comfort in from the standpoint of the overall NuScale design is -- got a high level of margin just kind of generically.

MS. WHITING: Yes, and I think throughout the FSAR, we're meeting the requirements and maintain margin limits for safety.

MEMBER HALNON: I just don't want people to misunderstand going from three to two, changing system pressures, and other things. You're not tightening up on this to the point where you're just barely acceptable. It's still --

(Simultaneous speaking.)

MS. WHITING: No, we optimize things to maintain margin and also we're meeting all the ASME code requirements in Chapter 5 as well.

MEMBER HALNON: I just wanted to get that out. Appreciate it.

MS. WHITING: Any other questions?

MEMBER HARRINGTON: This is Craig Harrington again. Just a question on the heat exchange -- heat removal, heat exchanger system. There's level instrumentation to look for the noncondensible gas, water interface.

But what does the operator do if they see that here or there? Is there any intended operator action in response to that? Or they just note that that's where it is and move on? You've analyzed presumably a limiting amount of noncondensible gas.

But just kind of what does the operator do with that information other than write it down?

MS. WHITING: There's a DHRS operability technical specification which would be where we maintain that.

MEMBER HARRINGTON: Do they have the ability, like, crack the valve open and flow water through our steam through to sweep out the gases?

MS. WHITING: I think Ben Bristol can address that.

MR. BRISTOL: Sure. This is Ben Bristol with NuScale. So there's a couple of options. Certainly we can down power and do some maneuvering that way in order to bleed that.

The other option at power conditions, there's a certain pressure drop across the steam generator. So operators can actually optimize the pressure drop in such a way that they can bleed the DHRS side by cracking the valve open and causing reverse flow and recover the level once the noncondensible is filled up. Bleed it out through the steam system.

MS. WHITING: Does that address your question?

CHAIR KIRCHNER: Dennis, go ahead.

DR. BLEY: Yeah, I want to address a question to Ron because I nosed around just a little bit. I don't know much about FXM-19. And I saw something about cracking in a chlorine-rich environment.

What kind of temperatures, Ron, are we talking about? And what kind of cracking? NuScale hasn't really responded to your statements here. I'm just curious if you can fill the committee in a little bit on that history and what the problems might be.

MEMBER BALLINGER: It's -- FXM-19, it's basically a better stainless steel than 304 or 360. It's cracked in some environments, and I have a paper which I sent them. But the temperature is way higher.

It's in PWR steam generator temperatures which is lower -- which is higher than the NuScale steam generator temperatures. And the rule of thumb is they're probably closer to, let's just say, military applications for PWRs. And so while you can crack it and it has -- there have been instances of cracking at PWR, U.S. PWR temperatures, the lower temperatures at NuScale operates at mitigates against having the same problem, although I think they need to be aware of it, especially when you do welding on this stuff where you get very high residual stresses.

DR. BLEY: Okay. That helps me a little bit. Thank you.

MEMBER BALLINGER: It's also known as Nitronic, I think, 50, yeah. And that's been used by --

DR. BLEY: Yeah, I saw that.

MEMBER BALLINGER: That's been used forever.

MS. WHITING: Next slide. And finally in Section 5.4, we address of the pillars of the DWO safety case. Both the real time monitoring and physical inspections are addressed in Chapter 5. For the DCA, the impetus of DWO is, as this body probably knows, there was a carve out that asked us to evaluate secondary site instabilities and also ensure steam generator integrity that was meant to -- that the COL applicant has to address that for the US-600 design.

We removed that COL item for SDAA. And our initial intent was to use the inlet flow restrictor to say that DWO is precluded across all operation. As we moved through the SDAA and gathered more information about DWO, we decided that, well, we couldn't preclude DWO throughout operation.

And so we used real time monitoring which is an approach temperature that's discussed in Section

5.4. There's a figure and a description of what it is. The use of safety signals ensures that we know when we are likely to have DWO.

And then there's a lifetime limit as was discussed in Chapter 3. And we ensure that the steam generator integrity is insured throughout that lifetime limit. We also added extra inspections for the steam generator tubes and the inlet flow restrictors to ensure that we are aware of degradation occurring. And we added a loss coefficient range as part of the audit because of audit questions where it made it easier for the staff to review exactly what that particular performance the IFR would be doing.

CHAIR KIRCHNER: So Erin, because of the approach temperature concept that you're using to kind of (audio interference) most of the operating range to ensure that you don't get into these DWO situations, does that then get reflected in tech specs somehow?

MS. WHITING: Yes. The requirement for the cyclic and transient limits in Chapter 3 are in Tech Specs 543, I believe. And Tech Specs 544 has the steam generator program which is the inspection requirements --

(Simultaneous speaking.)

CHAIR KIRCHNER: Yeah, so those are kind

of -- how should I say it -- not something that you -- that's something you monitor and inspect for after service. But during normal operation, is there a tech spec that requires that within a certain power operating profile that approach temperature has to be such and such?

MS. WHITING: Yes, there are regions that we'll discuss in the closed session. And actually in the FSAR, the Figure 5.4-16, I believe, has a region where DWO is precluded during operation and also a region where you could count time in DWO.

CHAIR KIRCHNER: Would the operator then have that figure or something equivalent, much like you have pressure, temperature limits and you operate within that band when you're operating a PWR like this design? Would there also be then some kind of tech spec operating limit somehow that the operator -- some reincarnation of that or --

MS. WHITING: There's no restriction on operating with DWO. We're just counting time in DWO in Chapter 3 and through those tech specs. And it's 5.5.3 and 5.5.4. I misspoke. I apologize. So it's not like a pressure-temperature limit where you have to stay under the curve for operation. It's not an LCO.

CHAIR KIRCHNER: You wouldn't -- why would you not have an LCO for the operators that check that? You had sufficient -- I'll call it subcooling or temperature difference between the main average coolant temperature and in that water inlet temperature.

MS. WHITING: So the limit of time in DWO for the lifetime of the NPM is 2,840 days. So it's over six years. And we can show that the steam generator integrity is maintained over that period.

So there's, first of all, not a safety concern saying we can't operate there until we hit that limit. Then we would not be able to. Does that answer your question?

CHAIR KIRCHNER: Yes and no. I'm just thinking from an operator standpoint, yes, we can go through the cycle and have some confidence that we're not going to eat up our margin in terms of fatigue and vibration and wear and so on. But that's something you inspect for after, say, a refueling cycle or whatever. But as the operator, what guidance is out there to the operator to support this safety case?

MS. WHITING: The operator would be counting time in DWO against the tech specs limits for the cyclic and transient operations. So that's a normal thing that operators do. So they'd be counting time in DWO to ensure that they're not hitting the 2,840 days.

CHAIR KIRCHNER: How would they measure time in DWO while you're operating the plant? I get the shutdown you inspect and all the rest. And you've done analysis to show you're not eating up the margin in terms of structural integrity. But what does the operator do with this?

MS. WHITING: So they would evaluate the approach temperature. And if they're above the limit where they have to count time, they're fine. There's really no chance of DWO in that operating space. When they're below the curve, they would count time in DWO against the cyclic limits in Chapter 3.

CHAIR KIRCHNER: Okay. So they're counting time. But that suggests to me then there's a tech spec that somehow they're monitoring at temperature and --

(Simultaneous speaking.)

MEMBER HALNON: Yeah, this is Greg. The tech spec or if you will those limits could be not in days. But you have an operating curve that they'll probably be operating their plant to.

And they're in the region of concern. A

ticker will go off, probably automated to some extent I would imagine. And then when they get above it, the ticker stops. And at the end of the day, you say, okay, I added one day to the 200,000, whatever hours I can have. So my sense is it's an operating curve that applies towards a limit, whether it be in tech specs or a safety limit, whatever the case may be.

MS. WHITING: Yes, and so --

MEMBER BALLINGER: This is not unlike counting fatigue cycles in PWRs. And yeah, it's the same concept.

CHAIR KIRCHNER: No, I get that part. I'm just in my own sense of operating a plant and you have the peak heat curves you typically use. There would be some three dimensional plot that shows steam generator feedwater inlet temperature versus --

MS. WHITING: That's not one of the parameters. The approach temperature is the difference between RST hot and being steam temperature.

CHAIR KIRCHNER: Yeah, I'm sorry. I misspoke. So yeah, main steam exit. So anyway, in my mind for an operator, that's something that they would be monitoring.

MS. WHITING: Yes. I guess the cyclic

limits in Tech Spec 5.5.3.

MR. BRISTOL: This is Ben Bristol. So a little context to how we anticipate this folding into the operational scheme. We will have control systems, and we'll get into in the closed section.

But we're planning to define the operating path of the steam generator a long way from where the limit is. As Erin kind of mentioned, we view the -- the concern is mostly being a long term accumulated degradation type concern. Therefore, we don't necessarily want operators immediately responding to space where we're getting close or may dip into that region.

That's something that can be analyzed on the back end. So it's not something that is acutely important to safety and something that operators should be worried about. But we will devise control systems that maintain a level of margin and keep the steam generator controlled in the stable zone, the Region 2.

MS. WHITING: And we'll discuss this more in closed session --

(Simultaneous speaking.)

CHAIR KIRCHNER: Thank you.

MS. WHITING: And I believe that concludes our open presentation. Are there any further questions?

CHAIR KIRCHNER: Members? Okay. Thank you very much, Wendy and Erin. And we'll just pause here a moment and ask the staff to come forward.

MR. DRUCKER: Hey, Mike Snodderly. Can you see the screen? I'm sharing my screen right now.

MR. SNODDERLY: Yes, we can see your screen. Can you make it presentation mode?

MR. DRUCKER: Yes.

MR. SNODDERLY: That's good thank you. Yes, better.

CHAIR KIRCHNER: Okay. Who's going to lead off for the staff?

MR. DRUCKER: I am. Good morning. My name is David Drucker. I'm a senior project manager in the new reactor licensing branch at NRR and the lead project manager for the Chapter 5 review.

This slide shows the main contributors for the review of Chapter 5. And names in shown in blue are today's presenters. The NRC staff completed the review of Chapter 5 and issued an advanced safety evaluation to support this ACRS subcommittee meeting.

There are no significant changes between the draft safety evaluation provided to ACRS on

January 4th and the safety evaluation published on January 29th. There are four sections that constitute Chapter 5. And the remainder of this briefing will focus on the deltas between the design certification and the SDAA. Next up is Nick Hansing.

MR. HANSING: My name is Nicholas Hansing. I'm a mechanical engineer.

MR. DRUCKER: (Audio interference.)

MR. HANSING: Excellent, thank you. My name is Nicholas Hansing. I'm a mechanical engineer in the Mechanical Engineering and In Service Testing Branch. I've been with the NRC for over ten years.

Again, Section 5.2.1 which is compliance with the codes and standards rule and SME code cases. Significant differences between the DCA and the SDAA include the particular codes of record that are used as discussed in the NuScale presentation. They use the 2017 edition as opposed to the earlier editions that were for the DCA.

Additionally, the selection of ASME Code Cases that used are different in this application. However, they are all accepted for use in the appropriate NRC regulatory guides. The conclusions remain the same for the SDAA as compared to the DCA.

There are no matters to discuss for 5.2.2, 5.2.4, or 5.2.5. So my next slide will be on 5.2.3. This is the reactor coolant pressure boundary materials.

As we heard earlier from NuScale, the materials have changed. They're outlined here. NRC has found them acceptable for the particular applications.

They're compatible and suitable for the intended use. And the conclusions remain the same between the DCA and the SDAA. That concludes the 5.2 slides.

MEMBER PALMTAG: This is Scott Palmtag. So you mentioned the FXM-19 looks just fine for this application. And Ron says there may be issues with cracking. So how do you reconcile this?

MR. HANSING: I will note I am a mechanical engineer, not a materials engineer for this. So I'm going to turn to my colleague here.

MR. WIDREVITZ: We'll discuss that more in 5.3 --

MEMBER PALMTAG: Okay.

MR. WIDREVITZ: -- which is next.

MEMBER HARRINGTON: This is Craig Harrington. One quick comment. In the version of the SER that I reviewed, it still speaks to the reactor safety valves as pilot operated. Does that mean updated to spring operated?

MR. TESFAYE: This is Getachew Tesfaye. Yes, those have been changed to spring valves. They put that in the final version and change it to spring operated.

MR. DRUCKER: Dan, are you ready? Next slide?

MR. WIDREVITZ: Take it forward to 5.3. All right. Section 5.3 is focused on materials, ensuring aspects of the reactor vessel itself. The significant differences between the DC and SDAA were principally the use of FXM-19 austenitic stainless steel for the lower reactor vessel. Also, there were several exemptions, 6 and 15. The slides are correct here from the ferritic steel requirements which are inapplicable to austenitic stainless steel through the material change.

These generally interact with requirements of 10 CFR 50.60, 10 CFR 50.61, and 10 CFR 50 Appendices G and H which don't apply to austenitic stainless material. So we had to find a way of syncing that back up which the applicant did for their exemptions. Also, you'll notice that there's a fairly large COL item, 5.3-1 which is partially transcribed here on the slide.

That gets to a number of remaining details that would have to be verified at the COL stage versus the information that was available during the SDAA review. Next slide, please. So if we went two slides --

MEMBER BALLINGER: Can you go back a slide?

MR. WIDREVITZ: So the NuScale SDAA SE conclusion is different from the DCA generally because of the material change for the lower RPV. Consequently, there's a whole discussion of exemptions that do not exist in the design certification application. In addition, there are some differences in how pressure-temperature limits methodology was constructed and reviewed. Next slide, please. I'll take a significant pause.

MEMBER BALLINGER: This is Ron Ballinger again. I'm going to keep pounding this dead horse. 2017 version of the ASME code, now I've got to remember whether that's true or not. If you go from 2017 to 2019 version of the code, there are changes related to API 579 and 580, including it's called FM-1.

These numbers that require you to deal

with environmental effects which didn't -- which were required in past times. So I'm curious as to whether -- I don't see anything in there in the SE and everything that says other than the issues, the difference between stainless steel and ferritic steel with respect to embrittlement and those kinds of things. There's nothing in there where it said did you look at environmental effects and did you disposition those environmental effect possibilities and the reasons for doing that.

MR. WIDREVITZ: Well, I can't speak directly to that because that is the 2019 edition. That's talking off the top of my --

MEMBER BALLINGER: Well, we're talking about --

MR. WIDREVITZ: I'll try and answer you in a technical way which is moving to FXM-19 is totally unique because everyone else is using these traditional OLI ferritic steels, right? Clad with stainless, nobody is making a vessel in our commercial industry yet until NuScale does out of Nitronic 50. So what we did do is we tried to conduct a -- I'll call it thorough, you can debate that term. We try to conduct independent literature research. And also quite a bit of information was provided by the applicant as part of their support for the exemptions.

And so we were looking for things like how does it age, particularly the temperatures of interest, right? Now we know that austenitic stainless steel is obviously going to be better in corrosion properties. It's a lot more tough than ferritic materials that use vastly more fluence before you can measure any effects in terms of toughness, right? And we were looking to verify those through essentially literature review, and that's what we did.

MEMBER BALLINGER: Yeah, I mean, and that's perfect. All I'm saying -- that's perfect. All I'm saying is, is that there's not much mention in there of the potential for environmental effects which I just didn't see it.

MR. WIDREVITZ: Yeah, we don't --

MEMBER BALLINGER: So --

MR. WIDREVITZ: -- specifically address that, no.

MEMBER BALLINGER: Yeah, but that's not necessarily a good thing.

MR. WIDREVITZ: If the -- if our literature indicated aging considerations, you'd bet they'd be in there.

MEMBER BALLINGER: I'm sitting on my desk in my office here. Okay, thanks.

MR. WIDREVITZ: I can only speak to what we did and I don't have my time machine handy. So that brings us to the pressure-temperature limits methodology itself. There were a lot of significant differences mainly because of the change in material.

And I think from a -- I'm going to say the words that might get folks excited -- fracture mechanics standing, changing to austenitic stainless steel gives you a lot of advantages. And that sort of change where you're interested in looking in terms of pressure-temperature limits where you're limiting locations are change from what is traditionally ferritic materials are outlined where you're receiving a lot of fluence just near the fuel and lower reactor pressure vessel to some sort of geometric discontinuity where there's a stress riser in the rest of the power module. And so that was quite a big difference in pressure-temperatures.

It's in some sense more robust design which makes verification pressure-temperatures just move to a more sophisticated analysis question but not necessarily a riskier question. So with that, the SDAA design is never beltline limited in the lower reactor pressure vessel. That's very different from every other design that's operating in the fleet.

The pressure-temperature limit curves are limited by geometric discontinuities and locations where potentially no neutron embrittlement. The aging of those thermal and neutron is going to be inconsequential based on our review relative to what you'd expect from limiting locations in a traditional design. And of course, there's this enlarged COL item 5.3-1 with a lot of details that need to be verified because the location of interest is very different and how it interacted with the information and the various things that were presented for our review.

So ultimately, the SDAA SE conclusion is different from the DCA, not because of anything necessarily more risky or safety considerations but just how the whole case and the details that need to be validated for the COL stage are different from what you'd expect from all of the other designs. And that's a bad thing. That concludes my slides for 5.3. I'll take a significant pause here.

MEMBER PALMTAG: Scott Palmtag. I didn't really hear an answer to that question. Is there a cracking issue?

MR. WIDREVITZ: Not that I'm aware of, though I would love to see that paper because I did not find it.

MEMBER BALLINGER: So let it be written, so let it be done.

(Laughter.)

CHAIR KIRCHNER: Okay.

MR. WIDREVITZ: Next up is Greg Makar.

CHAIR KIRCHNER: Go ahead, Greg.

MR. MAKAR: My name is Greg Makar. I'm a materials engineer in the Division of New and Renewed Licenses and Corrosion and Steam Generator Branch. And I'd like to thank and acknowledge my coworker on this review, Leslie Terry, also in Corrosion and Steam Generator Branch, and also a reviewer for the thermal hydraulics area for the approach temperature limits, Tim Drzewiecki, who's in the senior reactor systems and engineer in the Division of Advanced Reactors and non-power production and utilization facilities.

The regulatory basis for our review focuses on the integrity and the inspection of the reactor coolant pressure boundary. Staff reviewed FSAR Section 5.4.1 in accordance with the design specific review standard, Section 58.21, to ensure the integrity of steam generator materials is maintained and that the steam generator materials meet the relevant regulatory requirement. We also reviewed

Sections 5.4.1 and 5.4.1.6 in accordance with the SRS Section 5.4.2.2 to ensure the steam generator is designed to permit periodic inspection and testing of the tubes and other critical areas and that it includes features to assess structural and leakage integrity of the tubes.

And we also reviewed the tech specs and bases as they relate to incorporating the steam generator program. This slide focuses on the differences from certified design that we consider most significant. I'll start with the inlet flow restrictors.

These are a different design in that they -- in the certified design, they were -- the flow was around the restrictor. Now it's through a central orifice. And there's now contact with the inside surface of the tube.

But the materials are 300 series austenitic stainless steel. They're compatible with the secondary coolant. Although the new design involves contact with the tube, there are design features designed to prevent it from coming loose and becoming a source of loose parts in the tubes or from damaging the inside of the tubes.

They will be inspected visually during

steam generator tube inspections or cavitation. And based on the IFR design, those materials approved by ASME code, compatible with the environment and features to prevent tube damage and the proposed periodic inspections, staff found this design acceptable with respect to tube integrity. Added to the steam generator program, combined license item 5.4-1 is additional inspections in between the first -- the inspection of the first refueling outage which is 100 percent of the tubes.

And the next inspection that's required in the tech specs which is no tube can go beyond 72 effective full-power months. There's an addition of at least 20 percent of the tubes being inspected at each refueling outage for the first module to undergo a refueling outage. This was introduced in the context of density wave oscillations, DWO.

But it's a good idea regardless of whether DWO is a concern. Without operating experience early in life, it's more difficult to assess the significance of tube degradation or the lack of tube degradation. And so these additional inspections will be valuable for understanding the form and rate of degradation that's needed for condition monitoring and the forward looking operational assessment.

And it's also -- I also want to mention these are primarily performance-based tech specs that are required to maintain tube integrity. So we really can't say going in when the second inspection will be. But it's -- because it's going to depend on the findings of the first inspection. But we know that there will be at least this minimum amount of inspection in the subsequent inspections to the first. And so we find these as an acceptable way to address the uncertainty early in life of the steam generators and to help ensure integrity is maintained.

MEMBER HALNON: Greg, this is Greg. Outside of the tubes, you mentioned visual. Is that the expectation is that there'll be a comprehensive visual inspection on the outside of the tubes supports?

MR. MAKAR: Well, I'm not sure any -- their comprehensive visual inspection is very difficult in steam generators on the outside because of the proximity of the tubes to one another. Normally, there are lanes without tubes installed, these long vertical passes that they have in traditional steam generators. So there are -- you can put cameras in and look into this. The expectation is that they'll do that, tube sheets and where they can at supports. But I'm not sure there's a way to look -- I'm not sure it would be possible to look everywhere.

MEMBER HALNON: I guess -- so we're going to rely on eddy current through the tubes to tell us. If we are seeing any kind of flow issues between the outside of the tubes, you may not see those after only one cycle. So my question is, you think the 72 month -- it was 72 long months -- do you think that's going to be adequate to prevent any kind of failures if there is something starting to occur?

MR. MAKAR: I think not 72 months alone. I think that's the importance of this. Well, there's the first outage where 100 percent of the tubes.

And then the tech specs say you could go up to 72 effective full-power months until your next inspection provided that you have an operational assessment to support that. It's not automatic. Now with this first module having additional inspections at 36 months, 54 months, 72 months, then it gives you some more -- a better idea of what -- if there's nothing happening.

That's one of our concerns is that nothing happens in the first cycle. And then it looks like nothing is going to happen forever. And this helps to prevent that scenario. And I would say the COL item also has subsequent modules. The COL applicant will have to justify if they believe at that 20 percent doesn't apply to them.

MEMBER HALNON: We've heard earlier that the subsequent modules will be treated as prototypes until such time that the first modules proves it's an adequate prototype. I'm kind of paraphrasing. How's that going to work with this scheme?

MR. MAKAR: As far as I know, that prototype system does not affect this tech spec program.

MEMBER HALNON: It's only the CVAP, not the inspection portion. I'll have to think on that. Thanks.

MR. MAKAR: And one scenario for the outside of the tubes is sometimes you -- if there's -- of course, we're looking for -- and a very common thing to see is where from support structures. But there's also where it could occur from a loose part or a foreign object. And sometimes those are protected from the inside with eddy current.

MEMBER HALNON: Yeah.

(Simultaneous speaking.)

MEMBER HALNON: -- that all bets are off.

You're going to find it --

(Simultaneous speaking.)

MR. MAKAR: And then you have a targeted area where you can try to get a camera in and see what's going on.

MEMBER HALNON: Okay, thanks.

DR. SCHULTZ: Greg, this is Steve Schultz. The inspection program is described in a number of different places now. And in one place, I thought I saw that there was a pre-operational inspection, a 100 percent inspection, pre-operational so that when you perform that first 100 percent inspection after the first outage that you would know that something had changed, not that something was wrong because you've done it pre-operational.

MR. MAKAR: Yeah, or pre-service inspection --

DR. SCHULTZ: Pre-service.

MR. MAKAR: -- PSI. That's done after the tubes are installed and after hydrostatic pressure testing has been performed, either in the shop or in the field. So you get that look at any flaws in the tubes or imperfections in the tubes before they go on the surface.

DR. SCHULTZ: Then I thought I saw

something that suggested that if you're performing a 20 percent inspection anywhere in the process and you find something, then your inspection program is advanced. In other words, you need to look in other areas right away. Is that true?

MEMBER BALLINGER: I think that's a Section 11 requirement, right?

MR. MAKAR: I don't know. The industry does have guidelines for how to expand the scope of the --

(Simultaneous speaking.)

DR. SCHULTZ: Okay.

MR. MAKAR: And the details of the 20 percent, I'm sorry, I don't remember. But yes, there are in the steam generator program the industry framework. There are expansion guidelines. And that's the expectation here. And that's why I say we can't say for these steam generators any for sure when a second 100 percent inspection would occur because it depends on what they find in that first inspection.

DR. SCHULTZ: Okay. That's fair. I think we've got another presentation that's going to come back to this. Thank you.

MR. MAKAR: Okay. The next topic was changes in the technical specifications. There's one that is a change in the structural integrity performance criterion. The structural integrity performance criterion include a safety factor on primary, secondary pressure differential for normal steady-state full-power operations which is the most limiting of the criteria for NuScale. This is -- for conventional steam generators, this is a factor of 3.0 for burst because higher pressure is on the inside.

And that was also used for NuScale in the DCA, although they applied it to collapse or external pressure. But NuScale is -- the thickness of the tubes are determined by the ASME code case and 759-2. And that allows a stress reduction factor of 1.7 to 2.0, so lower than 3 for externally pressurized cylinders.

That's not different. That code case was also used for the DCA. But they didn't make use of that provision for the lower safety factor. So they're doing that now, but they're not taking any exceptions.

That's the code case approved by the NRC without conditions. It's consistent with some other parts of the ASME code such as pressure vessel design. So the staff finds this acceptable based on being designed in accordance with the approved code case with no exceptions and because the other safety factors and structural integrity performance criterion were not changed.

MEMBER BALLINGER: I have sort of a technical question which it may have been addressed in the B&C. These are externally pressurized tubes. And there's this 40 percent through-wall requirement which is basically for original other internally pressurized tubes.

And it's for environmental degradation. Now the issue you're going to have is not that. It's wear. So if you have 40 percent through-wall wear, does that affect the collapse criteria? Is that safety factor in the ASME code that allows you to reduce the differential pressure on everything, if you had a 40 percent through-wall region now, wear region, would that affect the collapse criteria? Because it's really collapse, not rupture.

MR. MAKAR: Well, I'm glad you asked about that because the 40 percent plug-in criterion has not changed from the DCA.

MEMBER BALLINGER: Yeah, that's what I was thinking.

MR. MAKAR: And it's a bracketed value in the technical specification which means a COL applicant has to either justify its use or justify something else. But what's different, and the reason I'm presenting it on the slide, is that the -- it's based on a new analysis. Now the 40 percent widely used in the industry is a value that's found to be bounding. And it is thinning around the tube from the outside. So that's limiting over --

(Simultaneous speaking.)

MEMBER BALLINGER: It's the volumetric criteria.

MR. MAKAR: Yes, yeah, yeah. And so they apply that same -- the same concepts that were used and the same approach that's used in the determination of the plugging criterion here when operating plants look at that because the thinning is coming from the -- still coming from the outside.

MEMBER BALLINGER: I'm just wondering about the collapse criteria.

MR. MAKAR: Well, and NuScale has performed an analysis, looking at those criteria, looking at the different loading conditions. And with thinning what they expect to be the most likely location of where in the dimensions they expect that wear to take. And then they performed an analysis -- finite element analysis to calculate the collapsed pressure.

MEMBER BALLINGER: We're kind of using a set of rules that were designed for one configuration and applying it to a different configuration.

CHAIR KIRCHNER: Greg, isn't the wall thickness or wall thinning allowance different than the crack depth allowance?

MR. MAKAR: Cracks are normally not allowed to stay in service because of the difficulty sizing them and evaluating them. There are some exceptions. But in this case, NuScale did look at cracking.

It's hard for them to get -- they've looked at conditions where they could potentially get a tensile stress in the presence of different types of cracks and found that this thinning is still bounding. And cracks will be -- would be plugged on to (audio interference) for protection. And in looking at this plugging criterion, the staff, we reviewed NuScale's analysis.

We also performed some calculations of our own based on our relationship between yield stress and geometry from collapsed tests that were performed at Pacific Northwest Laboratories years ago. And we had -- during the DCA review, we had a finite element analysis support to justify using that data, applying that data for a collapsed test on Alloy 600 to Alloy 690. So our calculations indicate that they can maintain this factor of 2.0 when we looked at more wall thinning than in their analysis.

CHAIR KIRCHNER: And Greg, that also includes the higher operating approach?

MR. MAKAR: Yes, yes. So for those reasons, we found that 40 percent bracketed through-wall. Plugging criterion and a reasonable preliminary value is that COL applicant will have to justify that. Next slide, please. And so the next slides are on the density wave oscillations and our staff's evaluation of approach temperature limit.

The definition and use and the definition of approach temperature is here which is the difference between the reactor coolant system hot temperature and the exit temperature from the steam generators. And so the review focused on whether this approach temperature limit is a way to protect against the onset of -- the effects of the onset of DWO. Next slide, please. This is an organization chart to show how our review was organized.

See the main questionnaire is -- the goal of the evaluation, the finding we were seeking to make is whether the approach temperature provides reasonable assurance or protection against the onset of DWO, not to -- not a finding on that DWO onset itself is accurate predicted. So a little different there. The next two slides have more information about three of these four topics.

You see that there are different elements identified for each topic. For the third one, static instability coupling, staff determined that no detailed review was necessary for that one. And then on the last slide, it summarizes the conclusions. Next slide, please.

This has two topics. They're not in order. I think that probably because they fit on the slide well in this configuration. But this first topic is whether there is margin between the approached temperature limit and calculations of DWO onset.

And there's a table added to in the SER that lists the five parameters and compares operating range to the analysis range. And this shows that there were different elements. This shows the elements that were applied to this review area, 1.1, the approach temperature.

And we'll get back to the second one, 1.2

on the next slide. But here, 1.4 address the uncertainties and whether the uncertainties in -- if we look at risk associated with DWO onset, are they reasonable? The uncertainty and the prediction of DWO onset, are they reasonable considering the risk associated with DWO? And there are these four elements that we looked at in our review. Next slide, please.

DR. BLEY: This is Dennis Bley.

MEMBER DIMITRIJEVIC: I have a question about risk associated with the DWO. This is Vesna Dimitrijevic. So my question is, did you -- what the sensitivity performed for these estimates? Well, when you risk is small, is this risk associated with DWO was evaluated to be small?

MR. MAKAR: When we speak of risk in this part of our -- in this safety evaluation section, I think it refers to the risk that was determined -- associated with the failure of a tube which we're not presenting that here. But I think it's risk associated with tube failure.

MEMBER DIMITRIJEVIC: Well, so my question is related to this. This is just associated with a frequency of estimated steam generated tube failure. It's not associated with the number of the tubes which assume fail or the likelihood that those tubes can be in both steam generators.

MR. MAKAR: I don't think so. But I don't know. I'm sorry. I didn't -- I wasn't -- I didn't perform the review. So I think we can --

MR. TESFAYE: This is Getachew Tesfaye. So risk associated with the failure of steam generator is discussed in Chapter 19.

MEMBER DIMITRIJEVIC: And --

(Simultaneous speaking.)

MEMBER DIMITRIJEVIC: Yeah, finishing the Chapter 19 and then how they address in Chapter 19 is different than where we discussed the phenomena. So this is why I want to bring it here because in the Chapter 19, already DWO was considered, is shown to be unsensitive to frequency or steam generator tube failures, approximation. However, there is no analysis of the sensitivity to map out the tubes assume fail.

So in the risk analysis, it's assumed that only one tube is fail and it's only in the steam generator. So I was really -- my question was, was it considered that this DWO could affect this -- the DWO consideration will affect those assumptions. That was my question.

MR. TESFAYE: This is Getachew again. I believe that will be addressed in Chapter 19. We just take advantage of the results in Chapter 19 to justify the DWO condition here. There's no specific risk to my knowledge. We can get back to you on that.

MEMBER DIMITRIJEVIC: I understand this, Getachew. I just want to say using this as one of the things which makes you feel more sure that this issue can be closed. But this is without looking and what assumptions were done.

So I mean, the Chapter 19 would not address the closure of DWO issues. Chapter 19 is separate thing. So I just thought it will make sense to bring it here. But it's all right. Chapter 19 is coming in two weeks. So we will look in it.

MR. SNODDERLY: Mr. Chairman, I think someone from -- Tom Griffith from NuScale would like to speak.

MR. GRIFFITH: This is Tom Griffith from NuScale. I think Sarah Bristol can add a little context here. I do think it's appropriate maybe to talk of 19. But I think now is a fine a time as any to talk a little bit about one of the audit responses that we have related to, I think, this question. So Sarah, can you step in?

MS. BRISTOL: Yes, this is Sarah Bristol, manager of PRA. Can you hear me?

MEMBER DIMITRIJEVIC: Yeah.

(Simultaneous speaking.)

MS. BRISTOL: All right. Hi, Vesna. Yes, we did get three audit questions related and supporting this DWO topic. Ultimately, as you know, we do look at the single tube failure in our PRA in Chapter 19.

But ultimately, we did do and look at additional failures or other potential considerations because of DWO. And so ultimately, we do a sensitivity in 19 where we increase the initiating event frequency. So therefore, if, for instance, DWO were to result in additional initiating events, we do look at that impact.

And again, that is in the sensitivity table in Chapter 19. But ultimately, looking through the various data and the history, NuScale knows no known failure mechanism that could lead to this. And there hasn't been those examples in the industry as described in SECY 93-87.

So we started there. However, we still did look at initiating event frequency and the potential for an increase there. In addition, we also do the various failures of system response.

And so we also look at all DHRS failing and the impact of what happens if we didn't have steam generators at all to respond. And also based on that, there still was nothing substantially different between a single tube failure and multiple tube failures. So we looked at it from initiating event frequency.

We looked at it from plant response. We looked at it from multiple tubes. We had about three audit questions with multiple questions asked that we worked with the staff to confirm that there was no safety or risk concern with respect to this potential phenomena.

MEMBER DIMITRIJEVIC: Thanks, Sarah. We're looking forward to check those when we reviewed the Chapter 19 and discussed it. I just thought since this was one of the -- on the previous slide, this was one of the elements which were supporting finding the (audio interference) to discuss here.

So because let's say in Chapter 19 you find there's some sensitivities where no kind of impact. I don't believe that will be the case. I mean, that will have to go back to reflect on the conclusion on this. So that's why I think even this is a part of Chapter 19. The results of this belongs here as long as they're presented as one of the elements supporting finding. Okay. Thanks.

MS. BRISTOL: Understood. Thank you.

DR. BLEY: This is Dennis Bley. Just a follow-up on that one. If, in fact, what we're looking at is uncertainties in the prediction of DWO, if the problem is in some systematic error that's hidden in the uncertainties, then certainly you could get multiple tube failures. And I'm not sure why we think it'd be two or three or something like that. So I look forward to Chapter 19 too.

MR. MAKAR: Any suggestions for -- I wonder. Did I hear a suggestion that there should be more in Chapter 5, safety evaluation, about this topic?

MEMBER DIMITRIJEVIC: My suggestion was the reference to this one should be maybe provided in Chapter 5. But in that case, our review would not be completed until we complete the review of Chapter 19. The same thing happened with the LOCA thing when the sensitivity to DHRS was said it will be addressed in Chapter 19.

And a lot of those risk analysis refer to Chapter 19. But then there is no feedback connection back to the chapters where they have been initiated. So they sort of go as I say to Chapter 19 to die there. It's just like there should be some feedback in connection between those.

MR. MAKAR: Well, we've moved to the next slide. And this is on the -- whether the onset calculations find reasonable insight into the likelihood. And so that's made up of two main parts, adequacy of the modeling capabilities and assessment of the model against experimental data.

So there were a number of areas we looked at. You can see there are 15 elements that go into these two parts. Next slide, please. The first four bullets on this say that based on these four review areas and all those 23 elements that we reached a conclusion that the approach temperature limit provides reasonable assurance of adequate protection against DWO onset. But it goes on to say the finding does not extend to the general use of NRELAP5 evaluation model for DWO calculations or for thermal hydraulic condition calculations during the DWO conditions.

MEMBER HALNON: This is Greg. What I take away from this is that this approach limit that they're measuring days against in concert with the inspections you talked about earlier provides a real solid margin for us to present. Is that fair to say?

MR. MAKAR: That's fair to say, yes.

MEMBER HALNON: Because if you take any one in isolation, you can say it's okay to put them together. It's really solid.

MR. MAKAR: Okay. Well, next, Brian Nolan is going to present the staff review on the heat removal system.

MR. NOLAN: Thanks, Greg. My name is Ryan Nolan. I'm in the Nuclear Methods Systems Branch for new reactors. I've been doing new reactor licensing reviews for 15 years now. Prior to that, I was a systems engineer in the NSSS group licensee in the northeast.

I was one of the reviewers who performed the systems review for the decay heat removal system. While there are changes to the system, overall functionally, it has not changed. The purpose of the DHRS is to remove decay heat when the secondary side is not available.

So that all stays the same. Regarding the changes, it kind of falls into three different categories, actual physical design changes, some analytical approach changes, and, in addition, modeling changes. And so for the physical changes, I note them here.

There's things like increases in the number of tubes. They shorten the tubes. The condenser sits a little bit lower. And then the ultimate heat sink water level, the initial water level has decreased.

As far as changes to analytical approaches as the staff briefed the subcommittee last month when it presented the LOCA topical report, DHRS is now credited in the LOCA evaluation model. It is a safety-related system. It was a safety-related system in the DCA.

NuScale is just taking credit for that for the SDA. And then regarding modeling changes, this is a topic that will be covered in more detail next month when the staff presents the non-LOCA topical report. But there were some significant modeling changes with respect to DHRS.

I note a couple here such as additional heat structures, changes to pool nodalization. Overall, taking a more realistic look how the plant responds and how DHRS functions. So as far as the conclusions are concerned, they're very, very similar conclusions to the SDA with respect to the functional requirements and meeting typical GDCs too for 34, for example.

We just -- because it's now credited in the LOCA EM, we also include some explanation on 50.46 and GDC 35. And that's all I had to share on DHRS. I'll pause for some questions.

CHAIR KIRCHNER: Given the previous topic, have you thought through what would happen to the DHRS performance with a tube rupture or multiple tube ruptures?

MR. NOLAN: Yes, the staff did look at that. I was not the one to perform that review. So I can't speak to it in detail. But we certainly did ask questions, not just on tube ruptures themselves.

But if you do enter a DWO condition, does that impact DHRS? And we concluded it would not impact the DHRS overall and particularly some of the loss coefficients from the IFR. And ensuring that is captured in the FSAR was something that the staff did do as part of this review.

CHAIR KIRCHNER: So the DHRS performance is based on evaporating, essentially condensing. If you just pressurize the system from the primary side, then you would just have single phase heat transfer conditions which is nowhere near as good as condensation in the heat exchanger and then draining the fluid back, so --

MR. NOLAN: This is in reference a tub rupture scenario?

CHAIR KIRCHNER: Yes.

MR. NOLAN: If someone wants to correct me in the back, feel free to step up. But in those instances, you would just consider that train lost and not effective anymore.

MEMBER DIMITRIJEVIC: So if you have a tube rupture in both steam generators, you will consider total loss of decay heat removal.

MR. NOLAN: Right. I don't believe that's something that's considered within the design basis. So we're getting into, like, Chapter 15 area.

(Simultaneous speaking.)

MEMBER DIMITRIJEVIC: Well, I'm getting into Chapter 19.

CHAIR KIRCHNER: Yes, Vesna.

(Simultaneous speaking.)

CHAIR KIRCHNER: Thanks, Ryan.

MR. NOLAN: Yeah. Like, the main purpose of this is really establishing the design criteria the system has regarding response to the system, the various transients that will come in future subcommittee meetings. And if there's no further questions, I'll pass it back to David to close out the presentation.

CHAIR KIRCHNER: Dave, do you want to make a summary statement or just let us read the view graph?

MR. TESFAYE: David, you're on mute.

MR. DRUCKER: Thanks. So overall for Chapter 5, the staff found that all applicable regulatory requirements were adequately addressed. And this concludes the Chapter 5 presentation.

CHAIR KIRCHNER: Thank you. Members, any pressing questions right now?

Okay. We've gone over schedule. I take responsibility for that. But -- oh, I didn't see you.

MS. WHITING: That's okay. Erin Whiting from NuScale. As it relates to FXM-19 and the lower RPD, we do have a technical report in SDA, TR130721, entitled Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel, which assesses the impact of using FXM-19 and a location of welds within the RPD when subjected to radiation and thermal embrittlement.

And we have documented that concluded that FXM-19 is substantially safer than use of ferritic material. And there are no safety concerns when reviewing all of the applicable available documentation. Chapter 5 does require pre-service inspection and in-service inspection of each vessel and the welds on the vessel. And in addition, the lower RPD was analyzed for all applicable ASME code and environmental fatigue criteria. And we demonstrated that the design meets those criteria.

MR. DRUCKER: Thank you.

CHAIR KIRCHNER: Thank you, Erin. So at this point, we'll take the opportunity to ask for public comments. Anyone in the room or anyone on the line, just state your name, affiliation as appropriate, and make your comment.

Not hearing anyone trying to make a comment. Okay. Then at this point, we have completed our open session. And we are going to break for lunch. For those listening online, if you are authorized access, we will re-engage at 1:00 o'clock Eastern Time. And with that, we are recessed.

(Whereupon, the above-entitled matter went off the record at 12:09 p.m.)

C E R T I F I C A T E

This is to certify that the foregoing transcript

In the matter of: ACRS NuScale Subcommittee Open Session

Before: NRC

Date: 02-04-25

Place: teleconference

was duly recorded and accurately transcribed under my direction; further, that said transcript is a true and accurate complete record of the proceedings.

Neal R Gross

Court Reporter

January 29, 2025 Docket No. 052-050

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SUBJECT: NuScale Power, LLC Submittal of Presentation Material Entitled "ACRS Subcommittee Meeting (Open Session) Sections 3.7, 3.8, 3.9.2, and Chapter 5 (Including the Pressure and Temperature Limits Methodology Technical Report and the Density Wave Oscillation Safety Case)," PM-178795, 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 4th, 2025. The materials support NuScale's presentation of the subject sections, and technical report for the US460 Standard Design Approval Application.

The enclosure to this letter is the nonproprietary presentation entitled "Sections 3.7, 3.8, 3.9.2, and Chapter 5 (Including the Pressure and Temperature Limits Methodology Technical Report and the Density Wave Oscillation Safety Case)," PM-178795, 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

Distribution: Mahmoud Jardaneh, Chief New Reactor Licensing Branch, NRC
Getachew Tesfaye, Senior Project Engineer, NRC
Michael Snodderly, Senior Staff Engineer, Advisory Committee on Reactor Safeguards, NRC

Enclosure 1: ACRS Subcommittee Meeting (Open Session) Sections 3.7, 3.8, 3.9.2, and Chapter 5 (Including the Pressure and Temperature Limits Methodology Technical Report and the Density Wave Oscillation Safety Case), PM-178795, Revision 0, Nonproprietary

Enclosure 1:

ACRS Subcommittee Meeting (Open Session) Sections 3.7, 3.8, 3.9.2, and Chapter 5 (Including the Pressure and Temperature Limits Methodology Technical Report and the Density Wave Oscillation Safety Case), PM-178795, Revision 0, Nonproprietary

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ACRS Subcommittee Meeting

(Open Session)

February 4th, 2025

Sections 3.7, 3.8, 3.9.2, and Chapter 5

(Including the Pressure and Temperature Limits Methodology Technical Report and the Density Wave Oscillation Safety Case)

Acknowledgement and Disclaimer

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Chapter 3

Design of Structures, Systems, Components and Equipment

(Sections 3.7, 3.8, and 3.9.2)

February 4, 2025

Presenters:

Haydar Karaoglu and Emily Larsen

Chapter 3 - Design of Structures, Systems, Components and Equipment

- Section 3.7 – Seismic Design
- Section 3.8 – Design of Category I Structures
- Section 3.9.2 – Mechanical Systems and Components - Dynamic Testing and Analysis of Systems, Components, and Equipment

Note: The presentation does not include Section 3.8.1, Concrete Containment, and Section 3.8.3, Concrete and Steel Internal Structures of Steel or Concrete Containments. The US460 NuScale Power Plant design does not use concrete containments or internal structures.

Overview of Key Design Features and Updates

- The Standard Design Approval Application (SDAA) is a derivative of the certified design.
- SDAA structures reflect 6 modules (12 modules in the DC), which necessitated updated structural analyses.
- For the SDAA, the Reactor Building (RXB) uses steel-plate composite (SC) walls along with reinforced concrete (RC) members.
- The site layout in the SDAA reflects the updated building designs.
- Seismic analyses for the SDAA are performed for a double-building model, featuring the RXB and Radioactive Waste Building (RWB) and a separate surface-based Control Building (CRB) model, while the design certification (DC) used a triple-building model and individual building models.
- Presentation will focus on high level design and methodology changes and important audit questions and requests for additional information (RAIs).

Section 3.7 – Seismic Design

Section 3.7.1 – Seismic Design Parameters

- Percentage of Critical Damping
  - The DC used separate fully cracked and fully uncracked models, and the RC members had the same damping ratio of 7 percent.
  - The SDAA employs hybrid models with both cracked and uncracked members. The damping in the structural members varies based on their cracking status and whether the calculation is for developing in-structure response spectra (ISRS) or performing design calculations.

    "*Building Design and Analysis Methodology for Safety-Related Structures*", *TR-0920-71621-P-A*
- Supporting Medium
  - The DC included four generic soil profiles, Soil-7 (Rock), Soil-8 (Firm Soil/Soft Rock), Soil-9 (Hard Rock), and Soil-11 (Soft Soil).
  - In the SDAA, Soil-8 is removed and the soil-separation scenario with the Soil-7 profile is introduced.
- No audit questions or RAIs for Section 3.7.1

Section 3.7 – Seismic Design (Continued)

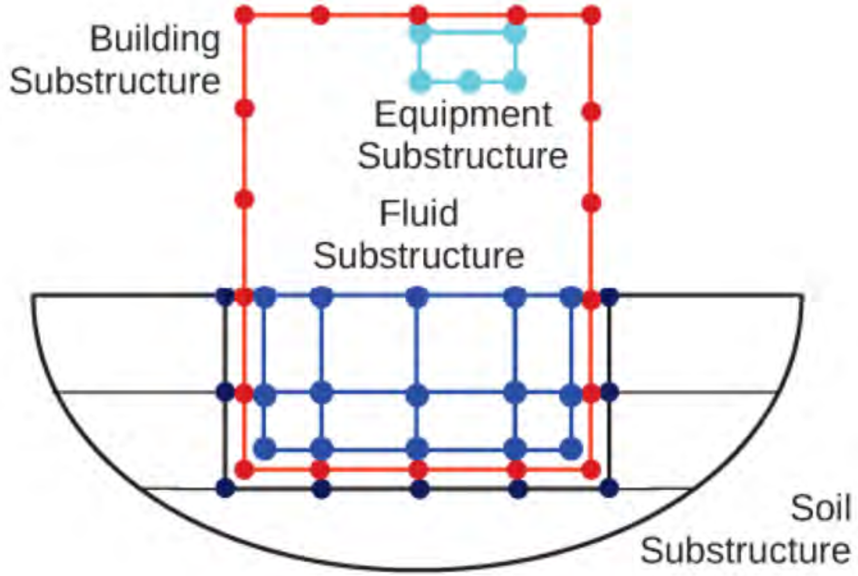
Section 3.7.2 – Seismic System Analysis

- Seismic Analysis Method
  - In the DC, soil-structure interaction (SSI) analyses were performed using the extended subtraction method with SASSI.
  - In the SDAA, the SSI analyses are performed using the soil library methodology, a robust approach equivalent to the direct method. The soil libraries are built using SASSI and the simulations are performed with ANSYS.

    ***"**Improvements in Frequency Domain Soil-Structure-Fluid Interaction Analysis", TR-0118-58005-P-A*

- Three Components of Earthquake Motion
  - In the DC, the maximum responses were calculated using the square-root-of-the-sum-of-the-squares method.
  - In the SDAA, the SSI responses from the three, statistically independent-components of the ground motion *are algebraically added.*

TR-0118-58005-P-A, Figure 4-1:
Idealized Soil, Structure, and Fluid Substructures



Section 3.7 – Seismic Design (Continued)

Section 3.7.2 – Seismic System Analysis (Continued)

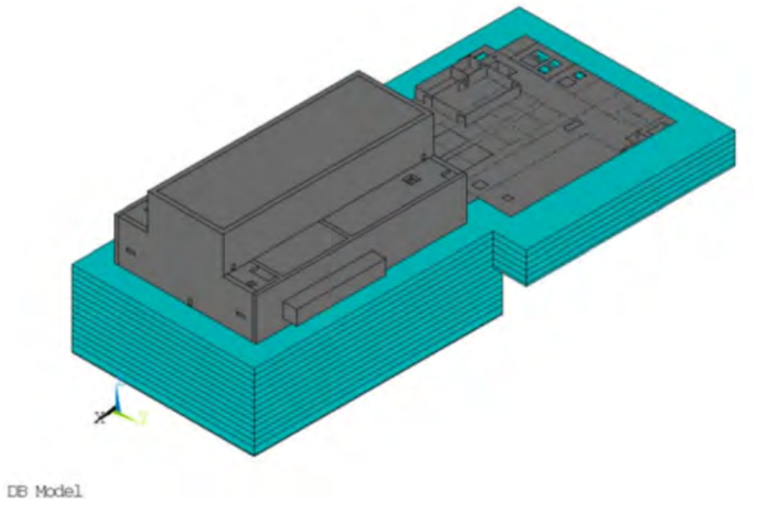
- SSI Numerical Models
  - In the SDAA, the reactor pool is modeled with FLUID elements of ANSYS and using the fluid-structure interaction (FSI) technology. The 6 NuScale Power Modules (NPMs) are modeled in detail using advanced features of ANSYS.
  - In the DC, the pool was modeled as distributed mass. The 12 NPMs were modeled using mass, spring, and beam elements (simplified beam model).

- Audit Responses

  33 questions resolved in audit, resulting in the following details and updates added to the Final Safety Analysis Report (FSAR)
  - modal analysis, double building model dimensions, and pool sloshing

- No RAIs for Section 3.7.2

Figure 3.7.2-2a: Isometric View of the Double Building (DB) Model



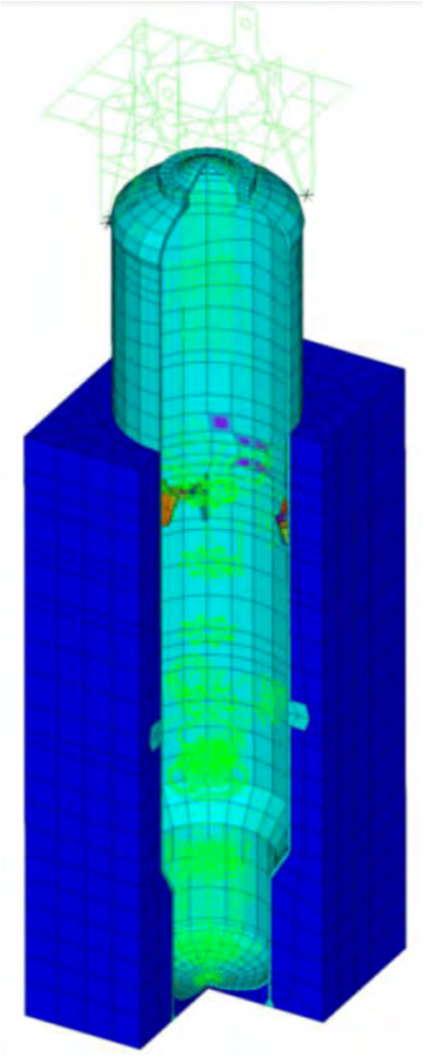
Section 3.7 – Seismic Design (Continued)

Section 3.7.3 – Seismic Subsystem Analysis

- The SDAA includes updates to major subsystems, including the bioshields, the reactor building crane, and the NPMs.
- Three different NPM models have been developed
    - Simplified NPM model is used in SSI analyses to calculate seismic responses on RC and SC structural members.
    - A detailed NPM model is used in SSI analyses to calculate the seismic response around the pool.
    - A detailed NPM model with the use of the superelement technology of ANSYS is used for the nonlinear transient analysis.

(content reflected in Appendix 3A)

"US460 NuScale Power Module Seismic Analysis", TR-121515-P



Simplified NPM Model
(TR-121515 Figure 3-1)

Section 3.7 – Seismic Design (Continued)

Section 3.7.3 – Seismic Subsystem Analysis (Continued)

- In the SDAA, the nonlinear NPM seismic analyses are conducted using a local model that includes the 6 NPMs, the pool, and the surrounding structural members.

  "US460 NuScale Power Module Seismic Analysis", TR-121515-P

- In the DC, the NPM seismic analyses were conducted using a local model that included only one NPM at a time, the pool, and a rigid plane under the NPM.

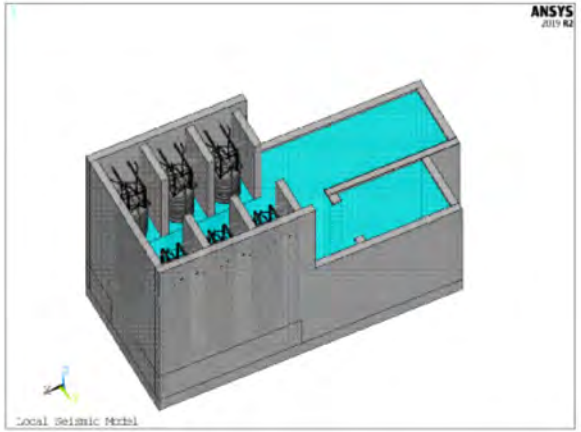
  "NuScale Power Module Seismic Analysis", TR-0916-51502-P-A

- Audit Responses

  4 questions resolved in audit, resulting in additional bioshield details in the FSAR

- No RAIs for Section 3.7.3

Figure 3.7.2-7: NPMs within UHS
(Local Seismic Model)



Section 3.7 – Seismic Design (Continued)

Section 3.7.4 – Seismic Instrumentation

- In the SDAA, the locations and descriptions of the seismic instrumentations are updated due to the new layout of the buildings.

- No audit questions or RAIs for Section 3.7.4

Section 3.8 – Design of SC-I Structures

Section 3.8.2 – Steel Containment

- Differences from DC
  - Increase in design pressure and temperature for power uprate
  - Material change from carbon steel with cladding to combination of austenitic and martensitic stainless steels
  - Changed pre-service/in-service inspections from Class 1 to Class MC vessel with augmented requirements in some areas
  - Removed hydrogen detonation from load combinations because of added passive autocatalytic recombiners (Chapters 6 and 15)
  - Majority of nozzles changed from welded to integrally forged

- Audit Responses
  - 12 questions resolved in audit

- No RAIs for Section 3.8.2

Section 3.8 – Design of SC-I Structures (Continued)

Section 3.8.4 - Other SC-I Structures

- In the SDAA
  - o The RXB incorporates SC walls designed according to AISC N690-18 using element- and panel-based approaches.
  - o The RC members are designed according to ACI 349-13 using the section-cut forces at critical locations.
  - o The forces are calculated from numerical models with different cracked states associated with different load combinations.
  - o The simulations are performed using ANSYS with the use of SASSI for soil library calculations.

  (content is also reflected in Appendix 3B)

  "*Building Design and Analysis Methodology for Safety-Related Structures*", *TR-0920-71621-P-A*

- In the DC, the major structural members were of RC type and designed according to ACI 349-06 using an element-based approach. The simulations were performed using SASSI and SAP2000.

- Audit Responses

  15 questions resolved in audit, resulting in the following updates to the FSAR
  - o dynamic soil pressure, differential settlement analysis, definition of the supporting medium used for calculating the static load demands, and the design and analysis procedure (Appendix 3B)

- No RAIs for Section 3.8.4

Section 3.8 – Design of SC-I Structures (Continued)

Section 3.8.5 - Foundations

- Differences from DC
  - In the SDAA, the nonlinear stability analysis is performed only for the SC-I portion of the surface-based CRB.
  - In the SDAA, the peak bearing pressure values are calculated using a methodology tailored to the capabilities of the software utilized, ANSYS.

- Audit Responses
  - 12 questions resolved in audit

- No RAIs for Section 3.8.5

Section 3.9.2 –
Dynamic Testing and Analysis of Systems, Components, and Equipment

- Differences from DC
  - Updated requirements from Regulatory Guide 1.20 Revision 3 to 1.20 Revision 4
  - Updated requirements from the American Society of Mechanical Engineers (ASME) Operations and Maintenance (OM) Code, 2012 Edition to ASME OM Code, 2017 Edition
  - Comprehensive vibration assessment program (CVAP) startup instrumentation changed from strain gauges and accelerometers to dynamic pressure sensors
  - Removed Combined Operating License (COL) Item 3.9-14 (DC density wave oscillation (DWO) carveout)
  - Reactor vessel internals (RVI) were evaluated for updated US460 loads
  - Revised flow-induced vibration (FIV) analyses with US460 design changes and updated flowrates and operating conditions
  - Added inlet flow restrictor (IFR) cavitation evaluations with consideration of DWO to CVAP analysis report
  - Added an analysis case of both reactor vent valves (RVVs) actuating to TR-121517-P, "NuScale Power Module Short-Term Transient Analysis"

Section 3.9.2 – Dynamic Testing and Analysis of Systems, Components, and Equipment (Continued)

- Audit Responses
  - 35 audit questions resolved
    - Added reference to startup test abstracts from Section 14.2 to FSAR 3.9.2.1
    - Updated language of NPM prototype classification options to match TR-121353-P, "NuScale Comprehensive Vibration Assessment Program Analysis Technical Report"
    - Provided summary of TF-3 (steam generator fluid-induced vibration (SGFIV)) flow testing results for review
    - Provided tube sliding and wear evaluation caused by the DWO transient
    - Provided DWO fatigue usage for tube-to-tubesheet weld, tubes, and tubesheet in the feedwater plenum

- RAI Results
  - RAI 10111 (Question 3.9.2-1) - Confirmation that steam generator (SG) integrity is maintained during Service Level D events
    - Provided preliminary Service Level D fatigue results for RVI and SG components
    - Resulted in no changes to the SDAA

Section 3.9.2 – Density Wave Oscillation

- 10 audit questions resolved
  - 1 in Section 3.9.1, 9 in Section 3.9.2
- No DWO RAIs in Chapter 3
- Analyses
  - Section 3.9.1
    - DWO Service Level A Transient
    - NPM lifetime limit for time in DWO
  - Section 3.9.2
    - Structural integrity of steam generator during DWO

Acronyms

ASME	American Society of Mechanical Engineers
COL	Combined Operating License
CRB	Control Building
CVAP	Comprehensive Vibration Assessment Program
DB	Double Building
DC	Design Certification
DWO	Density Wave Oscillation
FIV	Flow-Induced Vibration
FSI	Fluid-Structure Interaction
IFR	Inlet Flow Restrictor
ISRS	In-Service Response Spectra
ITP	Initial Test Program
NPM	NuScale Power Module
NRC	Nuclear Regulatory Commission
OM	Operations and Maintenance
RAI	Request for Additional Information
RC	Reinforced Concrete
RVI	Reactor Vessel Internals
RVV	Reactor Vent Valve
RWB	Radioactive Waste Building
RXB	Reactor Building
SC	Steel-Plate Composite
SG	Steam Generator
SGFIV	Steam Generator Fluid-Induced Vibration
SSI	Soil-Structure Interaction
SDAA	Standard Design Approval Application

Chapter 5

Reactor Coolant System and Connecting Systems

February 4, 2025

Presenters:

Wendy Reid and Erin Whiting

Chapter 5 – Reactor Coolant System and Connecting Systems

- Section 5.1, Summary Description
- Section 5.2, Integrity of Reactor Coolant Boundary
- Section 5.3, Reactor Vessel
  - Pressure and Temperature Limits Methodology Technical Report (TR-130877-P, Revision 1)
- Section 5.4, Reactor Coolant System Component and Subsystem Design

[1] Denotes changes made in revision 2 of the Standard Design Approval Application (SDAA) Final Safety Analysis Report (FSAR)

Chapter 5 and Pressure and Temperature Limits Technical Report Review

- Audit Questions
  - 59 questions in Chapter 5
  - 20 questions on Pressure and Temperature Limits Methodology Technical Report (PTLR)
- Request for Additional Information (RAI)
  - 1 RAI in Chapter 5
  - No RAIs on PTLR

Section 5.1 - Summary Description

- Change in primary and secondary operating pressures, temperatures, and flow rates as a result of the power uprate
- Design pressure is the same for primary (inside the reactor vessel) and secondary (inside the steam generator tubes. Both design pressures changed from 2100 psi to 2200 psi
- Classification change for upper steam generator (SG) support for manufacturing concerns, requirements are consistent with American Society of Mechanical Engineers (ASME) code. [1]
- Reactor coolant system (RCS) volume change [1]

Section 5.2 - Integrity of Reactor Coolant Boundary

- Adopted 2017 ASME Boiler and Pressure Vessel and Operation and Maintenance Codes
- Change to leakage detection sensitivity requirement
  - No change to the equipment or system capabilities
  - No change to Technical Specifications for RCS leakage
- Change from three to two reactor vent valves
- The set points and design of the reactor safety valves (RSVs) changed
  - Setpoints increased with the design pressure increase and staggered
  - Minimum design capacity per valve increased[1]
  - Design change from pilot operated to spring operated RSVs
- Added the containment isolation test fixture (CITF) [1]
- Augmented preservice examination for the Class 1 containment isolation valves (CIVs) and CITF on each of the four chemical and volume control system lines [1]
- Augmented examinations applied to welds between containment vessel (CNV) and CIVs to support Branch Technical Position 3-4 requirements as discussed in Section 3.6 [1]
- Low temperature overpressure protection setpoints changed due to material change for lower reactor pressure vessel (RPV)

Section 5.2 - Integrity of Reactor Coolant Boundary (Continued)

Changes to Table 5.2-3 reporting materials for reactor coolant pressure boundary components and support materials

- Lower RPV change discussed in Section 5.3
- Added additional permissible materials to increase manufacturing flexibility for the combined license applicant
- Changes for consistency and completeness in response to audit questions
- Reconciled naming conventions with internal design documents

Section 5.3 - Reactor Pressure Vessel

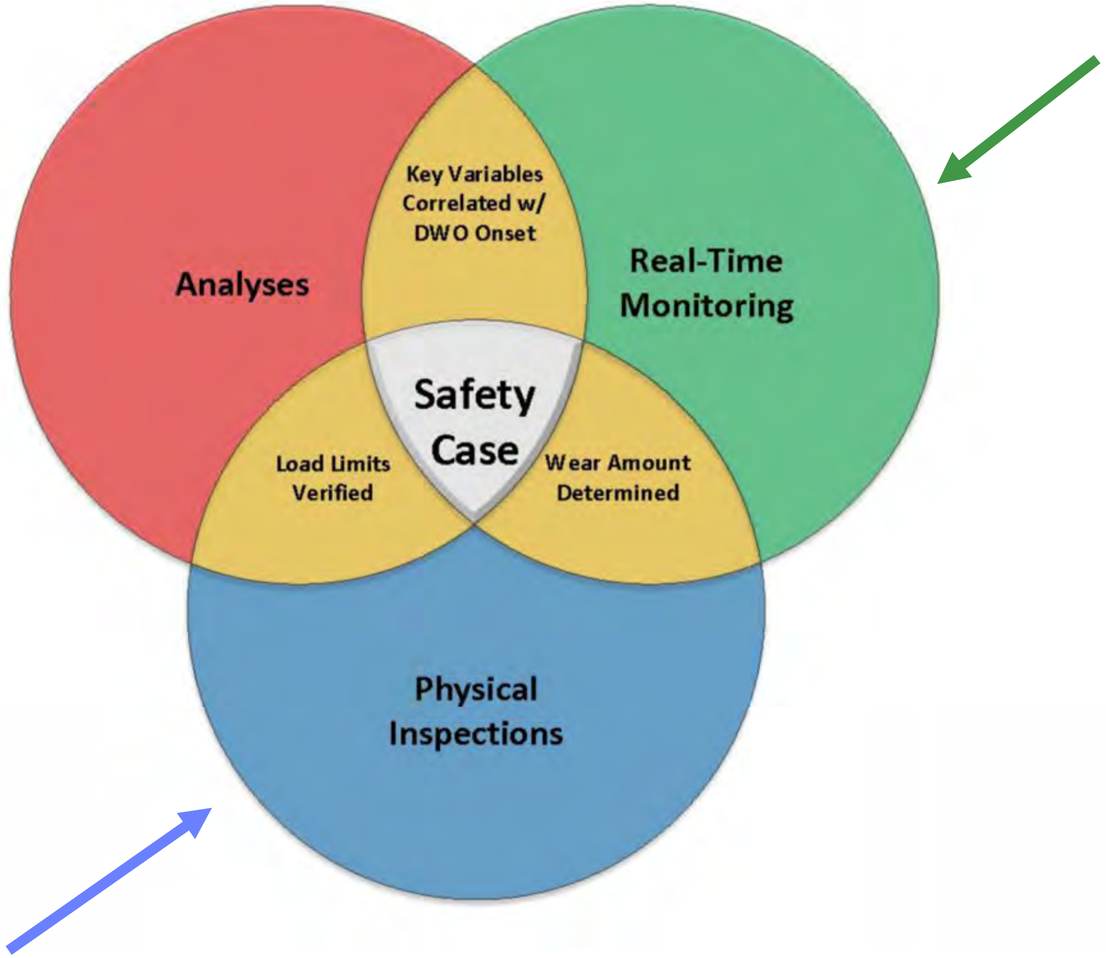
- Material change for the lower RPV to FXM-19 austenitic stainless steel
  - Change reflected in the PTLR methodology Technical Report
    - Upper RPV limiting ferritic component susceptible to fluence effects [1]
  - Expansion to Combined Operating License (COL) Item 5.3-1 for PTLR [1]
  - Exemptions for 10 CFR 50.60 fracture toughness (Appendices G and H) for and 10 CFR 50.61 pressurized thermal shock
  - Use of austenitic stainless steel in lower RPV
    - Superior ductility compared to ferritic materials
    - Less susceptible to the effects of neutron and thermal embrittlement than ferritic materials
    - Regulatory beltline concerns not an issue
    - No Appendix H material surveillance program required
- Removal of COL Item concerning onsite cleaning of the RPV during construction
- Removal of the flow diverter [1]
- Change to seismic restraint feature between lower CNV and lower RPV [1]

Section 5.4 - Reactor Coolant System Component and Subsystem Design

- Decay heat removal system (DHRS)
  - System size change
  - Credited in safety analysis; required for containment peak pressure response to a loss-of-coolant accident (LOCA) (added to Chapter 5)[1]
  - Details on emergency core cooling system (ECCS) venting to limit hydrogen accumulation in the RPV during containment isolation [1]
  - Design meets the intent of SECY 94-084 by achieving passively cooled, safe shutdown conditions within 36 hours [1]
    - DHRS performance cases achieve a passively cooled, safe shutdown condition within 36 hours.
    - Added off-nominal cases, including the worst case DHRS case (single train, high inventory), which provides sufficient cooling to below 450 degrees Fahrenheit RCS average temperature in 36 hours.
  - Actuation valve accumulator pressure details added [1]
- Expanded description of SG supports [1]
- Added description of flow paths between the riser and downcomer [1]
- SG tube plugging criterion description changed due to bracketed value in Technical Specifications [1]

Section 5.4 - Reactor Coolant System Component and Subsystem Design (Continued)

- Design Certification (DC) approach
  - Ensure density wave oscillation (DWO) preclusion with inlet flow restrictor (IFR) sizing
  - DWO onset evaluation subject to future analysis
  - SG integrity to be determined during operation with DWO
  - COL Item 3.9-14 (DC DWO carveout)
- DWO Safety Case [1]
  - Three pillars provide defense-in-depth safety case
  - Real-Time Monitoring
    - Approach temperature description and figure
    - Link to Section 13.5.2 procedure development
  - Physical Inspections
    - Augmented examination requirements for SG tubes and IFRs
  - Added IFR loss coefficient range



Acronyms

ASME	American Society of Mechanical Engineers
CITF	Containment Isolation Test Fixture
CIV	Containment Isolation Valve
CNV	Containment Vessel
COL	Combined Operating License
DC	Design Certification
DHRS	Decay Heat Removal System
DWO	Density Wave Oscillation
ECCS	Emergency Core Cooling System
FSAR	Final Safety Analysis Report
IFR	Inlet Flow Restrictor
LOCA	Loss-of-Coolant Accident
NRC	Nuclear Regulatory Commission
PTLR	Pressure-Temperature Limits Report
RAI	Request for Additional Information
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
RSV	Reactor Safety Valve
SG	Steam Generator
SDAA	Standard Design Approval Application

Presentation to the Advisory Committee on Reactor Safeguards Subcommittee

Staff Review of NuScale's US460 Standard Design Approval Application Final Safety Analysis Report, Revision 1

Chapter 3, Sections 3.7, 3.8, 3.9.2

February 4, 2025
(Open Session)

NuScale SDAA FSAR Chapter 3 Review

(Sections 3.7, 3.8, 3.9.2)

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NuScale SDAA FSAR Chapter 3 Review

(Sections 3.7, 3.8, 3.9.2)

Overview

- NuScale submitted Chapter 3, "Design of Structures, Systems, Components and Equipment," Revision 1, of the NuScale SDAA FSAR on October 31, 2023.
- NRC performed a regulatory audit as part of its review of Chapter 3, from March 2023 to June 2024.
- Questions raised during the audit were resolved within the audit. All RAI responses were acceptable.
- Staff completed the review of Chapter 3 (Sections 3.7, 3.8, 3.9.2) and issued an advanced safety evaluation to support the ACRS meeting.
- Since providing draft SE to ACRS on 1/4/2025, **Section 3.7** was updated regarding acceptability of strong-motion time history being less than 6 seconds; **Section 3.8** was updated regarding demand over capacity ratio (DCR) values for Reactor Building (RXB) calculated and assessed by both element-based and panel section-based approaches.

NuScale SDAA FSAR Chapter 3 Review

- 3.7 – Seismic Design
  - Section 3.7.1 – Seismic Design Parameters
  - Section 3.7.2 – Seismic System Analysis
  - Section 3.7.3 – Seismic Subsystem Analysis
  - Section 3.7.4 – Seismic Instrumentation
- 3.8 – Design of Category I Structures
  - Section 3.8.1 – Concrete Containment (N/A)
  - Section 3.8.2 – Steel Containment
  - Section 3.8.3 – Concrete and Steel Internal Structures of Steel or Concrete Containments (N/A)
  - Section 3.8.4 – Other Seismic Category-I Structures
  - Section 3.8.5 – Foundations
- Section 3.9.2 – Dynamic Testing and Analysis of Systems, Structures, and Components

Section 3.7.1 – Seismic Design Parameters

Significant Differences Between NuScale DCA and SDAA:

1. <u>Structural Damping Values Used in Seismic Analysis:</u>
   - ❖ DCA used reinforced concrete (RC) for safety-related structures and applied a uniform 4% damping for both cracked and uncracked RC members to generate in-structure response spectra (ISRS).
   - ❖ SDAA used RC and steel-plate composite (SC) for safety-related structures, utilizing a hybrid damping scheme to generate ISRS; 7% and 5% for cracked RC and SC, and 4% and 3% for uncracked RC and SC, respectively.
   - ❖ In both cases, cracked and uncracked ISRS are enveloped to establish design-basis ISRS.
   - ❖ Staff finds the SDAA damping values (percent of critical damping) for both cracked and uncracked RC and SC cases acceptable, as they align with the guidance in RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants."

Section 3.7.1 – Seismic Design Parameters

Significant Differences Between NuScale DCA and SDAA:

2. Supporting Media for Seismic Category I Structures:

- ❖ DCA considered four supporting media types: soft soil, firm soil/soft rock, rock, and hard rock.
- ❖ SDAA, by contrast, utilized three supporting media types: soft soil, rock, and hard rock.
- ❖ In both cases, seismic responses for each soil type were enveloped to generate the design-basis seismic demand.
- ❖ Staff finds the SDAA supporting media for Seismic Category I structures acceptable, as they adequately represent the range of expected site soil conditions.

Section 3.7.2 – Seismic System Analysis

Significant Differences Between NuScale DCA and SDAA:

1. <u>Different Methodologies for Seismic Soil-Structure-Fluid Interaction (SSFI) Analysis:</u>
   - ❖ DCA employed a two-step methodology to address SSFI effects, involving separate soil-structure interaction and fluid-structure interaction analyses, which included simplifications and approximations.
   - ❖ SDAA adopted a single, integrated methodology to evaluate SSFI effects under design-basis ground motion.
   - ❖ SDAA methodology is based on Topical Report (TR-0118-58005), "Improvements in Frequency Domain Soil-Structure-Fluid Interaction Analysis," which was approved in 2022.
   - ❖ Staff verified that seismic SSFI analysis for US460 standard design was performed in compliance with the applicable limitations and conditions specified in the approved topical report.

Section 3.7.2 – Seismic System Analysis

Significant Differences Between NuScale DCA and SDAA:

2. <u>Different Analysis Models Due to Design Changes:</u>

- ❖ SDAA incorporates significant design changes from DCA, including six NPMs, updated NPM models, resized UHS, relocated CRB, and new SC walls.
- ❖ DCA employed a Triple Building Model (including RXB, CRB, and RWB) for design-basis seismic demand calculations, whereas SDAA used a Double Building Model (including RXB and RWB) with an independently modeled CRB.
- ❖ Staff determined that updated models used in seismic system analysis for US460 standard design are acceptable, as they adhere to applicable industry standards and DSRS acceptance criteria.

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Section 3.7.2 – Seismic System Analysis

Significant Differences Between NuScale DCA and SDAA:

3. <u>Different Approaches to Addressing the Results of Parameter Sensitivity Studies:</u>
   - ❖ Both DCA and SDAA conducted in-structure response spectrum (ISRS) sensitivity studies to evaluate parameter variations, including structure-soil separation, empty dry dock, and modularity.
   - ❖ In both cases, the soil-separation scenario resulted in a noticeable exceedance of the design-basis ISRS.
   - ❖ DCA addressed this exceedance by including a COL Item, requiring that site-specific ISRS in soil-separation conditions be demonstrated to remain bounded by the DCA design-basis ISRS.
   - ❖ SDAA addressed the exceedance differently, incorporating the soil-separation scenario into the design-basis ISRS analysis cases. The staff found this approach acceptable, as it directly integrates soil-separation effects into the design basis.

Section 3.7.3 – Seismic Subsystem Analysis

Significant differences between NuScale DCA and SDAA:

- Seismic Analysis of Buried Seismic Category I Piping, Conduits, and Tunnels:
  - DCA did not include buried piping or conduits, and the tunnel connecting RXB and CRB was analyzed as part of CRB.
  - SDAA, however, included an underground reinforced-concrete duct bank containing conduits that connect RXB and CRB.
  - Staff determined the seismic analysis of SDAA buried Seismic Category I structures and systems is acceptable, as it was conducted in accordance with applicable industry standards and DSRS acceptance criteria.

Section 3.8 - Design of Category I Structures

(Control Building (CRB) and Reactor Building (RXB))

- ## Section 3.8.1 - Concrete Containment: N/A
- ## Section 3.8.2 - Steel Containment
  - Significant differences between NuScale DCA FSAR and SDAA FSAR include:
    - Reconfigured boundary condition between the bottom heads of CNV and RPV.
    - Design parameter
      - /operating parameters: (50 psig/1,200 psig/600 °F vs. 60 psig/1,050 psig/550 °F)*

    **(external design pressure/internal design pressure/design temperature)*
  - SDAA SE conclusion is the same as DCA SE conclusion.

Section 3.8.4 - Other Seismic Category I Structures

- Significant differences between NuScale DCA FSAR and SDAA FSAR include:
  - Methodology for the evaluation of seismic Category I and II structures (RXB and CRB) is per the requirements provided in TR-0920-71621-P- A, Rev. 1, "Building Design and Analysis Methodology for Safety-Related Structures."
- SDAA SE conclusion is the same as DCA SE conclusion.

Section 3.8.5 - Foundations

- Significant differences between NuScale DCA FSAR and SDAA FSAR include:
  - The embedment of CRB:
    - » In the SDAA, the CRB is modeled as a surface-founded structure, conservatively ignoring the 5-ft embedment of the foundation for its stability analysis.
    - » In the DCA, the CRB with an embedment depth of 55 feet is modeled as an embedded structure with backfill surround it for its stability analysis.
- SDAA SE conclusion is the same as DCA SE conclusion.

Section 3.9.2 - Dynamic Testing and Analysis of Systems

- Piping Vibration, Thermal Expansion, and Dynamic Effects
- Comprehensive Vibration Assessment Program (CVAP) of Reactor Vessel Internals (RVI) and Steam Generators (SG)
  - Dynamic Response Analysis under Operational Flow Transients and Steady State Conditions
    - TR-121353, Revision 2, “NuScale Comprehensive Vibration Assessment Program Analysis Technical Report”
  - Flow-Induced Vibration (FIV) Validation Testing and Inspection
    - TR-121354, Revision 1, “NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report”
- Dynamic System Analysis of the RVI and SG under ASME Service Level D Conditions
  - Seismic Loading Analysis
    - TR-121515, Revision 1, “US460 NuScale Power Module Seismic Analysis”
  - Short-Term Transient Loading Analysis
    - TR-121517, Revision 1, “NuScale Power Module Short-Term Transient Analysis”
  - Stress and Deflection Evaluations
    - RAI 10111, Question 3.9.2-1 (Resolved)

Section 3.9.2 – DCA Deferred or Unresolved

- CVAP-Steam Generator Qualification
  - Qualification of SG components due to DWO-induced dynamic loads carveout in the DCA
  - SG validation testing deferred to COL applicant
    - Elimination of significant SG tube FIV not demonstrated
- Service Level D evaluations
  - Did not include hard rock (there is a COL item for site-specific seismic analysis)

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Section 3.9.2 – CVAP - Dynamic Response Analysis

- Significant differences between NuScale DCA and SDAA FSARs:
  - Higher flow speeds (25% more power) –> stronger FIV loads
  - Reduced DWO-induced dynamic loads and impacts on SG
  - SG inlet flow restrictors (IFRs) redesigned – no longer at risk for FIV
  - SG tube support system redesigned
  - Secondary flow piping and valve systems redesigned to minimize FIV risk
- SDAA SE conclusion is complete, unlike DCA SE conclusion
  - Qualification of SG due to DWO-induced dynamic loads is no longer a "carveout"
  - TF-3 SG validation testing shows minimal risk of significant FIV

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Section 3.9.2 – CVAP – DWO-Induced Loads

- DCA (and early SDAA) concerns:
  - During reverse DWO flow the boiling boundaries in SG tubes might approach the SG inlets leading to:
    - Cavitation erosion
    - Condensation-induced water hammer (CIWH)
  - Significant number of DWO cycles initially allowed over plant life

Section 3.9.2 – CVAP – DWO-Induced Loads

❖ Three-tiered SDAA safety finding:

- ❑ Boiling boundaries are highly unlikely to approach SG inlets; cavitation and CIWH are therefore highly unlikely
  - Chapter 5 finding confirms NuScale’s analysis methods are acceptable for simulating boiling boundary heights
  - NRC Office of Research independent analysis confirms CIWH is highly unlikely
- ❑ In the unlikely event cavitation or CIWH occurs, NuScale estimates low tube and IFR wear
  - Reduced number of allowable cycles, small loads
- ❑ Finally, the SG inspection program is sufficient to capture any unexpectedly high wear (Section 5.4.1)
  - Full inspection during first refueling outage
  - Afterwards, full inspections over 72 effective full power month intervals

Section 3.9.2 – CVAP – TF-3 SG Validation Testing

- On-site staff audit of facility and flow testing at SIET in Piacenza, Italy in October 2024
  - Facility is a reasonable representation of a partial NPM SG
    - Tightly fitting SG tubes and supports, no need to account for SG support system design differences
  - Test data are sufficient to evaluate risk of significant FIV

- Tested over a comprehensive range of flow rates up to 250% of equivalent NPM 100% power
  - No evidence of Vortex Shedding (VS) or Fluid-Elastic Instability (FEI)

Section 3.9.2 – CVAP – FIV Validation Testing and Inspections

- Significant differences between NuScale DCA and SDAA FSARs include:
  - Replaced internal vibration sensors with dynamic pressure sensors for initial startup testing
- SDAA SE conclusion
  - SG TF-3 testing demonstrated that dynamic pressure sensors should "hear" unexpectedly high RVI or SG vibration during initial startup testing

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Section 3.9.2 – Dynamic System Analysis of the RVI and SG under Service Level D Conditions

- Significant differences between NuScale DCA FSAR and SDAA FSAR:
  - Different building, fewer NPMs (6 vs 12)
  - Seismic loads include soft soil and hard rock ground conditions
    - Hard rock events include significant higher frequency loads which align with SG modes of vibration
  - Upper and lower riser interface redesigned
  - RVI hanger plate interface redesigned
  - Different (but improved) modeling approaches

- SDAA SE conclusion is more comprehensive, unlike DCA SE conclusion
  - Thorough assessment of RVI and SG stresses and deflections show minimal risk of damage

Non-Proprietary

Section 3.9.2 – Dynamic System Analysis of the RVI and SG under Service Level D Conditions

- Seismic loads:
  - Simpler, more comprehensive and accurate modeling approach than in DCA
  - Bound all soil types and NPM locations
- Transient loads:
  - Short blow-down events
  - Loads order of magnitude lower than seismic

Section 3.9.2 – Dynamic System Analysis of the RVI and SG under Service Level D Conditions

- RVI stress analyses:
  - Bounding response spectrum method for overall structure
    - Confirmed to be reasonably bounding by comparing to single transient analysis
  - Bounding engineering calculations for joints and simple structures
    - Highly conservative
- SG stress analyses:
  - Full transient analyses for bounding soft soil and hard rock load cases – comprehensive and accurate
- All stresses within allowable limits

NuScale SDAA FSAR Chapter 3 Review

(Sections 3.7, 3.8, 3.9.2)

Conclusion

- While there are some differences between the DCA and the SDAA, the staff found that the applicant provided sufficient information to support the staff's safety finding.
- The staff found that all applicable regulatory requirements were adequately addressed.

Presentation to the Advisory Committee on Reactor Safeguards Subcommittee

Staff Review of NuScale's US460 Standard Design Approval Application Final Safety Analysis Report, Revision 1

Chapter 5

"Reactor Coolant System and Connecting Systems"

February 4, 2025
(Open Session)

NuScale SDAA FSAR Chapter 5 Review

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NuScale SDAA FSAR Chapter 5 Review

Overview

- NuScale submitted Chapter 5, “Reactor Coolant System and Connecting Systems,” Revision 1, of the NuScale SDAA FSAR on October 31, 2023
- Responses to Audit questions and RAIs were acceptable
- NRC staff completed the review of Chapter 5 and issued an advanced safety evaluation to support the ACRS Subcommittee meeting
- No significant changes between draft SE provided to ACRS on 1/4/25 and SE submitted on 1/29/25

NuScale SDAA FSAR Chapter 5 Review

Sections

- Section 5.1 – Summary Description
- Section 5.2 – Integrity of Reactor Coolant Boundary
- Section 5.3 – Reactor Vessel
- Section 5.4 – Reactor Coolant System Component and Subsystem Design

NuScale SDAA FSAR Chapter 5 Review

Section 5.2.1 Compliance with Codes and Cases

- Significant differences between NuScale DCA FSAR and NuScale SDAA FSAR include:
  - ASME Codes of Record (2017, vice 2013 BPV/ 2012 OM)
  - Use of ASME Code Cases used (while different, all approved in RGs)
- SDAA SE conclusion same as DCA SE conclusion

NuScale SDAA FSAR Chapter 5 Review

Section 5.2.3 Reactor Coolant Pressure Boundary Materials

- Significant differences between NuScale DCA FSAR and NuScale SDAA FSAR:
  - Lower RPV section flange shell RPV bottom head was SA—508 Grade 3, Class 1 for the DC vs. Lower Vessel (Lower Head, Shell and Flange) is SA-965 FXM-19 for the SDAA. This material is acceptable for ASME Code Class 1 applications
  - Welding material is SFA-5.4 Type E209, E240/SFA-5.9 Type ER 209,ER240 and is compatible to SA-965 FXM-19
  - FXM-19 and Type 2XX weld filler metal specify 0.04 maximum carbon and a Ferrite Number in the range of 5FN to 16FN which meets ASME Code
  - TR-130721 Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel concludes the US460 SDAA design meets the requirements of GDC 14, GDC 15, GDC 31 and GDC 32
  - Section 5.3 covers additional technical information in more detail

- SDAA SE conclusion same as DCA SE conclusion

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NuScale SDAA FSAR Chapter 5 Review

Section 5.3 Reactor Vessel

- Significant differences between NuScale DC FSAR and NuScale SDAA FSAR include:
  - Use of austenitic stainless steel for the lower NPM
  - Exemptions 6 and 7 from ferritic steel requirements inapplicable to austenitic stainless steel lower NPM
    - » Requirements of 10 CFR 50.60; 10 CFR 50.61, and 10 CFR 50 Appendices G (fracture toughness requirements) and H (reactor vessel surveillance program), do not apply to the lower NPM
  - At the COL stage, the final as-built design transients, and material properties of the reactor pressure vessel will be evaluated to confirm that they are bounded by those used in the PTL methodology (SDAA COL Item 5.3-1)

NuScale SDAA FSAR Chapter 5 Review

Section 5.3 Reactor Vessel (contd.)

- NuScale SDAA SE conclusion is different from NuScale DCA SE conclusion because the SDAA design includes austenitic stainless steel lower NPM instead of ferritic steel lower NPM in the DCA
  - Consequently, the SDAA SE includes granting exemptions from some ferritic requirements for the lower NPM
  - In addition, pressure-temperature limits methodology approval differs (next slide)

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NuScale SDAA FSAR Chapter 5 Review

Pressure Temperature Limits Methodology Report

- Significant differences between NuScale DCA FSAR and NuScale SDAA FSAR include:
  - SDAA design is never beltline limited in the lower NPM
  - Pressure-Temperature curves are primarily limited by geometric discontinuities in locations with essentially no neutron embrittlement
  - At the COL stage, the final as-built design transients, and material properties of the reactor pressure vessel will be evaluated to confirm that they are bounded by those used in the PTL methodology (SDAA COL Item 5.3-1)
- SDAA SE conclusion is not the same as DCA SE conclusion because of changes to the design and expanded COL Item 5.3-1

NuScale SDAA FSAR Chapter 5 Review

Section 5.4.1 Steam Generators

- Significant differences between NuScale DCA FSAR and NuScale SDAA FSAR
  - Inlet flow restrictor (IFR) design
    - New center-flow orifice design
    - IFRs expanded against the tube inside surface, not attached to a plate outside the tubes
    - Removed for SG inspection and maintenance activities, including IFR inspection
  - SG Program COL Item 5.4-1 includes additional inspections for first module to undergo a refueling outage
    - 20 percent of the tubes will be inspected during each refueling outage over the 72 effective full-power months after the first refueling outage (100 percent inspection)
  - SG Program technical specifications
    - Structural integrity performance criterion (SIPC) for steady-state full-power operation is based on ASME Code for external pressurization (2xΔP) rather than burst (3xΔP)
    - Tube plugging criterion not changed from [40%] through-wall, but new analysis based on new support design and SIPC

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NuScale SDAA FSAR Chapter 5 Review

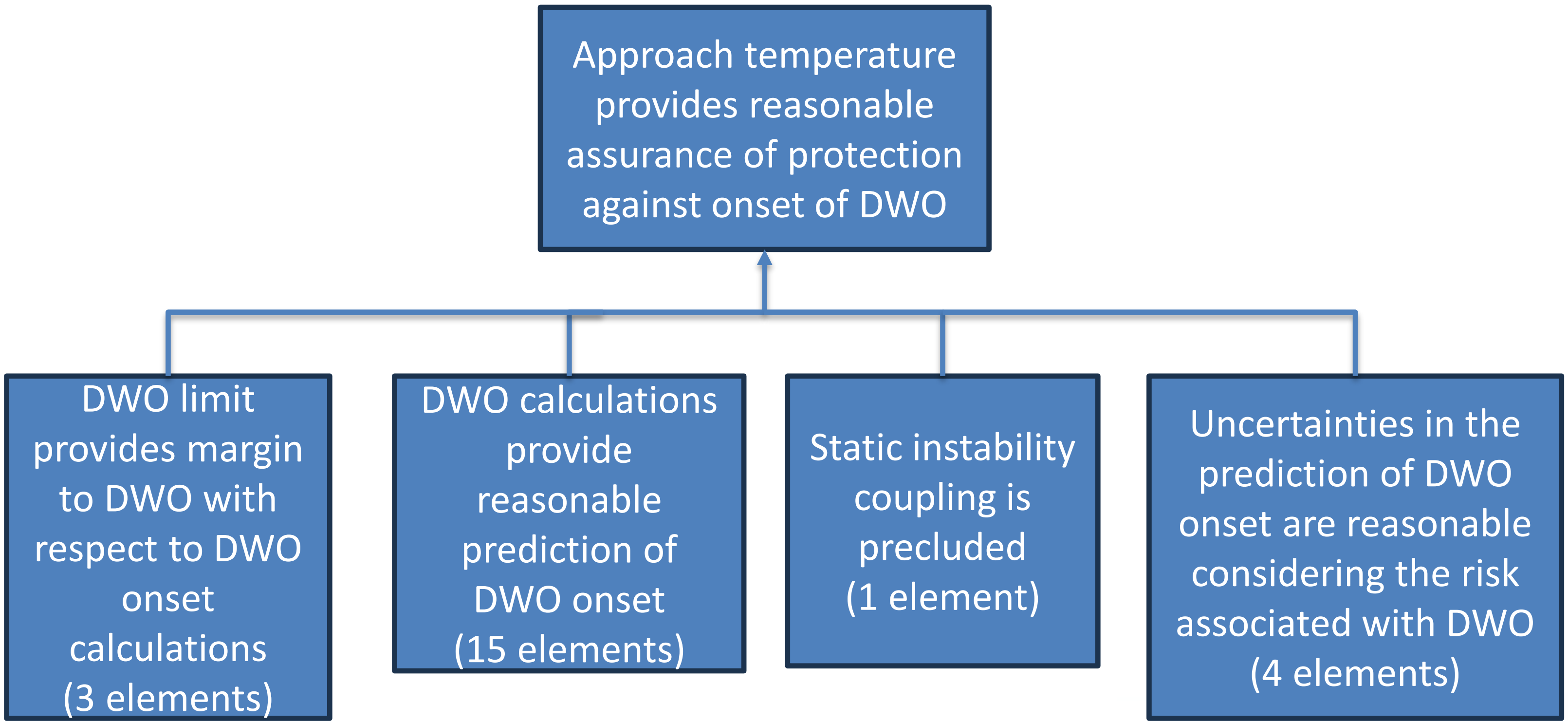
Section 5.4.1 Steam Generators (Continued)

Approach Temperature Limit for Density Wave Oscillation (DWO) Instability

- FSAR Section 5.4.1.3 describes the approach temperature
  - $\Delta T_{approach} = T_{RCS,hot} - T_{SG,exit}$
- Adequacy of approach temperature limit demonstrated through NRELAP5 calculations
  - Approach temperature limit demonstrates margin to DWO onset with respect to NRELAP5 predicted DWO onset

NuScale SDAA FSAR Chapter 5 Review
Section 5.4.1 Steam Generators (Continued)
Approach Temperature Limit Review Framework

- NRC staff evaluated 23 elements to support finding



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NuScale SDAA FSAR Chapter 5 Review

Approach Temperature Limit Review Framework (continued)

5.4.1.4.2.1.1	**The approach temperature limit provides margin to DWO with respect to DWO onset calculations**
	Approach temperature limit is always reached before DWO onset is predicted to occur
	Calculations cover an adequate range of operating conditions for the NPM steam generators
	Calculations use suitably conservative input

5.4.1.4.2.1.4	**Uncertainties in the prediction of DWO onset are reasonable considering the risk associated with DWO**
	Consistent with defense-in-depth philosophy
	Maintains sufficient safety margins
	Risk is small and consistent with the intent of the Commission's Safety Goal Policy Statement
	Performance measurement strategies

NuScale SDAA FSAR Chapter 5 Review

Approach Temperature Limit Review Framework (continued)

5.4.1.4.2.1.2	DWO onset calculations provide reasonable insight into the likelihood of DWO		
	5.4.1.4.2.1.2.1	The evaluation model contains the adequate modeling capabilities	
		4 elements	
	5.4.1.4.2.1.2.2	The evaluation model has been adequately assessed against experimental data	
		The experimental data used for assessment is appropriate	
			7 elements
		The evaluation model has demonstrated the ability to predict DWO over the analysis envelope	
			4 elements

NuScale SDAA FSAR Chapter 5 Review

Conclusions - Approach Temperature Limit Review

- Approach temperature limit provides reasonable assurance of adequate protection against DWO onset for the SG design
  - Approach temperature limit provides margin to DWO with respect to DWO onset calculations (see SER Section 5.4.1.4.2.1.1)
  - DWO onset calculations provide reasonable insight into the likelihood of DWO (see SER Section 5.4.1.4.2.1.2)
  - Static instability coupling is precluded (see SER Section 5.4.1.4.2.1.3)
  - Uncertainties in the prediction of DWO onset are reasonable considering the risk associated with DWO (see SER Section 5.4.1.4.2.1.4)
- The staff approval of the approach temperature limit does not approve the general use of the NRELAP5 evaluation model for use in DWO calculations
  - Limitation includes the prediction of DWO onset or the prediction of thermal-hydraulic behavior during DWO
  - The staff is unable to determine the adequacy of the evaluation model due to gaps in model assessment (see SER Section 5.4.1.4.2.1.2)

NuScale SDAA FSAR Chapter 5 Review

Section 5.4.3 Decay Heat Removal System

- Notable changes between NuScale DCA FSAR and NuScale SDAA FSAR include:
  - increase in number of condenser tubes, average shorter tube length, lower condenser elevation, lower UHS water level
  - credited in the revised LOCA evaluation model
  - new NRELAP5 basemodel changes related to DHRS such as additional heat structures and changes to pool nodalizations
- SDAA SE conclusion similar to DCA SE conclusion except with inclusion of LOCA-related requirement

NuScale SDAA FSAR Chapter 5 Review

Conclusions

- While there are some differences between the DCA and the SDAA, the staff found that the applicant provided sufficient information to support the staff's safety finding
- The staff found that all applicable regulatory requirements were adequately addressed

Meeting Title	**Open Session NuScale Subcommittee on Staff's Evaluation of NuScale Standard Design Approval Section 3.7, 3.8 and 3.9.2 and Chapter 5, including DWO**

Attendee

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Getachew Tesfaye	NRR
Matt Sunseri	ACRS
Shandeth Walton	ACRS
Ron Ballinger	ACRS
Larry Burkhart	ACRS
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David Yeager	
Courtney Goodwill	
Jim Osborn	NuScale
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Sarah Bristol	NuScale
Kevin Drost	NuScale
Rim Nayal	NuScale
Nicholas Mowers	
Eric Matthews	NuScale
R Snuggerud	NuScale
Taylor Zindren	
Wendy Reid	NuScale
Omer Erbay	NuScale
Daniel Diefendorf	NuScale
Cindy Williams	NuScale
Tammy Skov	ACRS
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