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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 DESIGN-CENTERED SUBCOMMITTEE

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WEDNESDAY

JANUARY 15, 2025

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

COMMITTEE MEMBERS:

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DENNIS BLEY

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

MICHAEL SNODDERLY

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

8:30 a.m.

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

I am Walt Kirchner, chair of today's subcommittee meeting. ACRS members in attendance in person are Ron Ballinger, Greg Halnon, Craig Harrington, Bob Martin, Scott Palmtag, and Tom Roberts. ACRS members in attendance virtually via Teams are Vicki Bier, Vesna Dimitrijevic, David Petti, Matt Sunseri, and myself. We have one of our consultants participating in-person, Steve Schultz, and one of our consultants participating virtually via Teams, Dennis Bley. If I have missed anyone, either ACRS members or consultants, please speak up now.

(No response.)

CHAIR KIRCHNER: Michael Snodderly of the ACRS staff is the Designated Federal Officer for this meeting.

No member conflicts of interest were identified for today's meeting, and I note that we have a quorum, as well.

During today's meeting, this Subcommittee will receive a briefing on the staff's evaluation of NuScale Power LLC's US460 Standard Design Approval Application; Chapter 3, Design of Structures, Systems, Components, and Equipment; Chapter 16, Technical Specifications; and Loss-of-Coolant Accident Evaluation Model Topical Report. We will also be briefed on the status of high-impact technical issues by the NuScale staff.

We previously reviewed the certified NuScale US600 design, as documented in our July 29, 2020 letter report, "Report on the Safety Aspects of the NuScale Small Modular Reactor." Like the staff, we are performing a delta review between the two designs, including a power uprate from 50 to 77 megawatts electric per module.

We are reviewing these chapters as part of our statutory obligation under Title 10 of the Code of Federal Regulations, Part 52, Subpart E, Section 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 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 in the NRC public website and can be easily found by typing "about us ACRS" in the search field on NRC's homepage.

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 a request to make an oral statement from the public, but we have set aside time at the end of this meeting for 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/or 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 the 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 star-6 if you're on your phone.

Please do not use the Teams chat feature to conduct sidebar discussions related to the presentations; rather, limit the use of that function to report IT problems. We ask everyone in the room, please put all your electronic devices on silent mode and mute your laptop microphone and speakers. In addition, please keep sidebar discussions in the room to a minimum, since the ceiling microphones are live.

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

Finally, if you have any feedback for the ACRS about today's meeting, we encourage you to fill out the public meeting feedback form on the NRC's website.

And now I just want to make a personal note. Here in Santa Fe, it's a nice, balmy 15 degrees. It's dark. We have a full moon out. On the West Coast, where our colleagues from NuScale are, it's probably much darker out there. I hope the moon's out, it's spectacular.

Thank you. We normally try and schedule the NuScale meetings for afternoon sessions, but, given the amount of material we want to cover today, NuScale has agreed to join us at -- I think it's about 5:30 a.m. out there in Oregon.

So, with that, thank you. And we'll now proceed with the meeting. I think we'll start with a opening statement from the NRC staff.

Greg, if you could direct things from there?

MEMBER HALNON: Go ahead.

MR. JARDANE: Good morning, Chair Kirchner, Vice Chair Halnon. Good morning to the ACRS Subcommittee members, NuScale participants, NRC staff, and members of the public. My name is Mahmoud Jardane, and I serve as the branch chief of the New Reactor Licensing Branch, responsible for the licensing of NuScale US460 design, in the Division of

New and Renewed Licenses, NRR.

Thank you for the opportunity --

CHAIR KIRCHNER: Could I interrupt you for a minute? Sorry. Would you pull your microphone closer? Those of us on Teams are not getting a good audio signal.

MR. JARDANE: All right, is that any better?

(No response.)

MR. JARDANE: Thank you for the opportunity today for the staff to present on their review of select NuScale US Standard Design Approval Application, SDAA, Chapters and topical reports. As you are aware, the staff is reviewing all chapters of the SDAA concurrently, with staggered completion dates based on the complexity of the chapter and the extent of the change from the certified NuScale US600 design.

Today, the staff will be presenting on their review of the fourth group of SDAA chapters and topical reports, including Chapter 16, Technical Specifications, and the Loss-of-Coolant Accident Evaluation Model Topical Report.

Earlier this year, the staff presented to the Subcommittee on Chapter 2, portions of Chapter 3, Chapters 7, 8, 9, 10, 11, 12, 13, and 14, portions of

Chapter 17, and Chapter 18. Staff is finalizing their review of the remaining SDAA chapters and topical reports and will inform the ACRS on the safety evaluation of the remaining chapters. Topical reports are available to the ACRS.

In today's meeting, the staff will focus on the deltas from the design specification that the NRC has approved and this Committee -- and the Committee reviewed in the past. Once again, thank you for the opportunity, and we look forward to a good discussion.

MEMBER HALNON: Thank you, MJ.

CHAIR KIRCHNER: Thank you. Greg, if I may, I noticed I made a mistake. We are not covering Chapter 3 today. That will be taken up next month, on February -- let me just get the date -- it will be on Tuesday, February 4. So, just Chapter 16, the LOCA Topical Report, and the HITI status. Thank you.

MR. TESFAYE: Excuse me. This is Getachew Tesfaye. Chair, I thought we were going to be following up on the past presentation on Chapter 3, to close up some --

(Simultaneous speaking.)

CHAIR KIRCHNER: Oh, yes. Yes, you're correct. I misspoke. Go ahead. Thank you.

MEMBER HALNON: Right. We have just a quick follow-up on Chapter 3. The Chapter 3 on February 4th will be the 3.7, 3.8, and 3.9.2 aspects of it. So it's a little confusing, the fact that we split Chapter 3 up in a couple different places. That's the clarification.

Tom, it's to you now, I believe.

MR. GRIFFITH: Yeah, good morning. Thomas Griffith, Licensing Manager, NuScale Power. I wanted to thank you for the opportunity to present on Chapter 16, the Loss-of-Coolant Accident Evaluation Model, and for the opportunity to present an update for the high-impact technical issues.

I would like to recognize the efforts by both the NRC staff and NuScale staff, and express my appreciation of the efforts that have gone into the review thus far. Furthermore, I would like to thank my NuScale counterparts for supporting such an early meeting. It is much appreciated for you all to be on the phone at such an early time, but it is essential that we have the opportunity to adequately discuss the topics that we're going to talk about today.

So, in conclusion, thank you for this opportunity, and thank you to the NRC staff, as well as NuScale.

MEMBER HALNON: Okay. Do you have -- who's presenting at this point?

MR. GRIFFITH: I have Gary Becker on the line.

MEMBER HALNON: Okay. Gary, you're up.

MR. BECKER: Good morning. Make sure I can be heard over there.

MEMBER HALNON: Yes. Give your name so the court reporter, too.

MR. BECKER: Thank you. Gary Becker. I am NuScale's Senior Regulatory Affairs Counsel. I've been with NuScale for almost 15 years, where I serve as our nuclear and licensing attorney. And, Walt, I can report that it is, indeed, 5:30 in the morning here, but unfortunately the moon is obscured, so it's just very dark.

I appreciate the opportunity to -- before you get into the heart of today's meeting - the opportunity to discuss the Subcommittee's draft Chapter 3 memorandum on the topic of standardization and downstream licensing reviews. We don't have any specific concerns with the memo's content, but I did want to take a few minutes to share our views on the topic and put those on the record.

NuScale recognizes the Subcommittee's

concerns for standardization and efficient licensing reviews. We agree. We were concerned with, and paid a lot of attention to, achieving standardization in our SDA application. We applied lessons learned from prior design reviews, up to and including the NuScale DCA, and from recent reactor construction projects. We drew upon the extensive experience of our engineering licensing personnel in those efforts.

Just a high level on our application approach, the NuScale design philosophy ensures standardization by consolidating as much of the safety-related SSCs as practical into the factory-fabricated NuScale power modules and the standardized reactor building, which will be constructed using modular construction.

Our SDAA follows a graded approach to design information, providing detailed descriptions in areas that are more important to safety and less information in areas of lesser safety significance. Reducing the bulk of our application in some areas optimizes the user's and NRC's attention on what is important versus what isn't.

As just one example, because our ultimate heat sink is inside of a protected seismic Category 1 structure, it doesn't really matter, for the sake of safe shutdown considerations, if a plant is water-cooled or air-cooled, next to a lake or in a desert. So we have left some of those specific design details to a downstream user.

The Subcommittee's Chapter 3 draft memorandum discusses some COL items in Chapter 3 as examples with respect to standardization. The design information provided in the SDAA prescribes a complete and final design, essentially complete and final design, that must be built by a licensed applicant referencing it, with limited and targeted allowances. One of those allowances: where final design information is not yet known or is not included in the FSAR description.

So, generally speaking, these are cases where it's not practical for NuScale to complete the design at this stage, or it's commercially prudent to provide some flexibility for the applicant. In such cases, the SDAA prescribes the safety implications, if any, that must be considered and completed in the detailed design.

Another allowance is where a license applicant changes an aspect of the design that is described in the FSAR, what we refer to as a departure. Where departures are anticipated, the SDAA prescribes confirmatory considerations in safety-significant areas. In all instances, the general requirements to address safety considerations for departures will apply.

I wanted to comment on -- note those areas, because these allowances and processes are central to making the standard design usable. While a high degree of standardization is important to realize the safety benefits intended by the Commission, and standardization is essential to achieve efficient plant licensing reviews, as you've noted, we view that undue stringency and rigidity in standardization would likely bring about the opposite result. That is, if standardization processes are too burdensome for future design applicants to use, they won't.

As the Subcommittee's draft memo observes, we expect a first-of-a-kind reference COL to complete any necessary design details and address site-specific features, and then commercial considerations will drive that design to be repeated to the extent possible in subsequent COLs. So these COL mechanisms compliment the SDAA to ensure standardized, as-built designs.

One comment on the topic of efficient

downstream licensing reviews. As noted in the Subcommittee's draft memo, efficient, nth-of-a-kind licensing reviews are essential to meeting the ADVANCE Act objectives, and processes to focus review on safety-significant issues would help in that regard.

For example, I discussed a moment ago the potential for COLA departures. If a departure doesn't alter the methodology or conclusions of the referenced design, then it should be dispositioned quickly in the staff and ACRS reviews. The design certification rules provide a 50.59-like process for that purpose, but NuScale has noted the SDA regulations do not provide something similar.

We've urged in a ongoing rulemaking to consider this issue, for the NRC staff to consider this issue, and that's the Alignment of Licensing Processes and Lessons Learned From New Reactor Licensing rulemaking, addressed by SECY-22-0052, and recently approved to go forward in a Commission Staff Requirements Memorandum.

I note that the staff's position in their draft-proposed rule falls short of our recommendation, and we look forward to weighing in on that when the proposed rule is published for comment. I bring it up because that might be -- that rulemaking might provide an area for efficient nth-of-a-kind licensing reviews to be considered and approved, and might warrant attention.

So, in conclusion, the NuScale SDAA provides a highly standardized, optimized licensing basis for future license applicants to follow. We agree the reference in subsequent COL processes are an important aspect of turning the standard design into standardized plants, and in ensuring timely and efficient licensing reviews in the process. Allowances for limited variations in site-specific designs are essential to using standardized designs, and commercial incentives will minimize, and licensing controls will address, the safety of such variations.

Importantly, our SDAA, and the standard design processes more generally, would not allow the degree of design divergence that occurred in earlier standardization efforts under Part 50.

And, finally, we agree that an efficient NRC review process for license applicants will be important for the industry and the agency to meet the objectives of the ADVANCE Act, while supporting our objectives for efficiently building safe nuclear facilities.

Thank you for the opportunity, and happy to answer any questions that -- if you have them.

MEMBER HALNON: Any questions from members in the room?

(No response.)

MEMBER HALNON: Online, any questions?

(No response.)

MEMBER HALNON: Gary, thank you. That was an excellent summary and conclusion. This is -- I'm sorry, James, this is Greg Halnon. Since I was the primary author of the memo, we'll be going through the memo in the next, I believe, full Committee meeting, during our P&P session, our Practices and Procedures session, so we'll get it fine-tuned there.

But I think the learning process that we're going through, both in this one and what we did in Kairos, and probably in the next couple, on what an NOAK, or nth-of-a-kind, looks like, will be a key input to the rulemaking coming up. So, I appreciate that plug there, Gary. We think that's going to be important, and incumbent on us as an agency to define and understand how that's going to impact our resources going forward so we can meet the ADVANCE Act.

One last chance, any questions or comments?

(No response.)

MEMBER HALNON: Okay, Tom, it sounds like -- Walt, if you're okay, we'll proceed on with Chapter 16.

CHAIR KIRCHNER: Yes, go ahead, Greg.

MR. GRIFFITH: All right. This is again Thomas Griffith. Appreciate the opportunity to present Chapter 16. We will have Karl Gross presenting Chapter 16 in lieu of Gene Eckholt. And so at this point I'll turn over to Karl Gross to begin the presentation of Chapter 16.

MR. GROSS: Thank you, Tom. My name is Karl Gross. I'm with NuScale. I joined them back in 2014, originally, primarily working on the technical specifications. I'll be covering the tech specs and the changes that have occurred to those tech specs since the DCA was submitted.

Can I have the next slide, please? And we can go -- do that again. There we go. That's the DOE acknowledgment, of course, supporting our work. Next slide. There we go.

NuScale included a Part 4 Generic Technical Specifications with the SDAA, consistent with the statements of consideration for the 2007 rule change to 10 CFR 52, where the Commission expressed its expectation for the contents of the application for design approvals to contain, essentially, the same technical information required by a DCA.

We recognize that it's difficult to perform a review of a design without having accompanying technical specifications. Subpart E of 10 CFR 52 does not require technical specifications for consideration, but, as I said, we went ahead and included them.

We can go to the next slide, please. These are several -- there were several changes that resulted in changes to the GTS. Obviously, the most noteworthy driver was the increase in thermal power from 160 to 250. We'll cover the rest of these as we go through the slides.

Next slide, please. Development of the SDAA GTS, as we mentioned, started with the DCA model tech specs as a model, and then we incorporated changes resulting from the design and analyses using criterion 50.36, applying those criteria. We also considered industry Travelers, used the writer's guide, the industry writer's guide, for format and content, as appropriate, with, obviously, some little differences there, since that was written for the large PWRs and BWRs. And provided a summary of discussions of this process in our technical report which was submitted to the staff. That tech report highlights the differences between the DCA and the US460 GTS at the time of the SDAA submittal.

Next slide, please. The next couple slides highlight some of the differences and changes that occurred between the two sets of technical specification GTS. One of the important ones on this one I'd like to highlight is the remote shutdown station LCO was removed from the tech specs because of our passive design -- did a more careful review of it, the way we implement it.

And next slide, some more changes. A couple things to note here, one of them -- and most of these will be addressed in other discussions, obviously, the design itself or the analyses. But we reduced the number of reactor vent valves, those are the valves above the -- up high on the reactor vessel from three to two. We also added an LCO, or what we call ECCS Supplemental Boron System, which is important for extended cooling periods to maintain shutdown margin -- maintain a reactor shutdown.

Next slide, please. The NRC review of the Chapter 16 and the proposed GTS resulted in 68 audit items, which have all been resolved. There are no

RAIs outstanding specifically related to the tech specs, however, other changes are driving, or have a potential to drive, additional changes. So, as other RAIs are resolved, we're watching those carefully.

MEMBER SUNSERI: This is Matt Sunseri. I have a question about the review process.

MR. GROSS: Go ahead.

MEMBER SUNSERI: I know you're referring to the NRC review there. You know, when I look at the topical -- or the technical report and how they construct these tech specs, you know, you outlined clearly how you followed those. And I think, you know, in many cases this individual limiting conditions for operation, or whatever, are derived from technical, specific, you know, ways that the system functions. But some of them you have to put more thought into the situation or the circumstances or the operating parameters or, you know, what can or cannot happen at the plant. So it requires some careful thought on, you know, thinking about the scenario and then what would be bounding for including in a technical specification.

So, it looks like you did a really good job on that. But my question is, did you have any kind of independent peer review to validate that you didn't miss anything? Because you took out a lot of stuff, based on your experience. You added a few things, but, you know, it's the devil's in -- I don't want to say that -- but, you know, sometimes it can be hard to not know what you do not know from your internal perspective, so having that peer, independent peer, can open your eyes for things. Did you have a peer review or anything?

MR. GROSS: Not so much independent, but we do work very closely with our operating group and also with our safety analysis group. They spend extensive time, I'll say it this way, picking apart our proposed tech specs. As you know from working with operators, probably, versus licensing, we spend a lot of time working with them to make sure that our intent is clear and what the -- that it aligns with the safety analyses. So, that was carefully considered and we spent untold hours doing that.

It was -- the 50.36 identification process, as described in the DCA tech report, was basically reapplied for the SDAA. So there was also a, I won't call it a clean sheet, but it was close to a clean sheet review during the development of these tech specs.

MEMBER SUNSERI: Okay, that's good. Just

a short follow-up, then. A lot of -- not a lot of times, but it is possible that when you go into operation you get similar scrutiny like this, so should you find, when the plant is in operation, that you have a tech spec that doesn't quite cover you -- I don't want to use the word, inadequate tech spec, like it's commonly used, but, you know, you find something that you might have missed that you need to include, how would that be dealt with?

MR. GROSS: Depending on when it's identified, you know, if we were actually at the operating, where we've got fuel loaded -- in either case, we're going to have to go through NRC approval, obviously, to get a change identified.

I, personally, happen to have been through that. And you're right, there's a -- there will be changes. I hate to say it that way, but that's my experience, anyway, every plant I've ever heard of, and clarifications. Sometimes they can be made at the basis level, which we can do ourselves, under the rules. But if a change to the actual technical specifications, we'll have to get Commission approval beforehand. There'll be a continual feedback loop during that process.

MEMBER SUNSERI: Okay, perfect. Thank

you.

MEMBER KIRCHNER: Karl, this is Walt Kirchner. The third sub-bullet on the slide that's in front of us right now under "noteworthy changes," maybe this is a question for the closed session, but what caught my eye was that the geometric form of the boron pellets, could you elaborate on that?

I am presuming that just is a time, depending on the form of the pellets that would perhaps impact the time that it would take for that to dissolve and go into solution, is that what you are referencing here?

MR. GROSS: That's basically what it amounts to is the dissolution rate that was assumed in the safety analysis.

MEMBER KIRCHNER: Okay.

MR. GROSS: Our safety analysis people maybe can jump in if they feel they want to.

MEMBER KIRCHNER: But it manifested itself in the form of a time specification?

MR. GROSS: Actually I think we modeled it based on an expected and demonstrable dissolution rate.

MEMBER KIRCHNER: Right.

MR. GROSS: But we didn't, you know -- it will be in the COLA is where we are including that requirement, the actual specifics.

MEMBER KIRCHNER: Okay.

MR. GROSS: Right now it's kind of like, you all remember the baskets in the bottom containments and there was issues early on about them turning into a big slug, I guess is a nice way of saying it.

MEMBER KIRCHNER: Right.

MR. GROSS: We are trying to avoid that and we have avoided it by including requirements for the form to ensure that they dissolve in an appropriate timeframe.

MEMBER KIRCHNER: Okay, all right. Thank you.

MR. GROSS: Mm-hmm.

DR. SCHULTZ: Karl, this is Steve Schultz. Your last bullet here, the change that you had to make associated with the generic tech specs with regard to the steam generator program, could you expand on that, what was it that you had to address there?

MR. GROSS: I'll defer some of this to others, but it basically boiled down to the Staff had concerns with the initial inspection to ensure that we had adequate inspections during the early operational days. I don't know --

MR. SNUGGERUD: This is Ross Snuggerud with NuScale.

MR. GROSS: Thank you, Ross.

MR. SNUGGERUD: Yeah. What Karl said is basically correct. There was a desire to make sure that in support of our position on how the steam generators would function and how the steam generator would potentially be impacted by density wave oscillations at low power, the industry's behavior of spacing out the steam generator inspections is more implied than stated, and so in our generic tech specs it will state that we need to do certain fractions of the inspections every refueling outage.

So, even though that's the way the industry does it generically, it's implied in the industry and its industry behavior. For NuScale, they wanted to ensure that we actually have our owners doing partial inspections following the first and subsequent and not just waiting for the full inspection period to do the entire steam generator inspection.

MS. BLUMSACK: This is Erin Blumsack from NuScale. I just wanted to clarify what Ross said. We have a COL item in Chapter 5 indicating that we will do inspections in a staggered basis until full inspection of the tubes is complete.

In the technical specifications we didn't change that, we reduced the amount of time between 100 percent tube inspections and we also bracketed the tube plugging criterion.

DR. SCHULTZ: Thank you.

MEMBER BIER: Quick question from Vicki Bier. Can you talk about whether the staggered steam generator inspections have adverse cost implications for users or do you think the cost is about equivalent?

MS. BLUMSACK: This is Erin Blumsack from NuScale. We haven't evaluated the potential cost, however, the steam generator program indicates that we will complete a degradation assessment and that will inform future inspections.

So we're not expecting that anyone would do anything they --

(Audio interference.)

MEMBER BIER: Thank you.

MEMBER MARTIN: This is Bob Martin. I had a question, and I was planning on using it for the LOCA but I'll bring it in here, are there tech specs -- Someone's got a hot mic.

Anyway, are there tech specs associated with leak rates of the RVVs and RRVs and then the other part of the question that would be a stretch is how do you mitigate excessive leaks, leak rate?

MR. GROSS: This is Karl Gross again. The RVV and RRV leak rates would be the same as those for the RCS as a whole. So, yes, that is addressed in Section 3.4 of the tech specs.

As far as mitigation, as you know probably the containment is operated under vacuum conditions and we have systems to maintain it there and monitor the pressure there and remove leakage --

(Simultaneous speaking.)

MEMBER MARTIN: Okay.

MR. GROSS: Okay. So I guess those --

MEMBER MARTIN: Is that also for say possible air ingress? You have a vacuum, so presumably you have a line to, you know, vacate it.

MR. GROSS: Yeah.

MEMBER MARTIN: Same kind of thing, you monitor for potential air ingress into the containment?

MR. GROSS: Yeah, not specifically as air but we watch the pressure, I believe, unless something --

MEMBER MARTIN: Okay. That would be the same thing.

MR. GROSS: Yeah.

MEMBER MARTIN: Yeah.

MR. GROSS: Next slide, I guess, if we're ready. One change that has occurred as we determined it appropriate to include an LCO for the passive auto re-combiner, that it will be discussed further in the Chapter 6 presentation.

I know that's coming up in the future as to why it was included. I want to point that out there is a new LCO for that.

Next slide. I think that's it. Thank you, unless anybody has any more questions.

MEMBER SUNSERI: This is Matt Sunseri. I've got maybe one or two more. Excuse me. I noticed that the introduction to the technical report stated that the tech specs are for a single module and I understand that.

So I presume that each module would then have its own technical specifications, would that be accurate?

MR. GROSS: That is correct. The intent is to keep them aligned as much as possible. I recognize I think over the life of a facility there may be some small variations as changes are incorporated into the design.

If they are, that's obviously not our intent, but, yeah, there would be a complete set for each module.

MEMBER SUNSERI: Yeah. So -- yeah, well that's where my question, and I wouldn't call it, maybe concern is too strong of a word, but at least my experience with multiple reactor sites, I'll call it that, that, you know, in some cases you have a common control room but they are segregated enough such that, you know, you have Unit 1 operators, you have Unit 2 operators, you have Unit 1 tech specs, you have Unit 2 tech specs, whatever, and they are pretty separate and the operating crew can, you know, pretty much guarantee they know which unit they're on and which set of tech specs they're in, but the way your control room is going to be operated you're going to a few operators covering as many as six modules with six different tech, potentially six slightly different tech specs. That seems like a pretty challenging human performance challenge.

What are your thoughts on how you're going to manage that and is there anything that needs to be put back into the tech specs, into a single unit tech spec. Single module for you guys.

MR. GROSS: Right now we're obviously we're not aware of anything that would be required to distinguish between them. However, recognizing that there's a potential for that in the future, we're going to leverage heavily I think our automated systems.

As you know the control room is very, can be automated, almost all the functions. It's kind of spooky if you're used to old plants. But, yeah, so hopefully any changes we can address that way or incorporate appropriate controls to ensure alignment that way to support our operators.

Ross, would you like to jump in on any of this?

MR. SNUGGERUD: I mean I think the most likely condition and the one we have talked to our potential operators about is the fact that we are going to be implementing some change on a module and that's going to require us to cycle through the tech specs, which is one of the advantages about having individual tech specs.

If we need to make a modification and then we're implementing that modification systematically throughout the modules over a course of refueling outages we can keep the tech specs up to date for the individual modules that have had that configuration change.

We don't really expect a lot of module-specific technical specification changes, so, you know, at most I think we would have two versions of the tech specs at any given time.

But, you know, as Karl suggested, we are going to be providing the operators with lots of support through the interface and there are lots of different ways to address that.

Certainly it's not -- you know, your concern is valid if we started to have deviations between the tech specs, but the idea of keeping the tech specs individual to the modules was intentional and largely to support operations at the plant.

MEMBER SUNSERI: Yeah, that's good and I appreciate that. You know, having been in these kind of control rooms, or situations before, you might just give some consideration to how it would flag or highlight, you know, any potential differences so that you guys -- human performance, right, you know how that goes.

MR. SNUGGERUD: Yeah, absolutely, we agree.

MEMBER SUNSERI: Yeah. Hey, and I have one more technical question that I'm not sure it's going to come up in an individual chapter review so I'm going to ask it now.

It deals with the definition of MODE 2 and MODE 3, hot shutdown and safe shutdown and there is a various combination of passive cooling or not passive cooling, minimal temperature for criticality, whatever.

I understand all that, but I just want to ask one question. Is there a reactivity -- are both those conditions limited to less than 0.99 or, yeah, 0.99 k-effective?

MR. GROSS: That is correct.

MEMBER SUNSERI: Okay. It was -- I thought it was when I read the GTS, but, you know, some of the documentation I read it wasn't clear that that condition was always -- so it is truly less than 0.99, okay. I'm good. Thank you.

MR. GROSS: Yeah. Yeah, it's specified that way in the definition of MODES, that table at the end of Section 1.1 of the tech specs.

MEMBER SUNSERI: Yeah, I saw that but then I also know that those aren't the real tech specs, right, that's just a generic one that you submit for reference?

MR. GROSS: Yes. But the intent is that that's where it will be.

MEMBER SUNSERI: Okay, all right.

MR. GROSS: The MODE tables are pretty well defined.

MEMBER SUNSERI: Okay. That's all I had. Thank you.

MEMBER KIRCHNER: Members, any further questions of NuScale before we turn to the staff?

(No response.)

MEMBER KIRCHNER: Mike, correct me if I've got the agenda wrong. Are we going to complete 16?

MR. SNODDERLY: No, you're right. You're right, Walt. We found that --

MEMBER KIRCHNER: Okay.

MR. SNODDERLY: Unless the Committee gives us other feedback, but we find it's best to complete a chapter and then the staff going --

(Simultaneous speaking.)

MEMBER KIRCHNER: Right. Right. So if there is no further questions I think we're ready to move to the Staff's presentation.

MEMBER HALNON: Okay. Give us a few minutes to change the seats and computers and I'll let you know when we're ready, Walt.

MEMBER KIRCHNER: Okay, mm-hmm.

(Pause.)

MEMBER HALNON: Okay, Walt, we're going to get started again.

MS. SCHILLER: My name is Alina Schiller.

MEMBER HALNON: Please turn your microphone on. Speak right into it.

MS. SCHILLER: Good morning.

MEMBER HALNON: That's not close enough.

MS. SCHILLER: Good morning.

MEMBER HALNON: That's better. Thank you.

MS. SCHILLER: My name is Alina Schiller. I am a project manager with the Office of Nuclear Reactor Regulation, Division of New and Renewed Licensees, New Reactor Licensing Branch.

I would like to thank the ACRS Subcommittee, NuScale Power, LLC, and the general public for entertaining the NRC for the presentation of the Staff's safety evaluation of NuScale Standard Design Approval Application from Chapter 16 and the Part 2, Revision 1.

NuScale 70 Part 2, the Final Safety Analysis Report, Chapter 16, Technical Specifications and Part 4, Generic Technical Specifications, Revision

0, in December 2022 and Revision 1 in October 2023. From March 2023 to August 2024 the NRC conducted a regulatory audit of FSAR Chapter 16 and Part 4 which generated 68 audit issues. All audit issues were resolved in the audit.

Fifty-two audit issues resulted in NuScale submitting supplemental information to address questions raised during the audit. No requests for additional information were issued in Chapter 16.

We are here today to discuss the Staff's advanced safety evaluation of Chapter 16 and Part 4. The contributors were Craig Harbuck, the lead technical reviewer and today's presenter, supported by Steve Smith, Clint Ashley, Josh Wilson, all with the Technical Specifications Branch, and the project manager for Chapter 16 and Part 4 supported by Getachew Tesfaye, the lead PM for NuScale SDAA.

This slide lists the sections in the FSAR Chapter 16 and Part 4. I am turning it over to the NRC subject matter expert, Craig Harbuck.

MR. HARBUCK: Good morning.

MEMBER HALNON: Craig, you're going to have to move that real close, as close as you can move it to yourself comfortably.

MR. HARBUCK: How's that?

MEMBER HALNON: That's good if you keep it that way.

MR. HARBUCK: Okay. So I am going to go through changes that we thought were worth mentioning outlining the differences between the DCA certified generic tech specs, the US600 design, and then reviewed the SDA.

This slide is to point out one of the key differences in the definition of MODE 3. It was adopted in the SDA. Essentially what has happened is that the MODE 3 in the DCA began at the minimum temperature for criticality, but the MODES 1 and 2 covered temperatures above that.

There is one thing I would like to point out that might not be obvious to everyone about the MODE 1 definition. It actually corresponds to MODES 1 and 2 that you see in normal PWR plant for the operating fleet.

So as was mentioned earlier in NuScale's presentation, you can enter MODE 3 from 1 or 2 by initiating passive cooling without first cooling down the module to below the, well the minimum temperature for criticality.

That particular temperature which that is was lowered in the SDA from the DCA value of 420

Fahrenheit to 3.5 and the way you initiate passive cooling when you are at the normal operating temperatures would be to either open up the ECCS valves or by initiating the decay heat removal system.

Both of these would result in a significant transient and a fairly quick cool down and so if you're in an action statement and it says shut the plant down to MODE 3 and a lot of times it will indicate, it may give a caveat on a lower temperature range than MODE 3, but typically you wouldn't implement that action statement by initiating passive cooling.

The preferred method would just be to use secondary heat sink systems to do that.

MEMBER HALNON: So, Craig, this is Greg. What was the purpose then, I mean if the normal shutdown is going to be 48 -- or whatever, a k-effective less than 0.95 and you cool it down to less than 345 is that in there for an operational transient situation or reactor trip of like a loss of power or something?

MR. HARBUCK: I can speculate about what the operational conditions might be where this would be appropriate, but --

(Simultaneous speaking.)

MR. HARBUCK: There must be some operational advantage to being able to say that you're in safe shutdown until you run actually your MODE 2 and not have to move that MODE change as long as you're being passively cooled, but I would have to refer to NuScale to address that question.

MEMBER HALNON: Okay, that's fine. It's not a game changer, it was just more of curiosity. Go ahead.

MR. HARBUCK: Okay. But it is the cycle that the nature of the design and how to operate it because you do have to think about these things when you are deciding what's the appropriate way to account these information constraints from the definitions.

I also want to point out that the MODE 3 definition before you get to -- well there is a footnote on some instrumentation functions which tell you that certain reactor trip and ECCS functions or PSF functions don't have to be operable if you have just one control rod mechanism being energized in any fast term is being capable of withdrawal and typically that particular footnote is stated as any (audio interference) fairly unique NuScale design features related to coupling and uncoupling control rods.

They changed the footnote so that you

could have one rod that you were trying to couple or uncouple would not by energizing the control rod drive to manipulate the rod to do that would necessitate that then goes, is corresponding to instrumentation functions probabilities and that's one difference from the rule, that normal caveat in the footnote in operating.

Then going to the next slide, there was one that was adopted recently, recently approved, relatively recently, that clarified the definition of pressure boundary leakage and NuScale elected to adopt those changes.

So this mark up here shows the change from what we had in the DCA and what we have now and this is consistent with the industry's understanding of how, and the Staff's understanding of what, particularly that last added sentence.

It's something that actually in the bases for the LCO related to leakage that is simply up and made part of that definition.

Next slide. And going to the safety amendments chapter, this shows you that the comparison of the different correlations from the DCA to the SDA and the NSP4 correlation was maintained but the other ones were not and then for certain operational aspects, which NuScale could elaborate on, there was an additional correlation that was added on.

The other reactor safety limit on peak center line temperature, the fuel did not change, that's quoted there as saying that, and then because of the higher operating pressure in the higher powered design we moved from 1850 to 2000 psi.

So the safety limit under, on the reactor coolant system pressure has increased as noted there on the slide.

DR. BLEY: This is Dennis Bley. Could you back up a slide to Number 6? Thank you. No. There we are.

MR. HARBUCK: Yes.

DR. BLEY: Leakage past seals piping gaskets is defined here as not a pressure boundary leak and I don't know that you have specified equipment to the level that would allow us to know how big such leaks could be.

I've seen some pretty big ones, like if seals blow out or something like that. Why do you express it this way rather than in terms of maybe a pounds per hour or something, you know, a quantitative definition of what you mean by pressure boundary leak.

MR. HARBUCK: The way your point is

addressed is in the LCO for a leakage for the reactor coolant system --

DR. BLEY: And --

MR. HARBUCK: And so there is no pressure boundary leakage, that's not allowed. If you identify that then you would be on a shutdown track.

That brings up another point, is that currently NuScale in their containment evacuation system it's just what maintains the vacuum in the containment of their operation.

They don't currently have a way of determining any leakage that is collected about what the source of that leakage is in terms of is it pressure boundary leakage or is leakage past some, like this phrase says, or is it, you know, is it coming from a secondary system, is it coming from a leaky external system like that containment flood and drain system, or is it, let's see, is it coming from the CVCS, or is it coming from like a feedwater line or something, which is not part of the pressure boundary within the CF.

So they essentially will treat any leakage that's detected if I understand correctly, and correct me if I'm wrong, but I think it would essentially treat any leakage they collected in their containment evacuation system as unidentified leakage, and there is a limitation on that in the LCO.

MEMBER HARRINGTON: Dennis, this is Craig Harrington. That added sentence is consistent with the ASME code treatment of pressure boundary that packing seals, gaskets, that's not pressure boundary.

DR. BLEY: Okay.

MEMBER HARRINGTON: It is, but --

DR. BLEY: They are being consistent. I understand, but they are being consistent with the standard. Okay. You know, it just felt funny to me. Thanks, Craig.

MR. HARBUCK: Okay, now we can go to the next slide. I think we've covered this, yeah, so let's go to Chapter 3 now. The remaining slides will roughly focus on Chapter 3.

So earlier it was alluded to that some specifications were, some LCOs were removed that would have been an additional DCA to your tech specs and then the renewal LCO had it in the SCA and this provides a list of those.

The bullets under the two sections that were removed continue to provide some, point to some of the rationale for why those removed and why it is acceptable to do that and I'll just briefly mention those.

A remote shutdown station in a typical operating plant would have the ability, would have controls as well as indications that from which personnel who had evacuated the control room be able to shut down the plant and monitor its condition in a safe shutdown situation.

In the DCA, the imagined remote shutdown station was only going to have indication and no controls per se because of the envisioned scenario was that if there was a need to evacuate the control room, part of doing that would be to shut down all the reactors, and because of the design, where there really is no operator action needed to ensure you are in safe shutdown after that, then there did not seem to be a need for any duplicate controls in a separate station outside the control room.

The indication monitor, it was pointed out during the review that the I&C equipment rooms associated with each module will have, also have indications that are explained in the control room and you can find those in the same information and, therefore, that was seen as being sufficient to monitor the status of the modules in the event of a control room evacuation.

There was also -- and typically the controls you need to make sure that you achieve safe shutdown and operating plants would require, you would have in the safety analysis you might have some assumptions about actions that operators would take and for those you have what's called Type A post-accident monitoring variables that would be seen as needed to provide information the operators would need in terms of to properly conduct that shutdown, but there are no such, there are no Type A variables in the NuScale design, which is another reason they don't have a main LCO also.

The in-containment secondary piping leakage in the DCA we address this idea to monitor the leakage and you can use the leak-before-break method to provide yourself assurance that you would be able to recognize when you had the potential for a high energy line break into avoiding any resulting pipe movement that could damage any other equipment in containment and you'd be able to shut down and address that because there would be enough time to do that.

In the SDA they have determined that based on the Staff guidance note on the BTP 3-4 and industry's guidance, the standards, that address certain pipes, size of pipes, and that sort of thing, that they have this exclusion criteria and it was determined that the high energy lines in containment satisfy those criteria, so that LCO was deemed not being necessary and the Staff has agreed to that.

So that's what we have removed or omitted, but what's been added was, as they mentioned before, the ECCS Supplemental Boron System and that's for assuring that you have adequate shutdown margin for long-term cooling.

That system is also implemented to make sure that it's implemented after a reactor trip if you are in a situation where a combination of xenon transients and cool down could get you starting to approach your re-criticality situation with water flowing from the containment.

The idea was to make sure that the system actuated and that's a passive design, it's hands off for 72 hours. So after eight hours, unless the operator has determined it's not necessary and they can block the system from actuating, there is an 8-hour post-reactor trip actuation time on ECCS primarily to be able to initiate the dissolving of ESB pellets.

That timer is addressed, the verification is addressed in the actuation logic LCO in SR-3333.

The other, another LCO that was added that could very well have been included in the DCA but there was, or I guess NuScale recognized that during MODE 4 when you are moving the module, the passage of decay heat during that evolution is, requires there to be a sufficient volume of water in the containment and so to preclude also any inventory in the containment met within diminished heat transfer capability. They wanted to make sure that any isolation valves from the containment to the outside would be closed.

So that is what this LCO is designed to do. It doesn't have to meet the same leakage requirements that you have for like containment isolation valves in that LCO or to make containment operable.

It's just designed to maintain the critical inventory to ensure you have adequate cooling and to what the module on the disassembly stand over the refueling area and you've unseated the containment off of its lower portion and, therefore, then everything is filled up to whatever the pool level is.

Okay. Then it was also mentioned that the passive auto catalytic be combined and it was determined to satisfy Criterion 3. It was determined to be a safety-related system and it was also somehow justified as meeting this criterion in 50.44.

If anyone has a question about that I would ask NuScale to respond as I am not that familiar with what those very non-specific words in 50.44 correctly say. Any questions about that? Okay.

So next slide. So there were -- to me it would be I guess one more significant change was an instrumentation in terms of the instrumentation fluxes that actuate the ECCS and also decay heat removal system, which will be on the next slide.

But they changed the way they're measuring level in the riser in the SDA over what they had in the DCA and, therefore, they have ECCS initiation primarily occurring on the riser level.

A low level which is designed to protect and trying to prevent uncovery of the top of the riser is blocked and if you go below 500 degrees it's interlocked and it does that.

The low riser level is designed to actuate ECCS before you uncover these holes that are in the side of the riser down lower in the vicinity of the steam generator in the downcomer to make sure that there is adequate flow of the high, relatively high concentration reactor coolant in the riser into the downcomer region to alleviate any dilution effects caused by evaporation that any of them had which ECCS has now furnished. As was also mentioned --

(Simultaneous speaking.)

MEMBER MARTIN: Question real quick. This is Bob.

MR. HARBUCK: Okay.

MEMBER MARTIN: And maybe this is a NuScale question, but the level instrumentation is that really a DP or is it --

MR. HARBUCK: That is done with a specialized kind of discreet -- There are probes that detect --

MEMBER MARTIN: Okay.

MR. HARBUCK: And those functions actually have like a 60-second delay on when you actually get a change in the signal.

MEMBER MARTIN: Okay.

MR. HARBUCK: So I think that's built into the substance of the safety analysis. I don't know if there is any software or other I&C magic going on that would interpolate between those discrete levels because they're not that far apart.

MEMBER MARTIN: Okay. That's just something I wanted to clarify.

MR. HARBUCK: And that's another question which will have, may warrant further investigation for a COL applicant, but --

MEMBER MARTIN: Okay. All right, that answers my question. Thank you.

MR. HARBUCK: Yes. Okay. Again, they reduced the number of reactor vent valves from three to two and I don't know if that was a change in the size of the valve or if the third valve had simply been there for some other redundancy reason, but to accomplish the ECCS function you need only one reactor vent valve and one recirculation valve.

So from that standpoint they are redundant and then each valve itself is also doubly redundant on, that's redundant itself. There are two actuation solenoids on each valve that are separate to the channels of the module protection system that have to be de-energized because you reached the ECCS actuation setting level for those valves to open.

Now there is one thing I want to -- this was the subject that we were hitting on and I just wanted to mention it.

For most events your reactor vent valves continue to remain shut so that you don't introduce the containment into the initial response to the event and handle it like the decay heat removal system, a reactor trip, what have you, then you may not need to change anything with the containment.

And so to maintain those valves shut during those kinds of transients and also during normal operation, the EDAS to which is, that implies special available or quality controls, you know, augmented quality is what they call it.

So that -- because before the DC those valves would not open until the pressure difference between the containment and the RCS reached a certain level, a certain level, thereby that had the effect of -- even if you would -- if the actuation set point until you reach that IAB, which was the more mechanical kind of inhibiting of the opening of the valves, the valves wouldn't open.

But that's been removed in the SDA with the DES liability has been, combining the system and all of that has been in these areas in which it needs to be addressed or there was any kind of concern that was resolved here.

But the reactor recirculation valve still has an IAB and it's helping to still be delayed even if you get to set point. So that's all I wanted to say about that. Next slide.

MEMBER ROBERTS: Hey, Craig, it's Tom

Roberts. Can you clarify, you said there were two solenoids in each of the vent valves and recirculation valves?

MR. HARBUCK: That's correct.

MEMBER ROBERTS: Is there a tech spec that they both have to be in service?

MR. HARBUCK: Yes.

MEMBER ROBERTS: Okay. So the intent is that either one will trip a valve or either one will heat it up?

MR. HARBUCK: You would need both of them to actuate to cause the valve to open.

MEMBER ROBERTS: Okay. So from a tech spec perspective let's say the safety position of the valve would be open and so why would there be a tech spec that says those solenoids have to be in service?

MR. HARBUCK: Well if the solenoid -- if they are not in service they would de-energize and the valves are going to open. This is a fail/safe design on all valves to go to their safety position, so this result was not, did not be -- I suppose you could operate with one of them out of service.

As long as one of them’s keeping it closed, but it's not seen as being a safety-related function. It's called the ECCS holds function. So the -- we could put NuScale on to address this. They would probably do a better job.

MEMBER ROBERTS: I guess we'll renew the question later. It just seemed a bit odd. Certainly it's prudent, right, definitely it's prudent if you have the redundancy to have it because you don't want to add inadvertent actuation, but I'm not sure why it's a tech spec because the -- unless there is some concern that inadvertent actuation is a safety concern.

MR. HARBUCK: Well the tech spec is focused on not having both of them for the purposes of keeping the valve shut. It's there to make sure that they really de-energize than actuate it.

So for tech spec purposes the valves would be inoperable if for some reason you removed power and they didn't change position and, therefore, the valve could not, it would not open.

Those basically have a hydraulic lock on the valve keeping it shut and then you de-energize the solenoids the valves open and that removes the lock, so it is kind of complicated in terms of the logic, thinking about it. Anything more to add?

(No response.)

MR. HARBUCK: Okay. Next slide, decay

heat removal system. There is not much to say about this other than that we added a few more instrumentation functions which would cause an actuation of the decay heat removal system.

At the bottom there I list the three, those three functions and the events that those functions are designed to mitigate and they all boil down to putting yourself in a safe shutdown situation.

That's when you set your MODE to 1 and you would initiate a decay heat removal system that's going to put you in MODE 3, and so that -- any other questions about that?

(No response.)

MR. HARBUCK: I believe that's the last -- oh, yeah, there's one more slide. So we'll go to the next slide. Oh, there's two more slides. Maybe this is it. Yes.

Okay. Another change that happened, another change that I want to mention was the boron dilution control, LCO-319. There is a system in NuScale that allows you to heat up the RCS called module heat up system and the way the system works is you have the heat exchanger that on one side is supplied with steam from the non-safety source that generated that, then you take the discharge out of the

CVCS and the discharge, I mean the injection line, and the diverted flow by the heat exchanger and then it comes back downstream of where it left back to the injection line warmer and then the injection line terminates in the riser section of the reactor vessel.

So in that way you can heat up the unit since there are no reactor coolant pumps to do that sort of thing or to heat up the coolant, but they don't have a module heat up system heat exchanger for each module.

There is a common one, so it was recognized there was a potential that if you had errors in alignment you could connect one CVCS system from one module onto another module.

So this was added to the LCO and clarified in the surveillance requirements to check the alignment to make sure, you know, you never had more than one module aligned with the module heat up system.

That's usually for a relatively brief time when you're starting up and you want to heat up the system. So it's not a likely thing to occur, but it was to see if that was a potential error that could be addressed where safety could made operationally that could be addressed by highlighting it in the LCO for boron dilution control.

In the -- I think I mentioned earlier about the MODE 3 applicability of selecting a reactor trip in a de-mineralized water system isolation instrument functions and that the footnote for those functions for MODE 3 is incapable of withdrawal of more than one control rod assembly.

So that was a difference from your regular, the usual definition, and also a difference from what they had in the DCA, because I think this particular concern was not identified in the DCA or it was not being needed to -- it just didn't come up.

The last thing, it was alluded to earlier there were some changes from what was in the standard tech spec and what was in the DCA relative to the steam generator requirement and some items here that were changed was the period between of having to inspect all of the tubes after the initial -- I think at the end of the first refueling outage there is, or at the first refueling there was supposed to have been another full inspection of all the tubes.

But subsequent to that you have 72 effective full power months in which I remember they were going to do staggered in the sense of the valves on the tubes until, so by the end of 72 effective full power months you would have all your tubes expected to be in.

So that's -- and previously had been like 96 months, so it was reduced. We have folks in the room that maybe could address why that is. The other parts of the description of the inspection discussion in this program specification talked about the fact that RCS pressures on the outside of the tubes and the tubes are susceptible primarily to collapse from collapse or buckling rather than bursting.

So that does allow for some other differences that, you know, I think it's Chapter 4, five, Chapter 5, where they go into more detail about this.

Then this -- as we did in the DCA we put a value of 40 percent for the criteria recognizing that a COL applicant might have a different -- might have had new information or new rationale and might have a different number, so that's what the use of the bracketed information is in the tech specs and in the bases also. That falls under COL item 60.1-4.

I believe that's the last information slide and this is a conclusion listing the regulations that govern the tech specs that we have determined that we are in compliance with those.

MEMBER MARTIN: This is Bob Martin. Now did you have RAIs --

MR. HARBUCK: We have no RAIs as far as the process.

MEMBER MARTIN: Yeah.

MR. HARBUCK: I think we've managed to resolve everything in the context of audit follow-up.

MEMBER MARTIN: Okay, all right. A pretty straightforward review?

MR. HARBUCK: Yes, it was. A lot of our issues were related to problems in other chapters, but we always hear about it last. That concludes --

MEMBER HALNON: We have a NuScale person with their hand up. Did you have a clarification, Tyler?

MR. BECK: Hi. This is Tyler Beck with NuScale. I just wanted to clarify about the ECCS trip valve and the solenoids. So for the ECCS valves to be operable they need to be closed and capable of opening and that is the operability requirement straight from the tech spec bases.

So, you know, technically I guess you could say that one trip valve was not energized but one was and in that case the ECCS valve would still be operable.

MEMBER ROBERTS: Okay. Thank you. Yeah, in the meantime I was looking through the tech spec document and that's consistent with the document that operable is not really defined in that level of detail, and so that does make sense.

MEMBER HALNON: Thank you, Tyler. Walt, we are in your time.

MR. BOWMAN: I have one more thing to say.

MEMBER KIRCHNER: Dennis has his hand up. Dennis, go ahead.

MEMBER HALNON: I don't think that's Dennis. It's another one of the NuScale folks.

DR. BLEY: No, it's not me.

MEMBER HALNON: Doug from NuScale, do you want --

MR. BOWMAN: This is Doug Bowman.

MEMBER HALNON: Go ahead, Doug.

MR. BOWMAN: Can you hear me?

MEMBER HALNON: Yes. Do you have a clarification?

MR. BOWMAN: Yes. This is Doug Bowman. I am the plant manager for Service, thank you, for Services Operation, wow -- Services Manager for Plant Operations at NuScale.

We did want to make one clarification for the record. Monitoring during a control room evacuation event occurs at the alternate operator work stations and those are located at either the module maintenance center or in the rad waste control room. Thank you.

MEMBER HALNON: Thank you, Doug. Okay, Walt, now it's to you.

MEMBER KIRCHNER: Matt, you are our lead on this, have you any further questions of the Staff?

MEMBER SUNSERI: Thank you, Walt. I don't have any. I think the Committee has asked all the appropriate questions. Thanks.

MEMBER KIRCHNER: Other members?

(No response.)

MEMBER KIRCHNER: Well, then at this point we have come to a logical break point in our schedule, and so let's take a break until 10:15. That will allow those of us on Mountain and Pacific Time to get some coffee and refuel. And we'll reconvene at 10:15 and we'll take up the LOCA TR. Thank you.

(Whereupon, the above-entitled matter went off the record at 9:59 a.m. and resumed at 10:14 a.m.)

CHAIR KIRCHNER: Okay. The meeting will come back to order, and we are going to turn to the topic of the Loss of Coolant Accident Topical Report and turn to NuScale.

Sarah, are you ready?

MS. TURMERO: Yes. Good morning.

CHAIR KIRCHNER: Good morning. Go ahead.

MS. TURMERO: All right. Thank you. My name is Sarah Turmero. I'm a licensing engineer for NuScale covering topics on Chapter 4, 9, 15, and the related topical reports. I've been with NuScale for about two-and-a-half years and have a background in PWR reactor engineering.

And with me, I have Meghan McCloskey and Ben Bristol from the System Thermal Hydraulics Group to assist with any questions if needed.

We'll be covering a summary of the significant changes since the approval of the Revision 2 LOCA Topical Report. These changes include those related to the scope, relevant design changes from NPM-160 to NPM-120, and changes to the phenomena identification and ranking table evaluation model structure assessment basis updates and adequacy assessment updates.

So the LOCA analysis method for a pipe break inside containment and the associated event classification figure of merits and key regulations were maintained from the approved topical report. The scope that was modified from the approved topical report includes the incorporation of the analysis method for the inadvertent opening of a reactor valve scenario. And with that the event classification figure of merit and associated key regulation was also incorporated into the scope. Some key updates associated with the IORV analysis is the implementation of the new NSPN1 critical heat flux correlation and modeling cross-flow between the hot and average channel, and of course the scope that is associated with the valve opening and inadvertent ECCS actuation.

For NuScale from the approved topical report the containment vessel pressure and temperature response analysis methodology was incorporated into this topical report. It was previously a separate technical report and the associated figures of merit and key regulations were also included as it relates to the containment response methodology. Additionally, the response to the LOCA pipe break, secondary line breaks, and valve opening events -- those are added scope specifically related to crediting DHRS.

Scope that is outside of the LOCA Topical Report related to long-term cooling and subcriticality are covered in the Extended Passive Cooling and Reactivity Control Topical Report.

And the figure on the left shows how we've defined phases of the event progression for the LOCA and IORV events with Phase 0 being where NSPN1 is implemented, and it's the first 10 seconds of the transient. And then it occurs in conjunction with Phase 1.

Since the DCA submittal, NuScale made incremental improvements to the design and analysis methods, resulting in margin improvement from overly conservative methods or assumptions while maintaining the same level of safety. So we have the power uprate with no significant changes to the module, system, structures, and components. The operating conditions listed are nominal conditions and changes are a result of the power uprate. And then for the containment vessel the design pressure and temperature increased and the upper material was changed.

Next slide?

MEMBER MARTIN: Just real quick, for the open session we're oftentimes quiet because we get to jump on you during closed, but given that we just talked about Chapter 16, the tech specs, could you briefly give an overview how you integrated the tech spec into safety analysis? I know in the report itself obviously you have. You may kind of highlight maybe the major kind of the usual suspects as far as initial conditions. I'm not sure there was a lot of mention of how it relates back to tech specs, but of course I know that they do. But I wanted to give you the opportunity to just kind of talk about how you incorporated the uncertainties which ultimately get integrated into tech specs and into your initial conditions, and that gets vetted to your safety analysis.

MS. TURMERO: I was going to ask a clarifying. Is there a specific tech spec or in general?

MEMBER MARTIN: Well, maybe in closed session.

MS. TURMERO: Okay.

MEMBER MARTIN: But just again this is a public meeting. We just had a discussion on tech specs and I thought it might be appropriate just to segue from one to the other with this point, otherwise -- I didn't want to save all my questions for closed session.

MS. TURMERO: I think I'll start and then we'll ask Karl to jump in from the tech spec side of things.

But from the safety analysis perspective, our initial conditions and the ranges of initial conditions and the SSCs that we credit in the safety analysis are largely consistent between the DCA and the SDA, and we focused -- and the design changes have changed operating conditions for design limits to accommodate the increased power, but that structure is largely the same. And then the one -- the new piece of the system is related to the supplemental boron in the ECCS that was incorporated into the tech specs.

Ben or Karl, do you want to add to that?

MR. BRISTOL: Sure, this is Ben Bristol. So I think generally the flow goes the other direction in our view, so safety analysis works pretty closely with the design team on understanding the constraints around the actual module itself as well as the system team with the constraints around power production targets things of that nature. And then we integrate that with the I&C team, right, that helps set up and establish what types of measurements we have, what protections then we can derive from that. And all of that package then goes into what ends up in tech specs. Because it's a PWR, it largely looks a lot like PWR tech specs.

But we work closely with the tech spec group on defining where those operational boundaries are that accommodates the safety margins that we demonstrate along with the consideration of the required operational margins necessary to support the plant systems and power production requirements.

MEMBER MARTIN: And there are a couple other facets. Tech specs, some people might say, oh, well, LOCA drives tech specs. That's not completely true. LOCA is a limiting for all things, right? So you get a mix of LOCA and non-LOCA informing tech specs. When it comes to initializing for safety analysis, you also -- contemporary approach is following PIRT. You'll talk about that of course. And not everything would justify biasing with all the uncertainties and best estimates.

So part of the answer I guess I was kind of probing was something to say that, well, you know, we look at the PIRT and we kind of look at what's important with regard to the influences coming in from the initial conditions and we select -- you don't have to bias everything because then it kind gets to become a administrative nightmare if you try to be too cute about it. But I was looking for a if-we-do-everything-kind of answer, but at least an acknowledgement that there are priorities when it comes to initializing the problem and it ties back to obviously the physics of what to -- the problem that you're solving and their influences throughout the event. But that's fine.

MR. BRISTOL: Yes, and to build on that, Sarah mentions the three different avenues of analysis that's covered by the LOCA TR being the pipe break scenarios and their figures of merit, which are different than the containment analysis and its figures of merit or the IORV in the short-term transient core response figures of merit. And the bias is -- the conservative bias directions are not consistent between those three different (audio interference).

MEMBER MARTIN: Of course. You have to reconcile that sometimes, yes. Thanks.

MS. TURMERO: Next slide? All right. So there were ECCS actuation signal modifications. So with decreased RCS inventory, ECCS actuates on riser level early in the transient progressions. The Tcold interlock prevents ECCS actuation for extended DHRS cooldown events. And the RCS level indication of decreasing RCS inventory can generate ECCS signal before significant containment level increase.

To highlight some of the ECCS valve design changes, the inadvertent actuation block was removed from the vent valves and flow venturis were added to the reactor vent and recirculation valves and the third reactor vent valve was removed.

For long-term cooling, which is covered with the scope of the Extended Passive Cooling Topical Report, one of the relevant design changes is the lower pool level. And even with the lower pool level, the containment surface area below the pool level provides ample core cooling and maintains ECCS cooling.

DR. SCHULTZ: Sarah, this is Steve Schultz. Could you just provide a general overview? A lot of changes that you've described here, they just came about as good ideas or did they come about as a result of analysis evaluations that pertain to the uprate?

MS. TURMERO: I can't speak to the specific changes, but the improvements in the design and analysis methods were done to gain margin improvement, improve our analysis methods so that we could spread margin across the design, operation, and analysis.

But if you would like to -- Meghan?

DR. SCHULTZ: Could you speak also to the design changes themselves?

MEMBER MARTIN: I had a very similar question, Steve.

PRA. Some changes are going to be strictly to accommodate the power uprate and other changes, given that of course you had a mature PRA at DCA, that you might be able to go here I have an opportunity to take advantage of this new insight. And that could be another set of changes of a different sort. So I think that was the perspective that I was coming from was a very similar idea that Steve had.

MR. BRISTOL: This is Ben Bristol again. So we have a list. Maybe I'll just start with the top. The removal of the IAB from the vent valves. This was a key safety improvement that we found. Now there are trade-offs. Depressurizing the vessel from high pressure is something that we add -- that we weigh very heavily. However, what we found -- Bob mentioned the PRA insights -- the ability to actuate ECCS on demand was a feature that we had precluded to some degree with the IAB in the DCA design.

So it was a big improvement for accident scenarios to have some assuredness on the timing of

ECCS actuation. There were event sequences where we had a broad range of uncertainty based on when the IAB would release due to its pressure-based lockout-blockout-type physics.

Generally for safety, right, we want lots of water over the core, so we recognize that keeping the IABs on the recirc valves was a good feature. However, also part of safety is being able to depressurize the reactor on demand. So removal of the IABs from the vent valve allows us to depressurize the reactor on demand, which allows us to reduce the uncertainty of the timing of ECCS for a broad range of events. And it really improves the response for certain event sequences that we found to be very beneficial. So that's just one. I could keep going.

Maybe another one of interest is the pool level change that gets a lot of kind of consideration, right? Seems like higher pool level would be better for safety. As it turns out, the containment surface area is ample. And we had plenty of heat removal capability in the containment with margin to reduce the pool level.

What we recognized in the design is that -- you'll notice on the module here, we've got these -- the big covers are access ports. And so one of the challenges we had with the other design was the ability to get in and service the equipment on top of the reactor head. So dropping the pool level allowed us to add those access ports that really improved the maintainability of the design and some of the required sequences.

And back to a different consideration of safety, right, ALARA considerations where operators are getting dose, getting them in and out of the vessel effectively, efficiently was something that we recognized is also a consideration of safety. And so that was one of the trade-offs there, where we took some of the margin that existed in the design from a safety perspective and added it to a different element of consideration of the design.

MEMBER HALNON: So, Ben, this is Greg. How did you deal with the uncertainty at the tail end of an accident when you're needing that extra volume and (audio interference)?

MR. BRISTOL: Yes, that's a good question and gets actually to one of the other bullets there is the venting capacity. Very late in the design, we depressurized the whole vessel down to sub-atmospheric conditions. The containment effectively is a big condenser, and so it can really draw the pressure down. The physics of what happens there is it actually adds stress to the venting capacity of the vent valves. So the long-term core level response is directly related to the flow or the pressure drop across the vent valves.

So having a reduced pool level actually allowed us some more margin in the sizing of the venting needs, and that's what allowed the removal of one of those valves, which has knock-on effects of improving maintenance in space on top of the reactor vessel with two valves instead of three and simplifies the design, the number of components. It also reduces the effect of the depressurization transient we talked about if we were to have an inadvertent ECCS actuation.

MEMBER MARTIN: Just a clarification. As a matter of fact, you brought up that access ports facilitate inspections and maybe maintenance on the top. Now they have the RRBs below. Is that also accessible for inspection at least and maybe some maintenance if there was any issue down there?

MR. BRISTOL: Yes, so as part of refueling we have a space where the upper module inclusive of the recirc valves goes over to a dry dock essentially. And I think most of those maintenance activities are performed there.

MEMBER MARTIN: Okay.

MR. BRISTOL: But that's about as deep as I can go on that topic.

MEMBER MARTIN: Okay. Yes, that's why -- you brought it up, so I -- you opened the door.

MR. BRISTOL: Sure. Most of the components are located on the reactor head, which is sort of where we identified that optimization.

DR. SCHULTZ: Thank you, Ben and Meghan. That additional information is very helpful. Thank you.

MS. TURMERO: Next slide, please? The NuScale PIRT was reviewed for the NPM focusing on topics like the break spectrum comparison, some scaling analyses. The PIRT panel convened for a focused evaluation on the phenomena associated with valve opening events during initial rapid depressurization. And regarding the impact of design changes to LOCA, NuScale evaluated changes to the PIRT geometric parameters and system state parameters, such as pressures and temperatures, and found that these changes did not introduce new phenomena or significantly impact phenomena ranges.

DR. SCHULTZ: Sarah, this is Steve

Schultz. Was there overlap between the first PIRT panel and this one, or was it the same panel that you invited through the process?

MS. McCLOSKEY: There were a couple of different PIRT panels as the design evolved and its LOCA methodology evolved. I believe that the first PIRT panel was actually in 2010. And then it was updated in 2013 and updated in 2015. And so some of the folks who were involved in the 2015 work were also involved in the IORV-focused PIRT as well as the review of the LOCA PIRT. So there was a little bit of overlap. It wasn't a total reconvene of all of the members. The updates were -- the PIRT panel members were all internal to NuScale at this point in the update space.

DR. SCHULTZ: So in this case, they reviewed your -- NuScale's conclusions first that you listed here, and then they looked specifically at the topics that you provided?

MS. McCLOSKEY: I'd say we went the other way around in terms of looking at the design changes and the event progressions and the body of work that NuScale had developed in terms of the NPM-160 work that was done to support the DCA submittal and review. And then we could build on that and compare it to the

-- what the break spectrums looked like with the updated design and all the design changes incorporated and evaluate whether there were significant changes that warranted an update in the PIRT space.

DR. SCHULTZ: And in the last bullet, the methodology changes identified, is that meaning that the PIRT panel identified some changes that you then implemented and evaluated, or is that something that they evaluated?

MR. BRISTOL: So just to clarify, the IORV phenomena were reviewed by a NuScale internal -- internally staffed PIRT panel.

DR. SCHULTZ: Okay.

MR. BRISTOL: The LOCA PIRT was reviewed by our team essentially looking back through what the PIRT panel had originally identified, the basis of those rankings compared with the updated analysis results to confirm that they were consistent. And the limited scope methodology changes -- we'll get into more detail on that in the closed session.

DR. SCHULTZ: Fine.

MR. BRISTOL: I don't think they were primarily driven from phenomena rankings. Largely they were driven by either needs of the power uprate and assessment basis for areas where the margins had required pencil sharpening or margin improvement.

DR. SCHULTZ: That helps. Thank you.

MS. TURMERO: Next slide, please? For the DCA, NRELAP5 Version 1.4 was approved, and the current evaluation model uses Version 1.7. The NIST-2 test facility was upgraded -- the NIST-1 test was upgraded to NIST-2, which allowed us to expand the NRELAP5 assessment bias by using the NIST-2 LOCA and IORV test series. Additional benchmark calculations or sensitivity cases were performed as needed to support these evaluation model changes.

MEMBER MARTIN: Question. Bob again. I saw on their topical that the RELAP5 that you received from Idaho is Version 4.13, one I'm familiar with. It's also 13 years old, and there's been many updates since that time. And I've worked with that code. I know some of the limitations and that they were resolved in some of the later condensations. Always a challenge, particularly under low pressure. Of course, you own NRELAP5. I mean, so as part of these updates are you working with Idaho or are you -- you're taking their changes and incorporated it. So it's really not 4.13 anymore. It actually embodies some of the newer versions, and you're kind of keeping up with what's going on with the development at Idaho.

I see you nodding, but you can go ahead and say it for the record.

MS. McCLOSKEY: Yes, so I'll say for the record we -- I know that we do continue to get the bug reports and code fixes that Idaho -- INL publishes to the users group and incorporate that as part of our normal (audio interference).

MEMBER MARTIN: All right. Some would certainly be more important than the others if your code was crashing. I mean, it's a reality of working with system codes. I think it would be challenging. And low pressure has always just been a huge frustration. They're a lot better than they were when we all started.

MS. TURMERO: Next slide, please? For the evaluation model adequacy assessment, the bottom-up and top-down evaluations that were performed for the NPM-20 builds on the previously approved LOCA adequacy assessment and the non-LOCA evaluation model development for the steam generator and DHRS heat transfer phenomena.

The top-down scaling analyses demonstrate the important PI group similarity between the NPM-160 and NPM-120. There were no significant changes to the field equations or numerical solutions and NRELAP5, with the overall conclusion being that NRELAP5 and the updated evaluation model are applicable and adequate for our defined scope.

Next slide? To conclude, the updated LOCA Topical Report describes the evaluation model using NRELAP5 to analyze the NPM-20 LOCA and valve opening events for Phase 0 and Phase 1A/B and the secondary pipe break for the containment pressure and temperature response.

We've covered a high-level summary of relevant design changes that drive changes to NRELAP5, including an expanded validation basis using the NIST-2 tests, and overall the LOCA Topical Report continues to provide a robust methodology to analyze the NPM response to LOCA valve opening events and the containment pressure and temperature response analysis.

With that, are there any additional questions?

MEMBER HALNON: Walt, I don't see any questions.

CHAIR KIRCHNER: Okay. Greg, then I think at this point if there are no questions from the members, we would turn to the staff for their open presentation on the TR.

MR. SNODDERLY: Walt, we're going to -- Tom Griffith is going to give us the HITI open slides, and then we're going to go to the staff since NuScale's all set up right now.

CHAIR KIRCHNER: Okay. Yes, that's more efficient. Okay. Thank you, Mike.

MEMBER HALNON: Go ahead, Tom.

MR. GRIFFITH: Thank you. Thomas Griffith, Licensing Manager of NuScale Power.

Just a little bit of background about myself. Roughly 15 years' experience in the nuclear industry. I held former positions as a senior reactor operator, I&C manager, worked in safety analysis, reactor engineering, and now work for -- in licensing at NuScale. In charge of the US460 standard design approval application.

What I intend to present here in the open session is some high-level updates on the high-impact tactical issues. There is a set of slides for the closed session, where we can discuss some of the aspects than what I'm going to present right now in more detail.

Next slide, please? So if you recall, in August when we discussed last the high-impact technical issues. We have not identified any new high-impact technical issues since that time. And at this point there are effectively two high-impact technical issues: No. 2 and 10 related to LOCA break spectrum that we continue to work through defining a clear path forward.

I did mark on this slide that the high-impact technical issue related to the IFR design changes and the ASME qualification, the helical coil steam generator, considered resolved by NuScale and NRC management, that decision does officially take place during our quarterly meetings which is next week, but my understanding in discussions with my counterparts at the NRC is that both sides do recommend closure of the item, and enhanced with the timing here in the presentation I wanted to at least highlight that.

The high-impact technical issues related to DWO we would consider resolved and we look forward to presenting the material related to DWO here in some of the upcoming ACRS meetings.

Next slide, please? So I do want to take an opportunity here to provide an overview of the approach to DWO. The purpose of this is kind of to start some conversation. And we do have closed slide presentations that accompany this presentation material, but for the purpose of transparency in the open session I want to walk through kind of what -- how NuScale views the resolution of DWO.

And so effectively what I'm showing here is that NuScale's approach to DWO -- and we're calling it our safety case -- by establishing three separate pillars. We consider those pillars to be analysis, real-time monitoring, and physical inspection. And our approach is not limited to one specific area. So for example, in analysis we've defined a DWO transient. We have analyzed the steam generator integrity with that transient, and we've defined a time under which the steam generator can handle the transient that we've analyzed.

We have also defined real-time monitoring, which is -- I think I would characterize it as -- it was a lot of feedback that we've gotten from both the staff, our internal NuScale individuals, and in discussion that we had in August with the ACRS. We took a hard look at what we were providing to operators, and we've now defined a pillar that we're calling real-time monitoring. And that's effectively a way for the operating team to infer where the steam generator is operating with respect to DWO.

And then the last piece is physical

inspections. And so to accompany the analysis of the real-time monitoring that's provided we've also defined a set of inspections in intervals to provide another layer of confidence that the steam generator, being that it's a reactor coolant pressure boundary -- its integrity is maintained.

Next slide, please? So as I stated before, we'll walk through some of the pillars a little bit here. I do have more detailed slides in the closed session, but effectively under analysis, like I said, we've defined a DWO transient. We've done the evaluations with the transient to demonstrate steam generator structural integrity. We've defined real-time monitoring using a comparison between our RCS hot temperature and main steam temperatures, which are safety-related indications. And then we've also defined a limit, and that limit is required by tech specs for how long a particular steam generator could operate in a region where there's the potential for DWO.

Next slide?

MEMBER MARTIN: Bob Martin. A clarification. My understanding is that with the DWO, you were going to more or less design it out of normal/abnormal operation or design-basis conditions.

And of course part of that would be you would have protection systems to otherwise ensure that this doesn't happen, as opposed to conditions, or allowing for conditions that would result in DWO, right? To clarify, basically take it out of a DBA space or DBE space. Is that correct? Do I understand that properly?

MR. GRIFFITH: So a current approach to DWO, the way I would define it, is that we have established what we -- a representative transient for DWO and analyzed the steam generator for a particular time frame that that transient could occur.

MEMBER MARTIN: Could occur with assumptions that make it a DBE-kind of thing, or is it a beyond-design-basis condition?

MR. GRIFFITH: I think some of the specifics you -- we would want to get into are probably more appropriate for closed, but I would say that the loads induced from the phenomena are relatively low and not impactful. In fact, when we look at the expected operation -- and we have the figure up right now that has been placed in the FSAR 5.1-16 -- we expect the majority of operation to occur in Region II, and Region II is defined as the region that precludes DWO.

That does not mean -- and I don't want to misrepresent -- Region I does not mean you have DWO. Region I simply is that there is less margin to DWO. And there are some slides in the closed session that walk through exactly how those margins are defined. But what we've done is in that Region II -- even if -- based upon the total amount of time that we would expect an applicant to operate, there is sufficient margin that roughly seven years or so of additional margin that DWO could continue or could occur continuously before hitting our acceptance criteria for wear.

So what I'm trying to say is that 90-percent-plus of the operation would be in this DWO is precluded. We had to set an analysis limit somewhere with acceptance criteria. The amount of margin in there before we would -- we would say internally, hey, we need to look at this further is such that there -- or there's on the order of multiple years or many years. And it's kind of tricky because there's different impacts from DWO like, you know, for example, sliding wear is different than thermal fatigue. So I think that we need -- we would need to get into some of those finite details, but there is substantial margin before DWO would cause any sort of concern with the steam generator integrity.

MEMBER MARTIN: I guess where I was going with it, although your answer is what I expected, we've gone through a lot of discussion on design changes. Were there any design changes to support the strategy and specifically related to I&C monitoring that might result in some sort of protective action?

MR. GRIFFITH: I think I can get into that in the closed session.

MEMBER MARTIN: Okay. I mean yes or no might work, but --

MR. GRIFFITH: To some extent the answer is yes.

MEMBER MARTIN: Good enough.

MR. GRIFFITH: Next slide, please? So lastly, the other piece we'd talked through a little bit is we've set a number of physical examinations that include the steam generator and associated components and specified frequencies that would inform what future inspections may need to look at and provide the assurance of steam generator integrity.

Next slide?

MEMBER HARRINGTON: One question real quick. This is Craig Harrington. Are you envisioning that inspection schedule as specific to the first module that operates or something that would be appropriate long term?

MR. GRIFFITH: So we do have specifics on the first module.

Erin, I don't know if you can step here on the exact language for the first module?

But there are specifics for the first module under operation. And the frequency is set to complement what we've done for analysis.

MS. BLUMSACK: Yes, this is Erin Blumsack from NuScale. For the first NPM that undergoes a refueling outage, after the first 100 percent tube examinations at the first refueling outage they're required to inspect at least 20 percent of the steam generator tubes at each outage with the tech spec requirement that they have to get to 100 percent of tube inspection by 72 EFPM after the first outage. That is a COL item in Chapter 5. Tech specs have not changed and that's only for the first NPM that undergoes refueling.

MEMBER BALLINGER: This is Ron Ballinger. I'm not sure -- I don't remember, so I'll ask the question: Since it's externally pressurized tubes, a lot of the inspections for various phenomena that occur in a commercial PWR -- it's not applicable. But there are others. For example, dimensions. Is there inspection going to occur that looks at the ID? In other words, just to find out if there's any kind of pre-collapse or creep-down that's going on in the tubes?

MS. BLUMSACK: Details of examinations will be developed for a COL applicant as part of the Steam Generator Program.

MEMBER BALLINGER: I guess -- okay.

MS. BLUMSACK: Does that address your question?

MEMBER BALLINGER: Yes, I guess. We have -- a lot of us have issues every time somebody says that's up to the COL applicant, so --

MS. BLUMSACK: Understood. We expect to be able to use examination techniques that the industry uses, but that will be developed in more detail during the Steam Generator Program.

MEMBER BALLINGER: You'll find out the first time a bobbin coil gets stuck in one of the tubes.

MS. BLUMSACK: That is true.

MR. GRIFFITH: And I believe that that was the last slide that I had, so if there's any further questions for the open session, any update?

MEMBER HALNON: Walt, I don't see anything in the room.

CHAIR KIRCHNER: Other members, any questions?

(No response.)

CHAIR KIRCHNER: Okay. Bear with me. I'm on a small NRC computer screen. I have to move so I can look at the agenda, so bear with me.

Okay. With that then, I believe we are ready, Mike, to go to the staff's evaluation of the LOCA TR in the open session.

MR. SNODDERLY: Yes, sir, that's --

CHAIR KIRCHNER: Is that correct?

MR. SNODDERLY: I just need a minute to switch presenters.

MEMBER HALNON: I'll let you know when we're ready, Walt.

CHAIR KIRCHNER: Okay. So when you're ready, just go ahead. I can't see --

MR. SNODDERLY: I'll let you know. And then also an opportunity for public comment after the staff's presentation. And then --

CHAIR KIRCHNER: Yes.

MR. SNODDERLY: -- that will be end of the open session.

CHAIR KIRCHNER: Right. Correct. Okay.

MEMBER HALNON: Okay. Just give us a minute.

(Pause.)

CHAIR KIRCHNER: Okay. Staff, whenever you're ready.

MR. VIVANCO: Good morning, everyone. My name is Ricky Vivanco, and I'm a project manager in the NRR New Reactor Licensing Branch. I am the PM assigned to the staff's review of NuScale's Loss of Coolant Accident Evaluation Model Topical Report.

The technical reviewers a part of this review are: Dr. Shanlai Lu, Dr. Sean Piela, Dr. Dong Zheng, Mr. Carl Thurston, Mr. Ryan Nolan, Dr. Syed Haider, Dr. Joshua Kaizer, Dr. Peter Lein from the Office of Research, and Dr. Leonard Ward from Numark. Again, I am the project manager assigned to this topical report supported by (audio interference) for the overall project.

A review of this topical report. The Revision 3 of the LOCA Evaluation Topical Report was submitted on January 5th, 2023, and the topical report was accepted for review on July 31st, 2023.

The staff conducted an audit from March 2023 to August 31st, 2024. Fifty-seven audit issues were generated resulting in supplemental information submitted by NuScale. Of those items not resolved during the audit, four RAIs were generated. I want to be clear here, all audit items and RAIs were resolved/closed that were specifically related to the LOCA Topical Report. I do want to point out though there are confirmatory items that are awaiting confirmation and upcoming revisions or related to open items in other areas of the review -- in other topical reports.

Due to the technical and proprietary nature of the topical report, the details of the staff's review are going to be covered in the closed session, however we will go over the conclusions here.

Subject to the closure of those open and confirmatory items I noted, and along with 11 limitations and conditions identified, the staff concludes that the methodology is acceptable for meeting the requirements of 10 CFR 50.46 and the associated portions of Appendix K evaluated in the topical report.

For evaluation of the ECCS performance in the NuScale NPM-20 for design-basis LOCAs, the proposed LOCA evaluation model is conservative to determine CHF and collapsed liquid level above the reactor core. Further, the staff finds that the containment response analysis methodology is conservative and acceptable and that the NRELAP5 computer code and the NPM-20 model are acceptable to evaluate the MCHFR and IORV in LOCA events.

And that's the end of the staff's presentation. I'll defer any questions to Dr. Shanlai Lu.

DR. LU: As our project manager mentioned, there are a lot of details we can present in the proprietary session, but if there are any questions for the staff at this point in open session, I'm here to answer.

MEMBER HALNON: I don't see any in the room, Walt.

CHAIR KIRCHNER: Members online, any questions?

Hearing none, I think we are at a juncture where we should ask for any public comment.

So members of the public either in the room or online -- just those of you online, un-mute yourself and state your name and affiliation as appropriate and ask your question. Or make your comment. Excuse me. Not question.

MEMBER HALNON: We have no one in the

room, Walt.

CHAIR KIRCHNER: Okay. Online, any comments from the public?

(No response.)

CHAIR KIRCHNER: Okay. I think, Greg, we don't have any public input today.

With that, then, I think we're at the juncture where we can close this open session and move to an actual closed session. And that will take up the LOCA Evaluation Model first.

So with that, for those of you that attended on the open session, thank you.

Again for the record, I want to thank NuScale for joining us so early this morning. And this open session is closed. Thank you.

MEMBER HALNON: Okay. We'll be logging off this one and be logging on -- there should be a new link for those that are invited to the closed session. This session will be closed.

(Whereupon, the above-entitled matter went off the record at 11:04 a.m.)

January 09, 2025 Docket No. 052-050

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Presentation Material Entitled "ACRS Subcommittee Meeting (Open Session) Chapter 16, Part 4, LOCA LTR, and HITI Status," PM-177830, 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 January 15, 2025. The materials support NuScale's presentation of the subject chapter, topical report and status of the US460 Standard Design Approval Application.

The enclosure to this letter is the nonproprietary presentation entitled "ACRS Subcommittee Meeting (Open Session) Chapters 16, Part 4, LOCA LTR, and HITI Status," PM-177830, Revision 0.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Jim Osborn at 541-360-0693 or at josborn@nuscalepower.com.

Sincerely,

Thomas Griffith
Director, Regulatory Affairs
NuScale Power, LLC

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Michael Snodderly, Senior Staff Engineer, Advisory Committee on Reactor Safeguards, NRC

Enclosure 1: ACRS Subcommittee Meeting (Open Session) Chapters 16, Part 4, LOCA LTR, and HITI Status, PM-177830, Revision 0

Enclosure 1:

ACRS Subcommittee Meeting (Open Session) Chapters 16, Part 4, LOCA LTR, and HITI Status, PM-177830, Revision 0

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ACRS Subcommittee Meeting
(Open Session)

January 15, 2025

Chapter 16, Part 4, LOCA LTR and HITI Status

ACRS Subcommittee Meeting
(Open Session)

January 15, 2025

Chapter 16
Technical Specifications

Presenter
Gene Eckholt

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NuScale SDAA Part 4, US460 Generic Technical Specifications (GTS)

- Subpart E of 10 CFR 52, Standard Design Approvals, does not require submittal of Technical Specifications for consideration.
- In the Statements of Consideration for the 2007 rule change to 10 CFR Part 52, the commission expressed its expectation that the contents of applications for design approvals should contain essentially the same technical information that is required of design certification applications.
- NuScale included Part 4, Generic Technical Specifications, in the SDAA.

Noteworthy US460 Design Changes Affecting GTS

- Rated thermal power increase from 160 MWt to 250 MWt
- Modification of ECCS design from three to two reactor vent valves
- Addition of ECCS supplemental boron system
- Addition of passive autocatalytic recombiner in containment

US460 GTS Development

- Started with NuScale US600 Certified Design Technical Specifications as model
- Addressed US460 design changes
- Applied 10 CFR 50.36 criteria to plant design, operations, and safety analyses
- Used industry STS Writer’s Guide format and guidance
- Incorporated recent industry STS changes as appropriate
- Technical Report TR-101310 Rev 0 describes the differences between US600 and US460 GTS at the time of SDAA submittal

Noteworthy GTS Changes Described in TR-101310 Revision 0

- The MODE definition revised to better align with the plant response behavior
- The reactor core critical heat flux correlations and limits, and the RCS pressure safety limits revised to reflect the increased reactor power and changes to the plant design
- New Surveillance Requirement to ensure isolation of Module Heatup System between modules
- Module Protection System requirements modified to align with design changes
- Remote Shutdown Station LCO removed
- RCS Operational Leakage LCO and definition modified to align with industry standards to the extent appropriate for the NuScale Design

Noteworthy GTS Changes Described in TR-101310 Revision 0 (continued)

- LTOP and ECCS LCOs modified to reflect reduced number of reactor vent valves
- UHS LCO modified to reflect design changes
- New LCO to ensure OPERABILITY of ECCS Supplemental Boron System
- New LCO to ensure containment closure during module movement between operating location and containment closure tool
- LCO 3.7.3 removed due to change from leak-before-break to break exclusion
- Chapter 5 Administrative Controls modified to reflect approved control room staffing plan

US460 GTS Review

- Audit Results
- 68 audit items resolved
- Most changes were editorial or clarifications
- Noteworthy changes included
  - Core reactivity balance surveillance frequency clarified
  - Module Heatup System flow paths added to Boron Dilution Control specification
  - ECCS Supplemental Boron specification revised to include requirements for the geometric form of boron pellets

- RAI Results
- No RAI questions on Chapter 16 or GTS
- GTS change resulting from RAI associated with another FSAR Chapter
  - Steam Generator Program revised to update the SG tube integrity discussion

Noteworthy GTS Change Not Associated with SDAA Review

- Added LCO to ensure OPERABILITY of passive autocatalytic recombiner (PAR)
  - NuScale determined the PAR mitigates design-basis events, making the component safety-related and appropriate for inclusion in the GTS

Acronyms

CFR	Code of Federal Regulations
ECCS	Emergency Core Cooling System
FSAR	Final Safety Analysis Report
HVAC	Heating, Ventilation, and Air Conditioning
LCO	Limiting Condition for Operation
LTOP	Low Temperature Overpressure Protection
MWt	Megawatts Thermal
PAR	Passive Autocatalytic Recombiner
RAI	Request for Additional Information
RCS	Reactor Coolant System
SDAA	Standard Design Approval Application
STS	Standard Technical Specifications
TS	Technical Specifications
UHS	Ultimate Heat Sink

ACRS Subcommittee Meeting
(Open Session Session)

January 15, 2025

Loss-of-Coolant Accident Topical Report

Presenter: Sarah Turmero

Agenda

- Summary of significant changes since TR-0516-49422-P Revision 2 approval
  - Analysis scope addressed by topical report
  - Design changes from 160 MWt NPM-160 design to 250 MWt NPM-20 design
  - Summarize effects on PIRT
  - EM structure and assessment basis updates
  - Adequacy assessment process and conclusions
- Conclusions

LOCA Topical Report: Analysis Purpose and Transient Class

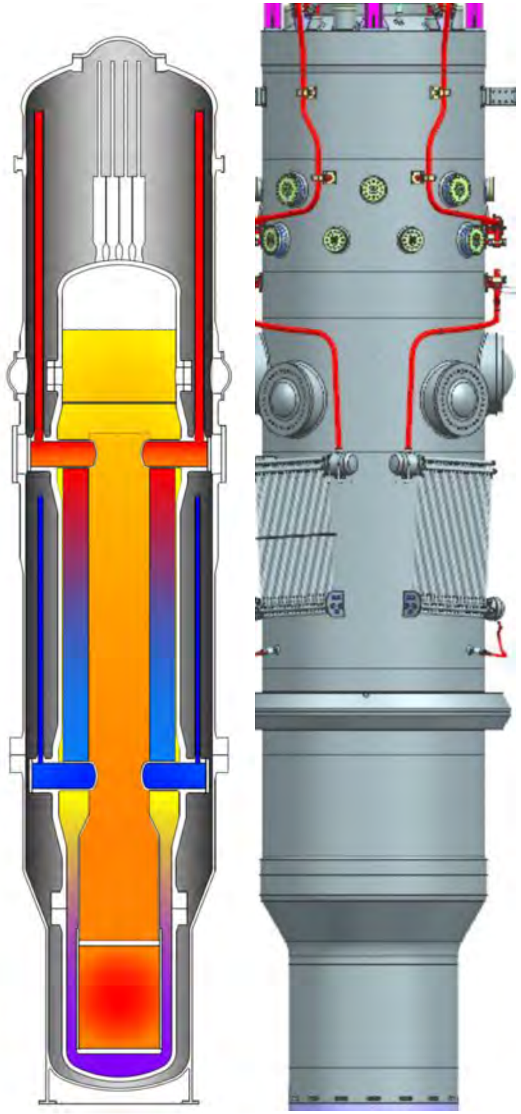
- Scope maintained from DCA Topical Report:
  - Loss-of-coolant accident (LOCA) pipe break inside containment analysis method
  - Event classification: Postulated accident
  - Figures of merit: Phase 1a, 1b collapsed liquid level (CLL) over top of fuel, minimum critical heat flux ratio (MCHFR)
  - Key Regulations: 10 CFR 50.46, 10 CFR 50 Appendix K, GDC 35
- Scope modified from DCA Topical Report:
  - Inadvertent opening of a reactor valve (IORV) analysis method
  - Event classification: conservatively classified as Anticipated Operational Occurrence (AOO)
    - Realistically not expected to occur during a module lifetime
  - Figure of merit: MCHFR during ‘Phase 0’ initial blowdown
  - Key Regulations: GDC 10

< Event Initiation	< RTS		< ECCS Actuation	< Recirculation
Phase 0	10 s	(*ends 10 seconds after rods are fully inserted)		
Phase 1a			Phase 1b	LTC

- Scope added from DCA Topical Report:
  - Containment vessel (CNV) pressure/temperature response analysis method
    - Similar to method used in DCA technical report
  - Response to LOCA pipe break, secondary line breaks, IORV events, or inadvertent ECCS actuation
  - Figures of merit: Maximum CNV pressure, maximum CNV wall temperature, CNV pressure reduction over time
  - Key Regulations: GDC 16, PDC 38, GDC 50
- Scope addressed elsewhere:
  - Extended passive cooling and reactivity control (XPC) topical report addresses long-term core cooling and subcriticality

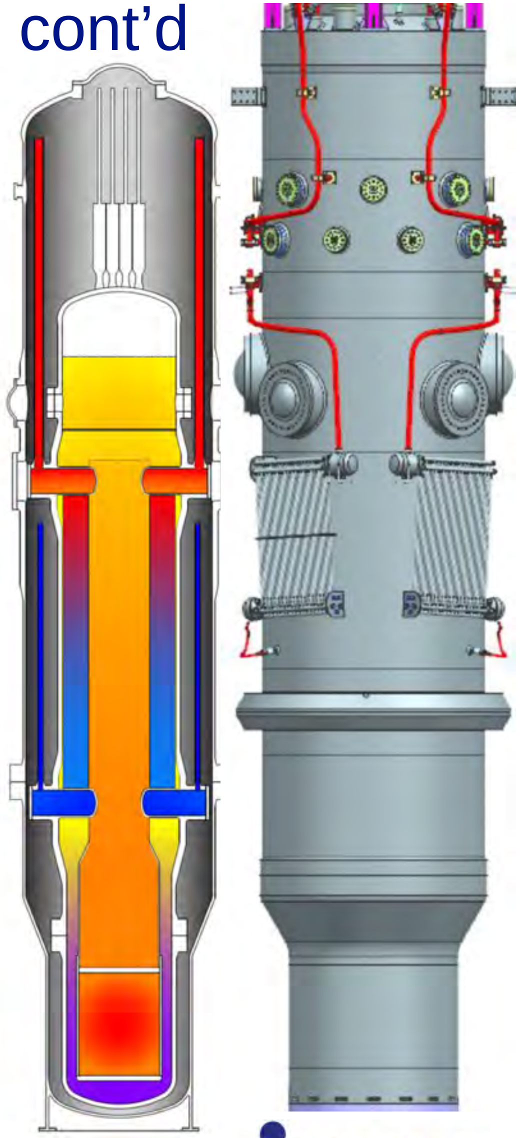
Power Uprate and Design Changes Summary NPM-160 to NPM-20

- Power uprate from 160 MWt to 250 MWt
- Module SSC design essentially maintained
- Operating conditions
  - Increased primary pressure from 1850 psia to 2000 psia
  - Primary and secondary side design pressures increased from 2100 psia to 2200 psia
  - Use T_{avg} control instead of T_{hot} control (RCS Tavg change from ~545°F to 540°F)
  - Decreased secondary side feedwater temperature at 100% power from 300°F to 250°F
  - Reduced minimum temperature for criticality from 420°F to 345°F
- Containment vessel
  - Design pressure increased from 1050 psia to 1200 psia
  - Design temperature increased from 550°F to 600°F
  - Upper containment material change from SA-508 to SA-336 F6NM



Power Uprate and Design Changes Summary NPM-160 to NPM-20, cont'd

- ECCS actuation signals modified
  - Low RCS level (top of riser), Tcold interlock
  - Low-low RCS level (mid-riser) always active (no interlocks)
  - Timers:
    - 8-hour timer on all reactor trips; operators can block the actuation if subcriticality at cold conditions is confirmed and combustible gas mixture in RPV is precluded
    - 24-hour timer after loss of AC power supply (unchanged from DCA)
- ECCS valve design changes
  - Removed IAB from vent valves to enhance depressurization capability in DBE and BDBE
  - Modified IAB threshold/release pressures on recirculation valves
  - Added second trip valve to each ECCS valve to prevent inadvertent opening on solenoid failure
  - Added flow venturi to RVVs and reactor recirculation valves (RRVs)
  - Removed third RVV
- Long-term passive cooling enhancements for collapsed liquid level and subcriticality FOM
  - Addressed by Extended Passive Cooling and Reactivity Control methodology
  - Design changes reduce but maintain ample CNV cooling capacity:
    - Lowered reactor pool level from ~68 ft to ~53 ft
    - Reduced conductivity in upper CNV due to material change
  - Mitigation of boron redistribution during DHRS and ECCS cooling with riser hole flow paths
  - Supplemental ECCS Boron (ESB) to maintain subcriticality during extended passive cooling



Phenomena Identification and Ranking

- NuScale reviewed the NPM-160 LOCA PIRT for applicability to the 250 MWt NPM-20 design based on:
  - Plant break spectrum comparisons
  - Scaling analyses
  - Other NPM-160 work subsequent to the PIRT development
- NuScale convened a PIRT panel for focused evaluation of IORV phenomena during rapid initial depressurization
- Overall conclusions:
  - Existing phenomena identification and rankings applicable for 250 MWt NPM-20 design
  - Clarified phenomena importance based on work for NPM-160 and NPM-20 designs performed after development of NPM-160 LOCA PIRT
- Limited scope of methodology changes identified, and supported by additional NRELAP5 validation and sensitivity analyses

EM Structure and Assessment Basis

- NRELAP5 v1.4 previously approved in DCA
- Current EM employs NRELAP5 v1.7
- Upgraded the NIST-1 integral effects test facility to NIST-2
- Expanded NRELAP5 assessment basis:
  - NIST-2 LOCA test series
  - NIST-2 IORV test series
- Additional benchmark calculations, sensitivity cases as needed to support EM changes

EM Adequacy Assessment

- Bottom-up and top-down evaluations performed
  - Builds on LOCA EM adequacy assessment performed for DCA
  - Builds on the non-LOCA EM development for SG/DHRS heat transfer phenomena
- Compared NPM-160 and NPM-20 geometry, operating conditions, range of conditions for LOCA spectrum
- Evaluated scope of NRELAP5 code changes since v1.4
- Top-down scaling analyses demonstrated important PI group similarity between NPM-160, NPM-20, NIST-2
- No significant changes to NRELAP5 field equations or numerical solution
- NIST-2 LOCA and IORV tests expand the NRELAP5 assessment basis for NPM integral response
- **Conclusion**: NRELAP and updated EM are applicable and adequate for the defined scope.

Conclusions

- Updated topical report describes evaluation model for use of NRELAP5 to analyze:
  - NPM-20 LOCA or valve opening events, to assess Phase 0 MCHFR, Phase 1a/1b MCHFR, collapsed liquid level,
  - NPM-20 LOCA, valve opening, and secondary pipe break containment pressure response
- Design changes from NPM-160 to NPM-20 evaluated for effect on LOCA or valve opening transient and important phenomena
- NRELAP5 code changes incorporated as necessary to support the NPM-20 EMs
- NRELAP5 validation basis expanded with NIST-2 tests
- Topical report provides robust methodology to analyze NPM response to LOCA and valve opening events, and for containment pressure/temperature response analysis.

Questions?

Acronyms

AC	Alternating Current
AOO	Anticipated Operational Occurrence
BDBE	Beyond Design Basis Event
CLL	Collapsed Liquid Level
CNV	Containment Vessel
DBE	Design Basis Event
DCA	Design Certification Application
DHRS	Decay Heat Removal System
ECCS	Emergency Core Cooling System
EM	Evaluation Model
ESB	ECCS Supplemental Boron
FOM	Figure of Merit
IAB	Inadvertent Actuation Block
IORV	Inadvertent Opening of an RPV Valve
GDC	General Design Criteria
LOCA	Loss-of-Coolant Accident
MCHFR	Minimum Critical Heat Flux Ratio
NIST	NuScale Integral
NPM	NuScale Power Module
PDC	Principal Design Criteria
PIRT	Phenomena Identification and Ranking Table
RAI	Request for Additional Information
SDAA	Standard Design Approval Application
SDA	Standard Design Approval
SSC	Systems, Structures, and Components
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
RRV	Reactor Recirculation Valve
RVV	Reactor Vent Valve
XPC	Extended Passive Cooling

ACRS Subcommittee Meeting
(Open Session)

January 15, 2025

Update – High Impact Technical Issues

Presenter: Thomas Griffith

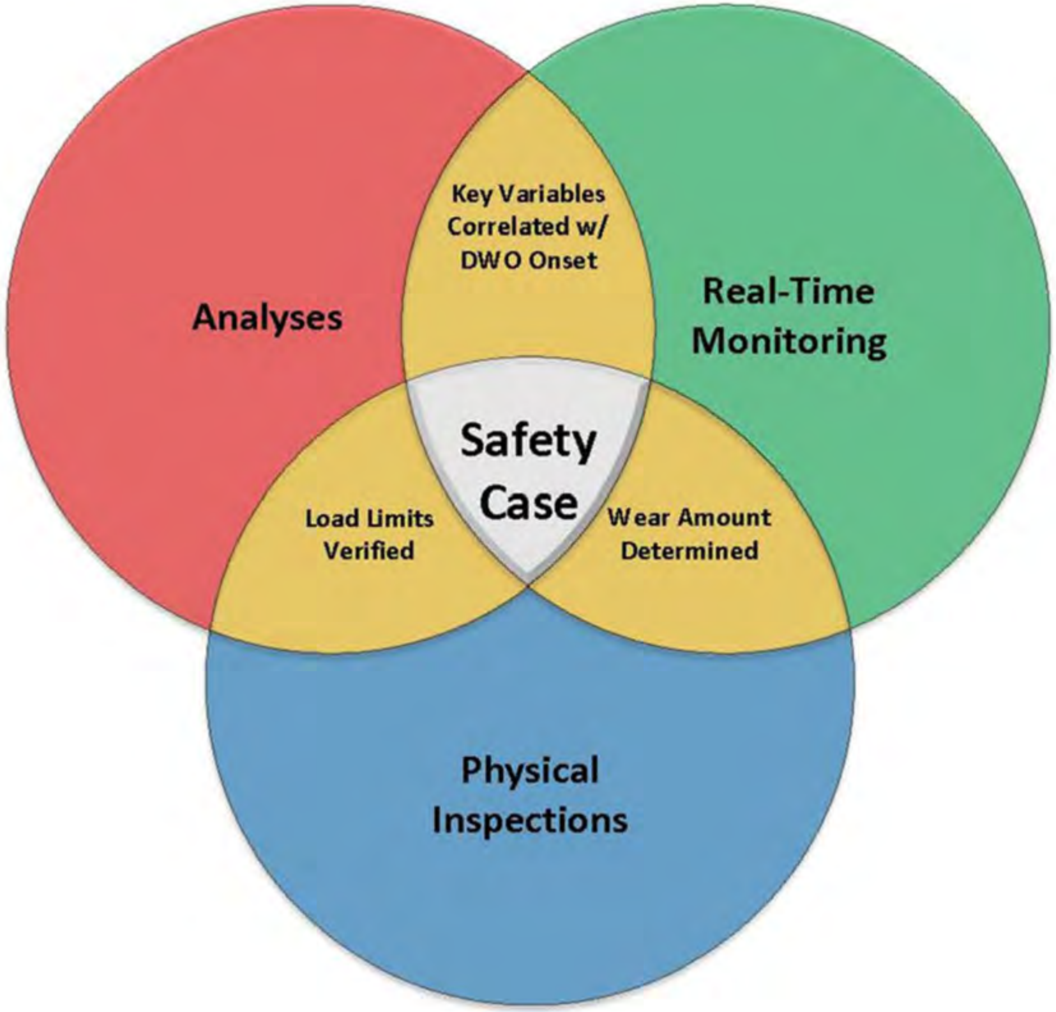
High Impact Technical Issues (HITIs)

1. Design and classification of the augmented DC power system (EDAS)
2. Loss-of-Coolant (LOCA) break spectrum
3. Incorporated by reference (IBR)
4. Containment Vessel (CNV) material change
5. Lower reactor pressure vessel (RPV) material change
6. Secondary side controller design for density wave oscillation (DWO) events
7. DWO and steam generator inlet flow restrictor (IFR) design changes
8. ASME qualification of the helical coil steam generator for the onset of DWO-induced loads
9. Upper-to-lower RPV flange bolted joint shear loading that results from differential thermal expansion
10. LOCA break at CVCS/CIV connection

- Note: Green indicates issues that have been considered resolved by NuScale and NRC Management

Density Wave Oscillation Safety Case

- Analyses – DWO transient, which is used to assess SG structural integrity. SG structural integrity ensured for longer than NPM lifetime limit for time in DWO.
- Real-Time Monitoring – Defined operational space where DWO is precluded and where time in DWO is conservatively accounted for against the NPM lifetime limit.
- Physical Inspections – Examinations of SG tubes and IFRs ensure RCPB integrity is maintained. Degradation assessment will ensure that any damage to the tubes will inform future examination locations and frequencies.



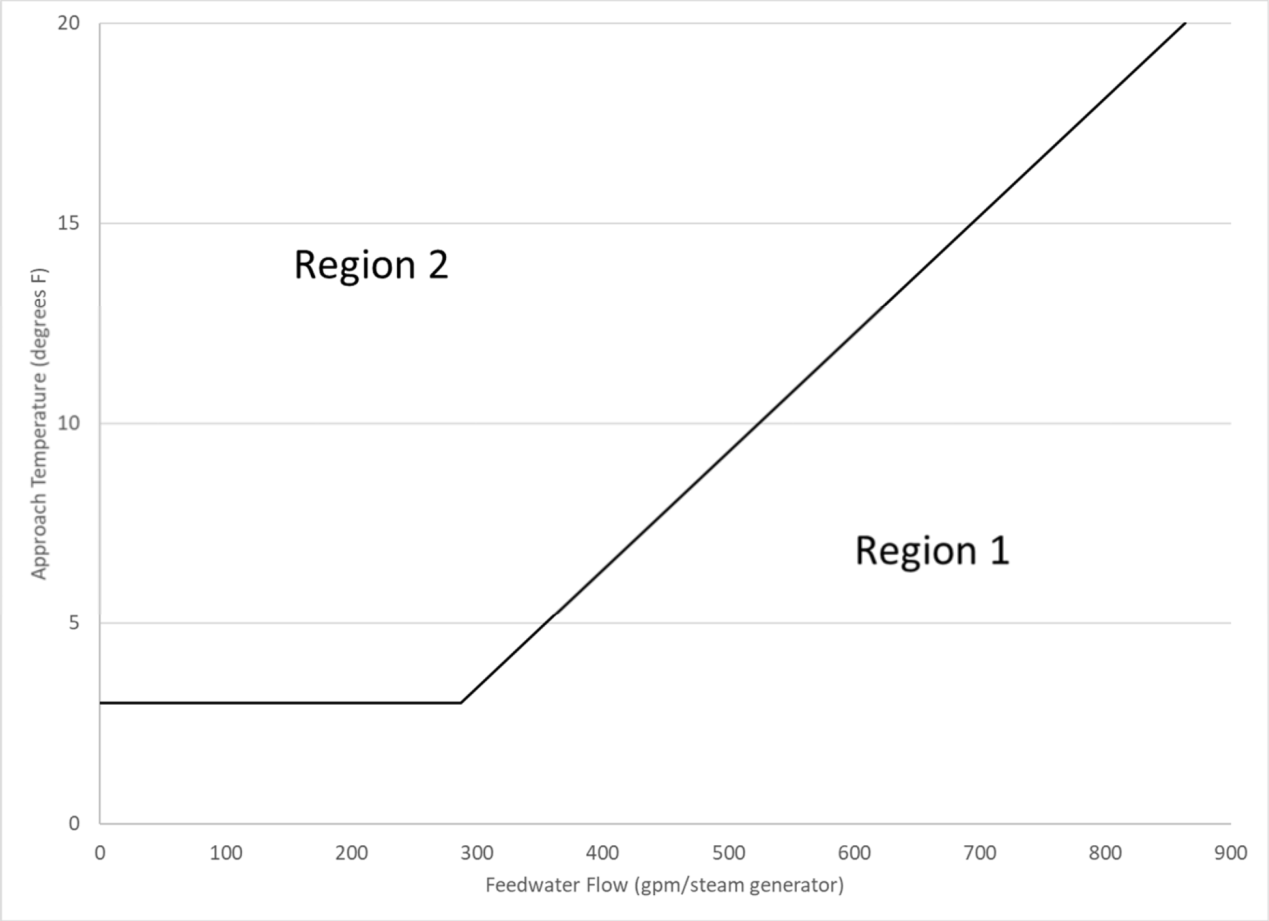
DWO Safety Case (Continued)

- Analyses
  - DWO transient defined in SDAA FSAR Section 3.9.1 and lifetime limit specified in Table 3.9-1
  - SG structural integrity is evaluated beyond the DWO lifetime limit for the NPM 60-year design life.
- Real-Time Monitoring
  - SG approach temperature
    - Comparison between RCS hot temperature and main steam temperature
  - Time is counted in DWO against FSAR Table 3.9-1 “Summary of Design Transients” 60-year US460 design life limit of 2840 days in DWO.
    - Technical Specifications 5.5.3 cyclic limits

DWO Safety Case (Continued)

- Real-Time Monitoring
  - DWO is precluded during normal operations by maintaining an adequate SG approach temperature.
    - DWO is precluded in Region 2
    - Margin between normal operation and the Region 1/Region 2 boundary; Margin between the Region 1/Region 2 boundary and DWO onset.
    - Operation with DWO is avoidable for most of the NPM operating life.

SDAA FSAR Figure 5.4-16



DWO Safety Case (Continued)

- Physical Examinations
  - SG tube examination requirements in Technical Specifications 5.5.4
    - 100 percent SG tube examination at first refueling outage
    - 100 percent SG tube examination over 72 EFPM (~ 6 years) after first refueling outage:
      - Maximum time below approach temperature boundary (2840 days or >7 years) is greater than maximum time between SG tube examinations.
      - Additional requirement to inspect at least 20 percent of tubes per outage for the first NPM to undergo a refueling outage
    - Degradation assessment program will ensure that examination results factor into future examination frequency and location.
    - VT-3 examination of IFRs

Acronyms

ASME	American Society of Mechanical Engineers
CIV	Containment Isolation Valve
CNV	Containment Vessel
CVCS	Chemical and Volume Control System
DWO	Density Wave Oscillation
EDAS	Augmented DC Power System
EFPM	Effective Full Power Months
FSAR	Final Safety Analysis Report
HITI	High Impact Technical Issue
IBR	Incorporate by Reference
IFR	Inlet Flow Restrictor
LOCA	Loss-of-Coolant Accident
NPM	NuScale Power Module
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
SDAA	Standard Design Approval Application
SG	Steam Generator

Presentation to the ACRS Subcommittee Staff Review of NuScale SDAA Part 2, Chapter 16, “Technical Specifications,“ and Part 4 , “US460 Generic Technical Specifications,” Volume 1, Specifications, and Volume 2, Bases Revision 1

January 15th, 2025

(Open Session)

NuScale SDAA Part 2 Chapter 16 and Part 4 Review

Overview

- NuScale submitted Part 2 (FSAR), Chapter 16, “Technical Specifications” (TS), and Part 4, “US460 Generic Technical Specifications” (GTS), Revision 0, of the NuScale SDAA on December 29 and December 31, 2022, respectively, and Revision 1 on October 31, 2023
- NRC regulatory audit of FSAR Chapter 16 and Part 4 was performed from March 2023 to August 2024, generating 68 audit issues
- All audit issues were resolved in the audit
- 52 audit issues resulted in NuScale submitting supplemental information to address questions raised during the audit
- No RAIs issued
- Staff completed review of FSAR Chapter 16 and Part 4 and issued an advanced safety evaluation to support today's ACRS Subcommittee meeting

Non-Proprietary

NuScale SDAA Part 2 Chapter 16 and Part 4 Review

Contributors

- **Technical Reviewers**
  - Craig Harbuck, Lead Reviewer, NRR/DSS/STSB
  - Steve Smith, NRR/DSS/STSB
  - Clint Ashley, NRR/DSS/STSB
  - Josh Wilson, NRR/DSS/STSB
- **Project Managers**
  - Alina Schiller, PM, NRR/DNRL/NRLB
  - Getachew Tesfaye, Lead PM, NRR/DNRL/NRLB

NuScale SDAA Part 2 Chapter 16 and Part 4 Review

Part 2, FSAR Chapter 16, TS

- Section 16.1 Technical Specifications
- TR-101310-NP, Revision 0, “US460 Standard Design Approval Technical Specifications Development”

Part 4, GTS Volume 1, Specifications

- Chapter 1.0 Use and Application
- Chapter 2.0 Safety Limits (SLs)
- Chapter 3.0 Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs)
- Chapter 4.0 Design Features
- Chapter 5.0 Administrative Controls

Part 4, GTS Volume 2, Bases

- Chapter B 2.0 SLs
- Chapter B 3.0 LCOs and SRs

NuScale SDAA Part 2 Chapter 16 and Part 4 Review

Significant Changes from DCA to SDA **GTS Chapter 1**

Definition of MODE 3 – Safe Shutdown

DCA	SDA
Module is shutdown (k_{eff} < 0.99)	Module is shutdown (k_{eff} < 0.99)
All indicated reactor coolant temperatures < 420 °F (minimum temperature for criticality)	All indicated reactor coolant temperatures < 345 °F (minimum temperature for criticality) OR PASSIVELY COOLED ➢ Any indicated reactor coolant temperature may be ≥ 345 °F

NuScale SDAA Part 2 Chapter 16 and Part 4 Review

<u>Significant Changes from DCA to SDA **GTS Chapter 1 (cont'd)**</u>

Definition of Reactor Coolant System (RCS)

pressure boundary LEAKAGE

- Industry Technical Specification Task Force (TSTF) traveler 554, Rev. 1, approved on December 18, 2020 (ML20324A083) and incorporated into Revision 5 of NUREG-1431, "Standard TS Westinghouse Plants," changed the definition – as shown:

> "LEAKAGE (except primary to secondary LEAKAGE) through a ~~nonisolable~~ fault in an RCS component body, pipe wall, or vessel wall. <u>LEAKAGE past seals, packing, and gaskets is not pressure boundary LEAKAGE</u>."

NuScale SDAA Part 2 Chapter 16 and Part 4 Review

Significant Changes from DCA to SDA **GTS Chapter 2**

Reactor Core Safety Limits

DCA US600		SDAA US460	
Critical Heat Flux Ratio		**Critical Heat Flux Ratio**	
Correlation	Safety Limit	Correlation	Safety Limit
NSP2	≥ [1.17]	-------	
NSP4	≥ [1.21]	NSP4	≥ [1.21]
Extended Hench-Levy	≥ [1.06]	-------	
-------		**NSPN-1**	**≥ [1.15]**

Peak fuel centerline temperature ≤ { 4901 - (1.37E-3 × Burnup, MWD/MTU) } °F.

Reactor Coolant System Pressure Safety Limit

DCA US600	SDAA US460
Pressurizer Pressure ≤ 2285 psia	**Pressurizer Pressure** ≤ **2420** psia

NuScale SDAA Part 2 Chapter 16 and Part 4 Review

Significant Changes from DCA to SDA **GTS Chapter 3**

LCO Subsection Omissions and Additions

DCA LCOs Omitted in SDAA

- 3.3.5 **Remote Shutdown Station**
  - Indication-only monitors in I&C equipment rooms
  - No Type-A PAM variables
- 3.7.3 **In-Containment Secondary Piping Leakage**
  - Leak-before-break (LBB) methods not used
  - High energy pipe break exclusion criteria met (consistent with BTP 3-4)

Additional LCOs in SDAA

- 3.5.4 **Emergency Core Cooling System Supplemental Boron** (ECCS ESB)
  - 3.3.3 Function 1 ECCS – 8-hour post reactor trip actuation timer; SR 3.3.3.3
- 3.6.3 **Containment Closure**
  - Mode 3 and Passively Cooled; Mode 4 before unseating of upper module assembly from lower containment vessel flange
  - Maintain reactor coolant inventory to ensure adequate core cooling
- 3.6.4 **Passive Autocatalytic Recombiner**
  - Meets 10 CFR 50.44(d)

NuScale SDAA Part 2 Chapter 16 and Part 4 Review

Significant Changes from DCA to SDA **GTS Chapter 3**

Instrument Functions that Initiate **ECCS**

DCA Revision 5	SDAA Revision 2 (draft)
22.a **High Containment Water Level**	----------
23.a **Low RCS Pressure**	----------
----------	23.a **Low RPV Riser Level** *(if above 500 °F)*
----------	24.a **Low Low RPV Riser Level**
----------	25.h **Low AC Voltage to EDAS Battery Chargers**
Three reactor vent valves (RVVs)	Two RVVs
Two reactor recirculation valves (RRVs)	Two RRVs
• Inadvertent Actuation Block (IAB) on each valve delays valve opening on ECCS actuation signal until RPV-CNV pressure difference below unblock setting	• No IAB on RVVs; EDAS DC power ensures ECCS hold function until ECCS actuation signal or reactor trip occurs • RRV opening delayed by IAB

NuScale SDAA Part 2 Chapter 16 and Part 4 Review

Significant Changes from DCA to SDA **GTS Chapter 3**

Instrument Functions that Initiate **DHRS**

DCA Revision 5	SDAA Revision 2 (draft)
7.b **High Pressurizer Pressure**	7.b **High Pressurizer Pressure**
----------	11.c **Low Pressurizer Level**
13.b **High Narrow Range (NR) RCS T_{HOT}**	13.b **High Narrow Range (NR) RCS T_{HOT}**
16.b **High Main Steam Pressure**	17.b **High Main Steam Pressure**
----------	22.c **High NR Containment Pressure**
25.b **Low AC Voltage to ELVS Battery Chargers**	25.c **Low AC Voltage to EDAS Battery Chargers**
----------	26.c **High Under-the-Bioshield Temperature**

Low Pressurizer Level

- Steam Generator Tube Failure

High Under-the-Bioshield Temperature

- High-energy line breaks under the bioshield

High NR Containment Pressure

- Loss of containment vacuum
- Feedwater System pipe break
- Inadvertent RVV opening

NuScale SDAA Part 2 Chapter 16 and Part 4 Review

Significant Changes from DCA to SDA **GTS Chapters 3 and 5**

SDA GTS Improvements Over DCA GTS

- 3.1.9 **Boron Dilution Control**
  - Added configuration constraints on the Module Heatup System (MHS) to **ensure the MHS is never aligned** to the CVCS injection line of more than one NuScale Power Module
- 3.3.1 **MPS Instrumentation -** accommodating control rod coupling and uncoupling
  - Mode 3 Applicability of selected reactor trip and DWSI instrument functions
  - "when capable of withdrawal of more than one control rod assembly (CRA)"
- 5.5.4 **Steam Generator (SG) Program**
  - 72 effective full power month inspection interval for all SG tubes
  - RCS pressure is on the outside of the SG tubes, so the tubes are susceptible primarily to collapse or buckling rather than burst
  - Tube plugging criterion of 40 percent through-wall thickness is bracketed as part of COL Item 16.1-1

NuScale SDAA Part 2 Chapter 16 and Part 4 Review

Conclusion

NuScale US460 Standard Design GTS and Bases are acceptable because they comply with

- 10 CFR 50.34, "Contents of Applications; Technical Information";
- 10 CFR 50.36, "Technical Specifications"; and
- 10 CFR 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors."

Presentation to the ACRS Subcommittee
Staff Review of NuScale's Loss-of-Coolant Accident (LOCA) Evaluation Model Topical Report (TR 0516-49422-P)

January 15th, 2025

(Open Session)

NuScale LOCA Topical Report Review

Contributors

- **Technical Reviewers**
  - Dr. Shanlai Lu, Lead, NRR
  - Dr. Sean Piela, NRR
  - Dr. Dong Zheng, NRR
  - Mr. Carl Thurston, NRR
  - Mr. Ryan Nolan, NRR
  - Dr. Syed Haider, NRR
  - Dr. Joshua Kaizer, NRR
- **Project Managers**
  - Ricky Vivanco, PM, NRR
  - Getachew Tesfaye, Lead PM, NRR

NuScale LOCA Topical Report Review

Overview

- NuScale submitted Loss-of-Coolant Accident (LOCA) Evaluation Model Topical Report (TR 0516-49422-P), rev. 3, on January 5, 2023. The topical report was formally accepted for review on July 31, 2023
- NRC conducted an audit of the topical report from March 2023 to August 31, 2024
- 57 audit issues were generated, resulting in supplemental information being submitted by NuScale
- For items not resolved during the audit, 4 RAIs were generated
- All audit items and RAIs are resolved closed
- Staff completed review of the LOCA Topical report and issued an advanced safety evaluation to support today's ACRS Subcommittee meeting

Non-Proprietary

NuScale LOCA Topical Report Review

Conclusion

NRC staff completed the review of "LOSS-OF-COOLANT ACCIDENT EVALUATION MODEL" topical report. Subject to the closure of the open and confirmatory items noted in the draft SER, and, with eleven limitations and conditions identified, the NRC staff finds the following:

- The proposed LOCA analysis methodology is acceptable for meeting the requirements of 10 CFR 50.46 and the associated portions of Appendix K* evaluated in this TR, for evaluation of the ECCS performance in the NuScale NPM-20 for design-basis LOCAs. The proposed LOCA EM is conservative to determine CHF and collapsed liquid level above the reactor core.

- The proposed containment response analysis methodology is conservative and acceptable.

- The proposed NRELAP5 computer code and the NPM-20 model are acceptable to evaluate the MCHFR for IORV and LOCA events.

*Note that certain portions of Appendix K require exemptions as specified by limitation/condition (post-CHF phenomena).

Meeting Title	**Open Session NuScale Subcommittee on Staff's Evaluation of NuScale Standard Design App Chapters 3, 16 and LOCA Evaluation Topical Repo**

Attendees

Michael Snodderly	ACRS (DFO)
Thomas Dashiell	ACRS
Sandra Walker	ACRS
Larry Burkhart	ACRS
R Snuggerud	NuScale
Jim Osborn	NuScale
Rob Meyer	NuScale
Scott Barnes	
Tyler Beck	NuScale
Kyle Hoover	NuScale
Freeda Ahmed	NuScale
Cindy Williams	NuScale
Andrea Torres	ACRS
Dennis Bley	ACRS
Karl Gross	NuScale
Dave Petti	ACRS
Doug B	NuScale
Shandeth Walton	ACRS
Kevin Spencer	NuScale
JJ Utberg	NuScale
Kevin Lynn	NuScale
John Fields	
Wendy Reid	NuScale
Meghan McCloskey	NuScale
James Cordes	Court Reporter
Eric Lantz	NuScale
Yi-Lun Chu	NRR
Craig Harbuck	NRR
Leonard Ward	NRR
Ron Ballinger	ACRS
Erin Blumsack	NuScale
Tom Case	NuScale
Emily Larsen	NuScale
Alina Schiller	NRR
Thomas Griffith - NuScale	NuScale
Mahmoud -MJ- Jardaneh	NRR
Matt Sunseri	ACRS
Andrea Mota	NuScale
Elisa Fairbanks	NuScale
Gary Becker	NuScale
Vicki Bier	ACRS
Steven Bloom	NRR
Ricky Vivanco	NRR

Walt Kirchner	ACRS
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Raul Hernandez	NRR
Warren Erling	NRR
Gordon Curran	NRR
Prosanta Chowdhury	NRR
Thomas Hayden	NRR
Joy Jiang	
Milton Valentin	NRR
Jorge Cintron-Rivera	NRR
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Stacy Joseph	NRR
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Sarah Bristol	NuScale
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Tammy Skov	
Leslie Terry	
Jeff Luitjens	NuScale
Timothy Polich	
John Bozga	
Derek Widmayer	ACRS
Brian Wolf	NuScale
Marie Pohida	NRR
Kris Cummings	NuScale
Sarah Turmero	NuScale
Ben Bristol	NuScale
Ata Istar	NRR
Clint Ashley	NRR
John Honcharik	NRR
Greg Makar	NRR
C Basavarajh	NRR
Sunwoo Park	NRR
Shanlai lu	NRR
Sean Pieta	NRR
Mike Swim	NRR
Rebecca Patton	NRR
Carl Thurston	NRR
Antonio Barrett	NRR
Joshua Kaizer	NRR
Syed Haider	NRR
Justin Coury	RES
Cole Takasugi	RES

Alfred Krall	ERI
Getachew Tesfaye	NRR

Chapter 16 and Technical Specifications	
Noteworthy Changes from DCA to SDA	**Discussion**
Technical Report TR-101310, "US460 Standard Design Approval Technical Specifications Development," Rev 0 describes differences between US600 and US460 Technical Specifications at the time of SDAA submittal.	The reasons for changes are described in general terms, and includes removals, relocations, and new requirements.
LCO 3.1.2 Core reactivity balance surveillance frequency was clarified.	The response to Audit Item A-16.3.1.2-1 revised SR 3.1.2.1 by removing the note associated with adjustment of predicted reactivity values to correspond to measured core reactivity prior to exceeding a fuel burnup of 60 EFPD. NuScale has no basis for the inclusion of this note other than consistency with the Standard Technical Specifications. The note implied that adjustment of predicted reactivity values is prohibited beyond 60 EFPD. There is no restriction on the timing of the revision of predicted reactivity values. The revision to SR 3.1.2.1 also removed a note in the frequency column. The note described when the surveillance is to be performed, and was unnecessary. The Surveillance Frequency Control Program (SFCP) establishes the surveillance frequency.
TS 3.1.9 modified to incorporate additional controls on possible dilution flow paths associated with the Module Heatup System (MHS).	Responses to Audit Items A-16.3.1.9-2 and A-16.3.1.9-3 revised TS 3.1.9 to include: • New LCO related to MHS flow paths • Revision to Mode 3 Applicability to include "with any dilution source flow path not isolated" • Changes to Actions to address new MHS LCO • Changes to SR 3.1.9.5 to clarify verification that MHS flow paths to and from cross-connected systems are isolated. • Supporting Bases changes

	The MHS heats the RCS to assist in developing natural circulation through the core before nuclear heat addition. The MHS is shared among NPMs and, when in service for a module, could represent an inadvertent dilution source for other modules. The revisions to LCO 3.1.9 ensure the modules not being heated by MHS are isolated from the MHS by two closed valves.
TS 3.5.4 modified to address the form of the emergency core cooling system supplemental boron (ESB) pellets and the associated requirements to be specified in the core operating limits report.	Response to Audit Item A-16.3.5.4-1 revised TS 3.5.4 and associated Bases to address the form of boron pellets and the associated requirements to be specified in the core operating limits report. The pellet dissolution rate depends on the geometric form (dimensions and shape) of the boron pellets.
TS 5.5.4, “Steam Generator (SG) Program,” revised to update the tube integrity discussion, plugging criterion and inspection requirements.	To determine an appropriate steam generator tube plugging criterion for the US460 design, NuScale performed a finite element analysis specific to the US460 design. TS 5.5.4 was updated to reflect the analysis, and bracket the tube plugging criterion Revisions to inspection requirements increased inspection frequency and specificity.

Supplement - ACRS – Loss-of-Coolant Accident Evaluation Methodology Topical Report Review Presentation – Closed Session, January 15, 2025

The staff is providing this supplement to highlight differences between the draft Advance Safety Evaluation Report (ASER) for NuScale, LLC. Topical Report "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422, Revision 3 that was submitted to the Advisory Committee for Reactor Safeguards (ACRS) for review on December 14, 2024 and the version that was published on January 7, 2025 (ML24312A002). These differences do not change any of the staff's conclusions. The staff's presentation during the January 15, 2025 ACRS subcommittee's closed session accurately reflected the staff's conclusions in the published version of the ASER.

In summary, the differences include editorial changes, clarifications, and refinement of language in the limitations and conditions. The technical differences are the inclusion of additional supporting evidence of the staff's conclusions, specifically for scaling analysis of the LOCA EM, N-RELAP5 code version use, and containment response analysis methodology (CRAM). The changes are mainly due to additional information submitted by the applicant that supported the staff's conclusions in the draft ASER.

The table below lists the SE sections where changes were made, summary of the changes , and the slide number where this information was presented during the January 15, 2025, closed subcommittee meeting:

SER Section	**Summary of Change**	**Presented on Slide #**
Section 4.4.2, Phenomenon Identification and Ranking Table Rankings	As a result of additional information submitted by NuScale and confirmatory analysis performed by the staff, the staff found the existing PIRT for the in-vessel flow and heat transfer is not impacted by the generation and transport of the small amount of radiolytic gas.	14
Section 4.5.1.5, Helical Coil Steam Generators (HCSG)	The staff confirmed that the DHRS modeling and coupled pool nodalization is sufficient to model the overall decay heat removal responses and heat transfer capability.	15
Section 4.5.1.6, Containment Vessel and Reactor Pool	The staff confirmed that the uncertainty in natural convection heat transfer modeling from the CNV and DHRS to the pool due to thermal stratification would not be safety-significant with respect to the containment pressurization and DHRS capacity.	29
Section 4.6, NRELAP Computer Code	The staff confirmed code update and basemodel version-to-version benchmark results and determined that the code version update and model changes are acceptable and consistent with this NPM methodology.	9

SER Section	**Summary of Change**	**Presented on Slide #**
Section 4.7.5.1, Test Facility	The staff confirmed, based on the results of the various assessment sensitivity studies, the applicant's conclusion that the NRELAP5 model responses are consistent with physics-based expected results and that there are very negligible effects on the event FOMs.	11-12
Section 4.8.2.7, **[[ ]]**	The staff confirmed that heat transfer from the lower head to the reactor pool has a minor impact on the CNV pressure response and that using the **[[ ]]** for modeling heat transfer from the lower hemispherical CNV head does not have any safety-significance with respect to the CNV T/H response.	29
From Section 4.8.3.2.4, Reactor Coolant System Depressurization Scaling	The staff confirmed **[[ ]]** and conservatism. Therefore, the conclusion in the distortion analysis is acceptable.	12
Section 4.8.3.3, Assessment of NuScale Facility Integral Effect Test Data	The staff confirmed that the applicant's extensive assessments in the LTR with these NIST-2 tests, and the code-to-data agreement is excellent for the figures of merit.	12
Section 4.8.3.4, Evaluation of NuScale Integral Effect Tests Distortions and NRELAP5 Scalability	As a result of additional analysis and justification provided by the applicant, the staff confirmed that NIST-2 chronology scaling is maintained as it was in the NIST-1 scaling.	11
Section 7. Limitation and Condition Section 4.5.2, Analysis Setpoints and Trips.	L/C modified to include "*unless the method is followed that is described in section 5.2 of the LOCA EM TR that models the riser level instrument setpoint based on mixture level in the riser, using one of the approaches described in detail in section 5.2 (not including the application – specific alternate approach).*" (This addition is also reflected in Section 4.5.2.). Additionally, L/C #4 and #9 were combined into one L/C.	13