Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title:	Advisory Committee on Reactor Safeguards, Nuscale Design-Centered Committee, Open Session
Location:	teleconference
Date:	Tuesday, April 1, 2025

Work Order No.: NRC-0289 Pages 1-215

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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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TUESDAY

APRIL 1, 2025

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

COMMITTEE MEMBERS:

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MICHAEL SNODDERLY

A-G-E-N-D-A

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

8:30 a.m.

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

I'm Walt Kirchner, Chairman of today's subcommittee meeting. ACRS members in attendance in person are Ron Ballinger, Vicki Bier, who I expect shortly, Craig Harrington, Robert Martin, Scott Palmtag, Dave Petti, Thomas Roberts and myself.

ACRS members in attendance virtually via Teams are Vesna Dimitrijevic, Greg Halnon who will join us in person later and Matt Sunseri.

We have two of our consultants participating virtually via Teams -- Dennis Bley and Stephen Schultz. If I've missed anyone, members or consultants, please speak up now.

Michael Snodderly is the ACRS staff that's the Designated Federal Officer for this meeting. No member conflicts of interest were identified and I also note that we have a quorum.

During today's meeting, the subcommittee will receive a briefing on the staff's evaluation of the NuScale Power, LLC's US460 Standard Design

Approval Application, Chapter 1, Introduction and General Description of the Plan, Chapter 4, Reactor and Chapter 15, Transient and Accident Analysis.

We previously reviewed the certified NuScale US600 Design as documented in our July 29, 2020, letter report on the safety aspects of the NuScale small modular reactor. Like the staff, we are performing a delta review between the two designs including a power upgrade 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 applications which concern safety.

The ACRS was established by statute and governed by the Federal Advisory Committee Act or FACA. The NRC implements FACA in accordance with our regulations. Per these regulations, and the Committee's bylaws, the ACRS speaks only through its published letter reports. All member comments, therefore, should be regarded as only the individual opinion of that member, not a committee decision.

All relevant information related to ACRS activities, such as letters, tools for meeting participation and transcripts are located on the NRC website and can be readily found by About Us ACRS in the search field on NRC's home page.

The ACRS, consistent with the Agency value of public transparency and regulation of nuclear facilities provide opportunity for public input and comment during our proceedings. We have received no written statements or requests to make an oral statement from the public; however, we 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 Sunshine Act. Attendance during the closed portion of the meeting will be limited to NRC staff and its consultants, applicants and those individuals in organizations who entered into an appropriate confidentiality agreement. We will confirm that only eligible individuals are participating in the closed portion of the meeting.

The arcs will gather information, analyze relevant issues and facts and formulate proposed conclusions and recommendation as appropriate for deliberation by the full committee.

A transcript is being kept and will be

posted on our website. When addressing the subcommittee the participants should first identify themselves and speak with sufficient clarity and volume so that they may be readily heard. If you are not speaking, please mute your computer on Teams or by pressing *6 if you're on your phone. Please do not use the Teams chat feature to conduct sidebar discussions related to the presentations, rather limit the use of that function to report IT problems.

For everyone in the room, please put all your electronic devices in silent mode and mute your laptop microphone and speakers. In addition, please keep sidebar discussions in the room to a minimum since our ceiling microphones are live.

For the presenters, welcome back. As you know, these microphones are unidirectional so you'll need to speak directly into the front of the microphone so that our court reporter can identify who you are and maintain his records.

Finally, if you have any feedback for ACRS about today's meeting, we encourage you to fill our public meeting feedback form on the NRC's website. With that, we will now proceed with the meeting and I will ask Michelle Sampson, Director of the Division of New and Renewed Licenses in NRR to make an opening statement. Please, Michelle.

MS. SAMPSON: Thank you. Good morning, Chair Kirchner, members of the ACRS Subcommittee, NuScale representatives, NRC staff and members of the public. My name is Michelle Sampson and I serve as the Director the Division of New and Renewed Licenses in the Office of Nuclear Reactor Regulation.

I would like to begin by expressing my sincere appreciation to the ACRS members and staff for their flexibility in collaboration and accommodating the staggered completion schedule of the Standard Design Approval Application or SDAA chapters. Your support has been instrumental in allowing us to stay on schedule and we look forward to presenting the final three chapters of the NuScale US460 design SDAA today.

As you are aware, NRC staff has been reviewing all chapters and associated topical reports on the SDAA concurrently with completion dates staggered based on chapter complexity and the extent of changes from the previously certified NuScale US600 design. Today, the staff will present their review of the eighth and final group of SDAA chapters, specifically Chapter 1, Introduction and General Description of the Plant, Chapter 4, Reactor and

Chapter 15 Transient and Accident Analyses.

Previously, the staff presented to this subcommittee on chapter 16 of the 19 SDAA chapters as well as the three SDAA topical reports. In today's meeting, the staff will focus on the key differences between the NuScale US600 design certification which was previously reviewed by this committee and approved by the NRC and the NuScale US460 design.

Also, as part of their Chapter 15 presentation today, the staff will present on a differing view regarding the role of EDAS. As you may recall, in Chapter 8 that was presented to the ACRS in November last year, the staff confirmed that EDAS is a non-safety-related system, structure and component and given the augmented quality requirements in place, the staff concluded that classifying EDAS as safety-related is not necessary for adequate protection. The staff's differing view has been captured in a nonconcurrence which is currently under management review. We are considering the function of EDAS as compared to the approved US600 design certification and whether this issue was resolved in the issuance of the design certification in preparing the management response. As this is nearly our final ACRS meeting, I want to confirm that regardless of the outcome of the differing view, we do not expect the decision to impact the overall analysis or design of the NuScale US460 reactor.

Also, I would like to note that our two reactor systems lead reviewers were unexpectedly unable to be here today, so we appreciate Dr. Adam Rau, Dr. Zhian Li, Mr. Josh Miller and Mr. Tom Scarbrough for stepping in to present and answer questions.

Once again, I want to thank the committee for the opportunity to present this important work and we look forward a productive discussion today.

CHAIR KIRCHNER: Thank you, Michelle, and with that, I think we'll turn to NuScale and Tom, right? You're going to lead it off?

MR. GRIFFITH: That's correct. Thank you.

CHAIR KIRCHNER: Please go ahead.

MR. GRIFFITH: Good morning, ACRS members, NRC staff, the public and NuScale. Today, we present on the final chapters for the SDAA at ACRS Subcommittee Chapters 1, 4 and 5.

This is a huge milestone for NuScale to reach and we're very excited to be at this point in the review. There have been countless hours spent by both the staff and NuScale reviewing the design and answering questions. I appreciate the thorough review by the NRC and all the effort by the NRC staff members, by the ACRS in reaching this moment today.

I'd like to start a little bit with my background. I am Thomas Griffith the project manager and licensing manager for the US460 design. I have 15 years of experience in nuclear power. I have experience in safety analysis, probabilistic risk assessment, held a senior reactor operator license and addressed a nuclear power station, worked as the I&C shop manager at Dresden and ultimately have taken over as the licensing manager and project manager for the US460 design.

Again, I really appreciate all of the efforts by the staff and the ACRS in getting to this point and look forward to the presentation. With that, I'll turn it over to Tyler Beck.

MR. BECK: This is Tyler Beck presenting virtually and we can go to the next slide. Before we get started with the presentation materials, I wanted to recognize and acknowledge the DOE for their support and award of which this work is supported by. Next slide.

As I said, my name is Tyler Beck and I am the licensing engineer for Chapter 1, which is

Introduction and General Information. I've been with NuScale for about two and a half years and prior to my time in NuScale, I was the reactor assistant engineer at the NRC in the General Communications and Operating Experience Branch. Next slide.

We've got a short time frame to get through Chapters 1 and 4 and so, we're going to go ahead and get right into the presentation materials. Similar to the previous presentations, we're going to focus on changes from the DCA and with that, we're just going to go through the Chapter 1 sections one by one.

For Section 1.1, Introduction, similar to past FSAR content, we've optimized the FSAR concept to remove what is redundant from other sections and so that type of change exists for many of the Chapter 1 sections. The other big change for Section 1.1 is a discussion of multi-module considerations that exists now. In the DCA, that discussion was in SR-Chapter 21 and we relocated it to Chapter 1.

Section 1.2 is General Plant Description and you've got high level descriptions of various SSC and plant figures. You'll notice that that content is changed to reflect the US460 standard design. An example would be that the plant overview figures will show a one-turbine building instead of two-turbine buildings.

For Section 1.3, Comparison with Other Facilities, it's similar to Section 1.2. We have high level things describing the plant, so you'll see that it now reflects, for example, 250 megawatts thermal for the power output of the module.

Section 1.4, Identification of Agents and Contractors. This section is unchanged from the DCA and it is just a COL item. Next slide.

Section 1.5, Requirements for Additional Technical Information. It talks about some verification and confirmation tests of unique design features, for example, ECCS supplemental boron. For changes from the DCA, there are two noteworthy test program additions. We have perform boron dissolution testing that has been described before, I believe, for the XPC topical and that was performed at the NIST facility, so that's one new test program. The other is we've performed additional ECCS valve functional testing that was performed with fully prototypic valve assemblies and that is the other additional test program.

For Section 1.6, Material Referenced, this section is where the incorporation by reference high impact technical issue lived and ultimately, we resolved that issue with the NRC staff during the audit and now in that section, the majority of Technical and Topical Reports are incorporated by reference.

Section 1.7, Drawings and Other Detailed Information. This section has no significant changes from the DCA.

Section 1.8, Interfaces with the Standard Design. The main change from the DCA is that we have removed conceptual design information that was listed in the DCA. An example would have been the potable water system, this isn't listed because, I believe, mainly to do with requirements. For DCA, you have to list conceptual design information. Next slide.

Section 1.9 is Conformance of Regulatory Criteria and this talks about conformance with various reg guides, SRP criteria, DSRS criteria. There's not enough time in this presentation to be able to go through every change in that section, but to give a couple of examples of how conformance with regulatory criteria has changed, for Reg Guide 1.7, which is Control of Combustible Gas Concentrations and Containment, you'll now see that the verbiage reflects our PAR and whereas in the DCA, there was not a specific control system.

In DSRS 5.3.1, there's an acceptance criteria that pertains to the reactor crush vessel material surveillance program and due to the design change to austenitic stainless steel and the lower RPV and the associated exemption. Now for the SDAA that applicable acceptance criteria is no longer applicable, whereas in the DCA, it was applicable.

Lastly, for Chapter 1, is Section 1.10, Sites with Multiple Nuclear Power Plants. This section is not changed from the DCA and it's just a COL item, I believe. That is all of the prepared content for Chapter 1 if there are any questions.

CHAIR KIRCHNER: Members? Virtual members?

MEMBER BALLINGER: Yes, this is Ron Ballinger. I have a question related to the austenitic stainless steel, but I have to reserve it for the closed session.

CHAIR KIRCHNER: Okay. Tyler, since this is one of the last presentations, do you want to highlight one significant change from the DCA? You've addressed in the SDAAs, you're not taking the exemption or seeking an exemption on GDC27. Do you want to address that at all?

MR. BECK: I don't know, I think that's previously been covered, but it may not be. I'm not sure if we have anything planned for our Chapter 15 presentation, but ultimately in the DCA, if you'll look at SR Section 3.1, for GDC27. In the DCA, there was a potential return to power and in the SDAA design with the addition of the ECCS supplemental boron, there is no return to power. I believe that's the gist of that change from the designs.

CHAIR KIRCHNER: Okay, thank you. I just wanted you to put that on record. Thank you.

MR. GRIFFITH: This is Thomas Griffith, NuScale. I'd like to highlight a little more than that on the ESB change. Specifically, that we heard the feedback on the design with the US600 and ultimately the change adding ESB makes the design safer and that ultimately led to the extended Passive Cooling Topical Report that exists in Chapter 15, which will have some follow up slides on today, to discuss. Overall, it's a substantial improvement to the safety of the design.

CHAIR KIRCHNER: Thank you. Sarah, we're turning to you next.

MS. TURMERO: Yes.

CHAIR KIRCHNER: Go ahead.

MS. TURMERO: Good morning. If we can go to the next slide for Chapter 4. My name is Sarah Turmero. I'm a licensing engineer covering topics in Chapter 4, 9 and 15 in the related Topical Reports.

I've been with NuScale for about two and a half years and was previously a reactor engineer at Waterford 3. Today, I have Ken Rooks and Allyson Callaway from fuels engineering to support Chapter 4 questions. Next slide.

Chapter 4 consists of the fuel, nuclear, thermal and hydraulic design including materials that are interior to the reactor and the functional control design of the control rod drive system. Next slide.

A summary description in Section 4.1 was simplified from the DCA to the SDA by removing information that was repeated in subsequent questions, so this slide provides a guide of where that information lives in the SDA. There were no audit questions or RAIs for Section 4.1.

For the fuel system design, the majority of the fuel design remains the same and the control rod design remains the same. So, for the fuel, things like fuel rod array, rod assembly, spacer grids, active fuel lengths remain the same from the DCA.

There were administrative changes to

Section 4.2 that incorporated the classification of the SSCs and removal of redundant information. The fuel rod length increased by about 1 inch in the upper portion of the fuel pin and that is related to an anti struggle feature that was added. The core loading change associated with the power uprate and the faulted limits used in the fuel NCRA technical report were updated to use the ASME boiler pressure vessel code based on the boiler pressure vessel code and previously the limits were derived from the Framatome Topical Report.

The Fuel and Structural Response Methodology Topical Report was incorporated into the SDA that was previously approved.

There were 21 audit questions and no RAIs and 11 of those audit questions were on the fuel NCRA technical report. Next slide.

For Nuclear Design in Section 4.3, the fuel pellet density changed and other parameters related to the power uprate change, such as the linear heat rate, peaking factors, cycle length. The emergency supplemental boron was added for GDC27 compliance for shutdown cooling considerations. Calculations such as the vessel influence were revised to use the US460 design inputs but those methodologies remain the same.

There were 29 audit questions resolved and one RAI that was resolved. This RAI requested a limiting condition for operation on the heat flux hot channel factor or Fq and Fq does not require an LCO per 10 CFR 50.36 because it's not used as a direct input into safety analyses, but NuScale does continue to use the approved methodology from the DCA to develop and analyze bounding axial power shapes and the implementation of a conservative enthalpy rise hot channel factor or F delta H value and those two factors influenced Fq.

The figure on the left is the DCA and the figure on the right is for SDA and shows that they could have come from the same work.

MEMBER MARTIN: This is Member Martin. Just a point of clarification, the pellet density change, you're getting the fuel from the Framatome, that's the plan, this is just their number?

MS. TURMERO: That's correct.

MEMBER MARTIN: Okay, so 96 was something old and 6.5 is just what they're, I think, they're advertising today?

MS. TURMERO: That's correct.

MEMBER MARTIN: Okay, nothing to do with

power uprate and all that other stuff, it's just maybe, yeah, okay. Thank you.

MS. TURMERO: Next slide. This slide provides the example nuclear design parameter comparisons and things such as the core average layer heat rate, heat flux hot channel factor, limiting shut down margin available and cycle length have changed between the MPM-160 and the MPM-20.

MEMBER BALLINGER: This is Ron Ballinger again. If it's not proprietary, what is the peak linear power? In other words, the average is two and one-half or two point whatever, but what is the peak? Didn't you say --

MS. TURMERO: I think that's proprietary.

MEMBER BALLINGER: Well, say that in the closed session.

MR. ROOKS: This is Ken Rooks for NuScale, it's roughly the product of the 3.9 and 2.196 on there, so roughly --

(Simultaneous speaking.)

MEMBER BALLINGER: That's what I kind of figured. Okay.

MR. ROOKS: Yep.

MS. TURMERO: Next slide. For the thermal and hydraulic design, the Approved Statistical

Subchannel Analysis Method Topical Report was implemented and as a result, a new technical report was provided to outline the statistical uncertainties to satisfy the limitation and condition from the Subchannel Topical Report. As discussed the ACRS Subcommittee for the Loss of Coolant Accident Topical Report, a new critical heat flux correlation was implemented, NSPN-1, for rapid depressurization events. The analytic limit for NSPN-1 is 1.20 and the analytical limit for an NSP-4 is 1.43. A flow reduction that's applied to the limiting fuel assembly in the subchannel analysis is 20 percent for the SDA whereas the DCA that was 15 percent.

There were three audit questions resolved and no RAIs. There are a few additional comparisons of reactor parameters that have changed between the DCA and SDA. Next slide.

This is a comparison of the analytical design operating limits between the DCA and the SDA and note that the minimum temperature for criticality has decreased and overall the operating range for temperature has shifted. Next slide.

This shows the thermal margin limit math comparison between the DCA and SDA. Note, that for the SDA, the X axis starts at 20 percent. One item to note is that the lines for the minimum and maximum flow relative to the core inlet temperature have decreased for the SDA. Next slide.

Section 4.5 is specific to the reactor vessel internals and control rod drive system materials. For the CRDM, the changes are related to be design standards that are applied and the additional alloy option to improve strength. For the reactor pressure vessel internals, there are not significant changes from the DCA to the SDA and the materials are austenitic stainless steel of various grades, types and classes. Next slide.

Section 4.6, Functional Design of the Control Rod Drive System. The mechanical changes were covered as part of FSAR Section 3.9.4. Two changes to note are the pressure housing is now bolted instead of welded to the reactor pressure vessel head and the addition of the rod hold out device was added to facilitate storage of the drive shaft in the upper NPM during refueling outages.

The safety function of the CRDM remains the same between the DCA and the SCA. The release of the control rod assembly is during a reactor trip and to maintain the pressure boundary of the reactor pressure vessel.

There were three audit questions resolved and no RAIs. That is all I have for Chapter 4.

CHAIR KIRCHNER: Thank you, Sarah. Members? Don, do you have any comments or questions?

PARTICIPANT: I reviewed the chapter, but that looks pretty standard, so.

CHAIR KIRCHNER: Okay. Members? Online, any comments, questions? Okay. Thank you very much. With that, now we will turn to the staff, so those online, we'll just have a pause here for a few moments and we'll have the staff present their review. Getachew, are you ready?

MR. TESFAYE: Yes. Put the slide on. Good morning, my name is Getachew Tesfaye, I'm the lead project manager for NuScale Standard Design Approval Application, SDAA, Review.

Thank you for the opportunity to present the NRC staff's safety evaluation of NuScale's FSAR Chapter 1, which is Introduction and General Description of the Plans. Chapter 1 is a high level summary of the safety evaluation of NuScale's certified design discussed in detail in the remaining parts of the SDAA.

Before I present Chapter 1, safety evaluation differences between the NuScale certified design, DCA, and the SDAA, for the record, I would like to recap the US460 SDAA review activities. Next slide.

The US460 SDAA review process began with the pre-application activities in 2019 starting with submittal of the Regulatory Engagement Plan Revision 0, followed by a public meeting. Since then, the process has included submissions of eight Topical Reports during the pre-application phase and completion of a staged SDAA submittal in January 2023 that included four new Topical Reports.

Key milestones include the NRC staff issued its acceptance review results and request for supplemental information, RSI, on March 17, 2023. A detailed safety evaluation of parts of the application not impacted by the RSI began in March 2023, that includes Chapter 1. After receiving the supplemental information on July 14 and July 17, 2023, the NRC issued a docketing letter on July 31, 2023, which outlined a four phase, 24-month review schedule. Next slide.

Our approach for the SDAA follows a four phase process contrasting with the six phase review used for the NuScale Design Certification Application. The key enhancement in our approach extended the audit process using NuScale's electronic reading room, eRR. That facilitated efficient access to calculations and supporting documents aligned for a more streamlined review and also a reduction in the number of requests for additional information enhancing efficiency while maintaining a regulatory and safety focus.

For the most part, this approach was a success that allowed the staff to conduct a thorough review under a compressed schedule.

MEMBER MARTIN: Getachew?

MR. TESFAYE: Yes?

MEMBER MARTIN: It's Member Martin again. I'm kind of curious, the NuScale scale design specific review standard, of course, was established for US600, I don't believe -- were there any updates for that?

MR. TESFAYE: No.

MEMBER MARTIN: So, it was still used kind of to guide the review --

MR. TESFAYE: For the SDAA.

MEMBER MARTIN: And you found it applicable, continued to be applicable across all chapters?

MR. TESFAYE: Yes, correct.

MEMBER MARTIN: Thank you.

MR. TESFAYE: Across all applicable

chapters.

MEMBER MARTIN: Of course. Thank you.

MR. TESFAYE: At this point, I would like to take this opportunity on behalf of the project team, to extend our sincerest thanks to ACRS members and staff for the exceptional cooperation and the flexibility throughout this process. Your willingness to adapt has played a crucial role in enabling us to maintain our aggressive schedule. Thank you helping us move so efficiently. Now, I'll go to my chapter, Chapter 1.

NuScale submitted SDAA Chapter 1, Revision 0 on December 31, 2022 and Revision 1 on October 31, 2023. Our regulatory audit of Chapter 1 was conducted between March and August of 2023, resulting in one audit issue, which was successfully resolved during the audit. More importantly, no RAIs resulted for this review.

The draft SE provided to ACRS on March 4, 2025, was updated to include supplemental information submitted by NuScale on March 17, 2025 and is reflected in the SC submitted on March 25, 2025. The March 17 submittal is a Chapter 1 Revision 1 to Revision 2 snapshot, which allowed the SC to reflect the content of Revision 2 that NuScale plans to officially submit this spring. Next slide.

Now, notable differences between the NuScale DCA and the SDAA FSAR include the elimination as discussed earlier by NuScale, the elimination of Chapter 20, Mitigation of Beyond-Design Basis Events, and Chapter 21, Multi-Module Design Consideration from SDAA.

For Chapter 20, NuScale has instead submitted a January Topical Report to be used by future applicants that will use the NuScale design. That Topical Report was presented to this committee and is currently under administrative review before the final SDAA is issued.

As discussed earlier by NuScale, Chapter 21 was removed as most of the content of DCA Chapter 21 is addressed in other chapters of the SDAA and also in Chapter 1.

The second difference is unlike the DCA, SDAA did not use their approved Topical Report on Safety Classification of Passive Nuclear Power Plant Electrical Systems. As a result, the limitations and conditions that were evaluated in the DCA Chapter 1, SE is not applicable for SDAA.

Two exemption requests in the DCA were not requested for the SDAA. For DCA, NuScale requested an exemption from 10 CFR 50.54 regarding minimum licensed operator staffing requirements. For SDAA, as discussed in Chapter 18 of the SE, all applicants referenced in the US460 Standard Design Approval will seek an exemption from 10 CFR 50.54 using NRC approved Topical Report, NuScale Control Room Staffing. As you recall, that Topical Report approved the decrease in the operator licensing the control room from six to three.

Another exemption that was not included in the SDAA is the one that was discussed earlier, which is an exemption request from GDC27. That was requested in the DCA because US600's design does not rely on poison additions through ECCS for the SDAA, the ECCS supplemental boron function provides a passive source of boron that compensates for the positive reactivity added by the cool down. This is discussed in Chapter 15 of the SE. Next slide, please.

Continuing on the differences, three new exemption requests were added in the SDAA that were not in the DCA. As discussed previously during the Chapter 5 presentation, NuScale requested an exemption from 10 CRF 50.60 which required that light water reactors meet the fracture toughness and material surveillance program requirements of the reactor coolant pressure boundary set forth in 10 CFR 50 Appendices G and H.

Also, as discussed during the Chapter 5 presentation, NuScale requested an exemption from 10 CFR 50.61 which provides fracture toughness requirements to protect against pressurized thermal shock PTS events. Also, a new exemption request as part of the resolution of HITI number two and number 10 regarding LOCA break spectrum. NuScale requested an exemption from the requirements of 10 CFR 50.46(a)(1)(I) that requires the most severe postulated loss of coolant accidents are calculated. This will be discussed the Chapter 15 presentation later on today.

One other notable change is the staff's evaluation of the exemption requests for GDC19, in the DCA it was in Chapter 1 and now it's included in Chapter 6(s)(e).

Lastly, as discussed, as part of the briefing of HITIs to this committee, for the SDAA the only applicable sections of topical reports and technical reports are incorporated by reference or IBR. For the DCA, all sections of topical and technical reports were incorporated by reference.

This has made the SDAA FSAR manageable giving future US460 design license orders flexibility to make minor modifications to these reports using appropriate regulatory process.

In conclusion, the only significant evaluation that was done in SDAA in Chapter 1 is the tables section, Section 1.8, the IBR, and that adequately addresses the applicable regulatory requirements of the content of FSAR. Chapter 1 does not include a safety finding as SDAA safety findings are contained in Chapters 2 and 19.

That concludes my presentation. I'll be happy to address any questions you may have at this point.

CHAIR KIRCHNER: Members, any questions of the staff's review of Chapter 1? Online, any questions or comments? Thank you.

MR. TESFAYE: Thank you.

CHAIR KIRCHNER: Let's move on to Chapter 4.

MS. JOSEPH: Thanks, Getachew. My name is Stacy Joseph and I'm a senior project manager in the Office of New Reactors and I'm the PM for both Chapters 4 and 15 today.

Like the other chapters that have already been presented to the ACRS, the staff began their review of Revision 0 and then Revision 1 of Chapter 4 of the SDAA with the start of a regulatory audit in March 2023. Over the next year and a half, the staff generated 76 audit issues and in the end, issued one RAI, which has since been resolved.

The staff completed the Chapter 4 review and issued an advanced safety evaluation. There has been one significant change in the staff's evaluation between early March when the staff provided the draft SER to ACRS and last week, when the draft SER was submitted and made public.

Between revisions of the SER provided to the members, the staff was able to close the one Chapter 4 open item. Section 4.3 was revised to evaluate the RAI response provided by NuScale and to provide an assessment for why an LCO is not needed for the heat flux hot channel factor Fq. This will be discussed in more detail later in the presentation.

There are a number of technical staff who contributed a great amount of time to the review of Chapter 4 and completion of the safety evaluation. I'll leave this slide up to display the team names while I ask the presenters here today to go through and introduce themselves. Adam?

DR. RAU: My name is Adam Rau. I'm a technical reviewer in the Nuclear Methods Systems, New Reactor branch.

MS. SUGRUE: Hi, Rosie Sugrue. I'm also a technical reviewer in the Reactor Systems branch.

DR. LI: My name is Zhian Li. I'm a senior nuclear engineer in the same branch.

MR. HONCHARIK: Hello, my name is John Honcharik, senior materials engineer in the Division of New and Renewed Licensing.

MS. JOSEPH: There are six sections in Chapter 4. The staff will be focusing their presentations today on those sections of the FSAR that has significant differences between the DCA and the SDAA.

SDAA Section 4.1, Summary Description and SDAA Section 4.6 on the Functional Design of the Control Rod Drive System do not have significant differences from the DCA, so we will not have separate presentations on those sections. For these sections, the conclusions of the SDAA were the same as the DCA.

In addition to the discussion during the open session, there will be an additional presentation during the closed session on Section 4.2 related to fuel seismic analysis and thermalomechanical performance.

We will start the presentation on Section 4.2, Fuel System Design and I'll turn it over to Dr. Adam Rau.

DR. RAU: Thanks, Stacy. This is Adam Rau. I'll quickly be going over the staff's review of FSAR Section 4.2 covering Fuel System Design and Fuel Thermal Mechanical Analysis. The staff's review of this section was supported by the help of engineers from the Pacific Northwest National Labs, who are in the audience here today. The NPM-20 uses the new fuel HTT2 fuel design, which is very similar to the fuel used in the certified NPM-160 design.

The NPM-20 FSAR implements the Topical Report TR-108-553, which was approved prior to the submission of the Standard Design Approval Application. This topical report extends the applicability of fuel thermomechanical and fuel seismic methodologies used in the NPM-160 Design Certification Application to the upgraded NPM-20 conditions.

NuScale's results are summarized in the technical report that is incorporated by reference into Chapter 4 of the FSAR. Analysts at PNNL performed confirmatory analysis of NuScale's current calculations using the FAST code. Confirmatory analysis was performed for a specified criteria with the FAST code. In general, the analysis showed good agreement with the applicant's analysis.

During the review, staff also considered the effect of the uprate on other fuel through mechanical analysis and audited calculations. Reviewers did note a change to some cladding stress intensity limits which will be discussed in additional detail in the closed session.

Additionally, the fuel seismic analysis was revised for the NPM-20 due to changes in the building footprint, pool dimensions and levels and other changes that led to revised floor plate movements. PNNL analysis also performed LS-DYNA confirmatory analysis of the fuel's seismic evaluation and found agreement in terms of load and overall margin.

With that, I will pass it onto to Zhian Li to discuss --

MEMBER MARTIN: Just a quick question. When you use a term like good, good results, good comparison it's obviously bit qualitative and sounds subjective. Could you just briefly say what you're looking for, maybe do more detail? Are we talking like conservative? Are you looking at best estimate to best estimate, best estimate to EM? What's behind good?

DR. RAU: It's varying between -- for different cases. I think PNNL folks could speak to some of the details, but in some cases, they saw agreement that they were able to replicate the calculation and just one for one support it and confirm it. In some cases, their calculation is able to confirm that NuScale's calculations and margins to the acceptance criteria.

DR. LI: Good morning, ACRS members. Good morning, Chairman. My name is Zhian Li, again I'm here to present our review of the Nuclear Design for the NPM-20 reactors.

Together with me, Dr. Adam Rau, we reviewed the design and the FSAR and also we have support from the Office of Research with Dr. Andy Bielen and Dr. Nate Harrison. Nate is not here, but Andy is here and thank you for the support.

Compared with NPM-160 design, the NPM-20, design as we all know, has the higher power density and then some slight change on the fuel allotment and then, of course, you have the linear power generation rate which also changed.

One of the things we clarified with NuScale during our review is the use of actual low enriched or natural uranium blanket fuel. In the FSAR, it says they may have variation of enrichment in the axial direction. We clarified that with NuScale because we see all the safety analyses are based on a uniform axial enrichment. If you have with blanket at top, bottom or variation, your safety analysis would have to consider this potential impact. We clarified with NuScale there will be no blanket design for today.

Also, there is the added new supplemental boron system that is for criticality safety during the heat flux or boron dilution process which will be discussed later on in more detail.

Another item that's not described is there as a COL item which required the applicant or the future COL applicant to analyze non-baseload case design for now, it's for baseload only because there is some implication, for example, the fuel burnout control rod worked and the peaking factor, these all come into play if you have a non-baseload. Next slide, please.

MEMBER PALMTAG: This is Scott Palmtag. I just have a question on the blankets.

DR. LI: Yes?

MEMBER PALMTAG: My understanding of reading it too was the equilibrium cycle did have blankets, but is there now a restriction that they can't have blankets or is this something they can't have in the future?

DR. LI: Well, we cannot say they cannot have it in the future, but the current design does not have blankets.

MEMBER PALMTAG: You mean the equilibrium design?

DR. LI: Equilibrium and the FSAR from the initial set. All the way up, they provided these cycles design.

MEMBER PALMTAG: Okay, so it's just the equilibrium cycle does not have blankets, but they can have blankets in the future?

DR. LI: Yes, I guess that's where they would have to go to through their design change of 50.59 through the regulatory process.

MEMBER PALMTAG: Thank you.

DR. LI: Thank you. So, we reviewed and audited the updated calculation. We looked at the normalized power, assembly pin power, axial power distribution and also we looked at the control rod work and the potential control rod depletion and include the integral control rod with the differential control rod and yeah. So, we did our review and the conclusion from NuScale -- and the control rod worth loss was limited by the design specification from Framatome, which is the control rod vendor. Also, we reviewed the shutdown margin and that's presented and also long term shutdown margin that will be discussed in Chapter 15.05.

We looked at also Doppler moderator temperature and the power deficient coefficient and then some of the parameters have been confirmed by our research folks and those marked with stars are the parameters we did confirmatory analysis.

Also, the review team reviewed the fluence calculation on the reactor vessel. This is a pretty good review and NuScale basically used the MCNP computer code which is widely used for fluence calculation and the NF4B7 cross section, that's the state of the art cross section library to our knowledge. Next, please.

One of the challenging issues we have been looking through is about the use of Fq, which is the heat flux hidden factor and the NuScale design does not include the heat flux hot channel factor in their tech spec. We questioned what's the reason because typically the current license, the PWR, all use Fq as their limit in the tech spec. So then we asked NuScale to provide justification why this design is special. NuScale basically said they do not use this as input for the LOCA analysis and I think that's the fact. Also NuScale provides some calculations to show the linear power generation rate is lower, much lower than the current operating reactor fuel and the fuel is designed with the same standard for the current operating reactors, so therefore, based on this effect, the staff found this acceptable and not including Fq in the tech spec. But also based on our review, we determined this is acceptable for the baseload application if, in the future, there's non-baseload case and all the factors have to be considered.

Based on our review, I think the staff concluded there is a reasonable assurance the Nuclear Design meets the design criteria and meets the regulatory requirement. On this basis, we concluded our review is complete. Thank you. Any questions?

MEMBER MARTIN: This is Member Martin. This is linked to the LCO question. We all appreciate the margin that is there, given maybe a hypothetical, I might imagine there might be some, you know, other LCO, power-related LCO, that might actually come into play well before, maybe not power-related, but operational-related tech spec or LCO that would come into play well before any concern with Fq. Did you go through that thought process or did you just kind of take this kind of more on existing margin? I mean the heat transfer and CHF are not just dependent on power, you know, this is a different reactor and its flow is considered.

But, like I said, I think we acknowledge this large margin, but I would also expect that something else might come into play that would otherwise prevent operation much beyond where we're at, correct? And did that come into the thought process?

DR. LI: Yes it did actually. We discussed with NuScale about this aspect in NuScale's, you know, kind of rationale was the safety in ER would have to play first after damage and it did get in, but our decision is primarily on the low part average end rate. This is consistent with the DCA or NPM-160 design, to which the staff requested NuScale to provide justification for not including Fq in the tech spec and NuScale's response to that at the time was there's a very low power generation rate compared with up in the reactor.

MEMBER MARTIN: Okay, yeah, you wouldn't want to draw conclusions based on work done by Framatome and just because you're using the same fuel product, it's a different plant and there would be different conditions for its operational ranges and stuff, but I wanted to get you to say it.

DR. LI: Thank you.

MEMBER MARTIN: Thank you.

MR. SUGRUE: This is Rosie Sugrue. I'll briefly talk about 4.4 which is the Thermal-Hydraulic design.

The two main significant differences between the DCA and the SDA were the statistical CHF analysis limit. This was applicable to the NSPN-4 correlation which is used for VIPRE subchannel analysis. They had a new CHF correlation, NSPN-1, which the development, range of applicability and so on was all approved in the LOCA topical. Move onto the next.

Our main review items were those two items, the subchannel analysis, the CHF and statistical analytic limit and then the new CHF correlation NSPN-1. We took a look at the bypass flow calculations also. Those were all accounted for in subchannel analysis. They really had a minimal impact on the overall results.

We also took a look at the effects of crud. We just noticed some minor differences in the wording, but it turns out that there wasn't much difference from the DCA.

MR. HONCHARIK: Hi, I'm John Honcharik again. I'm going to talk about the fracturing materials, specifically with the control rod drive mechanisms from materials.

There was basically one significant difference and that was with the CRDs are now bolted instead of welded to the upper head. In addition, these bolted connections consist of stainless steel threaded inserts in the reactor pressure vessel head and have a seal weld applied to it. That's to mitigate any fluoridated water from contacting the alloy steel upper head. Degradation of that bolt connection would include in the stainless thread insert and also go into the alloy steel vessel head. This degradation could impact the CRD and its safety function.

There is, per the ASME code, to visually inspect these, but these but these CRDMs are not routinely disassembled to allow the inspections to monitor this degradation underneath. Next slide.

To monitor this degradation, at the bolted connection, the applicant proposed to perform an augmented VT-1 visual examination of threaded inserts and its seal welds on other ASME class I components that are actually routinely disassembled, such as the steam generator, Feedwater Plenum Access Covers and so on. The augmented inspection plan will be able to detect defects which would indicate that there's an underlying degradation in a reactor vessel's alloy steel and could compromise the bolted connection. Inspection of these areas would basically provide a statistically significant number of threaded inserts and seal welds to provide an adequate assurance of the integrity of these threaded inserts and seal welds.

If defects or corrosion are found in these areas, the threaded inserts, it would be expanded then to the upper head including the CRDMs and other areas, such as the RRV valves and RVV valves in order to verify the integrity of these threaded inserts and the seal welds.

With that, staff found that this provided adequate assurance of the integrity of these threaded inserts and its seal welds. That's the conclusion.

CHAIR KIRCHNER: So, John, what do you think? That's a pretty busy space in the top of the reactor vessel to start looking to do a visual inspection from the access covers. What's involved in that visual inspection? They'll actually have a probe that goes in somehow and allows them to make a detailed reconnaissance and inspect, say the central control rod assembly or the inner assemblies which wouldn't be visually accessible from the access covers.

MR. HONCHARIK: Yeah, I think right now, because you're talking about the CRDs on top?

CHAIR KIRCHNER: Yeah.

MR. HONCHARIK: Where they're bolted in, you know, if they do have to remove them for whatever reason, they will do an inspection which would be a VT-1.

CHAIR KIRCHNER: Right.

MR. HONCHARIK: They could do it manually or with robotics as long as they have the procedure qualified to the ASME code and such, but I think the issue --

(Simultaneous speaking.)

CHAIR KIRCHNER: But do we routinely expect them to disconnect those --

MR. HONCHARIK: They really would, right and that was the whole issue --

CHAIR KIRCHNER: Yeah.

MR. HONCHARIK: We wouldn't know because like you said, it's very busy up there. There's a lot of bolted connections up there and before it was just welded.

CHAIR KIRCHNER: Just welded, yeah.

MR. HONCHARIK: Now, since they do have to do a lot of disconnecting of bolting, especially for steam generator and pressurizer, they use the same exact inserts. What they're going to do is, they have to inspect any time they disassemble one of those bolted connections, they'll have to do that visual inspection of those threaded inserts and seal welds. If they do find stuff there, then they have to have a plan to go and expand it to the upper head and look at X number and if they do find anything more in there, they have to further expand it to more of those.

CHAIR KIRCHNER: Thank you.

MEMBER HARRINGTON: This is Craig Harrington. The bolted connections for the CRDs, what kind of a gasket or an O-ring, what's the sealing mechanism between the CRD and the insert?

MR. HONCHARIK: Oh, I can't remember, but

I think it's more of a fit with the gasket. I'm not sure if it's a metallic gasket, but I think we weren't really looking at that part because if it's leaking that's going to be found out through the leak detection system. What we're looking at is the threaded insert basically provides the barrier to the alloy steel, so that's what we were concentrating on because that would be the point of the weakest link.

MEMBER HARRINGTON: Okay, thank you.

MEMBER BALLINGER: This is Ron Ballinger. I hate to bring up ancient history, but for Davis-Besse, the leakage which, there were a lot of other complications, but it never exceeded the unidentified leakage limit. So, in this case, you've got flanges on top of the head, the same kind of arrangement and in theory, you could get leakage up there which would be below the unidentified leakage limit, but which you would find instantly with a visual inspection.

CHAIR KIRCHNER: They're going to try and keep this at a vacuum --

MEMBER BALLINGER: Yeah.

CHAIR KIRCHNER: And contain it, so --

(Simultaneous speaking.)

MEMBER BALLINGER: So, they'll see

something --

CHAIR KIRCHNER: That's different --

MEMBER BALLINGER: Yeah.

(Simultaneous speaking.)

CHAIR KIRCHNER: Than having an allowable tech spec leak rate and be able to keep the vacuum.

MEMBER HARRINGTON: And the vessel is clad on the inside and outside, so you don't have the corrosive concern that you had at Davis-Besse.

CHAIR KIRCHNER: And your bolted CRD connections, they would leak a lot.

MEMBER BALLINGER: Notorious.

CHAIR KIRCHNER: That was the genesis of the question. We can pursue it with NuScale. Thanks.

MR. HONCHARIK: Thank you.

MS. JOSEPH: All right, thanks, John. This is Stacy Joseph again. After completing the review of Sections 4.2 through 4.6 of the NuScale SDAA FSAR, the staff found that the applicant provided sufficient information to support the staff's safety finding and that all applicable regulatory requirements were adequately addressed.

That concludes the staff's presentation on Chapter 4. Are there any additional questions for the staff?

CHAIR KIRCHNER: Scott, have you had any further questions for staff?

MEMBER PALMTAG: No. This is Scott Palmtag. I did a pretty thorough review of Chapter 4 and it looks pretty standard.

CHAIR KIRCHNER: Thank you.

MEMBER MARTIN: I just had one.

CHAIR KIRCHNER: Yeah, go ahead.

MEMBER MARTIN: Curious question in regard to Section 4.4 and the mention of crud. Is the intent to kind of generically address crud with the fuel or was it just that they had a method to address it down the road when they are operating and this becomes an issue?

MS. JOSEPH: Yeah, we just wanted to make sure that it was included in their assumption and their correlations.

MEMBER MARTIN: Okay, it's not something that they're trying to generically address for --

MS. JOSEPH: Yeah.

MEMBER MARTIN: Way down the road. Okay.

CHAIR KIRCHNER: Okay, I sense everyone is saving questions for the closed session. At this point, we're ahead of schedule. Rather than embark on Chapter 15, which is much more material, we'll take an early break and reconvene at 9:55 Eastern Time. We are in recess for 15 minutes. Thank you, everyone.

(Whereupon, the above-entitled matter went off the record at 9:41 a.m. and resumed at 9:55 a.m.)

MEMBER HALNON: Okay, we're going to start back up again on Chapter 15. My name is Greg Halnon, Vice Chair of the Committee, and just came from online to fill in for Walt Kirchner, who's had to excuse himself for a few minutes.

So, turn it over to NuScale to start the Chapter 15 presentation.

MR. LYNN: Good morning, my name is Kevin Lynn, I'm a Licensing Engineer with NuScale and have been for the past three and a half years. Prior to that, I have experience in operating plant licensing, Part 52 new plant licensing for a different design center, and Part 54, reactor license renewal for operating plants.

And I'll allow my colleagues here to introduce themselves as well.

MS. MCCLOSKEY: I'm Meghan McCloskey, I'm a Thermal Hydraulic Engineer with NuScale Power. I've been in the industry for about 18 years, focused on safety analysis, thermal hydraulic methodology development and implementation. Ten years at NuScale and Westinghouse prior to that.

MR. BRISTOL: This is Ben Bristol. I've been with NuScale for about 13 years. I'm working in safety analysis and system duct thermal hydraulics.

MR. IRVING: Thomas Irving, the licensing manager for the US460 Design and Project Manager of the US460 design.

MR. LYNN: Thank you, next slide, please.

The agenda today in the open session for Chapter 15 is we will start with a summary of the review and the current status. Then we'll provide some overview of the analysis results, focusing on primary and secondary pressure, MCHFR, as well as the results for the LOCA and inadvertent valve opening events. And then provide the dose consequence overview as well.

Consistent with the focus of this meeting, we will keep the focus on the key differences from the prior review of the US600. Several of the topics we'd like to cover during today's open session is the long-term cooling without return to power; the LOCA break spectrum high impact technical issues, that was HITIs 2 and 10; and the secondary side oscillation analysis that was performed new for this design.

In addition, as mentioned by the staff

this morning during their discussion, there is some interest in the augmented DC power system, or EDAS, and its relation to the safety analysis. So we'll be covering that as an additional topic this morning. Next slide, please.

In terms of the Chapter 15 review, there was a total of 105 questions received by NuScale during the audit. Ninety-six were resolved during the audit and nine of the audit questions were sent to RAI.

Of those nine, one was split in two, such that we had a total of 10 RAI questions. Eight of those questions were resolved, and two of the RAIs were in draft form on the LOCA break spectrum HITIs, the two HITIs, and those were resolved by supplemental audit responses, such that the draft RAIs were not issued formally. So there's only eight RAIs issued formally, but there are ten questions at the beginning.

The total numbers here don't necessarily reflect all the effort spent by NuScale and the staff because in some cases some of the questions had some feedback and multiple interactions back and forth and supplemental response, etc. So there was definitely a lot of action in Chapter 15.

MEMBER MARTIN: Member Martin. Softball question, but I had asked the staff earlier about the applicability of the DSRS. And of course they stated there were no changes to support the US460.

Chapter 15, you get pretty detailed with description of the events and what have you. And you, you know, made some tweaks, but feel like despite the last four or five years, really hasn't been anything so significant as to require revisiting of the review standards.

Or, do you feel like maybe through the experiences that you've had with the review process, it might have helped to have that updated?

MR. LYNN: I think that the overall intent of the DSRS was firstly to identify the major differences for NuScale compared to an operating plant. So in that respect, I think we covered, with the original DSRS, I think we covered the big topics, right.

I think you are correct in that there's probably some content in the DSRS that could use updating, just in general for one, but also specifically to this design change. So I think there could be some benefit to that, however, it's also a significant undertaking, right, to work with the staff and get that.

So I think from our perspective, we thought those, you know, minor differences or enhancements can be just addressed as part of the review without having to revise DSRS.

MEMBER MARTIN: And even NUREG-0800 gets revised once in a while, so maybe five, ten years down the line you can sharpen the pencil and on these sort of things to help improve the engagement with the staff and of course us. Thanks.

MR. LYNN: Thank you. Next slide, please.

Okay, so we'll dive into the results. First here we're starting with the primary and secondary pressure results for the non-LOCA events. LOCA events, primary pressure decreases, so it's not really a concern. So we'll focus on the non-LOCA events.

At the bottom of the graph, you can see is event numbers 15.1, 15.12. Those correspond to the section numbers in the FSAR, which also correspond to the SRB section numbers. And on the left is the pressure.

I'll start at the top and work down. The top line in red is the postulated accident limit. The blue line is the AOO limit. And the dotted gray line is the RSV lift set point, the reactor safety valve lift set point. And the blue circles, we'll start with those, is the primary pressure, the peak RCS pressure for those events.

You can see that even though some of the events identified in the list are actually postulated accidents, all of the event results meet the lower AOO limit. And in most cases, we don't even have the RSV lift. In the middle there, you'll see the 15.2 events, which are the heat-up events. Those are the ones most likely to result in an RSV lift.

But with the set point selected there, we have minimal overshoot, such that there's still margin to the acceptance criteria.

If you look lower, you'll see the green squares, which is the secondary pressure. In our design, secondary pressure has the same design limit as the primary pressure. So there's only one set of limits on this graph.

You can see that there's approximately 800 pounds of pressure margin for all of the secondary results compared to the limit. The exception being on the very far right is the tube failure, which is 1563, and that event, due to the initiating event, the primary and secondary pressure are essentially linked.

And so the secondary pressure is much closer to the primary pressure. Next slide, please.

MEMBER ROBERTS: Kevin, Tom Roberts. Just to clarify. Is the steam generator safety valve lift point also the same as the reactor safety valve?

MR. LYNN: The steam generators themselves do not have safety valves. There are safety valves on the main steam lines, essentially outside of containment and downstream of the MSIVs. So I don't have the set points offhand, but typically those don't come into play in the safety analyses, and so we don't look at that in Chapter 15.

MEMBER ROBERTS: Yeah, I was wondering if you were to plot that line, if that includes -- the safety valve would lift in any of those transients.

MR. LYNN: No, we wouldn't expect them to lift in these transients.

MEMBER ROBERTS: Okay, thanks.

MR. LYNN: Okay, moving on, the next set of results we'll look at is MCHFR for acceptance criteria. This is for the non-LOCA events, as discussed in the Chapter 4 meeting that we just completed. The MCHFR limit for the non-LOCA events is 1.43, which is there, the red line. The LOCA events have a separate limit, and we'll cover those in a minute.

But you can see there with the blue dots, which are the event results, there's sort of a trend, sort of groupings on the far left. The 15.1 events are the cooldown events. Those tend to be limiting and challenging, but the limit is still met.

In the middle there we have the 15.2, which are the heat-up events, those are not typically challenging for MCHFR. So although we report the results, you can see there's much more margin for those.

And then shifting back to 15.4, there's another grouping there, which are the reactivity insertion events, which are more limiting for MCHFR. But in all cases, the limit is met.

If you look overall, there's a margin of at least 5% for all of the results shown.

I'll also take the opportunity during this slide while we're talking about MCHFR, there was a question during the Chapter 4 discussion, I think it was directed at the NRC staff, about FQ and the tech specs. And there was a question about if there's other tech specs that may be reached first.

When it comes to MCHFR, the protection is provided by tech specs that we have for AO, the exit offset window, as we as F delta H. And then there's also LCOs related to power flow pressure temperature, so we do have it protected quite well in terms of our safety analysis.

MEMBER ROBERTS: That's what I expected. Thanks.

MR. LYNN: Next slide. So we covered the non-LOCA results, so now we'll talk about the LOCA and the inadvertent valve opening events.

In terms of LOCA, we have our design-basis LOCAs are inside containment. And there is essentially four areas, four lines of interest. We have the CVCS discharge and ejection lines, so breaks in those location are liquid space breaks.

We also have a pressurizer spray and a reactor pressure vessel high point event lines. Breaks in those locations would be vapor space breaks.

In addition to the LOCAs, we also have what we call the IORV, or the valve opening events, that are also analyzed with the LOCA methodology. We have several different combinations there. We can have a single valve opening event where if you had a reactor vent valve or a recirculation valve or a safety valve open.

You can also have two valve open

simultaneously if the ECCS inadvertently actuates. That will cause both RVVs to open.

And then finally we have some combinations where you can get up to three valves opening. For example, if you had an RVV open plus a loss of EDAS, which is the DC power system, that would cause the RVVs to open. So you essentially can have three valves opening at the event initiation.

The results are shown in the table there at the bottom. The LOCA results are slightly more limiting for MCHFR, but we have plenty of margin to the acceptance criteria of 1.2. For minimum collapsed liquid level, we have eight feet of water above the top of the core for both LOCA and IORB events.

And we also provide the containment pressure and containment temperature results there, although those are not reported in Chapter 15, those are reported in Chapter 6, those limiting results. Next slide, please.

Now we'll take a look at the radiological consequence analysis results. There's two major categories. We have the offsite doses, which is the exclusionary boundary in the low population zone. And then the onsite dose being in the main control room.

On the left I've grouped some of the

events here. The failure of small lines, the steam generator tube failure, main steam line break, and the iodine spike design basis source term. Those are four separate events. Results for each of the four are presented in the FSAR.

But in this case, we've taken the maximum of all of the results, which has a dose of 0.83 rem TEDE. And the limit is 2.5 for events with a coincident iodine spike, and less than 25 for events with a pre-incident iodine spike.

So regardless of the spiking treatment, regardless of the event, we're well below that acceptance criteria for that grouping. And when it comes to the control room, the max of all those events is approximately a quarter rem, compared to a limit of 5.

Moving on to the fuel handling accident, we have slightly higher doses at 1.6 for the offsite doses, compared to the slightly higher limit of 6.3. And then 0.55 compared to the limit of 5 for the main control room.

And last we have the core damage event. Note that in the NuScale design, core damage is not expected to occur during the design-basis events. However, we do include the beyond-design-basis event, core damage event, as part of the radiological consequence in Chapter 15.

The maximum dose at the EAB is 2.39, and the maximum at the LPZ is 4.95. Both of those are well below the limit of 25 for the offsite doses. And the main control room dose of 1.31 is well below the limit of 5.

MEMBER PETTI: Kevin, just a question.

MR. LYNN: Yes.

MEMBER PETTI: Do you know, recall why the doses are different in the core damage event at EAB and LPZ but the other accidents it's the same?

MR. LYNN: Yeah, so the difference between the EAB and the LPZ is related to the timing. So the EAB dose is a two-hour maximum running limit. So in the fuel handling accident, essentially all of the releases are at time zero, so you get the same answer. The core damage event is spaced out over time, so you could get a different two-hour rolling compared to the total.

MEMBER PETTI: Thanks.

MR. LYNN: Next slide, please.

MEMBER MARTIN: So I don't know if it's similar question, same answer even. But just, you know, the simplest way to kind of look at a delta review is say, all right you're at your higher power when it comes to dose, and you kind of expect the proportional to power. There certainly are some other cases. I'm just looking at your, you know, table here.

You know, and the MSL break, for instance, it's about twice what it was, still well, well below acceptance criteria. What else maybe about the design might have been below its start? Or maybe the, how they're calculated could influence maybe a doubling in this case, as opposed to just something that just intuitively would just give a burst of power?

MR. LYNN: So for a lot of these things is the, especially the events on top, the release is driven or the dose is permanently driven by a primary coolant release, so that at the primary coolant, it's the activity of the coolant, right.

So it's not necessarily one to one, but essentially, you know, the increase of power drives the increase in the normal coolant. But then we also have the iodine spiking on top of that. So it's not necessarily a one to one. So it is a slightly higher dose. Even if, you know, maybe the power wasn't a doubling, you get slightly higher effects there.

But it's also driven by the containment

response, right, and how much is held inside containment. And then containment leak, right.

MEMBER MARTIN: From a methodology standpoint, I don't recall anything that looked significantly different from what you did before. But it's just that, you know, a slight change of operating conditions, you know. I don't know, like you're at a higher pressure, that might contribute.

MR. LYNN: Well, so the other thing to point out I forgot to mention is that all of the doses that we present in the Chapter 15 table include a shine-based term. The shine-based term we develop from the core damage event, and we apply is conservatively to all the events.

So that shine-based term ends up contributing for these doses where the dose is rather small, that shine term from this core damage event contributes a lot to the lesser event. So that adds into the – so even if the primary coolant may have not doubled, the addition of that shine term then makes the doubling I think –

(Simultaneous speaking.)

MEMBER MARTIN: You're implying that shine wasn't considered in the previous?

MR. LYNN: No, it was considered in the

previous, but again, would the power – would the power operate that shine-based term --

MEMBER MARTIN: Another component --

MR. LYNN: – another component of it. And then those were added, right, so it's not a straight multiplier.

MEMBER MARTIN: Yeah. Okay, no, I can appreciate that, thanks.

MS. MCCLOSKEY: This is Meghan. I would add that one other factor that we'll talk about a little more in the closed session too is the mass release associated with these events. We used a bounding mass release approach for the SDAA analyses, as opposed to specific transient event conditions. So the masses are much greater than what we might have had in the DCA.

MEMBER MARTIN: Okay, so that is maybe a new conservatism. All right, thank you.

MR. LYNN: But in general in terms of your question about methodology changes, no, we didn't really have any methodology changes.

MEMBER PETTI: Yeah, I think we had our consultant Steve's hand up. Steve?

DR. SCHULTZ: Yeah, this is Steve Schultz. The only question I have here is that when I look at the chart here, it demonstrates that the fuel – fuel handling accident has the least margin to the acceptance criteria.

Is there anything associated with the conservatisms within the analysis of the fuel handling accident that you'd like to point out?

MR. LYNN: Fuel handling accident is performed for the standard Reg Guide 1.183 treatment. So it has all the conservatives as required there. I don't have anything to add I don't think for that.

DR. SCHULTZ: But basically a standard evaluation using the regulatory guidelines.

MR. LYNN: Correct. And in our case, we release the inventory of one full fuel assembly.

DR. SCHULTZ: Thank you. So you assume that all of the rods fail when the assembly is dropped.

MR. LYNN: In the dropped assembly.

DR. SCHULTZ: Thank you.

MR. LYNN: And just to point out that the other events listed above the fuel handling accident don't result in fuel failure, so that's why the fuel handling accident has a higher release than those. Those are driven primary by the primary coolant releases discussed earlier.

DR. SCHULTZ: Thank you, appreciate the difference. I appreciate that conservatism, too, where you have all the rods break. Thank you.

MR. LYNN: Next slide, please. I'll turn it over to Meghan.

MS. MCCLOSKEY: This is Meghan. So, one of the main areas of difference in this application was related to long-term cooling. For the US600 design, NuScale -- that design was certified with an exemption to GDC 27, as was mentioned earlier this morning. And as part of that, NuScale demonstrated that SAFDLs were not challenged during the return to power with worth rod stuck out, while accounting for limiting cold thermal-hydraulic conditions and uncertainties in the reactivity balance.

But for the US460 design, we implemented the ECCS supplemental boron feature, and that uses the boiling condensing heat transfer during ECCS operation to dissolve boron oxide in containment, and re-circulate it back into the reactor pressure vessel. And so, we analyzed the design to demonstrate that the core remains subcritical following a design-basis event, assuming worst rod stuck out in order to demonstrate the conformance with GDC 27.

The Extended Passive Cooling and

Reactivity Control Methodology Topical Report that we discussed with the subcommittee about a month ago, describes the methodology used for the analysis. We evaluate the range of design-basis events in the NPM that can transition the ECCS cooling, so that includes AOOs, infrequent events, as well as postulated accident initiated conditions.

The final evaluation model that's being approved by the staff explicitly analyzes both nominal operating conditions from, like, a base load power operation during the cycle, as well as explicitly covering a wide range of off-nominal power operating histories that occur just prior to the reactor trip and can result in reduced to decay heat conditions.

So, our -- the methodology essentially results in a high biased critical boron concentration calculation, as well as a low biased core concentration that conservatively minimizes margins.

Some of the key plant initial conditions that affect the analysis are controlled by the tech specs and the cooler limits, and this includes the ultimate heat sink pool temperature needs to be between the minimum and maximum values that are specified in the tech specs as well as the level. And those two factors affect the module pressure and temperature thermal-hydraulic conditions.

And we have implemented a minimum RCS boron concentration as a function of integral downpower conditions to protect the initial RCS concentration that could be important in -- near the end of cycle.

And so, our limiting results for Chapter 15 are shown on the table on the right. And the non-LOCA events are more limiting than the LOCA events, because the LOCA events actuate ECCS earlier in the transient and there's more time at higher decay heat conditions to circulate boron back into the RPV.

The minimum margin in the non-LOCA cases occurs 28 to 42 hours or so after the event initiation. By this point, the fuel temperatures are quasi steady, but we still have reactivity insertion from the xenon decay that leads to those minimum margin conditions.

In all cases, we demonstrate that subcriticality is maintained. And you can see in the right-hand column, there's boron in the system that continues to recirculate back into the RPV over time. And so, the margin increases between the core concentration and the critical concentration between the limiting time and the 72-hour end point.

And then finally, inside the RPV. This design includes lower riser holes to assure that the fluid in the downcomer remains near the core boron concentration, and that eliminates potential concerns about dilute water in the downcomer that could be more challenging to manage during post-event recovery actions, as the operators work to restore the plant to a technical specification condition. Next slide.

MEMBER HALNON: Meghan, this is Greg. I know this is all analysis based. You mentioned operators taking action, is there any action they can take with such a low margin PPM that -- where uncertainty in the measurement could cause them to take a inadvertent action, like to either borate or de-borate based on some indication that they missed? I mean, is -- I'm just kind of thinking through EOPs, and monitoring the plant, watching for it to be within the guardrails. Anything you're worried about there, if the operator could take inadvertent action?

MS. MCCLOSKEY: I think that inadvertent action --

MEMBER HALNON: Maybe that's the wrong word, but any --

MS. MCCLOSKEY: Any action --

MEMBER HALNON: Just stay away from the

inadvertent part.

MS. MCCLOSKEY: So, I expect that the operators will be taking actions according to their procedures, as you said, in order to maintain the safe condition. And so, if they were concerned about subcriticality based on what they actually observe for the control rod insertions, in the vast majority of these events that we analyze they should have the normal CVCS injection path available to them.

The exception would be if there's a break in that piping inside or outside containment, and then if there -- in that case, there is also defense-in-depth capability through the CVCS or through the containment flood and drain system.

MEMBER HALNON: Yeah, as you were talking I was thinking, they wouldn't take action based on any one indication anyway, they would be looking for increased counts or something going on, they would take action. Okay.

MS. MCCLOSKEY: Yeah, this is a deterministic analysis that assumes the operators don't do anything for three days, which is not what we'd expect, particularly in response to an AOO where they want to get back in operational --

(Simultaneous speaking.)

MEMBER HALNON: Yeah, they'd at least be monitoring -- monitoring the reactor, at least, yeah. Thanks.

MEMBER MARTIN: I'll jump in here real quick. We'll probably talk a little bit more about this in closed session. Let's say the optics on this slide, you know, I look at the minimum margin to critical boron concentration, 30, 28 PPM doesn't sound like much when you consider uncertainty. Now, of course, you note on the slide all the conservatisms then, but you don't have, you know, kind of the -- have it as the benchmark. And we've seen your results and we know that it's rather significant.

So, the natural question that comes from looking at the slide is how do -- you know, what are in these uncertainties? Because it doesn't look like there's much. But you might want to reiterate, you know, the significance of the conservatisms that you indeed have considered here. Because, like I said, it doesn't look like much, you know, on the slide itself.

MS. MCCLOSKEY: Yeah, the analysis methodology stacks the conservatisms deterministically, and so they stack on top of each other to bring the critical boron concentration up and the core boron concentration down.

And so, these are margins associated with the conditions towards end of cycle, and if we just think about the control rod worth, with all control rods inserted in this design, that brings the critical boron concentration down to below zero. So, you don't need any additional boron compared to what's already in the system to maintain subcriticality, and it's not possible to reach this type of condition.

(Simultaneous speaking.)

MR. BRISTOL: I suppose -- yeah -- this is Ben Bristol. In the spirit of the delta review, this slide for the DCA presented margin in units of MCHFR to the limit, so recriticality was not precluded. And the margin was all about -- or the conservatisms were all about calculating the conservative critical power level, and then evaluating that SAFDLs limit.

So yes, we're presenting margin to the critical, calculated critical concentration, but there's also significant additional margin to actually a safety concern of where SAFDLs would be violated.

MEMBER MARTIN: Wanted to give you the opportunity get that on the record. Thanks.

MEMBER PALMTAG: This is Scott Palmtag, a couple questions to follow up on one that Greg just asked. What type of operator actions can happen?

Because I thought, especially with a LOCA, everything's sealed up and CVCS is not available.

MS. MCCLOSKEY: With the LOCA conditions, the containment is isolated, yes, but those conditions really aren't challenging for our minimum margin cases. This is a minimum margin case that assumes a LOCA occurs after some -- an off-nominal, one of the off-nominal power histories that we've evaluated.

MR. BRISTOL: So, and just to build on that, even in the LOCA scenario, which is fairly unlikely, there's a couple of things that can occur. There's operational bypasses that allow for CVCS injection, under conditions where operators are allowed to do that.

In addition, these cases are sort of preferentially biased to minimize temperature, which is the conservative analytical kind of condition for these calculations, under those conditions that the containment in the entire system naturally depressurizes to the point where all of those signals clear. And so, the isolation functions are no longer active once the system temperatures get down to around 200 degrees.

At that point, then, operators can realign the system. And in the case of the injection line, that's the one I'm looking at, boron can be added through the pressurizer spray via the injection bypass up there.

Most likely though, the Mode Four state would probably have an operator's flood containment with the flood and drain system restore the normal Mode Four refueling levels in the module -- that's done adding pool concentration water to the system, so that's at 2,000 PPM, and would quickly recover and restore shutdown margins.

So that's -- the simplest thing would be either bypassing once operators have confirmed they understand the event, understand the module's in a safe condition, go ahead and bypass and flood with the flood and drain system. Or, let the system continue to depressurize while evaluating, and then once the MPS signal's clear, then they can go ahead and restore the flood levels and return to repair conditions.

MEMBER PALMTAG: Thank you. I think that's important that, you know, there is operator actions that could occur -- this is assuming 72 hours with no operator interactions.

Another question about the core design we had with the NRC questions earlier, which core is this for? Is this for the equilibrium core or is this generic, all the cores you're looking at? There's going to be a wide variation between cycle one and your equilibrium, have you looked at all of those or is this just equilibrium cycle?

MR. BRISTOL: This is the equilibrium cycle. Meghan mentioned, and I think we have more details in the closed session as to how we generate the operating limits curve. But essentially, there's some additional constraints beyond what the normal tech spec core operating limits that go into supporting this analysis. Those will be updated on a cycle-specific basis to ensure that the margins are maintained, regardless of the different characteristics of the first cycle or second session.

MEMBER PALMTAG: Okay, so this does depend on a core loading, and you've -- this will be updated at each core loading?

MR. BRISTOL: Parts of this analysis, yes.

MEMBER PALMTAG: Okay. All right, thank you.

MEMBER HARRINGTON: This is Member Harrington, just to follow up on those questions again, can you clarify for me the sheer ability of the operator to take action? Then -- you said, you know, pressure drops, then the isolation conditions clear, can they override the isolation conditions or not?

MR. BRISTOL: So, the answer is yes. The specifics of the operating procedures, I think are subject to the development of those procedures and the concept of operations in a given plant design.

MEMBER HARRINGTON: Yeah. No, I just wanted to understand --

MR. BRISTOL: But there's ability within MPS to go in and override that isolation function, if the operators have deemed that that's the appropriate action to take, and restore a configuration where boron could be added through a variety of different means, as Meghan mentioned.

MEMBER HARRINGTON: Okay, thank you.

MS. MCCLOSKEY: Okay Kevin, back to you.

MR. LYNN: Next slide, please. We'll talk about the LOCA Break Spectrum HITIs. As discussed there's two HITIs not related to this topic, and essentially they cover two different regions or locations within the plant. This slide's divided in half, so we'll cover the left-hand side first, which is connections between the ECCS valves and the RPV.

So there's four total ECCS valves per module, so we're talking about breaks in four possible locations. In a design-basis valve opening event that we analyzed in Chapter 15, there's flow through the value but it's restricted by venturis which are in place within the valve, which is the figure on the left below. Those venturis are captured by a shoulder feature, such that they can't be dislodged within the valve.

But theoretically, if you had a hypothetical break at the flange, for example if you had the eight bolts that connect the valve to the flange to the RPV -- if you had those eight bolts simultaneously fail such that the valve completely detached from the vessel, at that point the flow path there in the blue line wouldn't be through the valve, it would just be through the opening.

And that opening is larger than the venturi itself, so therefore you would have a larger flow rate which could potentially be more limiting for MCHFR and the containment response. But it would not be more limiting for the liquid level above the top of the fuel, because of all of the loss of inventory is still captured within the containment vessel.

So, that's one region of interest for the LOCA Break Spectrum HITI topic. We'll talk more about that on the next slide. But first we'll cover the other area of interest, which is the connections between the CNV and the CVCS piping.

So, there are four CVCS lines total per NPM. If we look at the figure on the right and we start at the bottom, the bottom red line represents the edge of the containment vessel. So, breaks below that bottom red line are breaks inside the vessel, those are analyzed in Chapter 15 as design-basis LOCAs. So, everything below the red line is fine.

If we look at the top red line, those are breaks beyond the containment isolation valves. So, breaks there are isolatable and they are analyzed as non-LOCA events in Chapter 15, so those are fully analyzed as design-basis events. So, everything above the top red line and below the bottom red line is analyzed explicitly in Chapter 15 as a design-basis event.

So, the area of interest for this HITI is the area between the two red lines which consists of the containment nozzle to safe end, and then those welds, a containment isolation test fixture, and the containment isolation valves which is two valves inside one single body, and the welds between those connections.

So, if you were to assume a hypothetical break at one of those connections, or one of those components, that would have the effect that the inventory would be lost outside of containment but would not be isolated by the containment isolation valves. So, breaks in that location would have the potential to be more limiting for liquid level response, because the inventory wouldn't be retained within the containment vessel.

They're not more limiting for MCHFR containment response, particularly for MCHFR because it's the same break as the other locations -- it's the same size. But for containment response, because the release wouldn't initially occur inside containment, so it's not limiting, there.

So we'll go on to the next slide to talk about what the implications of those breaks are, understanding now where they are. NuScale's position is failures at these particular locations are very unlikely due to the design of the connections themselves, the design stress and fatigue limits that are applied at these connections, inspections that are performed as part of plant operation, and the ability for operators to detect leaks before breaks.

To resolve these issues for the US460 design, after discussion with the staff NuScale elected to take exemption from 10 CFR 50.46 and GDC 35 so that we could classify postulated failures in these particular regions as beyond-design-basis-events.

With them classified as beyond-design-basis-events, analyses have been performed but they have been performed using alternate acceptance criteria and alternative assumptions compared to their design-basis counterparts. And we'll cover in more detail in the closed slides what those particular differences are.

However, we do have the analysis results, which we'll also cover in detail in the closed session. But those results show that the event-specific acceptance criteria for core cooling, containment response, and dose are all met. And they are met with credit for only -- with only our passive safety-related design features. When you consider the potential for additional operator actions where makeup can be provided, that provides additional defense-in-depth.

Overall, our conclusion is that these failures are unlikely, but our NPM design can passively mitigate failures in these locations so they're not a safety concern.

MEMBER MARTIN: This is Member Martin. Just to kind of close the thought here, and I imagine these -- the likelihood of these events is captured in the PRA. But you have that data as well to support these statements.

MR. LYNN: So, in particular, the failure ECCS valves we don't provide a quantification of that inspection frequency within the PRA, the reason being that it's not a hard image sequence for the PRA, so that, essentially, doesn't contribute to reporting the frequencies, so there's no reason to quantify that particular item.

For the breaks outside containment, we did provide some quantification based on fracture mechanics, et cetera. There was some disagreement between NuScale and the staff as regarding those exact numbers. So, at the staff's request we did not include those quantifications as part of the FSAR.

However, the agreement between NuScale and the staff, despite disagreement on what the exact number, we all agreed collectively that there was no -- that the number is low in subsequent failures.

MEMBER MARTIN: Okay. And we'll see this in the closed session to support your statements later in just a little while.

All right, thank you.

MR. LYNN: We'll move on to the next slide, at which point I'll turn it over to Ben.

MR. BRISTOL: Yeah. This is Ben Bristol. We're going to switch gears here a little bit and talk about stability and the analysis presented in Chapter 15, 15.9 sections.

Typically, largely the content in this section is similar to what was performed in the DCA using a similar method. We have a reactor system stability analysis model that is used to perform steady state perturbation analysis, as well as AOO transient stability analysis that's evaluated to acceptance criteria to K ratio, with an additional protection on offset risers for cooling.

That's a combination of the low pressure at high temperature trips in the reactor protection system. Those analyses and conclusions were similar to the DCA.

In addition, there's some new scope that was added. The purpose of the scope is to address the posture in the event of a secondary side oscillation developing on the supporting balance pipe systems.

There is a COO item related to this for the DCA, 7.0-1, where it was recognized that plant details were required in order to perform this analysis and demonstrate the plant system stability had been achieved by the actual as built plant design.

There was a slight change in the arch we took here, rather than leveraging the actual plant design in order to perform this analysis to confirm stability, that's still a design requirement for the system, we instead took the approach of bounding the postulated events, evaluating an exhaustive series of oscillations looking for feedback between the primary and secondary system.

So, we imposed oscillations using a feedwater flow oscillation, as well as a secondary pressure, steam pressure oscillation. That imposed, basically, a steam generator power oscillation because of the feedback on the primary sides, specifically the core power.

What we found is that generally the core power would follow, either follow the steam generator power or be able to compare to the steam generator power. High frequency oscillations were less challenging.

And in the end we were evaluating SAFDLs and confirmed that under the most limiting events, usually larger amplitude, slower oscillations to run the secondary side. These events looked very much like the transients that were previously analyzed in the Chapter 15.1 sections.

And similar to those conclusions, SAFDLs were all vets and protection, adequate protection was provided via the existing module protection system.

MR. ROBERTS: Hey, Ben. Tom Roberts.

Can you clarify the support sub bullet under the second major bullet, variety of module -- of MPS signals provide protection to terminate oscillations? The analysis you did imposes the oscillations, so I'm not sure how the MPS would terminate them. Do we mean terminate the transient?

MR. BRISTOL: Yes. That's a good point.

So, the oscillation could be terminated by secondary isolation. Some of the signals provide secondary isolation.

But, but, yes, the protection is provided via reactor, reactor trip. And, so, that's what, that's what provides the protection.

MR. ROBERTS: Okay. Thank you.

MR. BRISTOL: Sure.

Okay, next slide and I'll turn it back to Meghan.

Oh, turn it over to Tom.

MR. LYNN: Yeah, I think before Tom starts, just at this point, again, we covered all the major differences between the US600 and the US460 that we intend to cover.

And now we'll transition, talking about the EDAS system in a little bit more detail.

Tom.

MR. GRIFFITH: Yeah. Thanks, Kevin.

Thomas Griffith, NuScale.

So, I want to start from just give a bottom line up front because we have quite a few slides here, and I want to make sure that we kind of hit some of the, the high level points up front here so that we all keep those in mind and we can walk through the slides and get into some of the details for these points.

I'd like to set the stage that, first of all, the US460 exceeds Commission Safety Goals by orders of magnitude. So, the Commission Safety Goals for CDF and large are on the order of E to the minus 4, E to the minus 6. Whereas, the US460 design has a core damage frequency and large release frequency on the order of E to the minus 9 and E to the minus 13.

I bring that up in this context here mainly because it's not just an exceedance by one or two orders of magnitude, it's a substantial margin to what the Commission has set forth as, as the safety goals.

And NuScale's design does that with a nonsafety-related DC power system. Power is not required to place the US460 design in a safe, stable condition.

So, in terms of the DC, the augmented DC power system EDAS, in comparison to the US600 design, for the US460 design NuScale included additional requirements for what needs to go into the OCRM above and beyond what was required for the US600 design.

When we talk about ECCS, the function, the specified safety function of ECCS is the same between the US600 design and the US460 design. There is a recognition that we did remove IABs on the reactor vent valves.

What that does is it allowed the vent valves to open earlier in an event progression than was, than it was capable of during the US600 design.

It improves safety by allowing those vent valves to be more predictable when they open.

It also allows the operators an opportunity to depressurize the vessel.

Those are both very important functions. And I want to reiterate that the safety function of ECCS is to open to establish a passive cooling path with inside of containment through the reactor vessel.

When we talk about what was reviewed during the US600 design, it was NuScale's understanding that a number of the issues related to, would point out reactor coolant pressure boundary integrity, were largely discussed as part of the passive electrical power topical report.

So, in NuScale's position what we did for the US460 design was we designed a DC power system that met the fundamental basics of what was, what was approved in the electrical power topical report. We included a technical report that outlined in detail how the conditions limitations were met for that topical report. And, in addition, included, included in the Appendix B of that technical report a substantial amount of detail for each of the conditions for that topical report, and how it could be used.

Any questions so far?

MR. ROBERTS: So, it's Tom Roberts.

MR. GRIFFITH: Yes.

MR. ROBERTS: You'll probably get to this later in your presentation. But the bottom line on up front stage I just wanted to note that the motivation for this discussion appears to be that there is a scenario where a failure that causes an RVV or the ECCS, whatever, to initiate during another transient, kind of CTHF on this, and that an untimely failure of EDAS is one way you could get there

Is that right?

MR. GRIFFITH: Yeah. Part of the motivation is that discussion that came up. And we'll reference that as we reference it as the smart failure scenario.

MR. ROBERTS: All right. Just want to keep that in mind that the real problem is, as I understand it, is not the EDAS failure. The real problem is EDAS failure range in that scenario.

(Background conversation on mike.)

MEMBER HALNON: Go ahead.

MR. ROBERTS: To make that point and make sure I understand this correctly, the real problem statement is not an EDAS issue, it's an any scenario that could cause this combined transient is something you would need to think about.

MR. GRIFFITH: Well, I mean, I, I would say, I would argue that Chapter 15 looks at EDAS as being unavailable at the, at the onset of an event in Chapter 15. Which means the solenoids are de-energized and the vent valves are open.

So, Chapter 15 analyzes with the vent valves open.

MR. ROBERTS: Right. I understand that.

The question is that motivates the discussion, though, is Chapter 15 does not assume that you have a scenario that could open an RVV during an unrelated transient.

MR. GRIFFITH: Correct, yeah. I think --

MR. ROBERTS: To me, that's the sane way to get there but it may not be the only way to get there.

MR. LYNN: This is Kevin of NuScale.

I think to get at your point, right, the crux of the issue is, is it necessary to assume two initiating events at the same time? And NuScale's position is it's not.

MR. ROBERTS: Right. And I wonder if two initiating events at the same time, or an event that causes the valve to open at an initiating event is semantics. I'm sure we'll get into the semantics later

But I just wanted to make that clear at the outset of the discussion that's really what you're talking about is that this is a way one could postulate to have that double event or event or event complicated. But I assume whatever you would call it.

But it's not the only way to get there.

MR. LYNN: Correct.

MR. ROBERTS: Okay, good. Thank you.

MR. GRIFFITH: Yeah. And so I want to, I want to kind of summarize this slide. And the last is to pull from SRM-SECY 19-0036 from the, from the DCA.

And that "...in any licensing review or other regulatory decision, the staff should apply risk-informed principles when strict prescriptive application of deterministic criteria such as single failure criterion is unnecessary to provide for reasonable assurance of adequate protection of public health and safety."

So, in the case of reactor coolant pressure boundary integrity we think that that was resolved here in the DCA. We have a combined failure frequency of an inadvertent ECCS actuation that is less than the frequency of once in a monitor lifetime, and substantially less than that.

From the standpoint of whether or not a smart failure needs to be taken, it does not add to safety. And as an extension, I think that there a few of these, just to put it on the record, that will resolve using paperwork, which did not necessarily improve safety at all.

And the point there is that I would, I would argue that the LOCA break spectrum heading was resolved through an exemption, did not require any design changes. The design was safe as is. And the principles of 19-0036 could have been applied there, and should be applied to the safety finding related to EDAS.

And with that, I'll turn it over to Kevin.

MEMBER HALNON: This is Greg.

Before you go, I'm not sure I understood what you said when you said a smart failure doesn't add to safety.

I know that in the present I believe the reactor's do smart shorts, and smart fires, and smart symmetry circuits, and all kinds of things. But isn't a nonsafety system not only can't be credited, but also can't affect a crashing of the transient as well.

So, explain, you know, what you meant by a smart failure doesn't add to safety?

MR. GRIFFITH: I would, stepping back, EDAS was analyzed as available and unavailable at the beginning of an event. In order to get a failure EDAS it's not -- it requires multiple independent failures in order to disable the system.

So, stepping back, you have two divisions with two channels of batteries. Each one of the channels, so, each one of the four channels has its own battery charger, supplied from the low voltage system, which has redundancy. The low voltage system feeds up to the medium voltage and high voltage, has redundancy.

MEMBER HALNON: What you're telling me is very reliable and it's redundant. That's great. But we're not talking about that. We're talking about the regulatory rules of applying nonsafety-related systems to transient analysis where you're not taking credit for acidity.

The smart failure is that they are both available, and one of them, or both of them, or all of them fail exactly at the right time and make it really bad. But you -- that's a smart failure.

And what you said, that that doesn't add to safety. I don't understand that connection.

MR. GRIFFITH: So, traditionally, in Chapter 15 analysis what has been done is the system is either available or unavailable at the beginning of the event. Your failures typically happen with, with -- because of some sort of cause. And I think we've talked about that before.

Like, a turbine trip results in a loss of offsite, offsite power due to grid instability, which then trickles down on a plant, requires your EDGs to start. And then at that point in time if the system is demanded, that's another opportunity where we have, traditionally, you have said you have a failure to start for a particular component.

What we're talking about in the context of this particular smart failure, and specifically for the RVV opening, is an event that, that slightly exceeds your MCHFR ratio and causes a, and results in a slight fuel heat-up that does not get anywhere close to regulatory levels. We take it a margin at a time.

There is a substantial amount of margin that will go into that. And the exact margin's in the site, but it is in excess of 1,000 degrees Fahrenheit.

MEMBER HALNON: So, just to rephrase, my understanding is that it doesn't take away margin of any concern for smart failure?

MR. GRIFFITH: Correct.

MR. LYNN: I think our position, in response to your question, would be the phrase "doesn't add to safety," it doesn't add to safety to make classification-based decisions on these events. In other words, we show that these events are not a problem, there's no significant consequences.

So, using those events in a deterministic way to make decisions about the classification of systems is unnecessary.

MEMBER HALNON: That makes sense. Thank you.

MR. LYNN: Next slide, please.

Tom highlighted some of the major points that we'll make over the next few slides, that will cover those in more detail.

In terms of the general background for loss of power considerations as it pertains to Chapter 15, the GDCs require that safety functions be performed with onsite or offsite electric power available.

GDC 17, in particular, identifies the safety functions to be performed assuming -- with one system assuming the other system is not functioning.

Corresponding GDCs 34, 35, 38, 41, and 44 identify system-specific performance of onsite or offsite power operation related to GDC 17.

Typically, an operating plant would affect the GDC 17 in safety analyses by assuming offsite power is available throughout the event, or offsite power is lost, which prompts your safety-related onsite power systems to take offer.

That loss, loss of power is assumed at event initiation or it's assumed at the time of reactor trip as a consequence of the reactor trip itself. In other words, the reactor trip from the initiating event causes, because the unit's so large, causes the disturbance of the offsite power grid, which essentially causes the offsite power grid to fail.

And then the incoming power to the plant is lost, which then causes the onsite power system to go down.

So, in other words, it's a causal case. And in often case, the operating plants get credit for the timing-based delay, whereas there's a time delay between the time of reactor trip and the time that we trust that power is lost because of the physical process by which it takes time for the grid to go down.

But noting here in the context of this discussion, the grid itself is not a safety-related entity; right? So, crediting that delay is in a sense crediting a nonsafety-related component to perform a function during that sequence.

Next slide, please.

How does NuScale approach loss of onsite power?

NuScale's design we can perform safety functions either with or without electric power.

So, therefore, we take an exemption from GDC 17.

We do identify in FSAR Section 3.1 that we meet the intent of GDC 17 in that with electric power unavailable, our safety-related SSCs can satisfy the requirements that SAFLs are met during AOOs, and that the design conditions of the RCP are being -- RCPB are not exceeded, and that we maintain core cooling and containment integrity during Pas.

We also have conforming designs to the principal design criteria that corresponds to that, that we can perform safety functions without electric power.

In our design we show that we can meet our safety analyses by assuming electric power is unavailable. To do that, we have to succeed by demonstrating that we can lose AC power either at the time of event initiation or at the time of reactor/turbine trip, similar to operating plants.

And then when it comes to EDAS, we show that we can survive with EDAS either available or unavailable at the start of the event, with event initiation.

The reason for the difference between AC power and EDAS is that AC, the loss of AC power at the time of reactor trip is assumed to be a cause of the event itself; it's causal. Whereas, EDAS we show that there is no direct cause where the initiating event causes EDAS to fail and, therefore, there's no valid reason to assume that it does fail.

During that process in our safety analysis we demonstrate that electric power is not credited to mitigate design-basis events and, therefore, AC or DC power supplies are nonsafety-related.

Next slide, please.

Okay. One of the things to consider in the definition on the previous slide was what does it mean to maintain reactor coolant pressure boundary integrity?

In our design, the ECCS valves are designed to open if power is lost.

So, for the previous design, the US600, power supply was called the EDSS -- a slight difference here. But in that same design it's the same, the key function of the valves is to open to establish passive cooling.

And, so, the fundamental safety feature of the ECCS valves and the power supply to those valves is the same in both designs.

During the review of the US600 DCA, the Commission determined that inadvertent ECCS was not considered loss of RCPB integrity.

This came up during the review of the passive electric topical report. It was approved ruing that review.

There was a question from the staff about whether it was acceptable if nonsafety-related systems was maintaining the RCPB integrity?

In particular, on the loss of EDSS, the ECCS valves opened. And the question was whether that constitutes a violation of the RCPB integrity.

In that design, due to the presence of the IABs, the ECCS valves opening happened at approximately 1,000 pounds. However, the ECCS valves would open on loss of EDSS, regardless.

GDC 15 requires that the design conditions of the RCPB not be exceeded during normal operation of AOOs. And we understand that to constitute a gross failure due to over-pressurization of the system.

ECCS valve opening doesn't challenge the design conditions of the RCPB.

However, the staff concluded the opening of the ECCS during an AOO may be -- may not be consistent with the defense-in-depth purpose of GDC 15. And, so, ultimately, during the DCA review that issue was resolved by requiring the expected frequency of occurrence of a ECCS valve opening following an AOO to be less than once in the lifetime of the plant.

That was enforced with a Limitation and Condition No. 4.4 on that topical report. And we did that, with that L&C satisfied, the NRC concluded during the previous review that no exemption was required for EDAS for EDSS to be nonsafety-related. And the opening of the RCPB -- or opening of the valves did not constitute a failure of the RCPB or challenge its integrity.

This was confirmed in Statements of Consideration for the US600 by the Commission.

Next slide, please.

Okay. The next slide covers what, if anything, changed in the US460 approach.

In terms of the ECCS valves and their design, there's no change in the approach. The valves are still designed to open if power to the valves is lost.

Our licensing basis approach overall still follows the approved topical report from the DCA.

We have similar augmented requirements to ensure that the EDAS, the system that supplies power is reliable.

And although we don't reference the topical report in the FSAR, we do continue to apply and meet the L&C that ensures that the frequency of an AOO to the ECCS is less than once in the lifetime of the module.

So, what did change?

Well, we removed the IABs from the ECCS RVVs. And this change was made to improve overall plant safety by enhancing the ECCS capabilities for some events and some sequences.

As a consequent is that on a loss of EDAS the valves open at a higher pressure than would occur for the US600.

So, previously, on a loss of power supply the valves would open at approximately 1,000 pounds. Now they open at approximately 2,000 pounds.

This, from our perspective this is not a change in how RCPB integrity definition and GDC 15 applies. We still meet that definition based on compliance to the L&C 4.4.

We also explicitly analyzed valve opening events as an AOO to show that there is substantial safety margins for those events.

So, overall, the conclusion is the same as it was for US600: the power supply for the valves is not relied upon to ensure RCPB integrity.

MR. ROBERTS: Thanks, Kevin. Tom Roberts.

Help me understand what you just said.

If you had an inadvertent valve opening during an over power for temperature transient, on the original design, would the valves never open?

MR. LYNN: The valves, if you lost power to the valves, the valves did open once you get below the set pressure of the valve.

MR. ROBERTS: Which wouldn't happen for loss of you SCRAM; right?

MR. LYNN: Correct. So --

MR. ROBERTS: So, you could operate for, I don't know, hours?

MR. LYNN: Well, no. So --

MR. ROBERTS: Not having the valves actually open if you had an initiation?

MR. LYNN: The loss of power supply would trigger NTS to actuate safety systems. So, DHRS would be one of those systems actuated. DHRS provides cooling which would quickly cool the plant, depressurize the plant. And, so, then you would get the valves opening as part of that process.

MR. ROBERTS: Okay, thanks. But you'll be screening for that point --

MR. LYNN: Yes.

MR. ROBERTS: -- well before the valves open?

MR. LYNN: The valves would open --

MR. ROBERTS: Seems like a significant change.

MR. LYNN: Correct. That is a significant change. This was part of the things that we addressed as part of the pre-application process with the staff.

MR. ROBERTS: Okay. Yes, coming up to the next slide. I just wanted to understand what you meant by this didn't change the safety. It seems like for the transient, if they needed to be considered it's going to be a change. It's going to be, the argument must be it does need to be --

MR. LYNN: Correct. In the concept of this in the context of this slide we were focused on the discussion of RCPB integrity, one of the things being that one of the discussions we had with the staff was the question of whether valves opening constituted violation of the RCPB integrity.

So, our perspective here is that, yes, the pressure is different, so it does affect that so-called smart scenario. But in terms of the RCPB integrity question and the application of GDC 15, it's a matter of timing, not a matter of overall sequence. Right?

MR. ROBERTS: Okay. Yeah, thank you.

One really quick clarification. If you lose EDAS, you SCRAM; right?

MR. LYNN: Correct.

MR. ROBERTS: So, the same circuitry that holds the solenoids, you know, holds the RVVs shut, also keeps the rods latched. So, you would have a simultaneous Scram and ECCS initiation if the cause was loss of power?

MR. LYNN: Correct. Yeah, well, I'll provide an overview of the timing of those different scenarios on the following slide.

MR. ROBERTS: Okay, thanks. Because if the cause of the RVV opening was not loss of EDAS but something else, then it would seem like the Scram would be delayed by whatever time's required to depressurize the plant and make other protective signals to cause the Scram.

Have you looked at that?

MR. LYNN: So, in the case where the, the valve opening is not caused by the loss of EDAS, you're correct that you wouldn't get a Scram for the same reason of the valve opening.

However, if you had a valve opening event, right, it would pressurize the containment. Containment pressure triggers reactor trip. And, so, those valve opening events could be analyzed inside the containment. Typically, trip occurs within the first 5 seconds. So, then --

MR. ROBERTS: Did you look at that? I guess when we get in the closed session you'll have the details of the transient, and you looked at the violation. Did you look at that scenario, looking at the temperature?

MR. LYNN: So, we have looked at valve opening scenarios as part of Section 16.6.6. We have valve opening scenarios where the trip is delayed until the next trip signal is reached, which feeds containment pressure.

What we, if I am interpreting your question correctly, what we did not look at is we did not look at an event where if you had a separate initiating event it caused power, temperature, et cetera, to be high, and then open a valve not related to the loss of EDAS.

MR. ROBERTS: Right. Okay. And if we could talk about that during the closed sessions, just the details on what you did look at.

Okay. Thank you.

MR. LYNN: So, next slide.

Again, it was mentioned that in the context that I think Mr. Roberts you mentioned the context of this, but it was a change in terms of the sequence of events and how things could happen.

So, that is something we thought of when we made this design change. But, overall, the design change motivation to remove the RVV IABs is to improve plant safety in the overall context of public health and safety.

We did address it in accident sequences. And, again, because of the nature of this change we did have multiple pre-application engagements with the NRC to discuss this, this difference, and to address concerns raised by the NRC during those meetings.

As part of the FSAR we submitted a new technical report that's ref -- that was referenced in FSAR Chapter 15. It provided the description to augmented requirements applied to EDAS, evaluation of how those augmented requirements protect EDAS from design-basis initiating events so that we could show that EDAS failure is not expected to occur following those events.

We also covered how Chapter 15 evaluates EDAS failure to show that the system is not relied up in the safety analysis.

We included a quantification of the frequency of an AOO with the ECCS to show that we meet the L&C 4.4 from the prior topical report. And, so, those numbers were provided in that technical report.

We also provided that quantification of the so-called smart failure sequence show that thee quantification of the frequency -- expected frequency of that event sequence is essentially E to the minus 8 per year.

So, regardless of the consequences of that event, the initiating event itself frequency is far lower already than the Commission safety goals.

We did with that provide an evaluation of the consequences of that smart failure.

Next slide, please.

So, this gets into a little bit of the timing discussion of this that was addressed earlier.

So, through the loss of EDAS, the NPM safety systems are designs to actuate to their safe position when the power supply is removed.

So, what does that mean for the safety components?

So, ECCS actuates essentially when EDAS power is lost. That results in the valves opening.

The approximate opening time of the valves is approximately 1 second.

The RRVs -- so that would be for the RVVs. The RRVs, although they still have IABs, so they would remain closed initially, once pressure decreases they would then release and open.

Reactor trip happens at the same time the EDAS is lost. The rod insertion timing is approximately 2 seconds.

And then containment, secondary system both isolate. DHRS actuates. And those systems have valve repositioning times on the order of 10 to 30 seconds.

So, a loss of EDAS, within 30 seconds all of the safety systems have been actuated to their safe positions, and we transition to a safe shutdown condition.

But due to the difference in timing there with the RVV opening time versus the reactor trip time, we essentially have a bit of a race where the valve is opening at the same time rods are going in.

We hope the depressurization caused by the valve opening causes as slight flow reduction that causes a short duration reduction in MCHFR.

We see that in our analyses that are performed for valve opening events. The same consequence would occur in the so-called smart failure. So, it's a limited duration, transient dip in MCHFR that is very quickly overcome by the rod insertion from the reactor trip at the same time.

Next slide, please.

We covered a lot of this material before. But how do we find the loss of power in Chapter 15?

For AC power we take a loss of AC power and event initiation as a deterministic assumption.

And then we also assume a loss of AC power at the time of reactor/turbine trip, which is a consequential failure. Not that the NPM itself is rather small in terms of its electric megawatt output to the grid. So, it's expected that as single trip of the single NPM would not cause a grid failure like it would for a 1,000 megawatt plant.

However, we have retained that assumption of the causal failures, part of the conservative approach and the traditional practice that's used in the industry.

Then comes the EDAS loss of timing.

We assume the EDAS was lost a event initiation as a deterministic assumption. But unlike the loss of offsite power, AC power, there's no direct causal failure where the event initiation progression causes EDAS to fail. So, therefore, we don't assume a failure subsequent to time zero.

This treatment is consistent with design and augmented requirements being applied to the system.

Next slide.

And as far as the technical report that was included with the FSAR, we did provide a consequence analysis of the random loss of EDAs at the time of worst conditions. So, even though we don't consider it to be a design-basis event, we show that the consequences are minimal.

So, regardless of the initial condition, no core damage occurs.

Through a variety of combinations we can show that even the conservative MCHFR limits for the conservative safety analysis, the limits are met, even from powers above 102 percent, starting above 102 percent.

We include a PCT calculation to show that there is significant margin to the 10 CFR 50.46 limit for PCT.

And we'll provide some figures in the closed session to go along with that.

The technical report that was originally included and referenced in Chapter 15 was later removed at the NRC request. We included that because we thought it would be helpful for the staff to make their safety findings. However, they requested and stated that it was no longer necessary, so we removed it based on their request.

In addition, the NRC asked us to consider whether tech specs should apply to the EDAS system. From NuScale's perspective, tech specs are not necessary for this system.

One of the main reasons being that we pertaining to operating power to systems not available.

So, none of the safety analyses events can occur if you're not in power. And, essentially, if EDAS is already unavailable, you're not in Mode 1.

If you are in Mode 1 and you lose EDAS, all of the safety functions are performed as designed: reactor trip, isolation, DHRS, ECCS. All of them occur quickly and safely to a safe, stable condition with no need for further operating actions. So, there's no need for a tech spec from that perspective.

However, we did commit to control EDAS under the OCRM and the maintenance rule program. And this was done to ensure the system remains reliable and available throughout the plant lifetime.

Obviously, the plant owners will have a great motivation to keep the system reliable because it's directly related to being online and to, essentially, making their plant economically viable. But we did include those requirements to control in the OCRM and maintenance rule just to provide that additional regulatory assurance.

MR. ROBERTS: Kevin, Tom Roberts.

Your arguments to not include the tech specs seem to all be based on complete loss of EDAS, which I agree, if you lose EDAS you cannot run the plant. So, it's kind of needless to put that in tech specs.

But I would think that reduced redundancy in EDAS might be something you'd want to cover. And I assume that's in your OCRM manual that you would have limited operation with one of the batteries out of service on a given side, and that kind of thing.

Did you consider tech specs for degraded redundancy occasions?

MR. LYNN: Correct. Yes.

The requirement to include that is part of the OCRM, is to ensure that we maintain the reliability in accordance with the L&C subsequent inadvertent ECCS doesn't occur during the life of the plant.

So, one of the reasons inadvertent ECCS isn't expected to occur during the lifetime of the plant is due to the redundancy of the design. Right? And it takes more than a single failure.

So, in the event that you have a, essentially, late failure already present, right, the OCRM requirements would drive you to assess that to ensure that you could still meet that L&C that you're not expecting to occur during the life of the plant.

MR. ROBERTS: So, it's kind of an administrative call, OCRM vs. tech spec to get to the same place; is that it?

MR. LYNN: Correct. Yes.

And the end is the incentive would be for the operator to assess the system, assess the risk, and decide what the appropriate course of action is to do.

MEMBER HALNON: Again, one question from

Greg.

As I see the EDAS systems seismic monitoring system -- this is the crux of the --

(Audio interruption.)

MEMBER HALNON: I think that was an inadvertent contact.

CHAIR KIRCHNER: Just warn the people listening in, please silence your mikes.

Thank you.

MEMBER HALNON: So, is it seismically designed?

MR. GRIFFITH: Thomas Griffith.

EDAS is seismically --

MEMBER HALNON: That's what I thought.

So, externally that's not a problem on this.

I guess the curious question -- and maybe, Tom, you can answer it -- is how far, safety class aside, how far are you from, from the design being equal to the safety class?

MR. LYNN: Not that far. The intent to design the system was to design it like a safety-related system. One of the key hangups is the batteries. There's only one type of battery that could be classified as safety-related. And those batteries are very large.

So, having a sufficient battery of that particular type requires a large amount of real estate, which is a large amount of weight, which requires changes to the reactor building design.

So, one of the motivations is with this nonsafety-related system we can use those different batteries, we can shrink the footprint, and change the reactor building design.

So, conversely, if the question was how much work would it be to make it safety-related? In reality, not that much, except that using the different batteries would require a redesign of the reactor building.

MEMBER HALNON: Okay. And so it's a battery technology issue?

MR. LYNN: Correct.

MEMBER HALNON: Rather than the designs around that?

MR. LYNN: Correct.

MEMBER HALNON: Thanks.

MR. LYNN: And, so, if you look at the augmented requirements that are applied to the EDAS system which are referenced in Chapter 8, those essentially mirror most of the requirements that you would have for a safety-related system.

MEMBER HALNON: And as you were talking and as Tom was describing reliability and redundancy, it was very, very close, just not there.

Thanks.

MEMBER BALLINGER: This is Ron Ballinger.

My understanding is from my memory is that the difference between these batteries is basically is one's vented and one's not vented. But it's the same thing.

MR. LYNN: Correct.

DR. BLEY: This is Dennis Bley.

CHAIR KIRCHNER: Yeah, go ahead, Dennis.

DR. BLEY: This first came up when we initially looked at this plant many years ago. Well, not this one. Just from you guys, or maybe somebody on your staff would want to comment.

Is there any effort moving forward to get these things qualified so the industry can use these more easily without having to make a big defense about them?

PARTICIPANT: So, I'll answer that, and if someone from NuScale needs to correct me, they can.

But there's a, essentially, like a 10-year class -- part of the testing is a 10-year program with the investigation that we worked, you know, NuScale's approximately 8 years into looking at these types of batteries. So, it just takes time.

So, there is an effort. But in terms of the timing of this review, it's not something that can be done during this review.

DR. BLEY: Yeah, I, I understood that. But I'm glad to hear that it's moving forward. It's not been a major obstacle that has caused a lot of extra work, I think.

PARTICIPANT: No.

CHAIR KIRCHNER: And for the record, we first looked at this in 2016, maybe when you submitted the topical report on that. Then it was called EDSS, I think.

PARTICIPANT: EDSS. Yeah, I remember.

CHAIR KIRCHNER: Right. Yeah, thank you.

MEMBER MARTIN: Member Martin.

It seems an obvious candidate for 10 CFR 50.69. But I have not heard that mentioned. You know, basically, you already did the work risk over and over again, which is -- but if you could test it by 10 CFR 50.69.

Why not? I mean, it almost seems like it's a paperwork exercise.

What's that? It's Part 52? Well, it must be a counterpart to this, Part 52.

But nonetheless, why not just capture it in that space? Maybe I'm ignorant about whether this exists in both, both spaces. But --

PARTICIPANT: Just ask the Department.

MEMBER MARTIN: Okay. Also, I'll just throw in, you know, I've asked a couple questions on the DSRS. It just seems like this is kind of what you can tackle, you know, and this whole discussion would not have been necessary if a little attention had been done on that. But that's, again, also a comment that's static.

MR. ROBERTS: I just want to mention we had a subcommittee meeting two weeks ago and the subject of electrical power work at the agency here, and this would be on batteries, obviously working on in the near term. And it all supports this application. But they are working on it.

MR. LYNN: Next slide, please.

So, one of the things we considered as part of our review was risk, as you just mentioned.

In SRM-SECY 19-0036, the Commission directed the NRC staff the IAB feature of the ECCS valves did not need to be assumed as a single active failure.

But as commented earlier, the SRM-SECY actually went further and identified that the staff in any licensing review or other regulatory decisions, not only that the staff could use review principles, but the staff should apply risk-informed principles when strict, prescriptive application of deterministic criteria are unnecessary to provide for reasonable assurance.

So, when it comes to this, again, you know, as you mentioned, it seems to be a bit of a painful work discussion. And, so, our perspective, NuScale's perspective is that the SRM-SECY certainly applies here when it comes to unnecessarily thinking about deterministically and prescriptively applying things, the SRM-SECY can be used to bypass all that and make the determination that the design is safe, that the classifications can exist without all the unnecessary paperwork.

Next slide, please.

So, in conclusion regarding the loss of power topic, the NuScale believes that the nonsafety-related classification of EDAS is appropriate.

The control of EDAs in the OCRM and under the maintenance rule program combined with augmented requirements is appropriate to ensure reliability and availability during operation.

The safety analyses considering EDAS either available or unavailable at the time of event initiation are sufficient to show that EDAS is not relied upon to mitigate design-basis events, consistent with its nonsafety-related classification.

The design-basis events do not require consideration of the so-called smart failure at the time of worst conditions.

But that even if a smart failure is assumed at the time of worst conditions, NuScale can show that the consequences of such a sequence are minimal and that core cooling is maintained.

Overall conclusion, right, that the removal of the IABs is driven by design motivation to make the plant overall safer.

And the Commission direction, in accordance with that, the Commission direction would identify that when you're trying to make the plant safer you shouldn't be hung up by strict, prescriptive deterministic criteria.

So, on the next slide we'll transition to --

MR. ROBERTS: Before you go there, I think we'll probably cover this next, too, but before we go there, I'll kind of get to what Gregory was asking about earlier, which is that the design rules pretty much do require you to assume single failures at the worst possible of the event sequence.

And I think what's key in this argument is that temporal, which you'd have a consequence evaluation. And the consequence is something you can live with. And since a consequence, even though it's, you know, it's a limited violation, you could live with the consequence of a limited violation. That's at least an important part of the story, and maybe the most important part of the story depending on how you parse this whole argument.

And, so, it seems like you see this whole set of conditions to make the argument.

And I guess the fourth bullet there isn't all that, you know, impressive to me because that, again, just my understanding of what you're always required to do.

MR. LYNN: This is Kevin.

NuScale would disagree that that's correct. Right? There was a mention of a smart failure with respect to, you know, fires, et cetera.

But that's a different set of criteria that's applied.

So, in terms of safety analysis, it's always been the traditional mode of operation that you only take one initiating event at a time. So, taking multiple initiating events at a time is beyond design-basis event.

MR. ROBERTS: Oh, I agree with that.

MEMBER HALNON: This is Greg.

When you're dealing with nonsafety systems, it's different. I mean, I agree with the safety-related single failure, you know, single failure at the worst possible times, initiation of the event period. So, when you're just dealing with nonsafety-related systems my sense is you get, look at worst case, it's not, there's nothing there behind it to allow you to say it's kind of last.

MR. LYNN: But in terms of the industry operating experience I would argue that that's not the case.

For example, if you consider an operating, a traditional operating PWR, consider a rod withdrawal event; right? You take a rod withdrawal and you withdraw all the way up to your peak power just before a reactor trip. So, let's say their trip set point is 115 percent.

So, if you have a traditional operating plant and you take their power all the way up to 114.9 percent, okay, and then at that time you assume a loss of power and a full close-down of all RCPs, superimposed on 115 -- 114 percent, 114.9 percent power, it would absolutely fail, too.

And that is not something that's addressed in the design-basis with any of those events.

MEMBER HALNON: That's very -- that may be right.

MR. ROBERTS: And I would argue that's not generally caused by a single failure.

MR. LYNN: But it's still in the perspective of Member Halnon's question, it's still a nonsafety-related system being credited continue to operate during that sequence.

MR. BRISTOL: Nor is EDAS failure a single failure.

MR. ROBERTS: No, I agree. As long as you've got the redundancy you've designed in, it's not.

MR. BRISTOL: Right.

MR. ROBERTS: No, I agree with that completely.

I think the question would be you've

chosen to operate with the redundancy removed. Then you start to get closer to where you are on a single failure event.

MR. BRISTOL: Correct.

MR. ROBERTS: And I know you've got some of this coming up in the slide. But in some way it's a semantics argument to some degree of a double initiating event was an initiating event compounded by a single failure. You know, sometimes initiating events are caused by single failure. You get into these, you know, arguments that I've been involved in before, I recognize.

But if you've got a single, you know, electrical system failure because of lack of redundancy or because you've chosen to remove redundancy, at least in my experience you've got to consider those to have occurred during the, you know, the transient, those initiated by the initiating event.

And there may be reasons why you don't, like that fifth bullet there, that the consequences having to deal with, and maybe, you know, case-basis exemptions for other reasons like the, you know, the two 19 SRM states. But it seems like we start with that's what's in the set of things we need to assume.

And that's where I -- again, and there's an example coming up with the slides. And, you know, we've got a couple more examples that we could talk about.

But the general theme is, you've looked at this area of the -- of the understanding then, combined with what you might call a single failure, or what you might call another event, whatever you want to call it, and find it's acceptable because there are consequences. In which case you've got a very good argument, I think, that is risk-informed, you know, exceptions at the single failure analysis because you've got it covered.

MEMBER HALNON: And don't take our questioning any more than violently agreeing with you.

MR. ROBERTS: That technicality that we're talk, that we're arguing is immaterial. I mean, like tom said, the consequence.

MR. LYNN: Yeah, this is Kevin.

I think the point, just to emphasize, is you can scan the industry and find examples where nonsafety--related systems are accredited to continue to operate during certain events, and they didn't analyze it the other way. And that's the point we're trying to make here, that what we're doing is not unique to NuScale.

MR. CROMWELL: Can I? This is Mike Cromwell. Can you hear me okay?

CHAIR KIRCHNER: Yes.

MR. CROMWELL: Okay. Gary.

MR. BECKER: Gary Becker with NuScale. I'm the Regulatory Affairs counsel.

So, I just wanted to, Member Halnon, give a little bit more specifics to your point because you used the phrase design rules. And that's kind of a central point in this conversation is that what you're describing comes more from staff past practice.

And as Kevin was getting to, there's, there are examples kind of on both sides of the traditional practice. But when you look at the actual regulatory rules, they are very specific on which failures you need to take, and which assumptions you need to make.

And that is, that is key to the argument here because, for example, GDC 17 the phrasing is "with power unavailable." But it's different than assuming a loss of power at any random time.

So, so look at it from a regulation perspective, we do not see a rule that requires this to be assumed. Perhaps in the application of traditional practice you could find something to support the notion, but that's where we'd say that that is what we are attempting to do. We can look at it through a risk-informed lens and demonstrate that it's not, it's not important from a risk perspective to take that additional failure that could be positive.

So, meeting the rules is the first prong. And then we can talk about the risk from other assumptions.

MEMBER HALNON: Thanks.

MR. LYNN: Next slide, please.

Here we cover a topic of some interest, I believe, to Member Roberts based on his questions from prior meetings, some of the prior ACRS meetings I believe it first came up potentially prior in LOCA, also in the non-LOCA meetings. We deferred discussion to Chapter 15. So, here we are today.

So, back onto that question. The ECCS valves have two in series safety-related trip solenoid valves.

The design is such that both of those trip solenoid valves much actuate to actuate ECCS. The purpose of that configuration is it presents a single failure, a single failure form causing inadvertent

ECCS.

However, both valves fail in safe position, i.e., the actuated position, so that configuration also ensures that as ingle failure does not prevent ECCS actuation.

So, we have, essentially, single failure proof in both directions.

So, the previous question, as we understood it from the previous meetings was what do you in the case where you have one solenoid valve already fails?

For the RVVs, if you operate it in that condition the subsequent failure of the other solenoid valve would cause that RVV to open.

For the RRVs, the IABs go in there and so they prevent the RRV from opening, even if the other solenoid valve subsequently failed.

So, if you have a known failure of a solenoid valve during operation, you are required to perform an operability determination for the supported ECCS valve under Tech Spec 3.5.1.

If the conclusion of that operability determination was that the supported ECCS valve was inoperable, in other words it was incapable of performing its open function, Tech Spec 3.5.1 would require the restoration of operability within 72 hours, or else the plant would have to be shut down.

If the determination was made that the supported ECCS valve is operable, in other words if it can still perform its function at open, Tech Spec 3.5.1 would have no specific time-limiting restrictions on that mode of operation.

So, conceivably, you could continue to operate in that scenario if one solenoid valve failed.

However, there is still the restriction in the licensing basis and the FSAR in Section 15.0.0.6.3 that requires an analysis to show that the expected frequency of an AOO with actuation of ECCS is once -- less than once in a lifetime event of an NPM.

So, in order to continue operating with one solenoid valve failed, you would have to show that you didn't increase the frequency of a valve opening event in violation of that requirement of the FSAR.

Next slide.

So, we've covered a lot of this material before in some cases.

MR. ROBERTS: Kevin, I was wondering if you have any thoughts of what kind of analysis would this require? How would you do that analysis?

MR. LYNN: So, the current analysis that's done is a, a PRA evaluation of frequency that uses inputs from the PRA and identifies all of the sequences where ECCS could open.

And, so, I can't remember what all, all of the events that add up to it. But one of them is loss of power supply itself.

One of them is failure of the solenoid valves, et cetera.

One is failure of the RSV, which would lead to depressurization, which then causes the valves to open.

But, essentially, all of those sequences, or some, and we get an answer that shows that the total frequency of that once, less than once, you know, one over 60 years.

So, in that particular calculation you would have to address that. So, the input that says here's the frequency of failure of a valve to open, if you only had a -- instead of being a two out of two you had a one out of one at that point, right, with one solenoid valve that failed, so that its frequency would be expected to increase for that particular contributor to that sum.

And, so, if you sum those and then show that you are more than 1 out of 60, that would be a prohibited condition by the license basis.

So, some of that would involve how long you're expected to be in that configuration; right? If you found out that that solenoid valve has failed in the, you know, 12 hours leading up to the -- your next outage, obviously that, that contribution is going to be negligible.

But if you discovered it an hour into your 8th month cycle, potentially that could change the math and show that you wouldn't be able to comply with that statement in the FSAR. So, that would be part of the, the math that goes into that.

MR. ROBERTS: This is the calculation that the operator can do as opposed to something you do at design time?

MR. LYNN: Well, this, so the, the calculation is already done to show demonstrations of the L&C 4.4 is done, as the design-basis, part of the design-basis.

This would have to be a, essentially, a risk evaluation, an update to that based on the operating information you have at the time. So, not by, not done by the operators, but essentially at the request of the operators as part of an operability determination.

MR. ROBERTS: And do you have any sense of what kinds of numbers might come out of that?

MR. LYNN: We have not done that particular calculation at this time to know what the, the limits might be for that, to what extent you could operate.

MR. ROBERTS: Yeah. I'm wondering if it might be just as restrictive as the 72 hours for the case of inoperability.

MR. LYNN: Right.

MR. ROBERTS: It depends on, obviously, what's in the PRA.

MR. LYNN: Correct. Yeah.

It would depend on the PRA. And, also, it would depend on the particular configuration at the time and their online PRA.

Another consideration, right, is the performance of other surveillances. So, throughout operation you have to perform certain surveillances. Plants typically don't like to be called as equivalent to a half Scram situation; right?

So, if you're an operating plant and you're half Scram, it greatly restricts your ability to do other surveillances, such to the point where eventually you can't defer those surveillances anymore.

And, so, even though you, you could operate somewhat indefinitely with a half Scram, in reality you can't do the other tech specs requirements.

So, we expect something similar here where at some point the ability to perform other surveillances might be compromised, in which case you would no longer want to operate there. And, in general, we wouldn't, just the same way an operating plant wouldn't want to operate half scrammed; right? They wouldn't want to operate half, actually, the ECCS.

It's certainly a lot easier to recover from a planned shutdown than a unplanned shutdown.

That covers the discussion on EDAS. Sorry for the extended discussion there. But we felt it was a topic worthy of consideration, some additional details, especially in light of some of the discussion potentially offered by the staff later in their presentation.

So, to conclude the Open Session for Chapter 15. To reiterate, all review questions have been resolved.

All the acceptance criteria have been met.

And the bottom line is the US460 NPM

design passively mitigates Chapter 15 events with a reasonable assurance of adequate protection for public health and safety.

CHAIR KIRCHNER: Members, further questions?

MR. ROBERTS: Yeah, one last question.

I think you skipped one slide. We talked about most of the content on it. We missed that.

So, if we need to go back, I'll follow up with the staff on the question of the interpretation of the repair criteria. Because I think that's actually an important discussion topic.

As I pointed out, there's a requirement, set of requirements in the regulation, including the front matter of Appendix A of 10 CFR 50. And there's a lot of, 50-plus years of practice. And I agree with you, it's not regulation, but the role of that also, you know, patches into what you presumably need to assume.

You know, I think the better question here is the contents of the consequence versus the enlightenment. I think you've done a thorough job there

So, thank you.

MR. LYNN: Thank you.

That concludes the presentation, if there's no other questions.

MEMBER MARTIN: Thanks. I just wanted to throw in a thank you for the staff shot you provided. Certainly very useful for us to, you know, very quickly glean through changes, and understand exactly, you know, what you're not only talking about today but, obviously, to support the final demo on Chapter 15. So, thank you very much for that.

CHAIR KIRCHNER: Okay. At this point we would transition to the staff. I'm just wondering whether we launch into it now or take another 15 minutes for lunch.

I think it would flow better if we just start again at 1:00 o'clock with the staff's presentation on Chapter 15.

So, with that, we have a little bit longer lunch hour. We have a mandatory stop for the committee at noon. So, that's the reason behind this decision.

So, with that, we are recessed until 1:00 Eastern Time.

(Whereupon, at 11:38 a.m., the above-entitled matter went off the record, and reconvened at 1:01 p.m.)

CHAIR KIRCHNER: Okay. Good afternoon. This is a meeting of the NuScale Design-Centered Subcommittee of the ACRS, and we are taking up Chapter 15 and the staff's review.

I am turning it to Stacy Joseph of NRR.

MS. JOSEPH: Thank you. Again, this is Stacy Joseph. I am the PM for the Chapter 15 review.

During the regulatory audit for Chapter 15, the staff generated 105 audit issues. Most of these issues were resolved during the audit. Following the conclusion of the audit, the staff issued eight RAI questions for Chapter 15, and all of those responses have been determined to be acceptable.

The staff completed the Chapter 15 review and issued an advance safety evaluation to support today's meeting. There are two significant changes in the staff's SE from the version that was submitted to the ACRS in early March and the SE submitted on March 25th.

Over the last month, the staff updated Section 15.0.5 related to the extended passive cooling. Over that time, several RAIs for the SSC topical report were resolved, and the Chapter 15 evaluation was updated to reflect resolution of those issues. In addition, Section 15.6.5.3 on beyond design basis events was revised to reflect closure and evaluation of the LOCA break spectrum open items and its related exemption. These sections will be discussed in detail later in the presentation.

I would like to thank the technical staff listed here who contributed to the review of the Chapter 15 and completion of the safety evaluation. We have two presenters denoted here, who unfortunately are not able to join us today, Antonio Barrett and Ryan Nolan. Their colleagues, Adam Rau, Josh Miller, and Sean Piela have stepped up and will be presenting in their absence.

Since Chapter 15 is extensive, we have two sets of reviewers for today, and we will be switching out halfway through. For the first set of presenters, you have already been introduced to Adam Rau and Zhian Li, so at this time I will ask Josh Miller and Tom Scarbrough to introduce themselves.

MR. MILLER: Hi. My name is Josh Miller. I’ve been at the agency for about 17 years, and I’m in the Reactor Systems New Reactors Division.

MR. SCARBROUGH: I’m Thomas Scarbrough. I’ve been at NRC for quite a long time, and I’ve been helping out on the PDAS aspect here. Thanks.

MS. JOSEPH: Thanks, Tom.

We'll introduce the next set of presenters after the switch. Again, this is Stacy Joseph.

There are 11 sections in Chapter 15. Similar to the approach taken in Chapter 4 presentation, the staff will not present slides on every section but will instead be focusing their presentations today on specific portions of the application.

There are a number of differences in the design and also methodologies that impacted the review of Chapter 15. These changes include power uprate in version base model -- version and base model changes to NRELAP, ECCS valve design, and the number of valves, ECCS actuation and new riser level actuation, crediting DHRS for LOCA and LOCA-like events, no return to power during extended passive cooling, the addition of the ECCS supplemental boron feature and additional riser flow holes, and a change to DC power availability assumptions and reliance on the augmented DC power system, also known as ES.

Today's presentation will discuss most of the areas of change but will also hit on some key chapter events and issues. Staff will start with the implementation of the extended passive cooling topical report in Section 15.0.5, and then move on to the rod ejection analysis, which includes implementation of the limitations and conditions from the rod ejection topical report.

We will circle back with that and discuss boron dilution and specific cooldown and reactivity events. Staff will discuss their Chapter 15 review of EDAS, which includes a staff differing opinion, as Michelle mentioned earlier today.

Following the EDAS discussion, we will switch our group of presenters who will discuss key aspects of steam water line break, steam generator tube rupture, inadvertent operation of reactor valve, and, finally, LOCA.

All right. Enough of the introductions. We'll start with staff presentation with extended passive cooling.

DR. RAU: Thank you, Stacy. Again, this is Adam Rau. I'm here to present the staff's review of the extended passive cooling calculations in Chapter 15. Other folks have been involved with this portion of the -- of the review and should be either in the audience or on the line to help me potentially answer any questions.

So the calculations focus on three acceptance criteria that are named in the extended passive cooling topical report; namely, the collapsed liquid level, the RPV, FER -- excuse me, the collapsed liquid level and the RPB riser remains above the top of the active fuel, that the reactor core remains subcritical, and that coolable geometry is maintained because the boron concentration in the RPV remains below the solubility limit for precipitation.

I'll be highlighting some aspects of the staff review of this analysis in the following slides. So starting with the first acceptance criterion, the staff results -- or, excuse me, the applicant's results showed that the steam generator tube failure is the event leading to the minimum collapsed liquid level. Staff performed an independent confirmatory analysis of this event, which indicated that the minimum level analysis was performed conservatively.

Staff found that limitations and conditions on the topical report are relevant to the acceptance criterion, to this acceptance criterion, and went back to the applicant's analysis. Additionally, the applicant's results show that in this event the collapsed liquid level remains 1.8 feet above the top of the aptitude arrangement.

Limited case for the boron precipitation criterion is an inadvertent opening of an RVV. Staff found that the calculation was conservative, assumptions for thermal hydraulic conditions. Staff performed confirmatory analysis and sensitivity studies, which supported the amount of mixing in the applicant's analysis necessary to keep the boron concentration below limits in the core.

Staff also noted that the calculation assumes an initial RCS boron concentration at the maximum operational limit, which provides some conservatism as the system would only be expected to operate near this RCS boron concentration for a limited period of time.

Based on the -- so summarizing the results of the analysis, the minimum margin that the applicant found was 6,250 ppm with the core peak concentration around 8,500 ppm.

CHAIR KIRCHNER: Adam, could you just address the confirmatory analysis? Any confirmatory analyses that you did on those first two categories?

DR. RAU: So on these two categories I wasn't personally involved in the confirmatory analysis. But I know that for -- at the very least, the minimum level analysis, we performed confirmatory analysis and RELAP.

CHAIR KIRCHNER: I thought the RELAP

results had shown more margin than 1.8 feet, or am I misremembering that?

DR. RAU: I thought it was more than 1.8 as well.

CHAIR KIRCHNER: Okay. When you present a number like that, though, then, you know, the figure of merit here is to have the collapsed liquid level above the active fuel.

DR. RAU: That's right.

CHAIR KIRCHNER: So this is -- this is -- how would you assess this? This is -- you have good confidence that this is a conservative result? I'm trying to get the NRC's assessment of this. 1.8 feet, what do you do with that number? Does this -- are -- do you have a high confidence that their -- they've met their figures of merit? Just put it in terms of regulatory assurance for the public.

MR. THURSTON: This is Carl Thurston with NRC staff. So, yes, so we conducted sensitivity analysis using the RELAP code and using the applicant's modeling. We also completed confirmatory analysis for the TRACE code by Research -- staff in the Office of Research.

CHAIR KIRCHNER: And did you get similar results? Did you get more conservative results?

MR. THURSTON: Yes. But we --

CHAIR KIRCHNER: Just for the public, you know, where were your results vis-à-vis the applicant's?

MR. THURSTON: Yes. So -- yeah. So the applicant's results we think are very conservative.

CHAIR KIRCHNER: Okay. Thank you.

You probably see where I'm going with this. I mean, what does the general public make of 28 parts per million? So give us some context of your assessment.

DR. RAU: So I guess going through and, yeah, commenting on the subcriticality analysis specifically, so 20 parts per million, as the applicant's results --

DR. LI: Adam, can we answer?

DR. RAU: Sure.

DR. LI: Thank you, Chairman. I think I understand your question. Pertaining to perspective what the 28 ppm means, I did an estimate, not based on actual calculation. So it is roughly equal to .0056 K effective. That's roughly about .8 parts in reactivity, net reactivity. That's the equivalent. That means you have .8 net reactivity, you have a reactor that was safely shutdown.

CHAIR KIRCHNER: So this is the applicant's result or this is your RELAP map script result.

DR. RAU: This is the applicant's.

CHAIR KIRCHNER: Applicant's, not yours, right?

DR. RAU: And there are other conservatisms in this case for the analysis that the staff is basing the finding of 28 ppm. Qualitatively speaking, this is somewhat low, but there are other aspects of the calculation.

So, for example, the staff reviewed the implementation of the new PRA reliability factor and the subcriticality analysis. There are SR requirements that there is minimal non-condensable gas in the CNB. There are -- there are other aspects as well related to conservative assumptions as far as the speed of the cooling, increase the critical boron concentration over the transient as well, so --

MEMBER PALMTAG: This is Scott Palmtag. Just to follow up on that, do you have an idea of what the margin of error is when you project critical power -- critical boron concentration in a PWR?

PARTICIPANT: So we know that for some operating PWRs the -- at the very least, the code uncertainty can be about 100 ppm.

MEMBER PALMTAG: So, in that perspective, 28 ppm is very small. The uncertainty is 100. I was thinking 50, but I -- I've heard -- you brought up some good points about there is a lack of conservatisms in there, and I've heard that from the NuScale people, too.

But it would be nice if those could be quantified. It's just it's hard to quantify that when it's, okay, 28 ppm is low, but then there's other conservatives -- conservatisms. So it's hard to understand just, you know, what that means unless those other pieces can be quantified, so we actually know how much conservatism is in there.

PARTICIPANT: In this case, the 28 ppm, it does include the applicant's work to address their code uncertainty. It includes the NRF I guess in that number itself.

MEMBER PALMTAG: That's one piece, right?

PARTICIPANT: Yes.

MR. THURSTON: I can tell you -- this is Carl Thurston again, NRC staff. So --

CHAIR KIRCHNER: Carl, just -- you need to get closer to the mic, please.

MR. THURSTON: I was going to say that we do have additional information that the staff will show in closed session, which is more straightforward and quantifies the uncertainty about things.

MEMBER PALMTAG: Yeah. We're going to talk more about this in the closed session. I don't know I don't know if you want to comments now or wait.

MR. GRIFFITH: Yeah. This is Thomas Griffith. I was -- I was going to add to that that NuScale also has some additional information with some better quantification in the closed session that we can get into those specific details.

MR. THURSTON: Okay. Thank you.

DR. RAU: So, in addition, I have a few slides discussing some aspects of the extended passive cooling analysis that was of interest to ACRS members during the previous meeting. So one condition in the topical report was that a test must be performed to demonstrate acceptable performance of the as-builts, ECCS supplemental boron system.

So this slide is showing the FSAR markup that establishes the requirement to perform a first-of-the-kind test in order to meet this condition on the topical report.

Next slide, please.

MEMBER PALMTAG: This is Scott Palmtag. Yeah. We talked about this in the SPC review Committee meeting, and this came up. Please help me understand. It sounds -- verify ECCS supplemental boron pellets dissolve following ECCS actuation. So how do you show that? Do you actually have to have an ECCS actuation with steam in the system to show that? Or are you thinking there’s a different way of showing that?

DR. RAU: So that’s my understanding of the -- of the test, is that it will be in prototypic conditions.

MEMBER PALMTAG: Will the core be operating? You’re going to have to have steam? That’s where I’m confused at. So you’re going to have to have an operating -- the core is going to be operational and you’re going to do an ECCS?

MR. GRIFFITH: Yeah. This is Thomas Griffith. So one of the -- one of the tests that we do, we build at the core operating, use the module heat-up system. And Tyler Beck, if you’re on the line, you can -- you can add in here a little bit if you -- if you need to.

But, effectively, use module heat-up system to get as high in temperature pressure as you can, open the ECCS valves, and perform all -- perform a blowdown.

MEMBER PALMTAG: Okay. All right. Thanks. Yeah. I had some questions during the review, but I'm -- my understanding is NuScale and NRC both agree that this test is doable, so I'm not going to pursue them any further. Thank you.

DR. RAU: The extended passive cooling analysis assumes a minimum initial core boron concentration based on the pre-transient operating history as NuScale has discussed in their previous slides.

Since the pre-transient operating history can affect the level of decay heat related to initial non-related heat delivery of xenon concentration during the transients, the topical report includes conditions that the technical specification and LCO should be established to reflect this operating restriction.

So in reviewing the Chapter 15 SR, staff found that the applicant's LCO of 3.5.4 meets this requirement. The LCO requires operability of the ECCS supplemental boron system. Condition A of the LCO is that the ESB operational limits which are established in the COLR not met. So the LCO includes a condition for -- requires the limits to be met.

The LCO is supported by surveillance requirements, the RCS boron concentration be within the ESB operational limits, and the tech spec bases provide a description of the purpose of these operational limits. And so staff noted that the example operating limit that's shown on the next slide was also -- was provided in the FSAR and is required with the analysis for this event as well and consistent with factors.

And so this slide shows NuScale's example. COLR limit on the RCS boron concentration is provided in the FSAR. So this is a limit that would be developed on a cycle-specific basis, shows the limit on the RCS boron concentration based on the integral downpower for operating history.

The applicant analyzed a wide matrix of cases in order to support the development of this limit. You can see the higher integral downpower, the RCS boron concentration rises as these cases allowed a lower decay heat during the transients. And, additionally, the note defines a separate limit that and an ultimate minimum boron concentration and the power ascent rate is of a pressure hold.

MEMBER PALMTAG: Scott Palmtag again. I

just had a question about this curve. Maybe you can help me understand it. So if you're operating the core, you're towards the end of cycle at 100 ppm, and you have some integral downpower, and that moves you into that not-allowed mine, you have to shut down. Is that what this means?

DR. RAU: So the action that -- I believe is to be in mode 2 in 24 hours if they're not within these limits.

MEMBER PALMTAG: I'm sorry. Can you explain what that is?

DR. RAU: Oh. They would -- yeah. They would have to be subcritical in 24 hours.

MEMBER PALMTAG: Okay. And that would add more integral downpower, right? So, in essence, if they get into that situation, they would have to shut down for the cycle.

DR. RAU: That's right.

MEMBER PALMTAG: That seems pretty restrictive on the cycle. That could -- you could end your cycle really early, if I understand this correctly. Okay. But thank you for clarifying.

DR. RAU: That is what I have for the extended passive cooling Chapter 15 analysis. I'll pass it on to Dr. Zhian Li to talk about the rod injection.

CHAIR KIRCHNER: So before you go on, let me plant a question, a hypothetical question. Maybe you can address it in the closed session. What if this boron dispenser is so efficient and effective that you get a very high concentration in the lower plenum of the containment vessel? Is there a danger that you could hit the precipitation limit and drop all the boron into the bottom of the containment? Have you looked at that?

DR. RAU: So I believe that that's among the set of cases that NuScale analyzes for the precipitation limit and include a very fast bias on the boron dissolution rate.

CHAIR KIRCHNER: But did you look at that?

DR. RAU: I don't believe we performed confirmatory analysis on that, but --

CHAIR KIRCHNER: All right. Thank you.

DR. LI: All right. Good afternoon. This is Zhian Li -- Zhian Li again. Myself and my colleague who -- we reviewed the rod ejection calculation, which implements the rod ejection topical report and methodology described there.

So, basically, the -- we review the -- NuScale's calculation and their assumptions. We assumed the most reactive rod was stuck -- stuck out. So that's one major assumption that's a conservatism. The other assumptions, the MPS, the module protection system, in actuation. So they basically rely on the module protection shutdown and reactor where you have the rod ejection passage.

And also, NuScale's code assumed the pressurizer sprayed down, which would delay the heat-up and then the pressure increased. So they would keep it high, even more limiting on the minimal critical heat fluctuation.

A NuScale study at the zero power, 20 percent of power, 50 percent, there's several cases, and also look at the BOC and the EOC in the cycle, beginning of cycle, middle of cycle, end of cycle, to determine the most sensitive or most limiting case. Then they did further analysis.

And then, so based on the analysis, the minimal critical heat fluctuation was the 3.13. The limit is 1.43. That's the minimal acceptable. So once you have the sufficient margin, the peak reactor cooling system pressure is like 2,231 psia.

Well, this is an estimate. I don't think this is -- there is some uncertainty associated with this, but we still have sufficient market because this is relative, short, present. And, therefore, you see their rod get ejected at the end of reactor shutdown. So there is -- really, the reactor pressure would not go too far, too high. This is very reasonable, you know, accurate result.

And also, look at the peak radial enthalpy and the PCMI enthalpy limit, threshold, and the all -- and the peak fuel temperature, where they all meet the figure of merit or limit with sufficient margin.

That's pretty much the calculation result, and then, you know, the NuScale calculated to the -- with some delay on the calculation assume the reactor shutdown would -- or reactor trip, there would be a delay, and I have some -- again, I will, you know, not get into those detailed numbers, but there is some additional conservatism. The staff feels that it is really more conservative, gives you more kind of conservative result.

The key of this methodology is the implementation of the rod ejection methodology, which was reviewed and approved last year by the staff, and we presented that to the rod ejection topical reports here our review. So there are three limitations and conditions.

The first one, in order to use that

methodology present in the topical report, the user of that methodology has to demonstrate they meet the NPM 20 design, because at that time the methodology was really developed based on the NPM-20 design features. That's why we have that here.

And then, it's basically another limitation that's -- we want to make sure the control rod has not experienced the -- or the -- yeah, the -- so the core design was based on -- baseload operation rather than the load follow operation when you have a load follow and you have substantial manipulation of the control rod and of core power.

Therefore, you have the potential baseload operation of the core and the control rod work, too. So that's a concern.

And then, so the first -- the third one will be the same for the statistical subchannel analysis methodology. That was part of the reference -- incorporated by reference the methodology and the rod ejection methodology. So we looked through -- into this one and we find that the analysis all meet all the requirements.

Basically, you look at the NPM-20 design. The methodology was developed based on the NPM-20 design, and then the NuScale used the statistical subchannel methodology, and that's exactly what they -- the limitation and the limitation will be requiring.

And also, this course design for baseload analysis. In the future, if they want to, would have to do some additional analysis to address this concern. That's what -- based on this review, we think the rod -- NuScale has followed the rod ejection methodology that is approved, and then the calculated results demonstrate the need to -- the figure of merit and the conservative.

That concludes my presentation. Thank you.

DR. RAU: This is Adam Rau again. I'll be discussing a few additional Chapter 15 events. We have highlighted the 15.4.6, boron dilution transient, because it's somewhat unique for the NuScale design compared to operating BWR. So this event looks at a CVCS malfunction gain to a dilution of boron in the coolant.

The analysis described in this section evaluates the remaining shutdown margin before automatic isolation of the dilution source, somewhat different than operating BWRs because typically in those analyses you would see that alarms give operators enough time to stop the dilution, and that would be the acceptance criterion for the event.

The NuScale design doesn't credit any operator action during an accident transient scenario and so here the dilution source is isolated based on safety-related demineralized water, isolation valve signals that actuate on any reactor trip signal.

So here NuScale's analysis of the event considers CVCS malfunction when the module is in every operating mode, and then in mode 1 considers operation for -- at zero power to full power, which is consistent with the standard review plan for this event.

During the review, staff noted that the calculations in Mode 5 and some of the slower Mode 1 dilution events appeared to credit operator actions to secure the dilution source and terminate the event. So staff raised questions about whether operator actions were credited in other longer duration events as well as other Chapter 15 transients.

Based on this, NuScale revised the calculations as necessary to ensure that operator actions were not credited.

Next slide, please.

Then, based on their revised calculation, there were some slight changes to the boron dilution transient. The Mode 1 response is dependent upon the time in cycle. At beginning of the cycle, initial boron concentration is high, so addition of unborated water causes a generally greater change in boron concentration, getting to more rapid reactivity insertion. Additionally, moderator temperature coefficient is near zero, so a larger change in moderator temperature is needed to offset a given reactivity insertion.

Because of these effects, we tend to see earlier reactor trips, even -- or less water isolations. Their end cycle response is lower. Because of this, NuScale proposed a simplified method for evaluating later points at the end of the cycle. It's based on a high pressurizer level trip.

In this method, the high pressurizer level due to dilution is used to identify the condition of the high pressurizer load trip, and this is based in part on the NPM design prohibiting automatic letdown when the demineralized water system is not isolated.

So, with this approach, the total reactivity insertion is greater when the initial boron concentration is greater earlier in the cycle. NuScale performed this analysis assuming a bounding initial boron concentration, which is really more representative of middle of cycle conditions to assure that later points in the cycle would be addressed by this analysis.

With the limiting case, 47 pcm shutdown margin remains when the demineralized water system is isolated. Again, based on earlier discussions, this does seem low, but I did want to note that the -- there are conservative assumptions inherent in the analysis itself.

So, for example, this analysis assumes an initial shutdown margin at NuScale's analytical limit when, in reality, if it were performed based on the equilibrium cycle, it would have substantial additional margin -- shutdown margin criteria.

So based on this, I was able to find that this was consistent with the regulations.

I'll pause for questions.

Okay. So next slides cover cooldown and reactivity events. Starting with the reactivity events, the limiting AOO is a static -- or, excuse me, was the control rod misoperation, which was evaluated in Section 15.4.3 of the FSAR. This evaluation comes as two different types of events, including static misalignment of control rod assembly, as well as a single control rod assembly and a control rod assembly drop that is either a bank drop or a single CRA drop.

For these events, staff audited the calculations and confirmed that the non-LOCA evaluation model was followed. Staff also audited the subchannel analysis of these events. The limiting case for the minimum critical heat flux ratio is a static misalignment of the control rod assembly. In particular, this case occurs or is evaluated at 100 percent within an hour of one CRA inserted six steps past the 20 percent power or minimum insertion limits, with all other CRAs fully withdrawn.

Scenario B, the regulating CRA is left behind during startup while all other CRAs are withdrawn. The set of scenarios evaluated is consistent with the setup that was evaluated during the NPM-160 review. Staff did not identify any changes or additional scenarios based on design changes.

Then, for the linear heat generation rate, there can be -- linear heat generation rate acceptance criteria, limiting case is a single CRA withdrawal. Limiting case was initiated, 45 percent rate of power, a reactivity insertion rate of roughly one cent per second. In this case, reactor trips, secondary system isolation, and DHRS actuation were initiated on the high pressurizer signal.

Next slide, please.

Cooldown events. Limiting cooldown events and increase in steam flow evaluated in FSAR Section 15.1.3. The increase is caused by an instantaneous opening of the turbine bypass valve. Full opening of the turbine bypass valve in the NPM 20 design could lead to up to 100 percent increase in the steam flow, so the range of steam flow increase is analyzed and the event is quite large.

Again, staff audited the applicant's detailed non-LOCA and subchannel calculations and confirmed they followed the respective topical reports.

In this event, cooldown produces a temperature and coolant in the downcomer which affects the calibration of the x-square detectors which are sort of used to assess the high power rate and high power trip signals. Accordingly, NuScale adjusts the high power and high power rate trip signals in their analysis to account for this effect.

The limiting new critical heat flux ratio occurs when power peaks, roughly 120 percent rated from the power for the limiting case. In this particular event, the power would stabilize below the high power trip setpoint all in the peak, so the event would conclude without causing a reactor trip.

Here the time-dependent reactor power for this case is pictured on the right side of the screen.

So, additionally, while most loss of power scenarios would terminate these events, in this case the EDAS system is relied upon to remain function during these cooldown and reactivity events.

So I will hand it to Josh to discuss this in some additional detail.

MR. MILLER: Thank you. This is Josh Miller again. So due to the removal of the IAB valves from the RVVs, the augmented dc power system, EDAS, is now directly supporting the ECCS valve function to remain closed when a valid actuation signal is not present.

This raised concerns regarding the design and safety classification of the system resulting in the identification of the high impact technical issue. Reliance on valve-regulated lead acid batteries is the first-of-a-kind application in a nuclear powerplant.

Operating plants and other nuclear facilities typically use vented lead acid batteries, which have a proven record of capacity, capability, and reliable performance, because ERLA batteries are not typically used in standby applications at nuclear powerplants, which is how they would be used in the EDAS. Additional review is warranted to ensure a reasonable assurance of public health and safety.

Therefore, an interdisciplinary review team, or IRT, was established. The team was comprised of reactor systems, electrical, and PRA reviewers. This team put considerable effort into performing Be RiskSMART and RIDM evaluations to address regulatory and technical issues in a risk-informed manner to address the appropriate scope for the regulatory treatment of EDAS.

Based on its review of the FSAR and audited documentation, the staff determined the EDAS is related on in the safety analysis to perform, at a minimum, the following safety functions. Relied on assure the integrity of the reactor coolant pressure boundary during power operation, and relied on to ensure the SAFDLs are not exceeded during certain AOOs.

EDAS has augmented quality and was evaluated in Chapter 8 of the SER. Staff differing view was raised during the review and will be discussed in the following slides. The staff's initiated exemption to safety-related requirements in Chapter 8 is a potential option under consideration to address the differing view.

Next slide, please.

MEMBER ROBERTS: Yeah. Before you move on, a question I asked the applicant, and I want to ask you guys now is, EDAS seems like only one of several potential causes for spurious actuation of an RPV during this event.

Did you conclude that the single failure criterion would not apply to any of the other scenarios? Things like intentionally operating where the solenoids trip and then circumventing the system or failure in the MPS that makes the MPS blow with a single failure, you know, tripping the solenoids.

MS. PATTON: This is Becky Patton. I am the reactor systems supervisor. Let me see if I understand. So I think the scenario that we're talking about here is like you -- you know, you have an AOO, like a heat-up AOO. So that's the actual AOO that's occurring, like the one shown in the previous slides.

And then, so you're at an elevated power level is where you end up, because you didn't -- you didn't trip out at your MPS setpoint. So, at that point in time, the EDAS system is holding, right, the RVVs closed, right? You don't have that loss of power.

So you're asking single failure of that system or how single failure applies during --

MEMBER ROBERTS: Single failure of any system at that point, not just EDAS.

MS. PATTON: Okay.

MEMBER ROBERTS: Because the control system is also holding those valves energized, is holding those energized. And a spurious trip in the control system would also cause the same consequence, which is generally held down because there is two trip valves, and redundancy in the MPS, but there is also allowances to bypass that redundancy for maintenance.

So it would seem like you'd get to the exact same scenario, so it doesn't require EDAS to be the failure, that there are control system failures that would cause the same consequence.

MS. PATTON: Okay. So just in terms of, first of all, just to be clear on the single failure criteria, right? That's applied to safety systems, right?

So, you know, I think we heard questions earlier, too, about how you treat non-safety systems during progression of events, and that's more typically what -- you know, if the -- if the non safety system is needed to remain functional for the event to progress, like it would otherwise fail and the event would just terminate, then you assume it -- you know, it hangs on, right, for the event to progress.

But, you know, so single failure is the way we looked at safety-related systems, and then there's a requirement to take a safety -- you know, single failure during the event's progression, right? So it's a little different how you deal with it in terms of the non-safety aspect.

So I don't know if Tom wants to say anything about the valves specifically or how they --

MR. SCARBROUGH: Sure. In terms of these, with the removal of the IAB, inadvertent actuation --

CHAIR KIRCHNER: Tom, identify yourself for the -- identify yourself for the --

MR. SCARBROUGH: Oh, I'm sorry.

CHAIR KIRCHNER: -- court reporter.

MR. SCARBROUGH: Thomas Scarbrough, NRC staff. In terms of the -- with the removal of the inadvertent actuation block valve, you know, these two valves in the EDAS system are now your primary reactor coolant pressure boundary. So anything that disrupted the current to those solenoid valves, if it did it to both, you know, they both would go down.

But if you -- if you're able to separate it, right, so the -- the failure that you're talking about, if it's only on one train, you have the other one to protect it. So that's -- you know, that's part of our concern that has been raised by the staff is -- is that, you know, what are the potential possibilities for both of those valves to open now that the IAB valve has been removed from the system.

So that's what -- that's what was the major change in this system from DCA. You know, we had a lot of discussions about the IAB valve and its proof of concept and testing the target rod, and we went through all of that, to demonstrate that it could hold those -- that pressure until the system dropped down to 900 psi, something like that.

But now you have the EDAS system. It is the main protection of the reactor coolant pressure boundary, and so all of those types of questions that you're raising in terms of, what are the possibilities for it to lose current to both of those solenoid valves, is part of the discussion of the reliability of the system. So that -- it all goes into that.

MEMBER ROBERTS: Sure. I understand all of that, and I need to ask -- it would certainly be a way to cause the two trip valves to trip. You know, I want them to -- I know it would be you're operating intentionally and going out of service. Now you've got single failure potential in the other valve.

That's the only thing left holding the solenoids energized. Or it could be that you kind of compromise in the protection system because you pulled a card down for maintenance and now you've got one remaining card that could fail and do the same thing.

So I'm just wondering why you're focused on EDAS. It seems like the real problem is anything that would cause the scenario you're talking about.

MR. SCARBROUGH: Yeah.

MEMBER ROBERTS: Is there something about the single failure criteria that you think does not apply here?

MR. SCARBROUGH: Well, since it's -- since it's a non-safety system, you wouldn't officially apply the --

MEMBER ROBERTS: From what Becky said, the MPS is a safety system. But then, in that case, you would be looking at, you know, failures in the safety system.

MS. PATTON: Right. So you've got to separate, right -- the EDAS system is non-safety, right? So whether it's one failure in the EDAS system or multiple failures, it's not safety. Okay?

Other things -- the other systems are safety-related, the trip valves, MPS, and everything like that, right? And those -- you know, when you take a single failure of those -- and somebody -- you know, Thomas can correct me if I'm wrong -- but those, you know, have redundancy built into their design. Right?

So, you know, in -- for those cases, but the EDAS is what was focused on because that's a non safety system, the failure of which can cause them to open. And, again, you don't just say one failure on an --

MEMBER ROBERTS: Yes. I understand the difference. And if you were looking at the redundancy of the safety systems, that's great. It meets the single failure criteria, until you decide to operate unrestricted for an extended period with one of the redundancies out of service.

And you could incorporate requirements like a time limit to minimize the risk, you could incorporate operational limits to minimize that risk, you could leverage other requirements, which is what the staff I heard -- or, I'm sorry, the applicant heard on it this morning, said they are leveraging another requirement to minimize the time to be vulnerable to an inadvertent, you know, ECCS, which would seem like another way to apply.

But it seems like there ought to be some, you know, accounting for those scenarios, and, you know, not just focus on the one non-safety system.

MS. PATTON: So there are, you know, technical specifications in place, right, for, you know, you have -- and I think we have somebody from tech specs online that can help me out on this if necessary. But there are operability requirements like on the valves, right?

So if -- you know, and the same -- there are ones for control systems, RPS, things like that, right? So if you have something out of service, there is a certain time limit that you're only allowed to have that. And so that's -- it's because, you know, those are limited specifically because, you know, they are protecting the initial conditions of transient accident Chapter 15 analysis. So that's why those are set up that way for those -- you know, those safety-related systems.

MEMBER ROBERTS: All right. But in terms of time limit, because the safety function on the RVVs is considered to be, can they trip, you know, not can they be helped, that's what we heard this morning.

Now we also heard there is an ancillary requirement to minimize the potential period for ECCS actuations, which might cover this, but, you know, this is a "might."

I also want to make sure that the staff has looked at that. I'm still not sure we -- you know, that you have.

MS. PATTON: Well, I think -- yeah. I think so the EDAS and whether that's, you know, where my tech specs is separate. Then, when you're asking control systems, RPS setpoints, other failures that could happen, then I think that the conclusion was that, you know, the tech specs will cover that. Right? They have, you know, time limits on different aspects, just like, you know, every other plant, right?

MEMBER ROBERTS: Okay. Yeah. The other scenario is the operator inadvertently had issued ECCS. That would be presuming an error -- in combination because no procedure was found to do that. But on the other hand, unless the operators understand that there is a potential downside, you know, taking a safety action would potentially seem like the right thing to do during an unexpected event. How hard has that been looked at?

MS. PATTON: I don't think we have our human factors people here, but they do look at things like, you know, operator actions for commission and, yeah, there are certain criteria for how complicated, you know, those actions need to be in order to be considered.

MEMBER ROBERTS: Okay. Then I guess I'll stop my questioning now. It's just the focus on EDAS seems to me to be asking the wrong question. And EDAS is certainly a part of the question, but the real question seems to be, have you looked at the potential of inadvertently initiating an RVV actuation and, you know, looked at that more holistically. And, as a consequence, which we heard this morning, is pursuant not to your consequence, and the likelihood blowing up to the fact of poor judgment and not just focus on EDAS. So I guess I'll stop with that.

MR. MILLER: Okay. Josh Miller again. AT this point, we are going to pause the presentation on the staff's Chapter 15, safety evaluation, and spend the next couple of slides discussing the staff's differing view related to the augmented DC power system.

At the end of the -- of last year, a lot of the members and staff from five different disciplines filed a non-concurrence on the Chapter 15 safety evaluation report. Specifically, the staff raised concerns regarding insufficient technical or regulatory basis for the acceptability of the EDAS classification and its regulatory controls.

Next slide, please.

Specifically, the staff determined that EDAS meets the definition of a safety-related SSC because it's relied on to maintain the integrity of the reactor coolant pressure boundary and is relied on to achieve and maintain safe shutdown conditions.

In addition, ECAS meets 10 CFR 50.36 for establishing an LCO. Specifically, it meets criterion 2 because power from EDAS to the RVVs is an operating restriction that is an initial condition of a design basis transient analysis that either assumes the failure of or presents a challenge to the integrity of the fission product barrier.

In addition, EDAS meets criterion 3 because it is a system that is part of the primary success path and which functions to actuate to mitigate a design basis transient that either assumes the failure of or presents a challenge to the integrity of the fission product barrier.

Requiring an LCO for EDAS would be consistent with the Commission's final policy statement on technical specifications of prudence for nuclear power reactors. Absent appropriate LCOs, the facility will not be restricted to operate in a manner that is consistent with the reliability and availability assumptions contained in engineering and safety analysis.

For example, an SSC is not viewed as single failure-proof if there are not operability requirements for the system channels, divisions, trains.

The non-concurring staff also raised concerns related to management's decisions made early in the SDAA review on the acceptability of EDAS because it did not provide technical -- defensible technical or regulatory basis and was not conducted in accordance with applicable policies, procedures, and regulations.

We do not plan on presenting any further on this specific item today, but my understanding is that the ACRS members have access to the non-concurrence filed by the staff and can read it if they are interested in further details.

With this said, staff recognizes that EDAS has the lesser importance of safety compared to typical Class 1E power systems. However, it still plays a role in safety and defense-in-depth by protecting multiple fission product barriers. Therefore, the non-concurrent staff believe it is appropriate to resolve this SSC classification issue with the use of risk-informed exemption or existing risk-informed classification process, such as 10 CFR 50.69.

To ensure EDAS is reliable, as assumed in the NuScale analysis, non-concurring staff have proposed several purchases to address this, including review the qualification testing that would provide assurance that the batteries can perform their intended function and demonstrate reliability during their service life.

Also, the establishment of inspections, tests, analysis, and acceptance criteria, ITAAC, would enable NRC staff to determine reasonable assurance of public health and safety for use of the first-of-kind batteries after installation but prior to initial plant operation.

Regarding technical specifications, the non concurring technical staff believes simple modifications to the technical specifications and supporting documents such as the bases, in lieu of dedicated electrical specifications traditionally seen and other technical specifications, would be sufficient to ensure proper operability requirements with the RVVs.

To summarize, an approach that addresses both the classification and technical specification issues is important not only for making a regulatory finding on the licensing matter at hand, but also for ensuring that any future changes will be appropriately controlled.

Significant experience with construction and operations of nuclear powerplants shows that changes to the design and operation of the facility is highly likely. Thus, an efficient means for evaluating and controlling changes to maintain reasonable assurance for safety is desirable.

Absent such a framework, it is unclear how a future licensee would be accountable for ensuring that any relevant changes would be consistent with the associated risks, given the lack of specific controls such as tech spec LCOs and associated surveillances or

ITAAC requirements.

MS. PATTON: This is Becky Patton again. We have an additional statement by one of the staff that's online. Sheila Ray, if you want to go ahead at this point between these slides.

MS. RAY: Thank you. My name is Sheila Ray, senior electrical engineer. I just wanted to clarify some statements about the qualification of the VRLA batteries. There is no IEEE standard related to the qualification of VRLA batteries, and we understand that NuScale has an ongoing test program that they're in year eight of ten.

We understand a very, very high level concept of a qualification for VRLA, but we don't have any details. So I just wanted to make that clarification that staff hasn't seen that information, and we have not been able to conclude on the VRLA batteries that they would -- or we don't have details on the qualification of those batteries.

Thank you.

MR. SCARBROUGH: Josh, before you leave that slide or the previous slide of 50.69, there was a question came up earlier today about 50.69. Talk about that a little bit. But just kind of go through some of those points, because it's so high level.

50.2 has the definition of safety-related, and the first definition is integrity of reactor coolant pressure valve. So because of the way the valves are set up now, and relying on EDAS to keep them closed, it's a critical part of the reactor coolant pressure boundary. So that's part of the reason.

And then, with regard to the bullet on 50.36 is, you know, EDAS is an active system that holds those valves closed. So it's an active system. So it fits into that 50.36 definition of tech specs. And the management decision area of that is demonstrated by that.

But the -- jumping down to the various approaches, one would be regulatory treatment of non safety systems, which is we use that a lot with AP-1000. And the reason why it was developed for AP 1000, because it has a gravity-driven cooling system that has never been tried before. And back when Dr. Murley was here, that was something that he emphasized was because there is no large-scale test of that. Right?

So, to me, this is a very similar situation. You have a -- for the blowdown system that's supposed to be the cooling system for it, and it has really never been tried before, so the regulatory treatment aspect, which has a -- the RTNSS echo provision talks about, could it have an adverse effect on safety systems, right? So we think it could fit into the RTNSS category and have the -- stay non-safety, but could have some developed improvements.

The other part there about 50.69, interesting about 50.69, it does not include design certification applicants in the scope. The Commission specifically excluded that for a number of reasons, but one of the -- one of them had to do with finality, because the Commission was concerned that once you start categorizing the valves, is that changing the design certification?

So the Commission, when they wrote up the Federal Register Notice for 50.69, they said that, well, it could be addressed, even on a case-by-case basis, for a design certification applicant or a COL applicant could come in and they -- a COL applicant is allowed to use 50.69.

So they could come in and use the certified design and make -- you know, pull in 50.69 into that certified design. So it allows that capability there. But the Commission had specific concerns about including it generically, design certification applicants, in 50.69. So they have a separate, you know, process for that.

Anyway, so it -- you can do it with a design certification application, but it would -- it would be a challenge. You'd have to go back and look at the rules and maybe go through a 50.12 exemption from the specific words of 50.69, but you could do it. But it would not be very -- would not be a simple process of just picking it up.

But those are some of the aspects of -- that we had that we raised and some of the concerns we had with EDAS.

MEMBER HALNON: With the application of a quality assurance program over these, so there will be design controls. There'll be requirements since it's in the FSAR, or on a current licensing basis review, just to get to know, if you would.

And all the options that we have here are basically paper issues. And we're not talking hardware at this point, we're just talking the classification issue, is that fairly characterized? I mean, from a paper perspective. And so, the only options on this are paper issues at this point.

MR. SCARBROUGH: Well, part of it has to

do with the regulations. The regulations were written for large reactors, right? And they don't fit very well for the smaller reactors.

The way the regulations were written, if a component is part of the reactor coolant pressure boundary in the safety-related. And so, you can't fully get around that, right? It's sort of like, okay, this is what it does and this is what the regulation says.

But there's a process that for exemptions, you don't meet the specific words of the regulation, but you can justify an alternative that provides a reasonable level of safety, in terms of that. And I think to me that's what this falls into. Because to me it couldn't be more clear that it meets the 50.2 definition for safety-related, because it is part of the reactor pressure boundary. So, if the EDAS doesn't operate properly, slow down the system.

So, that's what the staff is proposing here, is that we deal with it through the regulatory process, where it does meet the definition in 50.2; however, there's not a need to do it that way -- right? -- if there's an acceptable safety process to be able to grant an exemption where they justify that they have augmented capability.

And from what I've seen from the discussions, it sounds like they have pretty strong augmented capability, even though it's not safety-related. Because they want to make sure the plant stays up and operating. So, they have an incentive to keep it operating.

MEMBER HALNON: Because they said it was hinting on the words a little bit that the paper's going to assure safety. Paper doesn't do anything for safety. Paper just describes what's there in the hardware and it's assuring its safety.

So, I get what you're saying. These large light water reactors request margin, and this has shown that there's very low consequence to the issue.

It's not unlike other things we do from a risk-informed perspective, is that some things have to fall below the line.

MR. SCARBROUGH: Exactly. And this is a case where it doesn't fit the small reactor. It doesn't fit the overall regulations very well.

And so, we're trying to develop a process where they can show they have adequate safety, and since the regulation doesn't fit them very well, go through the process of exemption. And I think from a safety perspective, we could do that.

But it's important to follow the regulations, or do a rulemaking to change it. But that's sort of where we are right now. We're trying to come up with some way to be able to say, yes, this provides adequate safety, doesn't precisely meet the regulation, so we think an exemption process -- like the other ones, there's several exemptions in this process.

MEMBER HALNON: Yeah, we're also still concerned it's opening the door for thousands of exemptions, because it's not going to always fit. You've got to make some judgments.

MR. SCARBROUGH: Yeah, and that's true. And that's one of the things I've talked about, is that it would be helpful if the NRC staff developed a template -- right? -- for the small reactors, where you have a template for exemptions, where it doesn't fit very well.

And that way, everything would be streamlined in terms of reviewing those.

MEMBER HALNON: Template or not, it's still fighting a thousand exemptions. But I get where you're at. Thank you for --

MR. SCARBROUGH: Okay, thank you.

(Simultaneous speaking.)

MEMBER DIMITRIJEVIC: Hello? Can you hear me?

MEMBER ROBERTS: Yeah, Vesna, go ahead. I'll hold my question.

MEMBER DIMITRIJEVIC: All right, sorry. I just have a one single question. How would the 10 CFR 50.69 help you in this case? Because EDAS is not showing as very significant in this Chapter 19. It's not really -- so it will be in category of non-safety, non-risk-significant.

So, I mean, we mentioned this in the previews, and I was just wondering about -- I think that Bob asked question about that. But using the 50.69 would not help you in this case.

This system is not identify as very significant in the Chapter 19. That's just the comment I want to put out.

MR. SCARBROUGH: Okay, thank you. This is Tom Scarbrough. Yeah, there's a couple of different ways. The RTNSS process, it would be under the ECHO category of RTNSS, which isn't directly related to the PRA number, right? It's more of, could it have an adverse impact on a safety system?

And then for 50.69, if you went down this approach, you might say it's safety-related, but since it's safety-related low-risk, it would fall into Risk 3, and then you would follow the process of 50.69 for that.

But we'd have to get through the process where it's not applicable under 50.69, because of the condition-excluded design certification applications.

But I understand what you're saying. It's overall low-risk just because the entire plant is low-risk, right? So, it would be difficult to put it under that.

But it would be a deterministic reason to include it. Because it directly provides the reactor coolant pressure boundary, as opposed to a PRA number.

MEMBER ROBERTS: Yeah, on the second-to-last option, the RTNSS process, what the applicant described was essentially almost safety-related, except for the question of the batteries.

How's that different for RTNSS? It sounds like the same thing.

MR. SCARBROUGH: Yes. It's Tom Scarbrough again. To me, this would fit into that process. I know there's some discussions about, in terms of the guidance, does it really fit into RTNSS, because the entire plant is low-risk.

But the RTNSS under all that was written for a large reactor, once again. But it does fit under the ECHO provision, which is, could the system have an adverse impact on safety systems? And this would, because it would drive the entire ECCS system to go into operation.

So, you can say it does have adverse impact. So, you could put it under the RTNSS ECHO provision, and then follow through on the discussion the applicant had regarding the reliability of the system.

The only thing that's sort of missing is, in terms of overall reliability, is the batteries, right? And they would have to work on that. But to me, it would fit into RTNSS, because it sort of follows that ECHO provision, which is, could it have an adverse impact on a safety system?

MEMBER ROBERTS: Okay, so there's something more that we need to do, in addition to what they're already doing? And what they describe, sounds like they're already doing that.

Again, this sounds more like what you call it, as opposed to what it actually is.

MR. SCARBROUGH: Yeah, and part of it is what they're doing. But it has also to do with down-the-road. As you know, twenty or thirty years from now you go back and you look at, okay, what's the regulatory basis where an incision's made?

And it's always important to have it very well understood which category of the regulation this falls under. Right?

If it falls under RTNSS, there's a process to follow that, right? If it falls under 50.69, there's a process to do that. But right now, it's a little uncertain as to what process it's in regulatory-wise, for down the road.

So, that's why we want to make sure that whatever decision is made, it's very well within the regulatory basis, so that if there is an issue way down the road, it's clear what the regulatory process would be to address it. So, that's where we are.

MEMBER ROBERTS: Okay, thanks.

MR. SCARBROUGH: And some of that dialogue, suggesting there's a regulatory gap currently that is creating the situation, NuScale's in with this?

Personally, I think the regulations, when you look at them, they were written for large light water reactors. And all of these plants that I've seen as the small ones start coming in, they have different issues and different questions.

So, to me, to be prepared for these, to me, I would recommend developing a template. Say, okay, these are the areas where a small reactor coming in, like in NuScale, has challenges meeting the specific words, and the regulations are written for large water reactors.

And just have a template. And say, this is the process, this is what you do to obtain an exemption, and have these all lined up so that we don't have to have these types of discussions all the time for every small reactor, once again.

MEMBER MARTIN: You know, I've asked the questions about DSRS a few times. And the intent really was, like, how did this slip by? Because that's exactly what at least I've always thought that DSRS was about, was to capture these design differences, these novelties, and kind of get the agreement between the applicant and staff as to, what is the appropriate interpretation of design criteria or what have you, and as I had observed, it hadn't been touched since 2016.

But I think this was one of those things where other priorities, and we've kind of let this one kind of just sit,, and then here we are the last minute, and what are you going to do?

I mean, is it exemption? Is it approval with basically open item, you go to COLA and let them handle it? Or you just come down to an interpretation that is acceptable and we move on. It seems like those are --

MS. PATTON: This is Becky Patton again. So, I just wanted to point out that we do engage with applicants very early, especially new reactor applicants, on areas where they may need things, such as exemptions.

This was discussed and talked about very, very early in the review, I believe pre-application portion.

So, you're seeing it at the end, right? But we've lived this issue throughout the entire review, even before it began.

So, we don't have a lot of specific guidance specific to new reactor applicants, in terms of specific exemptions. But that's why we do have a lot of engagement with them.

And we do have exemption criteria. We've done this multiple times for many other new reactor applicants where somebody has needed an exemption in one area. We have discussions with them, and the staff does look at those in a holistic type of risk-informed approach. The exemption process does allow that type of look at it.

So, some of that I'm mentioning is of upper management, others, I am one of the non-concurring staff as well. And I believe the exemption process is the appropriate place for this to be in, and that the regulations are clear in this case.

And so, we do look at things under exemptions in a risk-informed manner. But that is appropriate when the regulations are clear.

But they've obviously done a lot of things. We've noted in the non-concurrence some other things that would need to be done, and we also believe, like was stated, that the regulatory footprint on this does need to be set, because there are going to be design changes for these facilities in the future, right?

And so, how EDAS is treated in regulatory space, whether or not it's relied on, and what the basis was for the staff's approval, needs to be clear, because there could be significant modifications to that going forward under the change processes. And that needs to be appropriately --

(Simultaneous speaking.)

MEMBER HALNON: Becky, what you just stated sounds like the exact purpose of an FSAR and an SER, which is document the basis of this FSAR, and the staff review it, and provide your basis for approval in the SER.

Especially for a standard design, it seems like that would just be all you have to do. And if someone's changing it down the road, it's going to be under a design review program that's going to be the appropriate analysis for that change.

But seem to me that everything you're talking about is exactly the purpose of why you take an FSAR and you document your basis behind whatever decision -- I guess you use this one specifically -- document why EDAS is the way it is. The SER says, okay, we understand that. It's documented for standard going forward.

I don't see the difference between that and filling in a separate document like an exemption. It's that purpose of the FSAR.

MR. SCARBROUGH: This is Thomas Scarbrough again. In the SER, we have to say specifically what regulation we're making our decision under.

And that's sort of where we are right now is, we think this comes under an exemption process, because it doesn't meet the 50.2.

(Simultaneous speaking.)

MEMBER HALNON: -- SER is just a matter of meeting the regulation. You just check, check, check, check, and you don't make any decisions.

MR. SCARBROUGH: No, in the SER you have to say, this meets the regulation. So, we have to make sure we understand which regulation we're under.

MEMBER HALNON: You can't in a SER say this is proposed to us? The FSAR, you never say that. Here's what it is in the FSAR, therefore, it's okay. You never say that? You have to connect it exactly to a regulation every time?

MR. SCARBROUGH: Yes. Yeah, we have to say which regulations we're meeting.

MEMBER HALNON: And you know the consequences are way below the line.

MR. SCARBROUGH: But that's why you use the exemption process. You could decide it does not meet the regulation. Then you say, okay, we're processing the exemption because it's low-safety significance.

MEMBER HALNON: Just seems to me like this is a merry-go-round that's going to not meet any of the advance reactor good regulation. It just seems like we're doing an exercise that is costing a lot of money, a lot of staff time, our time, and it's --

MR. SCARBROUGH: Yes, I appreciate that. That's why I think we need the -- and make a DSRS --

MEMBER HALNON: Can you do this separately then?

MR. SCARBROUGH: I'm sorry?

MEMBER HALNON: Can you do that on a separate path?

MR. SCARBROUGH: Oh yeah, on a separate path. Yeah, that's what I would hope, that they'd go back and look at DSRS and see if there's a way to improve it to make it more streamlined and be able to say, okay, this is what the regulation says, it doesn't meet the regulation, so we're going to process it through this exemption, and streamline that process.

So, we have these new reactors coming in with new applicants, and it's all very streamlined as to what they need to do.

MEMBER HALNON: We're not going to be able to redefine the entire regulatory process here. Again, it's --

MR. SCARBROUGH: Well, I think under Part

53, this would be non-safety with special treatment. But that's insight that benefits from the last sixty years and doesn't have to happen to be in Part 52, right?

(Simultaneous speaking.)

MEMBER PETTI: Members, I really think we ought to move on here. We've got a lot to go, we don't want to be here at seven o'clock. Or the staff will have to be here until seven o'clock.

MEMBER HALNON: Right. Let's go ahead and move on. Like I said, we're not going to resolve the entire regulatory process. Yeah, go ahead, Gary.

MR. BECKER: This is Gary Becker, senior counsel for NuScale again. Before we moved on, I wanted to clarify one aspect for the record.

The staff has asserted definitively that the solenoid valves for the RVV trip valves, definitively constitute part of the reactor pressure boundary.

I just wanted to reiterate our position that we documented in our presentation that the Commission decided during the DCA review, that maintaining the ECCS valves closed during the transients on a loss of power, was not a safety-related function, maintaining RVV integrity.

So, I am urging staff to address today how this is different in that respect, other than removal of the IVs. As we noted, that changed the timing of opening the ECCS valves.

But in the case of the DCA, if opening ECCS on a loss of power was a loss of RVV integrity, there would not be any exemption then, and we did not. And that was resolved by rulemaking, and we think the same logic applies here.

MEMBER HALNON: Thank you, Gary. Rather than circle back and do some more circular discussions, let's go ahead and move on to your next slide.

MS. PATTON: This is Becky Patton. Can I address the NuScale comment, since they said we didn't address it specifically in our presentation?

During the DCA review, where I think they'd have IABs, part of the consideration for the staff in the limitation and condition that was placed on the topical report related to that, was the fact that you would be at a significantly reduced pressure, like a thousand pounds or less, when those would reposition. When the IABs would reposition.

And we considered that all the parameters where that trending and positive direction, you can already shut down.

And we considered that to be basically, you were in a safe, stable condition at that point. And that was why also that limitation and condition that was placed in the topical report, both of those in conjunction, were why you didn't need an exemption in that case.

So, just to be clear, the removal of the IABs was directly related to why it's an issue now.

MEMBER HALNON: Okay. Thank you again for that. Thank you. Let's move on.

MR. MILLER: This is Josh Miller again. As an outcome of the differing views process, NRR management is evaluating whether a function is needed to treat EDAS as a non-safety-related. Information pertaining to EDAS design and its reliability and availability controls, would be sufficient, or the exemptions classifying EDAS as safety-related, is not necessary for adequate protection.

As staff initiated exemptions, we documented in SER Chapter 8 exemption from safety-related requirements in 10 CFR 55(a)H, exemption from safety-related requirements of 10 CFR 50, Appendix B, Criterion 3 through Criterion.

This approach would clarify EDAS as

exempted from safety-related classification, and therefore non-safety-related.

MEMBER MARTIN: Just to clarify, one of the concerns that I've heard the last fifteen, twenty minutes, is documenting for the record that the EDAS, or the whole circuit -- the whole system, rather -- keeping those valves closed is important. And I was wondering if the exemption would document that.

Because right now, one of the problems is Chapter 15 just says, this is not a single barrier that needs to be assumed. That's the end of discussion.

There's nothing that has it as regulatory in some of the slides this morning. We have all told a pretty good story, but they're not in the FSAR. So, would the exemption document that whole story?

MS. PATTON: This is Becky Patton again. So, there are a variety of things. One, there are some things documented in Chapter 15 right now if you look at some of the sections of 15-0.

However, an exemption does make it clear, or would make it clear, and have a more thorough documentation of all of the aspects that went into staff's consideration, also making it clear that it would've been required to be safety-related under that regulation. So, we believe that that, in terms of the non-concurrence, that was a consideration for that.

Now, in terms of the current management decision of what's documented in Chapter 15, it does mention the fact in 15-0 that it's relied on, and then points over to Chapter 8 for the determination that it has all of these augmented aspects to it. Right?

So, there is something in there currently that notes that it is being relied on for the Chapter 15 events that's factual, I think if you've seen some of the presentation.

But instead of evaluating it as an exemption under the exemption criteria, it instead points to eight to talk about the augmented following aspects -- and other aspects.

MEMBER ROBERTS: I think Greg's point is very well taken, that the real concern, at least what I'm seeing, is Life Cycle's part of this plan tonight.

And if some design change were made and made loss of the holding function more likely, or the design change were made to make the consequence worse, and some of you were to know about it in the evaluation, and that's not documented in the FSAR clearly, then you may not in fifty years from now see that as something that needs to be evaluated.

So, I just want to make sure that if you were to solve this issue with an exemption, that the exemption is clear as you think needs to be. Those are considerations that would need to be looked into for future design changes.

So, I used to work through the process. I want to make sure that's what you're thinking.

MS. PATTON: Yeah, that's correct. Each of the exemptions, and you can find other ones in the different chapters of the SE right now. They do go through a full documentation of the staff's rationale and the different considerations that go into granting each of those exemptions.

MEMBER MARTIN: Go while you can.

MS. JOSEPH: Okay, at this point we're going to do a switch of reviewer.

MEMBER PETTI: Should we take a break?

MEMBER HALNON: Yeah, let's go ahead and take a ten-minute break and come back about 2:35.

(Whereupon, the above-entitled matter went off the record at 2:23 p.m. and resumed at 2:35 p.m.)

MEMBER HALNON: Okay, let's bring this meeting back to order. Stacy, you're up.

MS. JOSEPH: Rosie, if you want to go ahead and get started if you would.

MEMBER HALNON: Okay, Rosie.

MR. SUGRUE: Hi, This is Rosie Sugrue again. I'm going to be talking about 15.2, which is the decrease in renewable by the secondary system.

We found that the most limiting case in this group is the feedwater system pipe breaks inside and outside containment.

The results within the SER was 2.4, which is above the 95-95 limit of 1.43. Maximum RCS pressure was below 110 percent of RCS design pressure. Max peak secondary pressure was also below 110 percent of secondary system design.

I'll briefly run through the key assumptions in this case 2 over here.

The initial power level is going to be 102 percent of nominal, to account for measurement uncertainty.

Conservative reactor characteristics, like maximum time delay, holding the most reactor brought out of the core, inbounding control rad drop rate, limiting, beginning of cycle reactivity feedback, the limiting power response analyses, AC power's last at the time of the break, immediate turbine and feedwater pump drip.

FWIV is assumed to fail close on the

faulty feedwater line, SSI valves are assumed to pause, and DHRS valves are assumed to open at their maximum times.

System biases include high RCS temperature, high fuel temperature, low-pressurizer pressure, low-pressurizer level, and minimum RCS flow.

This next part I'm going to skip over, because this actually refers to other cases within 15.2, so it’s a little confusing.

I'll move on to the steam generator tube failure, which is 15-6-3. MCHFR is not limiting for this case -- it's been screened out.

The limiting RPV pressure scenario is a 20 percent partial tube failure at the top of the steam generator, with a coincident loss of nocturnal AC power.

The limiting steam generator pressure scenario is 100 percent split break to failure at the top of the steam generator, with loss of normal AC power.

Last one, radiological consequences, are confirmed to be bounded by the SR-1503 assumption that we talked about earlier.

And the key assumptions here, right at the core power, is at 102 percent. The highest worth rod was stuck out, and assuming no single failure is conservative. Two failure at the top of the steam generator results in higher RCS and steam generator pressure.

MR. PIELA: I'm Sean Piela. I'm going to cover the inadvertent operation of reactor valves, Section 15.616. There are a few valves in the MPM design for this event category. The ECCS valves are the ones with inadvertent operation laws, and the biggest challenge to figures of merit.

If you look at all the valves available, it means that the limiting IORV cases is inadvertent operation of the ECCS system, and taken with the loss of DC power, time is equal zero, causing both RRVs to begin to open immediately, because they no longer have IAVs. That would be a limiting scenario.

This means -- probably makes sense that the limiting case would be insensitive to ECCS actuation time, because CCS is on. ECCS valves now have well-restricting internal to the valve body. We mention that just for continuity with previous slides. Next slide.

The IORV events are MCHFR challenge events. The LOCA methodology is what dictates how you analyze these transients, which has a special set of methodologies for phase 0 of LOCA, MCHFR analysis.

Some of the conservatisms and initial condition biases are a flow blockage, in light of the hottest sun way additional primary set thermal power, 102 percent.

The hydraulic losses of the primary loop are redistributed, and there's a specific CHF correlation used. And we'll have time for this. Next slide, please.

So, we have the results of the applicant and performances to the confirmatory analysis. The limiting case of all the IORV possibilities is the inadvertent opening of one IORV, both loss of power, and times equal to zero.

Due to the IAB still being present on the RRV valves, it is improbable that more than one RRV can open randomly. It's not incredible to shooting an event.

Using the LOCA topical in the future, LOCA topical report methodology for phase zero, we confirmed the applicant's results that there were many MCHFRs 1.41.

This is not the limiting Chapter 15 MCHFR, unlike the case for US-600, and we accept these criteria as 1.2 or greater, this CHF correlation.

Just for clarity and to put this into perspective with the rest of Chapter 15, the IORV events are not for designs of limiting transients for continuing response.

RCS pressure, steam generator pressure, COL, is rather high, and for the limiting IORV event, CHR is not a relevant factor. Next slide, please.

MR. ZHENG: Good afternoon. I'm Dong Zheng and I been with the Agency for two years. Before that, I work for a private company back home for more than ten years doing thermal hydraulic analyzing. And I'm covering the section 15.6.5, the loss of coolant accident LOCA.

LOCA is postulated accident, reactor coolant is lost through the break on the reactor pressure boundary.

Potentially, that will lead to the reactor core overheating. The LOCA event for the NPM20 is unique compared to the typical light water reactor.

The RCS pipelines through the reactor vessel, and should have the CVCS injection line on the discharging lines.

And these lines are also sized two inches or less with venturi flow nozzles integrated. This design helps to limit the chance and the consequence of a LOCA break.

Accumulation of default. The ECCS actuation logic change, in that the ECCS injection is triggered by lower riser levels instead of containment level signal. Our work uses ECCS logic there in that line.

In this SDA, each HRS system is also credited for operating during a LOCA. It's trying the passing of cooling in the RCS, especially for the small break LOCA accidents.

We have covered the staff, the effort on reviewing these LOCA-related design changes in the previous ACRS LOCA topical review.

Because of the ECCS operations, most of these LOCA scenarios are similar, there are three distinct phases.

The LOCA phase zero is a sharp flow-down phase initiated by a break. The break initiation result in reactor that is pressurization inside the reactor, and quick pressure surge in the containment side.

Most cases, the MCHFR occurred in this space, which the reactor core may express the shock period of voiding, even though the overall class liquid levels still remains high in this case.

And then LOCA Phase 1 is the continuation of the blow-down of the core inventory into the containment. And here, the riser level jumps below the levels at that point, and the triggers the ECCS.

The LOCA Phase 2 begins with ECCS actuation. On top of the RVV open, the reactor core quickly drops to equalize with containment. Once the pressure equalized, the RRV, the IAV releases and enable the coolant fall back from the containment, back to the reactor vessel, and keep core power. But most cases the minimum class liquid level and the peak containment pressure and temperature occurred after the ECCS actuation.

Following the method discussed in the LOCA topical report and using the NRELAP5 version 1.7, NuScale has analyzed the thermal hydraulic response of various LOCA events in the NPM-20. Our staff has reviewed this analysis and verified the included parameter and the initial conditions conservatively assumed in these calculations. The limiting cases for MCHFR is determined to be a hundred percent CVCS discharge line break at the one hundred percent of reactor power concurrent with the loss of AC and the EDAS. With the loss of EDAS power, both RPVs will open immediately after break.

The worst MCHFR value is determined to be 1.35 and the minimum collapsed liquid level is 1.7 inches above the top of active fuel. Additionally our staff in the last group also performed a TRACE confirmatory analysis for a few of the LOCA break cases. The TRACE simulated the LOCA trend well with NRELAP5. The staff found that overall NRELAP5 result is more conservative than the TRACE in predicting the LOCA figure of merits. The confirmatory study that NuScale’s LOCA method produced a reasonably conservative result.

CHAIR KIRCHNER: So, what was the TRACE result?

MR. ZHENG: We can show that the TRACE result --

CHAIR KIRCHNER: Just a number. I'm interested in your number level of both the active fuel.

MR. ZHENG: I see.

MR. LIEN: This is Peter Lien from Office of Research. The rebutting TRACE result for the same transient, TRACE predicts 680 PSI maximum pressure. Versus NRELAP5, 780. So, we are 90 PSI lower.

For the minimum level, TRACE predicts about 12.3 above top of active fuel. This is NRELAP

9.7. So, we are more conservative. We are less conservative from here to NRELAP5, but in terms of minimum CHFR, TRACE is not applicable for CHF calculation, because of different CHF correlations.

But we did verify the trend for different break sizes and different locations. So, we conclude that NRELAP is conservative.

MEMBER MARTIN: On that question's Member Martin. So, I believe we visited last month the non-LOCA, maybe, before that, PO LOCA.

There was a limitation and condition associated with a topical on NRELAP 1.7. And I think we were told at the time that you all were going to work through that and get that closed. Is that water under the bridge at this point? Has that 1.7, in your eyes, now considered acceptable and all that?

(Simultaneous speaking.)

MR. SUGRUE: I believe that's the end result, yes.

MEMBER MARTIN: Okay, the end result? Okay. I guess the SC would be revised. I mean, it was draft.

MR. SUGRUE: Oh, that topical report SC?

MEMBER MARTIN: Yeah, yeah, yeah, yeah.

MR. SUGRUE: Yeah, that will conclude at

that meeting. That'll be revised even further.

MEMBER MARTIN: All right, thanks.

MR. SUGRUE: And headed our way.

MEMBER MARTIN: Thank you.

MR. ZHENG: Next, please. Staff have mentioned the ECCS actuation level logic is a major SCA design changes. NuScale proposes a new method using thermal dispersion switch. These have the mixture level changes by the heat transfer rate difference between the liquid and the vapor state. A LOCA topical review. Staff adding a limiting condition to ensure the proper ECCA actuation. It states that the approach should follow the LOCA topical report determined levels setpoint for units. In the current LOCA model this ECCS actuation approach is implemented by the NRELAP code using the trip component. The ECCS actuation on low RPV riser levels are assumed to occur when the 90% of void near the void outlet. This is corresponding to our riser level of approximately 550 inches from the bottom of the RPV. If for some reason this low riser signal is not triggered you can bypass, the ECCS can also be actuated by the low-low riser signal level. Which is assumed to occur when the void fraction at upper riser node reaching 95%. This is corresponding to our riser level of approximately 473 inches. We know it's hard to predict the void changes during the LOCA transient due to the flashing and rapid depressurization inside the reactor core and the riser. Staff expects this could lead to certain ECCS actuation delay and may impact the reactor safety. Staff at NuScale have performed the timing evaluation. In this violation the lower riser level is either triggered late by using the lower end of the analytic limit all completely bypassed. In that case the ECCS actuation signal was triggered by the low-low riser level setpoint. All schedules resolved a certain ECCS actuation delays. The evaluation shows that all LOCA figures of merit are adversely impacted by the simulated ECCS actuation delay. It is not a factor the MCHFR, since MCHR usually occurs well before the ECCS actuation. The result also shows the collapsed liquid level and the containment pressure and peak clad pressure and temperature also not adversely affected by the ECCS actuation delay. As long as the ECCS actuated the timing delay resulted in additional cooldown which led to early RRV release in the last limiting containment response. After reviewing the NuScale’s ECCS actuation approach and the associated LOCA calculation result, staff has made our findings the level sensor responses corresponding to the specific setpoints and analytical limits results in acceptable collapsed liquid level above the core. That concludes my part.

MR. PIELA: Back to me, Sean from SNRB. This was mentioned at the beginning, one of our colleagues is unable to speak today, so I'm going to cover this part for them.

Yeah, so LOCA break exemption. The application staff thought that there were two locations that should be counted as far as a break spectrum. These are the ECCS file flanges and the CVCS piping between the containment vessel and the containment isolation valves. These became HITI number 2 and number 10, IM VAC tech coefficients.

The staff was open to support or consider, or encourage, a risk-informed alternative approach for these analyses. And Ms. Hill wanted us supporting an exemption request with supporting analysis to treat these locations as beyond-sign basis, initiating locations. Next slide, please.

So, for this slide, there's one that says -- there, my colleague says here there's something you'd like me to read it.

So, due to the first-of-its-kind and precedent-setting nature of the exemption of the site from the requirements of 10 CFR 50.46, staff used the following framework and philosophical approach to balance prevention and mitigation of LOCA's risk-informed map.

First, the staff recognizes that the NuScale SMR design contains multiple holes to design features that reduced the overall risk of LOCAs.

Second, the US460 has enhanced design and operational programs that provide assurance that failures at the location of interest are highly unlikely.

Staff presented its review of the enhanced stress limits and application of the concepts of the branch technical position 3.4, during the Chapter 3 ECRS meeting.

Third, realistic best-estimate analyses of LOCAs at the locations of interest is beyond design basis accidents, must demonstrate that the consequences are acceptable.

The analysis must demonstrate the core remains cool, and accounts for uncertainties to avoid cliff edge effects.

All three of these criteria are met. The staff believes that reasonable assurance of adequate production can be provided, and it is appropriate to pursue an exemption to the LOCA requirements of 10 CFR 50.46, and GEC 35. Okay, next slide.

Now, this skill develops analysis of this criteria, related to LOCA methodology, to look at these breaks. They demonstrated that the results following the methodologies with any changes, meet the acceptance criteria of the LOCI, and more of the modified acceptance criteria.

The acceptance criteria couldn't cover the LOCA line containments and radiological figures of merit.

The staff did a review and did sensitivity analyses around confirmatory analyses that follow more closely with the LOCA methodology.

We cleared the analysis acceptable for the on-design basis event and supports to the exemption. So, notes that my colleague wanted me to read here, "this skill will present the acceptance criteria and the results in the closed portion of the meeting. The staff's analysis was focused on parameters the staff believed were either highly sensitive, or when these scales' assumption may be non-conservative."

The staff observed they're using more realistic parameters, or when accounting for uncertainty, the timing of the events and exchanges of the overall trends and physics remain the same.

In addition, the design contains multiple inventory-addition systems, which were not modeled in the underlying use calculations, but in reality will be used or utilized by the operators to add inventory to the guard.

CHAIR KIRCHNER: That was good.

MS. JOSEPH: All right, thanks, Sean. This is Stacy Joseph. In conclusion, while there are some differences between the DCA and the SDAA, overall the staff found that the applicant provided sufficient information to support staff's safety findings.

The staff found that all applicable regulatory requirements in Chapter 15 were adequately addressed, and finally, on the staff non-concurrence, staff does not expect a decision on the EDAS exemption to change the analysis or design.

As an outcome of a non-concurrence process reviews, the staff will modify the relevant SERs to clarify the regulatory basis and document the justification that EDAS is not safety-related.

As mentioned, we're still waiting on management conclusion to the abnormal occurrence.

And that concludes the staff's presentation. Are there any additional questions at this time?

CHAIR KIRCHNER: Bob, as lead for fifteen?

MEMBER MARTIN: I think we've covered a lot of territory in open session, more than I thought we'd be getting into. So, I would say you've done a good job here this morning and this afternoon. I have no other questions appropriate here, we'll close the session.

CHAIR KIRCHNER: Scott?

MEMBER PALMTAG: This is Scott. Yeah, I agree, I thought we covered a lot.

CHAIR KIRCHNER: Any other members? Online, any questions from our members online, or consultants?

Okay. With that then, thank you very much. We will proceed to take public comments. If there is anyone out there or in the room who wishes to make a comment, please state your name, affiliation, as appropriate, and provide your comment. I see Ed Lyman. Go ahead, Ed.

DR. LYMAN: Thank you. It's Edwin Lyman from the Union of Concerned Scientists. Can you hear me?

CHAIR KIRCHNER: Yes, we can.

DR. LYMAN: Okay, thanks. Yeah, I just

wanted to point out that we appreciate the Committee's raising the issue of uncertainty. And when, again, the shutdown margin seemed to be very small compared to the uncertainties that were quoted. It would be very helpful if in the open sessions that uncertainty, that error bars were provided, because that provides important context and that shouldn't be simply put off until the closed discussion.

So, I think in the future, it would be very much appreciate if results like that were provided with the uncertainty bands, because a twenty-eight PCM margin, or whatever it was, with 28 PPM margin, boron concentration of 100 PPM uncertainty, is essentially the same as zero margin, as far as I can tell.

So, I appreciate that was highlighted and I hope the presentations will reflect that in the future. Thank you.

CHAIR KIRCHNER: Okay, further comments from the public? Yeah, you are clarifying your presentation? Or making us --

MR. GRIFFITH: Yeah, just a closing comment and identify.

CHAIR KIRCHNER: Yeah.

MR. GRIFFITH: This is Thomas Griffith,

NuScale power licensing manager. I appreciate the discussion that we had with EDAS. And I just want to remake the point that the US 460 design exceeds the Commission's safety goals by orders of magnitude. The Commission safety goals are for CDF, and large release frequency are on the order of E to the minus 4, E to the minus 6. Before the US 460 design, it's on the order of E to the minus 9 and E to the minus 13.

And with respect to the function of the ECCS valves, NuScale maintains the position that the safety function of the valves is to open. And if we look at insights from Chapter 19, you'd see that the majority of cutsets that result in core damage are due to primarily a failure of the ECCS valves to open, which substantiates NuScale's claim.

And so, overall, what I would offer is that irrespective of what occurs to result in the ECCS valves opening, the NuScale design places itself into a configuration that is safe and precludes core damage, and is safe for long-term. That's it.

CHAIR KIRCHNER: Any further comments from the public? With that, okay, then we are finished with the open session for today. We will take a break and then reconvene in a closed session. How much time do we need? You checked everyone? Probably twenty minutes?

CHAIR KIRCHNER: Ten minutes? Okay. Well, let's give ourselves a fifteen-minute break here, and reconvene at 3:20, Eastern Time. You're in recess.

(Whereupon, the above-entitled matter went off the record at 3:04 p.m.)

C E R T I F I C A T E

This is to certify that the foregoing transcript

In the matter of: ACRS Nuscale Subcommittee Open Session

Before: NRC

Date: 04-01-25

Place: teleconference

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

Court Reporter

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LO-180546

March 25, 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) Chapters 1, 4, and 15,” PM-180495, 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 April 1st, 2025. The materials support NuScale’s presentation of Chapters 1, 4 and 15 for the US460 Standard Design Approval Application.

The enclosure to this letter is the nonproprietary presentation entitled “ACRS Subcommittee Meeting (Open Session) Chapters 1, 4, and 15,” PM-180495, Revision 0.

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

If you have any questions, please contact Amanda Bode at 541-452-7971 or at abode@nuscalepower.com.

Sincerely,

Mark W. Shaver

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

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

Enclosure 1: “ACRS Subcommittee Meeting (Open Session) Chapters 1, 4, and 15,” PM-180495, Revision 0, Nonproprietary

Enclosure 1:

"ACRS Subcommittee Meeting (Open Session) Chapters 1, 4, and 15," PM-180495, Revision 0, Nonproprietary

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360.0500 Fax 541.207.3928
www.nuscalepower.com

ACRS Subcommittee Meeting (Open Session)

April 1, 2025

Chapters 1, 4, and 15

Acknowledgement and Disclaimer

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ACRS Subcommittee Meeting (Open Session)

April 1, 2025

Chapter 1 – Introduction and General Information

Presenter: Tyler Beck

NUSCALE™ Power for all humankind

Section 1.1 through Section 1.4

- Section 1.1, Introduction
  - Optimized to reduce redundant content from other sections
  - Multi-module considerations
    - Previously included in US600 Design Certification Application (DCA) Chapter 21
- Section 1.2, General Plant Description
  - Includes changes (e.g., figures of plant overview) reflecting the US460 standard design
- Section 1.3, Comparison with Other Facilities
  - Reflects US460 design features (e.g., thermal power output)
- Section 1.4, Identification of Agents and Contractors
  - Unchanged from US600 DCA

Section 1.5 through Section 1.8

- Section 1.5, Requirements for Additional Technical Information
  - Verification and confirmation tests of unique design features (e.g., emergency core cooling system (ECCS) supplemental boron)
  - Boron dissolution testing performed at NuScale Integral System Test (NIST) facility
  - Additional ECCS valve functional testing performed with fully prototypic valve assemblies
- Section 1.6, Material Referenced
  - “Incorporation by Reference” was an issue resolved during audit
  - NuScale incorporates by reference most technical and topical reports
- Section 1.7, Drawings and Other Detailed Information
  - No significant changes from the US600 DCA
- Section 1.8, Interfaces with Standard Design
  - Removal of “Conceptual Design Information” list from the US600 DCA (e.g., potable water system)

Section 1.9 and Section 1.10

- Section 1.9, Conformance with Regulatory Criteria
  - Includes comprehensive list of conformance with NUREG-0800 criteria, Design Specific Review Standard (DSRS) criteria, regulatory guides (RGs), generic communications, etc.
  - Changes in conformance reflect US460 standard design
  - Examples of changed conformance from US600 DCA to US460 SDAA:
    - RG 1.7, Control of Combustible Gas Concentrations in Containment: NuScale utilizes a passive autocatalytic recombiner in the SDAA, as opposed to no specific control system in the DCA
    - DSRS 5.3.1, Reactor Vessel Materials: Criteria pertaining material surveillance are no longer applicable because the design supports an exemption from 10 CFR 50.61 and 10 CFR 50.60 due to using austenitic stainless steel in the reactor pressure vessel (RPV) beltline
- Section 1.10, Sites with Multiple Nuclear Power Plants
  - No significant changes from US600 DCA

Acronyms

DCA	Design Certification Application
DSRS	Design Specific Review Standard
ECCS	Emergency Core Cooling System
NIST	NuScale Integral System Test
NPM	NuScale Power Module
NRC	Nuclear Regulatory Commission
RAI	Request for Additional Information
RG	Regulatory Guide
RPV	Reactor Pressure Vessel
SDAA	Standard Design Approval Application

ACRS Subcommittee Meeting (Open Session)

April 1, 2025

Chapter 4 – Reactor

Presenter: Sarah Turmero

Agenda for Chapter 4: Reactor

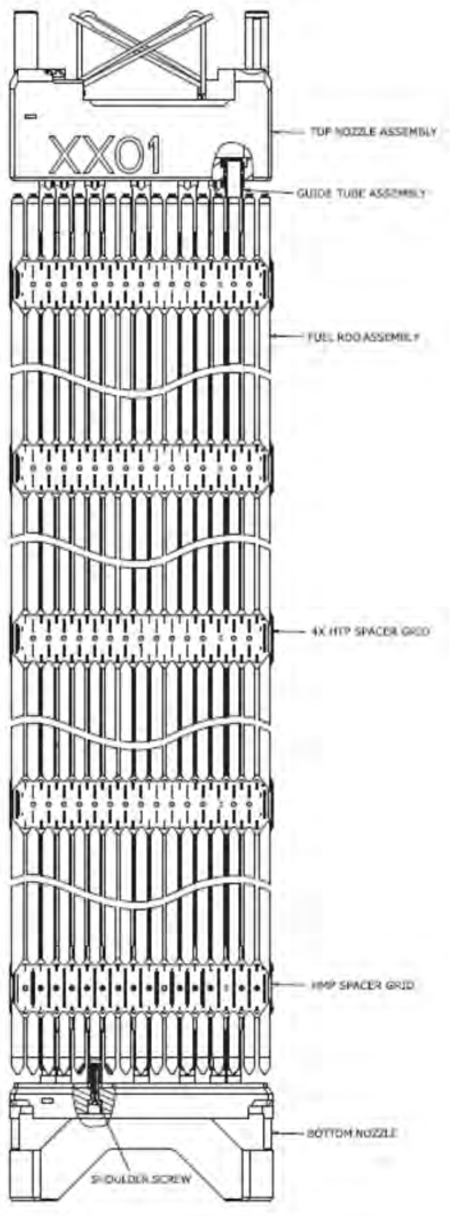
- 4.1 Summary Description
- 4.2 Fuel System Design
- 4.3 Nuclear Design
- 4.4 Thermal and Hydraulic Design
- 4.5 Reactor Materials
- 4.6 Functional Design of Control Rod Drive System

Section 4.1 Summary Description

- Information from DCA FSAR Section 4.1 was separated and incorporated into other sections of Chapter 4 – reduced redundancy in Chapter 4
- DCA FSAR Table 4.1-1 NuScale Reactor Design Parameters
  - SDAA FSAR Table 4.4-1 and Table 4.2-2
- DCA FSAR Table 4.1-2 NuScale Core Design Parameters
  - SDAA FSAR Table 4.3-1
- DCA FSAR Table 4.1-3 NuScale Reactor Control Rod Assembly Parameters
  - SDAA FSAR Table 4.2-3
- DCA FSAR Table 4.1-4 NuScale Core Design Analytical Tools
  - Provided in the text of SDAA FSAR Section 4.3.3 for Nuclear Analysis
- No audit questions or RAIs

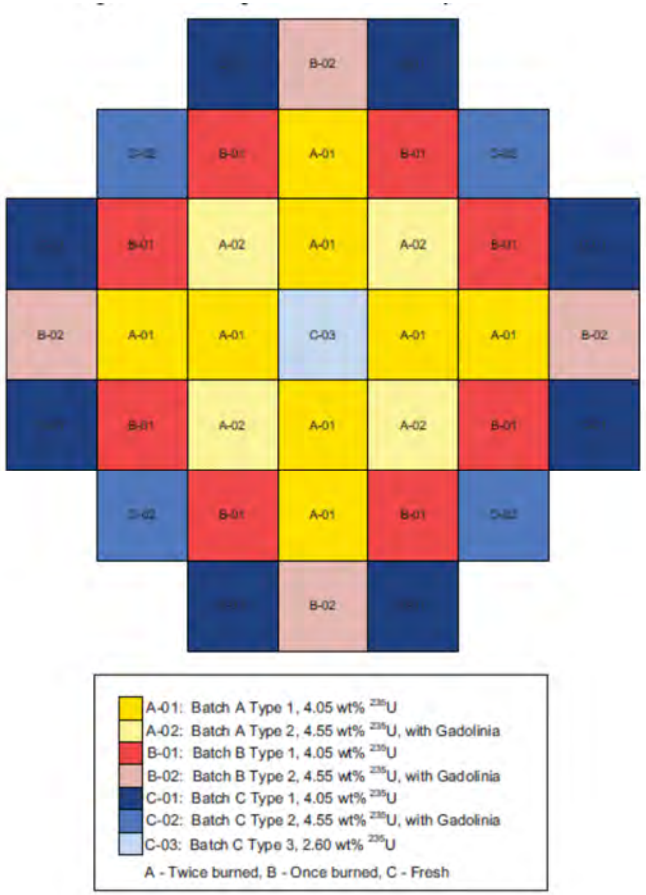
Section 4.2 Fuel System Design

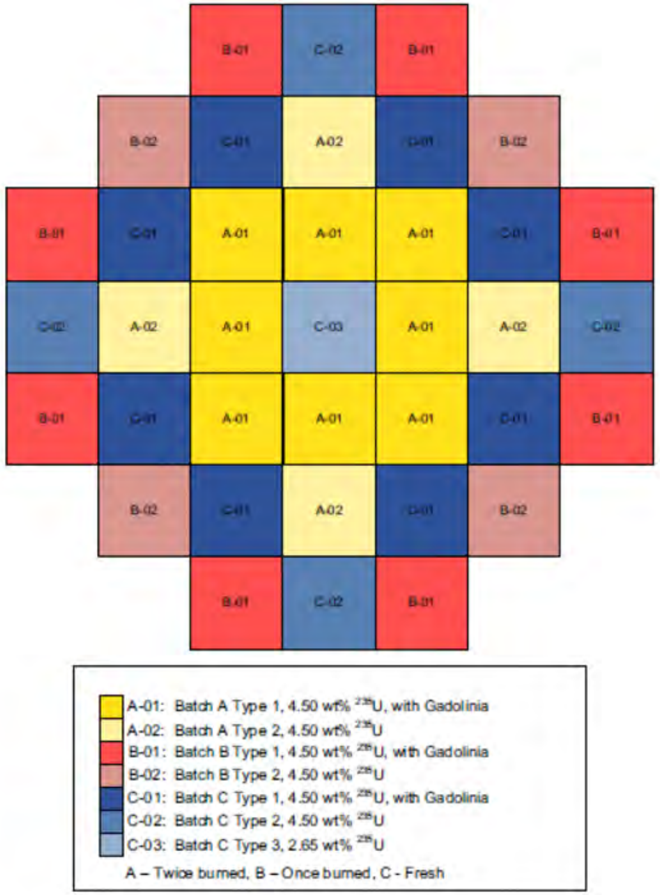
- Majority of the fuel design remains the same as the DCA design
  - Fuel rod array, rod per assembly, spacer grids, active fuel length, cladding material
- Control rod design remains the same as the DCA design
- Changes from DCA
  - Administrative – Incorporation of classification of SSC table, removal of redundant information
  - Fuel rod length increased by ~1 inch in the upper portion of the fuel pin where the plenum spring is
  - Core loading changed from 9,213 kgU to 9,269 kgU
  - TR-117605-P, “NuFuel-HTP2™ Fuel and Control Rod Assembly Designs,” Revision 1
    - Faulted limits applied to the fuel rod cladding are derived from ASME BPVC, Section III, Table XIII-3110-1 (2019)
  - TR-108553-P-A “Framatome Fuel and Structural Response Methodologies Applicability to NuScale,” Revision 0, for applicability of previously approved codes and methods to the SDAA design.
- 21 audit questions resolved and no RAIs
  - 11 questions on TR-117605-P



Section 4.3 Nuclear Design

- Changes from DCA
    - Fuel pellet density change from 96 to 96.5%
    - Changes related to power uprate
        - Linear heat rate
        - Peaking factors
        - Cycle length
- Addition of emergency core cooling system (ECCS) supplemental boron (ESB)
- 29 audit questions and 1 RAI resolved
    - RAI requested a limiting condition for operation (LCO) on the heat flux hot channel factor (F_Q) or justification for not having an LCO
    - F_Q does not require an LCO per 10 CFR 50.36(c)(2)(ii)(B), Criterion 2 because it is not a direct input or initial condition for safety analysis calculations





Nuclear Design Parameter Comparison

Parameter	NPM-160	NPM-20
Core Average Linear Power (kw/ft)	2.5	3.9
Heat Flux Hot Channel Factor	1.860	2.196
Maximum Enthalpy Rise Hot Channel Factor	1.386	1.400
Fuel pellet density (% theoretical density)	96	96.5
Doppler (least negative) ($/F)	-8.4E-03	-2.1E-03
Doppler (most negative) ($/F)	-1.4E-02	-4.7E-03
Shutdown Margin Available (pcm - EOC)	2696	2436
Cycle Length (months)	24	18

Section 4.4 Thermal and Hydraulic Design

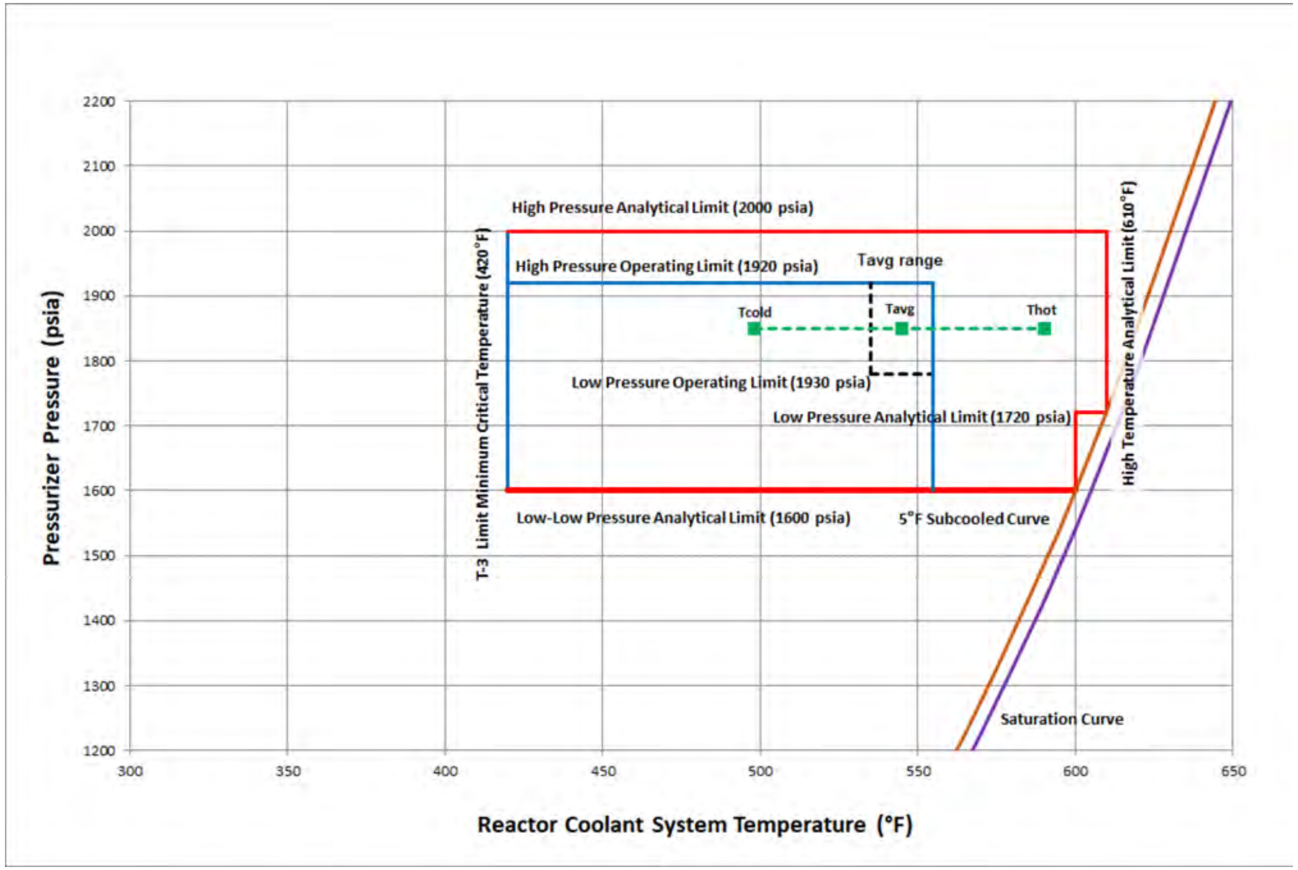
- Changes from DCA
  - Treatment of CHF uncertainties implementing TR-108601-P-A, Revision 4, “Statistical Subchannel Analysis Methodology, Supplement 1 to TR-0915-17564-P-A, Revision 2”
    - New technical report – TR-169856-P, Revision 0, “NuScale US460 Statistical Subchannel Critical Heat Flux Analysis Probabilistic Uncertainties”
  - NSPN-1 critical heat flux correlation for rapid depressurization events
    - NSPN-1 analysis limit – 1.20
  - NSP4 analysis limit – 1.43
  - Flow reduction of 20 percent applied to the limiting fuel assembly in the subchannel analysis
  - Changes related to power uprate
    - Flow rate
    - Average temperature
    - System pressure
- 3 audit questions resolved and no RAIs

Reactor Design Parameter Comparison

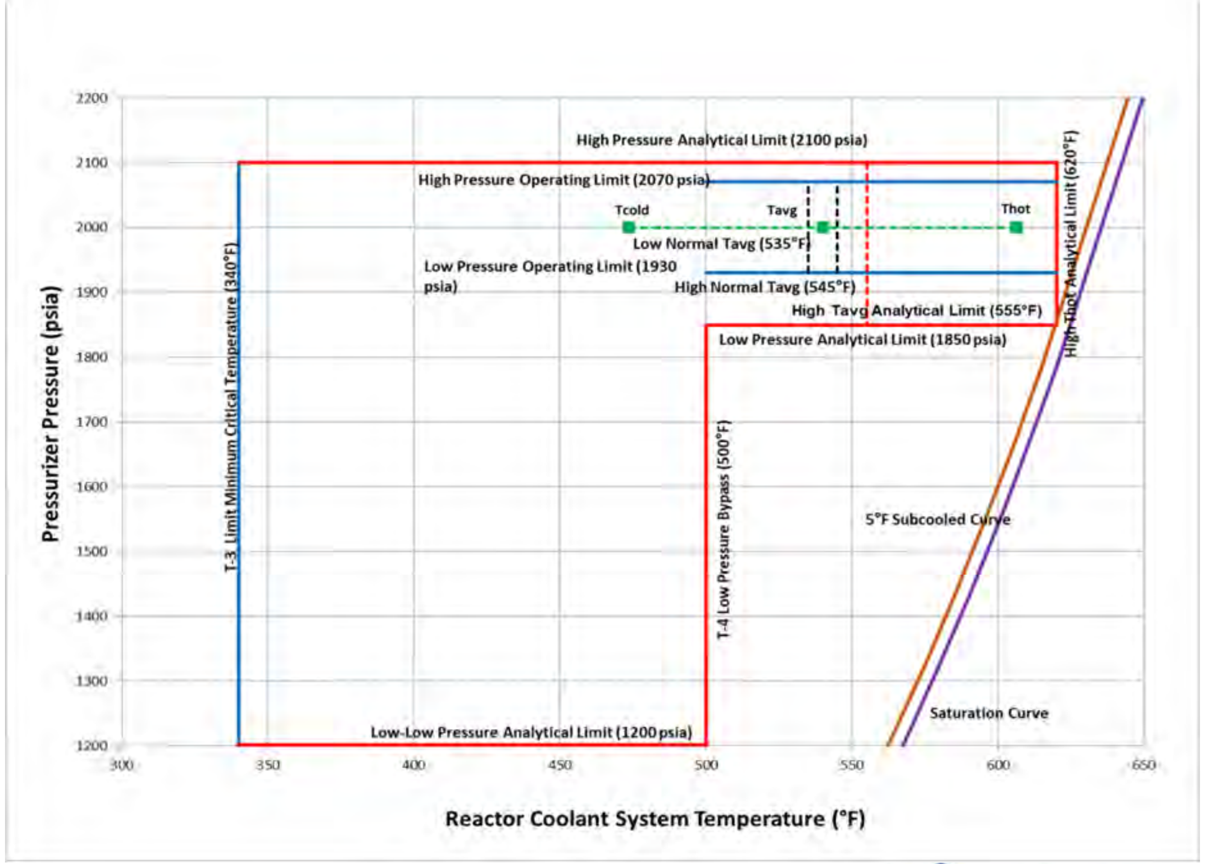
Parameter	NPM-160	NPM-20
Core thermal output	160	250
System pressure (psia)	1850	2000
Inlet temperature – best estimate flow (°F)	497	481
Core average temperature – best estimate flow (°F)	543	540
Core bypass flow (%)	8.5	7.5

Analytical Design Operating Limits

US600

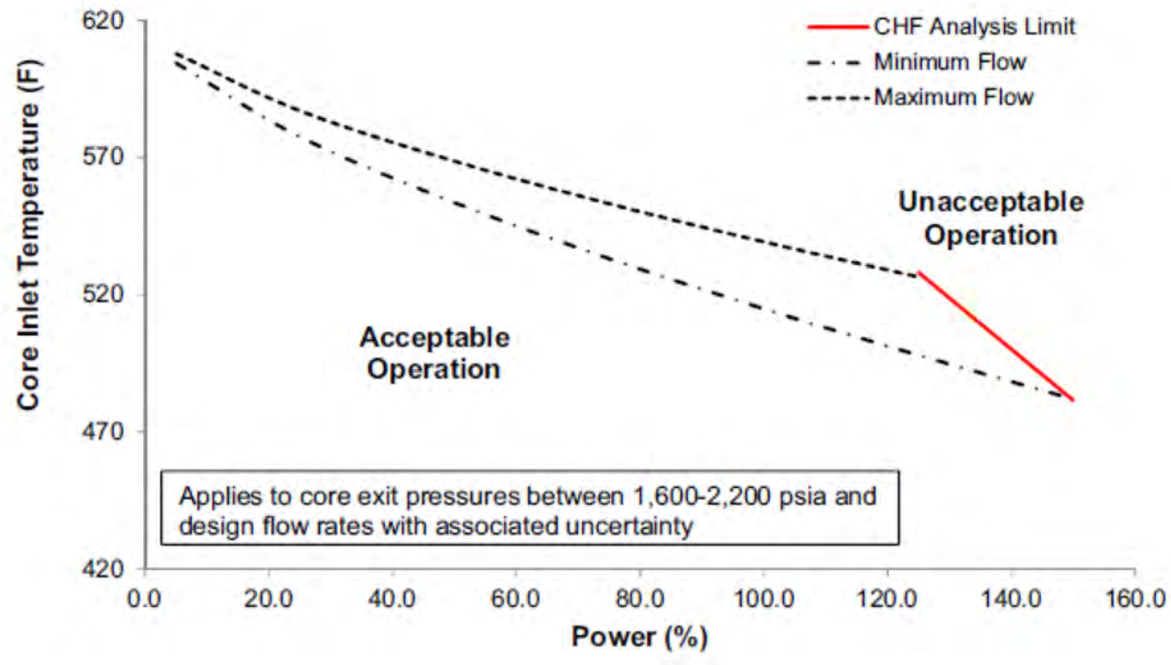


US460

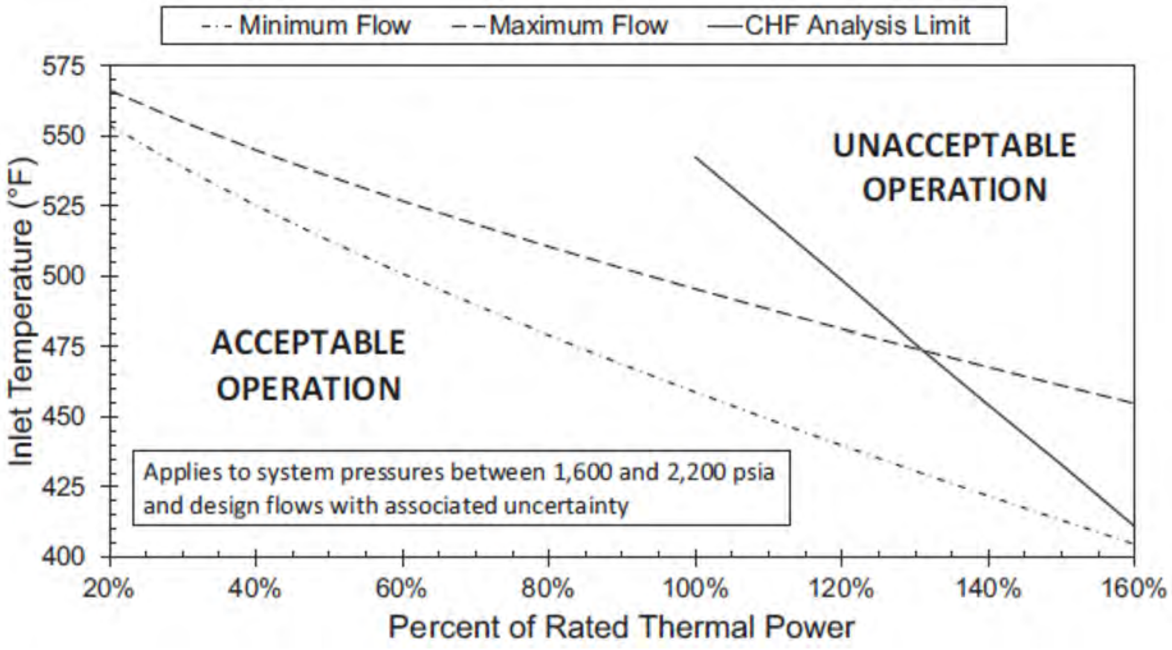


Thermal Margin Limit Map

US600



US460



Section 4.5 Reactor Materials

- Control Rod Drive System Structural Materials
  - Change from DCA
    - Control Rod Drive Mechanism (CRDM) cooling water pressure boundary components and water connections outside of the reactor coolant pressure boundary (RCPB) designed to ASME BPVC, 2018 Edition, B31.1.
    - Removed applicability of Paragraph NC-2160 and Subarticle NC-3120 for materials exposed to borated water
      - Materials selected for the SDAA comply with NB-2160 and NB-3120
    - Added additional alloy options such as Alloy 625, Alloy 718, and Type 440C to improve strength
- Reactor Internals and Core Support Structure Materials
  - No significant material changes from DCA to SDAA
  - RVI materials are austenitic stainless steel of various grade, class, or type
- 9 audit questions resolved and no RAIs

Section 4.6 Functional Design of Control Rod Drive System

- Changes from DCA
  - Mechanical design changes are described in SDAA FSAR Section 3.9.4
    - Pressure housing is bolted instead of welded to reactor pressure vessel (RPV) head
    - Addition of rod hold out device
- Safety function of the CRDM remains the same between the DCA and SDAA
  - Release the control rod assemblies (CRAs) during a reactor trip
  - Maintain the pressure boundary for the RPV
- 3 audit questions resolved and no RAIs

Acronyms

AO	Axial Offset
ASME	American Society of Mechanical Engineers
BPVC	Boiler and Pressure Vessel Code
CHF	Critical Heat Flux
CRA	Control Rod Assembly
CRDM	Control Rod Drive Mechanism
DCA	Design Certification Application
ECCS	Emergency Core Cooling System
EOC	End of Cycle
ESB	ECCS Supplemental Boron
FSAR	Final Safety Analysis Report
GDC	General Design Criterion
HFP	Hot Full Power
HZP	Hot Zero Power
LCO	Limiting Condition for Operation
RAI	Request for Additional Information
RCPB	Reactor Coolant Pressure Boundary
RPV	Reactor Pressure Vessel
RVI	Reactor Vessel Internals
SSC	Systems, Structures, and Components
SDAA	Standard Design Approval Application

ACRS Subcommittee Meeting (Open Session)

April 1, 2025

Chapter 15 – Transient and Accident Analyses

Presenters: Kevin Lynn, Meghan McCloskey,
Ben Bristol

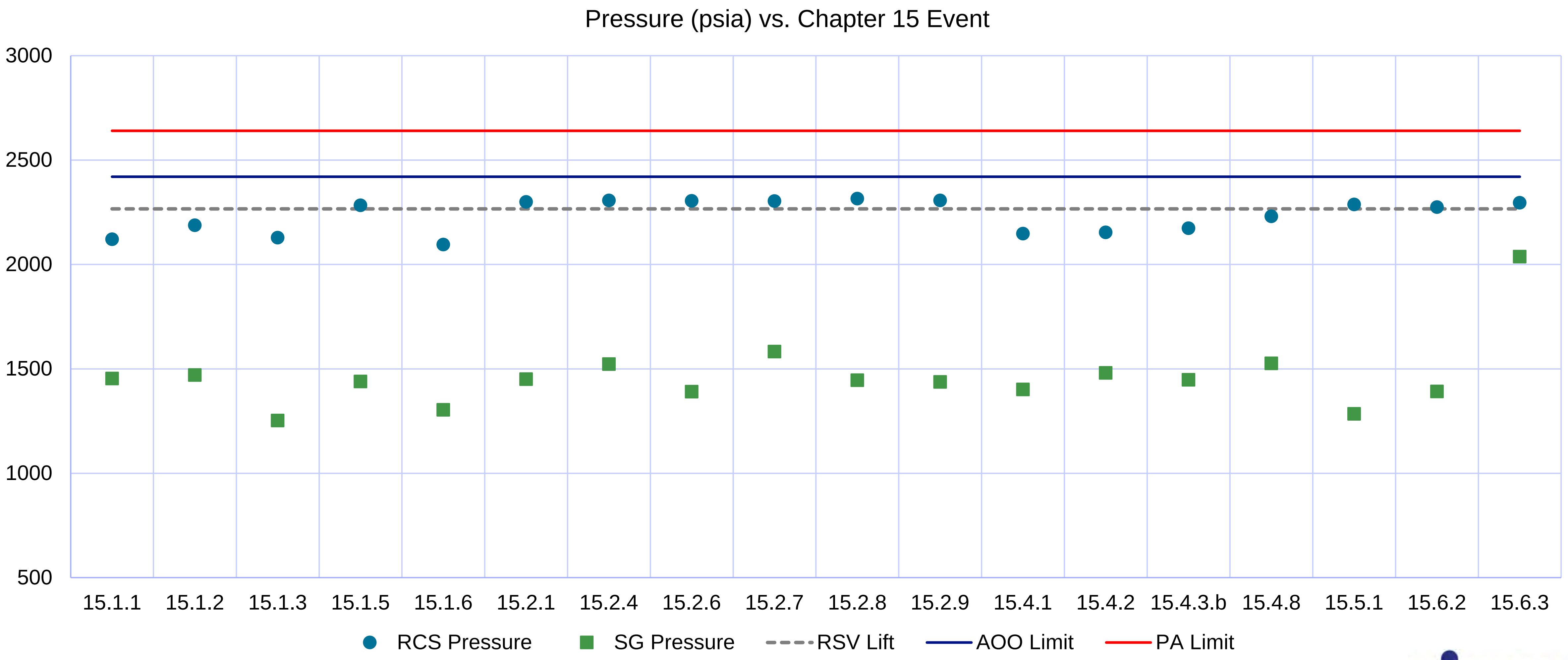
Agenda for Chapter 15

- Summary of review and current status
- Overview of analysis results
  - Primary and secondary pressure
  - Minimum critical heat flux ratio (MCHFR)
  - Loss-of-coolant accident (LOCA) and inadvertent opening of a reactor valve (IORV) event results
  - Radiological consequences
- Key differences from prior review
  - Long-term cooling without return to power
  - LOCA break spectrum high impact technical issues (HITIs)
  - Secondary side oscillation analysis
- Additional topic – augmented direct current (DC) power system (EDAS) considerations

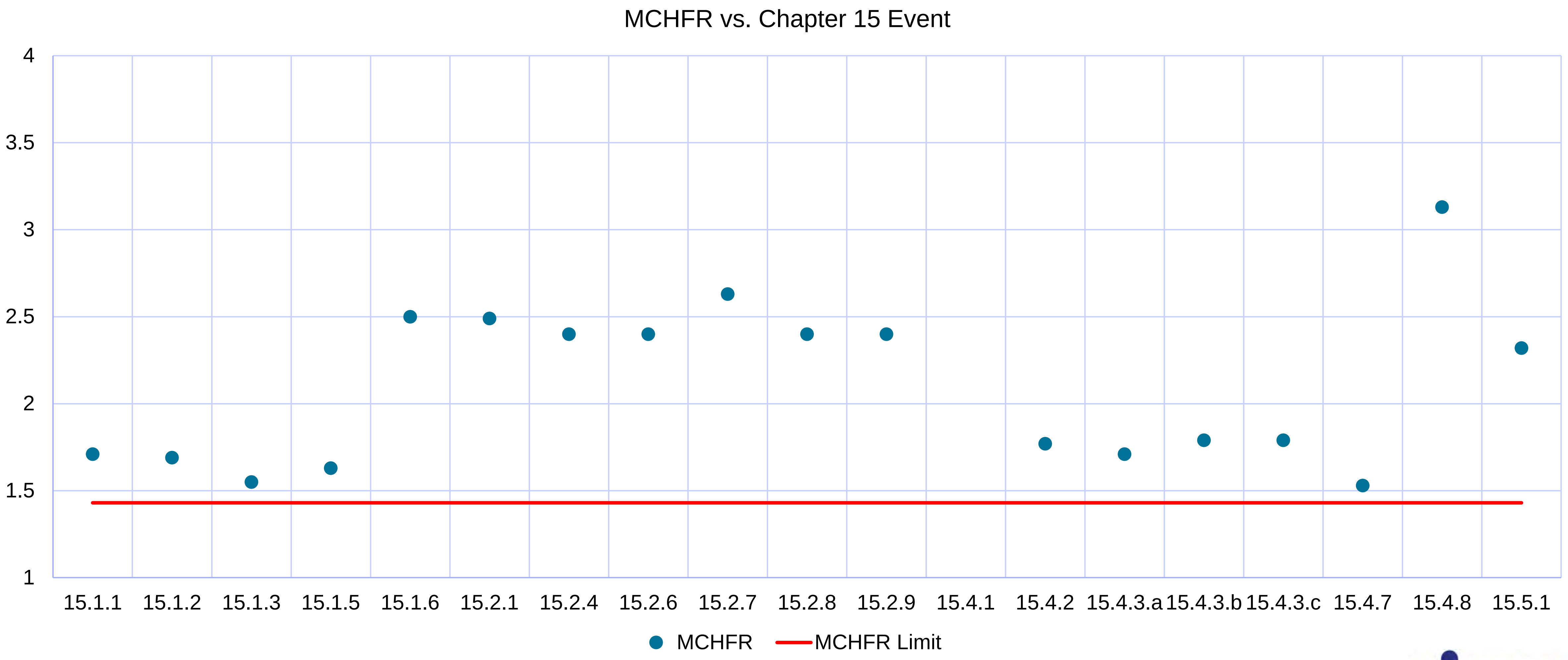
Chapter 15 Review Summary

- Total of 105 audit questions received by NuScale
  - 96 audit questions resolved during the audit
  - 9 audit questions sent to request for additional information (RAI) process
- Total of 10 RAI questions received by NuScale
  - 8 RAI questions resolved
  - 2 draft RAI questions on LOCA break spectrum HITI resolved by supplemental audit responses

Primary and Secondary Pressure Results vs. Acceptance Criteria (Non-LOCA)



MCHFR Results vs. Acceptance Criteria (Non-LOCA)



LOCA and IORV Events Results vs. Acceptance Criteria

- Design-basis LOCA break spectrum is for breaks inside containment
  - Chemical and volume control system (CVCS) discharge and injection lines (liquid-space breaks)
  - Pressurizer spray and reactor pressure vessel (RPV) high point vent (HPV) lines (vapor-space breaks)
- Design-basis IORV spectrum is for valve opening events
  - Single valve opens: reactor vent valve (RVV), reactor recirculation valve (RRV), reactor safety valve (RSV)
  - Two valves open: emergency core cooling system (ECCS) actuation (i.e., both RVVs open)
  - Multiple valves open: single valve opens (RRV or RSV) plus loss of EDAS causes RVVs to open

Parameter	Acceptance Criteria	LOCA Results	IORV Results
MCHFR	> 1.20	1.35	1.41
Minimum collapsed liquid level	> 0 ft above top of core	> 8 ft above top of core	
Containment pressure	< 1200 psia	< 920 psia (from Chapter 6)	
Containment temperature	< 600°F	< 535°F (from Chapter 6)	

Dose Results vs. Acceptance Criteria

Event	Offsite Exclusion Area Boundary (EAB) and Low Population Zone (LPZ)		Main Control Room	
	Results	Acceptance Criteria	Results	Acceptance Criteria
• Failure of small lines carrying primary coolant outside containment • Steam generator tube failure • Main steam line break • Iodine spike design-basis source term	0.83 (maximum of EAB and LPZ for listed events for either spiking)	< 2.5 (event with coincident spike) < 25 (event with pre-incident spike)	0.25 (maximum of listed events)	< 5
Fuel handling accident	1.60 (EAB) 1.60 (LPZ)	< 6.3	0.55	< 5
Core damage event	2.39 (EAB) 4.95 (LPZ)	< 25	1.31	< 5

*All values are in rem total effective dose equivalent (TEDE)

Long-Term Cooling without Return to Power

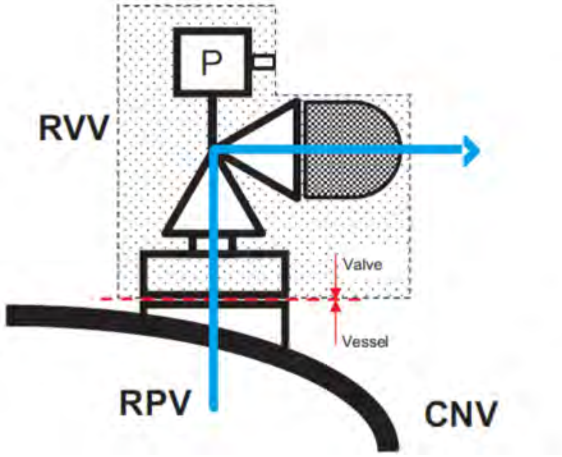
- US600 (certified design) evaluated return to power and took exemption from general design criterion (GDC) 27
- US460 prevents return to power and meets GDC 27
  - ECCS supplemental boron provides additional negative reactivity to maintain subcriticality, assuming highest worth control rod stuck out
- Conservative analysis scope and method per the extended passive cooling (XPC) evaluation model (EM)
  - Analysis bounds anticipated operational occurrence (AOO), infrequent event, postulated accident (PA) initiating events
  - Analysis bounds wide range of off-normal power operating histories
  - High-biased critical boron concentration (CBC) calculation and boron transport method results in low-biased core concentration to conservatively minimize margin
- Results:
  - Non-LOCA event analyses more limiting than LOCA due to later ECCS actuation
  - Minimum margin in non-LOCA cases occurs 28-40 hours after event initiation due to xenon decay; then margin increases as core boron concentration continues to increase
  - Lower riser holes assures fluid in the downcomer remains near the core boron concentration

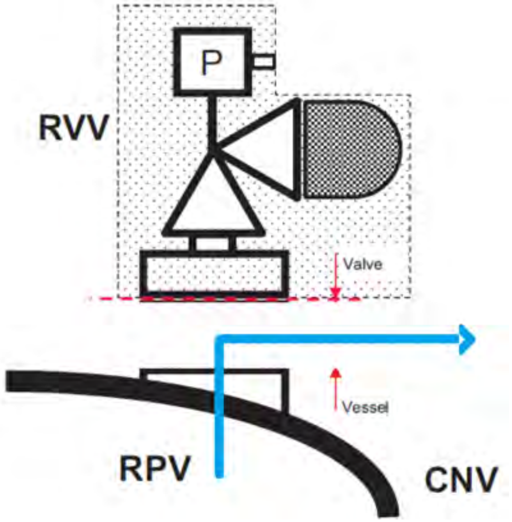
Event	Minimum Margin to CBC (ppm) [Time]	Approximate Margin to CBC (ppm) at 72 hours
LOCA Injection line break	134 [at 4.2 hours]	> 200
Non-LOCA Reactor component cooling water (RCCW) break Slow-biased ESB	30 [at 42.2 hours]	~ 50
Non-LOCA RCCW break Fast-biased ESB	28 [at 29.4 hours]	~ 150

LOCA Break Spectrum HITIs

Connections between ECCS valves and RPV

- Four valves total per NuScale power module (NPM)
- In design-basis valve opening events, flow is restricted by venturi (figure on left below)
- Hypothetical break at flange (figure on right below) would allow flow path without venturi
- Larger flow area has potential to be more limiting for MCHFR and containment (CNV) response (but non-limiting for liquid level above top of fuel)

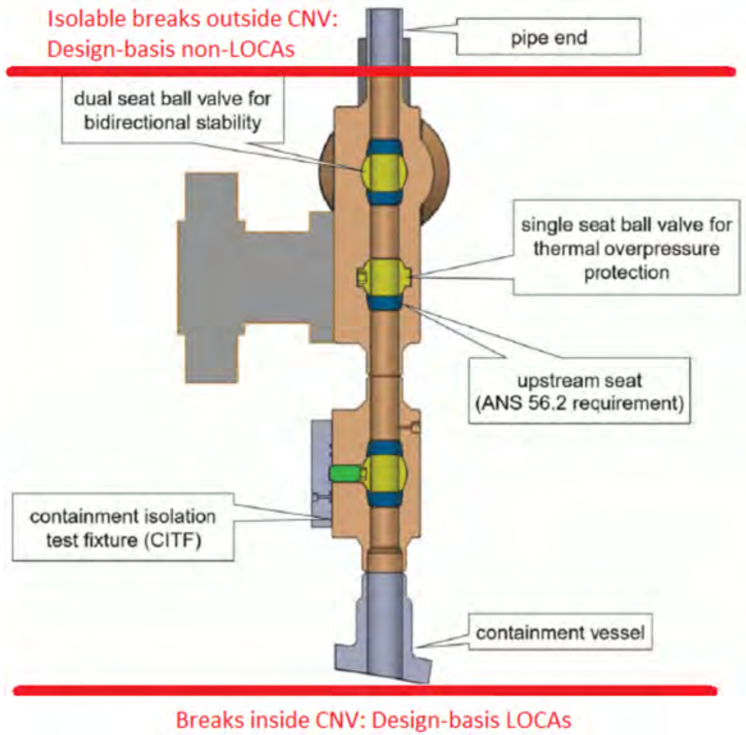




Based on FSAR Figure 6.3-3

Connections between CNV and CVCS piping

- Four CVCS lines total per NPM
- Hypothetical break would not be isolated by containment isolation valves and not all inventory would be retained within CNV
- Breaks in these locations have potential to be limiting for liquid level above top of fuel (but not for MCHFR and CNV response)



Based on FSAR Figure 6.2-4

LOCA Break Spectrum HITIs (continued)

- Failures at these locations are unlikely due to design of the connections, design stress and fatigue limits applied, inspections, and detection capabilities
- Exemption from 10 CFR 50.46 and GDC 35 requested to classify these postulated failures as beyond-design-basis events
  - Analyses are performed for these postulated failures with alternate acceptance criteria
  - Analyses are performed with alternative assumptions compared to design-basis events
- Results show that event-specific acceptance criteria for core cooling, CNV response, and dose are met
  - Met with credit only for passive, safety-related design features
  - Consideration of active makeup systems provides additional defense-in-depth
- Conclusion: these failures are very unlikely, but US460 NPM design can passively mitigate these failures

Stability and Oscillation Analysis

- Primary coolant and power stability analyzed with PIM code in same manner as US600 (certified design) using previously approved methodology
  - Stability to small perturbations during normal operation
  - Stability during operational occurrences
  - Analyses confirm acceptance criteria are met (decay ratio < 0.8 or reactor trip prior to loss of riser subcooling)
- New scope of stability evaluation: consideration of continuous secondary side oscillations
  - Addresses potential control system issues – was Combined License (COL) Item 7.0-1 for US600
  - Analyzed in NRELAP5 with secondary side oscillation imposed on steam pressure or feedwater flow
  - Spectrum of cases with varied oscillation amplitudes, oscillation periods, initial reactor power levels, and times in cycle
  - Variety of module protection system (MPS) signals provide protection to terminate oscillations prior to challenging specified acceptable fuel design limits (SAFDLs)
  - Limiting cases for SAFDLs look similar to existing Chapter 15 events
    - Example: oscillation induced cooldown causes control system rod withdrawal that behaves like other rod withdrawal events
- Conclusion: operational events do not result in unstable behavior or are terminated by MPS prior to challenging SAFDLs

Bottom Line Up Front – Augmented DC Power System (EDAS)

- Safety: US460 exceeds Commission Safety Goals by orders of magnitude
  - The design includes nonsafety-related EDAS
- EDAS: NuScale went beyond DCA requirements and included additional OCRM requirements to address failure modes, reliability, and test and maintenance unavailability
- ECCS: The fundamental function of ECCS is the same for the US600 and US460 designs
  - ECCS actuation establishes continual, passive recirculation, requires no operator action, and requires no electrical power
    - Removal of RVV IABs allows earlier ECCS valve opening and improves ECCS effectiveness
  - Both designs include nonsafety-related electrical power to ECCS valves
- RCPB Integrity :
  - ECCS valve actuation as it pertains RCPB integrity was raised and resolved by NRC staff during the DCA review of the *Safety Classification of Passive Nuclear Power Plant Electrical Systems* topical report
- SRM-SECY-19-0036:
  - "… In any licensing review or other regulatory decision, the staff should apply risk-informed principles when strict, prescriptive application of deterministic criteria such as the single failure criterion is unnecessary to provide for reasonable assurance of adequate protection of public health and safety."

Loss of Power Considerations – General Background

- The GDCs require safety functions to be performed with onsite or offsite electric power available
  - GDC 17 addresses electric power systems, generally: safety functions to be performed “assuming the other system is not functioning”
  - GDCs 34, 35, 38, 41, and 44 require system-specific performance for either onsite or offsite power operation

- Typical operating plant implements GDC 17 in safety analyses by assuming:
  - Offsite power available throughout event, or
  - Loss of offsite power (prompting safety-related onsite power to take over)
    - Coincident with event initiation
    - After reactor trip as consequence of the reactor trip and turbine trip with delay times crediting grid stability

Loss of Power Approach for NuScale

- NuScale design goes further: performs safety functions with or without electric power
  - Supports exemption from GDC 17
  - Intent of GDC 17 is met as described in FSAR Section 3.1: "With electric power unavailable, safety-related SSC have sufficient capacity and capability to ensure (1) specified acceptable fuel design limits and design conditions of the RCPB are not exceeded as a result of AOOs and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents."
  - Conforming PDCs require systems perform their safety functions without electric power
- US460 implements design-specific principal design criteria (PDC) in safety analyses by assuming electric power is unavailable
  - Chapter 15 event analyses evaluate availability of alternating current (AC) power and EDAS
    - Loss of AC power at time of event initiation or time of reactor/turbine trip
    - EDAS power supply available or unavailable coincident with event intiation
  - Conservative, nonmechanistic assumption
  - Demonstrates electric power is not credited to mitigate design-basis events, and therefore AC and DC power supply systems are nonsafety-related

Loss of Power Considerations – Maintaining Reactor Coolant Pressure Boundary (RCPB) Integrity

Background & US600 History

- The ECCS valves are designed to open if electric power (EDSS in the US600 design) is lost
  - o Ensures the key safety function of ECCS is fulfilled by establishing passive core cooling
  - o Fundamental safety feature of the US460 design, as with the US600 design
- In the review of US600 DCA, the Commission determined inadvertent ECCS operation was not a loss of RCPB integrity
  - o Staff considered during review of TR-0815-16497-P-A, “Safety Classification of Passive Nuclear Power Plant Electrical Systems”
  - o Staff questioned whether nonsafety-related was sufficient to maintain RCPB integrity
    - ▪ On loss of EDSS, ECCS valves opened when IAB set pressure reached (~950 psid)
    - ▪ GDC 15 requires that the design conditions of the RCPB are not exceeded during normal operation or AOOs
    - ▪ NuScale understands GDC 15 to concern gross failure of the RCPB due to over-pressurization
    - ▪ ECCS valve opening does not challenge the design conditions of the RCPB
  - o Staff concluded that ECCS opening during AOO “may not be consistent with the underlying defense-in-depth <u>purpose</u> of GDC 15”
    - ▪ Resolved by limiting the expected frequency of occurrence via limitation and condition (L&C) 4.4, requiring a probabilistic determination that the expected frequency of an AOO and an actuation of the ECCS is not expected to occur in the lifetime of the module
  - o With L&C 4.4 satisfied, NRC concluded RCPB integrity was consistent with requirements – <u>no exemption required for nonsafety-related EDSS</u>
  - o Commission’s Statements of Consideration for US600 Design Certification Rulemaking confirmed:
    - ▪ “The NRC reviewed topical report TR-0815-16497 and concluded that NuScale Power demonstrated that the safety-related systems do not rely on Class 1E electrical power.”
    - ▪ “Because no safety-related functions of NuScale rely on electrical power, NuScale does not need any safety-related electrical power systems.”

Loss of Power Considerations – Maintaining RCPB Integrity (continued)

US460 Approach

- The ECCS valves are designed to open if electric power (EDAS) to the ECCS valves is lost
- US460 licensing basis follows approach approved in TR-0815-16497-P-A
  - Similar augmented requirements to ensure reliability of EDAS
  - Applies and meets L&C 4.4 to ensure frequency of an AOO and an actuation of the ECCS is less than once in the lifetime of a module
- US460 design does not include IABs on ECCS RVVs
  - Improves overall plant safety by enhancing ECCS mitigative capabilities for some events
  - As a result, on loss of EDAS the ECCS would open at a higher RCS pressure than would occur for the US600 design
  - Not a material difference with respect to RCPB integrity:
    - "Underlying defense-in-depth purpose of GDC 15" still met by limiting frequency
    - Inadvertent ECCS on loss of EDAS is an analyzed event (assumed AOO) with substantial safety margins for core cooling and containment integrity
- EDAS is not relied upon to ensure RCPB integrity

Loss of Power Considerations – US460 Safety Analyses

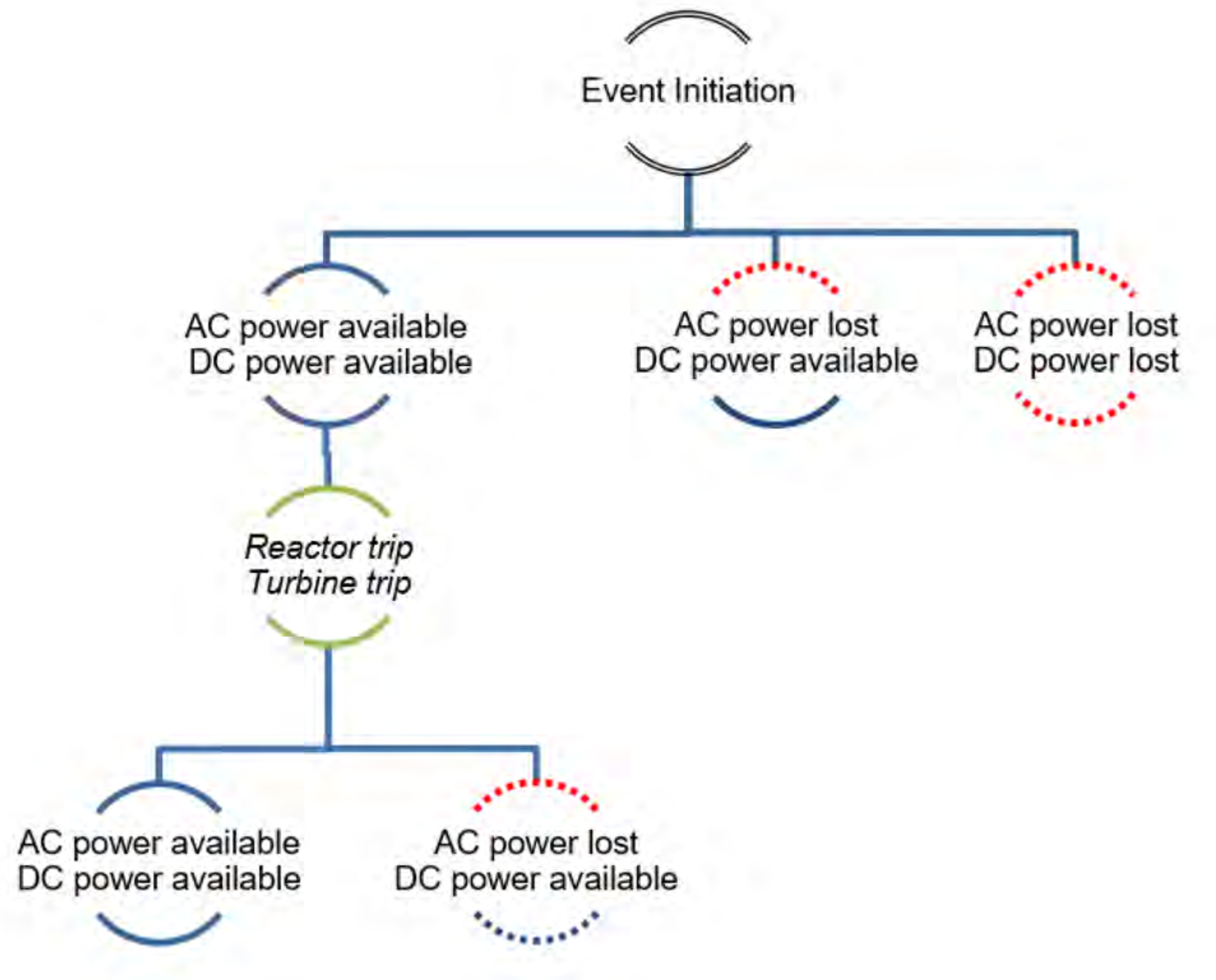
- US460 design change to remove RVV IABs improves plant safety overall in context of public health and safety
- Removal of IABs addressed in safety analysis event sequences
  - Multiple pre-application engagements with NRC discussing EDAS failure treatment in Chapter 15 analysis, and whether evaluating unrelated EDAS random failure was required to demonstrate that the system was not ‘relied upon to remain functional’ to assure RCPB integrity, in context of the 10 CFR 50.2 definition of safety-related
- NuScale submitted a new technical report, referenced in FSAR Chapter 15, describing:
  - Augmented requirements on EDAS
  - Evaluation of how the augmented requirements protect EDAS from effects of design-basis initiating events, to demonstrate that other initiating events are not expected to cause EDAS failure during the event progression
  - How Chapter 15 evaluates EDAS failure to demonstrate the system is not relied upon in the design-basis safety analysis
  - Quantification of frequency of an AOO and actuation of the ECCS as less than once in the lifetime of a module – providing assurance that the underlying purpose of GDC 15 is met, consistent with the L&C 4.4 on the previously approved topical report
  - Quantification of frequency of random EDAS failure and ECCS valve opening during a separate event: ~1E-8/year
  - Evaluation of consequences of assuming random EDAS failure and ECCS valve opening during a separate event under worst conditions

Loss of Power Considerations – US460 Safety Analyses (continued)

- NPM safety systems are designed to actuate to their safe position when power supply is removed from components
- Loss of EDAS power supply actuates the safety systems:
  - ECCS actuation – RVVs open (valves opening timing ~1 sec), RRVs remain closed initially due to IABs
  - Reactor trip (rod insertion timing ~2 sec)
  - Containment isolation, secondary system isolation, DHRS actuation (valve repositioning timing ~10-30 sec)
- Depressurization from RVVs opening reduces coolant temperature and causes flow reduction as power decreases due to rod insertion – very short duration (i.e., less than 2 sec) reduction in MCHFR

Loss of Power Treatment in US460 Chapter 15

- AC power loss timing consistent with regulatory requirements and guidance
  - Event initiation – Deterministic assumption
    In some scenarios, initiating event may disrupt normal AC power supply (e.g., seismic event).
  - Reactor/turbine trip – Consequential failure
    Normal AC power supply is disrupted after turbine trip because grid disruption is identified as a causal failure in the event progression (assumed consistent with traditional practice even though single NPM is small).
- EDAS loss timing consistent with regulatory requirements and guidance
  - Event initiation – Deterministic assumption
    Demonstrates not relied upon for safety functions.
  - Unlike loss of offsite power, there is no failure mode where the initiating event progression would cause the EDAS power supply to fail
  - Treatment is consistent with EDAS design and augmented requirements



Loss of Power Treatment in US460 Chapter 15 (continued)

- Random loss of EDAS at time of worst conditions in the event progression – not considered to be a design-basis event, but submitted technical report included bounding assessment of consequences
  - Regardless of initial condition, no core damage occurs
  - Conservative MCHFR limits met for a subset of power conditions exceeding 102%
  - Significant margin to peak cladding temperature (PCT) criteria of 10 CFR 50.46 even if MCHFR limit not met
- Technical report was originally referenced in Chapter 15, but was later removed at NRC request
- NRC requested consideration of Technical Specifications (TS) for EDAS
  - NuScale provided justification for no need for TS
    - Power operation is not possible if EDAS is not functional
    - Loss of EDAS during power operation ensures safety functions of reactor trip, containment isolation, secondary system isolation, DHRS actuation, and ECCS actuation occur as designed
    - On loss of EDAS plant is placed in safe, stable condition with no need for further actions
- NuScale committed to control EDAS under Owner Controlled Requirements Manual (OCRM) and maintenance rule program (10 CFR 50.65)
  - Ensures system reliability and availability is maintained throughout plant lifetime

Loss of Power Considerations – Risk-Informed Review

- In SRM-SECY-19-0036, Commission directed NRC staff that the inadvertent actuation block (IAB) feature of ECCS valves for NPM did not need to be assumed as a single active failure
  - US600 has IABs on RRVs and RVVs, NRC staff believed it necessary to treat IABs as an active single failure
  - Commission directed that treating IAB failure as a passive failure was consistent with risk-informed review principles
  - SRM-SECY-19-0036 went further by providing more general direction to NRC staff: “In any licensing review or other regulatory decision, the staff should **apply risk-informed principles when strict, prescriptive application of deterministic criteria such as the single failure criterion is unnecessary** to provide for reasonable assurance of adequate protection of public health and safety.”
- Strict, prescriptive application of RCPB integrity criterion is unnecessary to provide for reasonable assurance of adequate protection
  - US600 review established that ECCS opening on loss of power is an issue of “underlying purpose,” not compliance
  - A conflicting, stricter interpretation here does not advance public health and safety
    - As with IAB single failure, loss of EDAS is a low frequency event with insignificant consequences

Loss of Power Considerations – Conclusions

- Nonsafety-related classification of EDAS is appropriate
- Control of EDAS in OCRM and under maintenance rule program combined with augment requirements is appropriate to ensure reliability and availability is maintained during operation
- Safety analyses considering EDAS available or unavailable at event initiation is sufficient to demonstrate that EDAS is not relied upon to mitigate design-basis events, consistent with nonsafety-related classification
- Design-basis event progressions do not require consideration of random loss of EDAS during unrelated event at time of worst conditions
- Even if random loss of EDAS during unrelated event at time of worst conditions is considered, consequences are minimal (core cooling maintained)
- The removal of IABs was a design change made to improve overall plant safety
- Commission direction in SRM-SECY-19-0036 emphasizes that strict, prescriptive application of deterministic criteria are unnecessary when risk informed principles provide for reasonable assurance of safety

EDAS Related Topic – ACRS Question on ECCS Solenoid Valves

- ECCS valves have two in series safety-related trip solenoid valves
  - Both must actuate to actuate ECCS – prevents single failure from causing inadvertent ECCS actuation
  - Valves fail in safe (i.e., actuated) position – ensures single failure does not prevent ECCS actuation
- Previous ACRS meetings identified question regarding one solenoid valve failed
  - For RVVs, subsequent failure of other solenoid valve would cause that RVV to open
  - For RRVs, IAB would prevent that RRV from opening even if other solenoid valve subsequently failed
- Known failure of a solenoid valve during operation would require operability determination for the supported ECCS valve under TS 3.5.1
  - If supported ECCS valve is inoperable, TS 3.5.1 requires restoration of operability within 72 hours or else shut down
  - If supported ECCS valve is operable, TS 3.5.1 has no time-limiting restrictions, so continued operation may be possible. However, licensee remains responsible for compliance with licensing basis, including Section 15.0.0.6.3:

    An analysis … is conducted to quantify the frequency for which a combination of an AOO and an actuation of the ECCS is expected to occur, and the analysis concludes that ECCS actuation in response to an AOO or IE is not expected to occur in the lifetime of an NPM.

EDAS Related Topic – ACRS Question on ECCS Solenoid Valves (continued)

- In Chapter 15 safety analyses, single failures are applied to mitigating systems
  - For events where ECCS is needed – single failure of other solenoid valve opens the RVV (i.e., safety function met)
  - For events where ECCS is not needed – random single failure of other solenoid valve does not need to be considered
- Initiating events are analyzed separately
  - Event consequences are analyzed (e.g., if the initiating event results in failure of some other system or component)
  - Random component failures are not assumed to occur during the event
- Evaluating a reactivity insertion event or cooldown event with random failure of a solenoid causing the ECCS valves to open combines two initiating events and is not required in the deterministic design basis event scope
  - For example: Operating plants are not required to evaluate a reactivity insertion event with a random failure of feedwater flow, or reactor coolant flow, that could otherwise be postulated due to failure of the nonsafety-related pump or failure of the nonsafety-related normal AC power supply.
- NRC review focus on EDAS (not solenoid valve) failure due to interest in system safety classification per 10 CFR 50.2
- Consequences of a random solenoid valve failure (with one already failed) causing an ECCS valve to open would be similar to previous analyses

Chapter 15 Conclusions

- All review questions resolved
- All acceptance criteria met
- US460 NPM design passively mitigates Chapter 15 events with reasonable assurance of adequate protection of the public health and safety

Acronyms

AC	alternating current
AOO	anticipated operational occurrence
CBC	critical boron concentration
CNV	containment vessel
COL	combined license
CVCS	chemical and volume control system
DC	direct current
EAB	exclusion area boundary
ECCS	emergency core cooling system
EDAS	augmented DC power system (US460)
EDSS	highly reliable DC power system (US600)
EM	evaluation model
ESB	ECCS supplemental boron
FSAR	final safety analysis report
GDC	general design criterion
HITI	high impact technical item
HPV	high point vent
IAB	inadvertent actuation block
IORV	inadvertent opening of a reactor valve
L&C	limitation and condition
LOCA	loss-of-coolant accident
LPZ	low population zone
MCHFR	minimum critical heat flux ratio
MCR	main control room
MPS	module protection system
NPM	NuScale power module
OCRM	owner controlled requirements manual
PA	postulated accident
PCT	peak cladding temperature
PDC	principal design criteria
RAI	request for additional information
RCCW	reactor component cooling water
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RPV	reactor pressure vessel
RRV	reactor recirculation valve
RSV	reactor safety valve
RVV	reactor vent valve
SAFDL	specified acceptable fuel design limit
SDAA	standard design approval application
SE	safety evaluation
SG	steam generator
TEDE	total effective dose equivalent
TS	technical specification
XPC	extended passive cooling

Presentation to the Advisory Committee on Reactor Safeguards Subcommittee

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

Chapters 1, 4 and 15

April 1, 2025
(Open Session)

Non-Proprietary

Presentation to the ACRS Subcommittee
Staff Review of NuScale SDAA FSAR, Revision 1

Chapter 1, “Introduction and General Description of the Plant”

April 1, 2025

(Open Session)

NuScale SDAA FSAR Chapter 1 Review

Contributors

- Technical Reviewer
  - Getachew Tesfaye, Lead PM, NRR/DNRL/NRLB
- Project Manager
  - Getachew Tesfaye, Lead PM, NRR/DNRL/NRLB

US460 SDAA Review Overview

- US460 pre-application activities begun in 2019 with the submittal of a regulatory engagement plan followed by a public meeting
- Eight topical reports submitted during the preapplication phase
- SDAA staged submittal was completed in January 2023, including four new topical reports
- The NRC staff issued the results of its acceptance review with a request for supplemental information (RSI) on March 17, 2023
- The staff began detailed safety evaluation of portions of the application not impacted by the RSI on March 20, 2023
- Following the receipt of the supplemental information on July 14 and 17, 2023, a docketing letter was issued on July 31, 2023, that included a four phase, 24-month review schedule

Staff Review Approach for SDAA

- Four Phase Review for SDAA vs Six phase review for DCA
- Use of extended audit process via NuScale's electronic reading room (eRR) for efficient review of the application
  - Facilitated easy access to calculations and other supporting documents
  - Minimized the number of RAI

NuScale SDAA FSAR Chapter 1 Review

Overview

- NuScale submitted Chapter 1, “Introduction and General Description of the Plant” Revision 0 of the SDAA FSAR on December 31, 2022, and Revision 1 on October 31, 2023
- NRC regulatory audit of Chapter 1 was performed from March 2023 to August 2023, generating one audit issue that was resolved in the audit
- No RAI resulted from chapter 1 review
- Staff completed Chapter 1 review and issued an advanced safety evaluation to support today’s ACRS Subcommittee meeting
- The draft SE provided to ACRS on March 3/4/25 was updated to include supplemental information submitted by NuScale on March 17, 2025, and is reflected in the SE submitted on 3/25/25.

NuScale SDAA FSAR Chapter 1 Review

Notable differences between NuScale DCA FSAR and SDAA FSAR with Impact to Chapter 1 SE

- Elimination of Chapter 20, “Mitigation of Beyond-Design-Basis Events,” and Chapter 21, “Multi-Module Design Considerations” from SDAA
- SDAA does not use Topical Report TR-0815-16497-P-A, “Safety Classification of Passive Nuclear Power Plant Electrical Systems”
- Two exemption requested in the DCA were not requested for the SDAA.

NuScale SDAA FSAR Chapter 1 Review

Notable differences between NuScale DCA FSAR and SDAA FSAR with Impact to Chapter 1 SE (Continued)

- Three new exemptions requests were added in the SDAA
- Staff evaluation of exemption request for GDC 19 is in Chapter 6 SE. It was in Chapter 1 SE for DCA.
- For the SDAA, only applicable sections of topical reports and technical reports are incorporated by reference (IBR). For the DCA all sections of topical and technical reports were IBRed.

NuScale SDAA FSAR Chapter 1 Review

Conclusions

- Information from topical and technical reports incorporated by reference (IBR) in Section 1.8 adequately address applicable regulatory requirements
- Chapter 1 SE does not include a safety finding. SDAA safety findings are in chapters 2 through 19.

Presentation to the ACRS Subcommittee Staff Review of NuScale SDAA FSAR, Revision 1

Chapter 4, "Reactor"

April 1, 2025

(Open Session)

NuScale SDAA FSAR Chapter 4 Review

Overview

- NuScale submitted Chapter 4, “Reactor” Revision 0 of the SDAA FSAR on December 31, 2022, and Revision 1 on October 31, 2023
- NRC regulatory audit of Chapter 4 was performed from March 2023 to August 2024, generating 76 audit issues
- Questions raised during the audit were resolved within the audit. One RAI was issued, and the response was acceptable
- Staff completed Chapter 4 review and issued an advanced safety evaluation to support today’s ACRS Subcommittee meeting
- One significant change between draft SE provided to ACRS on 3/4/25 and SE submitted on 3/25/25

NuScale SDAA FSAR Chapter 4 Review

Significant differences between previously submitted SER

- One significant difference in Section 4.3.4 “Technical Evaluation” following closure of RAI question 4.3-28:
  - Section 4.3.4.1, “Power Distributions”, and Section 4.3.4.9, “Technical Specifications” - revised evaluation of the TS to include assessment of why a limiting condition for operation (LCO) is not needed for the heat flux hot channel factor (F_Q)

NuScale SDAA FSAR Chapter 4 Review

Contributors

- Technical Reviewers
  - Dr Zhian Li, NRR/DSS/SNRB
  - Dr Adam Rau, NRR/DSS/SNRB*
  - Antonio Barrett, NRR/DSS/SNRB
  - Dr Rosie Sugrue, NRR/DSS/SNRB*
  - Dr Cory Parker, NRR/DNRL/NVIB
  - John Honcharik, NRR/DNRL/NPHP*
  - Ryan Nolan, NRR/DSS/SNRB
  - Hiruy Hadgu, NRR/DSS/SNRB
  - Dr Andrew Bielen, RES/DSA/FSCB
  - Dr Kevin Kadooka, PNNL(Contractor)
  - Nicholas Klymyshyn, PNNL (Contractor)
  - Kenneth Geelhood, PNNL (Contractor)
- Project Management
  - Stacy Joseph, NRR/DNRL/NRLB
  - Getachew Tesfaye, Lead PM, NRR/DNRL/NRLB

*Presenters

NuScale SDAA FSAR Chapter 4 Review

Sections

- Section 4.1 – Summary Description
- Section 4.2 – Fuel System Design
- Section 4.3 – Nuclear Design
- Section 4.4 – Thermal-Hydraulic Design
- Section 4.5 – Reactor Materials
- Section 4.6 – Functional Design of the Control Rod Drive System

NuScale SDAA FSAR Chapter 4 Review

Section 4.2 Fuel System Design

- Significant differences between NuScale DCA FSAR and NuScale SDAA FSAR include:
  - Implementation of TR-108553-P-A: Applicability of Framatome methodologies for the new NPM-20 design
    - Approved in 2022 for NPM-20 operating parameters (power, pressure, flow)
    - NuScale Performance Calculation
    - FAST confirmatory analyses
  - Cladding stress intensity limits
  - Fuel Seismic Analysis with new core plate input motions
    - Changed building footprint, UHS dimensions and pool level, construction materials, hydrodynamic loads

NuScale SDAA FSAR Chapter 4 Review

Section 4.3 Nuclear Design

- Significant differences between NuScale DCA FSAR and NuScale SDAA FSAR include:
  - New equilibrium core design for higher power level
    - Increased power, power density, linear power generation rate
  - Fuel does not include axial blankets (i.e., reduced U-235 enrichment or natural uranium)
  - Added emergency supplemental boron (ESB) system

NuScale SDAA FSAR Chapter 4 Review

Section 4.3 Nuclear Design

- Staff reviewed & audited updated calculations for:
  - Normalized power distributions:* assembly, pin-wise, axial
  - Control rod worth and lifetime limit
    - Integral control rod worth*
    - Differential control rod worth
    - Loss of control rod worth is limited through exposure limits
  - Shutdown margin
    - Short term* (min 2436 pcm, most reactive rod stuck out)
    - Long term (Extended Passive Cooling (XPC) methodology) – discussed in 15.0.5
  - Doppler*, moderator temperature, and power defect coefficients
  - Updated RPV fluence calculation

* Indicates the staff performed confirmatory analyses with POLARIS/PARCS

NuScale SDAA FSAR Chapter 4 Review

Section 4.3 Nuclear Design

- US460 Generic Technical Specifications (GTS) include two power distribution LCOs:
  - Enthalpy rise hot channel factor ($F_{\Delta H}$)
  - Axial Offset (AO)
- Staff issued RAI 10269, Question 4.3-28 on the need for an LCO restricting peak linear heat generation rate (e.g., $F_Q(z)$, LHR)
- Staff findings:
  - Local peaking may exceed that considered in the AO window analysis
  - Higher peak LHGR may reduce MCHFR
- Staff is not requiring a US460 F_Q LCO because:
  - NPM-20 LHGR remains lower than operating PWRs
  - Safety analysis shows that fuel thermal limits would not likely be challenged

NuScale SDAA FSAR Chapter 4 Review

Section 4.4 Thermal-Hydraulic Design

- Significant differences between NuScale DCA FSAR and NuScale SDAA FSAR include:
  - Statistical critical heat flux analysis limit (SCHFAL)

Statistical Subchannel Analysis Limit Range of Applicability

Parameter	Applicability Range
Pressure (psia)	1800 to 2300
Inlet Temperature (degrees F)	431 to 568
Core inlet mass flux (Mlbm/hr-ft^2)	0.21 to 0.7
Core local mass flux (Mlbm/hr-ft^2)	0.20 to 0.721
Local equilibrium quality	< 68%

  - New critical heat flux correlation NSPN-1: used for rapid depressurization portions of applicable events. The correlation description and development is provided in the LOCA TR.

NuScale SDAA FSAR Chapter 4 Review

Section 4.4 Thermal-Hydraulic Design: Review Items

- Subchannel analysis
  - Statistical CHFR analytical limit
  - NSPN-1 CHF correlation
- Bypass flow calculations
  - Core bypass flow methodology and analysis was provided during audit
- Effects of Crud
  - Conservative heat transfer inputs for fuel rod conduction are used in COPERNIC to account for Crud

NuScale SDAA FSAR Chapter 4 Review

Section 4.5 Reactor Materials

- Significant differences between NuScale DCA FSAR and NuScale SDAA FSAR include:
  - Use of bolted connection for control rod drive mechanism (CRDM) in lieu of welded connection
  - Use of threaded inserts as part of bolted connection for the CRDM
- CRDM not routinely disassembled for inspection
- Degradation of the bolted connection (including stainless steel threaded inserts and alloy steel vessel head) could lead to shifting of the CRDM and could affect the safety function of the CRDM.

NuScale SDAA FSAR Chapter 4 Review

Section 4.5 Reactor Materials

- Augmented VT-1 examination on threaded inserts and its seal welds whenever an ASME Class 1 component is disassembled (routinely, such as):
  - SG Feedwater Plenum Access Covers, the SG Main Steam Plenum Access Covers, the Pressurizer Heater Bundles and the Instrument Seal Assemblies.
- Detection of defects in these areas requires sample expansion to include threaded inserts and seal welds for the CRDM connections.
- Staff finds this provides adequate assurance of the integrity of the threaded inserts and seal welds based on statistically significant number of threaded inserts being inspected

NuScale SDAA FSAR Chapter 4 Review

Conclusion

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

Non-Proprietary

Presentation to the ACRS Subcommittee
Staff Review of NuScale SDAA FSAR, Revision 1

Chapter 15, “Transient and Accident Analysis”

April 1, 2025

(Open Session)

NuScale SDAA FSAR Chapter 15 Review

Overview

- NuScale submitted Chapter 15, “Transient and Accident Analysis” Revision 0 of the SDAA FSAR on December 31, 2022, and Revision 1 on October 31, 2023
- NRC regulatory audit of Chapter 15 was performed from March 2023 to August 2024, generating 105 audit issues
- Questions raised during the audit were resolved within the audit. Eight RAIs were issued, and the responses were acceptable
- Staff completed Chapter 15 review and issued an advanced safety evaluation to support today’s ACRS Subcommittee meeting
- Two significant changes between draft SE provided to ACRS on 3/4/25 and SE submitted on 3/25/25

NuScale SDAA FSAR Chapter 15 Review

<u>Significant differences between previously submitted SER</u>

- Two significant differences
  - Section 15.0.5, "Extended Passive Cooling for Decay and Residual Heat Removal," revised to include evaluation of XPC TR RAIs
  - Section 15.6.5.3, "Beyond Design Basis Event Breaks," revised to reflect closure and evaluation of LOCA break spectrum open item.

NuScale SDAA FSAR Chapter 15 Review

Contributors

- Technical Reviewers
  - Antonio Barrett, NRR/DSS/SNRB*
  - John Parillo, NRR/DRA/ARCB
  - Ed Stutzcage, NRR/DRA/ARCB
  - Adam Rau, NRR/DSS/SNRB*
  - Zhian Li, NRR/DSS/SNRB*
  - Joshua Miller, NRR/DSS/SNRB
  - Rosie Sugrue, NRR/DSS/SNRB*
  - Ryan Nolan, NRR/DSS/SNRB*
  - Shanlai Lu, NRR/DSS/SNRB
  - Carl Thurston, NRR/DSS/SNRB
  - Sean Piela, NRR/DSS/SNRB*
  - Dong Zheng, NRR/DSS/SNRB*
  - Hiruy Hadgu, NRR/DSS/SNRB
  - Peter Lien, RES/DSA/CRAB II
- Project Management
  - Stacy Joseph, NRR/DNRL/NRLB
  - Getachew Tesfaye, Lead PM, NRR/DNRL/NRLB

*Presenters

NuScale SDAA FSAR Chapter 15 Review

Sections

- Section 15.0 – Introduction – Transient and Accident Analysis
- Section 15.1 – Increase in Heat Removal by the Secondary System
- Section 15.2 – Decrease in Heat Removal by the Secondary System
- Section 15.3 – Decrease in Reactor Coolant System Flow Rate
- Section 15.4 – Reactivity and Power Distribution Anomalies
- Section 15.5 – Increase in Reactor Coolant Inventory
- Section 15.6 – Decrease in Reactor Coolant Inventory
- Section 15.7 – Radioactive Release from Subsystem or Component
- Section 15.8 – Anticipated Transients without a Scram
- Section 15.9 – Stability
- Section 15.10 – Core Damage Event

NuScale SDAA FSAR Chapter 15 Review

Significant Changes Between DC and SDA

- Design and methodology changes that impact Chapter 15 include:
  - Power uprate and NRELAP version/numerous basemodel changes
  - Emergency core cooling system (ECCS) valve design and number
    - Removal of inadvertent actuation block (IAB) valves on RVVs
    - Addition of flow restricting venturis
  - ECCS actuation on riser level vs CNV level, new riser level instrumentation
  - Credit for decay heat removal system (DHRS) for LOCA and LOCA-like (IORV) events
  - No return to power during extended passive cooling
  - Addition of ECCS supplemental boron feature and additional riser flow holes
  - Change to dc power availability assumptions and reliance on augmented dc power system (EDAS)

NuScale SDAA FSAR Chapter 15 Review

Focus Areas for Review

- Extended Passive Cooling Analyses 15.0.5
- Analysis of Key Chapter 15 Events & Key Issues
  - 15.4.8 – Rod Ejection
    - Implementation of TR Limitations and Conditions
  - 15.4.6 - Boron dilution
    - Operator Actions
  - Cooldown & Reactivity Events (15.4.3 – CRA Misoperation & 15.1.3 – Increase in Steam Flow)
    - EDAS HITI
  - 15.2.8 – Feedwater Line Break
  - 15.6.3 – Steam Generator Tube Rupture
  - 15.6.6 – Inadvertent Operation of a Reactor Valve
  - 15.6.5 – LOCA
    - Thermal Dispersion Sensor
    - LOCA Break Spectrum HITI

NuScale SDAA FSAR Chapter 15 Review

Section 15.0.5- Extended Passive Cooling Analyses

- Limiting Minimum Level Event – Steam Generator Tube Failure
  - Staff performed independent confirmatory analysis
  - Xc value for RVV compressible flow expansion factor is part of the ASME QME-1 qualification program
  - **Collapsed Liquid level above TAF – 1.8 ft**
- Boron Transport Precipitation Analysis – Inadvertent RVV Opening
  - Conservative assumptions for thermal hydraulic conditions
  - Staff confirmatory/sensitivity studies show fair amount of mixing
  - Assumed initial RCS boron concentration at maximum operational limit
  - **Margin to precipitation limit – 6250 ppm**
  - **Core peak concentration – 8490 ppm**

NuScale SDAA FSAR Chapter 15 Review

Section 15.0.5- Extended Passive Cooling Analyses

- Boron Transport Subcriticality Analysis – RCCW Line Break
  - Staff sensitivity calculations performed for NRELAP and MATLAB script
  - Nuclear Reliability Factor implementation review
  - Minimal non-condensable gas in the CNV
  - Mixing delay due to liquid density differences accounted for
  - **Margin to critical boron concentration – 28 ppm**

NuScale SDAA FSAR Chapter 15 Review

Section 15.0.5- Extended Passive Cooling Analyses

- Initial startup test (first module) for CNV boron dissolution and transport (RAI-10350 R1, 6.3-7) FSAR Table 14.2-40, “Test #40 Emergency Core Cooling System”

3. The ESB feature supports the ECCS by providing boron to recirculated coolant during ECCS operation.	safety-related	40.02.01

6. Verify ECCS supplemental boron (ESB) pellets dissolve following ECCS actuation. 7. Verify boron concentration in the NPM following ECCS actuation.	3) Verify boron pellet dissolution through visual inspection or physical measurement. 4) Take a coolant sample at a sampling point in the NPM liquid space. Note 3 and 4 can be performed together, or in any order.	[ITAAC 02.01.14] [ITAAC 02.01.19] 6. ESB boron dissolution is within the bounds established for the test, accounting for test conditions and uncertainty. 7. Coolant boron concentration at a sampling point in the NPM liquid space is within the bounds established for the test, accounting for test conditions and uncertainty.

NuScale SDAA FSAR Chapter 15 Review

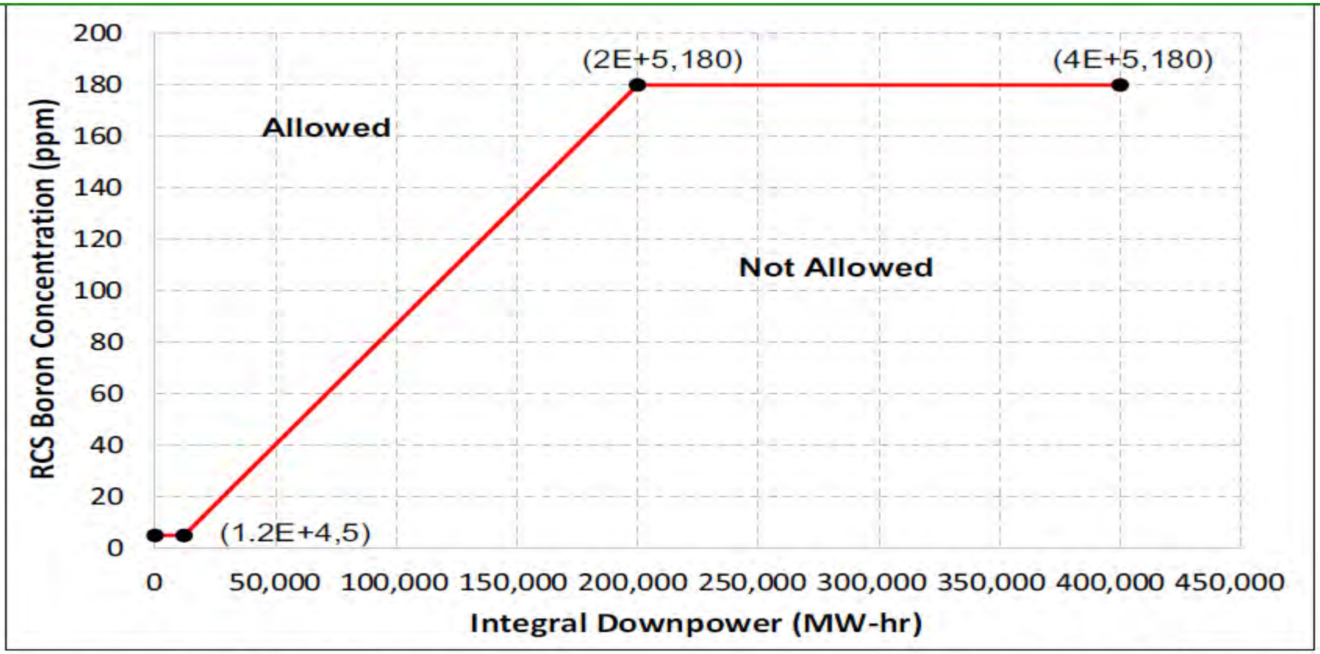
Section 15.0.5- Extended Passive Cooling Analyses

- Consideration of operating history of reduced-power impacts on short term xenon changes and potential for low decay heat
  - Technical specification LCO 3.5.4 – "The ESB shall be OPERABLE"
  - LCO 3.5.4 Condition A – "ESB operational limits specified in the COLR not met"
  - SR 3.5.4.2 – "Verify RCS boron concentration is within the ESB operational limits specified in the COLR"
  - Technical Specification Bases 3.5.4 – "Initial RCS boron concentrations greater than the ESB operational boron limit specified in the COLR, combined with other limitations associated with the boron limit, ensure core boron concentration remains above the critical boron concentration for at least 72 hours after event initiation."

NuScale SDAA FSAR Chapter 15 Review

Section 15.0.5- Implementation of XPC TR in Chapter 15

- SDAA Figure 15.0-16 – RCS minimum boron concentration limit requirements considering integral downpower (example COLR Limit)



Note: The limit in Figure 1 corresponds with a maximum power ascension rate limit of 25 MWt/hr. For power maneuvers with power ascent rates faster than 25 MWt/hr, the analysis uses a minimum RCS boron concentration limit of 300 ppm.

NuScale SDAA FSAR Chapter 15 Review

Section 15.4.8 –Rod Ejection Analysis

- Key Assumptions
  - Most reactive rod CRA stuck out
  - MPS actuation, pressurizer spray on
  - CRA ejection event with five initial power levels (0, 20, 50, 75, and 100%) and at BOC, MOC, and EOC.
  - Delay in core trip, most positive MTC
- 15.4.8 – Limiting Rod Ejection Analysis Results
  - MCHFR = 3.13 (Limit 1.43)
  - Peak RCS pressure = 2231 psia (Limit 2640 psia)
  - Peak radial enthalpy = 65 cal/g (Limit = 100 cal/g, RG 1.236)
  - PCMI failure threshold limit = 21 cal/g (Limit = 33 cal/g, RG 1.236)
  - Peak fuel temperature 2417 °F (Limit = 4791 °F)

NuScale SDAA FSAR Chapter 15 Review

Section 15.4.8 –Rod Ejection (Cont.)

- Implementation of Rod Ejection Methodology TR-0716-50350-P, Rev. 3
  - New peak radial enthalpy & PCMI failure thresholds per RG 1.236
  - All Limitations and Conditions are met
    - Demonstrate the applicability of the rod ejection methodology to the specific NPM design. **NPM-20 was used in TR development**
    - The rod ejection methodology is limited to evaluation of rod ejection accidents for fuel that has not experienced significant depletion with control rods inserted, such as from non-baseload operation. **SDAA only addresses baseload operation**
    - Rod ejection methodology must use TR-0616-48793-P-A, Revision 1, “Nuclear Analysis Codes and Methods Qualification,” and TR-108601-P-A, Revision 3, “Statistical Subchannel Analysis Methodology”. **These codes and methods are used in NPM-20 analyses**

NuScale SDAA FSAR Chapter 15 Review

Section 15.4.6 – Boron Dilution

- Evaluates remaining shutdown margin before automatic isolation of dilution source
- Considers Modes 1 through 5, HZP to HFP
- During the review, earlier calculations credited operator action to secure the dilution source for Modes 1 and 5
  - Staff issued questions to NuScale on crediting of operator actions for boron dilution and other events
  - NuScale revised necessary calculations to ensure operator actions were not credited

NuScale SDAA FSAR Chapter 15 Review

Section 15.4.6 – Boron Dilution (Cont.)

- Mode 1 analysis response dependent on time-in-cycle
  - BOC: faster response, higher initial boron concentration, smaller MTC
- Mode 1 "EOC" uses alternate method:
  - Isolation based on high pressurizer level
  - Automatic letdown prohibited when DWS unisolated
  - Assumes high initial boron concentration (bounds later times-in-cycle)
- Results:
  - 47 pcm SDM remaining at DWS isolation
  - No operator action required to terminate the dilution

NuScale SDAA FSAR Chapter 15 Review

Cooldown & Reactivity Events

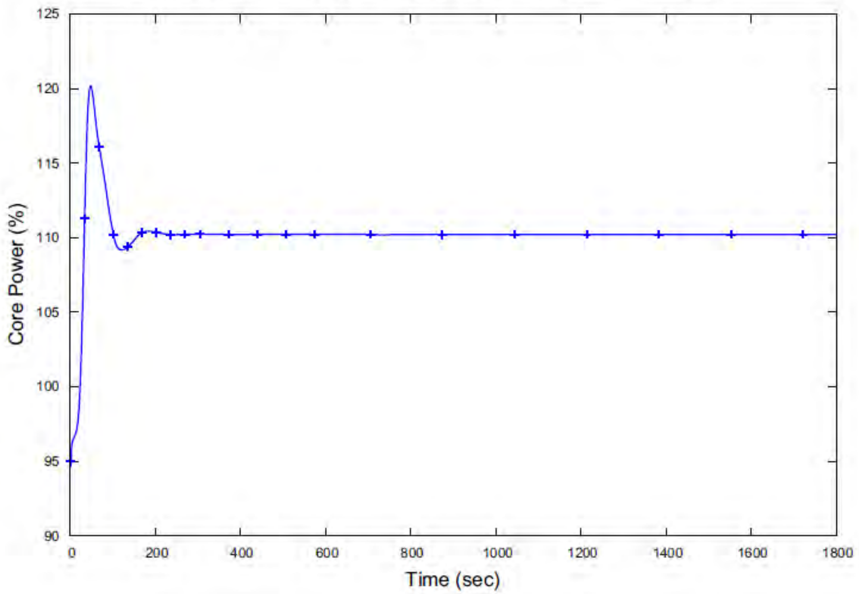
- 15.4.3: CRA misalignment, single CRA withdrawal, CRA drop (bank and single)
- Staff audited NuScale's detailed calculations and confirmed the non-LOCA EM TR was followed
- Limiting cases:
  - MCHFR: 1.71 (Limit 1.43) – Static CRA Misalignment
    - 102% RTP
    - One regulating CRA inserted to the 20% PDIL + 6 steps of rod position uncertainty, other CRAs fully withdrawn
  - LHGR: 14.0 kW/ft (Limit 15.0 kW/ft) - Single CRA Withdrawal
    - Initial power level: 45% RTP
    - Reactivity insertion rate: 0.0101 $/s
    - Reactor trip, SSI, and DHRS actuation on high PZR pressure

NuScale SDAA FSAR Chapter 15 Review

Cooldown & Reactivity Events (Cont.)

- 15.1.3: Instantaneous opening of TBV
- Staff audited the applicant's detailed calculations and confirmed they followed the Non-LOCA EM TR
- Analysis Results
  - MCHFR = 1.55 (Limit 1.43)
    No trip in limiting case
- Key Assumptions:
  - EDAS is relied on to remain functional during cooldown & reactivity events
  - EDAS loss during the event would cause blowdown from higher power, pressure, and temperature

Figure 15.1-23: Reactor Power (15.1.3 Increase in Steam Flow)



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Protecting People and the Environment

NuScale SDAA FSAR Chapter 15 Review

Cooldown & Reactivity Events - EDAS HITI

- Due to removal of the IAB valves from the RVVs, EDAS is now directly supporting the ECCS valve function to remain closed when a valid actuation signal is not present
- This raised concerns regarding the design and safety classification of the system resulting in the identification of a High Impact Technical Issue
- Based on its review of the FSAR and audited documentation the staff determined EDAS is relied on in the safety analysis to perform, at a minimum, the following safety functions:
  - Relied on to assure the integrity of the reactor coolant pressure boundary during power operation
  - Relied on to ensure the SAFDLs are not exceeded during certain AOOs
- EDAS has augmented quality and was evaluated in Chapter 8 of the SER
- A staff differing view raised during the review will be discussed in the following slides
- A staff-initiated exemption to safety-related requirements in Chapter 8 is a potential option under consideration to address the differing view

NuScale SDAA FSAR Chapter 15 Review

EDAS HITI – Staff Differing Opinion

- On December 13, 2024, the following staff submitted a non-concurrence on the NuScale SDAA Chapter 15 safety evaluation report:
  - Antonio Barrett, Senior Nuclear Engineer
  - Craig Harbuck, Senior Safety and Plant Systems Engineer
  - John Lehning, Senior Nuclear Engineer
  - Zhian Li, Senior Nuclear Engineer
  - Joshua Miller, Nuclear Engineer
  - Ryan Nolan, Senior Nuclear Engineer
  - Rebecca Patton, Branch Chief
  - Marie Pohida, Senior Reliability and Risk Analyst
  - Adam Rau, Nuclear Engineer
  - Sheila Ray, Senior Electrical Engineer
  - Thomas Scarbrough, Senior Mechanical Engineer
- Staff raised concerns regarding insufficient technical or regulatory basis for the acceptability of the EDAS classification and regulatory controls

NuScale SDAA FSAR Chapter 15 Review

EDAS HITI – Staff Differing Opinion (Cont.)

- The specific issues raised include:
  - EDAS meets the definition of a safety-related structure, system, or component prescribed in 10 CFR 50.2
  - EDAS meets 10 CFR 50.36 criteria for establishing limiting conditions for operation in the technical specifications
  - Management decision made early in the SDAA review on the acceptability of EDAS did not provide defensible technical or regulatory bases, and was not conducted in accordance with applicable policies, procedures, and regulations
- The differing view also provided acceptable risk-informed approaches to resolve the concerns, including:
  - Use of regulatory exemptions to applicable requirements and application of the RTNSS process
  - Use of the risk-informed classification process provided in 10 CFR 50.69

NuScale SDAA FSAR Chapter 15 Review

EDAS HITI – Staff Differing Opinion – Path Forward

- As an outcome of the differing views process, NRR management is evaluating whether an exemption is needed to treat EDAS as non-safety-related
  - Information pertaining to the EDAS design and its reliability and availability controls would be sufficient to support the exemptions
  - Classifying EDAS as safety-related is not necessary for adequate protection
- A staff-initiated exemption could be documented in SER Chapter 8
  - Exemption from safety-related requirements of 10 CFR 50.55a(h)
  - Exemption from safety-related requirements of 10 CFR 50 Appendix B, Criterion III through XVIII
  - This approach would clarify that EDAS is exempted from safety-related classification and is therefore non-safety-related

NuScale SDAA FSAR Chapter 15 Review

Section 15.2.8 - Feedwater System Pipe Breaks Inside and Outside Containment

- Most limiting case in group 2: Decrease in Heat Removal by the Secondary System
- Analysis Results
  - MCHFR = 2.4
  - Maximum RCS pressure = 2,316 psia
  - Maximum peak secondary pressure = 1,446 psia
- Key Assumptions:
  - Initial power level is assumed to be 102% of nominal to account for measurement uncertainty
  - Conservative reactor trip characteristics: maximum time delay, holding the most reactive rod out of the core, and bounding control rod drop rate
  - Limiting BOC reactivity feedback for limiting power response analyses
  - AC power lost at the time of the break, immediate turbine and FW pump trip

NuScale SDAA FSAR Chapter 15 Review

Section 15.2.8 - Feedwater System Pipe Breaks Inside and Outside Containment (Cont.)

- Key Assumptions (cont.):
  - FWIV is assumed to fail close on the faulted FW line
  - SSI valves are assumed to close and DHRS valves are assumed to open at their maximum times
  - System biases: high RCS temperature, high fuel temperature, low PZR pressure, low PZR level, minimum RCS flow
  - Limiting cases: double ended guillotine break:
    - RCS pressure case: FW line inside containment
    - Peak SG pressure case: FW line inside containment
    - MCHFR case: FW line outside containment
    - DHRS cooling case: FW line inside containment

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Section 15.6.3 – Steam Generator Tube Failure

- 15.6.3 - Analysis Results
  - MCHFR is not limiting for SGTF (screened out)
  - Limiting RPV pressure scenario: 20% partial tube failure at top of SG with coincident loss of normal AC power
  - Limiting SG pressure scenario: 100% split break tube failure at top of SG with loss of normal AC power
  - Maximum radiological consequences confirmed to be bounded by FSAR 15.0.3 assumption
- Key Assumptions:
  - Core power at 102%; highest worth rod stuck out
  - Assuming no single failure is conservative
  - Tube failure at the top of the SG results in higher RCS and SG pressure

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Section 15.6.6 – Inadvertent Operation of a Reactor Valve

- There are few valves in the design, and the ECCS valves are the ones that IO that cause the biggest challenge to FoMs
  - This means the limiting IORV event is an inadvertent ECCS operation
- A loss of dc power to MPS causes both RVVs, which do not have IABs, to open without delay
  - This means results will be insensitive to ECCS actuation signal timing
  - Note that ECCS valves now have venturi internal to the valve body

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<u>Section 15.6.6 – Inadvertent Operation of a Reactor Valve (Cont.)</u>

- IORVs are MCHFR-challenge events
  - LOCA EM has special sub-methodology for "phase 0" MCHFR analysis
    - Hot assembly inlet flow blockage
    - 102% initial thermal power
    - Distributed primary loop losses
    - Special, new NSPN-1 CHF correlation

Non-Proprietary

NuScale SDAA FSAR Chapter 15 Review

Section 15.6.6 – Inadvertent Operation of a Reactor Valve (Cont.)

- The worst IORV event is found to be an IO of one RRV with a loss of ac and EDAS dc power
  - IABs on the RRVs make inadvertent opening of more than one RRV improbable
  - Using the LOCA LTR methodology for "phase 0"
    - The limiting MCHR is 1.41
      - This is **not** the limiting Chapter 15 MCHFR (unlike for US600)
    - Acceptance criterion for MCHFR is 1.2 or greater for NSPN-1
- The IORV events are also not the design's limiting transients for:
  - Containment response
  - RCS pressure
  - Steam generator pressure
  - CLL (is about 10' for this worst IORV event)
- DHRS is not a factor *in the limiting IORV event*

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Section 15.6.5 – Loss of Coolant Accidents

- LOCA for the NPM-20 design characterized by:
  - Small break sizes <2”, and limited RCS pipe break locations
  - ECCS actuation logic changes – triggered by riser level
  - Credit DHRS during LOCA for passive cooling of the RCS (important for SBLOCAs)
- LOCA scenario and limiting analysis results:
  - Limiting case: 100% CVCS discharge line break w/o AC/DC
  - MCHFR > 1.35; CLL > 9.7” above TAF
- Staff performed confirmatory analysis using TRACE
  - NRELAP5’s LOCA FoMs are more conservative

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Section 15.6.5 – Use of Thermal Dispersion Switch for ECCS Actuation

- LOCA TR SE L/C - ECCS RPV Riser Level Instrument Setpoint Modeling
  - Method follows LOCA EM TR modeling setpoint based on mixture level
- Level Detection by heat transfer differences between liquid and vapor phase
- ECCS Actuation Trip Implementation
  - Low level signal trigger: 90% void near the riser outlet (CLL 540-552")
  - Low-low level signal trigger: 95% void (CLL 460-472")
- ECCS Timing Evaluation
  - LOCA not sensitive to ECCS actuation timing delay
- Staff’s Finding
  - the level sensor responses corresponding to the specific setpoints and analytical limits results in acceptable collapsed water level above the core

NuScale SDAA FSAR Chapter 15 Review

Section 15.6.5 – LOCA Break Spectrum Exemption

- Staff determined certain locations are subject to the requirements of 10 CFR 50.46 and GDC 35 and were not considered within the design-basis LOCA break spectrum. This resulted in two High Impact Technical Issues:
  - HITI #2: ECCS Valve Flanges
    - Applying the LOCA EM at these locations result in more severe consequences than IORV events
  - HITI #10: CVCS piping systems between the CNV and CIVs
    - Breaks at these locations result in the loss of coolant outside the CNV with more severe consequences than LOCAs analyzed inside containment
- Staff was open and supportive of a risk-informed alternative approach for the analyses of losses of coolant from these locations
- NuScale submitted an exemption request, with supporting analysis, to treat these locations as beyond-design-basis

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Section 15.6.5 – LOCA Break Spectrum Exemption (Cont.)

- Framework used to evaluate a risk-informed exemption to 10 CFR 50.46:
  - The design implements a holistic safety approach that reduces LOCA risk through both prevention and mitigation
    - Reduced penetrations, large volume of water above the core, slower accident progression that provides more time for operators to respond, etc.
  - Enhanced design and operational programs provide assurance that failures at the location of interest are highly unlikely
    - Limits on stresses at the locations beyond those specified in the ASME BPV Code, leakage detection, enhanced inservice inspections, etc.
  - Realistic, best-estimate analyses of LOCAs at the location of interest as beyond-design-basis accidents demonstrate that the consequences are acceptable
    - Analysis demonstrates the core remains cooled, consideration of uncertainties to avoid cliff edge effects

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Section 15.6.5 – LOCA Break Spectrum Exemption- Cont.

- NuScale Analysis and Acceptance Criteria:
  - Developed acceptance criteria for core cooling, containment, and radiological figures of merit
  - Thermal-hydraulic analysis was performed using the LOCA EM with modification to represent best-estimate initial conditions.
  - Demonstrates the results meet the acceptance criteria
- Staff Review:
  - Audited NuScale calculations to understand modifications to the LOCA EM and verified the results
  - Performed independent confirmatory and sensitivity analyses to confirm NuScale's assumptions and inputs do not result in cliff edge effects
  - Concludes the analysis is acceptable for a BDB event and supports the exemption to 10 CFR 50.46 and GDC 35

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Conclusion

- ❖ While there are some differences between the DCA and the SDAA, the staff found that the applicant provided sufficient information to support the staff's safety finding.
- ❖ The staff found that all applicable regulatory requirements were adequately addressed.
- ❖ Staff does not expect the decision on the EDAS exemption to change the analysis or design. As an outcome of the NCP review, the staff will modify the relevant SERs to clarify the regulatory basis and document the justification that EDAS is non-safety related.

Meeting Title	**April 1, 2025 Open Session Staff's Evaluation of NuScale SDAA Chapters 1, 4 and 15**

Attendees

Michael Snodderly	ACRS DFO
Shandeth Walton	ACRS
Larry Burkhart	ACRS
Thomas Dashiell	ACRS
Jon Bristol	NuScale
Tyler Beck	NuScale
Stacy Joseph	NRR
James Cordes	Court Reporter
Wendy Reid	NuScale
Andrew Dyszel	Contractor
Rose Charoensombud	NuScale
Getachew Tesfaye	NRR
Greg Halnon	ACRS
Carrie Fosaaen	NuScale
Ken Rooks	NuScale
Brian Wolf	NuScale
Leonard Ward	Contractor
Dennis Bley	ACRS
Thomas Scarbrough	NRR
Sheila Ray	
Ron Ballinger	ACRS
Stephen Schultz	ACRS
Jeff Luitjens	NuScale
Ron Ellman	
Glen Thomas	Framatome
Cindy Williams	NuScale
John Budzynski	NRR
Stewart Bailey	NRR
Matt Sunseri	ACRS
Marissa Bailey	ACRS
Tim Polich	RoPower
Jim Osborn	NuScale
Ryan Nolan	NRR
Dominik Muszynski	
Adam Rau	NRR
Kenny Anderson	NuScale
Angie Buford	
Vesna Dimitrijevic	ACRS
Angelo Stubbs	NRR
Sarah Bristol	NuScale
Rob Morrow	NuScale
Sean Park	NuScale
Stephen P O'Hearn	

Taylor Coddington	NuScale
Dan Lassiter	NuScale
Erin Whiting	NuScale
Dominik Muszynski	
Mahmoud -MJ- Jardaneh	NRR
David Benson	NuScale
Mark Shaver	NuScale
Robert Martin	ACRS
Freeda Ahmed	NuScale
Etienne Mullin	NuScale
Lisa Helfer	
Carlen Donahue	NuScale
Don Marksberry	RES
Amanda Bode	NuScale
Gene Eckholt	NuScale
Kyle Hoover	NuScale
William Deric Tilson	
Eric Baker	NuScale
Alissa Neuhausen	NRR
Paul Guinn	NuScale
R Snuggerud	NuScale
Samuel Lee	NRR
Caty Nolan	COMM
J.J. Arthur	NuScale
Thomas Hayden	NRR
Justin Mechling	
Wendell Morton	
River Rohrman	NRR
Matthew Mitchell	NRR
Alina Schiller	NRR
Elisa Fairbanks	NuScale
Chelsea Lockwood	NuScale
Ricky Vivanco	NRR
Craig Harbuck	NRR
Steven Pope	
Steven Alferink	NRR
Wren Fowler	NuScale
Carl Fisher	
Deric Tilson	
Michael Valleau	NuScale
Meghan McClosky	NuScale
Allyson Callaway	NuScale
Angi Cordillo	NuScale
Sarah Turmero	NuScale
Kris Cummings	NuScale
Rebecca Patton	NRR
Tom Griffith	NuScale
Ben Bristol	NuScale
Kevin Lynn	NuScale

Getachew Tesfaye	NRR
Nick Klymashyn	PNNL
Kevin Kadooka	PNNL
Zhian Li	NRR
Warren Erling	NRR
Sean Piela	NRR
John Honcharik	NRR
Josh Miller	NRR
Shanlai Lu	NRR
Rosemary Sugrue	NRR
Joshua Kaizer	NRR
Andrew Bielen	RES
Michele Sampson	NRR
Gary Becker	NuScale
Dong Zheng	NRR
Carl Thurston	NRR
C. Basavaraju	NRR
Edward Stutzcage	NRR
William Fork	Pillsbury Winthorp
Justin Coury	RES