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

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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NuSCALE SUBCOMMITTEE

+ + + + +

THURSDAY

AUGUST 22, 2024

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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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ACRS CONSULTANTS:

CHARLES BROWN

DENNIS BLEY

MYRON HECHT

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

MIKE SNODDERLY

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 Committee of the Advisory Committee on Reactor Safeguards.

I am Walt Kirchner, Chair of today's Subcommittee meeting. ACRS members in attendance in person are Ron Ballinger, Vicki Bier, Robert Martin, and Craig Harrington. ACRS members in attendance virtually via Teams are myself, Matt Sunseri, Vesna Dimitrijevic, Greg Halnon, and Scott Palmtag. We also have our consultants participating: Dennis Bley, Charlie Brown, Myron Hecht, and Steve Schultz.

If I've missed anyone, please speak up. I think that is the roster for today and attendance.

Michael Snodderly of the ACRS staff --

MEMBER ROBERTS: Hey, Walt, this is Tom Roberts. I'm on.

CHAIR KIRCHNER: Okay. My oversight, Tom. Thank you.

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

No member conflicts of interest were identified for today's meeting.

During today's meeting, the Subcommittee will receive a briefing on the staff's evaluation of NuScale Power LLC's US460 Standard Design Approval Application, Chapters 7, Instrumentation and Controls; Chapter 9, Auxiliary Systems; Chapter 12, Radiation Protection, and Chapter 18, Human Factors Engineering.

We will also be briefed on the status of 10 high-impact technical issues by the NuScale staff.

We previously reviewed the Certified NuScale US600 design, as documented in our July 29, 2020 Letter Report on the safety aspects of the NuScale small modular reactor.

Like the staff, we are performing a delta review between the two designs, including a power uprate from 50 to 77 megawatts electric per module. We are reviewing these chapters as part of our statutory obligation under Title 10 of the Code of Federal Regulations, Part 52, Subpart (e), Section 141, Referral to the Advisory Committee on Reactor Safeguards, and to report on those portions of the application which concern safety.

The Subcommittee will hear presentations by and hold discussions with the NRC staff and NuScale regarding these matters.

A portion of the presentations by the

Applicant and the NRC staff may be closed to discuss information that is proprietary to the licensee and its contractors, pursuant to 5 U.S. Code 552b(c)(4).

Attendance at the meeting that deals with such information will be limited to NRC staff and its consultants, NuScale, and those individuals and organizations who have entered into an appropriate confidentiality agreement with them. Consequently, we will confirm that we have only eligible observers and participants in the closed portion of the meeting.

The ACRS was established by statute and is governed by the Federal Advisory Committee Act, FACA. The NRC implements FACA in accordance with its regulations found in Title 10 of the Code of Federal Regulations, Part 7.

Per these regulations and the Committee's Bylaws, the ACRS speaks only through its published Letter Reports. We hold Subcommittee meetings to gather information and perform proprietary work that will support our deliberations and final decisions of whether to issue a Letter Report at a full Committee meeting. All member comments should be regarded, therefore, as the individual opinion of the member only, not a Committee position.

The rules for participation in all ACRS

meetings, including today's, were announced in a June 13th, 2019, Federal Register Notice.

The ACRS section of the U.S. NRC public website provides our Charter, Bylaws, member guidance, subcommittee structure, agendas, Letter Reports, and full transcripts of all full and subcommittee meetings, includes slides presented there.

The meeting notice and agenda for this meeting were posted there and can be easily found by typing about us ACRS in the Search field in the upper right corner of the website.

The ACRS, consistent with the agency's value of public transparency in regulation of nuclear facilities, provides opportunity for public input and comment during its proceedings. We have received no written statements or requests to make an oral statement from the public today, but we have set aside time in the agenda at the end of this meeting for any comments from members of the public listening into this meeting. The Subcommittee will consider all public comments, as appropriate.

The Subcommittee will gather information; analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

A transcript of the meeting is being kept and will be made available.

Today's meeting is being held in person and over Microsoft Teams for the ACRS staff, Applicant, and members of the public. The Teams link information with a telephone bridge line was placed in the agenda on the ACRS public 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. When not speaking, we request that participants mute your computer microphone on Teams, or phone, if you are on the bridge line, by pressing star-6.

Please do not use any virtual meeting chat features to conduct sidebar discussions related to the presentations. Rather, limit the use of the meeting chat function to report IT problems, such as inability to hear speakers or see presentations.

Also, for everyone in the room, please put all your electronic devices in silent mode, including muting your speakers and microphone on your laptops. In addition, please keep sidebar discussions in the room to a minimum, since the microphones in the ceiling are live for the course of the meeting.

Finally, for the presenters, your microphone at the tables, those of you there at headquarters, are unidirectional. So, you'll have to speak into the front of the microphone in order to be heard online.

And before we proceed with the meeting, I just want to reiterate Michael Snodderly's thanks to all participants. We are spanning four time zones.

A special shout-out to our NuScale colleagues. It's still dark here in Santa Fe, New Mexico, and I think it's probably darker in Corvallis at this moment. So, thank you very much for flexing with us to meet our schedule requirements.

And with that, I'm going to turn to a representative of the staff for opening comments.

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

I am MJ Jardaneh. I serve as the Branch Chief of the New Reactor Licensing Branch responsible for licensing the NuScale Units for design, the Division of New and Renewed Licenses in NRR.

Thank you for the opportunity today for the staff to present their reviews on select NuScale

US460 Standard Design Approval Application on Chapters 7, 9, 12, and 18.

The staff is reviewing of all the chapters of the SDAA concurrently. The staggered completion dates are based on the complexity of the chapter and the extend of change from the Certified NuScale US600 design.

Today, the staff will be presenting on their review of the second group of SDAA chapters, including Chapters 7, 9, 12, and 18.

The remaining chapters of the SDAA are still being reviewed by the staff and we will inform the ACRS on the Safety Evaluations when the remaining chapters are available to the ACRS.

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

Getachew Tesfaye, the lead Project Manager for the NuScale SDAA, will provide us with update about the project and walk us through the logistics of the review.

Once again, thank you for the opportunity and we look forward to a good discussion today.

MR. TESFAYE: Thank you, MJ.

Good morning, Chair Kirchner, ACRS Subcommittee members, and everyone that's participating in today's meeting.

My name is Getachew Tesfaye. As my Branch Chief MJ indicated, I'm the lead Project Manager for NuScale's Standard Design Approval Application.

In the way of background for today's meeting, NuScale completed the submittal of its Standard Design Approval Application for US460, small module reactor, that began in November 2022 to December 31, 2022.

NuScale submitted the SD application pursuant to the requirements of Title 10 of the Code of Federal Regulations, Part 52, Subpart (e). The application was formally accepted for an NRC review on July 31st, 2023. Following NuScale's submittal of supplemental information needed for docketing, the application was accepted and we started the review.

On March 29 of 2024, we presented Chapters 2, 10, 11, 17, not including Section 17.4. The full Committee deliberated on these four chapters in May.

About a month ago, we shared with the Committee the final Draft Safety Evaluations for Chapters 7, 9, and 12 that were still under management review and the completed Advanced SE for Chapter 18.

The three Draft SEs are still under legal and management review and were officially submitted and made public to support today's Subcommittee meeting. When management review and legal review of these chapters are completed, we will share the final versions with the Committee, highlighting the changes made to the drafts that are presented today.

We do not expect any material changes to the technical content of the Draft SEs as a result of review.

We do appreciate the ACRS's flexibility for allowing us to present them at this time to maintain our current schedule.

And I thank you again.

CHAIR KIRCHNER: Thank you.

Let me just reiterate, as we proceed, I'll ask each speaker to clearly identify yourself as you start your presentation, both since we have a large contingent listening in today and we're in disparate locations. So, just clearly identify yourself and affiliation as we start each presentation. That also benefits the court reporter, who is doing the transcription of today's meeting.

With that, I believe we're ready to start with Chapter 7 with NuScale. So, I'll turn to NuScale to begin.

MR. GRIFFITH: Good morning.

Thomas Griffith, Licensing Manager at NuScale for the US460 Standard Design Approval Application.

Background about myself: I have 15 years' experience in the nuclear industry; worked in a variety of roles, including safety analysis, reactor engineering, in maintenance as an I&C shop manager, and was a formerly licensed operator at Dresden Units 2 and 3.

NuScale is pleased to be presenting these chapters to the ACRS today, and I would like to extend my thanks to both the NRC staff for their thorough review and to the NuScale staff for their efforts in getting us to where we are today. There has been a substantial effort by everyone involved to get to the point we are, and I am thankful for all of that effort.

At this point, I'll turn it over to Tom Case to present Chapter 7.

MR. CASE: Good morning.

My name is Tom Case. I'm a Licensing Engineering with NuScale. I've been with NuScale for about two years and I've been in the nuclear industry for about 14 years, and I'm a Licensed Professional Engineer.

Before discussing Chapter 7, this slide acknowledges that NuScale is the recipient of financial assistance awards from the U.S. Department of Energy and is obliged to identify their support.

Next slide. I'll be presenting Chapter 7 of the final Safety Analysis Report from the NuScale US460 Standard Design Approval Application. The scope of this chapter covers safety-related I&C systems, which are the module protection system and the neutron monitoring system.

It also includes the non-safety-related, non-risk-significant I&C systems that perform specific regulatory required functions. These systems include the module control system, plant control system, plant protection system, safety display and indication system, in-core instrumentation system, and radiation monitoring system.

Next slide. Chapter 7 is divided into the three sections shown here. For each section, I'll be highlighting changes from the NuScale's US600 Design Certification and, also, covering the results of the NRC audit and review of this section.

Next slide. In Section 7.0, the remote

shutdown station is removed because the US460 design does not have a dedicated remote shutdown station. Instead, there are alternate operator workstations that allow for plant monitoring outside the main control room. And there are no audit items or RAIs specific to Section 7.0.

As denoted here, there is one audit item, Chapter 15 review, that is related to a combined license item, this section. And resolution of that audit item will be through the Chapter 15 audit and RAI process.

Next slide.

MEMBER HALNON: This is Greg Halnon. Before you go on, could you just confirm that your operating staffing connects up with this change, so that you have enough operators on shift to be able to do the monitoring you're expecting to be able to do?

MR. CASE: Yes. Can I just clarify? Is that with regards to the change listed here with the remote shutdown station?

MEMBER HALNON: Correct.

MR. CASE: Yes. So, basically, there's alternate operator workstations outside of the main control room. And in the event of a main control room evacuation, the operators have the ability to monitor the plant from outside the main control room in a safe shutdown condition.

As far as staffing goes, I'd like to introduce Doug Bowman to help with the staffing part of that question.

MR. BOWMAN: So, this is Doug Bowman, Plant Operations Manager from NuScale.

So, a staffing question, of course, we answered that with our revised staffing plan, Revised Staffing Validation that we performed that was done for topical report that was presented to the ACRS and already has an SAR on it. And that included, the testing included the remote shutdown station. So, there's adequate staffing to do remote shutdown and monitoring.

MEMBER HALNON: Okay. Thanks. Thanks.

Yes, Doug, just one quick follow-up. Did you guys also plan on installing appropriate communications, hard-wire communications between those stations or are you relying on repeaters, operator, I mean radio operating? Or what kind of communications are you looking at?

MR. BOWMAN: We have specified the same level of communication that we get in the control room for those alternate operator workstations.

MEMBER HALNON: What does that mean?

MR. BOWMAN: It means they get, you get, it gets exactly the same communication system that you would have at the main control room.

MEMBER HALNON: Okay. So, each of the alternate operator workstations will have some ability to communicate with the other operating stations and some central SRO who is still in charge of the plant?

MR. BOWMAN: Right. The SRO, the command function would move to the alternate operator workstation. So, there's not going to be another location.

Really, in terms of remote shutdown, there's really not much difference from the DCA to the SDA. We really didn't change the procedure very much The operators take the same actions before they leave. And they have the same actions they can implement afterwards, if they need to, as a compensatory measure. And really, the only thing that's changed is we not only have a dedicated remote shutdown station, we now have non-dedicated remote shutdown stations that we can implement, when needed, in a control room evacuation scenario.

MEMBER HALNON: Okay. Well, and that's the point. Rather than being located in a single area, you're going to be distributed around the plant, which causes a little bit more communication complications, coordination, and whatnot. And that's what I was just exploring, is if all of that was taken into consideration with hardware, as necessary.

MR. BOWMAN: I understand now. I'll try to get into a little more detail here and try to answer that.

There's two designated alternate operator workstations. One of them is with the module maintenance center, which is inside of the reactor building. One is at the rad waste control room, which is in the rad waste building.

We wouldn't implement both of those. The operators would all go to a single alternate operator workstations. So, it's just a matter of having choices just in case. You know, we could use this alternate operator workstations in a wide range of events -- from the loss of a large area to just a control room evacuation event. And depending on those conditions, you might pick one or the other, but they're not going to be at both during this event. They'll be at a single alternate operator workstations.

MEMBER HALNON: Okay. I get it.

MR. BOWMAN: Okay.

MEMBER HALNON: Yes, thank you.

MR. BOWMAN: Thank you.

MEMBER SUNSERI: This is Matt Sunseri. I had a follow-up question.

CHAIR KIRCHNER: Yes, go ahead, Matt.

MEMBER SUNSERI: So, regarding this control room evacuation, if the control room is evacuated in the event of a fire, please remind me if there was a way to lock out the controls in the control room, so that you don't get spurious actuations from the controls that are there.

MR. BOWMAN: Yes.

MEMBER SUNSERI: Is that the case?

MR. BOWMAN: Yes, there are.

MEMBER SUNSERI: Okay. Thank you.

MR. BROWN: This is Charlie Brown. Can I make a comment? Is it the appropriate time? Walt, is that all right?

CHAIR KIRCHNER: Yes, go ahead, Charlie.

MR. BROWN: Okay. On the overall architecture drawing, it still shows the remote shutdown station, and I presume that's the station you're referring to that's not going to be in place anymore. Is that correct?

MR. CASE: Can you clarify which figure is showing that?

MR. BROWN: It's 7.0-1, or something like that. I forgot. Hold it. I'll have it for you here in a second. I've got to scroll down to the bottom of your page. It's on page 7.0-36 and the figure is, let's see, 7.0-1.

MR. CASE: Yes, I believe that is not the current revision that we're presenting today. So, we're presenting Revision 1 of the US460 Standard Design Approval Application. And I'm looking at figure 7.0-1, which is on page 7.0-32 and it shows alternate operator workstations.

MR. BROWN: Okay.

MR. CASE: And it does not --

MR. BROWN: Okay. All right. I accept that. It's just the one I was able to get when we had the information.

But the question I have relative to that is, there were manual safety shutdown switches, so you could scram the plant and initiate safeguards, anything you wanted to do that you needed to operate, particularly tripping the plant. Do we still have manual control? Or now, you're saying it's all workstations? So, all remote actuation of any function would still be done software-based as opposed to any manual shutdown?

MR. BOWMAN: So, the remote shutdown station in the DCA did not have manual actuations for the safety functions. That was all done in the control room or had to be done locally at the MPS rooms. That's the only place they've ever had those safety-related manual actuations with the new design. So, that shutdown station in the DCA was nothing more than a monitoring, a place for the operators to go and monitor.

MR. BROWN: Yes, I'm just saying there was a Note 9 in the other ones that said you had these trip switches. So, I'm just trying to clarify, do we have the ability to scram the plant outside the main control room without going down to a switchboard and tripping all the circuit breakers?

MR. BOWMAN: We have the ability to locally -- we have the ability to trip the plant from inside the main control room and we have the ability to locally trip the plant and place it in safe shutdown from the MPS room.

MR. BROWN: Without software? It's a manual switch?

MR. BOWMAN: Correct.

MR. BROWN: But if the main control room is evacuated due to fire -- relating off to the previous comment -- is that still -- in other words, you're relying on somebody to not haul whatever they want to haul to get it out of the main control room to scram plant before they leave? Is that the point?

MR. BOWMAN: The operators will scram the plant prior to leaving the main control room.

MR. BROWN: Before they evacuate?

MR. BOWMAN: Sure.

MR. BROWN: Okay. All right. Thank you.

MR. SNODDERLY: Excuse me, Charlie. This is Mike Snodderly.

MR. BROWN: Yes?

MR. SNODDERLY: This is Mike Snodderly on the ACRS staff.

So, Charlie, in the ACRS SharePoint folder for Chapter 7 for this meeting --

MR. BROWN: Yes?

MR. SNODDERLY: -- that's Rev 1 of the SDAA, is in there, if you --

MR. BROWN: Yes, mine said, Rev 1 on it. So, I'm still a little puzzled. It's okay. I got my answer.

MR. SNODDERLY: Okay.

MR. BROWN: My version says, Rev 1, but it's got the little box up in the upper left-hand corner.

MR. SNODDERLY: So, if I could ask you, go to the SharePoint site for this meeting under Chapter 7 and --

MR. BROWN: That's where I got this.

MR. SNODDERLY: Okay. All right.

MR. BROWN: But don't worry about. I understand what you're saying and it's not a problem. Okay?

MR. SNODDERLY: All right. Thank you, Charlie.

MR. BROWN: Yes.

MR. SNODDERLY: All right. Bye.

MR. CASE: Next slide, please. In Section 7.1, MPS setpoints are changed as a result of the changes in operating pressure and temperature. In the updated safety analysis for the US460 design, some of the ECCS, DHRS, and RTS actuations are changed as a result of the updated Safety Analysis.

A timer is added to automatically initiate ECCS eight hours after a reactor trip to add supplemental boron, if needed to maintain subcriticality during long-term cooling.

And Type F post-accident monitoring variables are added as a result of adoption of an updated regulatory guide. Type F PAM variables monitor for fuel damage.

There was one audit item resolved in this section, and that audit item removed a note from a figure to clarify that the inadvertent actuation block is not applicable to the reactor bin valves.

Next slide. For Section 7.2, information from the Sensor Technical Report cited in the DCA is incorporated into this section.

Certain level and pressure sensors, shown here, are changed from digital to analog, based on additional design development of the US460 sensor design. And the quantity of reactor coolant system temperature sensors is reduced in each quadrant based on engineering evaluation that shows streaming effects do not require the use of multiple sensors per quadrant.

There are no audit items or RAIs specific to Section 7.2.

Next slide.

MEMBER ROBERTS: Yes, this is Tom Roberts. If I can go back to that previous slide, the second subset bullet says the D3 analysis was updated in 7.1

-- presumably to reflect the conversion from the digital to analog sensors for pressure.

Yes, my question is, the analysis change appeared to take away evaluation of the pressure sensor as a potential common cause failure, presumably because it's an analog sensor. I guess the question, is that right? And then, I was wondering why that would be valid because there still could be a common cause failure of the sensor because it's miscalibrated; there's a common cause, or something like that, that would still result in a loss of the pressure function due to a common cause.

MR. CASE: Yes, so that is correct. The diversity and defense-in-depth analysis and coping analyses were updated due to the change from digital to analog pressure sensors. And within that D3 analysis, the analysis looks for digital-based common cause failures. And because the pressure sensors in the US460 design were changed from digital to analog, they're no longer susceptible to digital-based common cause failures. Therefore, the coping analysis does not evaluate digital-based common cause failures of those pressure sensors.

With respect to, I guess what I would call, an analog-based common cause failure, that would be evaluated in other analyses, such as the failure modes' effects and effects analysis and the hazard analysis performed on the overall system.

So, I think it's appropriately excluded from digital-based common cause failure analysis and the D3 coping analyses, but failure of those sensors is addressed elsewhere in other analyses in Chapter 7.

MEMBER ROBERTS: Okay. Thanks.

You're about done with this chapter. I have what I call a clarifying question, and it kind of relates to the question I just asked, which has to do with ATWS. Give me a second here to get to this document. Okay.

So, the SAR Section 7.1.1, which is design basis, has not changed. But it says, The design meets the intent of 10 CFR 50.62 by demonstrating the redundancy and diversity of the MPS design, which avoids common cause failures and reduces the probability of a failure to scram. And then, it references Section 15.8. 15.8 says the same thing in three paragraphs. So, it doesn't really add to that.

You issued a Technical Report back in 2013 on ATWS. It was entitled, NuScale Power Plant Designs for ATWS and 10 CFR 50.62 Regulatory Compliance, and Technical Report 2196 was the number.

And it had two reasons or two rationales for the ATWS approach. One of them was the diversity and defense-in-depth in the MPS. The other one was preliminary RELAP calculations. It showed that the NuScale would meet ATWS success criteria, even if the event happened. And it quoted that the consequence would still maintain equivocal geometry; maintain radical and pressure boundary integrity; maintain containment integrity. And your SAR Chapter 19 has a similar discussion in 19.2.2.

And so, that leaves me a little bit confused as to what the technical basis is for the ATWS approach. The design of diversity and defense-in-depth in the MPS appears to be based on an assessment of digital common cause failures, which the ATWS rule predates digital I&C and was concerned about more generic common cause failures.

And your technical basis does show that the consequence of the event, if it occurred, would not be called catastrophic, which would support that being considered to be sufficient.

So, my question is, if you didn't have that analysis, or if that analysis had come up with a different conclusion in terms of the consequence, would you still consider the diversity and defense-in-depth of the MPS to be sufficient? Or are you really counting, also, on this separate study of the consequence of the ATWS?

MR. CASE: So, I think the justification provided in the SAR is what we're relying on to support our evaluation of 10 CFR 50.62 as it pertains to ATWS. And it specifies demonstrating the redundancy and diversity of the MPS design, which is digital, avoids digital-based common cause failures and it reduces the probability of the failure to scram. So, I think that's the justification we're providing in the application and the associated exemption from certain portions of that regulation with respect to ATWS.

With respect to the history of the additional analysis provided in there, I can't really speculate on how we would present this without that analysis, but what we are presenting here in the application includes the analysis that was done.

MEMBER ROBERTS: Right, and I'll have a similar question for the staff because they cited that analysis in Chapter 19 as part of their basis for accepting at least the exempt for part of the requirements there. Again, I'll ask staff, but it seems like they leverage that work as part of their basis for acceptance. So, it kind of leaves the question.

And we don't need to really ask the question for this plant design because the fact is you have both attributes. And so, if you didn't necessarily credit the diversity and defense-in-depth to the extent that it was initially intended by the ATWS rule, you've got the consequences as bounded. And so, the degree of diversity and defense-in-depth is probably more than sufficient for the ATWS rule, given the consequence of the event.

So, I just wonder -- I'm not concerned about NuScale -- my bigger concern is if there's a subsequent plant design that didn't have your plant characteristics and couldn't make that statement, would we all come out in a different place? And so, I just wanted to ask the question and I think you've answered it. So, thank you.

MR. CASE: Next slide. So, the module protection system is based on the Highly Integrated Protection System Platform described in the topical report shown here. This topical report was approved in June of 2017. There are no changes to the topical report and no changes since ACRS engagement in 2023.

Next slide. The Instrument Setpoint

Methodology Technical Report shown here is reviewed as part of Chapter 7. There have been no changes to the Technical Report since previous ACRS engagement in 2023.

Next slide. With respect to combined license items, or COL items, there are no changes to the COL items in Chapter 7. As previously mentioned, there is an audit item in Chapter 15 related to a Chapter 7 COL item, and that's being resolved through the Chapter 15 audit.

And that concludes my presentation of Chapter 7. I could take any additional questions.

CHAIR KIRCHNER: Well, hearing none at this point, Mike, we will turn to the staff, is that correct?

MR. SNODDERLY: That is correct. So, if staff could come to the --

MR. HECHT: This is Myron Hecht. I was on mute. I'm just wondering if I can ask a question.

CHAIR KIRCHNER: Go ahead, Myron.

MR. HECHT: Okay. All of the communications with both the auxiliary workstation and the primary control room are done using the HIPS communications modules, is that correct?

MR. CASE: So, can I just make sure I

understand the question? It's the communication between alternate operator workstations that we were --

MR. HECHT: Right.

MR. CASE: -- discussing previously?

MR. HECHT: Right.

MR. CASE: Yes. So, that communication is shown in figure 7.0-1 of the overall instrumentation and control system architecture.

The communication flow path is from the plant control system. So, the plant control system feeds plant control system workstations and alternate operator workstations. And the plant control system gets its information through the safety display and indication system hub, which gets its information from the module protection system. So, that's kind of the flow path of information from the module protection system through to the alternate operator workstations.

I will just note that, outside of the module protection system, the safety display and indication system hub, the plant control system, and the alternate operator workstations are all non-safety-related.

MR. HECHT: But they do rely on the HIPS communication systems in order to get that, in order to get data from the plant to the workstations, is that correct?

MR. CASE: Yes, that's correct. The information originates from the module protection system, which is based on the HIPS platform.

MR. HECHT: Okay. Thank you.

CHAIR KIRCHNER: Okay. Then, with that, we'll transition to the staff's evaluation of Chapter 7.

MR. VIVANCO: Good morning to the ACRS and staff and members of the NRC and NuScale.

My name is Ricky Vivanco. I am the Chapter PM for Chapter 7 of the NuScale Standard Design Approval Application.

To provide an overview, NuScale submitted Chapter 7, Instrumentation and Controls, Rev 0, on December 31st, 2022, and Revision 1 on October 31st, 2023.

The NRC performed an audit of Chapter 7 from March 2023 to August 2023. Here, I would like to note that during its review the staff recognized that the I&C architecture was, essentially, not changed between the US460 and the US600 design.

Use of the Highly Integrated Protection System Platform, diverse field programmatic gamma rays technologies, the number of groups, the number of divisions, data communications schemes, redundancy of data buses within a division, control of access design features, et cetera, in the US460 plant are the same as the Certified Design.

Because of this, only one audit issue was raised during the audit and resolved. NuScale submitted supplemental information to address the audit issue and no RAIs were issued as a result.

Joe Ashcraft and Dinesh Taneja are the technical reviewers for this chapter. Again, I am the responsible PM, Ricky Vivanco. Getachew Tesfaye is the Project Manager for the overall SDAA review.

Sections of the SAR:

Section 7.0 is the introduction and review process.

Section 7.1 is the fundamental design principles of the I&C system.

Section 7.2 describes the system characteristics of the I&C system.

To move to changes from the DCA to the SDA, the NuScale power module power uprate and safety analysis resulted in changes to the reactor trip setpoints and engineered safety feature actuation system actuation logic and analytical limits.

The setpoint methodology described by TR-122844-P documents the new analytical limits and setpoints.

There is a change to the common cause failure coping analysis due to the reductions in digital sensors.

And the US460 design added Type F post-accident monitoring variables.

Again, while there were some differences between the US460 and the US600 designs, the staff found that the Applicant provided sufficient information to support the safety findings and that all applicable regulatory requirements were addressed.

And that concludes the presentation for Chapter 7 from the NRC staff.

MEMBER ROBERTS: Yes, this is Tom Roberts.

I had the same clarification question on the section that didn't change from the previous one. I just want to understand what your basis is.

Pretty deep into Section -- let me find this -- I think it was page 7-63 in the SAR talks about your basis for accepting the ATWS approach. And it leverages pretty heavily the sensitivity study that the Applicant documents in Chapter 19 that shows that, for the more frequent events that could happen, complicated by an ATWS, the plant would take care of itself. And so, there's probably half a page on that, and then, you ended that, Based on this evaluation, the NRC staff finds the Applicant's assertion above to be reasonable. And then, you document a study the NRC did to confirm the analysis that was in Chapter 19.

So, it seems like, from the staff's perspective, an important part of the basis for accepting the approach to ATWS is the fact that the plant performance is not that bad; that as long as you have what would be characterized as an anticipated event, the plant will take care of itself. And so, the design of the I&C system certainly should be diverse, but that's kind of, you know, either secondary or ancillary to that.

So, I made a comment on that. Did I misunderstand what your intent was for that part of the discussion or is that really part of the basis for acceptance?

MR. TANEJA: Yes, this is Dinesh Taneja.

So, the requirement of the 50.62 of having a diverse means of achieving reactor trip and some of these other support functions, aux feedwater, which doesn't apply to NuScale, and turbine trip -- so, when we looked at the protection system design, it has diversity. So, that was a big review basis for that.

And also, in addition to that, the probability of achieving these functions, you know, so the regulatory requirements are that it needs to be better than 10 to the minus 5, I believe. And that's where the independent assessment was done by our PRA folks to assure that, yes, it also meets that requirement as well, achieving that function. So, I think that was the basis for approving this exemption request.

So, you know, it hasn't really changed at all from our DCA to the SDA. It's those features are still the same. They still have the same exact architecture for the HIPS platform and the arrangements. You know, it's using the same type of setup here. It's using a redundant setup of the equipment interface modules and the reactor trip breakers are aligned. So, it really is having the diverse FPGAs in the different divisions. So, it's not susceptible any kind of common cause failure and the reliability is pretty high.

MEMBER ROBERTS: Yes, thanks, Dinesh. I appreciate all that. And the question I'm asking would be hypothetical and it's probably not worth spending more time in this meeting talking about it.

But if you had a design where you didn't have that assessment in Chapter 19, or if the assessment showed completely opposite results, that if you had the event, the ATWS event, that the consequences would be really bad, you know, it seems like, from the way you've written the SER, that would be something that would certainly be a consideration in terms of whether it's acceptable, given that you've written that as part of the basis for acceptability. If they did the side study which you validated, that if the event were to happen, that it wouldn't be that bad, and that's what --

MR. TANEJA: Yes, but, you know, the thing is, when we looked at that rule 50.62, it was really specific to the reactor designs of the time. So, they were talking about the PWRs by Westinghouse and CE and their design features, and the BWRs. So, how do you diverse means of achieving that function was needed.

And I think it resulted from some events that happened at Salem, is that correct?

MR. VIVANCO: Yes, that is correct.

MR. TANEJA: So, like now, if you think about it from that point of view, that what if I do have a scenario where I have a transient without a trip, how does a plant cope with it?

So, you know, now we are looking at passive plants that have really a margin of safety and a margin of time that's there. So, all these factors really play into you achieving the safe shutdown condition or the ability to trip the plant, and there is a lot of time available, as opposed to the large light water reactors that we have had experience with.

So, you know, yes, it is a good way of asking this question, how do we cope with it, but, then, adequacy of defense-in-depth, I think we are looking at all the advanced non-light water reactors. Our safety goals are still to see that there is adequate defense-in-depth in achieving these safety functions. So, really, the fundamentals are still there. I don't think we are going away from that.

MEMBER ROBERTS: Yes, thanks, Dinesh. I think that answers the question.

Again, it would be hypothetical at this point to look at a plant design where a side study showed that the consequences of the event were really bad. Then, you would have to think about, okay, is the adequacy of either the plant design or the diversity in the trip system sufficient?

But, for NuScale, there's a pretty solid story, and you cite that in the Safety Evaluation, that there's kind of this two prongs of good diversity in the trip system, and if the event were to happen, the plant mitigates a consequence without the trip system, which it seems to me that both have a role. And that's really the point I wanted to make. And it sounds you're saying the same thing.

So, thank you.

MR. TANEJA: Yes.

MR. VIVANCO: Are there any additional questions on Chapter 7?

(No response.)

MR. VIVANCO: Thank you all again for your time.

CHAIR KIRCHNER: Thank you. Okay. With that, then, I think we should move on to Chapter 9. We'll begin, again, with NuScale, and then we'll hear from the staff. And depending on how much time that takes, we'll decide on taking a break. But let's proceed to Chapter 9 with NuScale's presentation.

MS. TURMERO: Good morning.

My name is Sarah Turmero and I'm a Licensing Engineer with NuScale Power. I've been with the company for about two years and I was previously a reactor engineer at Waterford 3.

MS. AHMED: Good morning.

My name is Freeda Ahmed. I've been with NuScale as a Licensing Engineer and I've been with NuScale for a little over two-and-a-half years. I have a degree in nuclear engineering and I've had over a decade of experience in the industry.

MR. GREEN: Good morning.

My name is Jordan Green. I'm a professional engineer licensed in Texas. I've been the with the Plant Systems Engineering Mechanical Group at NuScale for about two years. Before that, I was a systems engineer and programs engineer for about 11 years at South Texas Project.

MS. TURMERO: So, Chapter 9 is our auxiliary systems, and it's systems including the fuel storage handling, water systems, process auxiliary systems, air conditioning, heating and ventilation, and other auxiliary systems such as the lighting, communication systems, fire protection, and the fire hazards analysis.

Next slide. For Section 9.1, the focus is on fuel storage and handling, which includes the criticality of fuel storage and handling; the pool cooling and cleanup system; the fuel handling equipment, and the overhead heavy load handling system.

For Section 9.1.1, the criticality safety of fresh and spent fuel storage and handling, the criticality safety of fuel storage is addressed by COL Item 9.1-1, which is shown on the slide.

There were three audit questions. One focused on the design of the reactor flange tool and the criticality of the fuel while the reactor pressure vessel is in the reactor flange tool, and then, other audit questions were on the topic of pool criticality.

Section 9.1.2, the new and spent fuel storage. The storage is addressed by COL Item 9.1-2, which is shown on this slide. There were two RAI questions on the topic of pool inventory.

Section 9.1.3, pool cooling and cleanup system. The systems described in the DCA were consolidated as subsystems for the overarching pool cooling and cleanup system. The major components in subsystem remains the same as the DCA with the exception of the pumps and heat exchangers, which were reduced five trains to three trains.

And there were no specific audit or RAI questions for Section 9.1.3.

For 9.1.4, the fuel handing system, the design changes from the DCA, the new fuel elevator is now capable of handling irradiated fuel assemblies for inspection purposes. There are vertical travel limits. That ensures adequate shielding of the spent fuel assemblies.

And the new fuel jib crane classification changed from ASME NUM-1 Type 2 to ASME NUM-1 Type 1A with single failure-proof features.

And there were no specific audit questions to Section 9.1.4.

For the overhead heavy load handling system, the design changes include an increase of the reactor building crane capacity. The module-lifting adapter from the DCA was removed and is now integral to the crane itself.

The reactor building crane auxiliary hoist capacity increased. There were changes to the reactor building crane automated control system software that reduces probability of operator error.

Additional jib cranes were added to the overhead heavy load handling system and they are designed to ASME NUM-1 Type 1A.

And the heavy load exclusion zone above the spent fuel pool was removed.

There was one change to the COL Item 9.1-5 in relation to the heavy load exclusion zone. That wording was replaced with safe load paths.

MEMBER BALLINGER: This is Ron Ballinger.

We looked at this a very long time ago and we had a long discussion about the heavy load, the crane system itself. And I asked the question then, but I don't remember the answer. With respect to the safe load paths, are there mechanical stops for the crane as opposed to electronic interlocks that would prevent the crane from deviating?

MS. TURMERO: There are not mechanical stops.

MEMBER BALLINGER: There are no mechanical stops?

MS. TURMERO: That's correct. There are physical stops at end of travel, but nothing that would prevent going over the spent fuel pool physically.

MEMBER BALLINGER: Thank you.

DR. SCHULTZ: This is Steve Schultz.

This is a particular area where you have made changes that significantly affect the safety related to the operations of the crane, the capability of the crane, and so forth.

With regard to the COL item, can you describe the type of guidance that is provided by NuScale to the applicant to help them move through what you're asking them to do in the COL stage?

MS. TURMERO: For COL Item 9.1-5?

DR. SCHULTZ: Yes.

MS. TURMERO: We don't prescribe specific standards to be followed. However, we do anticipate that the zero applicant would follow Reg Guide 1.244, and that would be our recommendation for this COL item.

DR. SCHULTZ: Thank you.

MEMBER SUNSERI: This is Matt. I had a question here also. Matt Sunseri.

You made changes to the automated control software to reduce the probability of operator error. And as I recall on the previous version, operator error was a significant contributor to the risk of the station. So, in light of that, are there any limitations on whether or not the automated controls can be in override and the operator can take manual control, the operator and reintroduce those risks, or are there going to be administrative controls to prohibit or otherwise limit operator manual engagement?

MS. TURMERO: There are administrative limits. So, there's an override key that would be required and that would be administratively controlled.

MEMBER SUNSERI: And I'm presuming that use of that administrative control would require additional oversight or something of that nature to mitigate the incurred risk?

MS. TURMERO: That's correct. That would be part of the operating and maintenance procedures in zero Item 9.1-5.

MEMBER SUNSERI: Okay. Thank you.

MS. TURMERO: Next slide, please. For the audit and RAI results for Section 9.1.5, there were three audit questions on the use of ASME NUM-1, including demonstrating a compliance with Regulatory Guide 1.244, position C.1; one RAI on the deviations taken on ASME NUM-1 for the reactor building crane, and one RAI on the elimination of the heavy load exclusion zone terminology.

And that's all I have for Section 9.1. Before I turn it over to Freeda, are there any additional questions?

(No response.)

MS. TURMERO: Thank you.

MS. AHMED: Section 9.2 is water systems, and it consists of nine separate sections. The systems that are in bold text are the systems that have had changes from the DCA. All the systems that are not in bold have no major design changes. I will only be discussing the bolded systems, which is the potable and sanitary water system; ultimate heat sink; site cooling water system, and utility water system. However, I would be happy to answer any questions about any of the systems in Section 9.2.

And I would also like to point out that, other than the ultimate heat sink, all the systems are non-safety, non-significant systems.

Next slide, please. Section 9.2.4 is the potable and sanitary water systems and we had one change, which was that the potable and sanitary water systems piping, including the loop seals penetrating the control room envelope change from a Seismic Category II to a Seismic Class I. And the reason for this change was to ensure that the loop seals at the penetration could remain intact, thereby maintaining the leak tightness of the control room on load after a seismic event.

In addition to this change, we also removed two COL items. Because of the potable water system design, the Reg Guide criteria 1.206 and NUREG-800 are no longer applicable and all the general design criteria is included in the SDA. And we were able to successfully resolve one audit item.

Next slide, please. All right. Section 9.2.5 is the ultimate heat sink and we had a couple of changes from the DCA.

The first change would be that we had a smaller footprint of the UHS and the reason for this was that we went from 12 modules to 6 modules, so the lower inventory; and also, the UHS level was lowered from 68 feet to 53 feet from the bottom of the module.

And we had no audit or RAI questions specific to this section.

MEMBER SUNSERI: So, this is Matt. I have a question about this also.

I mean, I understand reducing the footprint and the boundary of the pool, but why would you lower the depth of the pool? I mean, it seems like that's giving away margin that would be easy to maintain.

Did you understand the question?

MS. AHMED: Could you just clarify one more time, please?

MEMBER SUNSERI: So, I understand why the total inventory of the UHS might go down because of the smaller footprint, but I don't understand why you would choose to lower the level from 68 feet to 53 in the remaining footprint, because that appears to me to be margin, giving away margin that is fairly cheap to maintain, unless I'm missing something.

MS. TURMERO: This is Sarah Turmero.

So, the pool level was reduced to help with the operation of DHRS. So, there was an issue of overcooling. And so, the lower level helps with the performance of DHRS.

MEMBER SUNSERI: Oh, okay. All right. Thank you.

CHAIR KIRCHNER: This is Walt.

Could you elaborate on that last point about the DHRS? You suggested that there was an overcooling problem.

MS. SWANSON: Hi. This is Mara Swanson, NuScale Power. I'm an engineer in the Mechanical Systems Group.

CHAIR KIRCHNER: Yes?

MS. SWANSON: Yes. I'm sorry, did you want to finish?

CHAIR KIRCHNER: No, go ahead, please.

MS. SWANSON: Yes. The lowering of the pool level was a result of overcooling concerns. So, during certain accident scenarios with an ECCS and DHRS actuation, the containment module would cool down too quickly, compromising the integrity of the containment module over long periods of time through lots of thermal cycling. So, the reduction of the pool level decreases the thermal stresses on the containment module.

CHAIR KIRCHNER: Thank you.

What I'm struggling with here, quickly, is the DHRS, basically, is immersed in the pool. Are you suggesting that going from 68 feet to 53 feet keeps some of it above the water level line?

I mean, pretty much, that heat exchanger affixed to the external part of the containment vessel will see a uniform temperature. So, if it's immersed, I'm not following how it could result in an overcooling transient. Your normal cooldown mechanism would not be using DHRS, would it?

MS. SWANSON: No, this scenario does not apply to a normal cooldown, and you're correct, the DHRS do remain completely immersed for a minimum of 72 hours during accident scenarios. So, it is for the ECCS system actuating, which involves the entire exposed surface area of the containment system. So, it's reducing the surface area of the UHS and the containment system itself, which is the conductive cooling that ECCS uses to --

CHAIR KIRCHNER: Correct.

MS. SWANSON: -- cool the primary fluid.

CHAIR KIRCHNER: But that would be an unusual event, for your ECCS system to be actuated.

MS. SWANSON: That's correct, that scenario applies only to very specific cooldowns.

CHAIR KIRCHNER: So, from what you've provided, like Matt, my concern is -- it's not a concern; it's just it seems to me you would have more margin.

So, the vessel, could you just kind of give us a feel for how much of the vessel, then, is immersed? The vessel, the containment vessel, is, roughly, 85-feet high -- I'm doing this from memory, which is not a good way --

MEMBER SUNSERI: I think he's got it there. I think it's on the chart there. The vessel would be 53-feet deep instead of 68-feet deep.

CHAIR KIRCHNER: Yes. Yes. Yes, essentially, roughly two-thirds of the vessel is immersed in water.

MS. SWANSON: That's correct, yes.

CHAIR KIRCHNER: Is part of the rationale just thermal performance; that if it's entirely immersed or more immersed at 68 feet, obviously, another 15 feet of water, that you're impacting your thermal efficiency? Well, you would have more heat loss to the pool during normal operation if it's more fully immersed.

I'm just trying to understand what design objective was being, what design objective was reached by reducing the water level.

MS. SWANSON: It was, yes, the design objective is to prevent the containment vessel failure over long periods of thermal cycling.

Regarding the other points you brought up, the thermal losses during normal operation, because our containment is kept at a vacuum, is pretty minimal. So, those --

CHAIR KIRCHNER: Pretty minimal? Okay.

MS. SWANSON: That parameter is largely unchanged from the DCA design.

CHAIR KIRCHNER: I see. All right.

MEMBER SUNSERI: So, maybe this is a question, then. I presume that, with the lower inventory in the pool, the water in the UHS is going to be warmer than it would be at 68 feet. Is that a fair assumption?

MS. SWANSON: The normal operating temperature is unchanged from 100 degree Fahrenheit.

During refueling operations, we did increase the maximum allowable pool temperature to 120 degree Fahrenheit.

MEMBER SUNSERI: So, the impact on the containment vessel is merely the temperature that metal would get based on only having 53 feet of it in the water versus 68 feet in the water. Is that --

MS. SWANSON: Yes, the rate of cooling during an ECCS actuation is that, yes.

MEMBER SUNSERI: Okay. All right. Thanks.

MEMBER HARRINGTON: This is Craig Harrington.

Not to belabor the point, but it seems like that portion of the vessel that's below water level, the temperature isn't really going to be different. So, how does that change the thermal cycling that I think you said you were attributing to essential failure over a number of cycles?

MS. SWANSON: I'm sorry, could you repeat the question? I'm not sure I understood.

MEMBER HARRINGTON: Well, by reducing the water level, the portion of the vessel out above the water level, that temperature may change, be different, but the part that's below water, it seems as it's immersed, would still be the same. So, I'm trying to figure out where the difference in thermal cycling that could lead to some vessel failure comes from.

MS. SWANSON: The concern is less with the temperature reached by the vessel and more related to the rate of change at which it cools. So, the increased surface area from a higher water level would cool the vessel down much more quickly than the reduced water level.

MEMBER HARRINGTON: It just seems like a bulk wall average temperature argument, as opposed to local temperatures below water and above water. Maybe I'm just missing it.

MS. SWANSON: Well, I think, for this level of detail, I'd need to defer this question until after the presentation. I can't speak to the temperature profile over the whole course of the cooling scenario.

MEMBER HARRINGTON: Okay. Thanks.

MEMBER SUNSERI: But you have an analysis that supports the change that will occur, is that correct?

MS. SWANSON: That's correct, yes.

MEMBER SUNSERI: Okay. Thanks.

MS. SWANSON: I see, in talking to some of the other staff, I think that this question at this level of detail is probably best addressed in the closed session.

MEMBER SUNSERI: Okay. We'll try to remember to ask then.

MS. SWANSON: Okay. Thank you.

MS. AHMED: Section 9.2.7 is the site cooling waters. And we only had one design change from the DCA. And that was we went from an open-loop system design in the DCA to a two-loop closed hybrid system in the SDA. And the reason for this is the design change was to better maintain the water quality for plant users.

We also removed two COL items, and the reason we removed the COL items is that, in the standard design, all (audio interference) system functions. And that will manage corrosion and fouling. And thus, there will no longer be a COL. And we were able to successfully resolve on audit question.

Next slide, please. Section 9.2.9 is the utility water systems. And we had a change from the DCA, and that was the removal of COL Item 9.2-5, which concerns the identification of the site-specific water source.

And the water treatment system was removed. And it was removed because there are no regulation requirements that pertain to the COL item, and the information is included in the SDA.

Additionally, we received two audit questions that we were able to successfully resolve.

Next slide, please. Section 9.3 is process auxiliary systems, and it consists of seven different sections. However, I will only be talking about the chemical and volume control system and the containment evacuation system, as they are the systems that had changed from the DCA.

I would also like to point out that, other than the DCS, all systems in this section are non-safety, non-risk-significant systems.

Next slide, please. Okay. Section 9.3.4 is chemical and volume control systems. And we had one change from the DCA and that was we removed -- the module heatup system was modified to use an electric heater in lieu of two heat exchangers that were providing steam from the auxiliary boiler.

And we were able to successfully resolve one audit item and three RAIs.

Next slide, please.

DR. SCHULTZ: Could you go back to that slide just for a moment, please?

Looking at 9.3.4, the additional design information in the last bullet there associated with flow-restricting venturis supporting probabilistic risk assessment, could you expand on that? What was provided and how did it affect the PRA? What was the question associated with that? How was the PRA affected as a result of the discussion?

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

So, what was added into Section 9.3.4 here was specific to the actual descriptions of the venturis, that the actual venturis themselves were not described as well as they could have been. So, we provided additional clarification in Section 9 related to the location of those components and a description of those components.

The venturis are operated in the PRA for line breaks where the venturis are installed, and they do provide a restriction for flow in those instances.

DR. SCHULTZ: That's going to affect the analyses that you perform, the heat associated with the LOCA and break, and so forth?

MR. GRIFFITH: So, it affects the small

line break analysis.

DR. SCHULTZ: Thank you.

MEMBER DIMITRIJEVIC: Hi. This is Vesna Dimitrijevic.

This is for the LOCA's outside containment, right, line breaks for the LOCA's outside containment? That's where those restrictive things are installed? It's my question.

MR. GRIFFITH: This is Thomas Griffith, the Licensing Manager.

That is correct. It is for the small line breaks outside of containment.

MEMBER DIMITRIJEVIC: And did that change, did they change the likelihood of those events?

MR. GRIFFITH: They did not.

MEMBER DIMITRIJEVIC: I mean did they think --

MR. GRIFFITH: Their impact is in the thermal-hydraulic analysis, not on the likelihood of an event.

MEMBER DIMITRIJEVIC: Okay.

MEMBER SUNSERI: This is Matt Sunseri.

These were installed as a result of the decision to remove the check valves in the line? That's my question.

MR. GREEN: This is Jordan Green.

So, the venturis and the two CIVs, and then, the third isolation valve, those are all considered containment system components. So, we're going to have more information on those in Chapter 5 -- Chapter 6, I'm sorry, 15 and 19.

The valves moved out of the nuclear power module bay are the CVCS valves and they were done so in support siting considerations, maintenance considerations, and reducing congestion. So, I think there is only one excess flow check valve, I think is the one that you're alluding to, and that was replaced with an AOV. And so, these materials were not added to compensate for that.

MEMBER SUNSERI: Okay. All right. Thank you.

CHAIR KIRCHNER: So, just for clarification -- this is Walt Kirchner -- we'll discuss this particular subject when we cover Chapter 6? Or later today in the closed session?

MR. GREEN: All the information on the venturis, their classification, their location, is included in Chapter 6 in the containment section.

CHAIR KIRCHNER: Okay. All right. Thank you.

MS. AHMED: Section 9.3.6 is the containment evacuation system. And we had a couple of changes.

The first change is that the CES inlet pressure instrumentation and connecting piping tube, up to and including the isolation valves, is designed to Seismic Category I standards. It was SC-III in the DCA. And this ensures that these components maintain capability to perform their function during and after safe shutdown only.

And the second change that happened is that same section of piping mentioned was increased to containment design pressure. And it was to act as a diverse, independent backup to CIVs in support of PRA. And we were able to successfully resolve three audit questions.

Next slide, please. Section 9.4 is air conditioning, heating, cooling, and ventilation systems. And it includes five separate sections. None of the sections are bolded because none of them had design changes. However, I would like to say that they had no design changes, but the physical size of the ventilation systems was altered.

And then, also, we removed four COL items, COL Items 9.4-1 through 9.4-4. And these COL items were in regard to the need for periodic testing and inspection requirements, and they were incorporated into the SDA, and thus, no longer needed as COL items.

And Section 9.5 is other auxiliary systems, which contains the lighting systems, communication systems, and fire protection. All three systems had some changes from the DCA. So, I will be discussing all three.

Next slide, please. Section 9.5.3 is lighting systems. And the main change from the DCA was that the main central room has a dedicated emergency lighting that is continuously on. It had an auto-transfer in the DCA, but now in the SDA it is continuously on. And we were able to successfully resolve two audit items -- two RAIs and one audit item.

Next slide, please. And Section 9.5.2 is the communication system. And the changes in that were that the sound-powered telephone system was removed. The health physics network was added to the communication system, and we removed COL Item 9.5-2 because it is now part of the Standard Design Application and not needed for a COL item.

And there were no audit or RAIs specific to this section.

Next slide, please. And I will now turn it over to Jordan to discuss Section 9.5.1, fire protection and 9A.

MR. GREEN: Thanks, Freeda. Section 9.5.1 is the fire protection program and reflects the building layout change. We successfully resolved three audit questions.

Next slide, please.

DR. SCHULTZ: Jordan, this is Steve Schultz.

Could you just expand on your last bullet there, that you discussed structural and electrical raceway fire barrier requirements? What was the topic of discussion and what were the results of the discussion that you had?

MR. GREEN: As I recall, it was just a reiteration that the raceway design meets requirements of ASME 119 and FPA 251 and is the responsibility of the Applicant. and per our Reg Guide 1.189 table 9.5.1-2.

DR. SCHULTZ: Thank you.

MR. GREEN: Section 9A is the fire hazards analysis. And again, this reflects the building layout change.

We successfully resolved one audit

question and one RAI -- with that last RAI being pretty extensive. We went through the fire hazards analysis, and then, explicitly stated each fire area that did not contain safe shutdown equipment and how propagation is mitigated.

And that concludes our presentation on Chapter 9. If there are any additional questions, we'll take those now.

CHAIR KIRCHNER: Members and Consultants, any further questions of NuScale on Chapter 9?

MEMBER SUNSERI: Walt, this is Matt. I don't have any.

CHAIR KIRCHNER: Okay. Thank you, Matt.

Hearing none, I see it's coming up on 10:00 Eastern time. While we transition to the staff, let's take a break until 10:10 a.m. Eastern time, and then, we'll pick up with the staff's evaluation of Chapter 9.

So, we are in recess until 10:10 Eastern time.

(Whereupon, the above-entitled matter went off the record at 9:56 a.m. and resumed at 10:10 a.m.)

CHAIR KIRCHNER: Okay, we are back in session and we are turning, next, to the staff's presentation their evaluation of Chapter 9, Auxiliary

Systems.

And I'll turn it to the staff.

MR. CRANSTON: Good morning. My name is Greg Cranston. I'm a Project Manager on the NuScale project, here for Chapter 9, Auxiliary Systems. I'm with the Division of New and Renewed Licenses, Licensing and Regulatory Infrastructure Branch.

Next slide, please. In conjunction with the overview, NuScale submitted Chapter 9, Rev. 0, SDAA FSAR on December 31st and Revision 1 on October 31st, 2023.

The NRC regulatory audit of Chapter 9 was performed in March 2023 through August 2023, generating 33 audit items.

When NuScale went through their presentation, they did identify those audit items and what systems they were associated with and, also, associated RAIs.

There were 10 audit issues opened, resulting in the NRC submitting supplemental information, and 13 RAIs were issued. All those issues have been resolved, and as a result, we do have four confirmatory items were, when the FSAR is updated, those confirmatory items will close to incorporate the feedback we got to resolve the audit issues and the RAIs.

Next slide, please. As was pointed out by NuScale, this auxiliary systems, there's quite a lot of differences to expand. I'm certainly not going to talk about them all today, either.

So, the number of contributors is probably quite a bit more than most other chapters. So, it was quite involved; a lot of people reviewed, basically, the 29 systems associated with the auxiliary systems.

Next slide, please. As mentioned previously, the main areas are fuel storage and handling; water systems; process auxiliaries, HVAC, and auxiliary systems. And then, Appendix 9 covers the fire hazards analysis.

Next slide. In conjunction with some of the significant items that the staff chose to identify, many of these, again, were previously discussed in the NuScale presentation.

On the fire barrier rating, there was a change from some areas being two-hour rates to now three-hour rated. And in both cases, the ratings were within code, whether it was associated with the DCA or the SDA.

And with the US460, there are only six nuclear power modules rather than 12 in the DCA. And as was discussed previously, the related parameters used in the analysis of the spent fuel pool and ultimate heat sink changed accordingly.

The reactor building crane capacity was increased to better handle the modules and a dry dock jib crane was added for refueling.

Next slide. The design included consolidating the spent fuel pool cooling and cleanup systems and combined the ultimate heat sink cooling systems into a single system that cools both the spent fuel pool and the ultimate heat sink.

And as discussed previously, the pool water level is lower and the operating temperature has increased. The fuel storage rack and design analysis will be done by the COL applicant.

And also, increased core thermal power will impact both the spent fuel pool and ultimate heat sink cooling, due to the increased heat loads associated with the nuclear power modules and spent fuel assemblies.

MEMBER SUNSERI: Hey, this is Matt. I have a question on that slide. Matt Sunseri.

MR. CRANSTON: Yes?

MEMBER SUNSERI: So, you're stating here in the third bullet UHS's operating temperature increased. But yet, when I asked the question during the NuScale presentation, they said it remained unchanged. Can you explain that difference?

MR. STUBBS: Yes. What is referred to there is the maximum allowable operating temperature. And I think they increased that up to 120 degrees.

Oh, and my name is Angelo Stubbs. I'm with Plant Systems, Containment and Plant Systems.

MEMBER SUNSERI: Okay. So, the maximum allowed did not increase, but the operating temperature did increase? That's what I would have thought, anyway. But okay. Thank you.

MR. HERNANDEZ: This is Raul Hernandez from Plant Systems.

It's the other way around. The normal temperature stays the same. The maximum allowable for evaluating the limits of the system, that is what increased from 110 to 120.

MEMBER SUNSERI: Okay. I got it. Thanks.

MR. CRANSTON: And the chemical and volume control system reconfiguration moved valves outside, again, as was discussed previously by NuScale, and the flow restrictors were added in conjunction with pipe breaks. And the flow-restricting venturis are credited in both Chapter 15 and 19, where they will be discussed further.

Regarding the SDA reactor building crane description, one item that we wanted to note was that, by going to controllers rather than total manual operation, that it had an significant improvement on core damage frequency, based on the calculated drop probability. And that was a big improvement, which, again, was discussed previously.

The application does not include specific spent fuel pool criticality safety design information and corresponding criticality safety analysis for the SDA. This is addressed in two COL items requiring the COL applicant to perform the criticality analysis for the new fuel and spent fuel pool.

So, in conclusion, while there are some differences between the DCA and the SDA, the staff found the Applicant provided sufficient information to support the staff's safety finding, which is that staff found that all applicable regulatory requirements were adequately addressed.

And that concludes my presentation.

MEMBER DIMITRIJEVIC: Hi. This is Vesna Dimitrijevic.

Can you go to the previous slide on the crane? So, they say that changes actually -- it resulted in operator error probability being negligible. So, how? Because this was the add-on of the Commission. So, how do those changes actually use the standard of the Commission?

MR. CRANSTON: Marie?

I was looking for a staff member who could --

MR. TANEJA: This is Dinesh Taneja.

MR. CRANSTON: Oh, Dinesh has got it. Go ahead.

MR. TANEJA: Hi. This is Dinesh Taneja. I am the I&C technical reviewer.

So, we audited the crane control system design. And basically, the crane is designed to meet the ASME Code requirements. And in accordance with that, the control system is designed to failsafe and all the safety features of the crane are implemented in a segmentation that is independent from the control system features that are used for controlling the crane.

So, by having that independence between safe follow-up protection on the control functions, and having the failsafe features implemented, I guess you can say that the reliability of the safety features and the control features is significant, you know.

I think, you know, from that point of view, you can kind of come to -- and in addition to that, the other thing that we were trying to confirm was that the software life cycle development activities for the control system are going to be performed in accordance with the software two-way program that is documented in Chapter 7. Specifically, I think it's Section 7.2.1. So, that really, also, kind of adds to the reliability of the development activities.

MEMBER DIMITRIJEVIC: So, I guess, then, when we review the Chapter 19, we're going to see this event developed in more details. So, we can see actually what you're talking about reflected in this probability. Is that what we should expect when we go to Chapter 19? Because that event wasn't developed in enough details to see something like this, you know.

And I guess, also, I was just really wondering, I mean, in operator actions, we're not really so specifically involved that you can see how the controls can prevent these events. I mean, you know, I was just wondering, are we going to see this in a little more detail somewhere?

MS. POHIDA: Good morning.

This is Maria Pohida, a Senior Reliability Risk Analyst in the Division of Risk Assessment.

Yes, we will be discussing the reliability and the reactability of the crane control system in the Chapter 19 presentation.

NuScale developed a special PRA for the reactability of the crane control system. I reviewed that with the I&C Branch and their technical expertise to verify the conclusions that operator errors are negligible contributors to module drop, given the assumptions of the reliability of the programmable logic control and the control system itself. But that will be discussed in Chapter 19.

MEMBER DIMITRIJEVIC: Okay. Thanks.

MS. POHIDA: You're welcome.

CHAIR KIRCHNER: While you have this slide up -- this is Walt Kirchner -- criticality safety is a COL item. I understand that.

This is a question for both the staff and the Applicant. I presume, in lowering the level of the ultimate heat sink, there were compensatory design changes to the spent fuel pool which communicates through a weir, such that the same margins in terms of shielding and water for cooling of the spent fuel, the assemblies in the spent fuel pool was maintained. Is that an accurate assessment of what was done with regard to the ultimate heat sink and the spent fuel pool?

MR. HERNANDEZ: This is Raul Hernandez from Plant Systems.

While we were reviewing the NuScale 600, the one with the higher level --

CHAIR KIRCHNER: Right.

MR. HERNANDEZ: -- the minimum water level needed for cooling was 55 feet, not 68.

CHAIR KIRCHNER: All right.

MR. HERNANDEZ: The additional water level was credited to provide buoyancy. So, the reactor building crane could handle the fully loaded nuclear power module.

CHAIR KIRCHNER: Right.

MR. HERNANDEZ: With increasing capacity of the crane, NuScale no longer needs to credit buoyancy.

CHAIR KIRCHNER: Right.

MR. HERNANDEZ: So, that additional margin of water was not credited for thermal or radiation protections. That additional margin was for buoyancy to move the fuel.

CHAIR KIRCHNER: Right. Hence, the

compensatory --

MR. HERNANDEZ: So, there is a -- sorry, go ahead.

CHAIR KIRCHNER: Well, so I understand the compensatory increase in load capability for the heavy lift crane because you would have less buoyancy. But my question was more directed towards the spent fuel pool and whether, if you're lowering the overall level in the ultimate heat sink and you're connecting the two pools via a weir, did they drop the spent fuel pool, the bottom of it, to the same level? Or maybe it's always been that, the same level as the bottom of the bay that accommodates the refueling and the six modules?

MR. HERNANDEZ: No, the bottom of the pool wasn't dropped. But what I'm saying is that the additional water was never accounted on the safe evaluation. We never credited that additional water.

CHAIR KIRCHNER: I see. Okay.

MR. HERNANDEZ: When we did our evaluation, the minimum water level was 55 feet, though they operated up to 68.

CHAIR KIRCHNER: Okay, okay. So, there's been no diminishment of margin, so to speak, with regard to the spent fuel pool?

MR. HERNANDEZ: There was a lower -- the water level has dropped. The minimum water level now is 48 feet for the thermal analysis.

CHAIR KIRCHNER: Okay. Yes.

MR. HERNANDEZ: So, there is still sufficient margin. And NuScale wants to talk, too.

MS. TURMERO: This is Sarah Turmero. With the flow level drop, we still maintain the required shielding above the fuel without any (audio interference).

CHAIR KIRCHNER: Okay. And then, heat load is not an issue?

MS. TURMERO: That's correct.

CHAIR KIRCHNER: Okay. It's shielding is the dominant design consideration then, not heat loading of the spent fuel pool?

MS. TURMERO: For criticality or?

CHAIR KIRCHNER: No, not for criticality. For just heat removal.

MS. TURMERO: No, shielding is not a dominant design factor.

Could you repeat your question to clarify?

CHAIR KIRCHNER: Okay. Thank you.

Members, further questions of the staff on Chapter 9?

MEMBER SUNSERI: None from me. This is Matt.

CHAIR KIRCHNER: Okay. Yes, I neglected to make this observation earlier for everyone participating. We assign leads amongst the members for each of the chapters. Our lead for I&C is Member Roberts. Our lead for the auxiliary systems is Matt Sunseri. Our lead for Chapter 12 on radiation protection is Dave Petti, and our lead on Chapter 18 for human factors is Vicki Bier.

And with that, I think we're ready to turn back to NuScale and Chapter 12.

(Pause.)

CHAIR KIRCHNER: NuScale, if you're ready to proceed, go ahead.

MR. SLOBE: Hello. My name is Erik Slobe. I'm a Licensing Engineer with NuScale Power. I've with NuScale for a little over a year and have been in the nuclear industry for eight years. I'm a Licensed Professional Engineer in Pennsylvania.

I'll be presenting on Chapter 12 for radiation protection of NuScale's Standard Design Approval Application. I will be focusing on the differences in the SDAA and the Design Certification.

In Chapter 12, we discuss ALARA and

radiation shielding requirements. Chapter 12 consists of five different sections. I'll be discussing each of the sections today.

This includes Section 12.1 on ensuring the occupational radiation exposures as low as reasonably achievable; Section 12.2 on radiation sources; Section 12.3 on radiation protection design features; Section 12.4 on dose assessment, and Section 12.5 on the operational radiation protection program.

We will be starting with Section 12.1 on ensuring the occupational radiation exposures as low as reasonably achievable. This section used the same methodology as what was used in the Design Certification Application. There were no significant changes made to this section.

Next, we'll be talking about Section 12.2 on radiation sources. This section also uses the same methodology as the DCA. The source term information, including tables 12.2-1 through -31, updated changes in source term information, are based mostly on the change in cycling, the increase in burnup rate, the change in thermal power, and the change in the number of NuScale power modules.

For source terms for systems shared between NuScale power modules, the tech spec design basis failed fuel fraction is applied to one NuScale power module, and the realistic failed fuel fraction is applied to the remaining NuScale power modules, as discussed in Section 11.1.

There were a few audit questions from the NRC that resulted in an update to the SDA. One audit question involved dose rates for workers on the fuel handling machine. The SDA was updated and clarified that the workers are not exposed to dose rates above 2.5 millirem per hour.

Another audit question involves clarification of the decay of N-16 in the CVCS and when it decays to insignificant levels. This occurs before the degasifier in the liquid rad waste system.

A third audit question involves the source terms for the low conductivity waste processing skid. Some of the source terms of the components of the LCW processing skid are now shown separate in the total of the skid in tables 12.2-12D and 12.2-13D.

Next, we'll be talking about Section 12.3 on radiation protection design features. As with the other sections, the methodology used is the same as the DCA methodology.

One difference from the DCA is there are now no very high radiation areas. This is due to more realistic modeling of the solid waste prepared for shipping and changes in the handling of the failed fuel fraction, and the reduction in the number of NuScale power modules.

There's also a reduction in the number of reactor building and rad waste building shield doors. This reduction in shield doors is due to the use of labyrinths, some lack of end room radiation sources, and some shielding analyses.

The number of fixed radiation monitors was also reduced based on an analysis of which fixed radiation monitors are necessary for providing adequate coverage of the localized areas of the plant. The reduction in fixed radiation monitors does not interfere with the plant operations, equipment, or personnel monitoring.

There were two COL items that were removed from this section. The first is COL Item 12.3-5, which concerned design criteria for additional area radiation monitors. The design criteria and regulatory criteria for area radiation monitors is included in Section 12.3.4.2, which makes this COL item unnecessary.

The second COL item that was removed is 12.3-8, which concerned radiation shielding for shield wall penetrations. This COL item is associated with a carve-out into the DCA, which is now resolved. The COL item is no longer needed due to the completion of more detailed radiation shielding analyses. SDA Section 12.3.2.4.1 has been updated to clarify that the calculations of the penetration show adequate protection from radiation streaming.

There were audit questions that impacted Section --

DR. SCHULTZ: Erik, can you just back up? When you say a more detailed shielding analyses, is that new methodology or is it more detail associated with the layout of the input to the methodology that you've used in the past?

MR. SLOBE: Yes, so this is more detail of analyses using similar methodology as other shielding analyses are done for the plant, but it did take in more detail of the layout of the plant.

DR. SCHULTZ: Okay. Thank you.

MR. SLOBE: In terms of audit questions that impacted 12.3, there was some language that was updated, based on a design change that replaced some break pot tanks and vent lines with hooded vents.

There were two RAIs associated with this section. The first concerned shielding. The language of the RAI was updated to clarify that the shielding was based on nominal concrete equivalent gamma attenuation.

The second RAI concerned radiation monitoring post-accident. It was clarified that the radiation monitors under the bioshield are PAM system Type F variables, which provide primary information to accident management personnel to estimate fuel damage and the effects of fuel damage. This is in addition to the Type B and C variables that were in the DCA. There are additional Type E PAM variable radiation monitors provided throughout the plant to effectively monitor the accident progression.

And next, we will talk about Section 12.4, which is on the dose assessment. The dose assessments were completed using the same methodology as the DCA. The results of the dose assessments differ from the DCA primarily due to changes in cycle length; increase in burnup rate; change of thermal power; the change in the number of NuScale power modules; building layout changes, and operational optimizations. The results of these are shown in tables 12.4-1 through -7 of Section 12.4.

Another change is that the vital areas for post-accident actions no longer include areas for initiating combustible gas monitoring. This was also related to a carve-out in the DCA regarding post-accident doses for the leakage associated with combustible gas monitoring systems. As discussed in Chapter 6, the SDA does not rely on combustible gas monitoring to assess core damage due to the use of the passive auto-analytic recombiners.

COL Item 12.4-1 was also updated as part of an audit question. The COL item was updated to say that the dose to construction workers from collocated existing NuScale power plants is the responsibility of the Applicant.

For Section 12.5 on the operational radiation protection program, there were no changes from the DCA.

And that concludes my prepared for Chapter 12. Are there any other questions?

CHAIR KIRCHNER: This is Walt Kirchner.

On behalf of Dave Petti, who is not with us today, I would like to ask a question going back to the change in level in the ultimate heat sink and the reactor pool. What impact does that have, or is it a negligible impact, on radiation, starting first with dose to the people doing the refueling operation? Is that a material impact or is streaming the largest component?

By streaming, what I mean is coming from the module itself rather than through the pool. Is that the dominant radiation exposure for -- source, I should say, for the refueling operation? You've removed the bioshields on that particular unit. So, I would presume streaming from the actual containment vessel in the vertical direction would be the major dose component?

MR. OSBORN: This is --

MR. BRISTOL: Go ahead, Jim.

MR. OSBORN: Yes, I'm sorry. This is Jim Osborn, Licensing Supervisor with NuScale.

No, the major contribution to worker dose in a refueling outage is going to be the evaporation off of the pool water itself. With the reactor shut down and removal of the bioshield, streaming -- and, of course, the containment is flooded; the reactor vessel is flooded. So, that provides sufficient shielding from any streaming pathways from the reactor core itself.

So, I think -- and Jon Bristol can corroborate me or correct me, if I'm wrong -- but I believe that the major contribution for a worker, you know, after removing the bioshield is going to be from the airborne in the evaporation from the pool.

MR. BRISTOL: This is Jon Bristol, NuScale, Engineering Radiological Manager.

For refueling, operator dose is a contribution of NPM component activation and source term in the ultimate heat sink from opening up the NPM. And then, there is a little bit of shining from the operating NPMs through the bioshield.

Ultimately, the lowering of the ultimate heat sink water level changed the shielding design criteria for the bioshield, which is adjusted to account for that water level difference.

But, in our design and analysis, we analyzed a variety of sources to operator dose.

CHAIR KIRCHNER: Okay. Well, it was mentioned by your colleague that atmospheric source from the pool was a contributor to that worker exposure. Previously, in an earlier presentation, we talked about the operating temperature of the pool. So, is that a factor in the evaporation rate?

MR. BRISTOL: So, there is evaporation from the ultimate heat sink and it does create airborne particulate in the area. We have the HVAC system designed to handle that, so that it's not a qualified airborne area.

Does that answer your question?

CHAIR KIRCHNER: Well, I'm just asking, if you operate at a higher pool temperature, then you're likely to have more evaporation; hence, more airborne particulate. You've got an operating HVAC system, I presume, continually during operation of the plant that may be adequate to keep the dose, the buildup of material in the atmosphere of the reactor building, prevent that from increasing dose. I presume that's the strategy there.

MR. BRISTOL: Correct.

CHAIR KIRCHNER: Let me reduce it to a simple question. Does operating the pool at a higher temperature increase the dose, the occupational dose? Or is that a negligible factor?

MR. BRISTOL: It's a negligible factor. The nominal ultimate heat sink temperature is consistent between the DCA and SDA. The upper allowable bound, if utilized, would be controlled through the operational programs to account for the radiological hazards of increased evaporation.

CHAIR KIRCHNER: Okay. And then, the second half of my question, then, is -- I'll make it a more direct question. Decreasing the water level, does that materially increase the dose to the refueling operating crew?

MR. BRISTOL: No.

CHAIR KIRCHNER: Okay. Thank you.

MEMBER MARTIN: To follow up on that, on the previous one, for clarification: so, if you've increased the maximum, that's usually where you would do your safety analysis. You've said that the evaporation is the largest contributor. I would otherwise expect that that's what shows up in those safety analyses.

You would have updated your analysis with 120, right, and that would have factored into what you presented?

MR. OSBORN: Yes, this is Jim Osborn.

Jon, I'm thinking that our normal operating temperature for the UHS didn't change.

MR. BRISTOL: Right, right.

MR. OSBORN: And I think we evaluated our evaporation for normal operations and normal worker doses at 100 degrees Fahrenheit, is that correct, Jon?

MR. BRISTOL: That's correct. And for Chapter 12.4, operator dose, you're not doing it at design basis conditions. It's a realistic representation of the operator exposure to radiation in the plant.

MEMBER MARTIN: Okay, well, I'm not a technical person, but I am a safety person, so I'm used to seeing that sort of thing being done at the limits, but I appreciate you said that.

MR. OSBORN: Right, this is not a strict cite to the analysis. This is radiation protection. This is different.

CHAIR KIRCHNER: Members, further questions? Okay, then I think we're ready to turn to the staff again, and their evaluation of Chapter 12. Just bear with us. It will take a moment to change out.

(Pause.)

MR. TESFAYE: Okay, the staff is ready. Can we start?

CHAIR KIRCHNER: Yes, go ahead.

MR. TESFAYE: Thank you, Chair. So just an overview of Chapter 12. NuScale submitted the Chapter 12, Radiation Protection, Revision 0 on December 28, 2022, and Revision 1 on October 31, 2023.

NRC regulatory audit of Chapter 12 performed March 2023 to August 2023, generating 13 audit items. Eleven audit issues were resolved in the audit. Nine audit issues resulted in NuScale submitting supplemental information to address questions raised during the audit. Two Requests for Additional Information were issued and resolved. Staff completed Chapter 12 review and issued an advanced safety evaluation to support today's ACRS Subcommittee meeting.

Technical reviewer and the final presenter today is Ed Stutzcage. Project manager, Chapter PM is Alina Schiller. Unfortunately, she's not here. She's on well-deserved annual leave this month. And I'm the lead PM sitting in for Alina.

So this section has been presented by NuScale. These are the various sections of Chapter 12. There's no point in going over them again. With that, I will turn over the presentation to Ed.

MR. STUTZCAGE: All right, thanks, Getachew. I am Ed Stutzcage in the Radiation Protection and Accident Branch. I have, essentially, five issues that we decided were significant to include in the presentation slide for changes.

The first two are related to carve-outs that were in the DCA. The first one here is related to the hydrogen and oxygen monitoring. If you recall, in the DCA, there was a carve-out due to the potential radiological implications of performing hydrogen and oxygen monitoring because you had to open up an --

isolate containment following a major accident and have a loop of the containment atmosphere through the CVCS system, the sampling system, and I think the containment flood and drain system.

In the SDAA, NuScale has requested an exemption from hydrogen and oxygen monitoring requirements. And that's being reviewed under Chapter 6, primarily, and I think Chapter 19, too, as well.

For the purpose of Chapter 12, the vital area mission dose requirements, we assume that exemption is going to be approved. It's still under active review, but it's -- I believe it's on a path. It's an issue that's changing, but it's on a path to resolution, I think. And they want me to do that hydrogen and oxygen monitoring. So, therefore, they want me to do this analysis, so that kind of resolves the carve-out that we had in the DCA.

Next slide, please. The second issue was -- and NuScale spoke to this -- this is the radiation -- the penetrations through the module bays. In the DCA, the radiation streaming through the penetration, what penetrations weren't explicitly considered. In the SDAA, they did update. They did provide calculations and assess the impacts of major penetrations. And we audited the calculations and, essentially, they weren't very significant to the dose. The major penetrations were really high in the module bay area and they were adequately assessed in the shielding and dose calculations.

Next slide, please. The third here is related to N-16. In the SDAA design, N-16 travels through the reactor coolant loop and the reactor vessel a lot quicker, significantly quicker than in the DCA. And, because of that, N-16 kind of makes it outside the CVCS lines in more significant concentrations than it did in the DCA. And we audited the NuScale calculation as part of the audit and made sure that the implications of this were assessed.

And, essentially, NuScale accounts for N-16 up until 10 half-life, when it's no longer significant. So the shielding and any radiological impacts to equipment qualification or anything, they assess the N-16 dose appropriately.

So that's that item. Next slide, please. Did we skip one? I think we might have skipped one. There we are. We skipped one earlier. Yeah.

This is just all the source terms and associated analysis with radiation shielding and doses and everything that were changed due to changes in the reactor power level, the cycle length, the number of units, all that. So, all the source terms, shielding calculations, doses analyses were all updated to account for this. And we reviewed the source term shielding based on the safety significance and found them to be acceptable.

Next item. And then the last one here is the change to the shielding approach for -- the way it's described in Chapter 12. So, Chapter 12 provides all the major radiation shielding for the plant. And, besides the bioshield shielding and water in the pool and stuff, the shielding is all described in terms of the concrete shielding thickness. It's evaluated for the appropriate -- for radiation attenuation based on concrete.

NuScale has updated their application to kind of say that if they want to use different shielding materials -- and through the audit we found that there may be a few cases where they want to use different shielding materials -- that it will meet the Chapter 12 radiation zoning and will meet all applicable regulatory requirements associated with whatever the shield wall is necessary for, just, you know, the shielding is needed for. So there's just changes in the FSAR language to describe the criteria that need to be met if an alternative shielding to concrete is specified. I think that's all for this issue.

So those are the significant items that we identified. There were some other changes that, most of them, NuScale described in their presentation that we didn't think were very significant, so we didn't include them in the slides.

Any questions?

DR. SCHULTZ: Edward, this is Steve Schultz. Just to go back to the catalytic converter. You mentioned that that issue is still being evaluated. I presume that what needs to be identified is how effective the converter will be and what's its reliability and availability, interaction, and conditions. Are those are the things that you're exploring?

MR. STUTZCAGE: It's not my review. I relied completely on those combustible gas reviewers in the Chapter 6 review. But I believe that's -- yeah, that's what they're -- that would be what they're looking into. And if, for whatever reason, that exemption from hydrogen and oxygen monitoring was determined to be -- you know, they still needed to do it, it couldn't be exempt from it, they would let me know, and this would then be an open item in Chapter

12 and Chapter 15.

So that's kind of where we are. Right now, we're in this spot where we think we'll get to where we won't need to do this monitoring, and we're assuming that for Chapter 12.

DR. SCHULTZ: As you understand, that review is on track?

MR. STUTZCAGE: I know there's been some ongoing review and changes going on there. I can't speak to specifics of that review. That would be in Chapter 6.

MR. TESFAYE: This is Getachew Tesfaye. There is no significant issue associated with that review that we know of. We presented these chapters earlier on. That's with the understanding we would go back. If something changed, we'd go back and reassess all the chapters as presented to you. So, that will happen when we complete the entire chapters that includes Chapter 6, Chapter 19, Chapter 15. We do anticipate some minor changes, to go back and tell you what the delta is. So when we evaluate Chapter 6, we will come back. If there's anything negative impact to Chapter 12 we will assess that then.

DR. SCHULTZ: But there's no expectation that there's going to be problems associated with this?

MR. TESFAYE: As it stands, I don't have any significant issue.

DR. SCHULTZ: Thank you.

MR. STUTZCAGE: So, yeah, I guess we didn't really do to the conclusion slide. While there is some differences, as we've discussed, we found that they provided sufficient information to support our safety finding, and that all applicable regulatory requirements were adequately addressed for the design within the scope of this chapter. And the COL items that are provided for the program and site-specific aspects are mostly similar to the DCA, and they're appropriately provided in the SDAA for the applicant to address.

MR. TESFAYE: That concludes our presentation. Any additional questions from members?

CHAIR KIRCHNER: Members, any further questions?

Actually, then, if there are no further questions, I'll use this opportunity to make a statement on behalf of the Committee. The Committee, April 28, 2020, wrote a letter on this topic that we were just previously discussing, combustible gas monitoring. And we raised serious concerns about the approach that was contemplated during the DCA, which would have required opening a large pipe, the containment evacuation system line, to set up a loop so that a realistic grab sample could be obtained to monitor hydrogen and oxygen content in the containment.

It was the view of the Committee that this presented a much greater risk of exposure to the workers, and potentially offsite, than the information that would have been gained in trying to assess what the hydrogen and oxygen content was.

So what the applicant has done, I think, addresses -- is a much better approach. We will review that subsequent to the exemption request and the details of the radiolytic -- I misspoke, the catalytic recombiner performance, in a subsequent chapter. But this change in design approach by NuScale and the review by the staff addresses a major concern that had been raised during the DCA.

And so a preliminary -- I won't draw any preliminary conclusions on behalf of the Committee, but I do just want to acknowledge the applicant's effort here to address what appeared to be a major risk-contributing process or procedure for post-accident monitoring. And so I'd just put that in the record.

Any further questions?

(No response.)

CHAIR KIRCHNER: Well, with that, let's turn then back to NuScale and Chapter 18. And bear with us again while we change both the presenters and the slides. Thank you.

(Pause.)

CHAIR KIRCHNER: Doug, if you're ready, please proceed.

MR. BOWMAN: Thanks, Walt. Good morning, everybody. My name is Doug Bowman. I'm Plant Operations Manager for NuScale Power. I've been at NuScale for 10 years now. All of my work at NuScale has been in the area of either human factors engineering, plant procedures, or training.

Prior to coming to NuScale, I spent 24 years in commercial nuclear power. I was SRO-licensed at both D.C. Cook and Byron Station, and also served many other positions along that path.

Next slide, please. So, Chapter 18 is broken up into 12 sections, and these sections are broken up as NUREG-0711 breaks them up, by element. The bolded sections you see there will be the ones we will cover, as they're the ones that have changes in.

The rest of them we will not cover. So those are functional requirements, analysis of functional allocation, task analysis, staffing and qualification, treatment of important human actions, and human-system interface design.

You can go on to the next slide. So, one of the big things we wanted to discuss here is, one of the things we did change from the US600 DCA to the SDAA is what we are submitting for this portion. So in the DCA, previously, we submitted a number of RSRs. We submitted all the elements, all the RSRs for the elements necessary get to the point of performing ISV. And then we performed the integrated system validation during the review phase of the DCA.

For this submittal, we did change that up a bit. We are submitting implementation plans. Implementation plans are allowed. They are strictly a description of methodology we will use, whereas Results Summary Reports describe both the methodology and results. And NUREG-0711 does allow for either one.

So we did take certain elements and we pulled them back to where we were only submitting IAP (phonetic) for the SDA, and this was done for a number of different reasons. One was a bit of lessons learned from the DCA. Another is that, if you look over what your human factors engineering program is supposed to cover, your human factors engineering program is supposed to be in place very early on in the design process, but it really needs the entire design complete in order to finish its portions of it to really understand how the humans will interface with it.

So we really wanted to move back to a more traditional model of human factors engineering, rather than the way we did it previously in the DCA. And we understood why we had to do that. We had novel design concepts and novel staffing plan, and we really had to be able to demonstrate that to the NRC to show that that would be acceptable. And now that we've done all that work, we feel like we can do our work in a more traditional manner.

Next slide, please. So our first section to discuss, then, is 18.3, which is functional requirements analysis and function allocation. The purpose of this element is to take those functions that the plant design has already been broken down into and understand how the operators will complete those functions. And, really, this breaks down into either it's going to be a manual operation by the operator to complete that function, a shared condition where you have both automation and manual actions, or a fully automatic function.

And the big thing we changed here is, as we matured our process, previously in the DCA we had separate databases tracking FRA/FA, task analysis, and procedures. And what we did was really combine all those together in one single interlinked database. And this is really an advantage for us for an efficiency and auditability standpoint. We now have direct connections from procedures to training, and procedures to human factors engineering elements. So it really becomes a very powerful tool for us to use down the road if we can continue to maintain human factors engineering programs as we move through the design process.

Next slide, please. 18.4 is similar. We actually use a very similar process to what we did in the DCA for task analysis. However, this, again, was combined as part of that large single interlinked database as we previously described in Section 18.3.

Next slide, please. 18.5, staffing and qualifications. S&Q determines the number and qualifications of licensed operators required for safe and reliable plant operation. Our minimum staffing numbers for the SDAA are one licensed reactor operator and two licensed senior reactor operators. All of these changes are a result of the topical reports that were submitted and was previously reviewed by both the staff and the ACRS. That staffing plan, that topical report number is listed there, and we also have the reviews by the ACRS, the ML numbers, back in April of 2021. So we essentially just leaned on that new revised staffing and plant validation methodology to develop our minimum operator-licensed staffing for this design.

Next slide, please. 18.6, treatment of important human actions. So identification of important human actions, we go and we look at Chapters 7, 15, and 19. Chapter 7, we're looking at the D3 analysis, looking to see if there's any operator actions listed there. In Chapter 15, we're obviously looking at the accident analysis to see if there's any actions that the operators would have to perform there. And then 19, we're looking for anything that is identified as risk-important in Chapter 19 would rise to that level of risk-importance of a human action. And the US460 standard design, unlike the DCA, has two important human actions. The U.S. standard design does not have any important human actions.

Importantly, that last bullet there I'd like to highlight. No matter what the action is, if we assume an action takes place, especially in PRA in Chapter 19, we cover that action in procedures, training, and development of the human-system interface. So, again, to reiterate, if there is an action assumed in Chapter 19, we still address those actions and still make sure the operators are capable of performing them correctly. So even though we eliminated two important human actions, those same two actions are still addressed by our procedures set and our HSI.

MEMBER DIMITRIJEVIC: This is Vesna Dimitrijevic. You didn't eliminate those actions; you just didn't classified them as important. But those actions are still in the PRA, right? Those were actions to make up, you know, for the coolant in the case of isolation and the LOCAs outside of containment. So those actions are still in the PRA, but they didn't show as important in the new Chapter 19. That's what you're saying?

MR. BOWMAN: That's exactly what I'm saying. That's an excellent --

MEMBER DIMITRIJEVIC: All right. One of

the points which I find a little strange is these actions that are identified as important to the Level 2 because of the LOCAs outside of containment. And based on some of the design changes, the math mentioned in, you know, the containment isolation, you would assume that LOCAs outside of containment become more important, but, actually, based on what you are saying, if those actions are not important anymore, it seems that LOCAs outside of containment are not any more significant contributors to the Level 2.

MR. BOWMAN: I'll say they're not as significant of a contributor. And I'm going to go into a little bit of detail and then I'm probably going to state that we should probably ask this question in Chapter 19. But there were changes made, especially in ECCS. So, we had changes in actuation set points, and what we learned through our process in PRA is that, if we had both trains of ECCS actuated, even with a containment bypass event, i.e., a LOCA outside of containment, we did not run into core damage, even if no operator actions were taken.

So that change in ECCS capability, and also the change in the actuation set point, it now comes off of riser level and it actually hits much earlier in the event, so you wind up not needing to take operator action, or it's not required to take the operator action in that case to prevent core damage.

Now, oddly enough, if we had those conditions, by our emergency operator procedures, we would still take the action to go ahead and initiate CBCS injection in those cases. So we still have --

MEMBER DIMITRIJEVIC: So, basically, what you're saying, maybe those actions are not anymore modeled in the PRA. They're not necessary for your assumed mission, then, right?

MR. BOWMAN: Correct. Right.

MEMBER DIMITRIJEVIC: I may be wrong in my previous statement that those actions are still in the PRA, because what you are saying, they may not be in the PRA anymore.

MR. BOWMAN: No, I'm sorry. They are still in the PRA. They are still assumed in the PRA, but the changes in the design allow us to -- even if we don't take the operator actions, if we take down the path, the fault tree path where the operators don't complete the actions, we still don't wind up in that core damaging situation, if we have both trains of ECCS actuated.

MEMBER DIMITRIJEVIC: I see. Okay. Well, we will pay attention to that in reviewing Chapter 19.

I have another question for you, what we also discussed previously in the, you know, previous discussion on that Chapter 9. And this is the operator commission actions and shutdown on the crane. And these actions are obviously considered, based on our previous discussion today, both in the chapter, you did not identify risk-important -- nothing risk-important for Chapter 19, but also you didn't get the actions from the Chapter 7 and 15, right, to be considered? And I was wondering why this action, commission action on the crane was not considered even, because if it was eliminated or reduced in the likelihood that would be good example or a example of human-system interface. I mean, why was not this action not even mentioned in Chapter 18?

MR. BOWMAN: At this point, I'd have to assume that, based on the information we have with the automations and the evaluation of the automatic functions for operation of the crane, that that would then cause a reduction of risk. But, again, I think we talked about this previously. It's probably better off discussed in Chapter 19 how those -- how all that combines to affect the risk.

MEMBER DIMITRIJEVIC: I do agree with your previous conclusion of makeup with the charging of, you know, the CFDS. But the thing is here, when it comes to the commission action, I thought you should identify this action through the Chapter 7, for example, 9 or something. And at least discuss it in the Chapter 18. I mean, we will look at that in Chapter 19, but this would be a very good example of the deterministic inclusion of the actions. So, that's just my comment and my personal opinion.

MR. BOWMAN: Thank you. I appreciate it.

CHAIR KIRCHNER: Doug, while we're on this -- this is Walt Kirchner again. I don't know that it rises to the level of a IHA, as rigorously defined, but for a fire in the main control room and evacuation, you do identify the alternate shutdown. Since you've taken away the shutdown -- there are no shutdown station kind of capability, you would have operators go to an alternate location to be able to effect a shutdown if they didn't do that from the main control room. I know that's part of the procedure.

So does that rise to the level of a IHA? The alternate shutdown locations and the training that would be necessary to ensure that when the operators went to that location, they operate the equipment to effect a shutdown and actuation of the ESFAS system.

MR. BOWMAN: So, I'll try to answer that

question. So, the alternate operator work stations, first of all, really there's no change in capability at the alternate operator work stations. The remote shutdown station that was described in the DCA was there strictly for operator monitoring. They were intended to complete the shutdown and placing the units in safe shutdown prior to leaving the control room. We do include in that procedure set a set of actions that can be taken in contingency if you are unable to ensure safe shutdown from the main control room once you got to the alternate operator work station.

CHAIR KIRCHNER: Right.

MR. BOWMAN: And those would include likely going to the MPS rooms to complete those actions if you needed to do that.

CHAIR KIRCHNER: Right.

MR. BOWMAN: But this is all in -- there's really no path that -- I guess, from a deterministic standpoint, it's not identified as an issue because it's not a -- no accidents are assumed to occur with a control room evacuation. That's in the regulation.

CHAIR KIRCHNER: Correct.

MR. BOWMAN: And so if you don't have an access, it's really hard to get to a situation where that would be important. So it's not a credited action anywhere in Chapter 15 since we don't have an accident to mitigate.

And from a D3 standpoint, again, D3 has a little bit difference of a focus, I guess. It's really surrounding the digital architecture and what's potentially not diverse, and what you have to do in order to address those potentially not-diverse digital assets.

So it really doesn't show up anywhere in any of those paths of important human actions. It's either a deterministic thing that you had to do for Chapter 15 to address an accident, or something you had to do to address a lack of diversity in your digital control system, or it's a risk-identified action. Those are the three paths we have to get there. It didn't rise up in any of those.

CHAIR KIRCHNER: Okay. Thank you.

MR. BOWMAN: All right. Next slide. And then 18.7, HSI design. The HSI design element takes all that other work we've done prior in terms of task analysis, FA/FRA, and treatment of important human actions, staffing qualification, and builds the HSI design in.

And really what we're highlighting here is the fact that there are changes in the US460. Those changes are either in response to operator feedback to the ISV we performed in DCA, or design changes we had to make from the US600 to get to the US460. So all those design changes you're going to hear about over the next month or so are all the things that we had to address in the HSI design.

That's really the only two paths we have. We either got operator feedback and we had to change something to address it, or we have a design change that came from the US600 which affects one of our HSI designs.

Next slide, please. And then we also added one ITAAC. This is a result of our implementation plan strategy. Previously, we had a single ITAAC that identified the design implementation element, and you had to ensure that the final control room, as built, matched the control room design that you had developed.

In this case, we added one to ensure that the integrated system validation test was completed once we had completed all the other elements that led up to integrated system validation. So, really, the final thing is that we have a report exists and it concludes that the acceptance criteria associated with the ISV has been satisfied. So that's our additional ITAAC.

Next slide, please. And then the summary of the audit. We did have 20 audit items that were successfully resolved. They included a wide range of questions from staff. And we also included a virtual demonstration of both some of the simulated changes we had made, and the staff review of those databases we talked about. So we did a pretty detailed virtual presentation with staff on the databases to show them how they were connected.

Next slide, please. Are there any other questions for me?

MEMBER BIER: I guess my question kind of goes beyond what your proposing and applying for right now kind of more long-range, which is regarding staffing levels. If there were to be increased numbers of units, would you envision that staffing levels would increase kind of linearly or maybe stay constant? Or you haven't really thought that far yet to have a plan?

MR. BOWMAN: We certainly have. Again, the revised staffing plan validation showed that, for up to 12 units, the minimum operators required for safe operation of the facility is three. That's one

RO and two SROs. Obviously, there are commercial considerations that would potential require you to bring additional operators on staff. We can envision a number of different staffing models from essentially an extra shift on day shift to deal with those maintenance and other commercial issues, to potentially -- if you've got a 12-unit plant, you might want to add an extra operator normally, just to deal with work. But, again, that's all commercial considerations. It's really up to the licensee on how they want to staff based on that workload. Again, our conclusion is that three is what's necessary for safe operation of the facility.

MEMBER BIER: Okay. And the lack of a designated SDAA is, again, based on kind of timeline and remote support options? Or can you just go over the rationale for that again?

MR. BOWMAN: So the SDAA, elimination of the SDAA was identified at the time that the TMI rule came out, what you had to do to do that.

MEMBER BIER: Yes.

MR. BOWMAN: And we showed in the staffing -- the topical report we did for staffing, that we had met those requirements, which really came down to SRO training upgrades and HSI upgrades. So we believe we showed and demonstrated in the topical report that we had met those requirements of the original TMI plan and that we no longer needed the capability of having an SDAA.

MEMBER BIER: So it's really based on the timing issues. The timing issues are maybe an additional advantage, but not necessary.

MR. BOWMAN: I wouldn't say the timing -- you're talking about the length of time for an accident to progress and the need for -- I would say was definitely a factor in that. It certainly helped the staff with their review that they knew that we could -- we had on the order of an hour before we really need any supplementary people, and even longer in most cases.

MEMBER BIER: So it's more of a kind of comfort level factor at the time?

MR. BOWMAN: Yes.

MEMBER BIER: Okay. Thank you.

MR. BOWMAN: Any other questions?

MR. BROWN: Yes, I'm going to backtrack -- this is Charlie Brown -- on two points, but I want to backtrack to the remote shutdown station issue again.

When we originally reviewed the HIPS systems and everything else, and we went through the first versions of these, we had the remote shutdown system. It was remote shutdown. That's the words that were used. That was not just monitoring. Maybe I misunderstood back five, six, seven years ago, but we were assured, based on the other comments and the notes in that earlier revision on the DCA as well, that there were manual control switches that then actuated and ran down -- I'm looking at the old figure again -- and actuated the scram for the scram breakers to get them shut down, hardwired. And that was the hard-wired module that went down.

Now you're telling me that there is no remote capability of manual shutdown through the hardware systems if you have to abandon the main control room under some circumstances.

So, I did not realize that when I did the review of this, so I'm just throwing that one back on the table, Walt. I'm not particularly enamored without having a hardware method for shutting down the reactor plant, tripping the scram breakers, if the main control room is unavailable for that actuation from internal.

So, I don't think that's a good idea. Maybe nobody likes that comment, but that's my thought process on that.

The second question was now on the staffing. There's one licensed reactor operator and two licensed senior reactor operators. But are those senior reactor operators always -- at least one of them, always in the main control room, along with the licensed operator?

MR. BOWMAN: So I'll take your second question first. I would like to address your first statement, too. Yes, there's always -- the minimum staffing inside of the main control room at all times is one SRO and one RO.

MR. BROWN: Okay, that's good.

MR. BOWMAN: One person could go out in the field to check on things or other things, but, yeah, our bare minimum staffing in the control room is that.

MR. BROWN: So there will always be one of the two -- or one of the three could leave and then come back, but there would always be at least two people in the main control room?

MR. BOWMAN: Correct.

MR. BROWN: Okay, you wanted to go back to question one again.

MR. BOWMAN: Yes, I do. So we absolutely have the capability outside the control room to open the reactor trip breakers in place the reactor in safe a shutdown.

MR. BROWN: How?

MR. BOWMAN: We would have to go to the MPS rooms and perform those tasks locally. But that's no different than the current plan --

MR. BROWN: What's MPS again?

MR. BOWMAN: Module protection system rooms.

(Simultaneous speaking.)

MR. BROWN: So you have to actually go to the reactor instrument rooms to do that, as opposed to where you had all the monitoring?

MR. BOWMAN: Correct. That's no different than what current plants do today.

MR. BROWN: Not some of the plants I'm familiar with. Okay.

MR. BOWMAN: The two plants I was licensed at you had to go --

MR. BROWN: I'm not talking about commercial.

(Simultaneous speaking.)

MR. BOWMAN: -- went down to the trip breakers and opened them locally.

MR. BROWN: Well, I understand you can do that. I'm referring back to my naval bona fides after that, which we could it at the cabinets. We could also do it in the main control room, or from a local station, which all you had to do is take three steps to do it.

Let me think about that a little bit. How far is the module protection system cabinets from the main control room?

MR. BOWMAN: I would have to get you that. I do not have a number on the top of my head.

MR. BROWN: But somebody has to exit -- you have to exit. You're out now, and you have to run somewhere, go up and down steps, down to the room, unlock -- do you have unlock a cabinet or unlock a switch that's protected with a lock so it's not inadvertently actuated?

MR. BOWMAN: There would be access control requirements on the door to the room. But, beyond that, the operators would have access to that so they could get in and perform the actions promptly. So it's nothing more than the time -- and don't forget -- I'll also back up. I'm going to have to state this. Those operators are complete -- those plants are completely safe without the operators in the main control room, in our design. That's very different than what's currently in the plant designs, right?

MR. BROWN: I understand that.

MR. BOWMAN: The protection system is fully operable. It will still shut the plant down automatically, if needed. In the meantime -- the plant is safe no matter what condition we're in, because the plant is safe without operators in our design.

MR. BROWN: I understand that. In spite of all those other types -- and I understand it. I've been through this for quite a few years now, and I'm always uncomfortable with having delays in order to be able to manually shut it down. In this case, you leave the room. You have to go somewhere up and down steps. If you have to have access to another room, does that require a key or a special somebody else to allow you in? It's just more difficult.

All right. I threw that back out if the Committee -- I'm just a consultant now, so if the Committee says this is okay, we'll proceed. I just wanted to get the discussion out on the table so that everybody had a common understanding of what this actually looks like now.

So, Walt, I'm done with that, if I've mouse-milked this enough.

CHAIR KIRCHNER: Thank you, Charlie. Members?

MR. BROWN: That means I'm done.

CHAIR KIRCHNER: No, it means you've been heard as well, Charlie. It's a concern that we share, I think.

MEMBER MARTIN: Hey, Walt. This is Bob. I just wanted to follow up, maybe just fill in my own lack of knowledge. You mentioned the MPS, and I assume physically it's located somewhere relative to the main control room and everything?

MR. BOWMAN: So it's on the reactor building. They're essentially immediately adjacent to the modules just outside of the --

MEMBER MARTIN: Now, should be expect that, like, the design criteria for the control rooms are the same for this space?

MR. BOWMAN: You mean for the space below the operating --

MEMBER MARTIN: Well, for the MPS room, if you have to go in. Obviously, they're not going to be there very long if they're doing an action like that, so that factors into it. But it's really my question to fill in a gap in my knowledge. Is there any requirement that that -- I'm really thinking about dose, related to that particular space and whether it aligns with what's otherwise required for the main control room?

(Simultaneous speaking.)

MR. BOWMAN: I can at least tell you from a EQ perspective it's a mild environment, right? So it should be reasonable dose rates if it were required to access. Temperatures are controlled, so it's not going to be hot or anything like that. And it is a vital area, so it is controlled from a security perspective. But, again, to kind of go back to Charlie's question, the operators will have access to those rooms. Normally, in my past life, you had a badge that you show to the reader and got access to the room so they can take the action.

MEMBER MARTIN: Thanks.

MEMBER ROBERTS: Yeah, this is Tom Roberts. Can you go back to that slide on the important human actions? I also have kind of an ignorance question I wanted to ask while we were here.

The third bullet says there's a credited deterministic human action from the D3 analysis. I was looking through the D3 analysis just now. So what are examples of deterministic human actions that are credited?

MR. BOWMAN: I guess that's a poorly written statement. There are no -- if there were deterministic human actions, they would have been credited D3, but there aren't any for us.

MEMBER ROBERTS: Okay, yes, that's pretty clear.

MR. BOWMAN: Sorry, that's a poorly written statement, I agree.

MEMBER ROBERTS: All right, thank you.

CHAIR KIRCHNER: Doug? This is Walt again. Just a follow up on Charlie's line of questioning. So at the NPSs, let me just suggest one scenario.

If you were to turn off the power to the, at the chassis, would that in effect, result in the scram of the system?

MR. BOWMAN: Yes.

Removing power from NPS system, all actions are failed to their safe condition. So, if you turn off power, everything's going to actuate.

So, yes, turning off the chassis would --

(Simultaneous speaking.)

CHAIR KIRCHNER: Yes, so that's what I would expect, a fail-safe design.

So Charlie, does that answer your question? If the operator is forced, the operators plural, the three-man crew, have to evacuate the main control room, then they could proceed to the NPS and also effect a manual scram by just turning off the power to modular, module protection system.

MR. BROWN: Well, I would have to go back and look, but the breakers that they have are if you lose power to the breakers, that means the under-voltage coils, whatever, they will trip the breakers.

It's a matter of how --

CHAIR KIRCHNER: Yes.

MR. BROWN: -- how it was delivered by the NPS and do you have to turn off all, how many channels do you have to turn off because they've got independent power supplies.

And each of those power supplies is fed by a separate breaker from some power panel somewhere.

So, it's not just one action. You have to take multiple actions to get them de-energized. You could get there. It's just the timing of it is a much longer period.

And the general argument that they're giving is, if everybody died in the plant, the plant is safe.

I'm saying that kind of, I'm maybe overstating it, but just to make the point of this plant.

CHAIR KIRCHNER: Well, this here --

MR. BROWN: Everybody, we could leave the site and this plant is safe. You don't have to worry about it. And I'm a typical skeptic just based on 35 years of previous experience that you ought to be skeptical.

So, that's why I like manual switches that just bypass all the other stuff that has to activate, or deactivate, in order to turn them off.

You ought to be able to just remove, trip the breakers by turning off all the under-voltage coils remotely, without any access from the main module protection system.

And back in the old days when we first looked at this years ago, there were manual switches. At least that's what they were advertised as being. And now they've disappeared.

That's the major change I see. I'm not saying your plant is unsafe, I'm not saying any of that. It's just I'm not particularly comfortable with that.

If I was on the committee and helping to write the letter, I would make that known. But in this circumstance, I'm only presenting it as my thought process.

But I don't like back stepping like this. So, I don't know if that answers Walt, but that's, that's just long time and a lot of plants being responsible for 35 years. So, in real time.

MEMBER ROBERTS: Well, let's ask the person who is responsible for writing this letter. I guess I wanted to respond to Walt's hypothesis with a question for the applicant.

If you really turned off the MPS power supply from the control room, does that deactivate the UPSs that, or holding the power for some time after loss of input power, or would you really be able to completely de-energize the MPS from the control room?

MR. BROWN: That's a good point, Tom.

CHAIR KIRCHNER: Well, I wasn't suggesting from the control room. I was suggesting de-energizing from the MPS panel. Presuming --

(Simultaneous speaking.)

MEMBER ROBERTS: Right. I was thinking if you can get to that room, I think you would just trip the breakers.

CHAIR KIRCHNER: Probably.

MR. BROWN: It would be much faster to manually trip the breakers, wherever the switchboard is for the scram breakers.

They are manually, I presume the scram breakers are manually operable. Is that correct, NuScale?

MR. BOWMAN: Yes, we can manually open the reactor trip breakers.

MR. BROWN: Just like going to your power panel in your house and tripping the breaker?

MR. BOWMAN: Well, little more involved than that because it's --

MR. BROWN: I got it.

MR. BOWMAN: -- a three-phase, it's a three-phase directable breaker that you have to go push a button on it, have springs popping open.

MR. BROWN: Yes.

MR. BOWMAN: But in function, yes, it's the same thing.

MR. BROWN: Okay.

CHAIR KIRCHNER: But we're talking about a scenario first of all, that requires evacuation of the control room.

So if they follow their procedures, and we're not reviewing procedures at this juncture, it's reasonable to expect that they can write a procedure that directs the operators to the MPSs, and/or the breakers to disable, disable isn't the right word, but to trip the reactor protection system and the S-class.

I don't think that's an unreasonable or unrealistic length of time to accomplish that task.

Is that a good assumption, Doug?

MR. BOWMAN: Yes. I would find it no more, I mean, I am aware of an operating plant where an operator from the control room had to walk down stairs, and a short walk to get to the trip breakers to trip them open. So to me, this is a similar action; similar timeframe; similar capability.

We're not asking, we're not trying to do anything different than what's been done in the industry in my 24 years of experience, for a very long time.

MR. BROWN: Walt, can I make another comment relative to that?

CHAIR KIRCHNER: Yes, you may.

MR. BROWN: I've gone back and looked at the older diagram. In addition to the remote station, at least in that diagram and I couldn't since I obviously messed it up with my review somehow on this one.

But there are theoretically, manual, safety manual control switches in the main control room. So you can bypass the entire main, at least you could.

And you could manually trip the scram breakers literally from the main control room, with the manual safety switches.

That's what was called out in the legend and everything in the original design of this stuff. And that part I couldn't, right now I was unable to get the rev 1 of this latest version.

Somehow that was not in the documents in my SharePoint when I went to SharePoint to get it. So I don't, are those manual safety control switches still available in the main control room?

MR. BOWMAN: Yes.

MR. BROWN: To bypass everything?

MR. BOWMAN: Yes, absolutely.

MR. BROWN: Okay.

MR. BOWMAN: That's what we would go, that's how we would take the action prior to leaving the control room.

MR. BROWN: Yes, if, that's, I guess you can argue if the fire is, somebody's got to leave the room and on the way out they turn out the lights.

That's the way I would think about it. Is that correct?

MR. BOWMAN: Yes, absolutely.

MR. BROWN: Okay.

So, there still is some manual capability with if you do it, execute it from the main control room before you evacuate?

MR. BOWMAN: Correct, and that's the way the procedure is written and that's our intended path to get to safe shutdown.

Those actions in the local are only there as a contingency. If you haven't achieved safe shutdown when you get to the remote monitoring station, then you would dispatch operators to go do that.

MR. BROWN: Yes.

CHAIR KIRCHNER: So, thank you, Doug.

MR. BROWN: I thought that might help as long as we know. Because I asked that question earlier and it didn't come out very clear back earlier in the meeting, so I apologize for that.

CHAIR KIRCHNER: I think we got the answer, Charlie.

MR. BROWN: Okay, thanks, Walt.

CHAIR KIRCHNER: Thank you. Okay, members and consultants, any further questions on the NuScale presentation for Chapter 18?

If not, let's transition to the staff. I think we can hear the staff's evaluation of Chapter 18 and then take stock and see if we'll take the lunch break, and then return for the rest of the agenda.

(Pause.)

CHAIR KIRCHNER: Okay, if you're ready to proceed, go ahead. Is this Tom Hayden speaking? Presenting?

MR. HAYDEN: Yes, that's correct.

CHAIR KIRCHNER: Yes, go ahead, Tom, if you're ready.

MR. HAYDEN: Thanks.

Yes, this is Tommy Hayden. I'll be presenting part of the Chapter 18 Human Factors Engineering staff review slides.

Reviewers for this effort from the NRC side, Amy D'Agostino, who is sitting to my left; Maurin Sheetz, who is remote and online today; Kamishan Martin, Brian Green, and then myself as the project manager for the chapter; And Getachew Tesfaye is the lead PM for the NuScale SDAA review.

All right, so these sections NuScale

covered the sections in Chapter 18. They matched the HFE, Human Factors Engineering, elements that you see in NuReg-1711 and 12 there, and then 18.0 as the overview for both in the submittal of the chapter, and then in our SDA as well.

So for an overview, NuScale submitted Chapter 18, Human Factors Engineering, Rev. 0 of the NuScale SDAA FSAR on December 31, 2022, and Revision 1 on October 31, 2023.

The NRC audit of Chapter 18 was performed from March 2023 to August 2023, generating 20 audit issues, all of which were resolved in the audit.

And 12 of which resulted in NuScale submitting supplemental information to address those questions raised during the audit.

As a result, no RAIs were issued and the staff completed their Chapter 18 review, and issued an advance safety evaluation to support today's ACRS Subcommittee meeting.

At this point, I'll turn it over to Maurin Sheetz, to discuss the significant changes in the DCA to the SDA. Maurin?

MS. SHEETZ: Thanks, Tommy. Hi, I'm Maurin Sheetz, an NRC technical reviewer for Chapter 18. I was also one of the technical reviewers on chapter 18 for the NuScale design certification.

This table summarizes significant changes in the area of human factors engineering from the design certification, to the standard design application.

So, in the area of main control room, the design certification application included a design-specific staffing requirement, in which six licensed operators operate up to 12 reactor modules from a single control room.

SDA for three operators to operate up to six modules from a single control room. And that includes the use of the NuScale control room staffing plan, which the staff approved in 2021.

The design certification application included two risk important human actions. For the SDA, no human actions met thresholds for risk significance, and none are credited in the Chapter 15 accident analysis.

During our audit, we learned that changes made to the US 460 design altered the risk significance of those two risk important human actions from the DCA, and they are no longer risk significant for the US 460 design.

The staff's review of important human

actions for HFE interfaces with the reviews of Chapters 7, 15, and 19, the latter two are still ongoing reviews.

So, the staff will verify the status of important human actions when these reviews are complete.

As we already talked at length, here the DCA included a remote shutdown station. The SCA does not have a remote shutdown station, however, as was the case for the DCA, the capability for remote shutdown exists locally at the module protection system cabinets.

And operators can monitor plant status at alternate locations on site, alternate operator work stations, which have very similar HSI design from the main control room to, compared to the main control room. Very similar.

Note that a designated facility for remote shutdown is not required by general design criteria 19, and in the event of a main control room evacuate, operators are expected to shut down each module from the main control room before evacuating.

I'll add that during our review of the Chapter 18 for the DCA, we observed an ISV scenario in which operators conducted a main control room evacuation, and shut down 12 units from the main control room before evacuating.

And we found that to be a successful path for 12 units, so the SDA is for six units, so even less to do there before evacuating.

And then, with regard to the applicant's strategy for using HFE during the design process, during the DCA, NuScale submitted what are called results summary reports for each of the HFE program elements.

Results summary reports summarize the results of using NUREG-0711, which is the HFE program review model.

So, they summarized using that model for each HFE element, include a brief description of the method used to achieve those results.

For the SDA, NuScale submitted implementation plans for five of the review elements, and results summary reports for two of the elements.

Those two are treatment of important human actions, and staffing and qualifications.

Implementation plans describe the methodology for conforming criteria, for conforming two criteria NuReg-0711, and they're submitted for work that's not complete at the time of the SDA.

However, and NuScale has committed to providing results summary reports before fuel load, excuse me.

Okay, I will talk more about that on the next slide, about the HFE strategy.

MR. BROWN: Can I ask a question?

MS. SHEETZ: Yes.

MR. BROWN: This is Charlie Brown again.

You talked about, I'm trying to go back to your statement about remote location. You can monitor all these remote like a technical support center, et cetera, et cetera, et cetera.

And I presume, it looks like from the other older diagrams, there's several locations where you can do this remote monitoring.

In the versions we saw before and I only had the non-proprietary diagram, architecture, to look at, all of the data, if you go backwards in the DCA and the earlier ones, all the data that went out to these remote monitoring stations, was transmitted via hardware-based non-unit directional-type data transmission services.

And I just wanted to confirm that that is still the methodology. Because that was back in the earlier HIPS design itself, when it was determined that we would at least that design was approved with the idea that any data going out of the main processing, main control system, MPS, everything else, would, any data, just data transmissions without requiring control functions, would be handled with unit directional-type data transmission.

Is that still the case?

MS. SHEETZ: So, this sounds like a Chapter 7 type question, but what I can say, what I do know is that these alternate operator work stations, there's no control there.

There wasn't control there for safety related systems in the DCA, and same thing, same exact situation for the SDAA.

It's a non-safety systems control in the main control room, but as far as --

(Simultaneous speaking.)

MR. TANEJA: Yes, this is Dinesh, Charlie.

MR. BROWN: Yes?

MR. TANEJA: So, there is no change to that scheme of data diode. So there's still, it's the same as what we had in the DCA.

MR. BROWN: Okay.

MR. TANEJA: So the data going out of that network, plant network, is all going through one-way data communication using a one-way.

MR. BROWN: That's fine.

MR. TANEJA: Yes, so that scheme is the same.

MR. BROWN: Okay, I was just trying to make sure since I didn't have rev 1 of the proprietary version to look at, and the other stuff was abbreviated somewhat.

MR. TANEJA: Yes.

MR. BROWN: So I went back and looked at it while we were doing this. So, I just wanted to confirm we were still in the same data transmission mode, that we were years ago.

MR. TANEJA: Correct, correct.

MR. BROWN: And your answer is yes.

MR. TANEJA: Yes, no change there.

MR. BROWN: Okay, I'm happy then. Thank you.

MR. TANEJA: Yes.

MR. BROWN: Sorry about that, Walt.

CHAIR KIRCHNER: No, that's fine. Please proceed.

MS. SHEETZ: Okay, Tommy, next slide. Okay, this is Maurin again, and thank you Dinesh, for taking that last question.

So, this slide explains the overall NuScale HFE program and their strategy. The staff conducted an in-depth review of the NuScale HFE program, as described in Chapter 18 of the SDA.

Six implementation plans, or IPs and two results in reports, or RSRs. Those were submitted to support Chapter 18 information. And we used the guidance and criteria in Revision 3 of NUREG-0711 to conduct our review.

Implementation plans describe a methodology for completing a Human Factors Engineering program element. They're submitted for work that's not complete at the time of the SDA.

To determine whether an IP is acceptable, we evaluate whether the implementation plan is complete, detailed, and verifiable. And then implementation plans must be followed by submittal of a results summary report to show that the associated activities are complete.

As I said before, the applicant has committed to submitting results summary reports before fuel load. And the staff can review them, if necessary. For example, during the verification of ITAAC closure.

Results summary reports summarize the

results of the applicant's conformance to a particular NUREG-0711 element, and include a brief description of the methodology used to derive the results.

And we reviewed two results summary reports for staffing and qualifications, and the treatment of important human actions.

Procedure development, training program development, and human performance monitoring are designated as COL items.

And then finally, the HFE inspections, tests, analysis, and acceptance criteria, or ITAAC, ensures that all remaining HFE activities are complete and produce adequate results.

So, for this, for the SDA, there are HFE ITAAC. One is a requirement for verification and validation of the main control room design, through the performance of an inspection of the as-built configuration of the main control room HSI.

And the second one is for a test called an integrative system validation, which is used to ensure that the final control room design culminating from the combined results of various HFE activities, supports the conclusion that operators can maintain plant safety.

Acceptance criteria for the ISV tests is discussed in the applicant's verification, and validation implement plan.

Together, these two HFE ITAAC confirm that the final design has incorporated human factors engineering principles, and minimizes the potential for operator error.

Next slide, please. In conclusion, while there are some differences between the DCA and the SDA, the staff found that that applicant provided sufficient information to support staff safety finding.

And we found that all applicable regulatory requirements were adequately addressed. And that's all I have for the staff's review of Chapter 18. I can take any questions.

CHAIR KIRCHNER: Members, consultants, any questions?

MEMBER BIER: Yes, this is Vicki Bier. A couple of questions. One, do you want to comment on the same question I asked the applicant on the staffing levels, and the justification for that based on test results, et cetera?

MS. SHEETZ: Could you repeat the question just so I?

MEMBER BIER: Sorry. When the applicant

was presenting earlier, I asked about the justification for the staffing level based on observed performance on tasks, et cetera.

And do you just want to comment on that, or you're satisfied with what they presented? Are there any issues that you may want to think about?

MS. SHEETZ: Well, I will as far as staffing goes, the staff, the NRC staff, this is Maurin, we reviewed the staffing plan, which is two SROs and one RO for up to, to operate up to 12 modules, as part of our review of a topical report, NuScale control room staffing plan.

So that was done in 2021, so our safety evaluation still stands there. And like I said, that's three operators for up to 12 units.

So here we are for the SDA, three operators for up to six units. That certainly falls within the scope of the applicability of that topical report.

MEMBER BIER: Okay, and kind of asking you to go beyond what the applicant is requesting now, if somebody someday wanted to operate a plant with I don't know, 24 units or whatever, I assume there would need to be a new justification and analysis presented for what the staffing levels would need to be?

Would it follow a similar process to what's been done already?

MS. SHEETZ: Yes, this is Maurin, NRC. Yes, for that hypothetical situation of how, what's the staffing needed for 24 units, that would have to go through another review.

Staffing applies, 10 CFR 50.54(m) applies to a COL. So the staff is going to look at this again when the COL comes in, and has another opportunity through the exemption process.

Because nobody can meet 50.54(m) as its written for this type of scheme of more than three units.

So, that review would be done again and the staff would most likely follow the process in NUREG, I think it's 1781 for how to process an exemption from, for staffing. How to look at alternate staffing models.

MEMBER BIER: And again, any comment on the lack of a remote shutdown station, that there's adequate ways for people to perform any tasks that are needed?

Or there's not anticipated to be any needed tasks so?

MS. SHEETZ: I'll just say, this is

Maurin. So on the topic of remote shutdown station, obviously we noticed that there was one for the DCA; there's not one for the SDA.

However, there was never a capability even in the DCA, to shut down the reactors from a remote shutdown station. It was just for monitoring.

MEMBER BIER: Got it.

MS. SHEETZ: So, GDC 19 requires the capability for remote shutdown. That has always been both for the DCA and the SDA, locally at the module protection systems.

But the expectation is operators shut down the units before they evacuate.

MEMBER BIER: Okay, thank you for the clarifications. I don't know if other people have questions or comments.

MEMBER DIMITRIJEVIC: I have one general question.

Just so I was wondering after we, the Chapter 15 and 19, the conclusion about important human actions change.

In your opinion, would that just require minimum change to this section like just the part on the important human actions, or require some additional viewing of the staffing procedures, the training, and things like that?

I mean, if this change, if important actions are going to, will be added back to it, in your opinion, is this section ready for it?

MS. SHEETZ: So, this is Maurin. That's a good question.

So, our review of the SDA is set up so we have an opportunity to alter our safety evaluation if there's any important human actions that come out of the other chapter reviews.

So, but that's like confirmatory items. So we would go back into that. If an operator action whether it's risk important or deterministic, is determined, we would expect the applicant would validate, would update their documents here.

For example, the treatment of important human actions, or at some point, show us how the operators can safely perform those important actions with their, the human system interfaces in the main control room. Or wherever they exist.

So, and we would also expect that during the integrated system validation test, that those would be tested. And they have to be tested during ISV.

So, there are other, there's other work

that would have to be done on both NuScale and the staff's, and the NRC staff's review of those.

MEMBER DIMITRIJEVIC: All right, thanks.

CHAIR KIRCHNER: Members, any further questions?

(No response.)

CHAIR KIRCHNER: Okay.

MR. BOWMAN: Oh, Walt, can you hold on for a minute?

CHAIR KIRCHNER: Yes.

MR. BOWMAN: This is Doug Bowman from NuScale. I just wanted to make one correction to a statement that I believe was made during our presentation.

And I'm not sure who it was, but somebody had mentioned that there was an uninterruptible power supply as part of MPS.

There is no uninterruptible power supply as part of MPS. It is solely powered from EDAS, those highly reliable batteries.

And so, to turn off the chassis, to get direct trip well, we have to simply open two breakers, one from two trains of EDAS and we could accomplish a trip that way.

So, just to make sure that's clear on the record. Thank you.

CHAIR KIRCHNER: Okay, thank you, Doug. Okay, this completes the chapter presentations and evaluations that were scheduled for today.

Mike, I need to confer with you. We are at the noon hour in East Coast time. The open presentation on the high impact technical items is relatively short, isn't it?

MR. SNODDERLY: One slide. Well, it's one slide but it depends on how you want to discuss it. I'm glad you brought this up. If I could just --

(Simultaneous speaking.)

MR. SNODDERLY: Okay, go ahead.

CHAIR KIRCHNER: Let me just say where I'm going. I would like to do the following. I would like to have, complete our open presentations and allow opportunity for public comment. And then take our lunch break and go to our closed session with the expectation we're not going to return to another open session, and ask the public to just stand by for several hours.

So, if we can conclude our open portion of the meeting and any deliberation by the members, and opportunity for public comment, then we could take a lunch break and go to our closed session.

MR. SNODDERLY: So, I'm on the same page but I would only add this. And I'd like us to stay in open session until after lunch because I'd like the members and you, to consider during lunch this, this, where we're at as far as next steps in the September full committee.

Right now, there are no planned presentations on this matter, meaning chapters, the review of Chapters 7, 9, 12, and 18.

CHAIR KIRCHNER: Right.

MR. SNODDERLY: Okay, based on that this was a delta review and any issues identified.

Now, if there is a need for further discussion at the September full committee, that needs to be identified now and in open session.

So, I'd like, so I guess what I'm saying is I would like some member discussion after lunch to clarify what if any, support, or what would be needed to support the September full committee meeting.

Because right now, the idea was that these letters would be fairly clean and would not require formal presentations.

Of course, the staff of NuScale will be there to support the memo writing, but no presentation.

So if that needs to change, meaning further presentations upon issues identified, then that needs to be discussed and put into place now.

So I guess what I'm saying is, think about it at lunch and if there's an issue that needs to be delved into further --

CHAIR KIRCHNER: All right, so let me outline my expectations and then if any member has additional requirements. My expectation is we will have the chapter write-ups from each lead member for our P&P session in September. Is that correct, Mike?

MR. SNODDERLY: Yes, sir, that's the current plan unless.

CHAIR KIRCHNER: Right, so, members and consultants, if that, if you feel the need, you can think about it over our lunch break, for further presentations, we need to make that decision and recommendation coming out of this open session, and then go from there to the closed session.

MR. MOORE: That's correct, Walt. This is Scott.

CHAIR KIRCHNER: Yes. So, I'll ask -- I'll have to speak on behalf of Dave Heddy, who is not with us today, unfortunately. But for the other three leads, do you see at this juncture, a need for further information or presentations from the applicant, and/or the staff for your particular chapters?

I'll start with you, Tom.

MEMBER ROBERTS: Yes, I think I'd echo something that Vesna said, which is I think there's a potential for some of the lay reviews to influence this write up.

And so, whatever we present or have available in two weeks, I assume the intent is that to be the current status that could change when for example, Chapters 15 and 19 get reviewed.

CHAIR KIRCHNER: Exactly. We always have that, that challenge when we do a serial review, and we don't have some of the more important chapters still to come, in particular 15. And we haven't had 4, 5, or 6, which are key chapters.

So, if you, my recommendation for our letters is that if there's an issue that you think may be impacted by a downstream review, be it Chapter 15 or the other core chapters, flag that not necessarily as a concern, but just as an item to pick up in a subsequent review.

That would be my recommendation for the chapter leads, in terms of their letter report at this juncture.

MEMBER ROBERTS: Okay, that, yes.

CHAIR KIRCHNER: We had the same challenge when we did this for the DCA, and did chapter write-ups. We did them serially. Chapter 15, Chapter 19, they came much later in the process.

So, we, if you will, we left pointers in our letters to potentially matters that should be looked at in conjunction with that subsequent review.

MEMBER ROBERTS: Okay, good. So with that caveat, I don't see a need for anymore presentations in September.

CHAIR KIRCHNER: Matt, from your perspective?

MEMBER SUNSERI: So, Walt, the only thing I see open right now is our understanding of the impacts of this ultimate heat sink water level, on the integrity of the containment vessel.

I think we're going to talk about that in the closed session, and I anticipate that we'll have a better understanding then.

The applicant appears to be convinced that it's okay, and they just are unable to talk about it because of the proprietary information.

So, outside of that issue, I don't see anything else that we will want presented at the full committee meeting, and I'll have a memo prepared to discuss for that meeting.

MEMBER DIMITRIJEVIC: Matt?

MEMBER SUNSERI: Yes, Vesna?

MEMBER DIMITRIJEVIC: Matt, you know in this charging thing, in this flow reduction, things like that can change in this containment is a relation.

And maybe that could be part of the, something which we will learn more later what's happening.

MEMBER SUNSERI: Yes, and that's a good point, Vesna.

I didn't bring that up because they're talking, I guess my understanding is, is that's an operational matter, not a system configuration issue at this point.

So we're going to talk about it in chapter, some other Chapter 15, I think is where that's probably going to be --

(Simultaneous speaking.)

CHAIR KIRCHNER: Well, I think the hardware aspects will be addressed in Chapter 6.

MEMBER SUNSERI: 6.

CHAIR KIRCHNER: And then, the safety

analysis will be addressed in 15. And that should be reflected, Vesna, I think in Chapter 19, as well.

MEMBER DIMITRIJEVIC: Right, right. But I wasn't sure what part, will any part go in Chapter 9 versus Chapter 6.

Yes, I'm just bringing this up because it was introduced in those flow reductions was introduced as a part of Chapter 9.

MEMBER SUNSERI: I think it's no doubt that's something that we want to know more about. It's just not related to Chapter 9 is all I'm saying, yes.

MEMBER DIMITRIJEVIC: Okay.

CHAIR KIRCHNER: Okay.

I addressed what was the major issue that we had with Chapter 12. It was the combustible gas monitoring. So, I don't want to repeat what the statement I made earlier.

And then, Vicki, on Chapter 18, what's your position?

MEMBER BIER: Yes, I don't see a need for further presentations in September. Seems pretty straightforward at this point.

CHAIR KIRCHNER: Okay.

So then, my expectation is we might would then have letters that have pointers in them, if there are concerns, or open matters.

But we would have those letters for the September full committee meeting.

MR. SNODDERLY: Thank you, Walt, that was very helpful and I think the staff at NuScale will appreciate that, too.

CHAIR KIRCHNER: Yes.

MR. SNODDERLY: Just to know what the expectations are for September. And thank you for having this discussion.

I still would suggest that, well, now the question is, do you want to go through the high impact technical issue open session, or do we take a break and maybe let the members think a little bit more about everything they took from this morning?

CHAIR KIRCHNER: Yes, let's take a break. We've gone for over two hours at this point. We'll take perhaps a shortened lunch break.

Is reconvening at 1300, at 1:00 o'clock Eastern Time acceptable?

MR. SNODDERLY: Yes, I think that would be, and we'll be in open session. We'll let NuScale do their open presentation on the HITIs, and then take public comment.

CHAIR KIRCHNER: Right.

MR. SNODDERLY: Close the session and go into closed session, and not go back into open session. Is that?

CHAIR KIRCHNER: Excellent, yes. That's the plan.

Okay, so with that, we will recess until 1:00 o'clock Eastern Time. And I thank all the presenters, and I once again thank the NuScale people for such an early start.

Thank you for your presentations. Thank you also to the staff.

We are in recess until 1:00 o'clock Eastern Time.

(Whereupon, the above-entitled matter went off the record at 12:13 p.m. and resumed at 1:01 p.m.)

CHAIR KIRCHNER: Okay, we are back in session. This is a meeting of the NuScale Subcommittee, and we are going to turn to NuScale for a presentation and discussion on their high impact technical issues.

And I'll turn to Thomas Griffith.

MR. GRIFFITH: Thank you, I'm Thomas Griffith, Licensing Manager at NuScale. Pleased to have the opportunity to provide an update on the US-

460 standard design approval application, high impact technical issues.

You may hear me use the phrase HITI. That's the shorthand way that we've been referencing the high impact technical issues, just for some clarity.

I do have with me Kris Cummings, do you want to introduce yourself?

MR. CUMMINGS: Yes, Kris Cummings, licensing engineer for NuScale. I've been with NuScale for about four and a half years now. Prior to that, I've had several, I've had roles with Westinghouse, Holtec, and NEI. My focus is generally nuclear-related stuff, but they seem to come to me with it whenever they get stuck on things.

So, I'm here to support Tom.

MR. GRIFFITH: Thanks, Kris.

Next slide, please. Appreciate the award from the Department of Energy, and appreciate their support in helping with NuScale with our mission.

Next slide, please. So, the HITIs are identified as specific topic areas that NRC and NuScale management have agreed, require elevated management attention.

The use of the term HITI does not imply a shortfall in the application, rather that there is a need to prioritize the resolution of the issue to support overall review schedule.

The HITI list in NS's opinion, is close to resolution because the NRC has provided NuScale audit items and RAIs, related to the HITIs.

Regulatory basis and acceptance criteria have been established, and due dates for final products to provide during the review, is largely understood.

To date, 10 high impact technical issues have been established. Three were considered resolved at the least quarterly management meeting between NuScale, and the NRC. That took place at the end of July.

Due to the hard work of the NRC and NuScale, two additional HITIs, HITI-1, the design and classification of the augmented DC power system, as well as HITI-3, are now considered resolved, as well.

I would like to defer to the closed session for a detailed discussion of each of the HITIs, as much of the detail could be considered proprietary.

CHAIR KIRCHNER: Okay, thank you, Thomas.

At this point, I think members and

consultants, we can hold detailed questions. Is there anything, Thomas, that you can say?

You indicated you have now five of these resolved. Is there anything that you want to say on NuScale's behalf about items 4, 5, and 9 in an open session?

MR. GRIFFITH: I think for items 4, 5, and 9, they're all related to material changes that NuScale has implemented with the containment vessel, the RPV vessel, respectively.

And item number 9 is a result of the upper and lower RVB material being different. The NRC as a result of some of the quarterly management meetings that we had, identified the need to review some of the detailed calculations from NuScale regarding the shear loading of results from differential thermal expansion.

And to that end, we believe that items 4, 5, and 9 are largely understood, and we do anticipate being able to provide detailed presentations on each of those items when we get to the chapters for ACRS.

CHAIR KIRCHNER: Okay. And there are no open ASME code issues with 4 and 5?

MR. GRIFFITH: Not that I'm aware of.

CHAIR KIRCHNER: Yes, just wanted to put

that on the record.

Okay, all right with that then, I think you only had one slide, Thomas, is that correct?

MR. GRIFFITH: That is correct.

CHAIR KIRCHNER: Just the list, yes.

So, Mike at this point, I think we can turn to the public and ask for any comments from the public, and proceed to that.

MR. SNODDERLY: Yes, Walt.

So, are there any members of the public that would like to make a comment on today's meeting?

MEMBER MARTIN: There appears to be one. Tim Polich, go ahead and unmute yourself and make your comment.

MR. POLICH: Yes, can you hear me?

MEMBER MARTIN: Yes.

MR. POLICH: Yes, I was encouraged to hear that one of the carve-outs was closed today. Still concerned with the other, the carve-outs that were left from the last approved design.

And I see a lot of work that looks like it needs to be done, because I've been monitoring not only the ADAMS, but also the SDA review dashboard.

I see responses seem to come in slowly. The average response time seems to be rather high. It appear the REIs, they seem to either need multiple responses or something because they go from awaiting response to reviewing, back to awaiting response.

And I'm just concerned that maybe there needs to be some more testing that needs to get done at this, and that just seems a little late in the game to be doing that.

But I'm still concerned, and it's probably some of the same concerns that I believe it's Dr. Dimitrijevic had in 2020 about the steam generators, and still haven't seen anything that resolves there.

So, that's my comment. Thank you.

MEMBER MARTIN: Okay, are there any others?

Hearing none, or seeing no hand raised, Walt, I'll return it to you to close.

CHAIR KIRCHNER: Okay, thank you for monitoring that, Bob.

Thank you for the public comment, Tim, and with that, we are at a juncture where we can adjourn the open session and for those people authorized to attend, go to our Teams link for the closed session.

So once again, I want to thank NuScale for agreeing to such an early start, and I thank you for your presentations as well. And thank you to the staff, likewise.

So with that, we are adjourned and for those who have permission, please go to the Teams link for the closed meeting. Thank you.

(Whereupon, the above-entitled matter went off the record at 1:09 p.m.)

LO-172559

August 19, 2024 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) US460 Standard Design Approval Application Chapters 7, 9, 12, and 18," PM-172558, 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 August 22, 2024. The materials support NuScale's presentation of the subject chapters of the US460 Standard Design Approval Application.

The enclosure to this letter is the nonproprietary presentation entitled "ACRS Subcommittee Meeting (Open Session) US460 Standard Design Approval Application Chapters 7, 9, 12, and 18," PM-172558, Revision 0.

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

If you have any questions, please contact Chelsea Lockwood at 541-452-7171 or at clockwood@nuscalepower.com.

Sincerely,

Thomas Griffith
Manager, Licensing
NuScale Power, LLC

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

Enclosure 1: "ACRS Subcommittee Meeting (Open Session) US460 Standard Design Approval Application Chapters 7, 9, 12, and 18," PM-172558, Revision 0

Enclosure 1:

"ACRS Subcommittee Meeting (Open Session) US460 Standard Design Approval Application Chapters 7, 9, 12, and 18," PM-172558, Revision 0

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ACRS Subcommittee Meeting

(Open Session)

August 22, 2024

US460 Standard Design Approval Application

Chapters 7, 9, 12, and 18

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Chapter 7

Instrumentation and Controls

August 22, 2024

Presenter: Thomas Case

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Instrumentation and Controls

- Section 7.0: Instrumentation and Controls – Introduction and Overview
- Section 7.1: Fundamental Design Principles
- Section 7.2: System Features

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Section 7.0: Instrumentation and Controls – Introduction and Overview

- Changes from DCA
  - Elimination of the remote shutdown station (across all of Chapter 7)
    - Alternate operator workstations allow for plant monitoring outside the main control room

- Results from audit and RAI review
  - No audit items or RAIs specific to Section 7.0 [1]

[1] One audit item in Chapter 15 related to COL Item 7.0-1

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Section 7.1: Fundamental Design Principles

- Changes from DCA
  - Module protection system (MPS) setpoint changes due to changes in operating pressure and temperature, and updated safety analysis
  - Emergency core cooling system (ECCS) actuation changes due to updated safety analysis
  - Additional decay heat removal system (DHRS) and reactor trip system (RTS) actuations due to updated safety analysis
  - Addition of an 8-hour timer for ECCS actuation to add supplemental boron if needed to maintain subcriticality
  - Adoption of Institute of Electrical and Electronics Engineers (IEEE) Standard 497-2016 as endorsed by Regulatory Guide 1.97, Revision 5, and the addition of Type F post-accident monitoring variables

- Results from audit and RAI review
  - Removal of a note from Figure 7.1-1aa regarding inadvertent actuation block (A-7.1-1)

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Section 7.2: System Features

- Changes from DCA
  - Information from the Advanced Sensor Technical Report cited in the DCA is incorporated into SDAA Section 7.2.16
  - Change from digital to analog sensors
    - Reactor pressure vessel riser level, containment vessel water level, reactor coolant system pressure, pressurizer pressure
    - Diversity and defense-in-depth (D3) and coping analyses updated in Section 7.1
  - Reduction in quantity of reactor coolant system temperature sensors based on updated engineering evaluation
    - Analysis of the reactor coolant flow determined that streaming effects do not require the use of multiple sensors per quadrant

- Results from audit and RAI review
  - No audit items or RAIs specific to Section 7.2

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SDAA Topical Report – Design of the Highly Integrated Protection System Platform TR-1015-18653-P-A Revision 2

- NRC approved Topical Report (June 2017)
- No changes since ACRS engagement in 2023

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SDAA Technical Report – NuScale Instrument Setpoint Methodology Technical Report TR-122844 Revision 0

- NRC reviewed as part of Chapter 7
- No changes since ACRS engagement in 2023

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COL items

- No change to COL Items in Chapter 7 [1]

[1] One audit item in Chapter 15 related to COL Item 7.0-1

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Acronyms

ACRS	Advisory Committee on Reactor Safeguards
COL	Combined License
D3	Diversity and Defense-in-depth
DCA	Design Certification Application
DHRS	Decay Heat Removal System
ECCS	Emergency Core Cooling System
IEEE	Institute of Electrical and Electronics Engineers
MPS	Module Protection System
NRC	Nuclear Regulatory Commission
RAI	Request for Additional Information
RTS	Reactor Trip System

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Chapter 9

Auxiliary Systems

August 22, 2024

Presenters: Sarah Turmero, Freeda Ahmed, and Jordan Green

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Chapter 9: Auxiliary Systems

- Section 9.1 Fuel Storage and Handling
- Section 9.2 Water Systems
- Section 9.3 Process Auxiliaries
- Section 9.4 Air Conditioning, Heating, Cooling, and Ventilation Systems
- Section 9.5 Other Auxiliary Systems (Lighting, Communication, and Fire Protection)
- Section 9A Fire Hazards Analysis

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Section 9.1: Fuel Storage and Handling

- Section 9.1.1 Criticality Safety of Fresh and Spent Fuel Storage and Handling
- Section 9.1.2 New and Spent Fuel Storage
- Section 9.1.3 Pool Cooling and Cleanup System
- Section 9.1.4 Fuel Handling Equipment
- Section 9.1.5 Overhead Heavy Load Handling Systems

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Section 9.1.1 Criticality Safety of Fresh and Spent Fuel Storage and Handling

- Criticality safety of fuel storage is addressed by COL Item 9.1-1.
- COL Item 9.1-1
  - An applicant that references the NuScale Power Plant US460 standard design will develop plant programs and procedures for safe operations during handling and storage of new and spent fuel assemblies, including criticality control.
- Audit and RAI Results:
  - Design of reactor flange tool is responsibility of COL applicant. Fuel remains in the lower reactor pressure vessel (RPV). Criticality of fuel while in the RPV is discussed in Section 4.3.2.6. (A-9.1.1-1)
  - COL Item 9.1-2 requires applicants to perform criticality analysis of fuel racks (A-9.1.1-2)
  - Criticality safety design for refueling pool is described in Section 9.2.5 (A-9.1.1-3)

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Section 9.1.2 New and Spent Fuel Storage

- Fuel storage is addressed by COL Item 9.1-2.
- COL Item 9.1-2
  - An applicant that references the NuScale Power Plant US460 standard design will provide the design of the spent fuel pool storage racks, including the structural dynamic and stress analyses, thermal hydraulic cooling analyses, criticality safety analysis, and material compatibility evaluation.
- Audit and RAI Results:
  - Demonstrated that the spent fuel pool (SFP) has >30 days of water above the top of fuel. (RAI 9.1.2-1.1)
  - Clarified the seismic classification between dry dock gate and the dry dock gate support in Section 9.1.2 and Section 9.1.3 (RAI 9.1.2-1.2)

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Section 9.1.3 Pool Cooling and Cleanup System

- Design changes from DCA:
  - Combined the spent fuel pool cooling system, the reactor pool cooling system, the pool cleanup system, and the pool surge control system into a single pool cooling and cleanup system.
    - Major components remain the same (filters, demineralizers, surge control tank) with the exception of the pumps and heat exchangers, which have been reduced from five trains to three.
  - Pool leakage detection system wall leak channels attach to the steel-plate composite walls.
- Audit and RAI Results:
  - None specific to Section 9.1.3.

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Section 9.1.4 Fuel Handling System

- Design changes from DCA:
  - New fuel elevator capable of handling irradiated fuel for inspection purposes
  - New fuel jib crane classification changed from ASME NUM-1 Type 2 to ASME NUM-1 Type 1A, single failure proof
- Audit and RAI Results:
  - None specific to Section 9.1.4

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Section 9.1.5 Overhead Heavy Load Handling Systems

- Design Changes from DCA:
    - Reactor Building crane (RBC) capacity increased from 850 tons to 950 tons
    - Module lifting adapter in DCA design was removed and is now integral to the RBC
    - RBC auxiliary hoist capacity increased from 15 tons to 40 tons
    - Changes to automated control system software reduces probability of operator error
    - Added additional jib cranes designed to ASME NUM-1 Type 1A
    - Removal of heavy load exclusion zone above the SFP
- Change to COL Item 9.1-5
    - An applicant that references the NuScale Power Plant US460 standard design will provide a description of the program governing heavy loads handling. The program should address
        - operating and maintenance procedures.
        - inspection and test plans.
        - personnel qualification and operator training.
        - detailed description of the **safe load paths** for movement of heavy loads.

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Section 9.1.5 Overhead Heavy Load Handling Systems, continued

- Audit and RAI Results:
  - Clarifies use of ASME NUM-1 within Section 9.1.5 including demonstrating compliance with Regulatory Guide 1.244 Position C.1 (A-9.1.5-1, A-9.1.5-2, A-9.1.5-8)
  - Justifies deviation of ASME NOG-1 design factor for plate buckling and the methodology for determining spacing of transverse stiffeners (RAI 9.1.5-3)
  - Eliminates heavy load exclusion zone terminology in 9.1.5, 15.7.5, and 17.4 (RAI 9.1.5-6)

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Section 9.2: Water Systems

- Section 9.2.1 Station Service Water System - Not applicable to US460
- Section 9.2.2 Reactor Component Cooling Water System
- Section 9.2.3 Demineralized Water System
- **Section 9.2.4 Potable and Sanitary Water Systems**
- **Section 9.2.5 Ultimate Heat Sink**
- Section 9.2.6 Condensate Storage Facilities
- **Section 9.2.7 Site Cooling Water System**
- Section 9.2.8 Chilled Water System
- **Section 9.2.9 Utility Water Systems**

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Section 9.2.4 Potable and Sanitary Water Systems

- Changes from DCA
  - Potable and Sanitary Water System piping (including loop seals) penetrating the control room envelope changed from Seismic Category II (SC-II) to Seismic Category I (SC-I).

- Audit and RAI Results:
  - Removal of COL Item 9.2-2 concerning source and pre-treatment methods of potable water
  - Removal of COL Item 9.2-3 concerning sanitary waste storage and disposal
  - The potable and sanitary water systems serve no safety-related functions, are not credited for mitigation of design-basis accidents, and have no safe shutdown functions. Site-specific characteristics do not impact ability to meet the identified requirements (A-9.2.4-1)

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Section 9.2.5 Ultimate Heat Sink (UHS)

- Changes from DCA
  - The number of modules was reduced from 12 to six, reducing the inventory of the UHS due to a smaller footprint.
  - UHS level lowered from 68 ft to 53 ft from bottom of module.

- No Audit questions or RAIs specific to Section 9.2.5

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Section 9.2.7 Site Cooling Water System

- Changes from DCA
  - Changed from one-loop open system to a two-loop system in SDAA to better maintain water quality for plant users
    - The two-loop system consists of a closed loop that removes heat from plant loads and an open cooling tower loop that rejects heat to the environment.
  - Removed COL Item 9.2-4 concerning long-term corrosion and fouling
    - The site cooling water system serves no safety-related functions, is not credited for mitigation of design-basis accidents and has no safe shutdown functions. Site-specific characteristics do not impact ability to meet the identified requirements.

- Audit and RAI Results:
  - Discussed that utility water provides makeup to the tower basin and demineralized water provides makeup to the closed loop (A-9.2.7-1)

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Section 9.2.9 Utility Water Systems (UWS)

- Changes from DCA
  - Removed COL Item 9.2-5 concerning identification of a site-specific water source and water treatment system
    - This item was written for a previous revision of Regulatory Guide 1.206, which does not apply to the SDAA. The UWS meets GDCs 5, 60, 64 and 10 CFR 20.1406.

- Audit and RAI Results:
  - The UWS provides raw water to the demineralized water system, site cooling water and fire protection for general washdown use. The system function determines the system chemistry controls. The selected source of raw water has no impact on the safety-related structures systems and components (SSC) (A-9.2.9-1)
  - Discussed UWS piping in the vicinity of safety-related or SC-I SSC; and protective measures to avoid impact on system from flooding (A-9.2.9-2)

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Section 9.3: Process Auxiliaries

- Section 9.3.1 Compressed Air Systems
- Section 9.3.2 Process Sampling System
- Section 9.3.3 Equipment and Floor Drain Systems
- **Section 9.3.4 Chemical and Volume Control System**
- Section 9.3.5 Standby Liquid Control System
- **Section 9.3.6 Containment Evacuation System**
- Section 9.3.7 Containment Flooding and Drain System

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Section 9.3.4 Chemical and Volume Control System (CVCS)

- Design changes from DCA:
  - Module heatup system modified to use an electric heater in lieu of an auxiliary boiler

- Audit and RAI Results:
  - Clarified how CVCS complies with General Design Criterion (GDC) 4 (A-9.3.4-1)
  - Revised demineralized water system isolation valves to Quality Group C (RAI 9.3.4-1)
  - Added the maximum boron concentration for the boron addition system to prevent boric acid precipitation (RAI 9.3.4-2)
  - Provided additional design information regarding flow-restricting venturis to support probabilistic risk assessment (RAI 9.3.4-3)

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Section 9.3.6 Containment Evacuation System (CES)

- Design changes from DCA:
  - The CES inlet pressure instrumentation and connecting piping, up to and including isolation valves, are designed to SC-I standards (SC-III in DCA), which ensures these components maintain capability to perform their function during and after a safe shutdown earthquake.
  - The same section of piping was increased to containment design pressure (Table 19.1-3) to act as a diverse independent backup to the CIVs in support of probabilistic risk assessment. (RAI 19.1-52)

- Audit and RAI Results:
  - Containment pressure correlates to a reactor coolant system leak rate (A-9.3.6-1)
  - Correction factor to account for water vapor bypass is calculated (A-9.3.6-2)
  - Vacuum pump removal of water vapor resulting from leaks inside containment (A-9.3.6-3)

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Section 9.4: Air Conditioning, Heating, Cooling, and Ventilation Systems

- Section 9.4.1 Control Room Area Ventilation System
- Section 9.4.2 Reactor Building and Spent Fuel Pool Area Ventilation System
- Section 9.4.3 Radioactive Waste Building Ventilation System
- Section 9.4.4 Turbine Building Ventilation System
- Section 9.4.5 Engineered Safety Feature Ventilation System

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Section 9.5: Other Auxiliary Systems

- **Section 9.5.3 Lighting Systems**
- **Section 9.5.2 Communication Systems**
- **Section 9.5.1 Fire Protection Program**

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Section 9.5.3 Lighting Systems

- Changes from the DCA
  - Main control room (MCR) has dedicated emergency lighting that is continuously on

- Audit and RAI Results:
  - Clarified illumination levels for normal and emergency lighting (RAI 9.5.3-1)
  - Explained how manual fire suppression would be handled with emergency lighting (A-9.5.1-2)
  - Clarified illumination levels outside MCR (RAI 9.5.3-3)

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Section 9.5.2 Communication System

- Changes from the DCA
  - Sound-powered telephone system was removed
  - Health physics network added to the communication system
  - Removed COL Item 9.5-2 concerning the location of security power equipment within a vital area
    - Now part of the standard plant design

- No Audit items or RAIs specific to Section 9.5.2

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Section 9.5.1 Fire Protection Program

- Changes from DCA
  - Building layout change

- Audit and RAI Results:
  - Confirmed containment cable design attributes (A-9.5.1.2.4-1)
  - Explained that fixed emergency lighting is not required for post-fire safe shutdown functions or alternative safe shutdown functions (A-9.5.1-2 & A-9.5.3-1)
  - Discussed structural and electrical raceway fire barrier requirements (A-9.5.1-1)

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Section 9A: Fire Hazards Analysis

- Design changes from DCA
  - Building layout change

- Audit and RAI Results:
  - Clarified safe shutdown requirements vs capabilities concerning MCR evacuation (A-9A.6.4.1-1 & -2)
  - Clarified the fire hazards analysis to state which rooms did not contain safe shutdown equipment and discussed how propagation is mitigated (RAI 9A.5-1)

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Acronyms

ASME	American Society of Mechanical Engineers
CES	Containment Evacuation System
CIV	Containment Isolation Valve
COL	Combined License
CVCS	Chemical and Volume Control System
DCA	Design Certification Application
GDC	General Design Criterion
MCR	Main Control Room
RAI	Request for Additional Information
RPV	Reactor Pressure Vessel
RBC	Reactor Building Crane
SFP	Spent Fuel Pool
SC-I	Seismic Category I
SC-II	Seismic Category II
SSC	Structures, Systems, and Components
SDAA	Standard Design Approval Application
UHS	Ultimate Heat Sink
UWS	Utility Water System

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Chapter 12

Radiation Protection

August 22, 2024

Presenter: Erik Slobe

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Chapter 12: Radiation Protection

- Section 12.1 Ensuring that Occupational Radiation Exposures Are as Low as Reasonably Achievable
- Section 12.2 Radiation Sources
- Section 12.3 Radiation Protection Design Features
- Section 12.4 Dose Assessment
- Section 12.5 Operational Radiation Protection Program

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Section 12.1 Ensuring that Occupational Radiation Exposures Are as Low as Reasonably Achievable

- Same methodology as the Design Certification Application (DCA)

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Section 12.2 Radiation Sources

- Same methodology as DCA
- Updated source term information in Tables 12.2-1 through Table 12.2-31
  - Updated for change in cycle length, increase in burnup rate, change in thermal power, and change in number of NuScale Power Modules
  - Design basis failed fuel fraction is applied to one reactor for shared system source terms (11.1).
- Audit results
  - Dose rate for workers on the fuel handling machine (A-12.2.1.8-1)
  - Decay of N-16 to insignificant levels in the chemical volume control system flow path (A-12.2-5)
  - Source terms for components of the low conductivity waste (LCW) processing skid, including LCW filters, ion exchanger, accumulators, and LCW polishers (A-12.2-3)

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Section 12.3 Radiation Protection Design Features

- Same methodology as DCA
- Differences from DCA
  - No very high radiation areas
  - Reduction in number of reactor building and radwaste building shield doors
  - Reduction in fixed radiation monitors
  - Removal of COL Item 12.3-5 on additional area radiation monitors
    - Design criteria for area radiation monitors are included in Section 12.3.4.2
  - Removal of COL Item 12.3-8 on radiation shielding for shield wall penetrations
    - Completion of more detailed shielding analyses
- Audit Results
  - Break pot tanks in phase separator tank and spent resin storage tank vent lines replaced with hooded vents (A-12.3.1.1-2)
- RAI Results
  - Shielding based on nominal concrete equivalent gamma attenuation (RAI 12.3-1)
  - Radiation monitors under the bioshield are post accident monitoring system B, C, and F variables (RAI 12.3.4.2-1)

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Section 12.4 Dose Assessment

- Same methodology as DCA
- Changes from DCA
  - Updated for change in cycle length, increase in burnup rate, change in thermal power, change in number of Nuscale Power Modules, building layout changes, and operational optimizations
  - Vital areas for post-accident actions do not include areas for initiating combustible gas monitoring
- Audit Results
  - An update to COL Item 12.4-1 includes changing the dose to construction workers from co-located existing operating NuScale Power Plants to a responsibility of the applicant. (A-12.4.1.9-1)

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Section 12.5 Operational Radiation Protection Program

- No changes from DCA

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Acronyms

COL	Combined License
DCA	Design Certification Application
LCW	Low Conductivity Waste
NRC	Nuclear Regulatory Commission
RAI	Request for Additional Information

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Chapter 18

Human Factors Engineering

August 22, 2024

Presenter: Doug Bowman

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Introduction

- Chapter 18 Overview
  - Section 18.1 – Human Factor Engineering (HFE) Program Management
  - Section 18.2 – Operating Experience Review (OER)
  - **Section 18.3 – Functional Requirements Analysis and Function Allocation (FRA/FA)**
  - **Section 18.4 – Task Analysis (TA)**
  - **Section 18.5 – Staffing and Qualifications (S&Q)**
  - **Section 18.6 – Treatment of Important Human Actions (TIHAs)**
  - **Section 18.7 – Human-System Interface (HSI) Design**
  - Section 18.8 – Procedure Development
  - Section 18.9 – Training Program Development
  - Section 18.10 – Human Factors Verification and Validation (V&V)
  - Section 18.11 – Design Implementation (DI)
  - Section 18.12 – Human Performance Monitoring
- There are not slides for the areas that have not changed
- Other Items

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Comparison of HFE Program for US600 DCA and US460 SDAA

Program Element	US600 DCA	US460 SDAA
Operating Experience Review	RSR Submitted	IP Submitted
Functional Requirements Analysis and Function Allocation	RSR Submitted	IP Submitted
Task Analysis	RSR Submitted	IP Submitted
Staffing and Qualifications	RSR Submitted	RSR Submitted
Treatment of Important Human Actions	RSR Submitted	RSR Submitted
Human-System Interface Design	RSR Submitted	IP Submitted
Procedure Development	COL Activity	COL Activity
Training Program Development	COL Activity	COL Activity
Verification and Validation	IP Submitted/RSR Submitted	IP Submitted
Design Implementation	COL Activity	IP Submitted
Human Performance Monitoring	COL Activity	COL Activity

- Implementation plan (IP) describes methodology
- Results summary report (RSR) describes methodology and results
- NUREG-0711 allows for submittal of an IP or RSR
- DCA: submitted RSRs for all HFE elements that are predecessors to V&V
- SDAA: NuScale is following the traditional model for HFE

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Section 18.3 – Functional Requirements Analysis and Function Allocation

- The purpose of this element is to verify those functions needed to satisfy the plant's safety and commercial goals, and the assignment of those functions to personnel and automation, takes advantage of human and machine strengths and avoids human and machine limitations
- **As the FRA/FA process has matured, a single, combined and interlinked database has been developed that aligns the HFE task analysis, FRA/FA database, the Operator Training Task Analysis and the Plant Operating Procedure set**

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Section 18.4 – Task Analysis

- TA identifies specific tasks (human actions) required to satisfy the functions from the FRA/FA element
- Similar process to US600 DCA
    - Now combined in a single, interlinked database as described in Section 18.3

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Section 18.5 – Staffing and Qualifications

- S&Q determines the number and qualification of licensed operators required for safe and reliable plant operation
  - **Minimum staffing requirement is one licensed reactor operator and two licensed senior reactor operators**
- Changes from the DCA:
  - For the DCA: Minimum staffing requirements are located in the DC rule (Part 52 App. G)
  - **For the SDAA: Technical basis and approach for minimum staffing requirements is approved topical report → NuScale Control Room Staffing Plan, TR-0420-69456-NP-A**
  - Previously reviewed by ACRS (ML21139A226 and ML21139A232 [April 2021])

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Section 18.6 – Treatment of Important Human Actions

- Identification of IHAs within the scope of Chapters 7,15, and 19
- Probabilistic Risk Assessment (Chapter 19) determines risk-important human actions
- Deterministic human actions are those credited in Chapter 15 and D3 (diversity and defense-in-depth) coping analyses of Chapter 7 (e.g., those required for long-term decay heat removal or reactivity control)
- US460 Standard Design does not have IHAs
- Changes from the DCA:
    - DCA included two IHAs → No longer RIHAs in the SDAA (see Chapter 19)
    - **The DCA IHAs are still addressed and mitigating strategies accounted for in the current SDAA generic technical guidelines, HSI design, HFE and training task analysis**

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Section 18.7 – HSI Design

- HSI design element establishes the HSI design
- Substantially similar main control room and HSI as the US600
  - **US460 changes are in response to ISV and design changes from the US600**

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Other Items

New HFE-related ITAAC:

02.	The MCR design incorporates HFE principles that reduce the potential for operator error.	An integrated system validation (ISV) test is performed in accordance with the Verification and Validation Implementation Plan.	A report exists and concludes that acceptance criteria associated with each ISV test scenario are satisfied upon initial performance of the scenarios or upon remediation of failures.

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SDAA Audit

- 20 audit items successfully resolved
- Included virtual demonstration of simulator and staff review of databases

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Acronyms

ACRS	Advisory Committee on Reactor Safeguards
COL	Combined License
D3	Diversity and Defense-in-depth
DCA	Design Certification Application
DI	Design Implementation
FRA/FA	Functional Requirements Analysis and Function Allocation
HSI	Human-System Interface
IHA	Important Human Action
IP	Implementation Plan
ISV	Integrated System Validation
MCR	Main Control Room
OER	Operating Experience Review
RIHA	Risk Important Human Action
RSR	Results Summary Report
S&Q	Staffing and Qualifications
SDAA	Standard Design Approval Application
TA	Task Analysis
TIHA	Treatment of Important Human Actions
V&V	Verification and Validation

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Presentation to the Advisory Committee on Reactor Safeguards Subcommittee Staff Review of NuScale's US460 Standard Design Approval Application Final Safety Analysis Report, Revision 1

Chapters 7, 9, 12, and 18

August 22nd, 2024

(Open Session)

Non-Proprietary

Acronyms and Definitions

- ACRS – Advisory Committee on Reactor Safeguards
- APLB – PRA Licensing Branch B
- APLC – PRA Licensing Branch C
- ARCB – Radiation Protection and Consequences Branch
- CFR – Code of Federal Regulations
- COL – Combined License
- COLA – Combined License Application
- DC – Design Certification (refers to NuScale US600 design)
- DCA – Design Certification Application (refers to NuScale US600 design)
- DEX – Division of Engineering and External Hazards
- DNRL – Division of New and Renewed Licenses
- DRA – Division of Risk Assessment
- DSS – Division of Safety Systems
- EEEB – Electrical Engineering Branch
- EICB – Instrumentation and Controls Branch
- ELTB – Long Term Operations and Modernization Branch
- EMIB – Mechanical Engineering and Inservice Testing Branch
- ESEB – Structural Civil Geotech Engineering Branch
- FSAR – Final Safety Analysis Report
- GDC – General Design Criteria
- NCSG – Corrosion and Steam Generator Branch
- NLIB – Licensing and Regulatory Infrastructure Branch
- NPM – NuScale Power Module
- NRLB – New Reactor Licensing Branch
- NRR – Office of Nuclear Reactor Regulation
- PRA – Probabilistic Risk Assessment
- SCPB – Containment and Plant Systems Branch
- SDA – Standard Design Approval (refers to NuScale US460 design)
- SDAA – Standard Design Approval Application (refers to NuScale US460 design)
- SNRB – Nuclear Methods Systems and New Reactors Branch

Presentation to the ACRS Subcommittee
Staff Review of NuScale SDAA FSAR, Revision 1

Chapter 7, “Instrumentation and Controls"

August 22, 2024
(Open Session)

Non-Proprietary

NuScale SDAA FSAR Chapter 7 Review

Overview

- NuScale submitted Chapter 7, "Instrumentation and Controls," Revision 0 of the SDAA FSAR on December 31, 2022, and Revision 1 on October 31, 2023
- NRC regulatory audit of Chapter 7 was performed from March 2023 to August 2023
- One audit issue was issued and resolved in the audit
- The audit issue resulted in NuScale submitting supplemental information to address questions raised during the audit
- No RAIs issued
- Staff completed Chapter 7 review and issued an advanced safety evaluation to support today's ACRS Subcommittee meeting

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NuScale SDAA FSAR Chapter 7 Review

Contributors

- **Technical Reviewers**
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  - Dinesh Taneja, NRR/DEX/ELTB
- **Project Managers**
  - Ricky Vivanco, PM, NRR/DNRL/NRLB
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NuScale SDAA FSAR Chapter 7 Review

Sections

- Section 7.0 – Introduction and Review Process
- Section 7.1 – Instrumentation and Controls – Fundamental Design Principles
- Section 7.2 – Instrumentation and Controls – System Characteristics

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NuScale SDAA FSAR Chapter 7 Review

Significant Changes from DCA to SDA

- NPM power uprate and safety analysis resulted in changes to Reactor Trip Setpoints & Engineered Safety Feature Actuation System actuation logic and analytical limits
- Setpoint methodology TR-122844-P documents the new analytical limits and setpoints
- Common Cause Failure coping analysis revised due to reductions in digital sensors
- Added Type F Post-Accident Monitoring variables

NuScale SDAA FSAR Chapter 7 Review

Conclusion

- While there are some differences between the DCA and 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

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U.S.NRC
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Protecting People and the Environment

Presentation to the ACRS Subcommittee
Staff Review of NuScale SDAA FSAR, Revision 1

Chapter 9, “Auxiliary Systems"

August 22, 2024

(Open Session)

Non-Proprietary

NuScale SDAA FSAR Chapter 9 Review

Overview

- NuScale submitted Chapter 9, Auxiliary Systems, Revision 0 of the NuScale SDAA FSAR on December 31, 2022, and Revision 1 on October 31, 2023
- NRC regulatory audit of Chapter 9 was performed from March 2023 to August 2023, generating 33 audit issues
- 23 audit issues were resolved in the audit
- 10 audit issues resulted in NuScale submitting supplemental information to address questions raised during the audit
- 13 RAIs were issued
- Staff completed Chapter 9 review and issued an advanced safety evaluation to support today's ACRS Subcommittee meeting

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NuScale SDAA FSAR Chapter 9 Review

Contributors

- **Technical Reviewers**
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NuScale SDAA FSAR Chapter 9 Review

Sections

- 9.1 Fuel Storage and Handling Systems
- 9.2 Water Systems
- 9.3 Process Auxiliaries
- 9.4 Air Conditioning, Heating, Cooling, and Ventilation Systems
- 9.5 Other Auxiliary Systems
- Appendix 9A Fire Hazards Analysis

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NuScale SDAA FSAR Chapter 9 Review

Significant Changes from DCA to SDA

- Fire barrier rating
  - In the DCA steel-composite material was limited to staircase walls with a required 2-hour rated fire barriers.
  - In the SDA the reactor building, control building, and radioactive waste building floors, walls and ceilings to be made almost entirely of reinforced concrete or steel composite walls to provide a 3-hour fire barrier rating.
- The SDA only houses six NPM and therefore reactor building size reduced compared to that of the 12 NPM DCA design.
  - Related parameters used in the analysis of the spent fuel pool (SFP) and ultimate heat sink (UHS) cooling analysis changed accordingly.
- Reactor building crane rated capacity increased; modular interface incorporated into crane design; dry dock jib crane added for refueling.

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NuScale SDAA FSAR Chapter 9 Review

Significant Changes from DCA to SDA

- SDA design
  - Consolidated SFP cooling and cleanup system and combined the UHS cooling systems into a single system that cools both the SFP and UHS.
  - UHS pool water level is lower
  - UHS operating temperatures increased
  - Fuel storage rack design and analysis assigned to COL applicant
  - Increased core thermal power which will impact SFP and UHS cooling due to increased heat loads associated with the NPMs and the spent fuel assemblies

NuScale SDAA FSAR Chapter 9 Review

Significant Changes from DCA to SDA

- SDA Chemical and Volume Control System (CVCS) reconfiguration of certain valves outside containment required compensating design changes. Multiple CVCS valves moved out of NPM bay. Valves included internal restrictions credited for CVCS line breaks outboard of containment isolation valves. Flow restricting venturis added into containment vessel nozzles inside containment for the injection and discharge lines. Flow restricting venturis credited in both Chapter 15 and Chapter 19 events.

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NuScale SDAA FSAR Chapter 9 Review

Significant Changes from DCA to SDA

- SDA reactor building crane - risk significant design change from DCA to SDA to include a Reactor Building Crane Control System. Reliability of programmable logic controller and associated components results in operator errors being negligible contributors to module drop.
  - In DCA, over 95% of the core damage frequency driven by module drop. Calculated drop probability dominated by operator errors of commission (e.g., overspeed, over-raise, overtravel and failure of instrumentation (interlocks/limit switches).
- The application does not include specific SFP criticality safety design information and corresponding criticality safety analyses for the SDA. This is addressed with two COL items requiring COL applicants to perform criticality safety analyses for the new fuel and spent fuel pool.

NuScale SDAA FSAR Chapter 9 Review

Conclusion

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

Presentation to the ACRS Subcommittee
Staff Review of NuScale SDAA FSAR, Revision 1

Chapter 12, “Radiation Protection"

August 22, 2024
(Open Session)

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NuScale SDAA FSAR Chapter 12 Review

Overview

- NuScale submitted Chapter 12, “Radiation Protection,” Revision 0 of the NuScale SDAA FSAR on December 28, 2022, and Revision 1 on October 31, 2023
- NRC regulatory audit of Chapter 12 performed March 2023 to August 2023, generating 13 audit issues
- 11 audit issues were resolved in the audit
- 9 audit issues resulted in NuScale submitting supplemental information to address questions raised during the audit
- 2 RAIs were issued and resolved
- Staff completed Chapter 12 review and issued an advanced safety evaluation to support today's ACRS Subcommittee meeting

NuScale SDAA FSAR Chapter 12 Review

Contributors

- **Technical Reviewers**
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NuScale SDAA FSAR Chapter 12 Review

Sections

- Section 12.1 – Assuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable
- Section 12.2 – Radiation Sources
- Section 12.3 – Radiation Protection Design Features
- Section 12.4 – Dose Assessment
- Section 12.5 – Operational Radiation Protection Program

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NuScale SDAA FSAR Chapter 12 Review

Significant Changes from DCA to SDA

- **Radiological Impacts of Hydrogen and Oxygen Monitoring**
  - The DCA rule includes an open item to be addressed by the COLA related to the radiological dose consequences to workers performing activities associated with hydrogen and oxygen monitoring following a core damage accident and potential dose consequences to the control room workers and the public resulting from activities associated with hydrogen and oxygen monitoring
    - Compliance with 10 CFR 50.34(f)(2)(xxviii), 10 CFR 50.34(f)(2)(vii), and 10 CFR 52.47(a)(2)(iv)
  - The SDAA includes a passive autocatalytic recombiner and NuScale has requested an exemption from the combustible gas monitoring requirements of 10 CFR 50.44(c)(4)10 CFR 50.34(f)(2)(xvii)(C)
    - The staff is reviewing the exemption under FSAR Chapters 6 and 19
      - If the exemption is approved, hydrogen and oxygen monitoring is unnecessary, and the associated radiological implications need not be accessed

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NuScale SDAA FSAR Chapter 12 Review

Significant Changes from DCA to SDA

- **Source terms**
  - Source terms throughout Chapter 12 were updated to account for changes in reactor power, the number of units, and other less significant design changes
    - Methodology for developing source terms remained mostly unchanged
    - Shielding and dose rates throughout the facility were adjusted accordingly
  - Staff reviewed the updated source terms, shielding, and zoning for significant radiation sources and found them to be acceptable

NuScale SDAA FSAR Chapter 12 Review

Significant Changes from DCA to SDA

- **Nitrogen-16 (N-16)**
  - The reactor coolant travels through the RCS significantly faster in the SDAA, impacting the N-16 source term
    - Results in significant N-16 concentration remaining in CVCS line exiting the reactor module bays during operation
    - NuScale adequately accounted for N-16 in shielding and dose calculations until N-16 reaches 10 half-lives, at which time N-16 doses are insignificant
  - Staff reviewed the N-16 concentrations and CVCS system shielding and found them to be acceptable

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NuScale SDAA FSAR Chapter 12 Review

Significant Changes from DCA to SDA

- **Concrete Equivalent Attenuation Shielding**
  - Radiation shielding is specified in terms of nominal concrete attenuation thicknesses.
    - FSAR specifies that materials used in place of the specified concrete provides the equivalent attenuation as the prescribed concrete shielding.
    - Alternative radiation shielding must meet the prescribed radiation zoning provided in FSAR Chapter 12 and must be verified to comply with all regulatory requirements.
  - Staff reviewed the shielding specifics provided in the FSAR and in audited shielding calculations and found them to be acceptable. Staff found the approach to be acceptable because it ensures that the zoning provided in Chapter 12 and applicable regulatory requirements continue to be met.

NuScale SDAA FSAR Chapter 12 Review

Conclusion

- While there are some differences between the DCA and SDAA, the staff found that the applicant provided sufficient information to support the staff's safety findings.
- The staff found that all applicable regulatory requirements were adequately addressed for the design.
- COL items are provided for programs and site-specific aspects, similar to the DCA application.

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Presentation to the ACRS Subcommittee
Staff Review of NuScale SDAA FSAR, Revision 1

Chapter 18, “Human Factors Engineering"

August 22, 2024

(Open Session)

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NuScale SDAA FSAR Chapter 18 Review

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NuScale SDAA FSAR Chapter 18 Review

Sections

- Section 18.0 – Human Factors Engineering Overview
- Section 18.1 – Human Factors Engineering Program Management
- Section 18.2 – Operating Experience Overview
- Section 18.3 – Functional Requirements Analysis and Function Allocation
- Section 18.4 – Task Analysis
- Section 18.5 – Staffing and Qualifications
- Section 18.6 – Treatment of Important Human Actions
- Section 18.7 – Human System Interface Design
- Section 18.8 – Procedure Development
- Section 18.9 – Training Program Development
- Section 18.10 – Human Factors Verification and Validation
- Section 18.11 – Design Implementation
- Section 18.12 – Human Performance Monitoring

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NuScale SDAA FSAR Chapter 18 Review

Overview

- NuScale submitted Chapter 18, “Human Factors Engineering,” Revision 0 of the NuScale SDAA FSAR on December 31, 2022, and Revision 1 on October 31, 2023
- NRC regulatory audit of Chapter 18 performed March 2023 to August 2023, generating 20 audit issues
- All issues were resolved in the audit.
- 12 issues resulted in NuScale submitting supplemental information to address questions raised during the audit
- No RAIs were issued
- Staff completed Chapter 18 review and issued an advanced safety evaluation to support today's ACRS Subcommittee meeting

NuScale SDAA FSAR Chapter 18 Review

Significant Changes from DCA to SDA

Area	DCA	SDAA
Main Control Room	12 units	6 units
Staffing	6 operators	3 operators
Important Human Actions	2 risk important actions	No important human actions
Other design changes	Remote Shutdown Station	No remote shutdown station
HFE strategy	Results Summary Reports and ITAAC for Design Implementation	Implementation Plans for: ▪ Operating Experience Review ▪ Functional Requirements Analysis & Function Allocation ▪ Task Analysis ▪ HSI Design ▪ Verification & Validation ▪ Design Implementation ITAAC for Integrated System Validation and Design Implementation

NuScale SDAA FSAR Chapter 18 Review

HFE Strategy

- The staff reviewed Implementation Plans for:
  - Operating Experience Review
  - Functional Requirement Analysis and Function Allocation
  - Task Analysis
  - HSI Design
  - Human Factors Verification and Validation
  - Design Implementation

RSRs will be available before fuel load

- The staff reviewed Result Summary Reports for:
  - Staffing and Qualifications
  - Treatment of Important Human Actions
- SDAA includes COL Items for programmatic elements:
  - Procedure development, training program development and human performance monitoring
- ITAAC ensure remaining HFE activities are complete
  - No. 03.15.01: the main control room HSI is consistent with design verified and validated by the integrated system validation including any changes reconciled during design implementation
  - No. 03.15.02: for integrated system validation of the main control room design

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NuScale SDAA FSAR Chapter 18 Review

Conclusion

- While there are some differences between the DCA and 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.

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August 19, 2024 Docket No. 052-050

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of Presentation Material Entitled "ACRS Subcommittee Meeting (Open Session), August 22, 2024, High Impact Technical Issues Discussion," PM-173236, 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 August 22, 2024. The materials support NuScale's presentation of the high impact technical issues identified during the US460 Standard Design Approval Application review.

The enclosure to this letter is the nonproprietary presentation entitled "ACRS Subcommittee Meeting (Open Session) High Impact Technical Issues Discussion," PM-173236, Revision 0. The proprietary version is provided in a separate submittal, under letter number LO-173238.

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

If you have any questions, please contact Chelsea Lockwood at 541-452-7171 or at clockwood@nuscalepower.com.

Sincerely,

Mark W. Shaver

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

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Enclosure 1: "ACRS Subcommittee Meeting (Open Session) High Impact Technical Issues Discussion," PM-173236, Revision 0

Enclosure 1:

"ACRS Subcommittee Meeting (Open Session) High Impact Technical Issues Discussion," PM-173236, Revision 0

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ACRS Subcommittee Meeting
(Open Session)

August 22, 2024

High Impact Technical Issues Discussion

Presenter: Thomas Griffith – Licensing Manager

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Acknowledgement and Disclaimer

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High Impact Technical Issues (HITIs)

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

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

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Meeting title	**Open Session NuScale Subcommittee Meeting on Staff's Evaluation of Chapters 7, 9, 12 and 18**

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