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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

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WEDNESDAY

MARCH 20, 2019

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ROCKVILLE, MARYLAND

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The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T3B50, 11545 Rockville Pike, at 8:30 a.m., Joy L. Rempe and Gordon R. Skillman, Co-Chairs, presiding.

COMMITTEE MEMBERS:

JOY L. REMPE, Co-Chair
GORDON R. SKILLMAN, Co-Chair
RONALD G. BALLINGER, Member
DENNIS BLEY, Member
CHARLES H. BROWN, JR., Member
MICHAEL L. CORRADINI, Member
VESNA B. DIMITRIJEVIC, Member
WALTER L. KIRCHNER, Member
JOSE MARCH-LEUBA, Member
HAROLD B. RAY, Member
PETER RICCARDELLA, Member
MATTHEW W. SUNSERI, Member

ACRS CONSULTANT:

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

MIKE SNODDERLY

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P R O C E E D I N G S

8:30 a.m.

CO-CHAIR REMPE: Good morning, this meeting will now come to order. This is a meeting of NuScale Subcommittee of the Advisory Committee on Reactor Safeguards. I'm Joy Rempe, and I'll be co-chairing the Subcommittee with Dick Skillman and Mike Corradini.

Members in attendance today are Vesna Dimitrijevic, Charlie Brown, let's see, I guess Ron Ballinger, Jose -- sometimes known as Roland -- Jose March-Leuba, Harold Ray, Dick Skillman, Mike Corradini, Dennis Bley, and Walter Kirchner.

In addition, we're joined by our consultant, Steven -- oh, excuse me. Matt Sunseri and Pete Riccardella. I forgot my colleagues up here at the front table. We also today joined by our consultant, Steven Schultz. And we may be expecting to be joined later today by Margaret Chu.

Mike Snodderly is the Designated Federal Official for this meeting. And today the Subcommittee will review the staff's evaluation of Chapter 9, Auxiliary Systems, and Chapter 16, Technical Specifications of the NuScale Design Certification Application.

We also expect to hear from members of the NRC staff, as well as representatives from NuScale. They'll be briefing our subcommittee.

The ACRS was established by statute and is governed by the Federal Advisory Committees Act. The rules for preparing today's meetings were announced in the Federal Register on March 18, 2019. It will be an open, and possibly then a closed, meeting, as we will, may close the meeting after the open portion to discuss proprietary material.

And presenters can defer questions that should not be answered in the public session to that time. And I'm going to ask for NuScale's help if some of our questions during the open session do go beyond what they should be and to just tell us you'd prefer to wait.

No written statement or request for making an oral statement to the Subcommittee has been received by the public concerning this meeting. A transcript of the meeting is being kept and will be made available, as stated in the Federal Register notice.

Therefore, we request that participants in this meeting use the microphone that is located at the front of the public seating area. And they need to first identify themselves, as well as speak with sufficient clarity and volume so that they can be readily heard when they do make comments.

We have a bridge line that's been established for the public to listen to this meeting. And to minimize disturbance, that line is being kept in a listen-in only mode. To avoid disturbances, I also want to request that attendees at this meeting either turn off or mute their electronic devices.

And we're now going to proceed with this meeting, and I'm going to call upon Scott Harris of NuScale to start us off. Scott.

MR. HARRIS: Thank you. It's a pleasure to be here today and present Chapter 9 on behalf of NuScale. My name is Scott Harris, I'm a supervisor for the Mechanical Systems Group.

Just to give background on myself, I graduated from Kansas State University with a degree in mechanical engineering. I started off my career as a field engineer at Fort Calhoun Nuclear Station for two years, and then transitioned to NuScale in 2012, where I've been a member of the Mechanical Systems Group ever since.

DR. NICHOL: My name is Corrie Nichol, I have a PhD in mechanical engineering from Penn State.

I spent 11 years as a research engineer at the Idaho National Laboratory designing, among other things, some of the remote handling equipment for the Yucca Mountain Waste Package Welding Project. And now I'm working for NuScale in the Remote Handling Group.

MS. FOSAAEN: Good morning, Carrie Fosaaen, I'm a supervisor of licensing at NuScale Power. I've been there for about three and a half years. Involved in Chapter 9 ever since I joined the company. I graduated from Perdue University in 2008 with a nuclear engineering degree, and then in 2010 with a masters in health physics.

MR. FIELDS: Good morning, I'm John Fields. I'm a nuclear nomad. I've been all over the industry for 35 years here, so I'm not going to go into my degree and all that stuff. But all of that in engineering and licensing.

MR. HARRIS: Just a high level review of Chapter 9. It includes fuel storage handling, water systems, HVAC, fire protection and other fire hazard analysis.

DR. NICHOL: So we're going to start off by talking about spent fuel storage and handling, then we'll talk about new fuel storage, spent fuel pool cleanup and cooling system, the fuel handling equipment, and the overhead heavy load handling.

So to start off the spent fuel racks, the design of the NuScale spent fuel racks prevents criticality without regard to loading patterns or zones. The spent fuel pool itself is connected to the ultimate heat sink, the operating pool of the facility, which essentially means the volume is large. The reason that's important is it prevents undetected boron dilution in the pool.

Fuel management is controlled by procedures, fuel storage and that sort of thing. There are a couple tech specs related to fuel storage. Tech spec 4.3 is fuel storage, tech spec 5.5.12 talks about the neutron absorber monitoring program.

And there is a combined license item related to this regarding the programs for fuel movement and the programs and procedures that govern that.

CO-CHAIR SKILLMAN: I'd like to ask a question, please. In this seven million gallon pool, you've got a number of fluid systems that feed to it. What ensures homogeneity in the boron concentration in that pool among the 12 module spaces, the refueling space, and the maintenance bay?

You've got a whole bunch of different

areas, and you've got a number of systems that touch that water. What makes sure the boron concentration, and I'm going to assume it's approximately 2,000 ppm throughout, what makes sure it's homogeneous?

MR. HARRIS: So I can speak to that. So there's two --

MEMBER CORRADINI: If I might just interject. If we're saying things that are proprietary, stop us.

MR. HARRIS: Okay, understood.

MEMBER CORRADINI: We can go and hold for the, but sorry.

MR. HARRIS: Okay, so there's two main systems that service the ultimate heat sink as far as cooling is concerned, the spent fuel pool cooling system and the reactor pool cooling system. Between those two systems, they have a number of suction and discharge points.

Suction points are taken off the spent fuel pool and refueling pool then discharged back into each individual operating bay. So that helps ensure homogeneity.

CO-CHAIR SKILLMAN: Thank you.

DR. NICHOL: One of the unique features of our plant, we don't have specific new fuel storage.

The new fuel is brought in and stored in the spent fuel racks. The racks, the pool liner, and the pool itself are seismic category I, and they are protected from non-seismic category I structures, systems, and components.

There are a couple combined license items related to this. The first one talks about the dry cask. We haven't specified that, that's something that the applicant will choose which model of dry cask, and that will then be spelled out in a combined license item.

There's also a combined license item for the site-specific rack designs.

CO-CHAIR REMPE: So before you go on to the next slide, on the prior slide, I know you've put off to the COL applicant to do the criticality evaluation, but in the staff's review, they noted that some of the information you prepared that there were some errors in some dimensions in that documentation, which you've corrected.

But I guess what I'm asking about is something that I saw a couple other places in the staff's SER, where there were some errors that I would have thought a QA program should have detected.

And although you've fixed the errors the staff identified, did you stop and think that there might some sort of underlying root cause that might be leading to such errors, and have you addressed those errors with some sort of fixes to your QA program to make sure that similar errors aren't existing that the staff might have missed?

MR. HARRIS: I'm not familiar with the particular errors, but I would say if they are identified, they would have been put into our CAP program, and depending on the severity, no matter the severity, would have had a condition evaluation performed.

CO-CHAIR REMPE: Okay, so as we go through today's presentation I'll bring up some other cases where the staff identified some things that I thought that the QA program would have detected or should detect and we'll see if you can, till I kind of get an idea that you really have thought about hey, we don't have those kind of things continuing to occur. Okay?

MR. HARRIS: Okay.

MS. FOSAAEN: And if I may just correct, so we did present a criticality analysis in there. The COL item is to confirm that it's still applicable with the site-specific conditions.

CO-CHAIR REMPE: And in that analysis the staff found some errors, right?

MS. FOSAAEN: Yes.

CO-CHAIR REMPE: And so you understand where I'm coming from, and yeah, okay, we all make mistakes, but are there other mistakes? And should the QA program have detected that?

MS. FOSAAEN: Yeah, I understand the question. And like Scott said, we did enter all of that into our corrective action system for evaluation, but I'm not familiar with the outcome of those actions and what actions we took, other than correcting the errors identified.

CO-CHAIR REMPE: Okay, thank you.

MR. HARRIS: And since we're talking about the --

MEMBER BLEY: Before you go ahead, I'd like to follow up on Dick's question. Put on that next slide, please. Yeah, thanks. There are a lot of separate areas here. It makes sense the answer you gave about why it ought to be mixed. But sometimes things that make sense don't turn out to be right.

Where is the concentration monitored and where will it be sampled and kind of how often? I mean, have you played with this? Do we know for sure that it's well mixed?

MR. HARRIS: Well, I can't --

MEMBER BLEY: Or will we know for sure when the plants operate?

MR. HARRIS: I can't speak to where it's sampled specifically. I imagine it's sampled on several different areas, in the spent fuel pool, refuel pool, and each individual operating bay.

MEMBER BLEY: I didn't see where that's spelled out. That's not part of the design. That's going to be left to the --

MR. HARRIS: That'll be part of an operating procedure.

MEMBER BLEY: Operating folks to figure out where to sample.

CO-CHAIR REMPE: And how will they --

MEMBER CORRADINI: Is that an item that's identified for the operator?

PARTICIPANT: No.

MR. FIELDS: Each applicant will have to develop their own operating and maintenance procedures. That's a COL item.

MEMBER BLEY: But there's nothing in COL items that tells them, that speaks to this issue of --

MR. FIELDS: Not specifically to that, your question, sir, no.

MEMBER MARCH-LEUBA: Generally if you're familiar with the criticality of the spent fuel pool for what you said. Do we take credit for the boron in the water, or are we not taking credit for it? For the spent fuel pool criticality calculation. Did you rely exclusively on the solid racks, or did you take credit for the soluble boron?

DR. NICHOL: So I believe we did take credit for the boron concentration.

MEMBER MARCH-LEUBA: In the water?

DR. NICHOL: For criticality control.

MEMBER MARCH-LEUBA: Okay, then these questions are really relevant. It should not be left over to the QA or the staff.

MS. FOSAAEN: Actually, we have someone in the audience we're going to have help us out here real quick with this question, if you're all right with that.

MR. SHAVER: Good morning, my name is Mark Shaver, I'm a supervisor of the Radiological Engineering Group at NuScale Power. In our crit safety analysis, the spent fuel pool is 1800 ppm of boron, but we only take credit in the criticality safety analysis for 800 ppm boron. And so that gives us 1000 ppm margin.

And also, the limits we work to is K effective value of .95 at the 100 ppm, and then under 1.0 without boron. So we do it both with and without boron, but the boron we do credit is conservative.

MEMBER MARCH-LEUBA: Do you take credit for burnup, burnup credit?

MR. SHAVER: No, we do not take any burnup credit.

MEMBER BLEY: All of this sounds pretty good, but, you know, I've spent a little time around the chemical process industry, and you sometimes see very surprising results in tanks that are assumed to be well mixed when you sample in different places.

MEMBER MARCH-LEUBA: Especially when you have thermogradients. The spent fuel pool is likely to need a hotter source -- to be insulated. And so whenever you have thermogradients in such a large pool, you have to worry about those things.

MR. HARRIS: Okay, so as we're talking about the spent fuel pool, I'll just lay out the ultimate heat sink configurations. So the ultimate heat sink consists of the spent fuel pool, our refueling pool, and the reactor pool. And they all freely translate with each other.

And we also have the dry dock, which we

use to put our module in for maintenance operations during refueling.

DR. SCHULTZ: Excuse me, a question. What latitude are you providing to the COL applicant with regard to the spent fuel pool racks, and what are they working to achieve when they do their own spent fuel pool criticality calculations?

MR. HARRIS: Do you want to speak to that?

DR. NICHOL: Sure, so if you actually go back one slide, this is spelled out. The COL applicant is responsible for submitting vendor-specific design that includes the seismic analysis, the criticality analysis, and those sorts of things. Does that?

DR. SCHULTZ: So they have the opportunity to change rack design different than what you're proposing at this point?

MS. FOSAAEN: Yes, they could change design, but that would require a reevaluation by the staff at the COLA stage. The COL item as represented here is really intended to look at the seismic structural aspect and ensure that the site seismic characteristics are, you know, adequately evaluated. And as part of that, we recognize that it could affect the other characteristics.

So this COL applicant item was really intended to be relative to the seismic aspect. And then if there was a change that wasn't bounded by what was already performed, we recognized they would need to look at these other things, and it could result in changes that would need to be evaluated.

And the standard review plan 384 spells out the criteria that they have to do this re-analysis for. So the applicant would need to use that guidance to ensure that they're appropriately selecting racks if they were changing design.

DR. SCHULTZ: And is NuScale also providing guidance based upon your analysis and your approach that would help the COL applicant to determine what they need to accomplish in those analyses?

MS. FOSAAEN: So we've spelled out certain criteria in the application that would need to be confirmed against, yes.

DR. SCHULTZ: Okay, thank you. So this really starts with things like external events that would not be considered in the normal design practice, based upon what you've assumed. And you've got racks that are seismic one and you've got criteria that match up with that.

MS. FOSAAEN: Yes.

DR. SCHULTZ: Thank you.

MR. HARRIS: So as previously mentioned, there's two systems which provide cooling for the ultimate heat sink, that's the spent fuel pool cooling and the reactor pool cooling systems. Both these systems are not safety-related.

In addition to cooling, they also maintain ultimate heat sink water level and provide a means for chemistry control and also provide reactor pool temperature information signals for post-exit monitoring.

In addition to the ultimate heat sink cooling systems, we have the pool cleanup system. Again, non-safety related. This is specifically used for removing impurities to reduce dose rates and maintain both chemistry and clarity.

We also have the pool surge control system, which is our system used to drain the dry dock during refueling operations.

The pool leakage detection system is used to monitor any leakage from the pool liner. It uses a series of channels to flow any leakage to our sumps within the reactor building for operator evaluation.

MEMBER CORRADINI: And does that, the way that's situated, does that determine the location or just gross loss?

MR. HARRIS: So you can isolate specific channels to help identify the approximate area within the reactor pool. And then from there you can use borescopes to go up and down the channels and identify any wet spots.

And those are my last two points there. Fuel handling.

DR. NICHOL: So the fuel handling equipment, essentially we're talking about equipment that handles receipt of new fuel, refueling operations, and loading of spent fuel into a dry cask. Some of the design considerations, these pieces of equipment are designed to the appropriate design standards, including ASME and ANSI design standards.

Essentially the goal is to reduce the risk of a drop or a criticality event or exposure to radiation. The equipment is also designed to support inspection of the equipment itself.

This is an illustration of the spent fuel pool and the fuel handling equipment. You can see starting on the left there is the new fuel jib crane. The area of hook coverage with a new fuel jib crane is illustrated there with a wedge cut out of the side of it that essentially prevents it from, by interlock, prevents it from carrying loads over the spent fuel racks.

MEMBER BLEY: Have you designed or specified what kind of interlock that's going to be? I mean, some interlocks are like a micro switch or something sensing where it's moved. And if you got a bad signal, it'll keep going. Other ones are actual physical interlocks, so you just can't move.

DR. NICHOL: So it is not designed.

MEMBER BLEY: Is it specified? If not, when will it be specified?

DR. NICHOL: That's a good question.

MEMBER BLEY: Thank you.

CO-CHAIR REMPE: So some background, for some background information, the fact that you put this interlock in was in result to the staff's review, right?

MS. FOSAAEN: That's correct.

DR. NICHOL: So I would think at the latest certainly at the time that the equipment is designed it will be specified. It's a specification to the equipment.

CO-CHAIR SKILLMAN: What is the distinction on the medium left of this image and the far right in this image where there is the green crosshatching, and on the far right, the blue crosshatching, and the combined crosshatching right in the middle? What is the left green crosshatching communicating to us?

DR. NICHOL: So there's a key at the bottom of the image. The green crosshatching is the coverage area that the fuel handling machine mast can access. You can see that covers all of the storage locations in the spent fuel racks, as well as a small portion of the refuel pool, which gives it access to the fuel when it's in the lower reactor pressure vessel during refueling.

The blue crosshatching is the auxiliary hoist, which is a piece of equipment on the fuel handling machine. And that's used for various operations, including handling control rod assemblies, that sort of thing. So it's physically located in a different spot on the fuel handling machine, so its coverage area is different.

CO-CHAIR SKILLMAN: Is there a safety-related connotation to this crosshatching image?

DR. NICHOL: No.

CO-CHAIR SKILLMAN: No. This is just, hey, this is where the equipment can reach, this is where the bridge and trolley are able to operate.

DR. NICHOL: Right.

CO-CHAIR SKILLMAN: It's simply that?

DR. NICHOL: Right.

CO-CHAIR SKILLMAN: Okay, thank you.

DR. NICHOL: The only other piece of equipment that I haven't mentioned is the new fuel elevator. You can see that illustrated there. The new fuel jib crane of course brings new fuel assemblies to the elevator. They're brought down to the level that the fuel handling machine can them access them and place them in racks.

The COL items associated with the fuel handling equipment, there are procedures that need to be developed related to transferring fuel to a spent fuel cask, and also one regarding the inspection of the, and periodic testing of the fuel handling equipment itself.

The overhead heavy load handling system. The main components of this system include the reactor building crane, the nuclear power module, the NuScale power module lifting fixture, which is the physical structural elements that attach to the lift tabs. And then the diagonal braces that attach to the module itself.

There's a module lifting adapter that interfaces essentially between the reactor building crane and the NuScale power module lifting fixture. And there's a wet hoist, which is used for any of the lifting operations that take place under water.

The reactor building crane specifically is designed to be a single failure-proof piece of equipment. It's designed to ASME NOG-1 Type 1 standards, that's the highest NOG-1 qualification, highest NOG-1 design standard.

The movement of that system is controlled via interlocks, and the path and the maximum lift height are limited to prevent or to maintain shielding when moving reactor modules. And there's a communication system between the operator of the reactor building crane and the Control Room.

MEMBER CORRADINI: What is the lift height normally?

DR. NICHOL: The lift height of?

MEMBER BLEY: In other words maximum.

MEMBER CORRADINI: But normal, well, I guess maybe both. When you move a module, it's how far off the floor of the ultimate heat sink floor.

DR. NICHOL: A foot off the floor.

MEMBER CORRADINI: And then I guess Dennis is asking the maximum height?

DR. NICHOL: Correct.

MEMBER CORRADINI: What is that?

DR. NICHOL: The maximum height of the, so I have the numbers in my head. The maximum hook height is --

MEMBER CORRADINI: Assuming you can tell us.

DR. NICHOL: Yeah, it's 145 foot off the, I mean, that's the site elevation, which puts the --

MEMBER BLEY: You said there was a limiter on how high it can lift. Where is that limiter set?

DR. NICHOL: Oh, so the lift height limit is going to be based on the load that's being picked up. So when a reactor module is being handled by the crane, there'll be a limit imposed on how high it's lifted.

MEMBER BLEY: So this is something an operator will have to manually set.

DR. NICHOL: No, it will be handled by an automatic system.

MEMBER BLEY: How does the automatic system know what it's going to lift?

DR. NICHOL: The system looks at the load that the crane is lifting and limits the maximum lift height.

MEMBER BLEY: By, just by the weight.

CO-CHAIR SKILLMAN: What is the quality level of that control system?

DR. NICHOL: That's a good question.

CO-CHAIR SKILLMAN: Yes, it is.

DR. NICHOL: So --

CO-CHAIR SKILLMAN: Okay, so I've got a 740-ton module I'm hoisting. I'm at 11 inches, I pass through 12, now I'm at 13, now I'm at 14. You have no module drop accident analyzed, and this control system has not stopped hoist. What do you do?

DR. NICHOL: So as an operator what do you do?

CO-CHAIR SKILLMAN: Yeah.

DR. NICHOL: So the operator will be informed of what is appropriate --

CO-CHAIR SKILLMAN: Now I'm at 14 inches.

DR. NICHOL: And the operator has the opportunity to hit a stop button.

CO-CHAIR SKILLMAN: How does he know to do that?

DR. NICHOL: He monitors the operations.

CO-CHAIR SKILLMAN: Okay. But he had an automatic system that he was or she was depending on to stop the hoist. So what's the quality of the control system? Let's just leave it at that.

DR. NICHOL: Let me get back to you on that.

CO-CHAIR SKILLMAN: Get back to us on that.

DR. NICHOL: Back to all of you.

CO-CHAIR SKILLMAN: Thank you.

MEMBER MARCH-LEUBA: I'd like to follow up on that because it's in the news all last week, all these computer controls with single switches cause a lot of problems. It really begs the question of what else we have single input computer controlled for an operation that could become dangerous if it runs away.

And that is not the responsibility of ACRS or the responsibility of the staff, it's your responsibility to find it. And we'll review what you've done. You cannot rely on the staff or ACRS to find all those problems.

It was on the news this morning that another plane, well, that like here, the condition of the crash was like there was another pilot. A day before the crash, that he went to the crash the day before because he knew how to handle it. And then the day after, that guy was not flying, so it crashed.

So having a crane handling one of these modules, I'm saying, well, the operator will push the red button.

MEMBER BALLINGER: So you say that it's load control, so it's a set round number 700-ton module. Most load cells 0.1% error, so that's about 7 tons. So what's the next heaviest device that the crane would have to lift?

Is it way beyond the uncertainty of the load system? In other words, is there something that close, that's within the 7 tons error that you could lift and make a mistake?

DR. NICHOL: No, no. I think the next lightest component is well below 400 tons.

MEMBER BALLINGER: Okay.

MEMBER CORRADINI: I was going to just, can I just generalize the question and then you can think about it. Operational procedures of the crew so that they know what's going on, even though they may or may not believe the computer control I think is kind of where we're going.

And so I'm curious, is there a, maybe that's not in this chapter, but I think it's related to the module. That's where we were a month ago, talking to another group of the NuScale folks about these topics. So I think that's the sense of what you're hearing from some of the members.

MS. FOSAAEN: Right, and we do have a COL item that identifies the need to develop procedures and operator training specifically for the crane in 915.

And if I may follow up on the question related to instrumentation, our software control will follow Reg Guide 1.168, and that's described in 7.2. So the crane has been identified, the crane operating software has been identified as needing to follow that information.

MEMBER RAY: Well, there's something I guess was implicit in something Dick said earlier. We've done a lot of heavy lifting in this business, and it's not a unique or first-time event. But when we did heavy lifting in the past, up until now as far as I know, that could result in core damage, we always off-loaded the core first. The exception of course is the removal of a head.

But this is different, because you are moving the core, basically. And so the questions about the quality of the specified requirements, let's put it that way, what are we certifying in this design is the question. If we're just certifying that oh, you have to come up with a crane that'll move a heavy load reliably, but that's the responsibility of each COL holder to do that individually, that's one thing.

But if there are more requirements that are placed in the design certification for this part of the system, we'd like to know what it is specifically.

DR. NICHOL: So I would actually like to speak to that. So in a standard operator system where the, you mentioned the core is offloaded before heavy loads are handled over the top of the core, what we do is different, and we recognize that. And we have spent a good deal of time analyzing the potential for core damage based on load drop scenarios.

Seen, though, from another viewpoint, it is not uncommon to move containers containing spent fuel in an operating reactor, in the operating reactor fleet. So seen from that angle, essentially what we're moving is a very large spent fuel container.

As I said, that doesn't mean that we're dismissing the importance of this, and we have done a lot of analysis looking at potential core damage frequency and those sorts of things. But we are essentially moving a large container of spent fuel.

MEMBER RAY: Well, I would disagree, but

this is not the place to debate that. It's not going to sit there for five years before you load the spent fuel cask. There are many reasons why I wouldn't agree, but this is not the place to have that debate.

The question I'm trying to ask very clearly is are you just going to tell the COL holder they need to address this, or are you going to, is the design certification going to provide specific requirements that must be met by the COL holder. And that extends over many, many issues from quality requirements for the control system to operator requirements and so on.

Because that has to do then with what is the scope of the design certification. Okay?

CO-CHAIR SKILLMAN: I do want to go on record and push this a little bit, because I find your argument flawed. I concur, an 840-ton cask or an 850- or 870-ton module can be safely moved. Happens all the time, big loads, shipyards, other parts of industry move large loads safely.

What's different is you are moving a, the current design moves a core that has abundant decay heat generation, and it is being moved adjacent to potentially 11 live cores. That's different.

And in the analysis, in Chapter 15, in

Chapter 19, module drop accident is not considered. And so I find the combination of your response, it's just a heavy load, kind of like a cask, combined with the absence of a module drop accident, to be problematic. I think NuScale has to own the notion they're moving a very heavy load adjacent to live cores.

Let me go one step further. I was involved in moving a heavy load repeatedly over a dead live core. And at TMI-2, we dropped equipment into the open reactor vessel in which there was destroyed fuel. So to the very best of our intention to have the single failure-proof crane, guess what, a load got away from us. It simply happens.

And so I guess I've got that orientation that I'll never let go of that a crane accident can be a very big problem. And the people at Arkansas Nuclear 1 will verify that. This is a heavy load, and it can have heavy consequences, and it's adjacent to other live cores.

MS. FOSAAEN: If I may, we did not evaluate it in Chapter 15 because of the guidance that says if we design to single failure-proof. However, we did evaluate model drop within our action with nearby modules in Chapter 19. And there is an analysis that does look at that under PRA, with the consequences.

MEMBER BLEY: There's one thing that kind of bothers me, and I don't expect you to answer this now. And I'm kind of guarding the staff, I'm going to ask them as well. It came up in one of my earlier questions about exactly what kind of limit device would be on the jib crane.

In Chapter 18, NuScale responded to the staff that details about the human system interface and essentially potential operator errors is being left to the designer of the crane, who will design to your specification. I assume some of these other issues are also in that ballpark. I don't know when they get looked at, and I'm going to ask the staff that when they come up.

If you've negotiated this and know about that, that'd be great. But some of these issues are potentially important, and putting it off on the designer of the crane at the time the crane's going to be built, I'm not sure where that fits in the review process by the NRC. And will it be reviewed?

Or is it just, I mean, they're not even right now reviewing your specification because they don't think you have it yet.

MEMBER CORRADINI: Can I ask Dennis's question a little bit differently? If it's as he said, is there an ITAAC then that has to come and be cleared, such that the specifications have been designated, met, and then checked by the staff? And we can hold off and ask the staff this, but I'm just kind of curious from your standpoint, because I think he makes a valid point.

MS. FOSAAEN: So I can check, but I'm quite sure that we actually do have an ITAAC for the crane specifically. I don't recall off the top of my head the specifics, but we can get back to that.

MEMBER CORRADINI: We'll come back.

CO-CHAIR REMPE: But I think Harold's point was that did you at least give guidance on what would be done? Not just have an ITAAC, but is there some high level guidance of what is expected is where I am at.

MS. FOSAAEN: We do have guidance in Chapter 14 that helps understand what's required to close the ITAAC. So there is guidance in 14 that specifies what should be done to help resolve the ITAAC.

CO-CHAIR REMPE: Thank you.

MEMBER BALLINGER: I couldn't find this in

Chapter 9, but maybe I should look in other chapters. And I'm quite familiar with the interaction between electromechanical and electrohydraulic systems, computer controls, and what happens when things can go wrong and all of a sudden you get very high temperature steam out into the laboratory.

And the way we solved the problem, invent stuff. The way we solved the problem was to put a mechanical limiter, mechanical limiter, in the system that made it virtually impossible. The crane operator, the computer guy could be stoned and try to move something, and it hits a mechanical stop and it just can't go any further.

Is there something like that with this crane? I couldn't find it, but it's so easy to do.

MEMBER BLEY: They told me earlier that's not specified.

MEMBER BALLINGER: I mean, that solves the problem, or largely solves the problem.

DR. NICHOL: So there are no physical, I'm trying to understand, you're saying, you're asking if there's like a physical?

MEMBER BALLINGER: Module a foot, right. But then somebody asked how high could you lift it. You said 100 --

DR. NICHOL: No, no, that's the top of the.

MEMBER BALLINGER: Yeah, but how high can you, you know, what limits how high you can actually go?

MEMBER BLEY: Ron's point, you know, this could be done by software. Looking at ovations and positions, it could be done by something like limit switches, electrical devices, or it could be done by physical devices --

PARTICIPANT: We had all three.

MEMBER BLEY: To keep it from moving and one works a whole lot more often than the other two. Is successful, the whole.

MEMBER BALLINGER: And it's simple.

MEMBER KIRCHNER: So what is the range of operation of the crane for other purposes? Isn't it used for shielding?

DR. NICHOL: Maybe we can get into that in, I have a figure coming up. I don't think it's the next one, but we'll get back to the range of operation of the crane. And that may answer some of the other questions about limits of where the crane can operate.

MEMBER RICCARDELLA: Before we go on, Corrie, you mentioned an ASME standard that I'm not familiar with. Could you tell us a little about that?

DR. NICHOL: I believe that was NOG-1, nuclear --

MEMBER RICCARDELLA: NOG, N-O-G?

DR. NICHOL: Yeah, nuclear overhead gantry, N-O-G-1.

MEMBER RICCARDELLA: Oh, okay.

MEMBER BALLINGER: But none of those documents mention a live core. 550 whatever it is, is two NUREGs and everything, there's nothing in there that says this is for a live core. So that's what makes it different, and I presume that your PRA, you're getting around it by having it a very low probability event in the PRA.

CO-CHAIR SKILLMAN: Well, no, the way they get around it is they communicate this is a single failure-proof crane and it won't happen.

MEMBER BALLINGER: Right, right, but the NUREGs don't have anything in there that says if you're lifting a live core, you got to do something different. It's just --

MEMBER BLEY: Lifting over a live, adjacent to it, lifting over a live core.

DR. NICHOL: If I could clarify, though, that we do, in Chapter 19, we do consider module drop accidents that would include core damage. We do look at that.

MEMBER MARCH-LEUBA: Yeah, and I'm looking at it. I'm not sure if it's proprietary or not, so I won't mention numbers, but they are ludicrous.

MEMBER DIMITRIJEVIC: Well, in your Chapter 19, that's the most important event from the point of core damage. How you go around is that it's crap and it's not going to be released and it's not considered as a safety measure. And so there is not really detailed analysis otherwise, as you tried to mention.

This is an estimate of probability of this happening, which is the low in ten to minus eight. But it's still the most important event because this is very safe plan. So this mostly, it seems this is happening in the big pool, in the scrapping there is no releases. That's what your estimate is that.

However, from the point of the importance of the thing, even things will just to lead to core damage without leading to releases should be considered important.

MEMBER BROWN: I've got one question.

MEMBER BALLINGER: I'll say it again, a mechanical stop cuts the head off of this snake.

MEMBER BROWN: But let me ask a question relative to that. Obviously if you've got a computer-controlled crane, you've got a processor of some kind in it, right.

DR. NICHOL: Correct.

MEMBER BROWN: A microprocessor or something. How do you handle a circumstance where the processor locks up? You call it single failure-proof, but what does that mean relative to the software? Do you have redundant software that runs this that has to be compared? Do you specify for that?

DR. NICHOL: So single failure-proof refers to the portions of the crane responsible for handling the load such that it isn't dropped.

MEMBER BROWN: Okay, well is that software-controlled?

DR. NICHOL: The single failure-proof portions of the crane are physical.

MEMBER BROWN: Are what?

DR. NICHOL: Physical. They're hardware, they're redundant ropes and those sorts of things.

MEMBER BROWN: So the answer is no. Well, what makes it move?

MEMBER MARCH-LEUBA: No, the answer is digital means. Digital means, software.

MEMBER BROWN: Software.

DR. NICHOL: Right.

MEMBER BROWN: So what, if it's moving, what prevents it from failing if the software locks up?

DR. NICHOL: So there will need to be a redundant system that monitors the operation of the existing software to ensure essentially that that doesn't happen.

MEMBER BROWN: Well, I know, but you just commented that the mechanical parts are designed to be backed up so that you don't have it, it's a single failure-proof.

DR. NICHOL: Right.

MEMBER BROWN: But you don't make any comment about the software being backed up, the single failure-proof.

DR. NICHOL: Right.

MEMBER BROWN: You kind of limited it between the software-based systems and the mechanical-based systems. And if anything we've ever learned, software locks up. It'll do it unexpectedly, you're moving your mouse and all of a sudden it doesn't move anymore, just randomly.

And the processes that you use, whether, are they going to use custom software, is it going to be commercially dedicated software with all kinds? You're going to get locked up if you don't provide some safeguards, I don't, safeguard's the wrong word, requirements to provide monitoring and/or backup and/or prevent lockup of that software, and have that lockup result in something that stops everything so that you can take manual control.

If you don't tell them to do that, how do you do it? Right now, it seems like there's a division of responsibility right now. Here all these things are going to be redundant and single failure-proof, but the software is just fine because it's not a problem. It's kind of a hard, that's a hard nut to follow. And I just throw that on the table and let you deal with it.

DR. NICHOL: Okay.

MEMBER CORRADINI: My thought is we need to move on, but I think you get the sense of it is specifications that are given by NuScale, go to the vendor, come back, are checked by NuScale and are inspected by the staff might be what I'll call a consensus opinion of the members.

So to make sure that you essentially come back all the way full circle so you have what you want in terms of your system design.

MEMBER KIRCHNER: It goes back to Harold's question, what are we certifying here, just a single module? Or what's unique about this plant is the idea of having multi modules and yet, the amount of detail that's being provided for critical functions is scant in this particular case. And yet, this is at the crux of multi-mode operation. I just find it deficient. That's one person's opinion.

MEMBER BROWN: You've really got to think about a fail-safe circumstance for the software-based systems, and that doesn't seem to be addressed based on the conversations I've heard. I'll stop at this point.

C0-CHAIR REMPE: So I think we've made our point. And maybe if we haven't, we'll come back with -- just let's go ahead. Unless you have some clarifying thing like oh, you should look at chapter such and such and you can see this, I think you need to move on, but you can understand there's some concern.

DR. NICHOL: Yes, let's move on.

C0-CHAIR REMPE: Okay.

CO-CHAIR SKILLMAN: Corrie, let me ask one other question before we move on.

DR. NICHOL: Okay.

CO-CHAIR SKILLMAN: I searched the word buoyancy and I find that intriguing because the way the documents are written, it seems to identify that buoyancy might be variable from one place to another. And I would just assert that if you're taking an imaginary one cubic foot of something to the Challenger Deep in the Marianas Trench or you put that same cubic foot of imaginary in your bathtub, it's still about 62.4 pounds of buoyant upward through the centroid of that mass.

Why is buoyancy a concern here? And what relationship would buoyancy have to the load-lifting capability of the crane, if any? My thought is none, but I'm asking the question because I searched on buoyancy and it shows up a couple of places.

DR. NICHOL: So we credit the buoyancy of the module when we assign a name plate capacity for the crane.

CO-CHAIR SKILLMAN: And so you consider the whole module voided?

DR. NICHOL: No. The condition that we analyze is the condition that it will be in in refueling which includes water in ECCS system and as --

CO-CHAIR SKILLMAN: So the containment is flooded?

DR. NICHOL: Right.

CO-CHAIR SKILLMAN: So what would be buoyant?

DR. NICHOL: We displace water with the steel that makes up the reactor module. And we don't flood the entire space in the containment vessel.

MEMBER CORRADINI: Is there a -- I figured you were going to say that. Is there a reason for that?

DR. NICHOL: To keep the -- I don't know, is that proprietary?

MEMBER CORRADINI: We can hold off.

CO-CHAIR SKILLMAN: Let's talk about that. I'm just curious because what is unique here is, if you will, a buoyancy discussion for a module you don't find that when you're refueling any other reactor.

DR. NICHOL: Right.

MEMBER CORRADINI: Let's talk about it in the proprietary section, please.

DR. NICHOL: Right.

MEMBER CORRADINI: Thank you.

MS. FOSAAEN: We recognize the importance of that as well and you'll find that that's reflected in the full-level bases as an important function.

MEMBER CORRADINI: I know. That's why I asked the question. That's where I went searching. Thank you.

DR. NICHOL: Okay, so briefly, I want to talk about the actual operating bay. In this image, you can see the safe-load paths are illustrated so maybe I can just talk through bringing a module in and then refueling operations will, of course, be the reverse of that.

So when a module is brought into the plant, it's on its side. It's brought in where the pointer is in the top left of the figure. It's brought in horizontally. It's brought in and then upended in the inspection rack.

C0-CHAIR REMPE: Is the crane used at this point or is something else used?

DR. NICHOL: Once the module is upended, the crane then picks up the upper module to move it to the next station.

C0-CHAIR REMPE: So at this point it's horizontal. It's not upended. So how are you bringing it in is what I'm asking?

DR. NICHOL: It's brought in horizontally and then upended.

C0-CHAIR REMPE: By not the crane.

DR. NICHOL: Correct.

C0-CHAIR REMPE: By other devices.

DR. NICHOL: Correct.

C0-CHAIR REMPE: A tractor or something, I don't know. And then a crane helps it upend?

DR. NICHOL: No, there's a piece of equipment that's the module upender.

C0-CHAIR REMPE: Okay. It's brought in or it always stays in there?

DR. NICHOL: It's a part of the refuel --

C0-CHAIR REMPE: Okay.

DR. NICHOL: Or the inspection rack.

C0-CHAIR REMPE: Okay. I didn't see this in Chapter 9. Maybe I missed it. But it's there?

MS. FOSAAEN: The upender is described in 3.

C0-CHAIR REMPE: Three. Okay.

DR. NICHOL: So once the module, the top portion of the module is brought in, I guess I should preface this, the lower reactor pressure vessel and the lower containment vessel are brought in separately and upended separately and placed in the reactor flange tool and the containment flange tool by means of the wet hoist.

Once the module is brought in and upended, the reactor building crane picks it up and brings it to the reactor flange tool where it's assembled to the lower reactor pressure vessel. Then the assembly is picked up and moved to the containment flange tool where it's assembled to the lower containment vessel and then it's moved to the appropriate operating bay.

The safe load paths are illustrated there. They're the fairly narrow darker bands that illustrates essentially where the module can be moved by the crane. The other cross-hatched area is the area of coverage for other operations that include wet hoist operations and other auxiliary hoist operations in the pool.

And I'll also point out there is a heavy load exclusion zone where the reactor building crane is prevented by interlock from the trolley traversing over that space where the spent fuel racks are in the spent fuel pool.

This is a chart that describes the design codes that are applicable to different pieces of equipment. You can see the reactor building crane main hoist. I mentioned ASME NOG-1, Type I. That's also the code that's used for the auxiliary hoist. The module-lifting adapter and the NuScale power module lift fixture are both designed to ANSI N14.6 which covers below the hook-lifting devices.

The wet hoist is an NOG-1 hoist and the jib crane is designed to the ASME NUM-1, Type II standard. NUM stands for Nuclear Underhung Monorail. So it's a different configuration crane, so it's covered by a different standard.

Also, noted on the chart are the seismic category, the different pieces of equipment and the maximum traverse and hoist speeds of the different equipment.

I would mention there are a few COL items on the crane. The first covers the process for handling receipt of critical loads including the module. The second one talks about spent fuel cask handling equipment including the procedures for safe handling of the spent fuel casks. The last one talks about the governing procedures. We mentioned this, the training and governing procedures that will be required for operation of the heavy load handling systems.

DR. SCHULTZ: It's these areas where I find it -- I'll call it strange or surprising that you're putting this to the COL applicant. I just have to believe you'd want to provide very specific instructions associated with these types of things. Project that you've got five NuScale facilities or you've got 10 or 100 NuScale facilities, to have COL applicants doing different things at different places, I just don't quite understand why you wouldn't want to have very specific instructions, training programs, procedures and so forth. These are very important things.

You may be able to survive an accident or a mishap associated with movement of a module, but if that happens and it will be significant for the NuScale industry, you don't want it to happen. So I'm surprised that if you don't have them, you would put off to a COL stage something that would be that significant and be totally involved in the process rather than have a COL applicant come up with this at that stage of the process.

MS. FOSAAEN: And I would agree. We at NuScale, we do intend to have a lot of services available at future times. At this point, the focus was on the design and we do have development in-house. And so those would be services we would intend on offering a future applicant.

And I think one thing that's important here is that you mentioned five potential reactors.

So there would be an advantage to standardization and something we didn't really mention is that for us refueling, I know you've said would be a frequent evolution. So the advantage to that is we'll have a lot of operating experience. We'll have crews that are very familiar, as they'll be constantly performing these evolutions, whereas in a traditional plant you're only doing a refuel 18 months or 2 years. And our cranes are always accessible as a result of not being inside a locked containment. So there are some advantages to our design. And I agree with the points. And thank you for that.

MEMBER BALLINGER: I guess I didn't notice this, but the last item on that slide, detailed description of the safe load paths for movement of heavy loads. I find that astounding because one of your previous slides showed the safe load paths.

MS. FOSAAEN: Right.

MEMBER BALLINGER: That's got to be tied to the PRA and every other thing.

MEMBER KIRCHNER: It has to be well specified.

MEMBER BALLINGER: Has to be specified just doesn't make any sense to me at all.

MS. FOSAAEN: So we did analyze and do

have specified safe-load paths. That item was added specifically in response to a staff question.

DR. SCHULTZ: Right. It was an RAI response. I appreciate what you're saying, but you say we can provide this as an offering to a COL. I just think it's too important to NuScale to leave it to the COLs.

I know the hubris of the industry is such that oh, we can do this. We don't have to have commonality, but think of the history of the industry and at some point we determined that we needed a PWR owners group and a PWR owners group activity in many, many different areas.

It would be nice if we didn't have to do that later, but if we did it now, and make it easier for the first COL applicant and those that follow to get this done.

MEMBER RAY: Not only that, Steve, but you know, we're focused here on the certified design of the scope and the content of that. The intentions that exist for providing information, help assistance, guidance and so on down the road, that's not part of the certification. It's not something we can credit. It just can't be because it's certified design. It has value as a certified design, regardless of what the holder of that design may ultimately do years in the future. And so we're just trying to get into the certified design, the things that we think are essential. Crediting what the present intentions are in the future just can't be part of that process.

C0-CHAIR REMPE: I think we made our point. Let's keep going. Thank you.

MR. HARRIS: Okay, on to 9.2 which is the wire systems. This includes station service water is not in NuScale design, but 9.2 includes reactor component cooling water systems, demin water system, potable and sanitary water, the ultimate heat sink, condensate storage, site cooling water, chilled water and utility water.

The majority of these systems are a typical industry design, so in the interest of time I'm going to only focus on 9.2.5, the ultimate heat sink. Otherwise, the remainder of those systems are non-safety related and non-risk significant.

C0-CHAIR REMPE: I'm going to stop you here though since I think that means you don't plan to talk about the reactor component cooling water system. Is that a true assumption?

MR. HARRIS: I can talk about it if you'd like.

C0-CHAIR REMPE: I'd like it.

MR. HARRIS: Okay.

C0-CHAIR REMPE: Just briefly, at the last month's meeting, I was told because of whatever, the multi-module chapter that you have said clearly you're going to install what's needed for when you have a less than 12 modules.

So in this component cooling water system, if you had 12 modules, you'd have 3 systems of 2 identical systems that each supports 6 modules and then you have a third system that's a backup. So in this case, even though I didn't see it clearly stated in the DCA, I would assume you will do the backup and the one system for whenever you even have one module installed. Is that true?

MR. HARRIS: I can't speak to particular construction activities about what, if you'd install one or both reactor component cooling water systems. I believe there's only -- should be only two reactor component cooling water subsystems, one for each side, north and south of the reactor cooling.

CO-CHAIR REMPE: Right, to the six, but there's a backup. When I have module number 1 that means I should have one of these two systems installed and the backup, right?

MR. HARRIS: I'm not sure what you're referring to as a backup. It's a non-safety system. There's no -- we have redundant pumps.

C0-CHAIR REMPE: The power train is not in operation, but it's kept in stand by and started, if needed. I'm quoting from the open DCA, okay? And so to me, that means clearly that I've got to have the backup as well as one of these systems installed from day one if I have one module. And that's all I'm asking because I mean that's what I think we read according to this multi-module chapter that you do, and I just am asking for clarification. Because sometimes there's a system that supports like three or four modules. Sometimes there's one that supports six. But then there's like sometimes a backup system and I'm just asking can we assume that, because we are certifying a very -- sometimes fuzzy -- DCA.

So I'm getting into another question of what are we certifying here, but just an example. And I would have expected you have said immediately, oh, of course, we'll have that back-up system when we -- before we start one of these modules up.

MS. FOSAAEN: I think it was just the terminology that was confusing. The system would need to have its basic functions. So if it was credited to have all three pumps installed, even for one module, that would be required.

C0-CHAIR REMPE: I can assume, and this is on the record, that again, before you start one module, we'll have one RCCWS and its backup.

MS. FOSAAEN: And by backup, we mean the full suite of pumps that would be in there.

C0-CHAIR REMPE: Okay. Okay. Go ahead, now. Sorry.

MR. HARRIS: And so I think like Carrie said, it's a difference in terminology, I think.

C0-CHAIR REMPE: Okay.

MR. HARRIS: So other than the ultimate heat sink, none of these systems are an essential source of water, nor are they required during or after a natural phenomenon event. And anything in proximity to a Seismic Category 1 component is designated Seismic Category 2.

On to the ultimate heat sink. Like I mentioned previously, the ultimate heat sink consists of the reactor pool -- refueling pool and spent fuel pool. It is a stainless steel lined reinforced concrete pool filled with borated water.

CO-CHAIR SKILLMAN: Scott, can you tell us about how thick that, if you will, the floor liner is under what will be the controlled load path for the module? Is it a one inch, two inch, a --

MR. HARRIS: It's less than one inch. I can't speak specifically --

CO-CHAIR SKILLMAN: It's less than an inch?

MR. HARRIS: Not off the top of my head.

CO-CHAIR SKILLMAN: Is it puncture proof, if that module gets away from the -- from your main hook?

MR. HARRIS: I wouldn't expect -- so -- I just want to clarify, the pool liner is non-safety related. So the safety-related structure is the reactor building.

CO-CHAIR SKILLMAN: So it's the concrete underneath.

MR. HARRIS: Right.

CO-CHAIR SKILLMAN: Got that. So now I've got a 740 ton and I drop it, do I puncture that less than one inch pool liner?

MR. HARRIS: I would expect if you dropped a 900-ton load, it would puncture steel less than an inch.

CO-CHAIR SKILLMAN: That's what I would expect. So is there going to be some armor plating under that safe-load path or something to make sure that no matter what, that pool liner cannot be punctured? Just think about it. You don't have to answer.

MR. HARRIS: Okay. Thank you.

CO-CHAIR SKILLMAN: Thank you.

MR. HARRIS: The ultimate heat sink consists of approximately seven million gallons of water, so it's sized such that active cooling systems are not required for accident conditions. And that combined volume of water provides sufficient coolant for greater than 72 hours without additional make up.

The ultimate heat sink is also provided with a seismically qualified make up line that can be connected external to the reactor building if additional water were to be added.

And during normal operations, spent fuel pool cooling and reactor cooling system services this body of water and in a design basis event involving loss of AC power decay heat removed from the modules and into the ultimate heat sink.

MEMBER CORRADINI: So let me ask just a quick question. So is there a procedure as to the replenishment after -- if there's change in level? There was a note made and the number is not important, but so many days before replenishment.

Is there a procedure for replenishment in terms of at some point at some level?

MR. HARRIS: So the level is covered by tech specs. So I believe the tech spec level is greater than 68 feet, normal level, so anything below that would trigger actions per the tech specs.

MEMBER CORRADINI: Thank you.

MR. HARRIS: And a number of COL items for chemicals used, sanitary waste disposal, corrosion, et cetera.

On to process auxiliaries. This consists of the compressed-air system, process sampling, equipment and floor drain system, chemical and volume control system, stand by liquid control is not applicable to our design, and also the containment evacuation system and containment drain system for 9.3.6.

Similar to the other water systems, I'm only going to focus on the items unique to our design so with the exception of chemical and volume control system. These are non-safety related, non-risk significant systems.

In any structures which could adversely affect Seismic Category 1 or designated Seismic

Category 2, and we have two COL items, leakage control program and post-accident sampling.

With the chemical and volume control system --

C0-CHAIR REMPE: I guess before you go there, just for the record, apparently, the sampling system, there's an issue about -- it's on hold because of the source term issue, right?

MS. FOSAAEN: That's correct. There is an open item from the staff in the review relative to that. And NuScale recently did submit an exemption in relation to that that is affecting that review.

C0-CHAIR REMPE: So you've requested an exemption, submitted all the documentation and the staff is reviewing it at this time.

MS. FOSAAEN: That's correct.

C0-CHAIR REMPE: And that's where the hold is right now. Thank you.

MR. HARRIS: All right, on to the chemical and volume control system. It's used to purify reactor coolant, maintain chemistry including boron concentration. Also provides for make up and let down and supplies pressurizer spray flow and also provides a de-gas operation.

There's one chemical and volume control

system per module and during start up is used in conjunction with the module heat system to raise the reactor coolant temperature to generate natural circulation. And also, it's used in conjunction with the boron addition system which is a shared system for all 12 modules.

MEMBER MARCH-LEUBA: Don't move. This system, the CVCS is dear to all of us, we tend to think about. Let's make sure I understood because typically I am the one that nobody understands when he speaks, but the acoustics in this room is so terrible that I don't understand you.

MR. HARRIS: Okay.

MEMBER MARCH-LEUBA: You have to speak more clearly. There is one CVCS system for the whole plant. Is that correct?

MR. HARRIS: No. There's one per operating module.

MEMBER MARCH-LEUBA: There's one CVCS system --

MR. HARRIS: So there's 12 total.

MEMBER MARCH-LEUBA: So there is one BAS system per module?

MR. HARRIS: No. One BAS per all 12 modules and --

MEMBER MARCH-LEUBA: It doesn't appear on the drawings. So each module has its own CVCS?

MR. HARRIS: Correct.

MEMBER MARCH-LEUBA: And they are not --- other than to the BAS system, they're not cross tied?

MR. HARRIS: BAS is just -- so correct, within the CVCS, they're not cross tied.

MEMBER MARCH-LEUBA: Completely independent. So if one fails, you cannot bring --

MR. HARRIS: It will not impact.

C0-CHAIR REMPE: But there's a backup power supply system, right, that includes -- there's two -- the backup power supply system, I guess, includes two redundant diesel generators and one auxiliary alternating current power source.

How much of that is installed when you have the first unit and six units and seven units, et cetera? Is all of that installed from day one? Because that is needed for the CVCS and I assume it's given credit for it.

MR. HARRIS: So any system, whether it's -- if it's shared or not, required for a module operation will be installed and as you bring other modules on line, module-specific systems would be installed for those modules. So for your example, backup diesel generator, that would be installed to support that initial first module coming on line.

C0-CHAIR REMPE: So the two -- for modules less than six, would you have one diesel generator, but you'd have the alternating current power source installed? Or would you put both diesel generators in from day one?

MR. HARRIS: I'm not sure of the specifics of the backup diesel generator, if there's one or two.

C0-CHAIR REMPE: Maybe it's an unfair question to ask you, but do you think it's documented in the DCA somewhere other than a vague statement?

MEMBER CORRADINI: Your question is where is it documented?

MS. FOSAAEN: I know that 20.1 discusses at a high level the multi-module in bringing additional modules on line, but I don't believe the level of detail you're requesting is specified in the DCA.

C0-CHAIR REMPE: So I'm just kind of wondering when they start building this, and I know, the economic case depends on being able to produce power with a few modules while you're still bringing in the other modules. But to make sure that they've appropriately done DCA, it seems to me some sort -- I think we've looked at that test in chapter -- was it 20, 21, whatever it was you said. And it's very high level and it kind of commits, yes, we're going to bring in everything that's needed for the modules. Okay, I might have accepted when you said well, we'll put one of the diesel generators, but that back up system, it seems like it's got to be there.

Somehow or other, I'd like to see a plan that provides some guidance on how it's going to be added on because we get a lot of single units sites in the U.S. where they've got a concrete pad and it's like the second unit never got installed.

CO-CHAIR SKILLMAN: Let me raise an issue here on CVCS. When I review, let's back up. CVCS provides the ability to maintain the chemistry of the reactor coolant system. It's a back up for adding boric acid to your seven million gallon pool. It serves a host of key functions.

If I'm an operator and I wish to add boric acid to the reactor coolant system, I would have to light off my boric acid system. I've got to work my way through a bunch of valves, find my way to the make-up pump, and at the make-up pump increase the pressure from 152 psi up to 2200. And only then do I get that boron, boric acid, into the core.

From Table 3.2-1, everything in CVCS is quality group D. It's 2 Bravo, excuse me, Bravo 2 out of .69 and it's all Seismic Category 3 for both CVCS and boric acid.

My point is a system upon which I very significantly depend is basically commercial grade, no augmented QA equipment. Why should I be comfortable with that? It's a key system.

MR. HARRIS: I agree. It's a key system. I do want to clarify a point. There are two safety related demineralized water isolation valves which are safety-related.

CO-CHAIR SKILLMAN: That's great. I'm not for putting in un-borated water. I agree with that.

MR. HARRIS: I agree.

MS. FOSAAEN: I think the key point --

CO-CHAIR SKILLMAN: I want a lot of boron and I want it now.

MS. FOSAAEN: Right. And our Chapter 15 analysis does not credit CVCS for any boron addition.

CO-CHAIR SKILLMAN: I know. Isn't that great. I want boron. I'm an operator. I want shut down. I want as much as negative reactivity as I can throw at that core for whatever reason. And I have no way to do it unless I go through a daisy chain.

Doesn't that spark a little bit of uncomfort? I would think it would to those who are people who have held reactor operator licenses. And every other application, you push a button, you get copious boric acid. And on the Bs (phonetic) you've got slick. And on this plant, I don't have that capacity. Is that something that ought to be looked at more thoroughly?

MR. HARRIS: I'd also like to add that there are redundant make-up pumps, redundant recirculation pumps, so there's redundancy within the system if an event a pump failed.

CO-CHAIR SKILLMAN: I still have a daisy chain to get boric acid to my core. I'll just leave it at that, okay?

MR. HARRIS: Okay.

MEMBER MARCH-LEUBA: And I want to second that opinion.

MR. HARRIS: Thank you.

MEMBER MARCH-LEUBA: We will see in Chapter 15. Thank you.

MR. HARRIS: So the chemical and volume control system is also the only system with connections to the reactor coolant system. The piping runs outside containment, so it is the total scope of the inner system LOCAs for consideration in NuScale design.

The containment isolation function for CVCS is also part of the containment system which is covered in Section 6-2.

As I mentioned, CVC is equipped with two automatic safety related fail closed demineralized isolation valves to ensure no inadvertent boron dilution concentration. Those are covered by Tech Spec 346 for containment isolation valves.

Section 936 is the containment evacuation system and containment flooding and drain system. These systems are used to transfer liquids and gases between the containment vessel free volume and other plant systems.

The containment evacuation system establishes and maintains a vacuum in the containment vessel by a moving water vapor and non-condensable gases in the CNV. It also has a function to perform leakage detection for a reactor coolant system.

The containment flooding and drain system is used to flood the containment vessel with borated water after shutdown in preparation to removing -- to moving the module and it's also used to drain the module prior to start up.

And the containment flooding and drain

system also has a function to provide borated cooling inventory to the containment vessel during a beyond design basis event.

The isolation function, containment isolation function is covered in Section 6-2 and in tech specs for these systems are 347, 349, and 373.

And here's just diagrams of the containment evacuation system. It consists of two redundant vacuum pumps. These pumps are operating continuously. Any liquid that's removed from the vessel is condensed and collected and quantified in the sample vessel and the non-condensable gases are sent to either the reactor ventilation system or gaseous rad waste system.

And the containment flood and drain system consists of two redundant pumps. There's one containment flood and drain system per six modules, north and south of the reactor building, so they each serve six modules apiece.

On to 9.4 which is air conditioning, heating and cooling, and ventilation system. This covers control room area ventilation, reactor venting and ventilation in the spent fuel pool, ventilation radiative waste building ventilation, turbine building ventilation. Section 945 engineering, continuous safety-feature ventilation system is not applicable to the NuScale design.

A common element to these systems are they are non-safety related or risk significant, not credited for public dose mitigation. Any components which could adversely affect Seismic Category 1 are designated Seismic Category 2 and we had a COL item to specify periodic testing inspection program requirements.

CO-CHAIR SKILLMAN: Let me ask this on the ventilation systems, one characteristic that people in the plant and the plants know real well is that ventilation systems are systems where you don't get wet and you don't get irradiated. And so most of all, you ignore them and they sit there and they turn away until the force draft fan of the blower breaks an axle and starts making noise, if the maintenance doesn't put some grease like they should. So it's squeaking and you say something needs to be fixed.

But the ventilation systems provide the negative pressure barriers and the contours for ventilation to make sure that the operators are safe. So here are these ventilation systems that are almost not too different than a ventilation system in an airplane hangar in this application.

What makes sure that these ventilation systems are performing the way they're supposed to? Because they actually do perform -- you might not call it a safety function, but it's certainly an important function for the operators.

MR. HARRIS: There's a number of assurance that you would use to ensure system performance flow, pressure, pressure across filter banks. And you're going to be continuously monitoring those systems and even non-safety alarms tied to anything that would identify improper performance.

CO-CHAIR SKILLMAN: Thank you. Right?

MR. HARRIS: On to 9.5. It includes communication system, lighting system, fire protection program, and fire hazards analysis. So the communication system includes components for inter-plant and plant off-site communication. This includes private branch exchange, public address, general alarm system, sound power telephones, distributed antenna, and radios for point-to-point communication. Serves no safety-related risk significant functions. It's not credited for design basis accidents or safe shutdown. And there are two COL items attached to the system.

CO-CHAIR SKILLMAN: Let's back up. 9.5.1

is your fire protection program. Chapter 9 is void of any description of your firefighting system. No identification of plumbing. I don't know how many diesel driven fire pumps or electrically driven fire pumps you have. I don't know where your fire mains are. You do communicate in Chapter 3.2 or Table 3.2 item 1. It's all non-safety and it's not seismic.

Fire protection is probably one of the most important systems you have in the plant. But there isn't anything in the design cert. application that points to this. What it does communicate is that some fire piping is likely Seismic II over I. And in that case, it won't be Seismic III. It will be something else. But that's a real loose requirement for what is at least in my judgment a critical system in this design.

Why isn't there any treatment of the fire system in Chapter 9?

MR. HARRIS: So the fire protection system is described in 9.5.1. We'll be getting to that. We kind of just reversed order so we could do the fire hazards analysis together in this presentation.

CO-CHAIR SKILLMAN: I would suggest that it's mighty sparse.

MR. HARRIS: There's a large matrix table of compliance with Reg. Guide 1.189. And there's where you get all of the specifications for how much water to be delivered to given areas. How many stand pipes, what the yard configuration is, the separation of the yard piping, and all that kind of stuff. All that is included in the compliance to Reg. Guide 1.189 table in Section 9.5.1.

MEMBER BLEY: It sounds as if the system is not designed yet. Is that right?

MR. FIELDS: I don't know how -- what full extent it's designed.

MR. HARRIS: It's designed to the point necessary for DCA.

MEMBER CORRADINI: Say that again?

MEMBER BLEY: There's no schematic. There's no --

MS. FOSAAEN: There is, in 9.5.1, there's Figure 9.5.1-1 and it shows the pumps, the tanks. So there are high-level schematics in 9.5.1 at the tail end.

MEMBER MARCH-LEUBA: Can you say again what table?

MS. FOSAAEN: 9.5.1-1 and 9.5.1-2.

MEMBER CORRADINI: But you made a point and maybe I didn't understand it. You said it's not required for DCA?

MR. HARRIS: I'm not saying that. I'm just saying that it's designed up to the point required for design specification application. It's not in detailed design.

MEMBER CORRADINI: Thank you.

C0-CHAIR REMPE: So I didn't come back and give you examples where I found the QA program might have a problem. I think that they throughout the staff's review, they noticed the seismic categories were incorrectly identified. There were plant layout drawings that had been updated and not reflected in the DCA.

I'm just kind of wondering again when you have something that's only designed up to a certain extent, is there going to be a good process in place that will say oh, I need to change something because of the fire protection system design, a wall needs to be thicker and you'll have noted it there, but it won't get updated other places in the DCA. And I'm bringing this up not to pick on you, but we've actually had examples for a certified design had not been updated and it's a real costly endeavor to change a certified design. And so that's why I'm wondering, do you have confidence you've improved your QA system that you'll be able to track requirements from different systems that if that's the actual plant design and say oh, I need to change it here, too, because whatever system?

I mean how -- we noticed some problems and say oh, yeah, we need to fix it and we have done that or you just don't think it's a problem yet?

MR. FIELDS: Yes, we've gone through a very painstaking process over the last year to ensure that the DCA is under configuration control and we've captured changes that we had missed in previous changes --

C0-CHAIR REMPE: That's the answer I was looking for.

MR. FIELDS: Yes, we've been going through that process to go ensure that and as we grow and as we change into customers and other applications, we're setting up systems to capture that so that it doesn't cross pollinate or doesn't cross boundary lines for one design versus another.

C0-CHAIR REMPE: Thank you.

MEMBER BROWN: I would presume the COMS systems that you talk about in your first paragraph there include general alarm systems, branch, public address, and it's intra-plant as well as plant to offsite.

Is there any definition of how those communication systems are ensured that they don't provide false information to plant people since your external to internal plant information? For instance, a general alarm could be sounded based on a cyber-attack that comes in and turns on the alarm and everybody puts their hair on fire, or is that say, you don't specify anything relative to that through the COL anywhere?

Are you putting it off or are you trying to isolate critical functions that could provide general alarms throughout the plant, so as to not inadvertently done by hacking. I'm sure your system is computer controlled. They're all done that way.

MR. HARRIS: Yes, if I may, I'd like to Jeff Ehlers, our electrical supervisor, who is on the phone?

MEMBER BROWN: Your who?

MR. HARRIS: Jeff Ehlers. He is our electrical supervisor on the phone address that question.

MEMBER CORRADINI: Jeff, are you out there?

MR. EHLERS: Yes, I'm sorry, we just got

off a minute. So yes, I would figure any software, any software that we use for our communication system will have to be under the software QA program and meet the same requirements as all the other software in the plant for cyber security.

And other than that, communication systems are all specified to be encrypted to add another layer of protection there.

MEMBER BROWN: By encrypted, what do you mean? It sounds good, but I'm -- encryption means if I send something somewhere, you have to have a fancy algorithm on the other end to decrypt it. And you have to encrypt it to start with. I presume -- I'm not a designer, but I assume that's the right thought process.

MR. EHLERS: Yes. I believe you're correct. We have not specifically made -- stated any specifics about that encryption, but there is a requirement in the DCA for all communications to be encrypted.

MEMBER BLEY: I've got a question. I spent time in the Navy and we used sound powered phones all the time. I don't usually see them in a commercial plant. Will they be throughout the plant to be used when the other com systems don't work well?

MR. HARRIS: Jeff, can you answer that one?

MR. EHLERS: Sorry, the beginning of the question --

MEMBER BLEY: Yes. I haven't seen sound powered phones in commercial plants. Of course, the Navy has had them forever.

Two questions about it. One, are they rigged throughout the plant so operators can use them when other communication systems are down? And two, all the ones I'm familiar with have really hideous frequency response and the first time you use them, it just sounds like gobbledy-gook and you have to practice for a week or more and pretty much routinely to be able to understand people. You can understand perfectly well when you're used to them.

And is there anything about training people on using these sound powered phones or maybe they're better than they used to be.

MR. EHLERS: Well, I think your first question is about the industry. They're already used in the industry. They were used at my Houston plant.

MEMBER BLEY: Oh, okay. I haven't seen them, but go ahead.

MR. EHLERS: Yes, they do have in the

nuclear industry. They do use them. There are designed -- they are jacks installed in various locations to be able to plug one in and use -- as long as you have a proper path that you can be able to use it.

MEMBER BLEY: Okay, so the jacks are around the plants such that you could use them for any operations you need to do?

MR. EHLERS: Correct. If you lose all the other battery pack communication systems, either wireless, fiber optic, or whatever, then yes, you have the sound powered to use.

MEMBER BLEY: Okay. What about legibility and practice on them?

MR. EHLERS: Legibility and practice?

MEMBER BLEY: Yes, the ones I'm used to, if you don't use them regularly, you can't understand what anybody is saying. You have to practice kind of routinely to be able to understand. Frequency response, or at least used to be horrible on them.

MR. EHLERS: We don't have any specification for, you know, how they would practice, but I'm assuming would be in their ERO drills and stuff like that where they would practice that kind of drill where they have to go to their sound powered phones. I know we have.

MEMBER BLEY: Okay, so that's left up to whoever is going to operate the plant?

MR. EHLERS: Yes, it's part of the emergency preparations they do.

MEMBER BLEY: Okay.

MR. HARRIS: On to 9.5.3, our lighting system, plant lighting system provides artificial illumination for buildings, room spaces, and outdoor areas of the plant. The functions include normal plant lighting, emergency plant lighting, normal and emergency main control room lighting. It's in accordance with NUREG post-700. We have emergency lighting fed from our DC power system for a minimum of 72 hours and a battery pack for one and a half hours for egress or exiting in accordance with NFPA 804.

And the plant lighting system also provides emergency lighting with battery pack for eight hours for post-fire, safe shutdown activities outside of the main control room.

Line 5.1 is our fire protection program. The objective of our fire protection program is to use defense in depth to achieve the required degree of reactor safety. Our safe shutdown relies on passive fire protection and redundant safe shutdown of equipment is separated by three-hour fire barriers. So part of that defense in depth is prevent fires from starting, to rapidly detect, control, and extinguish fires that do occur, and provide protection to components important to safety.

9.a, fire hazards analysis. Our fire hazard analysis demonstrates that the plant maintains the ability to perform its safe shutdown functions and minimize radioactive material release in the event of a fire. So our hazard analysis considers in-situ and transient fire hazards, determines the effect of any fire -- of a fire in any location in the plant and its ability to safely shut down the reactor. Also specifies measures for fire protection prevention, detection, and suppression.

Within 9.a, our special cases are a main control room fire area. We have switches that provide back-up control of systems, automatically controlled by module protection system.

In the event of a main control room fire, we have manual switches associated with module protection functions to isolate those actuations. Those switches are located outside the main control room.

For containment fire area, it's highly

unlikely due to our containment is held at a vacuum. Our electrical conductors are made of noncombustible material and a lot of the conduit is inaccessible during normal operation. And also during shutdown, we flood our containment which also helps prevent ignition or the spread of fire.

For the top of module fire area, also considered highly unlikely. It's enclosed in our bio shield and cabling under the bio shield is in a conduit or three hour rated cable. This top of module area is inaccessible during normal power operations. The hydraulic hood used in our containment isolation valves is specified to be noncombustible.

For our containment isolation valves and decay heat removal system actuation valves, these valves move to their safe position using stored pressurized nitrogen accumulators.

And between modules, there is a three hour fire barrier with the exception of the bio shield while facing the reactor pool.

And during shutdown, the bio shields are moved and administrative controls placed on that area for combustible -- for transient combustibles.

CO-CHAIR SKILLMAN: Where are the valves that are hydraulically actuated located?

MR. HARRIS: All of our containment isolation valves are hydraulically actuated and heat removal actuation valves are also hydraulically actuated. They're all located on top of the actual module.

CO-CHAIR SKILLMAN: So if you had a hydraulic leak, where would the hydraulic fluid leak to?

MR. HARRIS: Assuming it's gravity draining, just leak on top of the module and into the pool.

CO-CHAIR SKILLMAN: Into the pool. And what is the fluid? What is the hydraulic fluid? Is that like automatic transmission fluid?

MR. HARRIS: I can't speak to the specifics of the hydraulic fluid. It is noncombustible.

CO-CHAIR SKILLMAN: But it is a hydrocarbon.

MR. HARRIS: Again, I can't speak to the specifics of it.

MR. FIELDS: I don't know if it's been specified fully at this point. It's just that it's noncombustible

MEMBER CORRADINI: Do you specify it this way?

MR. FIELDS: I don't believe it's fully specified what we're going to use, but it's just a noncombustible hydraulic fluid.

MR. HARRIS: And just to reiterate, to go to the safe position for all these valves I mentioned does not require any hydraulic fluid. It's all done by nitrogen spring.

So there's a vessel on top or connected to each containment isolation valve and to actuate the valve you actually remove, you vent that hydraulic path and it has that hydraulic path vents during nitrogen spring -- moves the valve to the safe position.

CO-CHAIR SKILLMAN: At what pressure is the hydraulic fluid?

MR. HARRIS: I'm not prepared to answer that question.

CO-CHAIR SKILLMAN: I suggest you take a look at the operating experience from TMI II. We leaked hydraulic fluid and we ended up with leafy green vegetables because whatever is in that water that may be organic will go after that hydraulic fluid and it will grow.

MR. HARRIS: And there are instruments in place to identify if hydraulic pressure is reducing and thus identifying a leak.

CO-CHAIR SKILLMAN: I didn't realize there was hydraulic fluid over the pool water. So I'll leave that with you. Thank you.

MEMBER MARCH-LEUBA: Let's continue on that. I'm looking at this figure 9.5.1-1 and what I see here about it, fire protection system is two big water tanks and three pools that spray water into whatever is inside the plant, right? So am I to assume that you are spraying water on top of all of the control room computers and control system and the operators? Or is there a second fire protection system for the control room and the controllers?

What it specifies here is that you're going to be spraying water inside the control room and obviously, make it completely inhabitable and nonworking.

MR. FIELDS: If Ed Siener's on the line we could have him answer that question for you.

MR. SIENER: Could you repeat the question, please?

MEMBER MARCH-LEUBA: What type of fire suppression system do you have in the control room and all the controllers, computers, electrical system?

MR. SIENER: In the control room specifically, there's no automatic water suppression. It's a continuously operated control room so we rely on detection and manual suppression in the control room.

In electric rooms, it's somewhat similar depending on the fire loading. I would say generally in an electrical room, there is not automatic suppression. There's detection with manual suppression located per code.

MEMBER MARCH-LEUBA: Is that specified in the CVA somewhere or it's a COL item?

MR. SIENER: Well, one place it is located in is the 9A where we have every fire area in the plant and it says specifically whether it does or does not have automatic suppression.

MEMBER MARCH-LEUBA: Okay, I'll look it up now that I know where to look.

Second related question, when you look at the drawing of the plant, most of it is underground and most of it -- well, one would consider it water-tight rooms, for example, the battery room. It's underground and fairly water tight. When you turn on all those sprinklers in the event of a fire, what happens to the water that accumulates on the floor?

Because it's underground, so the only way you can drain that water is with an active pump system. Does nothing exist or do you just want to make a swimming pool of the battery room?

MR. SIENER: No, the NFPA code specifically address drainage and any of those rooms with automatic water suppression would have drainage that would not allow de-pooling of water. Most electrical equipment, other important equipment would probably be elevated so that we could deal with small puddling on the floor. But the drainage would take care of any buildup of water.

MEMBER MARCH-LEUBA: These are underground rooms, so it would have to be active draining. You have to have power and you have to have a pump working. But the point that mostly Dick is making which he has a lot of experience on this is that fire protection is probably the most important thing you can do in this plant for safety and it looks a little light on the CVA.

MR. SIENER: A huge advantage NuScale has is compartmentalization. There's a lot of rooms and we're able to separate all the safe shutdown equipment. We're using three hour fire barriers except where notified in those special cases. So our suppression -- we assure safe shutdown by separation of the equipment, so the suppression is really more for defense in depth and human safety.

CO-CHAIR SKILLMAN: But just remember, Browns Ferry was designed with that same, if you will, mental framework. TVA intended for those plants to be extremely safe and they underwent a terrible fire.

And my point would be that sometimes fires happen and they get away from you and they're insidious because they actually have a mind of their own. You can think you're going to drive that fire and the fire will drive you and you've got a couple of choices. You can prevent it from happening by having no combustibles or you can make the compartments so small that there is no real threat to the rest of plant, but now you are confined basically like a rat in a cage. You can't get to it.

And so the whole issue of firefighting is one that warrants a kind of a front lobe attack at the design stage to make certain that the fire cannot get to where it shouldn't get to. And where it's really problematical is in the control room and your switch gear and particularly where your cable spreading occurs. You don't need the sermon, but just recognize again it happened at Browns Ferry and those of us who have had maritime experience, the worst thing that can happen is that you're so compartmentalized you can't get to it which means you have to have ample either carbon dioxide or Cardox or wintergreen or whatever you're going to have or at Browns Ferry copious amounts of water readily available which means your headers and your hoses have to be ready to go.

C0-CHAIR REMPE: Colleagues, we're already like 30 minutes behind, so let's let them go through and then we need to decide whether we're going to bring the staff up or have a break before. But we are way behind, okay?

Go ahead.

MR. HARRIS: This is our last slide, just listing out our COL items, or no COL items for fire protection, no unique COL items for fire protection. We have generic items for fire protection program development, procedure development, implementation, QA program, training. And no unique COL items for the fire hazards analysis, just COL items to provide site specific fire hazards analysis and that's it.

C0-CHAIR REMPE: Do we want to have a quick break and come back? So let's do that and it's 10:16 by my clock, so let's get back here by 10:25, okay? And that will give us staff time, but I didn't want to cut into the chapter 16.

(Whereupon, the above-entitled matter went off the record at 10:16 a.m. and resumed at 10:25 a.m.)

CO-CHAIR REMPE: Okay. We're back. It's 10:25 and I'm going to bang the gavel. So, please sit down if you're in the near area. And the rest of the members will come as they can.

So let's restart. And I'm not sure which of the staff is beginning, but okay, please start.

MR. TESFAYE: Good morning everyone. My name is Getachew Tesfaye. I am the NRC Project Manager. Can you all hear me?

I'm the Project Manager for Chapter 9 of Auxiliary System. The staff has completed the Phase Two safety evaluation for the open items and submitted it to ACRS.

This report contains four one items. And one of them, one of the open items has since been closed. And the remaining open items will be addressed in Phase Four of this review.

Chapter 9 comprised of five major areas, review areas and 26 sections. And our presentation slides have 41 slides, but due to time limitation, we'll focus our presentation on Section that have open items and areas where we believe there is ACRS interest.

Those who are presented are -- who are presenting are sitting here with me. And the rest of the technical staff is up in the audience.

With that, I'll ask the presenters to first introduce themselves.

MS. SIWY: Alex Siwy, NRO Reactor Systems.

MR. STUTZCAGE: Ed Stutzcage, Radiation Protection Reviewer.

MR. NOLAN: Ryan Nolan, Reactor Systems.

MS. SACKO: Fanta Sacko, Electrical Engineering Branch in NRR.

MR. TESFAYE: And the rest of the contributors to this 41 slide package are Alissa Neuhausen, Raul Hernandez, Hanry Wagage, Chang Li, Angelo Stubbs, Bob Vettori, Tony Gardner, Alexander Chereskin, Andrew Yesnick, Nan Chien, and Dawnmathews Kalathivettil.

So, with that, if it's okay, we'll do the staff observation.

MS. SIWY: Thanks Getachew. Again, my name is Alex Siwy. And I'll walk you through the staff's review of Section 911.

The main objective of the review of this

Section is to verify that fuel will remain sub-critical during all credible storage and handling conditions in accordance with 10 CFR 50.68 and GDC 62.

The main guidance that the staff used for its review is SRP Section 911. And there are several topics in this review area as are listed on this slide.

As a result of the staff's review, the staff finds that the applicant's criticality calculation methodology is acceptable and adequately benchmarked. And the applicant's criticality models correctly incorporate design information and use appropriate assumptions.

There is an open item related to the analysis. And that is that the applicant is making the structural analysis of the racks, a COL item.

The implications of this COL item on the criticality analysis is that a COL applicant would have to show that any rack deformation or relocation due to a seismic event or a fuel assembly drop would be bounded by the existing criticality analysis.

And if it weren't, then the COL applicant would have to re-analyze. Right now this is an open item pretty much because the staff is working with the applicant and OGC on the wording of the COL item.

Other than that, the staff generally accepts the approach.

So, other than the analyses that are related to that COL information item, the staff finds that the applicant's criticality analyses do comply with 10 CFR 50.68 and GDC 62.

CO-CHAIR SKILLMAN: Alex, let me ask this question, please.

MS. SIWY: Sure.

CO-CHAIR SKILLMAN: On the safety evaluation on page 12, the safety evaluation communicates about two thousand PTM.

And then on page 13 is the statement, DCA Part Two, tier two, section 911 states that the large volume of water, about seven million gallons by the -- in the UHS prevents an undetected boron dilution accident from the normal -- I think it's nominal, nominal boring concentration of about 18 hundred to below eight hundred.

My question, is it two thousand or 18 hundred?

MS. SIWY: I'd have to double check the table in the DCA. I think --

CO-CHAIR SKILLMAN: I thought it was two thousand.

MS. SIWY: Yeah. I think they're aiming for two thousand. But I think there is one table in the DCA, in chapter 9 that says at least 18 hundred.

CO-CHAIR SKILLMAN: Thank you. All right. Thanks.

MS. SIWY: Okay. Yes, next slide. So, the staff also conducted an independent confirmatory analysis that support the applicant's conclusions with regards to the acceptance criteria.

The staff also notes that the tier one information is complete and consistent with tier two. And that the Tech Specs related to fuel storage adequately protect the assumptions in the criticality analysis.

With regard to materials for fuel storage, the staff notes that NuScale has not specified a neutron absorbing material. And there is a related open item in which the staff requested additional details on the impacts of manufacturing on neutron attenuation, materials qualifications for the spent fuel pool environment, and manufacturing process controls.

The staff and applicant recently agreed on a path forward for this open item by creating a COL item to address the staff's concerns. And like with the other open item, the staff and the applicant are working on wording of this right now.

The staff also notes that the applicant is implementing a neutron absorber monitoring program that's consistent with TSTF-577, Revision 1. And utilizes the staff approved NEI 16-03 guidance for monitoring a fixed neutron absorber and spent fuel pools.

And this provides the staff reasonable assurance that degradation of neutron absorbing material can be adequately detected so that the neutron absorber continues to provide the credit criticality control.

That concludes my presentation. Are there any questions?

MR. TESFAYE: Thank you Alex. And Section 9.1.2, new spent fuel storage, there was one open item related to the fuel gage protection system. The information we received was provided by NuScale. And we have since closed that open item.

And for the next sections, we don't have any open items. The slides are presented. If you have any questions, we'll be glad to address them.

No open items in water system either. So, the next open item is in Section 9.32. If it's okay, I would like to jump to that slide.

MEMBER CORRADINI: No. I think you're going the wrong direction.

MEMBER BLEY: On 9.1.2-1, RAI response, you say that one's closed now. Did you investigate how they ensure they have mixing in the -- in fuel pool so the boron concentration is the same everywhere?

MR. TESFAYE: Okay. Raul, would you please address that question?

MEMBER BLEY: And hot just the pumping, but how they do sampling to make sure it really is mixed.

MR. HERNANDEZ: This open item is not related to the boron solution issue. The open item in front of us is related to leakage detection.

And the spent fuel pool and the ultimate heat sink having an adequate leakage channel behind the walls to detect leakage. And the latest response they added.

For the boron solution issue, I don't have the exact location of the sampling point. So I cannot talk about that.

I can mention that the suction in the returns from the cooling systems are located at different points on the pool that would allow the flow of water to increase mixing and help prevent having stagnant areas of water.

MEMBER BLEY: How are you convinced they don't have any stagnant areas?

I've seen lots of tanks where people were surprised when they finally sampled at a different place then they'd been sampling and found very different concentrations.

MR. HERNANDEZ: Based on the drawings and -- I have not -- I don't expect there to be one. I don't have any specific details to give you right now at this moment.

MEMBER BLEY: So you don't know of really having mixing.

MR. HERNANDEZ: At this moment I cannot answer the question specifically.

MEMBER BLEY: Okay.

MEMBER CORRADINI: Is there --

MEMBER BLEY: I'm sorry.

MEMBER CORRADINI: Oh.

MEMBER BLEY: Go ahead Mike.

MEMBER CORRADINI: Well, I just wanted to ask, there's a COL information item, 9.1-1. I guess, and it was about procedures relevant to handling of storage of spent and new assemblies and criticality control.

This isn't related. But in some sense, their leak -- as I understood when we were asking NuScale, they're leaving it to the operator to develop procedures to determine that they have good sampling. And it is homogeneous.

That's how I interpreted.

MEMBER BLEY: I didn't hear them say that. They said, yeah, the operator will have to figure out where they're going to sample.

But, nothing about we have been mixing, we have checked for stagnant areas. We make sure you've got the uniform solution.

I didn't hear anybody say that that was a requirement that was laid out.

MEMBER MARCH-LEUBA: I guess my question is more for Alexander who is the technicality person. Is there any place in the UHS where we make ready for the boron other than the fuel suppression at the spent fuel pool?

MS. SIWY: I specifically focused on the spent fuel pool part. The other instance I could think of is during refueling.

MEMBER MARCH-LEUBA: Yeah. I don't -- I

don't think there is any transient that results in water around the pool that inside the vessel.

And the containment is always intact.

MS. SIWY: Um-hum.

MEMBER MARCH-LEUBA: So, just a recommendation, if I were going to sample it, I would sample it in the spent fuel pool drain line, the suppression line. That's where I would make sure that the boron is within specs, because that's where it's needed.

But that's my opinion.

CO-CHAIR REMPE: I believe there's someone from the audience who had a comment? Oh, okay. Sorry.

MR. TESFAYE: Okay. Are we ready to move on?

CO-CHAIR SKILLMAN: Getachew, what challenge, if any, did the staff raise about the adequacy of the thickness of the pool liner floor?

MR. TESFAYE: Yes. We have that question for. Raul, do you have the answer to that?

MEMBER BLEY: If we speak more directly in the microphones, everybody can hear you better. It's surprising how bad the acoustics are.

MR. TESFAYE: Yes. Okay, I'm sorry.

MR. HERNANDEZ: Can you repeat the question, please?

MR. TESFAYE: That how confident are we --

CO-CHAIR SKILLMAN: The question is, what consideration did the staff give to the adequacy of the thickness of the pool liner floor?

There's a stainless steel liner down there. It sounds like it's a quarter of an inch, maybe half an inch. And there are going to be multiple 730 ton loads --

MR. HERNANDEZ: Okay, --

CO-CHAIR SKILLMAN: Transported over or dragged over that very thin membrane.

MR. TESFAYE: Okay.

MR. YESHNIK: This is Andrew Yeshnik. I'm a materials engineer in the Office of New Reactors.

CO-CHAIR REMPE: Speak up.

MR. YESNICK: Closer? Oh, sorry. That was not credited. The liner itself is not credited with any sort of structural function in the case of a drop.

The drop is precluded so it's not part of the design.

MEMBER MARCH-LEUBA: So if it drops, it breaks? Or it has a high probability of breaking? Or at some point --

MR. YESHNIK: It will probably deform, yeah.

MEMBER MARCH-LEUBA: Huh?

MR. YESHNIK: I mean, it's a very heavy load.

MEMBER MARCH-LEUBA: The difference --

MR. YESHNIK: I have not seen the analysis of what that load specifically is, but.

MEMBER MARCH-LEUBA: I heard you say earlier something I liked very much. There is a leak detection system.

MR. YESHNIK: Um-hum.

MEMBER MARCH-LEUBA: I'm pretty sure there is one on the spent fuel pool. But is there one on the UHS?

MR. YESHNIK: Yes.

MEMBER MARCH-LEUBA: So there are channels underneath, if it breaks we will see it before the pool starts dropping?

MR. YESHNIK: No. Those channels are designed to detect any leakage of the wells. Not the entirety of the UHS.

MEMBER MARCH-LEUBA: Oh, so it's not an out leakage for this in use?

MR. YESHNIK: Um --

MEMBER MARCH-LEUBA: Ignore me.

MEMBER CORRADINI: If the well is cracked, it's swept along the weld line.

MR. YESHNIK: That's correct.

MR. HERNANDEZ: I just wanted to add that and this is not unique to this design. There are in other designs, the liner itself, it doesn't have the function of retaining the water.

The function is basically to protect the concrete. The concrete structure is the one that is holding the water and is designed to maintain the integrity.

We do monitor the wells between the plates to ensure that there is no long term degradation of the concrete because of all known leakage.

But, like Andrew just said, the liner is not credited to retain the water. It's the concrete structure.

CO-CHAIR SKILLMAN: I assure you my question was not aimed at suggesting the liner has any structural component whatever. The liner is nothing more than a membrane.

But if that membrane is injured because of whatever reason, that is a common mode failure for 12 live core reactors.

Now let me say that again. If you tear that liner, if you drop a module, if you drop a fuel assembly, if you drop a fixture, if you drop a reading device onto a one-quarter inch thick membrane that is well supported, I suspect that there will be a puncture wound.

And you will have to do something to repair that liner. And that's a common mode failure on a single pool that's cooling potentially 12 live, 160 megawatt reactors.

That's a problem. And that means that floor liner should probably be more then a one-quarter inch membrane.

It should be either armored for the load path. Or otherwise precluded from sustaining a puncture wound.

One man's opinion.

MR. TESFAYE: I think we've taken a note. So we'll look into that.

CO-CHAIR SKILLMAN: So the answer to my question is, the staff did not consider beyond a quarter of an inch or whatever that membrane thickness is.

And that was not challenged. I believe

the answer is no.

MR. TESFAYE: No.

CO-CHAIR SKILLMAN: Okay. Thank you.

MR. TESFAYE: Thank you. Any other questions on this?

(No response)

MR. TESFAYE: If there are no questions on the next Section 9.1.3 spent fuel pool cooling and clean up system and light load handling and systems, I'd like to move on to the next section that has an open item.

We have no open items on water systems. Okay. Ed, let's move onto.

MR. STUTZCAGE: Hi, I'm Ed Stutzcage. I'll be present 9.3.2. Which is process and post-accident sampling systems.

The purpose of the process imposed accident sampling system is similar to other designs. It is to collect and analyze the chemical and radiochemical conditions from the primary and secondary systems and other systems including the containment atmosphere. Next slide, please.

There's no dedicated post-accident process sampling system used in the NuScale design. The normal process sampling system is used during post-accident conditions.

I'll be focusing mostly on post-accident sampling, because we had no open items for normal sampling.

For post-accident sampling, we actually had seven open quest -- open items in Chapter 9. And nine open questions in Chapter 12.

They're all related to the post-accident sampling process, which is unique to NuScale's design. And you can go to the next slide, please.

So in NuScale's design to collect liquid samples during an accident, you'd have to un-isolate containment, and send the fluid through the chemical and volume control system. Through the sampling line where it would, you know, it would be sent through the sampling point and collected.

For gaseous samples, it would also have to -- the containment would have to be un-isolated. Liquid would have to be sent through the containment evacuation system to the sampling system and then returned to the containment through the containment flood and drain system.

There were numerous staff concerns related to this. Particularly related to the dose, to workers taking the samples and also the system operation and so on and so forth.

NuScale has recently requested an exemption from the requirements of 10 CFR 50.34(f)(2)(8), which are requirements to take post-accident samples and within specific dose limits.

Instead they would -- they're requesting to rely on continuously planning to take samples if they needed to.

And part of their basis for this is, they have other means of providing the inf -- of getting the information that sampling would provide, such as under the bio shield radiation monitors for radiochemical conditions, they have hydrogen monitors.

However, NuScale's hydrogen monitors are on their sample lines. So they'd have to use a similar process of opening the containment evacuation system to monitor for hydrogen.

The exemption request is somewhat similar to relaxations applied to the operating fleet in several topical reports. But like I said, there are some unique features in the NuScale design that are being considered.

So, we're still evaluating the exemption requests.

MEMBER CORRADINI: So, can I clarify this one? Because I didn't understand it. It's because you're operating at low pressure within the containment, and that creates an issue of how to take the sample?

MR. STUTZCAGE: There's different challenges depending on what the conditions are inside containment. There's a possibility and my understanding that you would have to send nitrogen into the system to push the gaseous fluid through.

And there's also potential challenges of liquids. If the water in the containment is too low, depending on where you're at in the accident, and what the accident is, you may have to do things to raise the water level.

So, there's -- like I said, there's -- it kind of depends on the situation and conditions.

MEMBER CORRADINI: So, their contingency -- would you help me out here? I should have asked them and I forgot.

Their contingency is to do what in terms of sampling? Or is it yet to be determined because we're in the middle of the RAI and the open items?

MR. STUTZCAGE: NuScale's approach is that they have other means to detect the information that sampling could provide them. However, if they really needed to, and they chose, it would be beneficial to take a sample.

They can, you know, they have the capability to do it. It's just they're only going to rely on it if they determine it was, you know, completely -- if they determine it to be beneficial.

They don't want to have a requirement to have to be able to do it within, you know, a short amount of time following an accident.

MEMBER CORRADINI: But this is -- this RAI is still going to remain open. This is --

MR. STUTZCAGE: Yeah --

MEMBER CORRADINI: Is 90.44.

MR. STUTZCAGE: All of the -- yes. All of the -- right. All of those questions remain open as we evaluate the exemption request.

And then yeah, we make a determination on that -- determination on that and we'll figure out what to do with that RAI.

MEMBER CORRADINI: Thank you.

DR. SCHULTZ: Is there a schedule for that Ed?

MR. STUTZCAGE: I can't say we have a definitive schedule at this point. Yeah, it came in at the very end of January and we've -- yeah.

DR. SCHULTZ: Got you. Thank you.

MR. STUTZCAGE: Any other questions?

(No response)

MR. TESFAYE: Okay. Thank you. Next slide is on CVCS. Ryan Nolan will be presenting that.

MR. NOLAN: Yes. My name is Ryan Nolan. I was one of the reviewers for the CVCS system.

And we don't have any open items in this review area. However, there is an exemption request that I'll spend a little bit of time discussing, as well as some previous ACRS questions that we received. I'll discuss some of those as well. So if you go to the next slide.

So NuScale requested an exemption to GDC 33 which requires having a reactor coolant makeup system to address for small breaks in reactor coolant pressure boundary.

And NuScale's design doesn't rely on any injection of ECCS from external tanks for adding inventory to the RCS. Instead, they rely on isolation of the system, inventory retention.

And then the ECCS and the containment vessels maintain the core coolability to meet the SAFDLs. And the way the staff approached this is we looked at it from both normal operation considerations, as well as off normal.

And so for normal operations, we would assume that CVCS is available to control inventory. And then for off normal considerations NuScale relies on the safety actuation isolation signals that would isolate CVCS as well as potentially containment. And then rely on ECCS and containment to meet the SAFDL requirements.

That's the finding we made on the exemption to 33.

MEMBER CORRADINI: If -- this is a kind of an off question. If they chose to un-isolate containment and they isolate a wait use list, is that selective?

Or is it un-isolated in all valves such that if they wanted to operate ECC -- CVCS selectively open the valves in there.

MR. NOLAN: I'm not as familiar in like recovery operation aspects of how they would actually do that. I would have to go back and look at the design.

MEMBER CORRADINI: Yeah. But it's not in the plan of how they want to address any sort of accident situation. They go to isolation and --

MR. NOLAN: Correct. And rely on --

MEMBER CORRADINI: To delay the high pressures and low pressure ECCS.

MR. NOLAN: Yes. And rely on ECCS for the long term cooling requirements.

MEMBER MARCH-LEUBA: Yeah, we -- we would look at these events during Chapter 15 review a little more. Because Chapter 15 is missing a few events.

MR. NOLAN: Um-hum.

MEMBER MARCH-LEUBA: One of them is normal shutdown. And under normal shutdown, without CVCS completely isolated, the reactor will start cooling down by itself. And it will start shrinking.

MR. NOLAN: Um-hum.

MEMBER MARCH-LEUBA: The water level will drop to 63 percent of volume. And I would like to see what happens then.

I mean, during normal operation they will have CVCS maintain the inventory in the vessel. And the vessel is going to drop significantly in liquid.

And I have not seen that analysis to ensure that it is acceptable.

MR. NOLAN: Well, we'll go to the next slide. We've received some questions --

MEMBER MARCH-LEUBA: Um-hum.

MR. NOLAN: About mixing phenomena. A low flow boron mixing phenomenon that we've seen in boiling water reactors.

And I guess the phenomenon is at really low flows, when you put boron into the lower vessel, the boron has a tendency to settle out and not make its way into the core.

And so in this drawing, what I'm trying to illustrate here is that the CVCS injection line is actually above the core. It injects eight feet above the core.

And so if there is a phenomenon of low flow or boron potentially setting, it actually has to go into the core first before.

And this is just sort of teeing up a discussion in Chapter 15. This is a thermal hydraulic phenomenon that we're currently looking at.

Boron dilution in long term, that's evaluated in Chapter 15. And then the mixing methodology that NuScale has used for the dilution events is evaluated in 15.4.6.

MEMBER MARCH-LEUBA: My point is, of course the vessel will never reach the temperature on the UHS. But, it might get close.

Because after, a mounting there eventually will reach. If you start at 1850 psi, the density of the vessel is 0.63 grams persistent. At the end of the operation, this is 1.

So if you have complete isolation of containment the water level will drop about 63 percent of what it was inside the vessel. So the operator will start CVCS eventually. And they will put a lot of water.

And that water better not be un-borated. And I don't see anywhere that that event is described. I don't see any event in Chapter 15 or Chapter 19. Maybe it is in Chapter 19.

I'm just putting it on the record, that there are events that I have not seen anywhere. Thank you.

MR. NOLAN: One more point I wanted to make. Member Skillman, you mentioned earlier this morning about -- you questioned the capability of putting boron into the core and why is CVCS not safety related, or questioning the pedigree.

And I'll just point out that that's a -- the staff looks at that as part of GDC 26 and 27. And that's eval -- GDC 26 is evaluated in Chapter 4. So, we'll have more discussions on that.

More or less the answer is, there's two elements of GDC 26 that requires independent means of using diverse principals for controlling reactivity. First is control rods, which are safety related.

And then the second one is for controlling rate of reactivity changes. And that's for our plan in normal operations.

And so in this case, CVCS could be credited for meeting that portion of the GDC. It doesn't specify that it has to be safety related.

And then GDC 27, NuScale is seeking an exemption for that. And that's evaluated in Chapter 15.

CO-CHAIR SKILLMAN: Yeah. And what kind of drove my comment this morning is my recognition that you -- that NuScale is seeking an exemption from GDC 27 and 33.

And those are basically key structural components for PWRs in terms of shutdown. And so here we have a proposed new design where the designer is saying, I really don't need that boron.

And guess what, I really don't need a make up system for a small break. I don't need the boron because I've got another way to shut it down. And I don't need Generalized Design Criteria 33 because the only small breaks I'm going to have are in the steam system.

And I'm choking on that. It would seem to me that a responsible reactor operator would say, I don't care what those words are. I want a backup shutdown system. And I want a pump that's going to put that boron in there when I tell it to go in there.

It would just seem to me that that would have been the construct for a passive design. And the addition of the boron does not need to be high volume. It just needs a little bit.

That's why we have the slip. That's why the newer reactor designs have direct vessel injection.

Well, to me it's just counterintuitive. And it seems like a very simple adjustment to what is a relatively robust design. Thank you.

MR. NOLAN: I understand. A lot of the long term review areas, that's something that we look at in Chapter 15.

And so, I'm sure we'll have lots of discussions on that in the near future.

CO-CHAIR SKILLMAN: Thank you.

MR. TESFAYE: Okay. Any other questions to Ryan?

(No response)

MR. TESFAYE: Thank you Ryan. And again,

we don't have any other open items in the rest of Section 9.3. And Section 9.4 HESE, we don't have any open items.

So, if you don't have any questions on those sections, I'd like to move onto the next section that has open items. Which is in Section 9.5.3, lighting. Fanta?

MS. SACKO: Okay. My name is Fanta Sacko, and I'll be discussing Section 9.5.3, lighting systems.

The NuScale design plan lighting system is a non-class one system that is not safety significant. Not risk significant and non-safety related.

And it includes normal pin lighting, emergency can lighting, normal and emergency main control lighting.

The normal pin lighting provides illumination on the plant site and for plain buildings. The emergency plan lighting provides illumination outside the control room or for loss of normal lighting.

The normal and emergency main control room lighting provides illumination under all operating MENs testing and emergency conditions in the MCR.

Staff reviewed the FSAR to determine

whether the plan levels are adequate in all ten areas. And whether the PLS can operate without adversely impacting the operational control and maintenance of FSCs.

We conducted these reviews in accordance with SRP Section 9.5.3 and NUREG 0700. In summary, we had one open item related to Reg Guide 175, which is the physical separation between non-safety related lighting circuits and safety-related circuits.

And we had one confirmatory item. So the completion of the staff's review of the plain lighting system is awaiting the completion of Chapter 8 open item regarding GDC 17 and 18 exemptions.

So, like I said, the open item in this section relates to the physical separation between the non-safety lighting and the safety-related circuits.

Regarding the staff verifying that the design does not require a safety-related power. The confirmatory item of plan illumination levels will be addressed in phase four of the FSAR.

This concludes this slide presentation.

MR. TESFAYE: Thank you Fanta. Any questions to Fanta on lighting systems?

(No response)

MR. TESFAYE: Okay. That completes our

presentation. But I'd like to add a couple of things on schedule. There was a question on when are we going to be completing the pass exemption requests.

Pass exemptions as all the other exemptions, are reviewed in the Chapter that it's in. And it's a phase four activity.

And phase four has begun for this chapter, of it will be completed in accordance to the Chapter 9 schedule, Chapter 9/Chapter 12 schedule for phase four. We do have a schedule.

CO-CHAIR REMPE: Thank you. And thank you for a very succinct presentation. Because we're back on schedule. I hadn't expected that.

MEMBER MARCH-LEUBA: I wanted to make one question. Just to put it on the record, because you don't know the answer of this.

I am looking at the design diagram of the CVCS. Which is Figure -- it's on page 9.3. That's 84, Figure 9.3.4-1. Which is a diagram of the chemical boron control system.

And it shows that there is a continuous sampling line on the discharge that comes from the vessel. So, we know what the concentration is in the vessel by measuring this.

But, it does not show a sampling line on the injection line, unless it's one of these heating ones that isn't a sample.

MR. NOLAN: It's just downstream of the regen heat exchanger. It's a normally closed.

MEMBER MARCH-LEUBA: The one for PSS?

MR. NOLAN: Yeah.

MEMBER MARCH-LEUBA: Okay. I take it back.

CO-CHAIR REMPE: Thank you. If there aren't any other questions, let's invite NuScale to come back and discuss Chapter 16.

(Off mic comments)

CO-CHAIR REMPE: So Ross, are you ready to begin?

MR. SNUGGERUD: Yes. All right. Hello, good afternoon. My name is Ross Snuggerud. I have a Master's degree in Nuclear Engineering.

I have been in the industry for 25 years. I spent 15 of those at a PWR in Michigan. And I've spent the last almost 11 years at NuScale.

MR. GROSS: My name is Carl Gross. I've been in the industry for 39 years now, including old licensing under Part 50 at Palo Verde.

I've worked in the operations and licensing.

CO-CHAIR REMPE: Carl, you're going to have to get closer.

MR. GROSS: Worked in operations and licensing out there. And have been mostly in licensing and operating programs since then.

Worked at about 25 commercial reactors and five or six DOE sites in nuclear safety and then TSR development there.

MEMBER BLEY: Please try to stay closer to the mic.

MR. GROSS: Sorry.

MEMBER BLEY: Because you disappear as soon as you back off a little.

MS. FOSAAEN: Again, Carrie Fosaaen, NuScale Licensing Supervisor for Chapter 16.

MR. SNUGGERUD: All right. So today we are going to talk about NuScale's Chapter 16. Which generally relates to the development of the technical specifications that will be used by the COLA applicant to submit specific technical specification for their facility.

The technical specifications -- who runs the slides? Do I? The technical specifications that we've built are designed to support individual modules.

So, the tech specs will behave for the operator much like they do in the current industry. There would actually technically be 12 copies of the technical specifications. Our intent would be that they are all the same.

But they deal with the technical specifications on a module by module basis. Shared systems like the pool would potentially cause actuations of the technical specifications across multiple modules. But they would be addressed individually.

Some of the things that are unique about the development of technical specifications for the NuScale design are the fact that we have a simpler, natural circulation design without the primary coolant pumps.

We have passive cooling systems. Both the ECCS and the DHR system. We have a completely digital module protection system.

We don't require any safety related AC power or any need for offsite power which has an impact on the tech specs. We don't have any credited HVAC systems and no active support systems.

And we do move the reactor when we refuel. So, all of those things impacted the execution of developing the tech specs.

Because the design is simpler and because we have fewer systems that are incorporated in the technical specifications, the ultimate document that we generated is roughly half the size of the typical set of instructions for a more conventional PWR design.

In developing the technical specifications, we used the criteria of 50 -- 10 CFR 50.36 to evaluate all the plant designs. Which was largely a review of Chapter 15.

And we developed a topical report that went through our evaluation of all of the designs. This was unique for us because NuScale doesn't have a predecessor plant.

So we didn't have a predecessor set of tech specs that we could start and work to modify. We started with a blank piece of paper.

So, we did a complete evaluation of the plant. We used the existing NUREGs for the base designs, both PWRs and BWRs as source input where appropriate.

And we used an operations group that incorporates -- I want to make sure I get the numbers right, but we have 18 previously licensed SROs with over 560 years of experience in operating nuclear facilities.

And that experience went into helping both the development of the technical report and the reviews of the technical specifications.

In addition, we followed the guidance of the standard tech spec writer's guide. And we monitored industry actions.

So, in addition to the standard NUREGs that exist, we also followed tech spec task force work. Both approved travelers and upcoming work that is available to the public.

And evaluated whether or not those changes were appropriate to the NuScale technical specifications. And incorporated what was appropriate to our design where that occurred.

One thing I didn't mention on the slide, but there were -- there's four criteria for 50.36. The fourth one is a criteria that's used because it's assumed to be prudent, not because it necessarily meets any of the prescribed requirements.

And there are two features of the NuScale design that we included for prudency. That's the manual actuation function and the remote shut down function.

I mentioned earlier, and the committee is clearly aware that we do move the module during our refueling process. And so that did change the way we went about developing our mode definitions.

Sometimes the mode definitions get confused because they're used very heavily in plant operations. But, the reality is, the mode definitions are chosen to make applicability of the Section Three tech specs easier for the operator to apply.

So you want to pick transition points that logically follow the plant design, and make it easier for the operator to know which sets of technical specifics apply to any given mode of operation.

As a result, we got rid of the classic mode two. Which is the low power hot mode. Because at NuScale there were no unique requirements that only applied to that mode of operation. So that mode was removed.

We added the mode transition which applies to the reactor module while it's being moved and prior to decoupling any of the bolts associated with the lower portion of the reactor vessel.

So basically from the time you complete your shutdown and prepare the module for movement, until the time that you start manipulating bolts associated with the reactor flange, that's the transition mode.

And then finally, we have a refueling mode. Which is not unlike the existing industry.

Just for clarity, the CRA is a control rod assembly, see --

MEMBER RAY: Wait a minute. You have a refueling load that's not unlike existing industry?

MR. SNUGGERUD: At the point that --

MEMBER RAY: What do you mean by unlike --

MR. SNUGGERUD: At the point that we're in refueling, the primary things that the tech specs are concerned about, are monitoring for reactivity changes in the core.

MEMBER RAY: Just tell me what you meant by that statement.

MR. SNUGGERUD: So, the kinds of things that are required in our Mode Five, look a lot like the kinds of things that are required in a refueling Mode Five for other plants.

Obviously, ours is compressed. But it's talking about the same types that evolutions is talking about, picking up and moving fuel inside a core. And what's necessary to ensure that that's a safe activity.

I understand that my choice of words may not have been appropriate.

MEMBER RAY: Those words were not appropriate. Period.

MR. SNUGGERUD: From what basis? Help me understand why you feel that strongly.

MEMBER RAY: Well, you've done refuelings before. You're moving fuel elements one at a time in and out.

This is moving an entire core and a reactor and its vessel with a core, you know, the entire core inside of it. That's in and of itself completely different.

MR. SNUGGERUD: I agree. And that's the transition mode. Which is unique and new to NuScale.

I transitioned from the transition mode, which incorporates all those activities you just talked about, to the refueling mode, which looks much more --

MEMBER RAY: Oh, after you get this --

MR. SNUGGERUD: Yes.

MEMBER RAY: I apologize. My fault.

MR. SNUGGERUD: I just want to make sure we are talking trans --

MEMBER RAY: I thought you were still

moving the reactor vessel when you said that.

MR. SNUGGERUD: Yeah. I'm sorry.

MEMBER RAY: Okay.

MR. SNUGGERUD: The transition mode is completely unique to NuScale.

MEMBER RAY: Okay.

CO-CHAIR SKILLMAN: Far right column. Indicated reactor coolant temperature, you reach Fahrenheit.

What is the distinction between all versus any? What do those words mean, please?

MR. SNUGGERUD: So, there is clarification in the basis document. We're talking about all safety related T colds and T hots.

Or any safety related T cold or T hot since you can't always guarantee that the hots will be hotter then the colds.

CO-CHAIR SKILLMAN: Okay. Thank you.

MEMBER MARCH-LEUBA: And mostly you said that any temperature indicated that this is less than 420? Is that --

MR. SNUGGERUD: All temperatures have to be less the 420 in order for you to be in Mode Three.

MEMBER MARCH-LEUBA: Okay.

MR. SNUGGERUD: Because there are certain pieces of equipment that are no longer required. But you have to prove that you've reduced the energy to below 420 at all locations. All operable indications.

MEMBER MARCH-LEUBA: And do you also have to guarantee that you -- the effect is .99? Because when you get really, really cold, you may not satisfy .99.

MR. SNUGGERUD: So, the --

MEMBER MARCH-LEUBA: How do you know where you are? Because what I'm saying is, I'm an operator and would like to know what temperature would be a threshold that would make me unsafe.

MR. SNUGGERUD: So we would have a guidance document not unlike what the PWR that's used in the industry that would tell the operators for a given temperature what the required boron concentration is.

It is demonstratable that under most cases the boron concentration necessary to meet this requirement is consistent with the operating boron concentration. Assuming that all of the control rods have fully inserted.

For refueling though, obviously we haven't --

MEMBER MARCH-LEUBA: At any temperature?

MR. SNUGGERUD: Right. You can go to a much lower temperature. And our expectation is that there would be the addition of boron to the system to do a normal plant shutdown to meet that requirement.

MEMBER MARCH-LEUBA: I got that impression too.

MR. SNUGGERUD: Yeah.

MEMBER MARCH-LEUBA: What we're concerned is, what equipment are you going to use to accomplish it?

MR. SNUGGERUD: We will be using the CVCS system.

MEMBER MARCH-LEUBA: A non-safety related CVCS?

MR. SNUGGERUD: Yes. Just like existing facilities.

MEMBER MARCH-LEUBA: You might want to put that on the record.

MEMBER KIRCHNER: I'm just puzzled. Why do you say our expectation? Why isn't this more definitive?

You could be at the end of a burn up cycle of a module and down to, if I remember right from Chapter Four, somewhere close to 10 ppm when you finish that cycle.

So, I would say that boron injection is required, not expected. Because your -- beyond the criteria --

MR. SNUGGERUD: The design is designed such that when all the rods insert, we maintain shutdown margin at all temperatures.

But, there are specific cases, and if you assume a rod missing and those types of things, where you would need slightly more margin.

In addition, our refueling process will involve removing control assemblies. And some of those assemblies will have control rods.

So the process includes margin for that boron. And because we want to ensure that the boron at the bottom of the vessel meets the boron requirements for the refueling operations, we would be adding boron during the shutdown process to ensure that.

I'm trying to be a little bit careful because the reactivity --

MEMBER KIRCHNER: But we asked -- where there --

MR. SNUGGERUD: That's where the expectation comes in. Because technically --

MEMBER KIRCHNER: I think it's a

requirement to do it safely. It's not an expectation.

MR. SNUGGERUD: I understand your point.

MEMBER KIRCHNER: Yeah. And therefore I would ex -- there I would use the word expected to be specifically clear either in tech spec space or procedures. Preferably tech spec.

MR. SNUGGERUD: The tech specs do provide requirements for concentrations of boron in shutdown modes.

MEMBER KIRCHNER: Right.

MR. SNUGGERUD: And what the required margins are. So if those couldn't be demonstrated with the boron concentration that had been exemplified during operation, additional boron would need to be added.

And we have the capability of doing that using the chemical volume control system.

MEMBER KIRCHNER: By using tech spec space.

MR. SNUGGERUD: It is in tech spec space.

MEMBER KIRCHNER: So there's no longer an expectation, it's defined.

MR. SNUGGERUD: Fair enough. Yes. The -- I mean, the mode definition is a piece of that. The boron requirements within tech spec are an additional piece of that.

MEMBER MARCH-LEUBA: We will review and -- we will be reviewing Chapter Four next month, I believe.

And I'm just giving you something -- maybe you could use some feedback to your colleagues. We hear that there will be a lot of questions about certain margins.

And it would be nice if they have some backup calculations that we can see. So, some actual numbers and actual calculations with actual assumptions. Not just statements of expectation.

MR. SNUGGERUD: Understood. You'd be more interested in what those numbers look like under different circumstances with different assumptions.

MEMBER MARCH-LEUBA: Not just the numbers, but how you arrived at those numbers. Because shutdown margin is the most important thing in a nuclear reactor.

I guess being an operator, you know that. Right? It cannot be an expectation. And it has to be well analyzed, very -- and how with uncertainty determination for it.

MR. SNUGGERUD: All true. Any other questions about the mode definition table that I have up there?

I was just going to clarify that CRA stands for control rod assembly. CVCS you probably recognize as chemical volume control system. And CFDS is the containment flood and drain system.

Part of what those are touching on are things that could potentially change the concentration of the water inside the module after you've established the boron concentration you intended to have in place for transition.

CO-CHAIR SKILLMAN: Is there a need -- this is an open question. And I don't -- I don't have an expectation for an answer.

But, is there a -- should there be an additional mode for NuScale where one of the modules or several are somehow compromised or in casualty?

These are operating modes. Most of us who have been around the industry kind of recognize this.

But, in this plant with 12 live cores, is there another mode that is -- ought to have its own unique name tag so that the leadership of the plant and the operators, the men and women that are working there, recognize this is different.

Different rules may apply. Or there may be some different ingress, egress, load handling, ventilation. Some other kind of requirement.

This is not a norm -- that's not the right word. This is so different. Does it deserve some kind of a moniker where you say, you know, this one's -- it's not in mode one, two, three, four, or five. This is in mode 12. And mode 12 means special rules apply.

MR. SNUGGERUD: So for the unaffected modules, the answer is no.

CO-CHAIR SKILLMAN: And I agree with that. Yeah.

MR. SNUGGERUD: So, for the affected module, and this is true for existing facilities, when you're in an accident, then you're in your emergency procedures and you're following your emergency procedures.

The technical specifications are designed to ensure that the plant started prior to the accident in a condition that allowed it to be in an analyzed position for the accident that occurred.

To the extent they can, operators will continue to try and maintain the plant within the requirements of technical specifications.

But, once you get into an accident, the function of the technical specifications to some extent, has been met.

CO-CHAIR SKILLMAN: But accident has a unique meaning also.

MR. SNUGGERUD: Absolutely.

CO-CHAIR SKILLMAN: So, maybe it's not an accident. Maybe you've got a module that's stuck. Your cable's failed.

Where the module is located is precluding some other operation. Or it's over in one of the storage stands. Your cable's unreaved.

You've got six of a number of bolts or detention. So, it isn't a full casualty. You don't have an accident. You might not be leaking radioisotopes.

But there is an alert in the plant where folks say, things are not right right now. We've got to get this situation modified. But it's none of these modes.

Don't answer.

MR. SNUGGERUD: You have a --

CO-CHAIR SKILLMAN: I'm just --

MR. SNUGGERUD: I've kind of given what I can answer.

CO-CHAIR SKILLMAN: Yeah. There's a -- you're in a different world in this plant.

MR. GROSS: Yeah. The role of tech specs though is not off normal conditions such as that. It's to keep the plant, as he mentioned, aligned with the assumptions before the event occurs and so that it will respond as appropriate if it hap -- if something occurs.

There is one different one, and that's of course to identify leakage from the RCS system. The -- those, the conditions you're describing would be off normal emergency procedures.

By definition if it affected the tech -- if it affected the safety analysis, we would have to take action under the emergency plan of procedures.

If a module is just not movable, it really doesn't have an effect. There's no special actions to take before we began moving or entering or changing conditions, all the other modules would have to comply with their tech specs.

And for example, the shutdown margin, the mode four transition definition, there --

MEMBER KIRCHNER: I think there might be a -- I don't know if you were thinking this. But, I'm thinking that if you build your plant and say install half the module, so you have six.

Now you're bring six more in. Or you put four in. There are a few instances where you do have shared systems. Boron addition comes to mind and such.

So, where you have special requirements in terms of isolation as you tap into those systems. So --

MR. GROSS: I don't believe any of those systems other then UHS are credited.

MEMBER KIRCHNER: Um-hum.

MR. GROSS: So, as long as we maintain water in the pool with the appropriate boron and the right temperatures, we've met our requirements for our safety analysis, which is what Chapter -- or what tech specs are driven to protect.

CO-CHAIR REMPE: Well, which mode is when you add another module? Is it transition?

Where am I when I'm bringing in a new one?

MR. SNUGGERUD: Yeah. So when you're bringing in a new module, it's not fueled yet.

So, it wouldn't technically be in technical specifications at that point. Once you put the lower pieces into the plant, and assuming you had completed whatever hot -- or start up testing and validation you had completed.

And you started loading in fuel, you would be in refueling.

CO-CHAIR REMPE: So, the other -- you've divided it up. And I've forgotten the term you had.

MR. SNUGGERUD: It's in the inspection station. Yeah.

CO-CHAIR REMPE: And then you put the fuel in there and --

MR. SNUGGERUD: No.

CO-CHAIR REMPE: Then bring it down. Are you bring -- what --

MR. SNUGGERUD: So, when you bring a new module in, obviously there's some testing that we're going to do before we ever put any fuel in it. Right?

We're going to do some training work and those kinds of things. But you get to the sixth, seventh module, you bring the components in.

You bring the lower containment vessel in. Low lower reactor vessel in, and then the upper reactor vessel and the upper containment.

The lower reactor vessel goes in its tool. The lower containment goes in its tool. At that point, with the upper module in the inspection bay, there's no difference between that and a refueling.

And as soon as you load that first bundle into the bottom of the reactor vessel, you would be entering mode five for refueling. And the equipment required to support refueling activity would have to be operable.

CO-CHAIR REMPE: And then you transition it back to its station. But -- and you hook things up just like you would the others.

MR. SNUGGERUD: And then before you left mode three, the required equipment would have to be available to support hot shutdown and so on.

CO-CHAIR REMPE: Okay. Thank you.

MEMBER SUNSERI: So the way you -- the way you have that mode five defined up there, it is equivalent to at least in my view, what plants call at all times right now.

Is that what was your intention?

MR. GROSS: Yeah. I believe it's slightly different. At all times is whether there's fuel in the core or not.

MEMBER SUNSERI: But that says less than one bolt fully tension.

MR. GROSS: So he's --

MEMBER SUNSERI: You might look at that.

MR. GROSS: I'll take a -- thanks. I'll take that back.

MR. SNUGGERUD: It's a good --

MEMBER BLEY: Karl said something that I'm not sure I read anywhere. And I'm not sure I heard you correctly.

But I think you were saying, if you have a module that's in a -- some kind of failed condition, that before you do anything with that module, all the others have to be meeting their tech specs.

Is that what you said?

MR. GROSS: All the others will meet their tech specs contin -- they need to -- everything needs to stay within its tech spec bound limits all the time.

MEMBER BLEY: Yeah.

MR. GROSS: If there's fuel in the pot basically. The -- if there's an upset module, it's by definition outside a -- pretty much outside of tech specs.

Not all -- that's not always true. But in general, that's where it is. And not fully out necessarily, but.

MR. SNUGGERUD: But the operators continue to apply tech specs to the --

MR. GROSS: That's right. Exactly.

MR. SNUGGERUD: Other parts of the plant. I think again, maybe part of your question goes to the mode definitions are for the applicability of the technical specifications.

They're not intended to describe all potential plant conditions. They're described -- intended to describe the required equipment in given plant conditions.

So, if I was in the condition where I had un-tensioned some bolts and six bolts I couldn't get un-tensioned, I know where I'm at on this table. The plant might be in some unique configuration that would require response of the operations organization and the maintenance organization.

But, I have defined where I'm at in the table. And that tells the operators what equipment is required under those conditions.

And then you get the multi-module issue, which you kind of hinted at as well. But now the other modules can't get to the refueling station because I have a module in the refueling station.

Those types of recovery plans are things that we're talking about in the design and development of our recover -- of our refueling equipment to make sure that the challenges to the other plants aren't experienced.

But, knowing that you can't refuel would preclude you from wanting to get into transition. So you would stay in your safe shutdown condition.

And that would define the requirements for that unit while it was sitting in its module space waiting for the other issue to be resolved.

CO-CHAIR SKILLMAN: I was just thinking back to what we got into when we started refueling at TMI-2, and we thought we had it all sewed up nice and clean, nice and tidy, and every time we turned around, we were in a new situation, trying to figure out where to go from here.

And a lot of the stuff, we were doing on the fly. We were writing procedures, we were writing stuff like this, trying to navigate our way through a safe and defendable path.

And then, here we are at the front end of this unique design, maybe a little more effort would stave off what would be really a complex situation in the future.

MR. SNUGGERUD: And it is our intent and our hope that the effort that we've put into this, I've been working on tech specs, at least part time, for all ten years I've been at NuScale, would, to a large extent, do that. I can't claim that we have thought of everything, because that would be a false claim.

But I did try and bring out the fact that we do have a substantial staff of previously licensed operators, and we have a very competent technical staff in the Licensing Department. It won't catch everything, but we have made a very big effort.

Like I said, we didn't just go through the PWR tech specs, we looked at the BWR tech specs, we looked at foreign types of things, to get input to make sure that we were evaluating this new design in as many ways as we could.

CO-CHAIR SKILLMAN: Thank you.

MR. SNUGGERUD: Let me get back into my presentation here. So, this is going into the sections of the technical specification that actually hold the limiting conditions of operation. Section 3.1 and 3.2 are not terribly different from things that you would normally see, so we have left those out, and I've jumped to Instrumentation.

I mentioned in my opening that we have a completely digital module protection system. And as a result, that changed, to some extent, the way we've organized the Instrumentation section of our technical specifications.

The manual actuation function is there, at

3.3.4. That was one that we don't credit in our Chapter 15 analysis, but as an operator, and our operating staff, we agreed that the operators need to have a way that's independent of the digital system to affect the actuation of the engineered safety features and the reactor trip system. I don't know if there's anything else you want to add on this slide?

Passive Core Cooling is another section that is unique to NuScale, in that our cooling systems don't involve the pumps. So, we have the emergency core cooling system, the decay heat removal system, and we put the ultimate heat sink under passive cooling because it obviously supports those two systems. There are elements of the ultimate heat sink that also support refueling operations, but they're covered under 3.5.

Containment Systems, again, is a little bit different. You're not going to find personnel hatch controls for our technical specifications. We're largely looking at the containment itself, its function as an isolation vessel, and then, the valves associated with isolating the containment.

Plant Systems, again, a smaller set. We have main steam and main feed water, because the isolation of the steam generators is important to the function of DHR and for the mitigation of steam generator tube ruptures.

And then, we have the in-containment secondary piping leakages, which is a specification that we added after working with the staff to ensure that the assumptions of our leak-before-break methodology are in place during normal operations.

3.8 typically is Electrical. For NuScale, we don't have that section. What we do have is Refueling Operations, which is normally 3.9.

MEMBER CORRADINI: So, is 3.8 then the transitional and refueling?

MR. SNUGGERUD: So, it's --

MEMBER CORRADINI: I'm trying to understand where movement fits into this and limiting conditions --

MR. SNUGGERUD: The movement is mostly controlled by the early technical specification on Reactivity Requirements. So, when we're moving the module, we ensure, prior to disconnection, that adequate boron is in place, and we ensure that adequate decay time, which is in 3.8, is in place.

And we ensure that the module temperature and systems are aligned as designed, to ensure passive cooling during transition. But in transition, specifically, there aren't any technical specifications, beyond ensuring that those things that were set up aren't changed.

MEMBER CORRADINI: And the conditions you want?

MR. SNUGGERUD: Yes, correct.

MEMBER MARCH-LEUBA: So, is the nuclear instrumentation live instrumentation? I mean, you have a cable pulled in for the incore detectors?

MR. SNUGGERUD: So, the incore detectors are only meaningful at power. So --

MEMBER MARCH-LEUBA: But that's better than not having nothing?

MR. SNUGGERUD: The types of self-powered incore detectors that we are using would not provide any indication.

MEMBER MARCH-LEUBA: So, do you add transition source level detectors?

MR. SNUGGERUD: So, we have source level detectors at the operating position and at the refueling position. Between those two, it is our position that reactivity changes are precluded.

MEMBER MARCH-LEUBA: Okay. So, the nuclear instrumentation specs don't say any -- say you don't need anything --

MR. SNUGGERUD: That's correct.

MEMBER MARCH-LEUBA: -- during transition?

CO-CHAIR SKILLMAN: Do the tech specs require that the reactor be borated to the 1,800 or 2,000 PPM before it is disconnected and then moved? I mean, clearly, you're depending on the rods to be bottomed, but are you depending on full-load boron as the preventative, so that you do not need neutron monitoring during module movement?

MR. SNUGGERUD: So, there is no means of changing out the fuel during the transition.

CO-CHAIR SKILLMAN: Oh, I understand that.

MR. SNUGGERUD: There is advantageous to have the boron concentration at the bottom of the vessel, at full boron concentration, prior to starting refueling activities, if your intent is to load new fuel, which requires the 1,800 PPM. The fuel that you're taking out, doesn't.

The current tech specs would require that you know that you have 1,800 PPM borated water in the lower reactor vessel. It doesn't currently specify the details of how you do that.

The easy one, which would be consistent with the current industry, would be to set up those conditions prior to taking it apart. Another option would be to find a way to ensure that you had flushed that vessel prior to moving anything and taking samples appropriately.

We haven't specified that level of detail in the tech specs, other than to ensure that the requirements for the new core would be met prior to the new core being inserted into the lower reactor vessel.

CO-CHAIR SKILLMAN: So, you're basically allowing credit for burnup, as reactivity depletion --

MR. SNUGGERUD: Yes.

CO-CHAIR SKILLMAN: -- and not necessarily requiring full-load boron before you move the module?

MR. SNUGGERUD: That's correct. And the existing fleet could do the same. Again, they know they're going to put fuel back in their modules and they don't want their cold legs sitting there with boron concentrations that don't meet the refueling boron concentrations for the new core.

So, they borate to refueling concentrations, not because the core they're taking out needs them to, but because the core they're putting in requires it. Same thing applies to NuScale. And the NuScale facility will have the same expectations that the existing facilities have.

CO-CHAIR SKILLMAN: Yes, except that you borate it up before you take the head off, and so, you're fully borated before you touch your first fuel assembly.

MR. SNUGGERUD: That's correct. At our plant, when you take the head off, you're going to be exposing it to a seven million gallon pool that meets all boron concentration requirements.

MEMBER MARCH-LEUBA: And in the operating fleet, they have pumps that they can kind of mix the boron.

MR. SNUGGERUD: Yes, but the pumps don't hit all portions of the shutdown plant, depending on the plant's design. Which, again, is why typically, and at the plant that I operated at, we borated to the new core's boron concentration before starting refueling processes.

MEMBER MARCH-LEUBA: But I've been reading in-between lines that you expect the boron concentration in the vessel to be non-uniform. And --

MR. SNUGGERUD: That's not what I'm saying.

MEMBER MARCH-LEUBA: -- how do you know what you --

MR. SNUGGERUD: What I'm saying --

MEMBER MARCH-LEUBA: -- do you say the

bottom of the vessels or --

MR. SNUGGERUD: The easy way to do it would be to do the same thing at NuScale, right? It would be to borate the module to 1,800 PPM on the way over.

As you had pointed out, during shutdown, there's going to be ample space made during the cooling process, that you could add as much boron as you chose to. I don't want to tell you that that's what they're going to do, because the tech specs don't force them to do that, and technically, it isn't required.

What is required is, before they touch a bundle and start putting new fuel into the reactor, that they ensure that that lower pole of the reactor that's sitting in the pool is at the boron concentration that they intended.

MEMBER MARCH-LEUBA: How they do that if instrument is on the discharge line of the CVCS?

MR. SNUGGERUD: So, as I suggested, the easy way to do that would be to borate the entire module prior to ever doing it.

MEMBER MARCH-LEUBA: That was my --

MR. SNUGGERUD: But there are other ways of doing it --

MEMBER MARCH-LEUBA: That was my

expectation --

MR. SNUGGERUD: -- and the tech spec --

MEMBER MARCH-LEUBA: -- before you started talking.

MR. SNUGGERUD: Yes. The tech specs don't tell them how they have to do that, they tell them what they have to do. I would agree with you, that would be the easy way.

MEMBER MARCH-LEUBA: Yes, if you don't have a sensor up there, that's the only way.

MR. SNUGGERUD: I can envision means by which you could ensure turnover of that lower module. It's not a very big space, you have hot fuel in it. It's going to be pulling water down the down comer and pushing it up through the core, just because of the flow design of that module.

I don't want to tell you that somebody in the future won't do the calculations to make that argument and do the demonstrations to prove that that's appropriate, but I haven't done them.

MEMBER MARCH-LEUBA: I have a running concern with the staff about how well the boron mixes when you don't have any flow, and indeed, experiments for BWR show the boron does not mix and it goes all to the bottom of the vessel. And this will have to be addressed somewhere --

MR. SNUGGERUD: Yes.

MEMBER MARCH-LEUBA: -- I expect Chapter 15.

MR. SNUGGERUD: And I've heard your concerns and I know NuScale is aware of them.

So, this is just a brief, the staff will cover this in more detail, but we have received approximately 60 Chapter 16 RAIs. A significant number of them have been responded to. We do have some outstanding and we do have some that were recently provided.

I will also mention that the number of RAIs isn't exactly the same as the number of questions, because one example up there, a single RAI had 80 subparts. So, we have more than 60 questions about the tech specs. The staff has -- well, I guess I have another slide that kind of covers some of the high level things.

There were a lot of discussions about how we chose to implement standard tech spec content into our design. And so, there were some formatting and some deliberations about the most appropriate way to do that, to ensure that the understanding is consistent with the existing fleet.

We got a number of questions related to instrumentation, which makes sense, again, as NuScale has a unique and newly designed module protection system, which is the subject of a Topical Report. And that spawned additional questions related to response time testing, because the barriers or the designations between where you do your testing are a little bit different for our design.

We discussed leak rate monitoring. Our containment is different, it's dry, it's in a vacuum. So, what are we going to use to give us evidence and make the operators aware of any leakage before it becomes a problem? We had discussions about that system.

And then, we added the RCS-specific activity limit that addresses dose rates for operators, which is something that the staff asked for.

And then, lastly, we added a tech spec, most recently, related to any containment secondary piping leakage that supports our leak-before-break methodology.

So, in conclusion for my side, NuScale has spent a lot of time to develop technical specifications and bases that reflect our design and the analysis and the operations as we envision them.

We've tried to ensure the safety in licensing basis will be implemented as designed, aligned with the standards that the industry uses, take into account industry experience, and prepare a document that will be easy for the COLA applicant to finalize, consistent with the intent of both the staff and NuScale.

CO-CHAIR REMPE: If there aren't any more questions, we're again running behind, and so, I'm going to ask the staff to come up.

MR. SNUGGERUD: Okay.

CO-CHAIR REMPE: Thank you.

MR. TESFAYE: Good morning. I'm also the Chapter 16 PM. Good morning, again. Craig Harbuck is my to my right and he's the technical reviewer, as you may all know. Greg's not here, he's our lead PM.

Staff has completed a Safety Evaluation Report and submitted it about a month ago. This report contains several open items when it was issued, but a good portion of the open items have since been closed.

The remaining open items will be addressed in Phase 4 of the review. Craig will discuss in detail the open items that have been closed and the remaining issues that will be addressed in Phase 4 and his overall technical findings.

Now, at this time, Craig has asked me, if it is okay with Committee, he would like to go through the presentation and then, the Q&A, if that's okay with you, in the interest of time.

CO-CHAIR REMPE: Thank you.

MR. TESFAYE: Okay.

MR. HARBUCK: I'll be discussing open items that are still open.

CO-CHAIR REMPE: Okay. So, please scoot very close to the mic.

MR. HARBUCK: Okay. Can you hear me now?

CO-CHAIR REMPE: Yes.

MR. HARBUCK: It's good? Okay. My name is Craig Harbuck. I'm in the Technical Specifications Branch and have been for a long time. Next -- no, previous slide. Previous slide.

This is the people who I was able to identify that were participating in supporting this review. If I've left anyone out, I apologize. Next slide.

The outline of this presentation, it consists of an overview of the tech specs and then, a number of technical topics that are mostly related to open items.

A brief overview of the tech spec chapter, sections, and subsections. This outline shows that the tech spec chapters and sections are mostly organized consistent with a PWR STS NUREG.

Notice that the NuScale tech specs are missing the LCO that allows operation for a limited period of time with an inoperable piping support snubber. That's normally LCO 308, but since they're not having that one, that's what it will be.

Also, NuScale tech specs do not have a Chapter 3 section for electrical power, for reasons we all know. Next slide.

This slide lists the Chapter 3 LCO subsections that were the result of applying the LCO Selection Criteria 50.36. NuScale's evaluation of its design and safety analyses against the LCO Selection Criteria is described in their report entitled Technical Specifications Regulatory Conformance and Development. Currently, we're up to Revision 1 on that report.

There's two LCOs that satisfy both Criterion 2 and Criterion 3. That would be boron dilution control and the ultimate heat sink.

I would also point out that LCO 3.4.1

differs from the equivalent STS LCO by only including verification of reactor coolant system flow resistance within limits at the beginning of each operating cycle, right after refueling.

And this is due to the use of natural circulation, where coolant flow is a function of core power, and therefore, we don't include a 12-hour surveillance that you typically have for verifying RCS flow.

Also, as mentioned previously in NuScale's presentation, they're using a leak-before-break evaluation, to ensure an in-containment secondary system line break is essentially precluded from occurring. To ensure this remains valid, LCO 3.7.3 is established, to specify a limit on the rate of secondary leakage into containment. Next slide.

This slide lists the sections and subsections of Chapter 4, Design Features, and Chapter 5, Administrative Controls.

In the Procedures Section, that's 5.4.1, NuScale agreed to add a requirement for procedures to control the availability and reliability of systems covered by the Owner Control Requirements Manual, because these systems are not covered by the Reg Guide 1.33 required procedures or the Quality Assurance Plan procedures.

And examples of systems that we would expect to see covered by the OCDM are control room habitability, ventilation, the CVCS system, demineralized water system, electrical power, the containment evacuation system, the core flood and drain system, and a number of others.

We'd also like to point out that the NuScale tech specs include a surveillance frequency control program, which is Risk-informed Tech Spec Initiative 5B, which is implemented using Traveler TSTF-425. Next slide.

CO-CHAIR SKILLMAN: Craig, if I could please, on that same slide.

MR. HARBUCK: Okay.

CO-CHAIR SKILLMAN: Under the procedures that you just enumerated, which is 5.4.1, Frank, F, would you also expect the procedures for the operation of the heavy load handling system, the crane, to be binned in that same category?

MR. HARBUCK: I'm not sure, that's not normally included in the procedures that are required. Well, let me put it this way, if that equipment was deemed to be safety-related, perhaps it would be covered already by --

CO-CHAIR SKILLMAN: Do you think it's safety-related?

MR. HARBUCK: -- Reg Guide 1.33.

CO-CHAIR SKILLMAN: Do you think it's safety-related? The crane?

MR. HARBUCK: Oh, I do think so, yes.

CO-CHAIR SKILLMAN: So, do you think the procedures --

MR. HARBUCK: But --

CO-CHAIR SKILLMAN: -- may be in here?

MR. HARBUCK: Well, everything they do at the plant should have procedures. What I'm concerned about in the tech spec arena is to ensure the things that we've normally included in tech specs, but that are not being called out as meeting any of the criteria, that there be some means of -- I mean, a tech spec requirement for there to be procedures. Otherwise, it's hard to know what level of control we're going to have, since none of these things fall within the other non-tech spec controls.

CO-CHAIR SKILLMAN: Okay, I'll leave it there. Thanks.

MR. HARBUCK: Okay.

MEMBER BLEY: I need a little help, I'm not well-versed here. Your Slide 5 is the Chapter 16 sections, yes?

MR. HARBUCK: Yes, that's correct.

MEMBER BLEY: And the last two slides you showed, your 6 and 7, are from Part 4, which as I think I understand it is going with the Generic Tech Specs in the NUREG and seeing which ones apply, is that right?

MR. HARBUCK: Well, what Slide 6 does, it lists the LCOs that were identified as meeting the criteria and one that doesn't, in 50.36(c)(2)(ii).

MEMBER BLEY: And if all these meet the criteria, except for one, shouldn't they all be in 5? I'm just a little confused about the relationship between --

MR. HARBUCK: Oh, the --

MEMBER BLEY: -- Part 4 and Chapter 16.

MR. HARBUCK: We refer to tech specs as having chapters, sections, and subsections.

MEMBER BLEY: Yes.

MR. HARBUCK: On Slide 5, I've gone down to the section level.

MEMBER BLEY: Okay. So, you've gone a layer deeper?

MR. HARBUCK: That's right.

MEMBER BLEY: You also said there's no

electric power?

MR. HARBUCK: There are no LCOs for electrical power, except for the Class 1E isolation devices to protect the MPS instrumentation.

MEMBER BLEY: And the reason there's not? I mean, there's something in here on CVCS --

MR. HARBUCK: It is --

MEMBER BLEY: -- is non-safety.

MR. HARBUCK: Well, that had to do with NuScale agreeing to add another provision for procedures --

MEMBER BLEY: Okay.

MR. HARBUCK: -- so we would have something to point to.

MEMBER BLEY: I'm linking back to things that aren't --

MR. HARBUCK: Yes.

MEMBER BLEY: -- in your area here, but --

MR. HARBUCK: Okay.

MEMBER BLEY: -- when we reviewed the chapter on electric power, they, for several reasons, decided not to have 1E electric power system and to have a highly reliable electric power system.

MR. HARBUCK: Yes, DC power.

MEMBER BLEY: That would imply to me that there ought to be something like a tech spec requirement on it.

MR. HARBUCK: The conclusions, I believe, on Chapter 8, were that there were no safety-related electrical systems. No electrical power is required, since everything that actuates in accidents does so by removing electrical power. So, it's my understanding that none of the criteria have been satisfied by the electrical power.

MEMBER BLEY: So, with this highly reliable electric power system, they could have it shut down and it would not affect the tech specs? It could be turned off?

MR. HARBUCK: Well, they might have a hard time operating if they didn't have that.

MEMBER BLEY: Yes, they might.

MR. HARBUCK: I had a thought --

MEMBER BLEY: Go ahead. Chapter 8 isn't finished, by the way, there were a lot of outstanding things.

MR. HARBUCK: Okay. Anyway, this reflects our understanding of the current status of Chapter 8 and its role in the safety analysis.

MEMBER MARCH-LEUBA: Chapter 8? I've been saying Page 8.

MEMBER BLEY: Sorry.

MR. HARBUCK: Okay. That's the end of the overview. Now, we're going to start on the technical topics. And before I begin, just let me follow up on something from the previous presentation.

The mode definition for Mode 4, that transition mode, requires shutdown margin to be whatever the reactivity condition, k-effective less than 0.95, which is going to dictate what your boron concentration is going to have to be set at before you start moving the module.

And then, when you disassemble the module, the requirements on the pool boron concentration in 3.5.3 would dictate the reactivity condition to maintain that requirement.

MEMBER MARCH-LEUBA: Do tech specs say you have to have a value for k-effective? Or do they say --

MR. HARBUCK: No, it says k-effective, I think that's --

MEMBER MARCH-LEUBA: Which you cannot measure during operation.

MR. HARBUCK: It's correlated with -- it's calculated. You know what the boron concentration is, you estimate --

MEMBER MARCH-LEUBA: It's --

MR. HARBUCK: -- how long since you shut down and --

MEMBER MARCH-LEUBA: I wanted to save it for the end, but she will make me shut up, because we'll run out of time.

MR. HARBUCK: Okay.

MEMBER MARCH-LEUBA: First principles, if you insert all rods, at any point in the cycle, but mostly towards the end of the cycle, when boron concentration is very low and the monitored temperature coefficient is very negative, you will have a temperature, a relationship between boron concentration and core temperature, which you regain criticality. So, for every boron concentration, you have a temperature you become critical again.

Why is this not on tech specs? There should be a three-dimension table that tells me temperature below which I cannot go, as function of boron concentration and burnup. And the statement, it says, you cannot reach is.

Tell me what it is and put it in tech specs, because the operator doesn't know what k-effective is, if he can measure the temperature.

MR. HARBUCK: Well, at the moment, these

values are not in the tech specs, they're in the Core Operating Limits Report, which we haven't seen.

MEMBER MARCH-LEUBA: Which you have not seen?

MEMBER CORRADINI: Because that's typical of most PWRs?

MR. HARBUCK: That has been the case since the 1990s, since we adopted this Core Operating Limits Report.

MEMBER MARCH-LEUBA: When will be able to see the COLR?

MR. HARBUCK: I would have to let NuScale answer that.

MEMBER MARCH-LEUBA: So, it's --

MR. HARBUCK: It's probably not something that's part of the DCA.

MEMBER MARCH-LEUBA: But do we know -- can we specify that something like this must exist in the COLR?

MR. HARBUCK: Yes, because the tech spec on shutdown margin says, you have to meet the values that are in the COLR. And then, there's a definition of shutdown margin, which provides the other assumptions you have to make in calculating what your shutdown margin is.

MEMBER MARCH-LEUBA: Okay. We'll talk about this at some point.

MR. HARBUCK: Okay. All right. So, enough of that.

Last thing I wanted to say was, the definition of mode also includes a phrase that says there's got to be fuel in the vessel. So, when you remove all your fuel, you're not in a mode. Okay.

I'll now discuss several topics of interest for which the staff and NuScale have not yet reached resolution. And first up are several defined terms.

The sentence that NuScale proposes to add to the pressure boundary leakage definition, it's indicated by the blue underlined text there, matches an STS change previously proposed by industry that the staff did not accept.

We do not believe that this relaxation, to permit leakage through an isolation device of a fault in the reactor coolant pressure boundary, is warranted. We are also not convinced that there are locations in the NuScale reactor coolant pressure boundary where this allowance could even be used. Next slide.

MEMBER CORRADINI: Can you -- I want to

understand that a little better.

MR. HARBUCK: Okay.

MEMBER CORRADINI: Okay, thank you.

CO-CHAIR REMPE: We can go on, Mike, or do you have a question?

MEMBER CORRADINI: No, I just was trying to catch up.

MR. HARBUCK: The next one is operability. And we believe that the indicated changes of the STS operability definition are appropriate for NuScale. This is a markup of the STS definition.

The most noteworthy difference with the STS definition is the removal of the phrase normal or emergency to describe electrical power. This makes sense, because, as we've previously mentioned, NuScale has no safety-related emergency electrical power sources and, thus, no LCO section for electrical power systems.

Also, notice this definition includes two lists of the various ways tech specs refer to a redundant part of a supported system. These lists should match, and I've added some markup to do that.

The second list should include the term Separation Group, as does the first list, it's a NuScale term. And then, both lists should include

Channel and Division, because the MPS instrumentation has four channels and the MPS RTS and ESFAS actuation logic functions have two divisions. Next slide.

NuScale proposed to admit the STS definitions for reactor trip system response time and engineer safety feature response time, because the digital portion of the MPS will contribute a conservatively allocated response time to the total ESF and RTS response times, and NuScale thereby concludes that these STS definitions are not needed to effectively state the Surveillance Requirements for verifying the I&C response times are within limits, as explained in the bases of the following Surveillance Requirements.

SR 3.3.1.3 is MPS implementation response time measurement. This surveillance verifies for each instrumentation function the response time of each channel from when the process variable exceeds its set point until the output from the channel analog logic reaches the input of the MPS digital logic. So, the response time for this surveillance is just handling the input to the digital system.

And then, SR 3.3.2.3 verifies for each reactor trip system actuation logic division, that the division response time from the output of equipment interface module until the reactor trip breakers are opened. So, it goes to the other end of the instrument loop and handles the output from the digital system.

And again, the digital system, they provide a conservative allocated time in calculating the total response time. Next slide.

MEMBER BROWN: Go backwards a minute.

MR. HARBUCK: Slide 11?

MEMBER BROWN: Is the stuff in blue replacing the lines above it or --

MR. HARBUCK: No --

MEMBER BROWN: -- is that just an explanation --

MR. HARBUCK: Well, what --

MEMBER BROWN: -- that we're going to get rid of the RTS and ESF terms?

MR. HARBUCK: Well, what this is meant to indicate, what I'll get to on the next slide, these are quotes from the bases and they are meant to indicate some material that was taken from the defined terms, the definitions, that have been put into the bases now. And I had a point to make about that.

CO-CHAIR REMPE: Let's let him get to the next slide, okay? I think it explains it.

MR. HARBUCK: Okay. Now, SR 3.3.3.2 verifies, and this is kind of interesting, so listen carefully, it verifies the pressurizer heater trip ESFAS function logic and actuation division response time, from the output of the equipment interface module to the loss of voltage at the output of the pressurizer heater breaker.

MEMBER BROWN: Can you -- that's on the previous slide?

MR. HARBUCK: That's on Slide --

MEMBER BROWN: You said SR --

MR. HARBUCK: That's on Slide 11, 3.3.3 --

MEMBER BROWN: 2 was on Slide 10.

MR. HARBUCK: Did I mix -- well, 3.3 --

CO-CHAIR REMPE: Well, if I can --

MR. HARBUCK: -- I believe it's 3.3.3.2, it might be 3.3.3.3, I could have -- anyways, the Surveillance Requirement number is not so important, but the response time surveillance for the ESFAS functions in the Instrumentation section only explicitly addresses the pressurizer heater trip.

Now, the surveillances for the other ESFAS functions, logic and actuation functions, that opens or closes valves are included in each associated system's LCO subsection.

Now, the reason for the blue highlighted text is because, unlike the Section 3.3 bases for response time surveillances, the bases of the surveillances to verify valve actuation time in LCO subsections of affected systems do not include descriptive language, which is consistent with the omitted STS response time definitions. And in the absence of those definitions, we are suggesting that the applicant add such language to the bases for those surveillances. Next slide.

CO-CHAIR REMPE: You okay, Charlie?

MEMBER BROWN: I'm still having trouble with that. Somebody else understands it better than I do. It seems to me, you've added -- the blue is not part of the overall STS? I'm not real familiar --

MR. HARBUCK: The quotations are taken from the bases, for that surveillance.

MEMBER BROWN: I'm trying to figure out what the intent of the blue part is. I mean --

MR. HARBUCK: It would --

MEMBER BROWN: -- maximum response time --

MR. HARBUCK: That phrase, that sentence is a phrase that's verbatim out of the definitions in the Standard Tech Specs.

MEMBER BROWN: Okay. Go back to Slide 10, I might --

MR. HARBUCK: Okay.

MEMBER BROWN: -- be able to understand that one better, 3.3.2. The Standard says, the maximum digital response time is described in the FSAR. Okay.

The SR encompasses the response time from the output of the equipment interface until the trip breakers open. Now, the next statement says, to me, that you can do it piecemeal or overall. Is that --

MR. HARBUCK: That's true. And the total response time there is referring not just --

MEMBER BROWN: Is that --

MR. HARBUCK: -- to what that surveillance is --

MEMBER BROWN: -- in the Standard right now?

MR. HARBUCK: Yes.

MEMBER BROWN: So, that's not an addition?

MR. HARBUCK: That's not an addition or anything, I'm just point out that --

MEMBER BROWN: Okay. I thought --

MR. HARBUCK: -- in compensating --

MEMBER BROWN: I'm sorry, I thought they were modifying the Standard.

MR. HARBUCK: No.

MEMBER BROWN: I thought they were modifying the Standard. I understand --

MR. HARBUCK: They were beefing up the bases --

MEMBER BROWN: -- whether I agree with that or not, that's another thing. But that's what's been used in the past?

MR. HARBUCK: Right, yes.

MEMBER BROWN: Okay, thank you.

MR. HARBUCK: Okay.

MEMBER BROWN: Jose, you were on the same page I was? Okay.

CO-CHAIR REMPE: We're a bit behind, so let's go ahead, okay? We're good now.

MEMBER BROWN: Yes, I got it. I got it, we're okay.

MR. HARBUCK: Slide 12. Okay. All right. So, very briefly here. Some other defined Surveillance Requirements.

The MPS, module protection system, continuously conducts automatic self-testing to verify channel and division operability between the channel calibrations and actuation logic test.

And this feature of the MPS is more

effective than performing the STS channel operational test every 92 days and the STS actuation logic test every 24 months, for verifying proper operation of the MPS instrumentation and the RTS and ESFAS logic and actuation divisions.

This is a reminder that in the Standard Tech Specs, the circuits are typically analog and so, these tests involve injecting a signal and seeing what happens to the logic and doing the various combinations.

MEMBER BROWN: Okay. If I understand --

MR. HARBUCK: And they don't --

MEMBER BROWN: -- this one --

MR. HARBUCK: And they don't do that one. Okay.

MEMBER BROWN: And if I understand --

MR. HARBUCK: All right.

MEMBER BROWN: -- in the old days, with the analog stuff, you had a periodic test that you had to do, where you input stuff.

MR. HARBUCK: That's right.

MEMBER BROWN: Since we've got self-testing, that's performed and the only check of the self-testing is on a two-year basis. That's the way --

MR. HARBUCK: The --

MEMBER BROWN: -- I'm reading it. They go -- that's how I'm reading this is that they have to do something to verify that the self-testing is actually self-testing.

MR. HARBUCK: That particular point in the Branch Technical Position that addresses the self-testing of digital systems would have to be addressed by Instrumentation reviewers. My understanding is that testing of the tester has been addressed in that discussion of that chapter.

MEMBER BROWN: Yes, I don't remember.

MR. HARBUCK: But I'm not in a position to --

MEMBER BROWN: It's been a while.

MR. HARBUCK: -- question that at this point. Okay.

Now, the STS actuation logic test definition has been revised for NuScale, since the NuScale actuation logic test has a reduced scope. It goes from the equipment --

MEMBER BROWN: Well, I'm reading the --

MR. HARBUCK: -- interface --

MEMBER BROWN: I'm reading the first paragraph, which says, you depend on self-testing and division operability between channel calibrations and actuation logic tests, which are on 24.

So, I think that reads, the way I'm reading it, am I missing -- Jose, do you see that the same way? They're relying two years on self-tests. Every two years, there's going to be an operability of the channel calibrations.

MEMBER MARCH-LEUBA: I understand what you're saying, I don't understand what they're saying. I think we're on the same page.

MEMBER BROWN: Yes, Page 12. Okay.

MR. HARBUCK: Okay. So, as I was saying, the actuation logic test in NuScale verifies the analog actuation priority logic in the equipment interface module and the various manual switches which have no automatic self-test features.

Now, the STS channel check definition is also revised, as shown by the markup on this slide, to account for the operation of the self-testing feature of the digital platform. So, they've kept the surveillance and they've kept essentially the same frequency, but how it's performed is a bit different than in typical plant. Okay.

Electrical Power Surveillance Requirements is a bit of misnomer, but I wanted to say it to make the point. NuScale protects the MPS from faults in the non-safety-related electrical power system by use of Class 1E isolation devices that automatically separate the MPS from its electrical power upon detecting an under voltage or overcurrent condition.

LCOs for the MPS reactor trip system and ESFAS include, for the associated Class 1E isolation devices, a Channel Calibration Surveillance Requirement, which verifies operability of each device.

They don't include such a surveillance for the manual actuation and we've asked them to consider whether that's appropriate.

Given the importance of maintaining water inventory in the decay heat removal system following its actuation, the main steam isolation and feed water isolation valves must not leak more than the leakage rate that will support decay heat removal system operation for the time period needed following an event that relies on the decay heat removal system for core cooling.

NuScale has not provided values as part of its application for these leakage limits, which are the acceptance criteria for the leak rate verification surveillances for the MSIVs and the feed water isolation valves.

We are still evaluating NuScale's position that these acceptance criteria will be established when the in-service test program is put in place by the COL applicant. NuScale has concluded, therefore, that these values do not need to be provided as part of the DCA.

CO-CHAIR REMPE: So, we're really running behind, can you highlight key points on the remaining slides, please?

MR. HARBUCK: Okay. In the next two slides, I'll just briefly put it -- make it very brief.

They've proposed, again, a provision that's in an unapproved industry Change Traveler to the STS that would allow Surveillance Requirements to be suspended and not required to be met if the isolation device or valve is put in its actuated position. And we're not convinced that this is needed or warranted. And so, we have open item related to that. Okay.

Now, Slide 17. The next six slides involve a listing of our open items, several of which have been discussed, as indicated by the blue color on their number.

16-30D, at the bottom of the page, is related to their adoption of a surveillance frequency control program. We asked them to provide, in the DCA, the initial frequencies and the bases for those frequencies, which is necessary in order for a COL applicant to actually implement that program. And they put this in a table in FSAR Section 16.1. And that table needs to be updated and so, this item is simply tracking that.

MEMBER BLEY: On this long list of open items, are there any on which you're really divided or is it on most of them or are you getting pretty close to agreement on how these are going to settle out?

MR. HARBUCK: I'll point that out, if I mention one --

MEMBER BLEY: Okay.

MR. HARBUCK: -- if there's a -- I'm not going to really talk about the ones that we've -- I mean, I guess, one would be on Slide 18, the one about providing the Surveillance Requirement acceptance criteria for the valve leakage in the steam and feed systems. That would be one where we'd perhaps are in disagreement.

MEMBER CORRADINI: And again, a little bit louder, that's the one that you're still in discussions with?

MR. HARBUCK: Right. On the -- Question 16-59 has to do with apparent disconnect on the applicability of this instrument function, which initiates the decay heat removal system, but there's going to be a design change coming that we've been made aware of that's going to alter their response to that question. They haven't, as a result, they haven't responded to that one yet and that's why it's open.

And on the next slide, we have some issues about the applicability of the LCO 3.1.9, we made a suggestion.

And the Mode 3 applicability for LCO 3.3.1, what's interesting there is that there's an interlock, which blocks it or bypasses it, and yet, the applicability goes beyond, goes to a lower temperature than at the point at which it's blocked, so that's a question here.

And finally, there was some confusion in an Action Requirement, which said, decrease power below this interlock, the interlock refers to 15 percent rated power and the question was, which interlock is the appropriate one to refer to?

On the Slide 21, these are -- we just

recently provided six new questions to NuScale in this RAI 9642.

And I'll just mention briefly that we have a concern about the design of their system to prevent putting water into containment when the reactor coolant system is at too high a temperature.

There's going to be in the PTLR a temperature limit and the way they preclude it now is that they have an interlock based on that temperature on a control valve, a non-safety control valve in the core flood and drain system. And we're asking, well, why didn't they put an interlock on the containment isolation valves, which are a higher level of safety grade?

And then, lastly, we're asking them to add some additional methodology references to the Core Operating Limits Report Specification 5.6.3.

The last one refers back to the leak-before-break, where we're still reviewing the limit they've proposed in LCO 3.7.3 of 1.5 gallons per hour. And we're still reviewing their supplemental response. And we're also, we think we're waiting for some additional input, that may be incorrect, but that's our understanding right now.

And then, the last thing doesn't have a

number, but the COL item list or the sub-item list, as it's been called, are all the places where information is bracketed in the tech specs and bases, where we want that list to be definitive as part of the design cert, so that the COL applicant knows exactly what they need to do to complete the tech specs.

CO-CHAIR REMPE: So, we have like one minute to finish this.

MR. HARBUCK: Okay.

CO-CHAIR REMPE: We've got --

MR. HARBUCK: I'm --

CO-CHAIR REMPE: -- another meeting, so maybe summarize.

MR. HARBUCK: Yes, I'm -- okay. The next slide is just your -- okay. Slide 23 is just for your information, to say we've got some more work to do, because they made some design changes.

And then, there's some general issues. One, I just mentioned, completing the COL item list. Also, we have some issues related to their adopting Travelers that aren't approved. And then, there's a bunch of administrative items that need to be cleaned up.

And so, we've done a pretty thorough review of the tech specs and we have a number of items that need to be resolved. I wouldn't say any of them are show-stoppers or very significant, in and of themselves.

And the Technical Branches listed will be providing some support for us in completing those. And when those are done, we'll be done with the tech specs. That concludes the presentation.

CO-CHAIR REMPE: Okay, thank you. I apologize for other people's sessions going longer.

MR. HARBUCK: I understand.

CO-CHAIR REMPE: Before we decide about having public comment, I want to have the members say whether they think we need to have a closed meeting to address, for example, your buoyancy question, Dick.

Do you want to explore that further today? So, any members see a need to have a closed meeting? Mike, did you have a comment before I start asking for public comments? Okay.

Do we have anyone on the phone line? Is it open? Yes, it's open? Okay. Do we need to verify that someone's out there or do we just --

MR. SNODDERLY: Is there anyone on the public phone line?

MR. LEWIS: Marvin Lewis, public --

MS. FIELDS: Sarah Fields.

CO-CHAIR REMPE: Okay. Do either of you have a comment you'd like to make at this time?

MR. LEWIS: Yes, I do.

CO-CHAIR REMPE: Please go forward.

MR. LEWIS: Well, let me put it this way, I've been listening to the ACRS since they went on the telephone and I've been attending ACRS meetings as a member of the public well before then, back to the 1970s. And I have to admit, today blew me away. I've heard better questions from the ACRS membership than I have ever heard before. And I thank you.

CO-CHAIR REMPE: Thank you for your comment. Are there any other comments? So, with that, I need to go around to each --

MS. FIELDS: Yes.

CO-CHAIR REMPE: Oh, sorry.

MS. FIELDS: This is Sarah Fields. I'd like to make a comment. Yes, I also believe that there were some very good questions, which I hope will be answered down the line.

Going back to the Chapter 9 presentation, 9.1-2, the slide indicated that the COL applicant will demonstrate that an NRC licensed cask can be lowered into the dry dock and used to remove spent fuel assemblies from the plant. And then, 9.1-3, they also reference the fuel being transferred to a cask.

And I found that there seems to be confusion between the term canister and cask. My understanding is that fuel first goes into a metal canister and is later transferred into a thicker containment cask.

So, I think that that should be clarified, whether fuel within the containment vessel is going to be transferred to a metal canister and then, possibly, later going into a cask.

That should be clarified, because a lot of discussions in other dimensions related to spent fuel, these fuels are confused and they should really be specifically defined.

CO-CHAIR REMPE: Thank you for your comment. Are there any other comments, individuals on the line? And I'm not seeing anyone in the room that has comments.

And so, we're going to go around and ask the members for comments. In particular, if there's some issues that rise to being mentioned in the progress or interim letter we're going to be writing, please emphasize those as we go around the room.

I heard several topics discussed today that I think might warrant being mentioned in this interim letter, but it would be good for you to highlight them if I've missed them. And I'm going to start with Walt.

MEMBER KIRCHNER: For this section, where I have questions, I think, on tech specs, I will take note and bring them up in Chapter 4 and 5 and 6, which we haven't yet reviewed. Issues like leak-before-break and breaks within containment.

And, here, we have things like leakage rates into the containment, but there's the more fundamental aspect of design, before you get to the leak, that is something that we can take up with the applicant and the staff when we get to Chapters 4, 5, and 6. Thank you.

CO-CHAIR REMPE: Thank you. Dennis?

MEMBER BLEY: Nothing to add.

CO-CHAIR REMPE: Mike?

MEMBER CORRADINI: Nothing to add at this time.

CO-CHAIR REMPE: Dick?

CO-CHAIR SKILLMAN: Nothing further, it's already on the record and --

CO-CHAIR REMPE: Okay. So, you've mentioned several points and you don't have any requests that you'd like to have emphasized in the full committee meeting? Because if we're going to have anything documented in the letter, this would be a good time to summarize some of those points.

CO-CHAIR SKILLMAN: Well, I'm going to pull those points out of the record when I see it.

CO-CHAIR REMPE: Okay.

CO-CHAIR SKILLMAN: I've identified them contemporaneously, my concern with homogeneity, the crane, the floor, the structure, not the strength of the floor, but the membrane vulnerability to load drop, and several others such as those. And those are already in the record --

CO-CHAIR REMPE: Okay.

CO-CHAIR SKILLMAN: -- so I'll pluck those from the record.

CO-CHAIR REMPE: Okay, thank you.

CO-CHAIR SKILLMAN: Thank you.

CO-CHAIR REMPE: Harold?

MEMBER RAY: Our last staff presentation included the excellent comment that the COL applicant needs to know exactly what they need to do. And I just have some concern on the larger areas covered and I think with regard to tech specs, that's being addressed, it seems, from what we've heard.

But we need to underscore, I think, that where a design certification credits an assumption, that a COL applicant will comply with requirements which are yet to be identified and will be provided later by the certificate holder, then that needs to be explicit.

We can't answer all their questions at the time of certification, but the fact that things are going to be provided later, requirements are going to be imposed on the COL applicant later, or provided to the COL applicant later, needs to be identified. And that's all I want to say at this point.

CO-CHAIR REMPE: Thank you. Jose?

MEMBER MARCH-LEUBA: Same as Dick, my comments are on the record and I will provide you with some comments for the letter later, when we can review them.

CO-CHAIR REMPE: Okay. Ron?

MEMBER BALLINGER: No further comment.

CO-CHAIR REMPE: Charlie?

MEMBER BROWN: No further comment.

CO-CHAIR REMPE: Vesna?

MEMBER DIMITRIJEVIC: No further comment.

CO-CHAIR REMPE: Before I get to Steve, what about Matt?

MEMBER SUNSERI: I don't have any comments, thanks, Joy.

CO-CHAIR REMPE: Pete?

MEMBER RICCARDELLA: I don't have any comments at this time, but I will bring out some thoughts and concerns I have at the letter writing, during the letter writing.

CO-CHAIR REMPE: Okay. So, let me hear from Steve, but hold that thought, I want to make a comment, but go ahead, Steve?

DR. SCHULTZ: No further comments that aren't on the record.

CO-CHAIR REMPE: Okay. So, we will be having an interim letter and waiting until that letter is always a little difficult. So, I know, Jose and Dick and Harold, if you'll send, and you also, Pete, some draft comments, it'll make it easier.

And I think Matt's going to combine this with the letter for 10, 11, and 12. So, copy me, but also send them to Matt, okay? Is that your understanding, the process, Mike, that you'd envisioned, that he would be leading --

MEMBER CORRADINI: I would appreciate it if I got to know what the members want.

CO-CHAIR REMPE: That would be true, too. Yes, copy Mike, Matt, and me.

MEMBER CORRADINI: But from the standpoint that we have, I think, on the full committee, we have the morning Thursday for the staff and the applicant to present.

So, to the extent that there are certain things we want to put in the letter, because again, my personal view is, if it's something that's significant, we want to issue it. If it's something that is a matter of discussion, we can enter it, but I'm looking for things significant that deviate from the open items, that we want or the staff wants.

CO-CHAIR REMPE: So, I think, Dick and Harold and Jose have emphasized those, about how you check about the boron mixing and different things like that, they've emphasized today, and that's why I wanted to have them bring it up again as we went around the circle. Is there anything else I've forgotten, Mike Snodderly?

MR. SNODDERLY: No, I'll just remind the Committee of the bylaws on letter writing and that when these comments are circulated, it should not involve more than the majority, no more than -- it has to be a less than a minority. Or then, you should wait until the open letter writing session for further discussions.

CO-CHAIR REMPE: Well, yes, we are just supposed to have it go to the Subcommittee, but the draft letter should be a product of the Subcommittee, so it is fine to have it sent to us that way. Okay? So, thank you and with that, we're going to adjourn.

(Whereupon, the above-entitled matter went off the record at 12:29 p.m.)

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SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Presentation: NuScale Chapter 9, Auxiliary Systems," PM-0319-64807, 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 March 20, 2019. The materials support NuScale's presentation of Chapter 9, "Auxiliary Systems," of the NuScale Design Certification Application.

Enclosure 1 is the presentation entitled "ACRS Presentation: NuScale Chapter 9, Auxiliary Systems," PM-0319-64807, Revision 0.

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

If you have any questions, please contact Carrie Fosaaen at 541-452-7126 or at cfosaaen@nuscalepower.com.

Sincerely,

Zackary W. Rad
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Enclosure 1: "ACRS Presentation: NuScale Chapter 9, Auxiliary Systems," PM-0319-64807, Revision 0

Enclosure 1:

"ACRS Presentation: NuScale Chapter 9, Auxiliary Systems," PM-0319-64807, Revision 0

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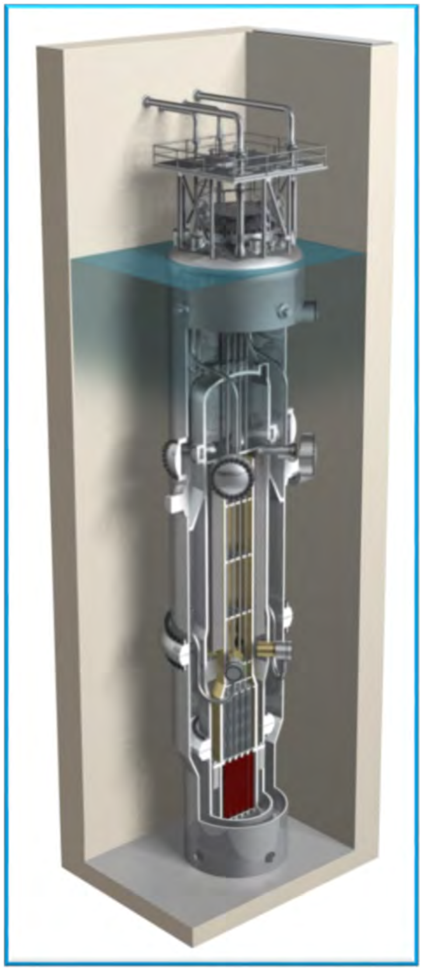
ACRS Presentation: NuScale Chapter 9, Auxiliary Systems



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March 20, 2019

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Template #: 0000-21727-F01 R5

Chapter 9: Auxiliary Systems

Section	Title
9.1	Fuel Storage and Handling
9.2	Water Systems
9.3	Process Auxiliaries
9.4	Air Conditioning, Heating, Cooling, and Ventilation Systems
9.5	Other Auxiliary Systems
9A	Fire Hazard Analysis

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9.1 - Fuel Storage and Handling

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

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9.1.1 - Criticality Safety of Fresh and Spent Fuel Storage and Handling

- Storage rack design does not require loading patterns or zones
- Spent fuel pool connected to refueling and reactor pools which forms ultimate heat sink - large volume prevents undetected addition of unborated water
- Max k_{eff} remains below applicable limits and no abnormal condition causes inadvertent criticality (TR-0816-49833, Rev. 1.)
- Fuel handling procedures place controls on movement of each fuel assembly and the designated storage location in a rack
- TS 4.3 – Fuel Storage
- TS 5.5.12 – Spent Fuel Storage Rack Neutron Absorber Monitoring Program
- COL Item 9.1-1: A COL applicant will develop plant programs and procedures for safe operations during handling and storage of new and spent fuel assemblies, including criticality control

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9.1.2 - New and Spent Fuel Storage

- Storage and handling facility for new and spent fuel assemblies located in the RXB - racks can store new or spent fuel assemblies
- The fuel storage racks, spent fuel pool, liner, and RXB meet Seismic Category I requirements and protected from non-Seismic Category I SSCs
- 9.1-2: A COL applicant will demonstrate that an NRC-licensed cask can be lowered into the dry dock and used to remove spent fuel assemblies from the plant
- 9.1-8: A COL applicant will submit an evaluation of the spent fuel storage racks that addresses structural, thermal-hydraulic, criticality, and material analysis aspects of the design. This evaluation is dependent on a vendor-specific design and the as-built configuration of spent fuel storage racks. The design of the spent fuel storage racks is considered acceptable when it meets the criteria of Appendix D to Design Specific Review Standard 3.8.4.

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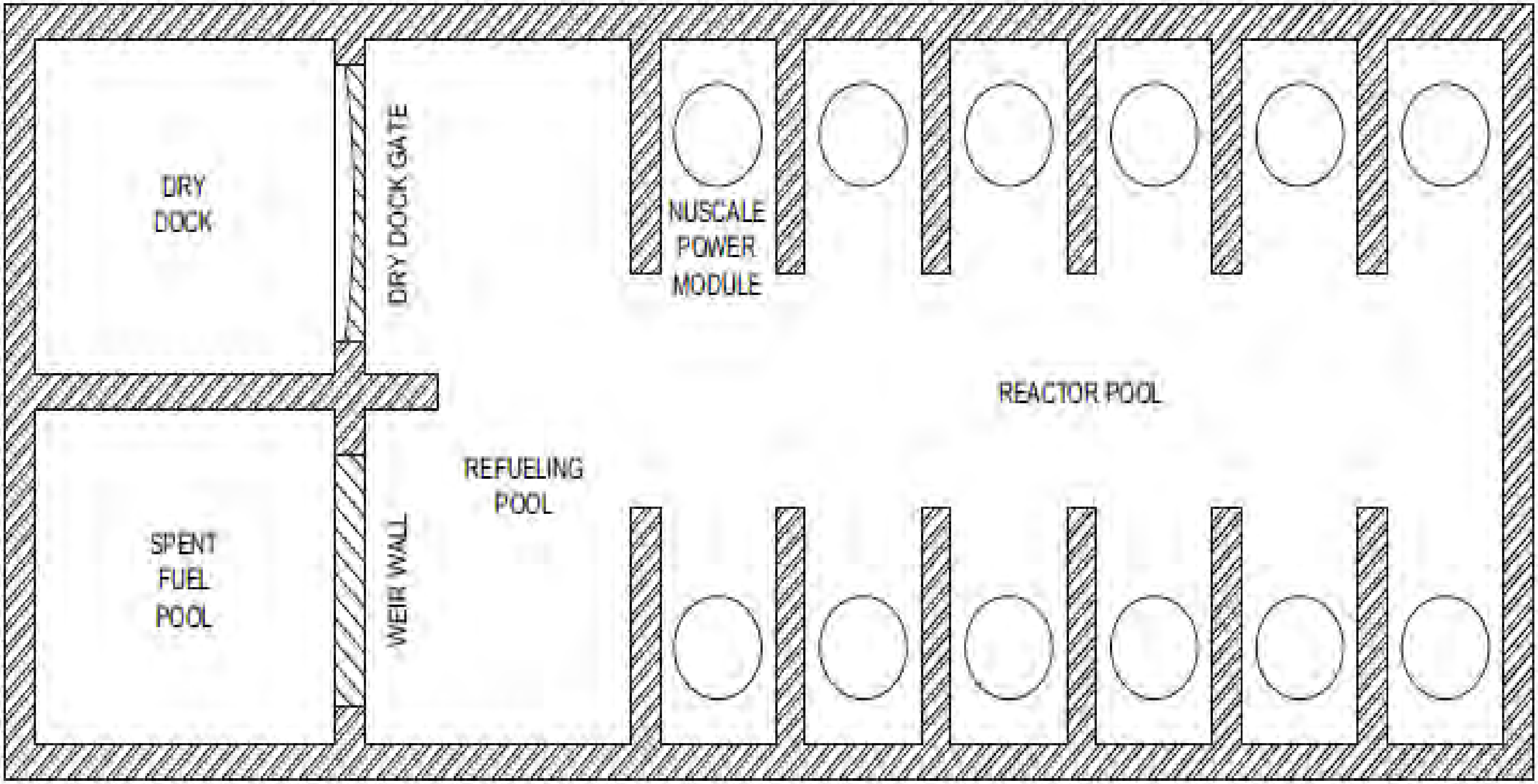
Ultimate Heat Sink Configuration

Figure 9.2.5-1: Ultimate Heat Sink Configuration



NOTE: NOT TO SCALE

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9.1.3 - Spent Fuel Pool Cooling and Cleanup System

Spent Fuel Pool Cooling System and Reactor Pool Cooling System

- Performs the following nonsafety-related functions:
  - Maintains reactor, refueling pool and spent fuel pool temperature during normal operations and refueling
  - Maintains ultimate heat sink water level by providing makeup from the demineralized water system
  - Provides chemistry control of pool including boron concentration
  - Provides reactor pool temperature information signals for post-accident monitoring

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9.1.3 - Spent Fuel Pool Cooling and Cleanup System

Pool Cleanup System

- Performs the following nonsafety-related functions:
  - removes impurities to reduce radiation dose rates and maintain water chemistry/clarity in pools and dry dock

Pool Surge Control System

- Performs the following nonsafety-related functions:
  - drains the dry dock using the evacuation pumps to support maintenance and refueling activities
  - transfers and stores excess water volume from the UHS to maintain pool level during surge events such as when an NPM is added to the pool

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9.1.3 - Spent Fuel Pool Cooling and Cleanup System

Pool Leakage Detection System

- Performs the following nonsafety-related functions:
  - provides for collection of water leaking from the pool liner
  - directs the flow to sumps for detection of collected leakage for operator evaluation
- Valves used to isolate each channel drainage line and leakage rate measuring line
- Channels collect leakage from pool wall and floor liner plates and direct it to a sump in radioactive waste drain system and routed to liquid radioactive waste system for further processing

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9.1.4 - Fuel Handling Equipment

- Functions
  - Receipt of new fuel
  - Refueling operations
  - Loading of spent fuel into a dry storage cask
- Design considerations - ANSI/ASME design requirements
  - Dropping fuel assembly following a safe shutdown earthquake minimized
  - Radiological shielding for personnel who operate the equipment
  - Protection from a criticality event
  - Capability to permit periodic inspection and testing of components
  - Speed of the fuel handling machine, trolley, and hoist motions are limited

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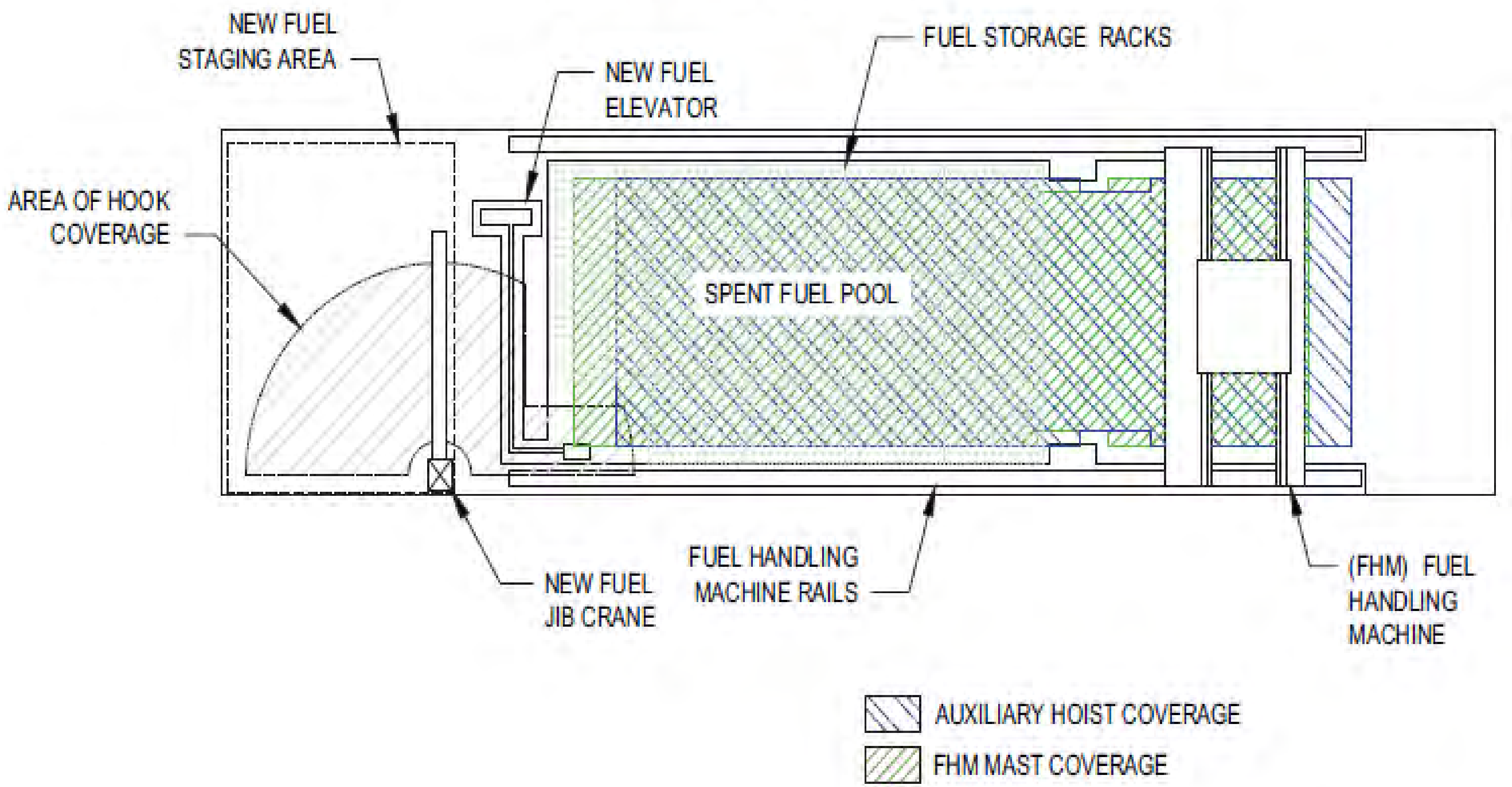
9.1.4 - Fuel Handling Equipment

Figure 9.1.4-1: Refueling Floor Layout



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9.1.4 – Fuel Handling Equipment

COL Items:

- 9.1-3: A COL applicant will develop procedures related to the transfer of spent fuel to a transfer cask
- 9.1-4: A COL applicant will provide the periodic testing plan for fuel handling equipment

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9.1.5 - Overhead Heavy Load Handling Systems

- Major components:
  - Reactor Building Crane (RBC)
  - NPM lifting fixture - comprised of the NPM lifting lugs and top support structure diagonal lifting braces - permanently installed
  - module lifting adapter (MLA) – connection for crane to lift and carry the NPM
  - wet hoist - main below-the-hook structure for lifting equipment underwater
- RBC:
  - Designed as a single-failure-proof crane – highest qualification
  - Movement uses position control system interlocks
  - Path and the maximum height limited – shielding maintained
  - Communication system provided between the MCR and RBC

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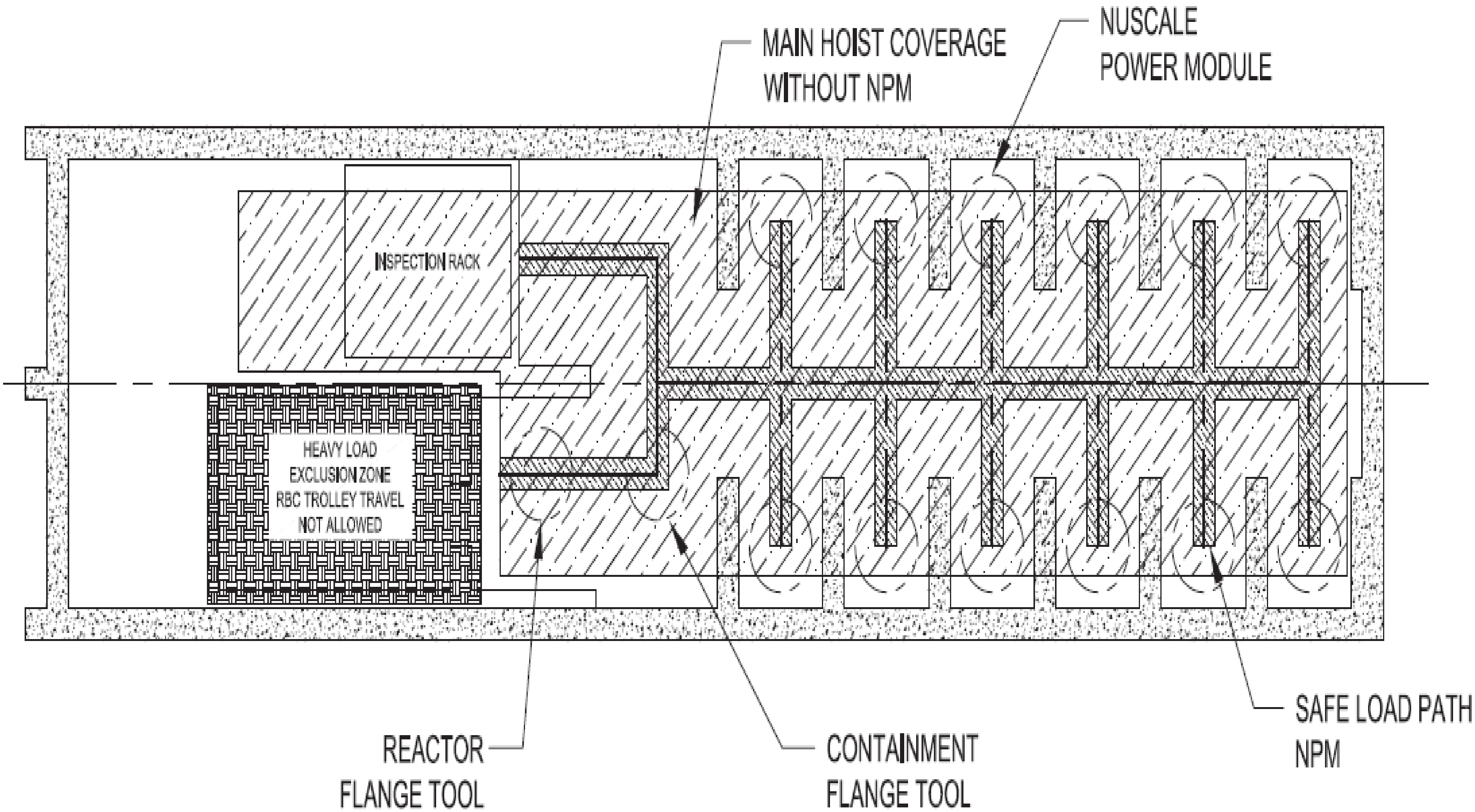
9.1.5 - Overhead Heavy Load Handling Systems



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9.1.5 - Overhead Heavy Load Handling Systems

Equipment	Rated Capacity (Tons)	Design Code	Single Failure Proof	Seismic Cat	Max Traverse Speed (fpm) [1]	Max Hoist Speed (fpm) [1]	Max Lift Height (ft)
RBC Main Hoist	850	ASME NOG-1, Type I	Y	I	30	2	50
RBC Aux Hoist	15	ASME NOG-1, Type I	Y	I	N/A	25[2] / 37[3]	65
MLA	790	ANSI N14.6	Y	N/A	N/A	N/A	N/A
NPM Lift Fixture	790	ANSI N14.6	Y	I	N/A	N/A	N/A
Wet Hoist	250	ASME NOG-1, Type I	Y	N/A	N/A	4	75
Jib Crane Hoist	1	ASME NUM-1, Type II	N	II	30	21	40

Notes
1. Bridge, trolley and hoist speeds are within recommended ranges of ASME NOG-1
2. With load on hoist
3. Without load on hoist

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9.1.5 - Overhead Heavy Load Handling Systems

- COL Items:
  - 9.1-5: The COL applicant will describe the process for handling and receipt of critical loads including NPMs
  - 9.1-6: The COL applicant will provide a design for a spent fuel cask and handling equipment including procedures and programs for safe handling.
  - 9.1-7: The COL applicant will provide a description of the program governing heavy loads handling. The program should address:
    - operating and maintenance procedures
    - inspection and test plans
    - personnel qualifications and operator training
    - detailed description of the safe load paths for movement of heavy loads

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9.2 - Water Systems

Section	Title
9.2.1	Station Service Water System – not in NuScale Design
9.2.2	Reactor Component Cooling Water System
9.2.3	Demineralized Water System
9.2.4	Potable and Sanitary Water Systems
9.2.5	Ultimate Heat Sink
9.2.6	Condensate Storage Facilities
9.2.7	Site Cooling Water System
9.2.8	Chilled Water System
9.2.9	Utility Water Systems

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9.2 - Water Systems

- Common Elements of NuScale Water System design (except for 9.2.5 - UHS)
  - Does not perform safety-related or risk-significant functions
    - Classified as nonsafety-related
  - Not an essential source of water to prevent or mitigate the consequences of accidents or to shut down the reactor and maintain it in a safe-shutdown condition
  - Not required during or after a natural phenomenon event
    - Includes provisions that ensure system failure will not adversely affect the functional performance of safety-related systems or components
    - Components in proximity to Seismic Category I SSCs designed to Seismic Category II

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9.2.5 - Ultimate Heat Sink

- Set of stainless steel-lined, reinforced concrete pools of borated water located below grade in the RXB (reactor, refueling, and SFP)
- Sized such that active cooling systems are not required for accident conditions; the combined volume of water in the UHS provides sufficient cooling for greater than 72 hours without additional makeup water
- Provided with a seismically qualified makeup line that can provide additional water to the UHS and with redundant water level instrumentation
- During normal plant operations, heat is removed from the UHS through the reactor pool cooling system and rejected into the atmosphere through a cooling tower or other external heat sink
- In a DBA involving a sustained loss of all AC power, decay heat is removed from the NPMs through passive heat transfer to the pool
- TS 3.5.3 – Ultimate Heat Sink

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9.2 - Water Systems

- COL Items:
  - 9.2-1: A COL applicant will select the appropriate chemicals for the RCCWS based on site-specific water quality and materials requirements
  - 9.2-2: A COL applicant will describe the source and pre-treatment methods of potable water for the site, including the use of associated pumps and storage tanks
  - 9.2-3: A COL applicant will describe the method for sanitary waste storage and disposal, including associated treatment facilities
  - 9.2-4: A COL applicant will provide details on the prevention of long-term corrosion and organic fouling in the SCWS
  - 9.2-5: A COL applicant will identify the site-specific water source and provide a water treatment system that is capable of producing water that meets the plant water chemistry requirements

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9.3 - Process Auxiliaries

Section	Title
9.3.1	Compressed Air Systems
9.3.2	Process Sampling System
9.3.3	Equipment and Floor Drain Systems
9.3.4	Chemical and Volume Control System
9.3.5	Standby Liquid Control System (Applicable only to BWRs)
9.3.6	Containment Evacuation System and Containment Flooding and Drain System

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9.3 - Process Auxiliaries

- Common Elements of NuScale Process Auxiliaries Design
  - No safety-related or risk significant functions, not credited for mitigation of design basis accidents, and has no safe-shutdown functions (except 9.3.4 – CVCS)
  - SSCs whose failure could adversely affect Seismic Category I SSCs during or following an SSE are designed as Seismic Category II
  - COL Item 9.3-1: COL applicant submits a leakage control program
  - COL Item 9.3-2: COL applicant develops PASS contingency plans for using the PSS and the CES off-line radiation monitor to obtain reactor coolant and containment atmosphere samples

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9.3.4 – Chemical and Volume Control System

- CVCS purifies reactor coolant, manages chemistry of the coolant (including boron concentration), provides reactor coolant inventory makeup and letdown, and supplies spray flow to the pressurizer to reduce RCS pressure
  - CVCS is provided for each NPM
  - CVCS is used in combination with the MHS during startup to raise reactor coolant temperature and to generate natural circulation flow in the RCS before nuclear heat addition
  - BAS is a shared system for all 12 NPMs
    - BAS prepares, stores, and transfers borated water for use by the CVCS or by the SFPCS for adding boron to the SFP as needed

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9.3.4 – Chemical and Volume Control System

- CVCS is only system with connections to the RCS and piping that runs outside containment
  - CVCS is the total scope for intersystem LOCA consideration in the NuScale design
  - CIV function part of CNTS - Section 6.2
- CVCS is equipped with two automatic, safety-related, fail-closed, demineralized water isolation valves to ensure CVCS operation does not inadvertently cause a dilution of the RCS boron concentration
  - CVCS required for NPM operation per TS 3.4.6

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9.3.6 – Containment Evacuation System and Containment Flooding and Drain System

- CES and CFDS are used to transfer liquids and gases between the CNV free volume and other plant systems.
  - CES establishes and maintains a vacuum in the CNV by removing water vapor and non-condensable gases from the CNV
  - CES Performs Leakage detection function
  - CFDS is used to flood a CNV with borated reactor pool water after shutdown in preparation for NPM refueling and to drain water back to the reactor pool in preparation for NPM startup
  - CFDS can provide borated coolant inventory to containment during BDBE for decay heat removal
  - CIV function part of CNTS - Section 6.2
  - CES required for NPM operation per TS 3.4.7, 3.4.9 & 3.7.3

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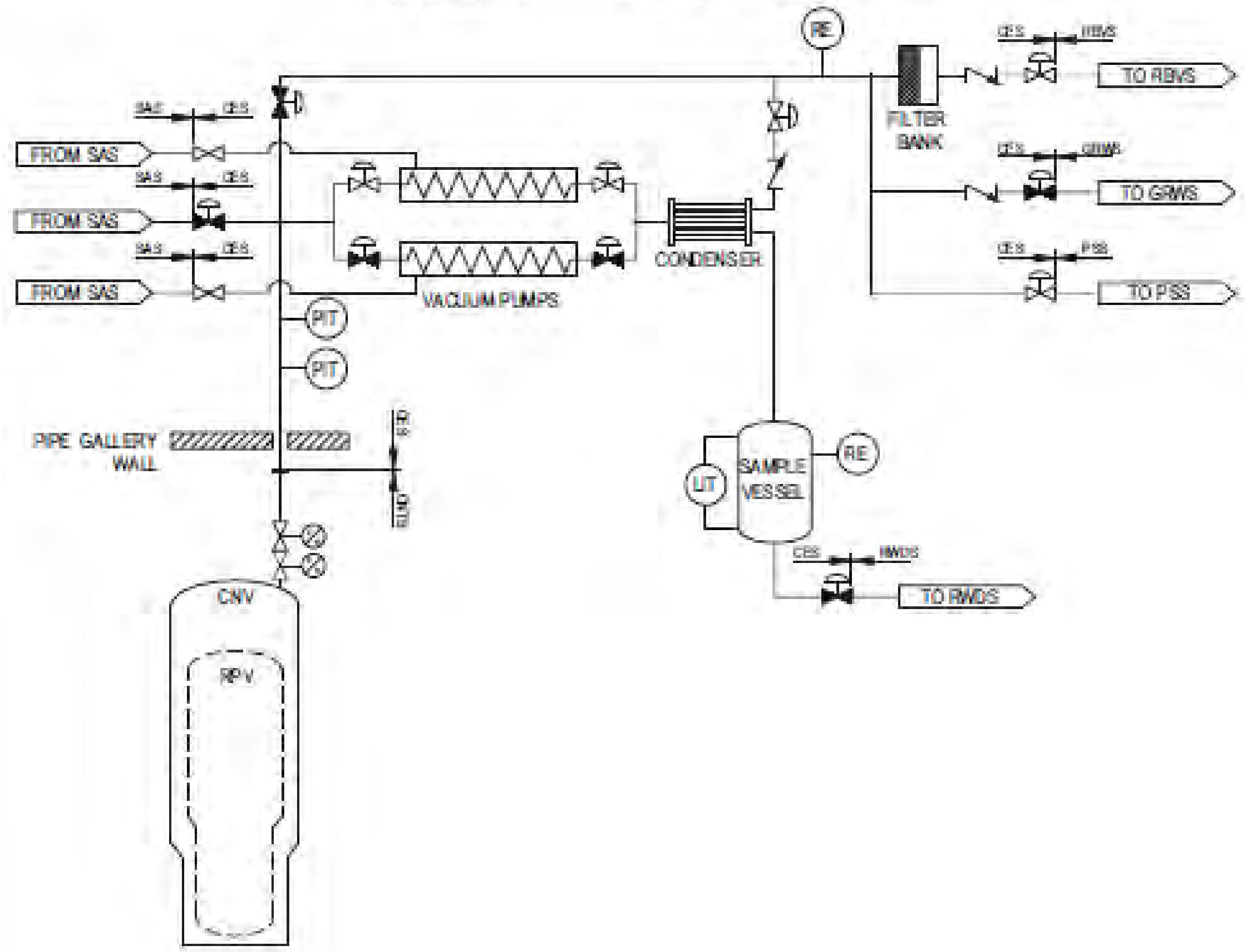
9.3.6 – CES Flow Diagram

Figure 9.3.6-1: Containment Evacuation System Diagram



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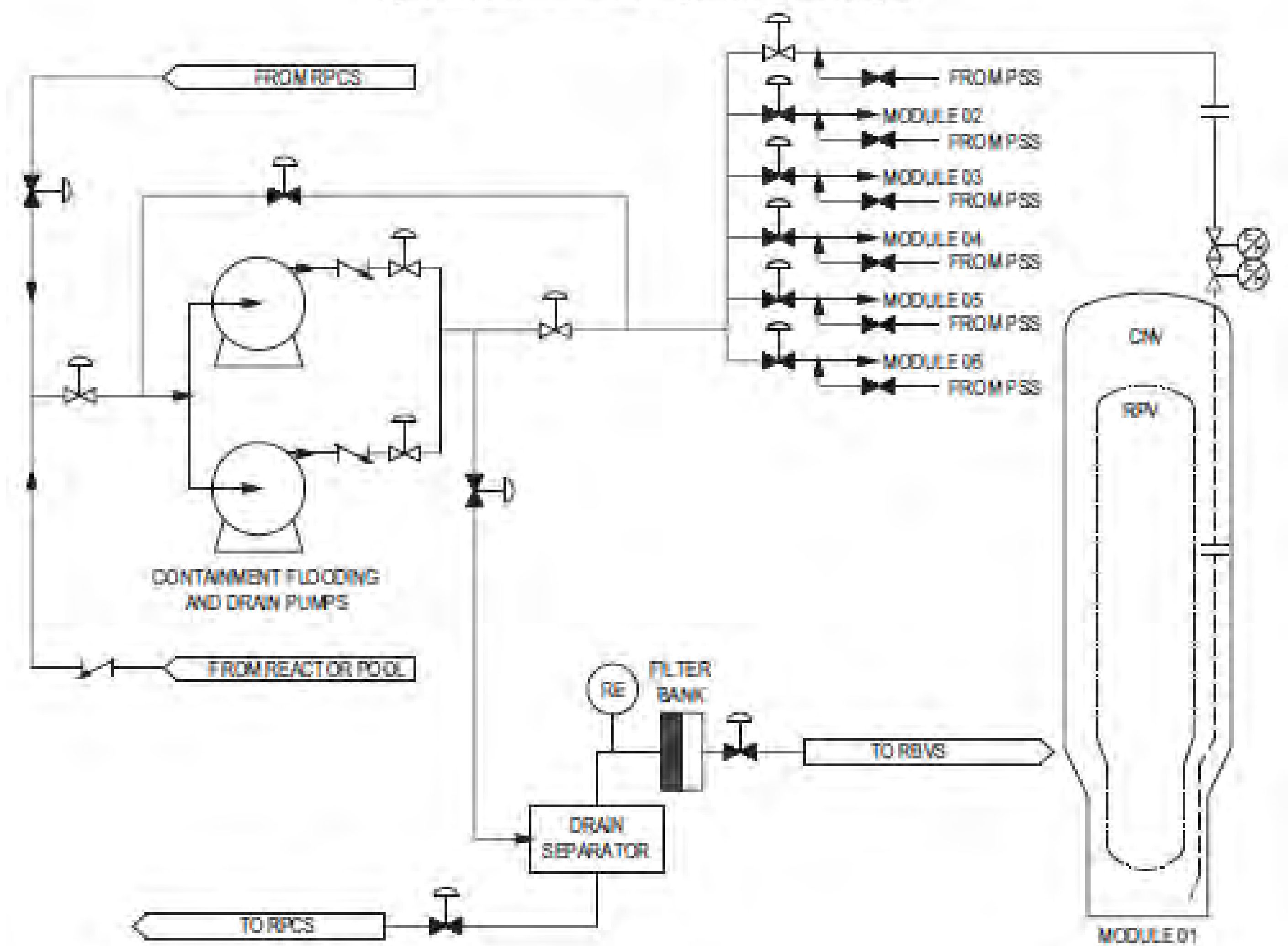
9.3.6 – CFDS Flow Diagram

Figure 9.3.6-2: Containment Flooding and Drain System Diagram



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9.4 - Air Conditioning, Heating, Cooling, and Ventilation Systems

Section	Title
9.4.1	Control Room Area Ventilation System
9.4.2	Reactor Building and Spent Fuel Pool Area Ventilation System
9.4.3	Radioactive Waste Building Ventilation
9.4.4	Turbine Building Ventilation System
9.4.5	Engineered Safety Feature Ventilation System – not in NuScale Design

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9.4 - Air Conditioning, Heating, Cooling, and Ventilation Systems

- Common Elements of NuScale Ventilation System Design
  - No safety-related or risk significant functions, not credited for public dose mitigation of design basis accidents, and has no safe-shutdown functions
  - Components whose failure could adversely affect Seismic Category I SSC or could result in incapacitating injury to occupants of the control room during or following an SSE are designed as Seismic Category II
  - A COL item for each system to specify periodic testing and inspection program requirements (COL Items 9.4-1, 9.4-2, 9.4-3 and 9.4-4)

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9.5 - Other Auxiliary Systems

Section	Title
9.5.2	Communication System
9.5.3	Lighting Systems
9.5.1	Fire Protection Program
9.A	Fire Hazards Analysis

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9.5.2 – Communication System

- COMS includes components for intra-plant and plant-to offsite communications. Communication methods include private branch exchange, a public address and general alarm system, sound-powered telephones, distributed antenna, and radios for point-to-point communication between plant personnel in vital areas of plant under expected conditions.
  - COMS serves no safety-related or risk-significant functions
  - COMS not credited for mitigation of design basis accidents and has no safe-shutdown functions
  - COL Item 9.5-1: COL applicant provides a description of the offsite communication system
  - COL Item 9.5-2: COL applicant determines the location for the security power equipment within a vital area

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9.5.3 – Lighting Systems

- PLS provides artificial illumination for buildings, rooms, spaces, and outdoor areas of the plant
  - PLS functions include normal plant lighting, emergency plant lighting, and normal and emergency MCR lighting
  - PLS provides illumination for each area of the plant in accordance with the guidelines of NUREG-0700
  - MCR emergency lighting is fed from the highly reliable DC power system for a minimum of 72 hours
  - PLS provides emergency lighting with battery pack for 1.5 hours for egress or exiting the area in accordance with NFPA 804
  - PLS provides emergency lighting with battery pack for 8 hours for the post-fire safe shutdown activities in areas outside the MCR and RSS (if required)
  - No COL Items

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9.5.1 - Fire Protection Program

- Objective:
  - Use the concept of defense-in-depth to achieve the required degree of reactor safety by using administrative controls, fire protection systems and features, and redundant safe shutdown capability
  - Safe shutdown relies on passive fire protection - redundant safe shutdown equipment is separated by 3-hour fire barriers
- Defense-in-depth principles achieve the following:
  - To prevent fires from starting;
  - To rapidly detect, control, and extinguish promptly those fires that do occur; and
  - To provide protection for SSCs important to safety so that a fire that is not promptly extinguished by the fire suppression activities does not prevent the safe shutdown of the plant

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9A - Fire Hazards Analysis

- FHA demonstrates that the plant maintains the ability to perform safe shutdown functions and minimize radioactive material releases to the environment in the event of a fire
- FHA has the following objectives:
  - consider in situ and transient fire hazards
  - determine the effects of a fire in any location in the plant on the ability to safely shut down the reactor or to minimize and control the release of radioactivity to the environment
  - specify measures for fire prevention, detection, suppression, and containment for each fire area containing safety-related and risk-significant SSCs, in accordance with NRC guidance and regulations

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9A - Fire Hazards Analysis

- FHA Special Cases
  - Main Control Room Fire Area
    - Manual switches that provide backup control of systems automatically controlled by the MPS are located in the MCR
    - MCR safety-related manual switches associated with MPS actuations (e.g., reactor trip and containment isolation) are isolated using switches located outside the MCR
  - Containment Fire Area
    - Fire in containment is highly unlikely – held at a vacuum, electrical conductors are of noncombustible construction or routed in conduit, inaccessible during operation
    - Flooding of containment for heat removal during shutdown prevents ignition or spread of fire

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9A - Fire Hazards Analysis

- FHA Special Cases
  - Top-of-Module Fire Area
    - Fire on the top of the NPM is highly unlikely – enclosed in bioshield, all cabling under bioshield is in conduit or 3-hr rated cable
    - Top of NPM is inaccessible during power operations
    - Hydraulic fluid for valve controls is noncombustible
    - CIV/DHRS valve actuation based on stored pressurized nitrogen accumulators on each valve – places valve in its safe position when hydraulic operating pressure is relieved by the remotely located hydraulic unit
    - Module has 3-hr fire barrier on each side with exception of bioshield wall facing the pool - Greater than 20 feet between intervening combustibles on pool side
    - During shutdown bioshield is removed and administrative controls limit transient combustibles on the top of the NPM

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9.5.1 and 9A

- COL items for FP
  - No unique COL items for FP
  - Generic COL items related to FP program development, procedure development, implementation schedule, QA program and FP training and qualifications
  - Provide Final Safe Shutdown Analysis – based on as-built design – cable routing
- COL items for FHA
  - No unique COL items for FHA
  - COL applicant is required to provide a site specific, as-built FHA per RG 1.189
  - COL applicant is required to provide a site specific, as-built SSA per RG 1.189

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Acronyms

- ANB – Annex Building
- ANSI - American National Standards Institute
- ASME – American Society of Mechanical Engineers
- BAS - Boron Addition System
- BDBE – Beyond Design Basis Event
- BPDS - Balance-of-Plant Drain System
- CAS - Compressed Air Systems
- CES - Containment Evacuation System
- CFDS - Containment Flooding and Drain System
- CFR - Code of Federal Regulations
- CFWS – Condensate and Feedwater System
- CHWS – Chilled Water System
- CIVs - Containment Isolation Valves
- CNTS – Containment System
- CNV - Containment Vessel
- COL- Combined License
- COMS - Communication System
- CRB - Control Building
- CRDM – Control Rod Drive Mechanism
- CRE - Control Room Envelope
- CRHS – Control Room Habitability System
- CST – Condensate Storage Tank
- CUB – Central Utility Building
- CVCS - Chemical and Volume Control System
- DBA – Design Basis Accident
- DHRS – Decay Heat Removal System
- DWS – Demineralized Water System

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Acronyms

- EFDS – Equipment and Floor Drain Systems
- ESF – Engineered Safety Features
- FDS – Fire Detection System
- FHA - Fire Hazards Analysis
- FPP – Fire Protection Program
- FPS – Fire Proteciton System
- FSS - Fire Suppression System
- GRWS – Gaseous Radioactive Waste System
- HEPA – High Energy Particulate Air
- HVAC – Heating, Ventilating and Air Conditioning
- IAS - Instrument Air System
- IEEE - Institute of Electrical and Electronic Engineers
- LOCA – Loss of Coolant Accident
- LRWS – Liquid Radioactive Waste System
- MCR - Main Control Room
- MHA – Maximum Hypothetical Accident
- MHS - Module Heatup System
- MLA - Module Lifting Adapter
- MPS - Module Protection System
- NDS - Nitrogen Distribution System
- NFPA - National Fire Protection Association
- NPM - NuScale Power Module
- OHLHS - Overhead Heavy Load Handling Systems
- PASS – Post-Accident Sampling System
- PLS – Plant Lighting System
- PRA – Probabilistic Risk Assessment
- PSS - Process Sampling System

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Acronyms

- PWS – Potable Water System
- RBC - Reactor Building Crane
- RBVS - Reactor Building Ventilation System
- RCCWS – Reactor Component Cooling Water System
- RCS - Reactor Coolant System
- RG - Regulatory Guide
- RWB – Radioactive Waste Building
- RWDS - Radioactive Waste Drain System
- RXB - Reactor Building
- SAS - Service Air System
- SCWS – Site Cooling Water System
- SFP – Spent Fuel Pool
- SFPCS - Spent Fuel Pool Cooling System
- SSA - Safe Shutdown Analysis
- SSC - Structure System or Component
- SSE – Safe Shutdown Earthquake
- TGB – Turbine Generator Building
- TS – Technical Specifications
- TSC – Technical Support Center
- UHS - Ultimate Heat Sink
- UWS – Utility Water System

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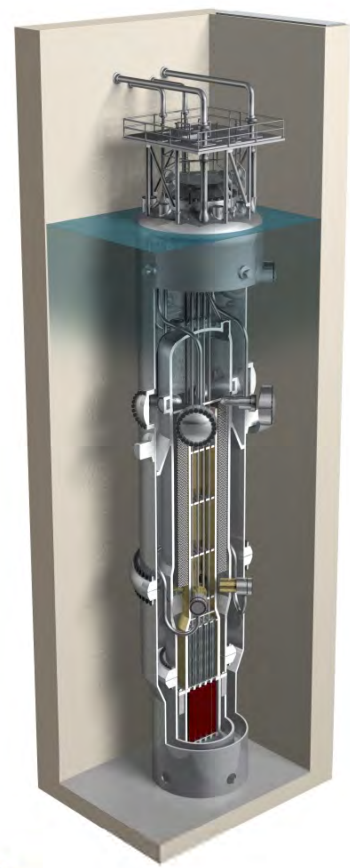

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Presentation to the ACRS Subcommittee

NuScale Power, LLC (NuScale)

Design Certification Application Review

Safety Evaluation with Open Items: Chapter 9

AUXILIARY SYSTEMS

March 20, 2019

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Alissa Neuhausen - NRO/DEI/SEB
Ryan Nolan - NRO/DSRA/SRSB
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Nan Chien - NRO/DSRA/SCPB
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Chapter 9: Auxiliary Systems

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Section 9.2 - Water Systems	Chang Li - SCPB Angelo Stubbs - SCPB Nan Chien - SCPB Bob Vettori - SPRA
Section 9.3 - Process Auxiliaries	Raul Hernandez - SCPB Tony Gardner - MCCB Edward Stutzcage - RGRB Bob Vettori - SPRA Hanry Wagage – SCPB Ryan Nolan - SRSB
Section 9.4 - Air Conditioning, Heating, Cooling, and Ventilation Systems	Nan Chien - SCPB
Section 9.5 - Other Auxiliary Systems	Robert Vettori - SPRA Dawnmathews Kalathiveettil - EICA Fanta Sacko - EENB Nan Chien - SCPB

Section 9.1: Fuel Storage and Handling

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Subsections:

9.1.1 - Criticality Safety of New and Spent Fuel Storage

9.1.2 - Fuel Storage System

9.1.3 - Spent Fuel Pool Cooling and Cleanup System

9.1.4 - Light and Heavy Systems and Heavy Load Handling System

9.1.1 - Criticality Safety of New and Spent Fuel Storage (1/2)

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Review objective: verify that fuel will remain subcritical during storage and handling in accordance with 10 CFR 50.68 and GDC 62

- Review guidance: SRP Section 9.1.1, Revision 3
- Review areas: computational methods and validation, scope of conditions analyzed, analysis models and assumptions, handling of bias and uncertainty, analysis results, storage rack materials

Staff Review and Findings:

- Criticality calculation methodology is acceptable and adequately benchmarked
- Criticality models correctly incorporate design information and use appropriate assumptions
- Open Item 9.1.1-2: Applicant is making the structural analysis of the racks a COL item, which the staff will evaluate in Phase 4

9.1.1 - Criticality Safety of New and Spent Fuel Storage (2/2)

Staff Review and Findings, continued:

- Except for analyses related to Open Item 9.1.1-2, applicant's criticality analyses comply with 10 CFR 50.68 and GDC 62
- Staff confirmatory analyses support the applicant's results
- Tier 1 information complete and consistent with Tier 2 information
- Technical Specifications adequately protect the assumptions in the criticality analysis
- NuScale has not specified a Neutron Absorbing material.
  - Open item 9.1.1-1: Request additional details on the impacts of manufacturing on the neutron attenuation, materials qualification for the SFP environment, and manufacturing process controls.
- The applicant is implementing a neutron absorber monitoring program that is consistent with TSTF-577, Rev. 1, and utilizes NEI 16-03, "Guidance for Monitoring of Fixed Neutron Absorbers in Spent Fuel Pools."

9.1.2 - New and Spent Fuel Storage (1/1)

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Review Objective

- The SFP has the necessary design features unique to fuel storage during initial receipt, refueling operations, and accident conditions, including maintaining cooling and limiting offsite exposure in the event of a fuel handling accident.

Items of major interest

- Staff reviewed the SFP in accordance with the guidance of SRP 9.1.2
- SFP is integral part of the UHS
- Shared system between all 12 modules

Open Items

- 9.1.2-1 – Related to the pool leakage detection system capability to monitor the UHS liner integrity - **Based on latest RAI response and proposed DCA changes, the staff finds this Open Item Closed**.

- Staff finds that the SFP conforms with the guidance of SRP 9.1.2

9.1.3 - Spent Fuel Pool Cooling and Cleanup System (1/1)

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Review Objective

- Ensure the safety-related Spent Fuel Pool Cooling System is capable of maintaining the spent fuel assemblies cooled and covered with water during all storage conditions.

Items of major interest

- Staff reviewed SFPCCS in accordance with the guidance in SRP 9.1.3
- Ensure system configuration ensures adequate water inventory in the SFP
- Evaluate all key assumptions of the thermal analyses
- Cooling capability during accident scenarios addressed in Section 9.2.5 UHS

- RAI responses found acceptable and proposed changes to DCD are being tracked as Confirmatory Items
- Staff finds that the SFPCS conforms with the guidance of SRP 9.1.3

9.1.4 and 9.1.5 - Light Load Handling Systems (Related to Refueling) (1/1)

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Review Objective

To ensure the design and operation of the load handling systems can prevent or minimize the likelihood of an event that could cause a release of radioactivity, a criticality accident, or an inability to cool fuel within the reactor vessel or spent fuel pool; or could prevent the safe shutdown of the reactor.

Items of major interest

- Responses of load handling equipment to a safe shutdown earthquake (SSE) event consistent with guidance in SRPs 9.1.4 and 9.1.5.
- Design criteria for single-failure-proof load handling components consistent with guidelines of NUREG-0554
- Procedures for safe load handling operations consistent with guidelines of NUREG-0612
- ALL RAI responses and DCA changes are found acceptable.

Staff finds that the light and heavy load handling conforms with the guidance of SRPs 9.1.4 and 9.1.5.

Section 9.2 - Water Systems

Subsections:

9.2.1 - Station Service Water System

9.2.2 - Component Cooling Water System

9.2.3 - Demineralized Water System

9.2.4 - Potable and Sanitary Water Systems

9.2.5 - Ultimate Heat Sink

9.2.6 - Condensate Storage Facilities

9.2.7 - Site Cooling Water System

9.2.8 - Chilled Water Systems

9.2.9 - Utility Water Systems

9.2.1 - Station Service Water System (1/1)

Key Design Features and Considerations

- The NuScale passive design does not rely on active systems such as a service water system to provide cooling to essential equipment.
- The NuScale Power Modules are partially submerged in the reactor pool portion of the plant UHS.
- This design configuration ensures passive heat transfer from essential systems and components directly to the UHS, with no intermediate heat transfer loop such as that provided by a typical LWR essential service water system.

Staff's Review and Evaluation Results

- The staff reviewed the NuScale system design and confirmed the above statements.
- The UHS, reviewed in Section 9.2.5 of this SER, performs the safety function of decay heat removal, which the service water system usually performs in LWR active designs.
- The site cooling water system (SCWS), reviewed in Section 9.2.7 of this SER, performs the heat removal function for the nonsafety-related systems, which the service water system usually performs in LWR designs.

9.2.2 - Component Cooling Water System (1/1)

Key Design Features and Considerations

- RCCWS is a closed loop cooling system comprised of two identical subsystems each supporting up to six reactor modules.
- RCCWS provides cooling to the CVCS non-regenerative heat exchanger, CES condenser and pumps, the PSS coolers and analyzer control units and the CRDMs electromagnetic coils housing .
- RCCWS is not safety-related, however the CVCS, CES, and PSS components cooled by the RCCWS are located in the RXB, the CRDM electromagnetic coils, are located inside containment and outside of the reactor vessel.

Staff’s Review and Evaluation Results

- Staff reviewed CCWS in accordance with SRP 9.2.2. Since the system is not credited for performing any safety related function the review focused on system design that ensures system failure would not impact important to safety SSCs.
- The system was found to be in compliance with GDCs 2 and 4 based on the RCCWS SSCs inside the reactor building and containment having appropriate seismic classifications and having appropriate environmental qualifications . The staff also found the system to be in compliance with GDCs 60 , 64.

9.2.3 - Demineralized Water System (1/1)

Key Design Features and Considerations

- The DWS is designed to treat the water from the utility water system (UWS) and provide and distribute high-quality demineralized water to the plant.
- The DWS provides plant support during abnormal conditions by providing additional makeup water to the SFPCS to compensate for inventory loss, and to the condenser for emergency fill.
- The DWS is not safety-related and is not required for any DBE.

Staff's Review and Evaluation Results

- Staff reviewed the systems in accordance with SRP 9.2.3
- The staff found the design to be in accordance with the guidance in SRP 9.2.3, and in compliance with GDCs 2, 5, 60 and 10CFR 20.1406.

9.2.4 - Potable and Sanitary Water Systems (1/1)

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Key Design Features and Considerations

- Systems provide potable water for domestic use and human consumption and collects and transfers site sanitary waste for treatment and discharge.
- The PWS piping in the control building penetrating the control room envelope and habitability boundary.
- Systems are site specific therefore only conceptual design information, interface requirements and design features to prevent radioactive contamination of the system is included in the application.

Staff's Review and Evaluation Results

- Staff reviewed the systems in accordance with SRP 9.2.4
- The staff found the conceptual design information and the interface requirements in the DCD to be in accordance with 10 CFR 52.47(a)
- Design provision made to prevent inadvertent contamination of the domestic water and sanitary systems with radioactive material was found to be acceptable and the design is in compliance with GDC 60.
- PWS uses passive design feature (loop seal) instead of isolation valves to ensure integrity of CR envelope.

9.2.5 - Ultimate Heat Sink (1/1)

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Key Design Features and Considerations

- The NuScale UHS is a set of safety-related pools of borated water that consists of the combined water volume of the reactor pool, RFP, and SFP.
- Up to 12 NPMs are located in the reactor pool and share the combined volume of water.
- During accident scenarios, the NuScale design credits the safety-related water inventory stored in the UHS to passively remove the decay heat.
- The applicant considered GDCs 2, 4, 5, 45, 46, 61, and PDC 44 in the design.

Staff's Review and Evaluation Results

- The staff evaluates the safety-related UHS function in this section of the SER.
- The staff's review confirmed that the UHS has sufficient water inventory for 30 days to remove the decay heat from the NPM and the stored fuel from the pool through boiling and evaporation, removing enough heat to maintain the spent fuel and fuel in the NPMs sufficiently cool to prevent fuel damage.

9.2.6 - Condensate Storage Facilities (1/1)

Key Design Features and Considerations

- The CST provides a volume for makeup and rejection of condensate to and from the condenser hotwell based on hotwell level.
- The CST does not serve a safety function, and it does not interface with other systems that could adversely affect safety-related or augmented equipment.
- In the event of failure of non-seismic CSF storage tanks, site grading and drainage will ensure SSCs important to safety will not be adversely affected

Staff's Review and Evaluation Results

- Staff reviewed CSF in accordance with SRP 9.2.6.
- The system was found to be in compliance with GDCs 2 and 60 and 10 CFR 20.1406.
- GDCs 5, 44, 45, and 46 were found not applicable because system is not safety-related and not credited for supporting safety-related systems.

9.2.7 - Site Cooling Water System (1/1)

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Key Design Features and Considerations

- Transfer heat from plant auxiliary systems to the SCWS cooling towers.
- Service loads for the SCWS include equipment in the RXB, Central Utility Building, North and South TGB, and Auxiliary Boiler Building.
- SCWS does not provide cooling to safety-related or risk-significant SSCs, and is not credited for performing any safety-related functions.

Staff's Review and Evaluation Results

- The staff reviewed the SCWS to determine if system failures could potentially impact import to safety SSCs (GDCs 2, 4 & 5) or result in the release of radiation to environment (GDC 60).
- Staff concluded that the SCWS complies with GDC 2, 4, 5 and 60 as well as 10 CFR 20.1406.

9.2.8 - Chilled Water System (1/1)

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Key Design Features and Considerations

- Primary CHWS provides cooling for the normal CRVS, RWBVS, RBVS, and other equipment in the radioactive waste processing systems.
- Standby CHWS is dedicated only to the CRVS in the event of a loss of normal ac power.

Staff's Review and Evaluation Results

- Staff reviewed CHWS in accordance with SRP 9.2.7.
- The CHWS has the heat removal capacity to provide, during plant normal operation, a heat sink for various air handling units and cooling loads in the radioactive waste processing systems
- The CHWS design complies with GDCs 1, 2, 4, 5, 44, 10 CFR 20.1406, and 10CFR52.47(b)(1)

9.2.9 - Utility Water Systems (1/1)

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Area of Review

- Staff reviewed the Utility Water System in accordance with the applicable guidance in SRP 9.2.4.
- Staff reviewed to ensure compliance with GDC 5, 60, 64, and 10CFR 20.1406.
- RAIs issued, all closed/resolved. No open items.

Staff's Findings

- The Utility Water System is in accordance with the applicable guidance in SRP 9.2.4 and is in compliance with GDC 5, 60, 64 and 10CFR 20.1406.

Section 9.3 : Process Auxiliaries

Subsections:

- 9.3.1 - Compressed Air Systems
- 9.3.2 - Process and Post-Accident Sampling Systems
- 9.3.3 - Equipment and Floor Drain Systems
- 9.3.4 - Chemical and Volume Control System
- 9.3.5 - Not Applicable
- 9.3.6 - Containment Evacuation System and Containment Flooding and Drain System

9.3.1 - Compressed Air Systems (1/1)

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Review Objective

- Ensure the quality of air does not prevent the safety-related air operated valves from performing their safety function

Items of major interest

- Staff reviewed compressed air and gas systems in accordance with the guidance in SRP 9.3.1.
- CAS has no safety function, but the system follows the guidance that ensures high quality air is available.
- The CAS is shared between 12 modules simultaneously.

- Staff finds that the compress air and gas systems conforms with the guidance of SRP 9.3.1

9.3.2 - Process and Post-Accident Sampling Systems (1/3)

System Description

The function of the process sampling system (PSS) is to provide the means to obtain representative liquid and gaseous samples from various primary and secondary process streams and components for monitoring and analyzing the chemical and radiochemical conditions.

Scope of Review

- Sampling locations and capabilities
- Detection and control of leakage outside containment
- Containment isolation

9.3.2 - Process and Post-Accident Sampling Systems (2/3)

Conclusions

- No dedicated post-accident PSS used. Normal PSS is used during post-accident conditions.
- There are sample points for collecting and analyzing liquid and gaseous samples from the primary coolant, secondary coolant, containment air, spent fuel pool, and other sources
- Includes the sampling locations in the SRP acceptance criteria, and other locations
- COL applicant will ….
  - Develop operational procedures and maintenance program for leak detection and contamination control
  - Develop post-accident sampling contingency plans for using the process sampling system and the containment evacuation system off-line radiation monitor to obtain reactor coolant and containment atmosphere samples.

9.3.2 - Process and Post-Accident Sampling Systems (3/3)

Conclusions (continued)

- The process of collecting liquid post-accident samples involves opening containment isolation valves and routing liquid to the CVCS system to the sampling lines for liquid samples.
- The process of collecting gaseous post-accident samples involves opening containment isolation valves and routing gases through the containment evacuation system to the sample lines and back to containment through the containment flood and drain system.
  - Staff has several open questions related to the post-accident sampling process, including radiological concerns related to post-accident sampling and the dose to operators when taking samples, in accordance with 10 CFR 50.34(f)(2)(viii). However, NuScale has recently proposed an exemption from 10 CFR 50.34(f)(2)(viii), which is currently being evaluated by staff. The staff will re-assess the status of the open questions following the review of the exemption.

9.3.3 - Equipment and Floor Drain Systems (1/1)

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Area of Review

- The EFDS is comprised of two separate, unconnected systems, the radioactive waste drain system (RWDS,) and the balance-of-plant drain system (BPDS).
- Staff reviewed the EFDS is accordance with the guidance in SRP 9.3.3.
- Staff reviewed to ensure compliance with GDC 2, 4, 5, 60, 64, and 10CFR 20.1406.
- RAIs issued, all closed/resolved. No open items.

Staff Findings

- The EFDS conforms with the guidance of SRP 9.3.3 and is compliance with GDC 2, 4, 5, 60, 64, and 10CFR 20.1406.

9.3.4 - Chemical and Volume Control System (1/3)

Findings

- **Design Basis**

  Only safety-related functions of the CVCS are containment and demineralized water isolation.

- **Functionality**
  - System functional design to include all components necessary for RCS chemical and volume control
  - Components, instrumentation and controls appropriate
  - Adequate chemistry/purity control
  - Utilized to establish natural circulation flow to support startup operations

- **Protection**
  - Protection against boric acid precipitation adequate
  - Appropriate seismic classification completed for CVCS
  - Purifying equipment appropriately protected
  - CVCS leakage control adequate to minimize contamination

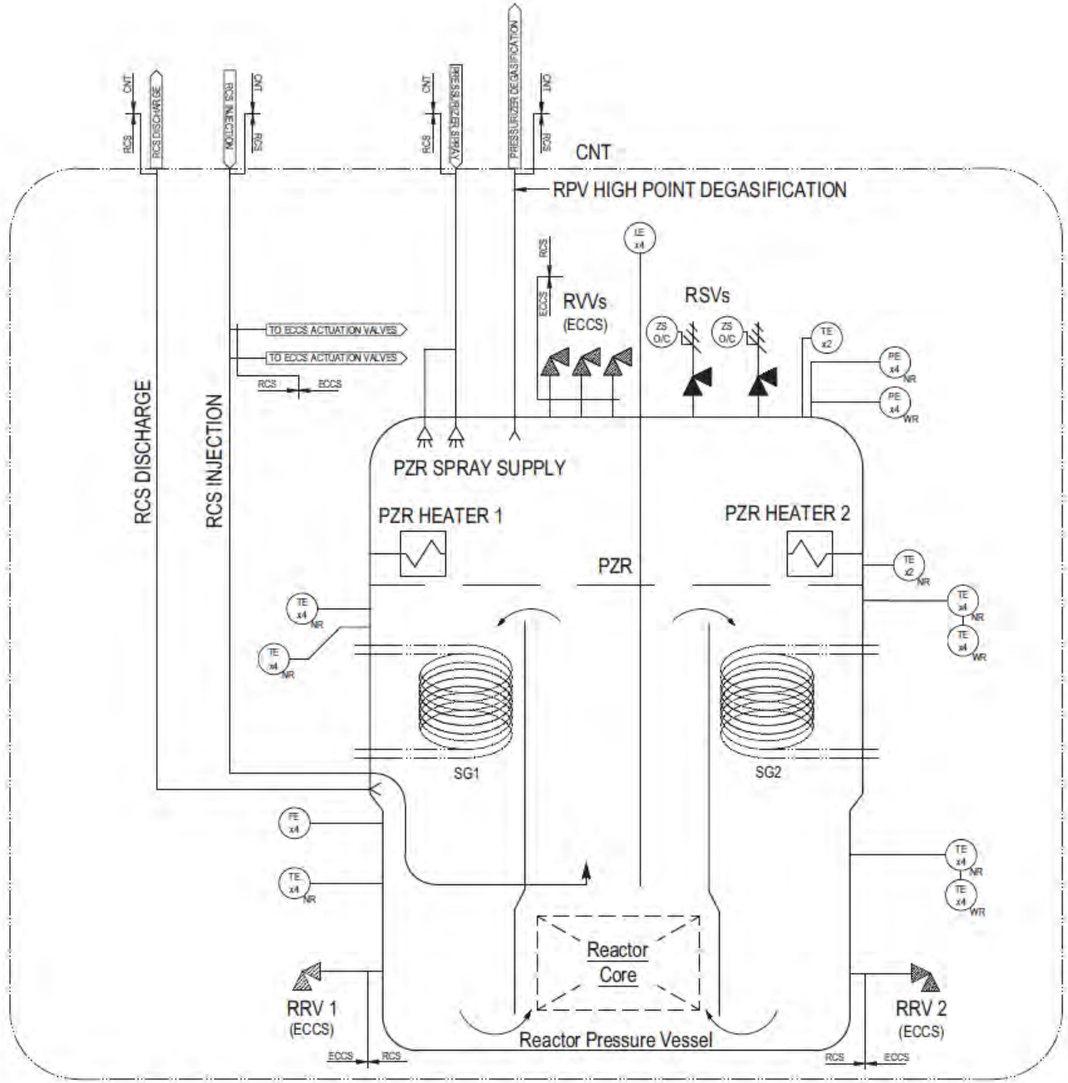
9.3.4 - Chemical and Volume Control System (2/3)

GDC 33 Exemption Request

- Not applicable to NuScale – RCS makeup from CVCS is not relied on to meet SAFDLs
- Underlying purpose of GDC 33 is met by:
  - CVCS isolation capability
  - CNV and ECCS capability to maintain RCS inventory and core coolability to meet the SAFDLs

9.3.4 - Chemical and Volume Control System (3/3)

- CVCS injects into the lower riser 8 feet above the core.
- Dilution mixing methodology during startup is reviewed in Section 15.4.6.
- T-H phenomena for boron dilution in the long term is reviewed in Section 15.0.6.



9.3.6 - Containment Evacuation System and Containment Flooding and Drain System (1/1)

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Area of Review

- Staff reviewed the CES and CFDS design function of transferring liquids and gases between the containment vessel free volume and other plant systems

Staff Findings

- In response to staff RAIs the applicant clarified and updated NuScale FSAR for the following:
    - CFDS and CVCS systems provide single failure proof capability to cool the nuclear power module from normal operating conditions to conditions equivalent to cold shutdown in a conventional plant
    - With the discharge line isolated, the radioactive gases are left in the CFDS to decay until the level is below the limit for release through the RBVS plant exhaust stack
    - Exemption from 10 CFR 50.34(f)(2)(xiv)(E) as applied to the CES is described in Part 7, Chapter 13
- The CES and CFDS conforms with the guidance of DSRS 9.3.6

Section 9.4: Heating, Ventilation and Air Conditioning Systems

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Subsections:

9.4.1 - Control Room Area Ventilation System

9.4.2 - Reactor Building and Spent Fuel Pool Area Ventilation System

9.4.3 - Radioactive Waste Building Ventilation System

9.4.4 - Turbine Building Ventilation System

9.4.1- Control Room Area Ventilation System (1/1)

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Area of Review

- Staff reviewed the CRVS design to determine whether it is capable of maintaining suitable environmental conditions for personnel under normal conditions.

Staff Findings

- NuScale does not credit operator actions to mitigate DBEs.
- CRVS is not credited to be operational during design-basis accidents.

Staff Conclusion

- Staff concluded that the CRVS complies with GDC 2, 4, 5, 19 and 60 as well as with 10 CFR 20.1406, 10 CFR 50.63 and 10 CFR 52.47(b)(1).

9.4.2 - Reactor Building and Spent Fuel Pool Area Ventilation System (1/1)

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Area of Review

- The capability of maintaining acceptable ambient conditions in the Reactor Building to support personnel and equipment.
- The ability of controlling airborne radioactivity in the area during normal operation and fuel handling accident.

Staff Findings

- RBVS spent fuel pool exhaust ductwork provides a protective function for the safety-related, Seismic Category I Reactor Building
- NuScale did not take credit to meet offsite dose limit due to leakage from the Reactor Building in case of a HELB event.

Staff Conclusion

Staff concluded that the RBVS complies with GDC 2, 5, 60, 61 and 64 as well as with 10 CFR 20.1406 and 10 CFR 52.47(b)(1).

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9.4.3 - Radioactive Waste Building Ventilation System (1/1)

Area of Review

- The capability to support personnel access and equipment functions by maintaining a suitable operating environment in the RWB.
- The ability to support the control of radioactive contamination by maintaining airflow from areas of lesser potential contamination to areas of greater potential contamination, maintaining the RWB at a negative pressure with respect to the outside atmosphere, and collecting potentially contaminated discharges vented from equipment in the RWB.

Staff Findings

- Staff finds that the RWBVS conforms with the guidance of SRP 9.4.3.

Staff Conclusion

Staff concluded that the RBVS complies with GDC 2, 5, and 60 as well as with 10 CFR 20.1406 and10 CFR 52.47(b)(1).

9.4.4 - Turbine Building Ventilation System (TBVS) (1/1)

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Area of Review

- Capability to maintain a suitable environment for all equipment and personnel in the TGB during startup, shutdown, and normal plant operation.

Staff Findings

- Staff finds that the TBVS conforms with the guidance of SRP 9.4.4.

Staff Conclusion

- Staff concluded that the TBVS complies with GDC 2, 5, and 60 as well as with 10 CFR 52.47(b)(1).

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Section 9.5: Other Auxiliary Systems

Subsections:

9.5.1 - Fire Protection Program

9.5.2 - Communication Systems

9.5.3 - Lighting Systems

9.5.4 - Emergency Diesel Engine (Backup Diesel Generators) Support Systems

9.5.1 - Fire Protection Program (1/1)

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Review Objectives

- Staff reviewed the Fire Protection Program and Fire Hazard Analysis in accordance with the applicable guidance in SRP 9.5.1.1, SECY 90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements." and Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants."
- Staff reviewed to ensure compliance with 10CFR 50.48, "Fire Protection," GDC 3, 5, 19, and 23.
- RAIs issued, all closed/resolved. No open items.

Staff's Findings

- The Fire Protection Program and Fire Hazard Analysis are in accordance with the applicable guidance in SRP 9.5.1.1, SECY 90-016, Regulatory Guide 1.189, and are in compliance with 10CFR 50.48, GDC 3, 5, 19, and 23.

9.5.2 - Communication Systems (1/1)

- The staff finds that the COMS designs are acceptable and meet the applicable requirements of 10 CFR Part 50, and 10 CFR Part 52.
- No open items.
- The NRC staff is present to address any questions or comments from the Committee members.

9.5.3 - Lighting Systems (1/2)

NuScale Design:

The plant lighting system includes normal plant lighting, emergency plant lighting, and normal and emergency main control room lighting.

- Normal plant lighting provides illumination on the plant site and for plant buildings
- Emergency plant lighting provides illumination outside the control room upon loss of normal lighting
- Normal and Emergency Main Control Room lighting provides illumination under all operating, maintenance, testing, and emergency conditions

Staff Review:

Staff reviewed the capability of the plant lighting levels to provide adequate illumination in all plant areas during all plant operating conditions, and to operate without adversely impacting the operation, control, and maintenance of structure, system and components.

9.5.3 - Lighting Systems (2/2)

Open Item:

- The open item in this section is related to RG 1.75 and the physical separation between nonsafety-related lighting circuits and safety-related circuits, and GDC 17 and 18 exemptions.
  - The completion of the staff's review of plant lighting system is awaiting the completion of Chapter 8 open item, regarding the GDC 17 and 18 exemptions.

Confirmatory Item:

- The confirmatory item will be addressed in our Phase 4 SER.

9.5.4 - Emergency Diesel Engine (Backup Diesel Generators) Support Systems (1/1)

- The applicant did not provide any physical or functional description for these systems in FSAR Section 9.5.
- In response to the staff's RAI, the applicant stated that the backup diesel generators (BDGs) are described in FSAR Section 8.3.1.1.2 as stand-alone, skid-based installations including support features such as lubrication and cooling. These support features are not assigned to discrete systems similar to those used for traditional large LWR safety-related diesel applications as presupposed by the guidance in SRP 9.5.4 through 9.5.8. Hence no need to include these SRP sections in the FSAR.
- The staff agrees with the applicant determination that the EDESS are nonsafety, non-risk-significant.
- No open items.
- The NRC staff is present to address any questions or comments from the Committee members.

ACRONYMS

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BDG – backup diesel generator
RCCWS – reactor component cooling water system
CDI – conceptual design information
COL – combined license
CSF – condensate storage facility
CST – condensate storage tank
CVCS – chemical and volume control system
CWS – chilled water system
DBA – design basis accident
DCD – design control document
EDG – emergency diesel generator
EFDS – equipment and floor drainage systems
GDC – general design criterion
RAI – request for additional information
RCS – reactor coolant system
SAFDL - specified acceptable fuel design limit
SFP – spent fuel pool
SFPCS – spent fuel pool cooling system
SFPCCS – spent fuel pool cooling and cleanup system
SRP – standard review plan
SSC – structure, system and component
SSE – safe shutdown earthquake
ITAAC – inspections, tests, analyses, and acceptance criteria
TBVS – turbine building ventilation system
UHS – ultimate heat sink

March 7, 2019 Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Presentation: NuScale Technical Specifications FSAR Chapter 16 and DCA Part 4," PM-0319-64414, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on Reactor Safegurards (ACRS) NuScale Subcommittee meeting on March 20, 2019. The materials support NuScale's presentation of Chapter 16 and Design Certification Application (DCA) Part 4, "Technical Specifications."

Enclosure 1 is the nonproprietary version of the presentation entitled "ACRS Presentation: NuScale Technical Specifications FSAR Chapter 16 and DCA Part 4," PM-0319-64414, Revision 0.

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

If you have any questions, please contact Carrie Fosaaen at 541-452-7126 or at cfosaaen@nuscalepower.com.

Sincerely,

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Enclosure 1: "ACRS Presentation: NuScale Technical Specifications FSAR Chapter 16 and DCA Part 4," PM-0319-64414, Revision 0

Enclosure 1:

"ACRS Presentation: NuScale Technical Specifications FSAR Chapter 16 and DCA Part 4," PM-0319-64414, Revision 0

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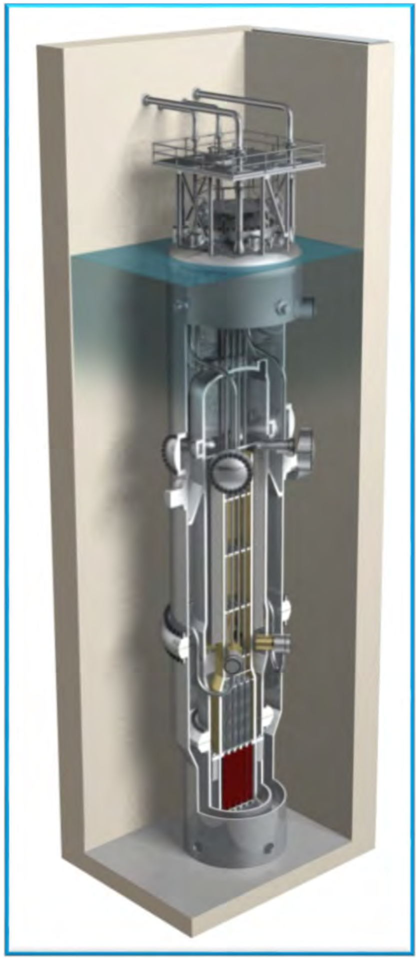
ACRS Presentation: NuScale Technical Specifications FSAR Chapter 16 and DCA Part 4



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March 20, 2019

PM-0319-64414
Revision: 0

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Chapter 16: Technical Specification

- Different TS
  - Simpler natural circulation and single vessel design
  - Passive ECCS with passive support systems
  - Digital module protection system
  - No safety-related electrical or offsite power needed
  - No credited HVAC, no active support systems
  - Reactor relocated for refueling activities
- Smaller than existing TS
  - Approx. 150 pages of Technical Specifications
    - only about 90 pages of LCOs
    - *(AP1000 approx. 350 pages)*
  - Approx. 275 pages of Bases
    - *(AP1000 approx. 670 pages)*

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Content Selection

- Applied 10 CFR 50.36 criteria to NuScale design, operations, and safety analyses
  - Technical report describing process and results
  - Includes LCO Criterion 4 content
    - Manual Actuation Function
    - Remote Shutdown Station
- Developed led by the Operating organization
  - Incorporate industry and staff operating experience
- Adoption of Industry Practices as Appropriate
  - Used STS Writer's Guide format and guidance
  - Monitored NRC/Industry working group on Tech Spec content
    - Considered, adapted, and incorporated concepts as appropriate

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NuScale MODE Definitions

MODE	Title	Reactivity Condition (keff)	Indicated Reactor Coolant Temperatures (°F)
1	Operations	≥ 0.99	All ≥ 420
2	Hot Shutdown	< 0.99	Any ≥ 420
3	Safe Shutdown[a]	< 0.99	All < 420
4	Transition[b][c]	< 0.95	N/A
5	Refueling[d]	N/A	N/A

(a) Any CRA capable of withdrawal, any CVCS or CFDS connection to the module not isolated.

(b) All CRAs incapable of withdrawal, CVCS and CFDS connections to the module isolated, and all reactor vent valves electrically isolated.

(c) All reactor vessel flange bolts fully tensioned.

(d) One or more reactor vessel flange bolts less than fully tensioned.

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Limiting Conditions for Operation

3.3 Instrumentation

- 3.3.1 Module Protection System Instrumentation
- 3.3.2 Reactor Trip System Logic and Actuation
- 3.3.3 Engineered Safety Features Actuation System Logic and Actuation
- 3.3.4 Manual Actuation Functions
- 3.3.5 Remote Shutdown Station

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Limiting Conditions for Operation

3.5 Passive Core Cooling Systems

- 3.5.1 Emergency Core Cooling System
- 3.5.2 Decay Heat Removal System
- 3.5.3 Ultimate Heat Sink

3.6 Containment Systems

- 3.6.1 Containment
- 3.6.2 Containment Isolation Valves

3.7 Plant Systems

- 3.7.1 Main Steam Isolation Valves
- 3.7.2 Feedwater Isolation
- 3.7.3 In-Containment Secondary Piping Leakage

3.8 Refueling Operations

- 3.8.1 Nuclear Instrumentation
- 3.8.2 Decay Time

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Technical Specification RAI Received and Addressed

- 60 Chapter 16 RAI Received
  - Many included multiple subparts,
    - e.g., 16-60 handled as 81 subparts.
  - 59 have been responded to
    - 16-59, and 2 of 81 subparts of RAI 16-60 pending
  - 13 supplemental responses submitted

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Key Technical Specification Questions

- STS Content Evolution Based on Operating Plants
- Instrumentation Surveillance Testing
- Response Time Testing
- RCS Leak Rate Monitoring
- RCS Specific Activity Limit
- In-Containment Secondary Piping Leakage

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Conclusion

NuScale has developed design-specific Technical Specifications and Bases that

- Reflect NuScale design, analyses, and operations
- Ensure safety and licensing bases will be implemented during operations
- Align with industry standard technical specifications format and content
- Includes consideration of industry experience
- Ready for COL applicant to finalize and submit with COL application

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Acronyms

CFDS	Core Flood and Drain System
COLA	Combined Operating License Applicant
CRA	Control Rod Assembly
CVCS	Chemical and Volume Control System
ECCS	Emergency Core Cooling System
ESFAS	Engineered Safety Features Actuation System
FW	Feedwater (as in FW Isolation Valve)
HVAC	Heating, Ventilation, and Air Conditioning
ICIS	In-Core Instrument System
LCO	Limiting Condition for Operations
MCS	Module Control System
MPS	Module Protection System
MSIV	Main Steam Isolation Valve
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RTS	Reactor Trip System
SFCP	Surveillance Frequency Control Program
SR	Surveillance Requirement
SSC	Structures, Systems, and Components
STS	Standard Technical Specifications (NUREGs as published by NRC)
TS	Technical Specifications

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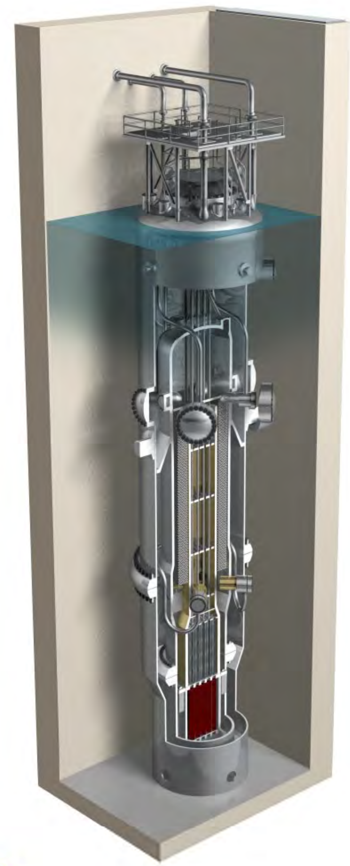

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Presentation to the ACRS Subcommittee

NuScale Power, LLC

NuScale Design Certification Application Review

Safety Evaluation with Open Items: Chapter 16

GENERIC TECHNICAL SPECIFICATIONS AND BASES

MARCH 20, 2019

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- **Project Managers**
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Outline

- Overview of Chapter 16
- Technical Topics
  - Defined Terms
  - RTS and ESFAS Surveillance Requirements
  - Electrical Power Surveillance Requirements
  - Allowed Leakage for Main Steam and Feedwater Isolation Valves
  - Exceptions to Meeting Surveillances
  - Open Item Listing
  - Pending MPS design changes
  - General Issues
- Review Status Summary

Overview of Design Certification Application (DCA) Part 2, Chapter 16; and Part 4

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1.0 USE AND APPLICATION
1.1 Definitions
1.2 Logical Connectors
1.3 Completion Times
1.4 Frequency

3.0 LCO APPLICABILITY
LCO 3.0.1
LCO 3.0.2
LCO 3.0.3
LCO 3.0.4
LCO 3.0.5
LCO 3.0.6
LCO 3.0.7
LCO 3.0.8

3.0 SR APPLICABILITY
SR 3.0.1
SR 3.0.2
SR 3.0.3
SR 3.0.4

2.0 SAFETY LIMITS (SLs)
2.1 SLs
2.1.1 Reactor Core SLs
2.1.2 RCS Pressure SL
2.2 SL Violations

3.0 LCO AND SR (#) – number of LCO subsections
3.1 REACTIVITY CONTROL SYSTEMS (9)
3.2 CORE OPERATING LIMITS (2)
3.3 INSTRUMENTATION (5)
3.4 REACTOR COOLANT SYSTEM (RCS) (10)
3.5 PASSIVE CORE COOLING SYSTEMS (PCCS) (3)
3.6 CONTAINMENT SYSTEMS (2)
3.7 PLANT SYSTEMS (3)
3.8 REFUELING OPERATIONS (2)

4.0 DESIGN FEATURES (3 sections)

5.0 ADMINISTRATIVE CONTROLS (7 sections)

Overview of DCA Part 4
LCO selection criteria results

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Criterion 1

3.4.7, RCS Leakage Detection Instrumentation

Criterion 2

3.1.1 SHUTDOWN MARGIN (SDM)
3.1.2 Core Reactivity
3.1.3 Moderator Temperature Coefficient (MTC)
3.1.4 Rod Group Alignment Limits
3.1.5 Shutdown Bank Insertion Limits
3.1.6 Regulating Bank Insertion Limits
3.1.7 Rod Position Indication
3.1.9 Boron Dilution Control (Boric Acid supply boron concentration; demineralized water makeup flow rate)
3.2.1 Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)
3.2.2 AXIAL OFFSET (AO)
3.4.1 RCS Pressure, Temperature, and **Flow Resistance** Critical Heat Flux Limits
3.4.2 RCS Minimum Temperature for Criticality
3.4.3 RCS Pressure and Temperature (P/T) Limits
3.4.5 RCS Operational LEAKAGE
3.4.8 RCS Specific Activity
3.4.9 Steam Generator (SG) Tube Integrity
3.5.3 Ultimate Heat Sink
3.7.3 In-Containment Secondary Piping Leakage
3.8.2 Decay Time

Criterion 3

3.1.9 Boron Dilution Control (Chemical and Volume Control System (CVCS)) demineralized water isolation valves)
3.3.1 Module Protection System Instrumentation
3.3.2 Reactor Trip System (RTS) Logic and Actuation
3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation
3.4.4 Reactor Safety Valves (RSVs)
3.4.6 CVCS Isolation Valves
3.4.10 Low Temperature Overpressure Protection (LTOP) Valves
3.5.1 Emergency Core Cooling System (ECCS)
3.5.2 Decay Heat Removal System (DHRS)
3.5.3 Ultimate Heat Sink
3.6.1 Containment
3.6.2 Containment Isolation Valves
3.7.1 Main Steam Isolation Valves (MSIVs)
3.7.2 Feedwater Isolation
3.8.1 Nuclear Instrumentation (refueling neutron flux channels)

Criterion 4

3.3.4 Manual Actuation Functions
3.3.5 Remote Shutdown Station (RSS)

No Criterion

3.1.8 PHYSICS TESTS Exceptions

Overview of DCA Part 4

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4.0 DESIGN FEATURES

4.1 Site Location
- 4.1.1 Site and Exclusion Boundaries
- 4.1.2 Low Population Zone (LPZ)

4.2 Reactor Core
- 4.2.1 Fuel Assemblies
- 4.2.2 Control Rod Assemblies

4.3 Fuel Storage
- 4.3.1 Criticality
- 4.3.2 Drainage
- 4.3.3 Capacity

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility
- 5.1.1 [plant manager]
- 5.1.2 [shift manager]

5.2 Organization
- 5.2.1 Onsite and Offsite Organizations
- 5.2.2 Facility Staff

5.3 Facility Staff Qualifications
- 5.3.1 *facility staff*
- 5.3.2 *licensed SRO and licensed RO*

5.4 Procedures
- 5.4.1 Written procedures shall be …
  - a. Regulatory Guide 1.33
  - b. Emergency Operating Procedures
  - c. QA and Environmental Monitoring
  - d. Fire Protection Program
  - e. Section 5.5 programs
  - **f. availability and reliability controls for SSCs in owner-controlled requirements manual**

5.5 Programs and Manuals
- 5.5.1 Offsite Dose Calculation Manual
- 5.5.2 Radioactive Effluent Control Program
- 5.5.3 Component Cyclic or Transient Limit
- 5.5.4 Steam Generator Program
- 5.5.5 Secondary Water Chemistry Program
- 5.5.6 Explosive Gas and Storage Tank Radioactivity Monitoring Program
- 5.5.7 TS Bases Control Program
- 5.5.8 Safety Function Determination Program
- 5.5.9 Containment Leakage Rate Testing Program
- 5.5.10 Setpoint Program (SP)
- 5.5.11 Surveillance Frequency Control Program (FSAR Table 16.1-1)
- 5.5.12 Spent Fuel Storage Rack Neutron Absorber Monitoring Program

5.6 Reports
- 5.6.1 Annual Radiological Environmental Operating Report
- 5.6.2 Radiological Effluent Release Report
- 5.6.3 Core Operating Limits Report (COLR)
- 5.6.4 RCS Pressure and Temperature Limits Report (PTLR)
- 5.6.5 Steam Generator Tube Inspection Report

5.7 High Radiation Area
- 5.7.1 ≤ 1 rem per hour at 30 cm
- 5.7.2 > 1 rem per hour at 30 cm, but ≤ 500 rads per hour at 100 cm

Technical Topics
Defined Terms

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- LEAKAGE
  - Mark up of STS definition of "Pressure Boundary LEAKAGE":

    "LEAKAGE (except ~~primary to secondary~~ SG LEAKAGE) through a nonisolable fault in a RCS component body, pipe wall, or vessel wall. A fault in an RCS component body, pipe wall, or vessel wall is isolated if LEAKAGE through the isolation device is ≤ 0.5 gpm per nominal inch of valve size up to a maximum limit of 5 gpm."

  - Based on traveler TSTF-534, "Clarify Application of Pressure Boundary Leakage Definition" Revision 0, rejected by the staff in 2012
  - Unclear where in NuScale reactor coolant pressure boundary this provision could be used, much less would be beneficial to unit operation or a safety enhancement
  - **Open Item** under RAI 9031, Question 16.2, Subquestion f (Supplemental response pending)

Technical Topics
Defined Terms (continued)

- OPERABLE–OPERABILITY
  - Mark up of STS definition plus **staff-suggested changes** for consistency with NuScale instrumentation design:

    "A system, subsystem, separation group, **channel, division,** train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, ~~normal or emergency~~ electrical power, cooling ~~and seal~~ water, lubrication, and other auxiliary equipment that are required for the system, subsystem, **separation group, channel, division,** train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s)."

  - **Open Item** under RAI 9031, Question 16.2, Subquestion f (Supplemental response pending)

Technical Topics
Defined Terms (continued)

- RTS RESPONSE TIME and ESF RESPONSE TIME
  - The applicant has proposed to omit these defined terms, and just use “response time” in the corresponding surveillances. Digital response time assumes an allocated value.
  - Bases for each surveillance explains the response time:
    - 3.3.1 MPS instrumentation Functions – channel actuation response time; B 3.3.1:

      “…the time from when the process variable exceeds its setpoint until the output from the channel analog logic reaches the input of the MPS digital logic.”

      “Response time may be verified by any series of sequential, overlapping or total channel measurements, including allocated sensor response time, such that the response time is verified.”

    - 3.3.2 RTS Logic and Actuation – RTS division response time; B 3.3.2:

      “The maximum digital time response is described in the FSAR. This SR encompasses the response time of the RTS division from the output of the equipment interface modules (EIM) until the RTBs are open.”

      “Total response time may be verified by any series of sequential, overlapping, or total channel measurements.”

Technical Topics
Defined Terms (continued)

- RTS RESPONSE TIME and ESF RESPONSE TIME (continued)
  - 3.3.3 ESFAS Logic and Actuation – only addresses response time for opening of pressurizer heater breakers; B 3.3.3:

    "…The maximum digital time response is described in the FSAR. This SR encompasses the **response time** of the ESFAS **from the output of the EIM to the loss of voltage at the output of the pressurizer heater breaker**."

    "Total response time may be verified by any series of sequential, overlapping, or total channel measurements."

    "The response time of valves actuated by the ESFAS are verified in accordance with the IST program" (and associated SRs in Subsections 3.4.6, 3.4.10, 3.5.1, 3.5.2, 3.6.2, 3.7.1, 3.7.2; Bases do not include language about response time, as provided in Section 3.3 Bases – subject of RAI 9642, Question 16-65)

  - Example: SR 3.6.2.3 – verify isolation time of each automatic containment isolation valve in accordance with the INSERVICE TESTING PROGRAM; B 3.6.2:

    "Isolation time is measured from output of the MPS EIM until the valves are isolated."

- **Open Item** under RAI 9614, Question 16-50 (ICE evaluating response)

Technical Topics
RTS and ESFAS Surveillance Requirements

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- MPS digital logic channels, RTS digital actuation logic divisions, and ESFAS digital actuation logic divisions rely on self-testing features to verify channel and division OPERABILITY between CHANNEL CALIBRATIONS and ACTUATION LOGIC TESTS (24 month frequencies)
  - **CHANNEL OPERATIONAL TEST** (92 day frequency) retained only for RCS leak detection monitors implemented in Module Control System (containment evacuation system (CES) gaseous radioactivity monitor channel, and CES condensate channels.)
  - **ACTUATION LOGIC TEST** verifies (B 3.3.2/B 3.3.3) "actuation and priority logic (APL) on all RTS/ESFAS EIMs, the enable nonsafety control switches, the main control room isolation switches, the [ESFAS] override switches, and the operating bypass switches. The ACTUATION LOGIC TEST includes a review of any alarms or failures reported by the self-testing features."
  - **CHANNEL CHECK** definition is revised as shown in markup of STS definition:

    A CHANNEL CHECK shall be the ~~qualitative assessment, by observation, of~~ verification through the absence of alarms from the automatic analog and binary process signal monitoring features used to monitor channel behavior during operation. Deviation beyond the established acceptance criteria is alarmed to allow appropriate action to be taken. This determination shall include, where possible, comparison of ~~the~~ channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter. This determination can be made using computer software or be performed manually.

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Technical Topics
Electrical Power Surveillance Requirements

- No LCOs included for electrical power systems, which are all nonsafety-related
- Module Protection System is protected from faults in the nonsafety-related electrical power system by use of Class 1E isolation devices
- Open on undervoltage or overcurrent
- Details of design of Class 1E isolation devices not included in DCA Part 2, Chapters 7 and 8
- Isolation settings are determined in accordance with the Setpoint Program
- OPERABILITY verified by CHANNEL CALIBRATION surveillance requirements in Subsections 3.3.1, MPS Instrumentation; 3.3.2, RTS Logic and Actuation; 3.3.3, ESFAS Logic and Actuation
  - support OPERABILITY of MPS Instrumentation Functions, RTS Logic and Actuation Functions, and ESFAS Logic and Actuation Functions
  - **Open Item** under RAI 9051, Question 16-28, regarding whether Class 1E isolation device CHANNEL CALIBRATION surveillance is needed for Manual RTS and ESFAS Functions of Subsection 3.3.4

Technical Topics

Allowed Leakage for Main Steam and Feedwater Isolation Valves

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- Verifying secondary system isolation valve leakage is within limits is required by SR 3.7.1.3 (main steam isolation valves (MSIVs)) and SR 3.7.2.3 (feedwater isolation valves (FWIVs), feedwater regulating valves (FWRVs), and feedwater check valves (FWCVs) in accordance with the Inservice Testing Program.
  - Leakage limit ensures decay heat removal system (DHRS) maintains sufficient inventory to remain OPERABLE following events that credit DHRS actuation.
  - Applicant's response to RAI 9614, Question 16-58, says that Surveillance acceptance criteria values for main steam and feedwater isolation valve leakage would be provided by a COL applicant to complete COL Item 3.9-5, by establishing "an Inservice Testing program in accordance with ASME OM Code and 10 CFR 50.55a."
- **Open Item** under RAI 9614, Question 16-58, to track completion of the staff's ongoing evaluation of the response.

Technical Topics
Exceptions to Meeting Surveillances

- While the unit is operating under *an exception*, operational impact may be reduced by allowing repositioning of the valve, breaker, or device *under administrative control*.

RTBs:

Not required to be met for RTBs that are open.

Class 1E Electrical Isolation Devices:

Not required to be met for Class 1E isolation devices that have isolated 1E circuits from non-1E power.

Pressurizer Heater Breakers:

Not required to be met for pressurizer heater breakers that are open or closed under administrative control.

CVCS Isolation Valves

Not required to be met for valves that are closed or open under administrative controls.

Verify each automatic CVCS valve … actuates to the isolation position on an … actuation signal *except for valves that are open under administrative controls.*

Technical Topics

Exceptions to Meeting Surveillances (continued)

Reactor Vent Valves and Recirculation Valves:

Not required to be met for valves that are open.

Containment Isolation Valves,

Verify the isolation time … is within limits *except for valves that are open under administrative controls*.

Verify each automatic containment isolation valve … actuates to the isolation position on an … actuation signal *except for valves that are open under administrative controls*.

Revision 2 of DCA Part 4 does not include such exceptions for

- DHRS Actuation Valves
- MSIVs, MSIV Bypass Valves
- FWIVs, FWRVs

- **Open Item** under RAI 9051, Question 16-28, because there are no apparent scenarios in which administrative control of an automatic valve, breaker, or device provides a safety benefit over entering the LCO action requirements. SR 3.0.1 already states that surveillances do not have to be performed on inoperable equipment.

Technical Topics
Open Item Listing (1 of 6)

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RAI #	Action	Issue
9033 16-2.f	NuScale	Definition of RCS Pressure Boundary LEAKAGE
9051 16-27	NuScale	Additional information about locations in FSAR that describe contents of Owner Controlled Requirements Manual (Note that creation of this document is the subject of COL Item 16.1-2)
9051 16-28	NuScale	Append the phrase “in accordance with the Setpoint Program” to Channel Calibration Surveillance statement
9051 16-28	NuScale	Include a Channel Calibration Surveillance for Class 1E electrical isolation devices in LCO 3.3.4, Manual RTS and ESF Functions
9051 16-28 9642 16-64	NuScale	Exceptions to meeting Surveillances and use of administrative controls while using an exception
9034 16-30.d	NuScale	Update FSAR Table 16.1-1, “Surveillance Frequency Control Program Base Frequencies”

Technical Topics
Open Item Listing (2 of 6)

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RAI #	Action	Issue
9614 16-50 9642 16-65	Staff NuScale	Omission of the Response Time definitions and defined terms, and the adequacy of the proposed response time verification Surveillances
9614 16-54	NuScale	(editorial) SR 3.7.2.1 should not contain "required" because both FWIVs are required OPERABLE by LCO 3.7.2, and the SR only applies to valves with pressurized gas accumulators
9614 16-54	NuScale	(editorial) Phrasing of surveillance statements to verify accumulator pressure for valves with the NuScale-specific actuator design
9614 16-58	Staff	Defer providing values for INSERVICE TESTING PROGRAM Surveillance acceptance criteria for main steam and feedwater isolation valve leakage limits

Technical Topics
Open Item Listing (3 of 6)

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RAI #	Action	Issue
9614 16-59	NuScale	Applicability of LCO 3.3.1, MPS Function 22b, DHRS actuation on High Narrow Range Containment Pressure in MODE 3
9634 16-60	NuScale	Editorial comments on SR 3.1.7.1, SR 3.1.9.1, SR 3.2.1.1, SR 3.3.2.3, SR 3.3.3.3, SR 3.4.10.4, and SR 3.5.1.4.
9634 16-60.2	NuScale	Clarify whether 'Low Pressurizer Level' or 'Low Low Pressurizer Level' initiates 'secondary system isolation (SSI)' actuation.
9614 16-52 9634 16-60.34 9634 16-60.63	NuScale & Staff	(editorial) Conformance to Improved TS writer's guide section 4.1.3.b. Remove unnecessary instances of the words "inoperable" and "required" in Condition, Required Action, and Surveillance statements; and also Action Notes on separate Condition entry.

Technical Topics
Open Item Listing (4 of 6)

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RAI #	Action	Issue
9634 16-60.37.1	NuScale	LCO 3.1.9, "Boron Dilution Control," Action B, requires isolating the dilution source, which completes the safety function of the DWSI makeup isolation valves. It appears that the Applicability of LCO 3.1.9 **for these valves** would be more accurate by stating: "MODES 1, 2, and 3 with any CVCS demineralized water isolation valve open."
9634 16-60.37.3	NuScale	Correct reference to interlock (either N-2H or N-2L) in LCO 3.3.1, Required Action E.1 ("Reduce THERMAL POWER to below the N-2L interlock. | 6 hours"). The affected MPS Functions (2a, 18a, 18b, 18c) are applicable in MODE 1 above the N-2H interlock (15% RTP)
9634 16-60.70	NuScale	Clarify rationale for Applicability not including MODE 3 for MPS Low Low Pressurizer Pressure Functions 9.b, DHRS, 9.c, CVCSI, and 9.d, PHT, of LCO 3.3.1, between 350°F and 420°F.

Technical Topics
Open Item Listing (5 of 6)

RAI #	Action	Issue
9642 16-61	NuScale	What are the appropriate means of addressing prevention and mitigation of the postulated inadvertent actuation of the containment flood and drain system (CFDS) to flood the containment vessel with RCS temperature above the RCS temperature limit in the PTLR? Why not use safety-grade interlock on CFDS CIVs to protect RPV from higher than analyzed thermal stress?
9642 16-62	NuScale	Clarify in the Subsection 3.3.1 Bases the relationship of the MPS Instrumentation Functions, and their bypassing or enabling interlocks and permissives, to the SP controls and Channel Calibration Surveillances.
9642 16-63	NuScale	(editorial) Clarify Bases for LCO 3.0.4 and SR 3.0.4. Should initiating and terminating PASSIVELY COOLED be treated as a MODE transition within MODE 3?
9642 16-66	NuScale	Request for adding two additional references to the Specification 5.6.3 COLR methodology list, LOCA analyses for LCO 3.1.6, LCO 3.2.2, and LCO 3.4.1.

Technical Topics
Open Item Listing (6 of 6)

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RAI #	Action	Issue
9201 5.2.5-7	Staff	Acceptability of 1.5 gallons per hour leak-before-break (LBB) limit on in-containment secondary system piping leakage, specified by LCO 3.7.3. The staff is reviewing March 5, 2019, supplemental response and may request additional edits to Subsection B 3.7.3.
9201 5.2.5-7	NuScale	Revised LBB analysis-related additional information in preparation.
None	NuScale	Completion of COL sub-item listing under COL Item 16.1-1

Technical Topics
Pending MPS design change

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- 13 DHRS-initiating MPS instrumentation Functions to support new ESFAS Function of "Secondary System Isolation (SSI)"
  - SSI closes isolation valves in main steam lines and feedwater lines without an unnecessary cooldown from DHRS actuation
  - Associated SSI operating bypass based on FWIV position indication, which is upgraded to safety-related to support bypass logic
  - Just 4 of these 13 MPS instrumentation Functions will initiate DHRS:
    - High Pressurizer Pressure (7.b)
    - High Steam Pressure (17.b)
    - High Narrow Range RCS Hot Temperature (13.b)
    - Low AC Voltage to ELVS Battery Chargers (25.b)
- Remove uncredited ECCS actuation on RPV riser level;
  - The safety analysis credited ECCS actuation on High Containment Water Level (23.a) is adequate to initiate ECCS by opening RVVs and RRVs

Technical Topics
General Issues

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- Combined License (COL) sub-item determination –
  Open Item
- Disposition of NRC-approved technical specifications task force (TSTF) traveler changes – adapted or not adapted
  - which have been incorporated in NUREG-1432 (digital), Revision 4; or
  - approved since issuance of NUREG-1432 (digital), Revision 4.
  - Open Items under RAI Questions 16-2f and 16-28
- Administrative Changes –
  - Open Items as listed above
  - Correction of grammatical and typographical errors
  - Replacement of inapplicable content taken from STS Bases
  - Addition of missing content to the Bases
  - Clarification of submitted content in the Bases
  - Conformance to STS style, punctuation, phrasing, formatting conventions
  - Resolution of inconsistencies, both within DCA Part 4 and DCA Part 2

Chapter 16 Review Status
SUMMARY

- The NuScale Generic Technical Specifications are based upon Standard Technical Specifications; differences are a result of design differences with plant designs considered in the STS
- A thorough review of the NuScale GTS has been conducted resulting in a safety evaluation chapter that includes open items in the following areas:
  - New and Revised Defined Terms and revised definitions
  - MPS instrumentation Function Applicability inconsistencies
  - I&C surveillance requirements & testing
  - Response time testing
  - Application of LCO selection criteria, TSTF disposition & COL action items
  - Administrative and editorial issues
- Resolution of the Open Items will be accomplished with the assistance of the technical branches (i.e., ICE, SRSB, SPRA, SCVB, and SPSB)