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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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NUSCALE DESIGN-CENTERED SUBCOMMITTEE

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TUESDAY, OCTOBER 1, 2024

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The Subcommittee met via Teleconference, at 1:00 p.m. EDT, Vesna B. Dimitrijevic, Chair, presiding.

COMMITTEE MEMBERS:

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ACRS CONSULTANT:

DENNIS BLEY

DESIGNATED FEDERAL OFFICIAL:

MICHAEL SNODDERLY

C-O-N-T-E-N-T-S

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

1:01 p.m.

CHAIR DIMITRIJEVIC: Okay. So the meeting will now come to order. This is a meeting of the NuScale Design-Centered Review Subcommittee of the Advisory Committee on Reactor Safeguards. I'm Vesna Dimitrijevic, chair of today's subcommittee meeting.

ACRS members in attendance in person are Walt Kirchner, Ron Ballinger, Vicki Bier, Gregory Halnon, Craig Harrington, Robert Martin, Scott Palmtag, Dave Petti, and Thomas Roberts. ACRS members in attendance virtually via Teams are Matt Sunseri and myself. I did not see Matt yet on the line, but I think he should be here with us any moment.

So we have also one of our consultants participating virtually via Teams Dennis Bley. Dennis, I see on the line. If I have missed anybody, the ACRS member consultants, please speak up now. But I think I listed everybody, so I couldn't have missed anybody.

So Mike Snodderly of ACRS staff is the designated federal officer for this meeting. No member conflict of interest were identified for today's meeting. And we have a quorum for today's meeting.

During today's meeting, the subcommittee will receiving a briefing of the staff advanced safety relation of the NuScale topical report, NuScale power plant design capability to mitigate beyond design basis events, DBE, defined by 10 CFR 51.55. The topical report describes design capabilities to mitigate beyond design basis events. Design features that provide enhanced capability for coping with an extended loss of electrical power, loss of normal access to the normal heat sink, and loss of launch area due to explosions or fire are discussed within this topical report.

10 CFR 51.55, mitigation of beyond design basis events, was put in effect in response to lessons learned from the Fukushima Daiichi accident. The ACRS was established by statute and is governed by the Federal Advisory Committee Act or FACA. The NRC implements FACA in accordance with our regulation.

But those regulations are the committee bylaws. The ACRS speaks only through each published letter reports. All members' comments should be regarded as only the individual opinion of that member and not as a committee position.

All of the information related to ARCS activities such as letters, rules, committee participation, and transcripts are located on the NRC public website and can be easily found by tying About Us ACRS in the search field of the NRC home page. The ACRS consisted with agency value of public transparency in regulation of nuclear facilities provides an opportunity for public input and comment during our proceedings. We have received no written statements or requests to make an oral statement from the public.

We have also set aside time at the end of the meeting for public comments. Portions of these meetings may be closed to protect sensitive information as required by FACA and the government in the Sunshine Act. Attendance during the closed portion of the meeting will be limited to NRC staff and its consultant, applicants and those individuals in the organization who have entered into an appropriate confidentiality agreement.

We will confirm that only eligible individuals in the closed portion of the meeting. The ACRS will gather information, analyze relevant issues and facts, and formulate proposed conclusions and recommendation as appropriate for deliberation by the full committee. A transcript of the meeting is being kept and will be posted on our website.

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

For everyone in the room, please put -- sorry, I have this distraction here -- please put all your electronic devices in silent mode and mute your laptop microphone and speakers. In addition, please keep sidebar discussion in the room to a minimum since the ceiling microphones are live. For the presenters, your table microphones are unidirectional and you will need to speak into the front of the microphone to be heard.

Finally, if you have any feedback for the ACRS about today's meeting, we encourage you to fill out the public meeting feedback form on the NRC website. We will now proceed with the meeting. So who will be the first presenter today from the NuScale?

MR. CUMMINGS: Hey, this is Kris Cummings from NuScale. Pete will be our presenter. But before that, I'm going to ask Jim Osborn to make opening statements.

CHAIR DIMITRIJEVIC: Okay. Thank you. Please.

MR. OSBORN: Yes, thank you, Vesna and Kris. Good morning, everyone. My name is Jim Osborn, Licensing Supervisor for NuScale Power. We thank everyone for this opportunity to present the mitigation of beyond design basis event topical report.

For a little background, NuScale has developed this topical report to generically address the requirements of 10 CFR 50.155 for a NuScale design plant. It is not specific to either the U.S. 600 or the U.S. 460. This topical report was developed from two technical reports that were found acceptable in the U.S. 600 DCA approved application.

However, the DCA review did not assess the capacity and capability of the U.S. 600 plant beyond 72 hours. This topical report provides a framework for a future licensee to demonstrate that their NuScale plant has the design features and capability to satisfy the requirements of 10 CFR 50.155 for a significantly extended period beyond 72 hours. So at this point, I would like to turn the floor over to my colleague, Pete Shaw, who will be the main presenter.

MR. SHAW: Good afternoon. My name is Pete Shaw. Thank you for the introduction, Jim. I have 15 years of experience in the nuclear industry, 10 of which were at the Vogtle 3 and 4 construction site.

I will be the -- I'm the lead presenter on mitigation on beyond design basis events topical report. Next slide, please. Once again, we would like to thank DOE for their award for NuScale and their contributions to the project. Next slide, please. So this is a brief agenda for everything that will be discussing today.

It will include an introduction and basis for the topical report. We will have a brief overview of the plant features for individuals to provide familiarity with the design. We'll then discuss the event mitigation features for design basis events which then perform for the accrediting of safety features for mitigation of beyond design basis events.

We'll then discuss the conditions of applicability and a summary of the plant systems relied upon for loss of AC power, a summary of how regulatory commitments are fulfilled, and then conditions of applicability that were brought by the NRC. And then last, we will have our statements. Next slide, please. So as Jim said, the basis for this particular topical report, we were looking at a consistent approach for mitigation of beyond design basis events.

And the importance behind this is that it would be applicable to any applicant looking to adopt a NuScale technology. This includes configurations and that implement the design features that are described. They are described at a level that is consistent for all NuScale power plants.

The safety features extend the baseline coping period past traditional reactor technologies. And the topical report discusses how these are relied upon for beyond design basis events. This use informed approached used by NEI guidance and the lessons learned from the Fukushima Daiichi accident, these are developed into the NuScale design and are a part of the basis that we will be discussing today.

An applicant will then be required to discuss and describe on how they fulfill these conditions for use in their applications. And as a note, the duration of the extending coping period is proprietary NuScale information. If there are questions about the duration beyond the 72 hours, we can answer those in the closed session. Next slide, please.

So familiarization with the design and what the basis is. There is a NuScale reactor. It's a light water reactor with enhanced passive safety features. And it implements in all of its safety functions passive reliability in a sequence that will be discussed later in more detail.

At the initiation of a design basis event, decay heat removal is actuated and it will be removed by the steam generators and the DHRS for three days. And after the three days, the decay heat will continue to be removed by the containment into the ultimate heat sink. This transition period is our topic of discussion today.

Obviously, once this is completed, the modules will then transfer into long term air cooling. But again, that's not subject to it. These water levers on this particular graphic are not indicative of final plant design. They're here for illustrative purposes. Next slide, please.

So this is an overview of an example, 6 pack power plant for the purposes of today's presentation. What's most important to focus on in this is the control building and reactor building are separate structures. Next slide, please.

MEMBER HALNON: Pete, before you go on, this is Greg. Just wanted to get your definition of indefinite on the previous slide. Is that assuming nothing -- no operator reaction to refill or in any way replenish the DHRS?

MR. SHAW: So yes, but that's also not covered in this particular topical report. So that's a topic for another discussion in the design of the NuScale power plants. We're focusing just on the boiling and the coping period as discussed in the topical.

MEMBER HALNON: Okay. I didn't put indefinite on the slide. You did. Can you explain indefinite to me?

MR. SHAW: I'd like for Meghan McCloskey who's on the phone to potentially answer that question.

MS. MCCLOSKEY: This is Meghan McCloskey, NuScale. One key clarification here is that the indefinite cooling is really focused on the module conditions that are -- the modules are isolated. The containments are closed. And so water inside the modules continues to cool the emergency core cooling system. Eventually, there could potentially come a point where additional liquid is needed in the pool for spent fuel pool cooling. Does that clarify your question?

MEMBER HALNON: I think so. I don't want to get into a lot of detail. But I was just trying to figure out at what point something had to get done. You see Dr. Martin is going to jump in there and bail me out or something.

MEMBER MARTIN: I don't know about bailing you out. But I had a question for Meghan. So just from what you described, we have oil off. We have closed containment. We do not expect some condensation and maybe return of that coolant back into the pool?

MS. MCCLOSKEY: The condensation and return of coolant potentially occurs. It's not a design feature of the reactor building. And it's not credited in any of the work that we've done to evaluate these types of extended cooling capabilities. And the focus of this topical report really was on the criteria and the design features of the NuScale plant to cope with these beyond design basis conditions. NuScale was not looking for approval of specific methods and requirements for the thermal hydraulic analysis specifically.

MEMBER MARTIN: So potentially conservative to kind of set design criteria.

MS. MCCLOSKEY: I didn't hear that clearly.

MEMBER MARTIN: I was trying to summarize and paraphrase what you're saying. Potentially conservative basically to establish a method of -- criteria.

MS. MCCLOSKEY: Yes.

MR. SHAW: Thank you, Meghan. Next slide, please. For familiarization, this is a reactor building example. This is typical features of a 6 pack power plant.

The specific features are discussed in the topic report are here. You have an overhead heavy load handling system, single failure proof in the event of a seismic event during a module move. There are reactor building walls that protect those modules and the ultimate heat sink.

The bioshields also provide an additional potential layer of protection in a beyond design basis event. The spent fuel pool and the ultimate heat sink are communicated together as the same unit water until the evaporation below the rear wall. And I'll note that the ultimately heat sink is also below grade in the event of a beyond design basis event.

That is protected from external events by the reactor building. Next slide, please. And then this is a graphical representation of a typical NuScale power module. As you can see, it has its own containment vessel.

It has emergency core cooling values for both the vent at the top of the reactor vessel and towards the bottom for recirculation and then the decay hey removal system. Next slide, please. So this is a brief summary of a high level of the mitigation features for design basis events. When an event occurs, safe and stable shutdown is achieved without any operator actions.

These include an automatic reactor trip, containment isolation that activates immediately, a decay heat removal system that passively removes decay heat for 24 hours. If AC power is not restored within those 24 hours, the emergency core cooling valves activate. And then the modules rely on no additional equipment to maintain core cooling.

The spent fuel pool shares a common water source with the reactor modules, thus only a single volume needs to be maintained for that coping period. We do not credit this for safe-shutdown. But for 72 hours, the system response is assured in the main control room.

And then last, all of these design-basis mitigation features are then relied upon for beyond design basis events to provide the capability for successful mitigation past large light water reactor design coping periods. Next slide, please. And so as an extension into the mitigation of beyond design basis events, these redundant reliable safety features then are relied upon to mitigate or damage. The immediate plant shutdown and containment isolation at the initiation event prevents release.

The containment vessels are protected by the reactor building from external events. And because no operator actions are needed to initiate safe shutdown, further actions aren't needed during the extended coping period. To reflect this, the NuScale implements a set of conditions for use an adopter of this topical report.

Firstly, they must have described all of the design features that are in scope of the topical report. They must then provide a thermal analysis to validate a proposed coping period. They must establish a maintenance fuel program in accordance with 50.65 to ensure reliability of the features that are relied upon. And then last, they must have an emergency plan in their application. Next slide, please.

MEMBER KIRCHNER: Peter, just to clarify, when you said an emergency plan, is that upper case emergency plan?

MR. SHAW: Yes.

MEMBER KIRCHNER: That's site specific, not an emergency plan for coping?

MR. SHAW: Exactly.

MEMBER KIRCHNER: Thank you.

MR. SHAW: Next slide, please. And so this is a brief summary of the plant systems that are relied upon for a loss of AC power event. The reactor building protects modules from design basis and natural phenomena. And it houses the ultimately heat sink.

The control building is relied upon to protect operators from design basis natural phenomena. And augmented direct power system provides continuous DC power for the equipment. The module protection system automatically actuates safe-shutdown functions and don't require new operator actions.

The plant protection system monitors reactor, refueling, and spent fuel conditions, and they're common to all modules. And the safety display and indications system provides non-safety-related accident monitoring.

Next slide, please. So, further, the containment isolation is required so that the passive systems can reliably initiate safe-shutdown and prevent an uncontrolled release and also dissipate decay heath.

The ultimate heat sink provides common water source for maintaining safe-shutdown. And it also has a Category 1 assured water make-up line. A decay heat removal system removes decay heat from the reactor into the ultimate heat sink. Emergency core cooling circulates reactor coolant into containment to initiate long term passive cooling.

A reactor building and control room building ventilation system maintain safe atmospheres for respective environments. An overhead heavy load handling system supports the module during lifts during design-basis natural phenomena. And then last, a communication system used to coordinate operator response and respond to a beyond design basis event. Every NuScale reactor configuration power plant would have to have these features in it to adopt the topical report.

CHAIR DIMITRIJEVIC: Hi, this is Vesna. So let me just ask you -- let me just summarize where we are now. So you're basically -- the premise is you can deal with just install existing equipment. You don't need any extra equipment to deal with those events, right?

And you can rely just on the plant equipment. And then you describe all of this plant systems which you rely on. So let me ask you a couple of the things.

Like, for example, the containment isolation, what does it -- in your TR, it says the containment isolation occur on the loss of AC power instantly. So is that something -- there is no signal needed for containment isolation. It's something which comes automatically with loss of AC power?

MR. SHAW: So that is given by the -- so described earlier in the previous slide, the module protection system automatically actuates the containment isolation for the reactor.

CHAIR DIMITRIJEVIC: For every trip or just loss of AC power.

MR. SHAW: I would actually like

clarification on that from Meghan McCloskey.

MS. MCCLOSKEY: This is Meghan McCloskey, NuScale. During a loss of AC power event, the module protection system will actuate containment isolation and then -- and reactor trip and DHRS actuation. Containment isolation could also be actuated on a number of other signals such as high containment pressure or I think --

(Simultaneous speaking.)

CHAIR DIMITRIJEVIC: The normal signals, right.

MS. MCCLOSKEY: The normal signals that you're familiar with.

CHAIR DIMITRIJEVIC: Yeah, okay. So the loss of AC power will automatically actuate. Let me ask you then also you -- I saw that on your augmented, the DC system, the two channels provide the -- remove power from emergency cooling system valves and open them about one hour. Does that mean it's assumed that decay heat removal will get pressure down enough to enable opening the ESSC valves in one hour?

MS. MCCLOSKEY: I'm not sure where the one hour comes from. If AC power supplied to the chargers is not restored, then the module protection system will actuate ECCS after 24 hours.

CHAIR DIMITRIJEVIC: Sorry, that's what I meant 24 hours. And if I said the one hour, that was a mistake. So you have those two channels, you know EDAS, IMS, right, which provide ECCS hold mode. And after 24 hours, they will sort of, like, de-energize those valves, right? Well, my question is would the pressure differential be low enough to enable opening that valve in 24 hours?

MS. MCCLOSKEY: Yes, 24 hours of normal DHRS cooling is very effective to reduce the module pressure. And so for designs that have inadvertent actuation blocks on all of the valves, we would expect the ECCS values to open at that time. And then the design -- the NuScale design that was submitted with the SDAA does not have inadvertent actuation blocks on the vent valves. And so the vent valves would open when actuated and continue depressurizing the reactor coolant system.

CHAIR DIMITRIJEVIC: Okay, thanks. So let me then ask you following which is connected with this slide. And that's why I'm asking this question here. You said you don't discuss this extended coping time. But we can discuss 72 hours, right, in the open session? Okay.

So what's happening, the 72 hours is

basically the batteries run out, right? We lost the ventilation system and lights in the control room, right? After 72 hours, it will be like a total blackout, right? But we don't really need operator in instrumentation that's a main premise, right?

(Simultaneous speaking.)

CHAIR DIMITRIJEVIC: I mean, after 72 hours.

MS. MCCLOSKEY: I think Dan Lassiter here can provide some additional information from the system design perspective.

MR. LASSITER: This is Dan Lassiter, NuScale design engineering. You're asking about the -- with regard to the time for ECCS operations?

MS. MCCLOSKEY: No, the --

(Simultaneous speaking.)

CHAIR DIMITRIJEVIC: No, no. I'm asking about the --

(Simultaneous speaking.)

CHAIR DIMITRIJEVIC: -- instrumentation and control room and things like that because I see here you have it here that somebody apply a system rely upon during loss of AC power. Some of those systems, you rely on after in the 72 hours, not for that extended time, right?

MR. LASSITER: Yeah, there's no requirement for operator action within the 72 hour design basis event is the bottom line.

CHAIR DIMITRIJEVIC: So let me just put my question a little more clear. We have two times which we're going to discuss. One is 72 hours which is what has been done before, right?

And now you have introduced another specified extended time which you said we can only discuss in the closed section, right? So I'm only talking now about 72 hours. After 72 hours when we're going to enter this other specified time to be discussing in the closed session, there would not be the control room, the ventilation systems, the PAM.

And everything will be gone because the augmented DC power system will be gone in 72 hours. Is that a true statement? After 72 hours, we will not have any instrumentation and control room will not be able to be habitable, right?

MS. MCCLOSKEY: Just a second.

CHAIR DIMITRIJEVIC: Sure.

MR. LASSITER: It is correct that the control room habitability system and the PAM battery provisions -- excuse me, post-accident monitoring battery provisions for the instrumentation is designed for 72 hours. After 72 hours, we expect there to be time for operators to take appropriate actions based on the accident.

MS. MCCLOSKEY: And this is Meghan again. The plant is in a safe -- the modules are in a safe, stable condition throughout that 72-hour duration and the spent fuel is being cooled. And so the plant personnel have that time frame to respond to the event as appropriate for whatever is going on.

CHAIR DIMITRIJEVIC: I see. Okay. Well, then I sort of -- that was my impression. You're qualifying all of this for 72 hours, not for that other extended specified time which we cannot discuss now. So all right. Well, okay, all right. I understand this now. So let me also ask you why do you have to rely on this load handling system for the loss of AC power?

MS. MCCLOSKEY: I think Erwin may have something.

MR. LAUREANO: Yeah, this is Erwin Laureano, Plant Office Manager for NuScale. The overhead heavy load handling system, that's just being mentioned in the event that a NuScale power module is in transition to the containment plant tool or it's being held in the containment plant tool which is seismically qualified.

MR. LASSITER: Yeah, it's not an active safety system. It's just designed to keep the module in a save condition in the event of loss of power in that scenario.

CHAIR DIMITRIJEVIC: Okay. All right. Thanks.

MR. LAUREANO: You're welcome.

MR. SHAW: Next slide, please. And so following from the loss of AC power mitigations, the filament of the other regulations actually follows a very similar mitigation. In this case, the station blackout has the same mitigations of a safe stable shutdown as said within that 72 hours, no operator reactions. Because the spent fuel pool shares a common water source, only a single volume needs to be maintained for that extended coping period.

For 10 CFR 50.155(b)(2) and (c), exterior concrete walls for the reactor building are designed to withstand aircraft impacts in similar events for loss of a large area. Next, the training requirements for these components are not needed because they are in the described systems that they operators will already be trained and qualified for. And then last for spent fuel pool monitoring until the final fuel removal from the reactor vessel, 51.55(e), a spent fuel pool monitoring is common to the ultimate heat sink and the instrumentation needs the guidance of NEI 1202. Next slide, please.

So the review with the staff, there were 24 audit questions that were asked, one docketed response. And there were no RAIs for the review. During their review, the NRC identified several site specific considerations based on plant location. Given that our extended coping period is so long, additional considerations were needed. For these limitations and conditions, most of these would normally be a part of a site selection and they would be --

An adopter would also supply why no preplanning is needed using this methodology 451.55. And if not, an adopter must identify water for an ultimate heat sink, necessary mode of equipment such as pumps and generators, equipment for debris removal, address the statement of operators past the 72 hours. Address all personnel, will ascertain plant conditions in a post-event, and address all spent fuel level instrumentation. Power sources will be replaced post-event.

And applicant must also provide a fire

protection in accordance with 50.48 and a training program. That includes this relied upon for beyond design basis event. Next slide, please. And so as summary, NuScale power plant design provides extended coping capabilities for mitigating beyond design basis events. Operator actions are not necessary to establish the safe shutdown condition. And the topical report establishes a consistent approach for remedy at a NuScale power plant to demonstrate compliance to 10 CFR 50.155.

CHAIR DIMITRIJEVIC: Okay. Thank you. And who will be presenting from NRC?

MR. HAYDEN: That's me, Tommy Hayden.

CHAIR DIMITRIJEVIC: Okay.

MR. SNODDERLY: We need maybe just a few minute break or we need to move out the NuScale folks and then move in staff able to come and put their slides up. So give us one minute and we'll let you know when we've made that --

(Simultaneous speaking.)

CHAIR DIMITRIJEVIC: Okay, okay.

MR. SNODDERLY: Thank you.

(Pause.)

MEMBER KIRCHNER: Are you ready, Tom?

MR. HAYDEN: Yes, I'm ready.

MEMBER KIRCHNER: Vesna, so I think the NRR staff is ready. So go ahead.

CHAIR DIMITRIJEVIC: Okay. Go ahead, please.

MR. HAYDEN: Yeah, sure thing. This is Tommy Hayden from the NRR DNRL NRLB branch, new reactor licensing branch. I'll be presenting the staff's evaluation of the topical report.

So for a number of you, NuScale submitted the topical report for the mitigation of beyond design basis events, Rev. 0 on September 11, 2023 and Rev. 1 on June 26, 2024. As NuScale mentioned, the NRC regulatory audit was performed on September 2023 to March 2024. And 24 issues were generated. Eleven of those were resolved via audit responses and 13 were resolved via limitations and conditions.

NuScale submitted one piece of supplemental information to the docket and no RAIs were issued. Staff completed the topical report review and issued an advanced safety evaluation to support today's ACRS subcommittee meeting. Here's the technical reviewers for the topical report: Hanry Wagage, Raul Hernandez, Angelo Stubbs, Josh Miller, Ryan Nolan, John Hughey, Thinh Dinh, Marie Pohida, Sheila Ray, Nick Hansing, and myself and Getachew

Tesfaye with our project managers for this topical report.

DR. BLEY: Hey, this is Dennis Bley. Can you go back two slides from there? One more, yeah. Is it pretty unusual, you have no RAIs? Were you able to -- I guess the audits gave you the flexibility to resolve issues that were concerning you. Is that right?

MR. HAYDEN: That's correct.

DR. BLEY: Okay. Go ahead.

MR. HAYDEN: Okay. Here's a layout of the safety evaluation for this topical report. You'll see the bolded sections are the sections that we've created slides for that we'll go into a little more depth on. So Section 2 is the background and then Section 3 is technical evaluation and each of the sections within that evaluation.

Section 4 is the conclusion. You'll notice Section 5, limitations and conditions, is not -- we don't have a slide made out for that explicitly. But inside each of the technical evaluation slides, the applicable L&C is discussed there.

Section 2 background, so 2.1 has regulatory requirements and regulatory and industry guidance. And then Section 2.2 is the summary of technical information. So the TR is applicable only to NuScale's small modular reactor designs with specific characteristics.

As mentioned in NuScale's presentation, there's also conditions of use that were laid out in the topical report. And the TR contains a description of the NuScale design capabilities and features, including passive safety systems capable of maintaining core cooling containment and cooling functions. And the large reactor pool serving as the ultimate heat sink.

The topical report specifies how these features enable a design to mitigate beyond design basis events for a specified extended duration without the need for AC power, special equipment, or additional guidelines and strategies. 3.1 is the plant baseline coping criteria for loss of all AC power. 3.1.1 is the assessment of the electrical power.

In the first 72 hours of a loss of all AC power is identical to a station blackout and no AC power is relied upon for performing safety functions. The initial conditions and assumptions in NE 12-06, Rev. 4 and Reg Guide 1.226, Rev. 0, assume that station batteries would remain available following a beyond design basis event since they are considered robust. In the TR, Section 3.1.5, Initial Event Conditions and Assumptions, NuScale states that for the baseline coping capability, station batteries and associated DC buses remain available for the design operating of the station batteries, and installed electrical distribution system, including inverters and battery chargers, remain available provided they are seismic Cat. 1.

Initial assumptions in the TR following a beyond design basis external event are consistent with NE 12-06, Rev. 4 guidance, endorsed by Reg Guide 1.226. 3.1.1 continued, the ultimate heat sink monitoring, including spent fuel pool level, is assured for a time frame in excess of the 7 days post-event minimum in NEI 12-02. The staff notes that the ultimate heat sink and spent fuel pool level instruments can be powered by multiple means.

The capacity and capability of such multiple means are not discussed. And therefore, an applicant must satisfy L&C 5.2. L&C 5.2 addresses how plan operators will ensure, during the initial coping phase, that the following can be achieved to provide makeup to the ultimate heat sink.

The source of water is identified and

available in sufficient quantity. Necessary motive equipment such as pumps and generators and the required electrical power and fuel, can be obtained, staged, and implemented. And required debris removal will be accomplished to support placement of equipment and access to site connections.

3.1.2 is plant design capabilities. Following a loss of all AC power event, automatic responses of safety-related equipment establish and maintain core cooling, containment, and spent fuel pool cooling by placing the reactor modules and spent fuel into a safe, stable, shutdown state with passive cooling. 3.1.2 also introduces L&C No. 5.1.

To accomplish this, an adopter must satisfy Conditions of Use in TR Section 1.3, by providing a plant specific design, as described within the report, plant specific thermal analysis, including configuration of the plant, number of modules, spent fuel pool capacity for all modes of operation, a maintenance rule program and an emergency plan. And I bolded the second Condition of Use there as that is the portion of L&C 5.1 that the safety evaluation in Section 3.1 captures. 3.2, plant systems and responses to a loss of all AC power event.

10 CFR 50.155 requires that equipment have

the capacity and protection to be used to support mitigation of beyond design basis events. TR Section 4.0 lists proposed equipment for a NuScale SMR design as NuScale just showed. The TR also lists the protection of this equipment.

The design specific and location specific nature of the MBDBE necessitates analysis from a COL applicant. The analysis would need to confirm the assumptions in the TR remain true for the proposed plant as well as contain analysis to show that the proposed plant systems would be able to respond to the progression of the event for the proposed period of time. Again, L&C 5.1 is referenced here, and this was added to address the need for the connection of the analysis and the plant system responses for the event.

3.3 is safety functions during a loss of all AC power. 10 CFR 50.155 states that applicants shall develop, implement, and maintain strategies and guidelines to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities. The TR discusses the strategies for maintaining these functions during the event and how it is accomplished with a NuScale SMR.

The site-specific nature of hazards and potential site layouts and locations requires that a

COL applicant address the potential events at their specified site. L&C 5.1 addresses the need for an applicant to provide analysis and potential guidelines and strategies dependent on that analysis. 3.3.2, indefinite maintenance of core cooling, containment, and spent fuel pool cooling capabilities.

The regulation requires that offsite assistance and resources be acquired indefinitely or until functional capabilities can be maintained without the need for mitigating strategies. The endorsed guidance in NEI 12-06 recognizes that FLEX strategies and offsite resources do not need to be explicitly planned for the period beyond 72 hours. The TR proposes that no pre-planning for mitigating actions is required because a NuScale SMR design provides beyond design basis external event mitigating capability beyond 72 hours.

3.3.2, continued. However, the guidance in NEI 12-06 also presumes that initial coping mitigating strategies have been established such that staging areas to receive offsite equipment are identified; means are established to transport equipment to deployment areas; the ability of an offsite organization to provide support has been ensured; and standard FLEX equipment such as mechanical and electrical connectors that are compatible with site connections are obtained. L&Cs 5.2, 5.3, 5.4, and 5.5 address the requirement for long term mitigating capabilities associated with the NuScale SMR design discussed in the TR.

MEMBER HALNON: Tommy, this is Greg. It appears that -- trying to reconcile their statement that no pre-planning is required, yet in their TR, they also say that pre-planning is required. Is that how you all took it? I mean, staging areas, means are established, stability offsite, that's all pre-planning stuff. When they say no pre-planning is required, what do you they mean? How did you take it?

MR. HAYDEN: Josh or John, if you have that -- if you're able to answer that. There's a mic right there.

MR. HUGHEY: John Hughey, PRA division, Oversight Branch. So my understanding of the TR is that the assertion was no pre-planning was required. And the basis for that was their design was able to cope with the -- cope without any planning for at least the 72 hour period.

So the guidance says after 72 hours we acknowledge that you're going to be able to continue to indefinitely get resources. You don't have to explicitly plan that. However, with the NuScale design, there was no pre-planning considered necessary at all.

Therefore, those characteristics that would have been in place with the initial mitigating strategies will not be in place according to the topical report. So the TR was essentially saying no pre-planning was required at all in RSE where we are essentially saying some pre-planning is necessary unless for some reason an applicant is at a site and for one or more or all of the elements, they can make a justifying argument for why they in their particular situation don't need pre-planning.

MEMBER HALNON: Okay. Thank you. That explains it.

DR. BLEY: This is Dennis Bley. In past discussions, and I think there was no commitments that I remember about FLEX safer and info. And then the statements certified by FLEX essentially a requirement if you're going to (audio interference).

CHAIR DIMITRIJEVIC: Dennis, you are breaking up. I cannot hear you.

MEMBER HALNON: Dennis, it sounds like your connection is poor. We can't catch any of the words.

(Audio interference).

DR. BLEY: They're working on it.

MEMBER HALNON: We got that they're working on it.

CHAIR DIMITRIJEVIC: Yes, Dennis, it's tough to understand what you're saying, so --

DR. BLEY: (Audio interference.)

MEMBER HALNON: Put it in chat, then we can read it.

MEMBER KIRCHNER: Dennis, this is Walt. If you can hear me, if you could just text your question to Mike Snodderly, we'll loop back and try and address it. But we're having problems with your audio. Tom, why don't you go on and then we'll try and loop back to Dennis' question.

MR. HAYDEN: Sure thing. Perhaps seeing -- and I'm seeing that are referenced on this slide, on the next might shed some light. So L&C 5.2 specifically addresses inventory makeup to the ultimate heat sink. So identification of an available and sufficient source of water; acquisition, staging, and implementation of necessary motive equipment; and provision of debris removal capability to deploy offsite equipment.

L&C 5.3 specifically addresses the

assigned control room monitoring function. So sustaining control room operators for the 72-hour period after the start of the beyond design basis external event, provision of debris removal to allow egress from the control room. L&C 5.4 specifically addresses the capability of site support personnel to ascertain plant conditions to determine necessary coping requirements once onsite power systems are depleted. And L&C 5.5 specifically addresses long-term support related spent fuel pool level monitoring instrumentation. And as John mentioned, a provision is included in the above L&Cs to allow an applicant to provide a justification that supports not implementing the elements of that L&C.

MEMBER HALNON: Tommy, this is Greg. I would assume that you'd want at least the equivalent capabilities of those L&Cs.

MR. HAYDEN: Right. And that would be encompassed in the justification.

MEMBER HALNON: You're talking about pre-planning the implementation. They need to be able to do those. That would be in the justification. We don't have to pre-plan for it. Is that correct?

MR. HAYDEN: Could you repeat that?

MEMBER HALNON: Yeah, you mentioned if

they had justification to where they did not have to implement those L&Cs. My point is that they're asking for that you don't have to pre-plan those elements. But you still need to be capable of doing those things.

MR. HAYDEN: Yeah, so I'd have to refer back to the L&C as they're specifically written. But that might be --

MEMBER HALNON: In general, you want at least the equivalent capability to be able to restore or replenish or whatever case --

MR. HAYDEN: Sure.

MEMBER HALNON: Yeah, okay. So it's not that you don't want to be able to.

(Simultaneous speaking.)

MEMBER HALNON: Sustaining operations for 72 hours. But you're not saying that they can justify not doing it for 72 hours. They just don't have to pre-plan for it. The way that I read the TR was that they didn't have to do pre-planning. I would assume that you would expect in that justification at least the equivalent capabilities to do those things.

MR. HAYDEN: Yeah, I understand your question. I think I'd want to re-read specifically how the L&C is worded --

MEMBER HALNON: Okay. That's fair enough.

MR. HAYDEN: -- before I answer that. John, if you're capable of answering that now, that's fine. Or I can take that and I can make sure I have an answer by the end.

MR. HUGHEY: No, I think I can answer.

MR. HAYDEN: Thanks, John.

MR. HUGHEY: You are correct. So for an example, the first L&C talks about identifying a makeup water source, an inventory source for the ultimate heat sink. So for example, if one of these plants was located next to a very large river, very large lake, the ocean, somewhere where it was obvious that was the water source. And in their operating procedures, they already had procedural direction for how to get water from there.

It was protected for many environmental -- so if they already had things like that built into the design that were obvious, then there would be no need for that pre-planning. But you are correct. The issue is pre-planning. The plant still needs to be able to perform the mitigating capability.

MEMBER HALNON: Thank you.

MR. HAYDEN: Yeah, I think it may be misleading how that last bullet there is worded, what is in the L&C. You can justify not accomplishing them in their entirety, but the pre-planning part.

MEMBER HALNON: Right. So I just want to make sure we're focused on just the pre-planning.

MR. HAYDEN: Okay. Thank you. Okay. 3.4, capability to respond to a loss of large area due to explosion or fire event as required by 10 CFR 50.155(b)(2). That requirement, in part, is in regards to a LOLA. Strategies and guidelines must address in a three-phase approach, Phase 1, enhanced firefighting capabilities, Phase 2, measures to mitigate damage to fuel in the spent fuel pool, and Phase 3, measures to mitigate damage to fuel in the reactor vessel and to minimize radiological release.

The TR follows guidance in NUREG-0800, Temporary Instruction 2515/168, and NEI 06-12. For Phase 1, the TR lists design features that cope with potential loss of large area events, but it did not provide plant-specific features to demonstrate adequate LOLA coping capability. For Phase 2, the TR provides generic NuScale plant spent fuel pool design features that mitigate damage to fuel in the spent fuel pool. But it did not provide certain plant-specific details, for example, minimum spent fuel pool water level.

MR. SNODDERLY: Excuse me. This is Mike Snodderly from the ACRS staff. I have Dennis Bley's question. His question is, so would a NuScale referencing this topical report, would they have to have access to a safer type facility to obtain the FLEX equipment or not?

MR. HAYDEN: I think the answer is not. They don't need a safer center. They have the equipment on site.

MR. WAGAGE: Yes, NuScale does not rely on FLEX equipment or the coping capability.

MEMBER HALNON: Hanry, you need to state your name for the court reporter.

MR. WAGAGE: My name is Hanry Wagage from NRR. NuScale does not need any FLEX capabilities for this presented strategies.

MEMBER HALNON: And that includes onsite equipment and safer beyond --

MR. WAGAGE: Because you can then talk in the closed session that for a long time.

MEMBER HALNON: I think that was Dennis' question was all the commitments the industry made for FLEX equipment safe for Phoenix and Memphis or really where those things are -- that's not required here.

MR. WAGAGE: It's not requiring FLEX

equipment.

CHAIR DIMITRIJEVIC: Well, maybe all of this will be easier to be discussed in the closed section because now I'm sort of, like, everything is for here, Like what is for closed or for the open session. But one of the main ideas in the TR is that after 72 hours, you're still not in a hurry. There's still time available to get all equipment you need and organize those things.

So it is a slightly -- and this is where they will introduce this addition of time in analysis available for the -- to take an action, get organized, and see what needs to be done. So I think that maybe this will be better discussed in the closed session where we can discuss this extended after 72 hours' time. That's opinion about the subject. All right.

MR. HAYDEN: Sure. So I'll continue on with 3.4, capability to respond to a loss of a large area event. For Phase 3, the TR provides generic NuScale design features that are used to mitigate damage to fuel in the reactor vessel and minimize radiological release by plant key safety functions. But it did not provide plant-specific design features.

So for those reasons, L&C 5.1 for Phases 2 and 3 and L&C 5.6 for Phase 1 were written into the evaluation. And 5.1, we've gone over. But it addresses plant-specific design features, including key design features that would preclude the need for an enhanced spent fuel pool mitigation capability such as diverse or portable makeup capability to mitigate a LOLA event and maintain key safety functions to mitigate potential fuel damage and radiological release due to a LOLA event.

And L&C 5.6 for Phase 1, and that addresses the Fire Protection Program, including plant-specific design features and procedures, that would provide assurance of adequate LOLA coping capability. 3.5 capacity, capability, and protection of equipment associated with mitigation of events described in the rule, as required by 10 CFR 50.155(c). This regulation requires the equipment that is relied on for the mitigation strategies and guidelines have sufficient capacity and capability to perform the intended functions and be reasonably protected from the effects of natural phenomena that are equivalent in magnitude to the phenomena assumed for developing the design basis of the facility.

The TR provides the equipment relied upon for the mitigation strategies and guidelines. The TR also describes the capacity, capability, and protection of the equipment from natural phenomena. Subject to satisfaction of L&C 5.1, and applicant or licensee adopting the TR for a NuScale design could demonstrate compliance with 10 CFR 50.155(c). 3.6, training requirements as defined by 10 CFR 50.155(d).

NEI 12-06, Section 11.6 describes the training to be provided to address this regulation. The TR provided justification why specific training is not needed to address 10 CFR 50.155(d), but did not specifically address the required activities related to replacement of spent fuel pool level monitoring power supply. For that reason, the staff introduced L&C 5.7 which requests an adopter of the topical report to confirm that the training program includes the required activities related to replacement of spent fuel pool level monitoring power supply.

3.7, spent fuel pool monitoring after final fuel removal from the reactor vessel. The staff has endorsed NEI 12-02 which describes the design criteria for spent fuel pool monitoring system that meets the requirements of 10 CFR 50.155(e). The TR discussed the level instruments design criteria and indicated that they meet the guidance of NEI 12-02, but it did not address the training requirements.

For that reason, L&C 5.7 was used in this section of the evaluation as well. So finally, Section 4, our conclusion. Based upon its review as discussed above, subject to the limitations and conditions described in Section 5 of the evaluation, the NRC staff concludes that this topical report provides a reasonable methodology for an applicant or licensee to demonstrate the NuScale plant design capability to mitigate beyond design basis events as defined by 10 CFR 50.155.

MEMBER HALNON: Tommy, what is the process now for their topical? I mean, a lot of those limits and conditions were what I would consider they should've been there. Someone should've thought it out. It's pretty -- establish nuclear cultures to take it one step further and make sure that these capabilities are all there.

Are they going to revise the topical so that the revised one will have those items? Or is it -- because the SER itself is not part of a licensing basis. It's the topical report and their response to the SER. Are you expecting a response to the SER with a revision or an acceptance letter or something?

MR. HAYDEN: I would let NuScale answer that. But I do not expect that.

MR. CUMMINGS: This is Kris Cummings with

NuScale. So I'd actually say the SER is part of the topical report. So we'll get the SER. We've seen it. We've provided our comments on it.

We will get the final SER, and then we will submit to the NRC for verification that the approved topical report will include the SER. It will also include the one audit response that we docketed and the topical report itself. So when we have an approved topical report, it will include the SER as part of that.

MEMBER HALNON: Okay. So in the future if an applicant uses it, they're going to have all three of those. It's going to be one on the one title cover. The topical report will have the SER and question and answers in there?

MR. CUMMINGS: That's correct. And an adopter of the topical report would need to when they adopt it show how they meet the limitations and conditions and the conditions of use --

(Simultaneous speaking.)

MEMBER HALNON: I just want to make sure that was clear because there's a lot of -- 13 limits and conditions is a lot for this focused report, I think. It felt like you had to tell them a lot.

MR. HAYDEN: I hear what you're saying.

And I think from NuScale's perspective many of these would be or will be answered in the site specific application.

MEMBER HALNON: Yeah, okay.

MR. HAYDEN: And so for them, that was part of their consideration.

MEMBER HALNON: So they roll it into the –

(Simultaneous speaking.)

MR. HAYDEN: And our evaluation would obviously look to ensure that satisfies the L&Cs, whatever is put into that specifically.

MEMBER HALNON: Okay. Thank you for that clarification. Thank you.

MEMBER PALMTAG: This is Scott Palmtag. I just had a question about the FLEX. You said that the NuScale reactors won't have FLEX?

MR. WAGAGE: The answer is yes.

(Simultaneous speaking.)

MR. WAGAGE: My name is Hanry Wagage.

MEMBER KIRCHNER: Yeah, just pull the microphone closer to you.

MR. WAGAGE: I said yes. And the question is how do you need the reminder for going to -- the question is -- the answer is yes.

MR. HAYDEN: It will not have it?

MR. WAGAGE: It will not. It will not need any FLEX. No, sir.

MR. HAYDEN: He kind of explained it to me because it seemed like the -- I don't understand the exact regulations and FLEX, I'm not familiar with them. But it seems like FLEX was for beyond design basis for some unanticipated situation. It also seemed like very low hanging fruit.

MR. WAGAGE: Yes.

MR. HAYDEN: Why would you know want FLEX?

MR. WAGAGE: I think you must be thinking about operating reactors at three phases. So addressing mitigation of strategies. The first one is Phase 1 is to use plant equipment.

Then after using the plant equipment, then all the capability is gone. And they will rely on FLEX equipment. FLEX equipment is other equipment available at the facility.

They can use, then the third phase is when getting help from -- getting resources from outside to help with the situation. But NuScale is passive reactor. It can more than what other operating reactors do because we think that needs a pinpoint that can keep between actors in the reactivities and the reason that passive design -- and they'll need --

MR. HAYDEN: I understand it's passive. But it just seems like FLEX is that extra layer in case something unanticipated comes along.

MR. WAGAGE: Yes, but the adopter of this topical report has to show that core cooling containment and spent fuel pool can be maintained for the expected duration. So they have to show by thermal analysis.

MEMBER HALNON: Scott, that point is really aimed at the large light water reactors where it could possibly have problems in those three areas. And this topical shows that those three areas are covered by analysis and in plant design. So there would be no reason -- those vulnerabilities don't exist.

MEMBER PALMTAG: I understand that. And it's not a requirement. I understand. It just seemed like that'd be a low hanging fruit to have that just in case. But if it's not required, it's not required.

CHAIR DIMITRIJEVIC: Well, the basic idea they're trying to say after initiating safe shutdown in the beginning, no active systems are required. Everything is managed possibly without operator actions. So basically, therefore there is no FLEX equipment that'd be necessary for this situation.

I'm not sure here why you call this TR that describes methodologies. It's not really methodology. It's sort of like the plant design which enables this to be a situation.

So in this statement, it says provides reasonable methodology. This is not really methodology. But this is just like some comments on that statement.

MR. CUMMINGS: This is Kris Cummings from NuScale. I just wanted to respond to Scott's comment about the safer facility since that's on the record in the open session. I want to be able to address that.

You're right. We don't anticipate needing to subscribe to the safer -- the FLEX facilities. At the end, you spend a lot of time and effort to do for the light water reactors.

And that's because it's a NuScale design itself, the plant and the extended coping period that we'll talk about in the closed session that provides the real capacity to mitigate beyond design basis events. So that extended duration allows you to have a lot of time to be able to use something on a plant specific basis whereas the FLEX facilities in the safer centers were to be able to get equipment to the large light water plants that needed it very, very quickly. They don't have those large time frames. So we think this is a very appropriate response because of the design of the NuScale plant.

MEMBER PALMTAG: Thank you for the explanation.

CHAIR DIMITRIJEVIC: All right. Any more questions from the members? And then --

MEMBER ROBERTS: This is Tom Roberts. I'm trying to follow up on Scott's question that there's probably, I'm trying to count, 13, 14 systems in the applicant's slides that all are accredited to avoid having to rely on any of the FLEX equipment. It seems like it'd cause low hanging fruit to have the capability of acting in one of those systems that are accredited. I was wondering if they had thought of that --

(Simultaneous speaking.)

MEMBER ROBERTS: -- reactor building is probably the close to obvious one. The assumption is that the reactor is capable of holding the UHS water -- spent fuel pool water and protecting all the SSEs inside the reactor building. And so if you had some sort of hole in the floor, that's suddenly going to take a lot of the water away potentially. And I assume the reason why you don't think that can happen. But having the ability to bring in your equipment would seem like come from one of those unknowns.

CHAIR DIMITRIJEVIC: I propose we can continue discussing this in the closed session when we will have a little better feeling what is the main idea in this proposal. I think NuScale doesn't want to make a comment on this latest proposal. Then I propose that we open for public comments.

MR. CUMMINGS: Well, Vesna, this is Kris. I guess I'm elected to let that comment stand without a response. So let me just comment, right? 51.55 is promulgated after Fukushima. And that was specifically to consider a loss of all AC power, right?

And the specific event by which you lose AC power is not considered, right? But there are certain things that you can credit within the context of 51.55. And that includes reliance upon your safety-related equipment.

So the UHS is safety-related. It's designed to cite us in the Category 1. So as we describe in the topical report, that sort of a structure, the reactor building is expected to survive whatever the beyond design basis event is because it's designed to essentially survive.

So that's the context of the rule. And so that's the context of the topical report. So we want to consider specifically that sort of a situation in this particular topical report for addressing 51.55.

MEMBER ROBERTS: Okay. Thank you. I assume if a NuScale reactor found itself in an unexpected position they would use resources available --

MR. CUMMINGS: That's right.

MEMBER ROBERTS: -- to get whatever they needed to --

MR. CUMMINGS: There's an emergency plan. There's FEMA. There's ways to get other equipment on site, things to mitigate events that go beyond that.

MEMBER ROBERTS: Okay. Thank you.

MEMBER HALNON: Tommy, before we go back to Vesna, could you go back to your slide 13 here? I just wanted to go back to the crux of Dennis' question ant that's standard FLEX equipment such as mechanical connectors are compatible. We've been talking about FLEX is not necessary. So why do that?

MR. HAYDEN: I guess I'll answer that by saying that this is meant to show what the guidance in the NEI, it proposes, right?

MEMBER HALNON: Okay. So that's the NEI guidance, not --

(Simultaneous speaking.)

MR. HAYDEN: And then we talk about the L&Cs that we are having adopted.

MEMBER HALNON: So that's fine. I think that will clarify for Dennis.

MR. HAYDEN: Okay.

MEMBER HALNON: So that's the NEI guidance. And we're saying that they don't need to do that.

MR. HAYDEN: Right. In the previous slide, it talks about what the TR provided. This slide, it talks about however we do understand, we recognize what this guidance states. And so here are the four L&Cs that we've adopted.

MEMBER HALNON: Okay. I think that addresses Dennis' question.

MR. HAYDEN: Okay, great.

MR. CUMMINGS: So Scott, I just had one more. This is a minor detail. But at the very beginning, you said this TR applies to every NuScale reactor. I'm not sure what the exact wording was.

MR. HAYDEN: If that was the wording, that was not necessarily intended.

MR. HUGHEY: It's specifically the 460 and the 600.

MR. HAYDEN: No, that is not correct. It's not specific to either of those two designs. But it's not specific to -- you couldn't say any design. There is a criteria that's established in the topical report itself. And it's proprietary, but --

MR. HUGHEY: Any NuScale reactor with these criteria, if I remember right?

MR. HAYDEN: Yes, I see Milton. I just wanted to make sure I answer Scott's question first.

MR. HUGHEY: I just wondered if that should be -- because they come up with a new change, major change.

MR. HAYDEN: Right, if they have a new change and then that change somehow removes one of these systems or design features that this topical report relies upon, then it would not be a topical report.

MEMBER KIRCHNER: The area would be a thermal analysis of what's going on with the UHS. Assuming that we're looking at a nominal NuScale module. Limiting actor here would be the passive cooling and the UHS capacity, right?

We should discuss some aspects, I guess, further in a closed meeting. But I think it is in the open that the, for example, NuScale or the latest SDAA has changed the level in the UHS. But it would seem to me that in general that you, the NRC, would look at whatever the proposed level is and then look at how that supports the coping analysis to see if they can use 72 hours and then go beyond as part of the COL review.

MR. HUGHEY: Right, yeah. L&C 5.1 references those four conditions of use. And those conditions of use are housed in the TR itself. So the TR has to -- as part of -- likely as part of their application -- include that thermal analysis.

MR. HAYDEN: So for example, if they had enough power for ten years down the road, your criteria would cover that. They would have to show that would be applicable?

MR. HUGHEY: Yeah, within that change process, yes. Go ahead.

MR. VALENTIN: I'm sorry. I'm Milton Valentin. I am the supervisor for the containment and plant systems branch. And I just wanted to provide some context.

When the staff was looking at this report, we need to keep in mind the possibilities of the different design features and the site characteristics that will then define what will be needed at the specific site, either the use of FLEX equipment, either the reliance of certain systems or components that are now being generally created for these capabilities. So we understand that we are going here assuming generic things that might not be as descriptive as ideally we have seen for operating reactors. But the reality is that we don't have the information right now.

This is something that we're trying to capture in a way that in the future when we see the specific designs for these sites using these designs at the sites, all the information, then anyone will take on and be fair on looking at all of the details that will be important to make a safety finding. So here this is a very unconventional approach that we -- again, we're trying to come up from a place of giving credit for what we know but also keeping in mind other possibilities that could happen once we have the details. So I hope that helps everyone understand with some of the challenges that the staff has dealt with in trying to look at this from a more futuristic or, like, thinking about what will be important to look at when we actually have the design information and all of the analysis that you typically see at this point in time.

MR. HAYDEN: Thanks, Milt. At this time, I think Jim Osborn gave it in his opening statement that this is a framework and that the TR when it is adopted in the future will obviously be supported by analysis that ensures that the TR is applicable.

MR. CUMMINGS: I'll just add. Kris Cummings from NuScale. We certainly have two designs, the U.S. 460 and U.S. 600, that we have those two, a 6 pack and a 12 pack. As long as they meet the plant equipment that are identified and you can go through the thermal analysis that you have the extended coping time as described in the topical report, then you could apply this and say you basically rely on the plant itself to be the mitigating strategy with a few things that need to be done on a site or plant specific basis.

MR. VALENTIN: And Milton again. And the fact that they are saying that FLEX is not needed doesn't prevent them from actually using it. Like, somebody could come in and identify the need for FLEX. And then they will have that resource available for them. But that is something that the COL applicant will have to decide when they do the actual evaluation. Hope that helps.

CHAIR DIMITRIJEVIC: Okay. Any more questions? All right. So shall we open for the public comments? Anybody from the public wish to make a comment? Please let us know.

MEMBER KIRCHNER: Vesna, this is Walt. We're not seeing any hands raised.

CHAIR DIMITRIJEVIC: I was just looking at that.

MEMBER KIRCHNER: Let me reiterate your request. If anyone on the line wishes to make a comment, please signal so. Speak up.

CHAIR DIMITRIJEVIC: Unmute yourself. All right. So hearing none, I think we can go on the break and then come back in closed section. So it is now 2:27. So let's come back at 2:45. And we will sign in on the closed section for the people who are virtual connection. Okay. So see you in 17 minutes.

(Whereupon, the above-entitled matter went off the record at 2:28 p.m.)

September 27, 2024 Docket No. 99902078

U.S. Nuclear Regulatory Commission
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Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Presentation Material Entitled “ACRS Subcommittee Meeting (Open Session) Mitigation of Beyond Design Basis Events Topical Report,” PM-174186, Revision 0

REFERENCE: 1. Letter from NuScale to NRC, LO-170692, “NuScale Power, LLC Submittal of Topical Report ‘NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155,’ TR-141299-P, Revision 1 and Docketing of Resolved Audit Responses” dated June 26, 2024

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 October 1, 2024. The materials support NuScale’s presentation of the Mitigation of Beyond-Design-Basis-Events topical report, as submitted in Reference 1.

The enclosure to this letter is the nonproprietary presentation entitled “ACRS Subcommittee Meeting (Open Session) Mitigation of Beyond Design Basis Events Topical Report,” PM-174186, Revision 0.

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

If you have any questions, please contact Kris Cummings at (240) 833-3003 or at kcummings@nuscalepower.com.

Sincerely,

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Enclosure 1: "ACRS Subcommittee Meeting (Open Session) Mitigation of Beyond Design Basis Events Topical Report," PM-174186, Revision 0

Enclosure 1:

"ACRS Subcommittee Meeting (Open Session) Mitigation of Beyond Design Basis Events Topical Report," PM-174186, Revision 0

ACRS Subcommittee Meeting
(Open Session)

October 1, 2024

Mitigation of Beyond Design Basis Events Topical Report

Presenter: Peter Shaw

Acknowledgement and Disclaimer

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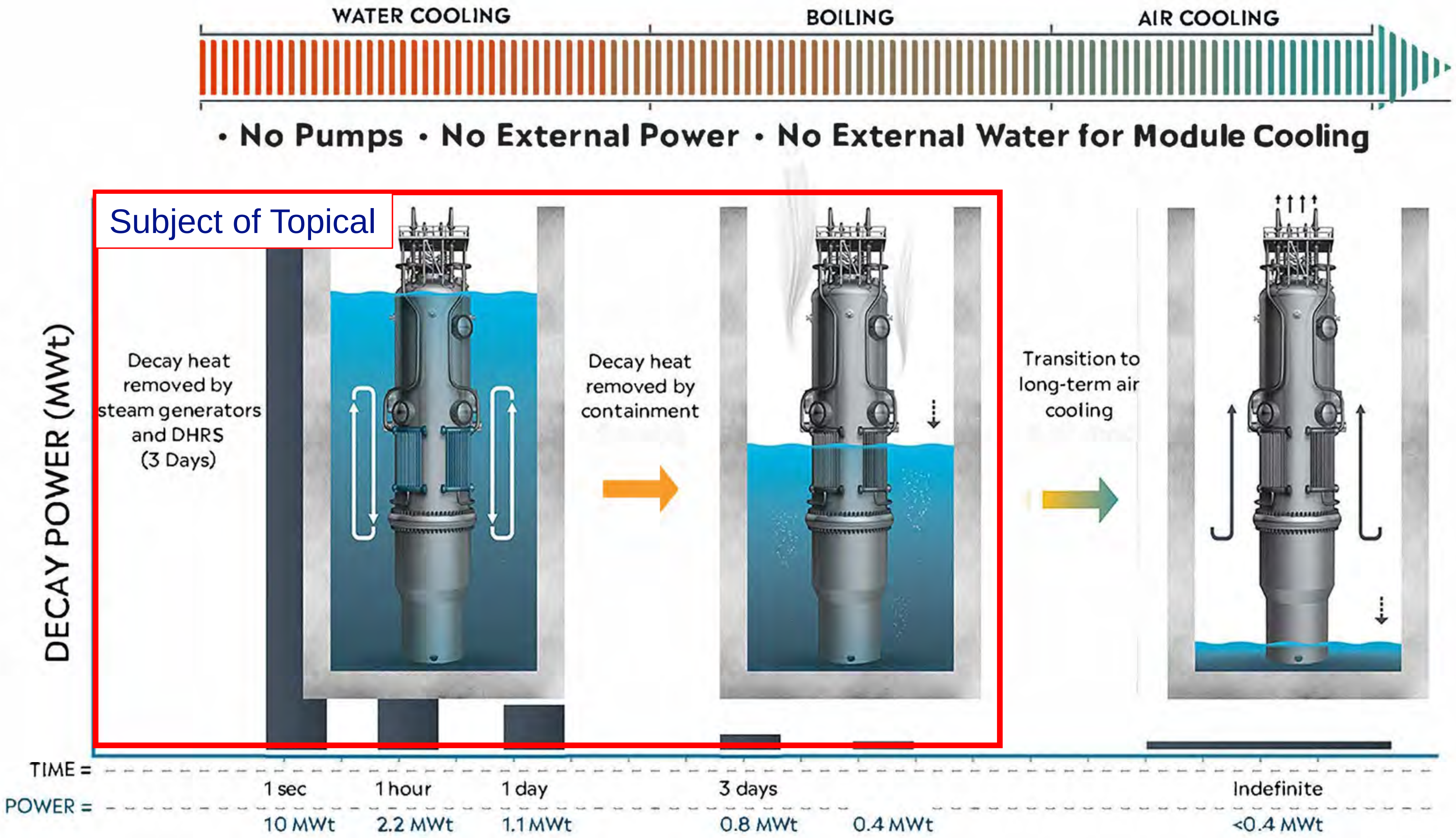
Agenda

- Introduction
- Overview of Plant Features
- Event Mitigation Features for Design-Basis Events
- Crediting of Safety Features for MBDBE
- Conditions for Applicability
- Summary of Plant Systems Relied Upon During Loss of AC Power
- Summary of Fulfilling Regulatory Commitments
- Conditions of Applicability (NRC)
- Conclusion

Introduction

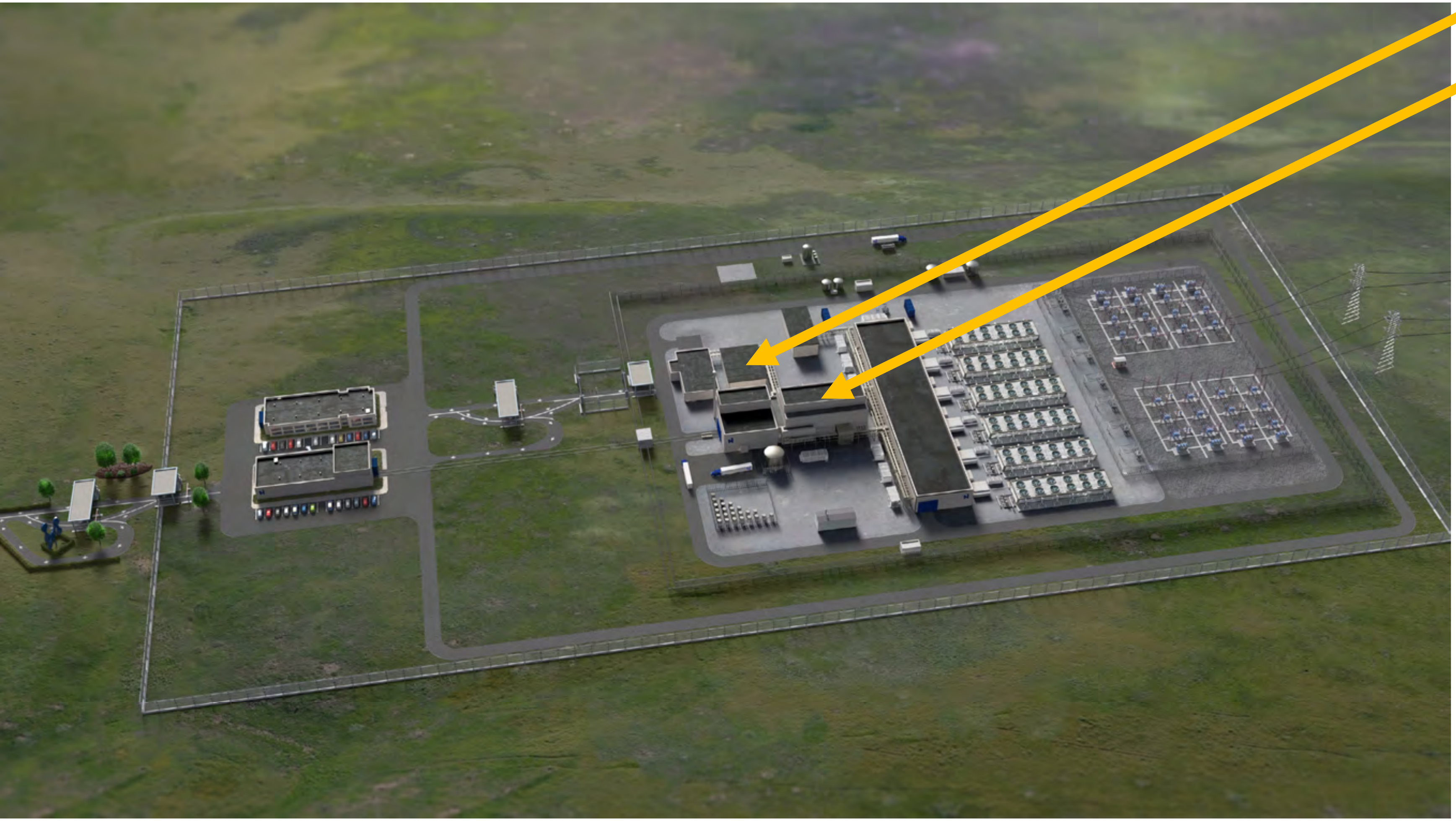
- Topical report establishes a consistent approach for mitigation of beyond-design-basis events (MBDBE) at a NuScale Power Plant to demonstrate compliance with 10 CFR 50.155
- Applicable to NuScale Power Plant configurations implementing design features described within the topical report
  - NuScale Power Plant passive safety features extend baseline coping period past traditional reactor technologies
  - Topical report describes how the systems are relied upon in a BDBE
- Informed approach using lessons learned from 10 CFR 50.155 guidance
  - NEI 06-12 “B.5.b Phase 2 & 3 Submittal Guideline”
  - NEI 12-06 “Diverse and Flexible Coping Strategies (FLEX) Implementation Guide”
  - IAEA “The Fukushima Daiichi Accident” Technical Volume 1/5 Description and Context of the Accident
  - NEI 12-02 “Industry Guidance for Compliance with NRC Order EA-12-051, To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation”
- An applicant decides how to fulfill the conditions of use (e.g. thermal analysis)

NuScale Reactor Module for Mitigation of Beyond-Design-Basis Events



*water level and decay heat are not indicative of final plant design

Plant Layout Overview*

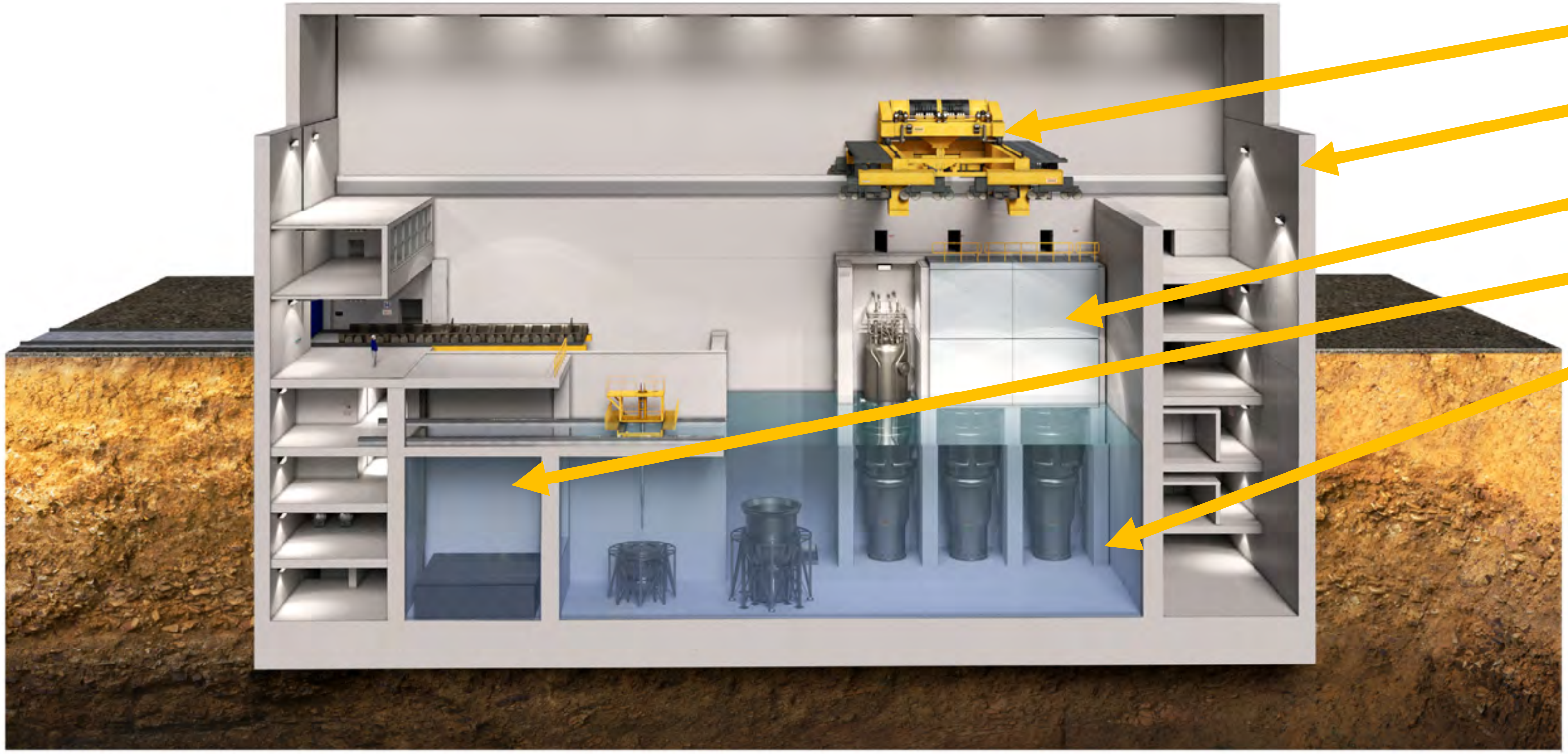


Control Building

Reactor Building

*Figures are representative of a typical 6 pack power plant

Reactor Building Overview*



Overhead Heavy Load Handling System

Reactor Building Walls

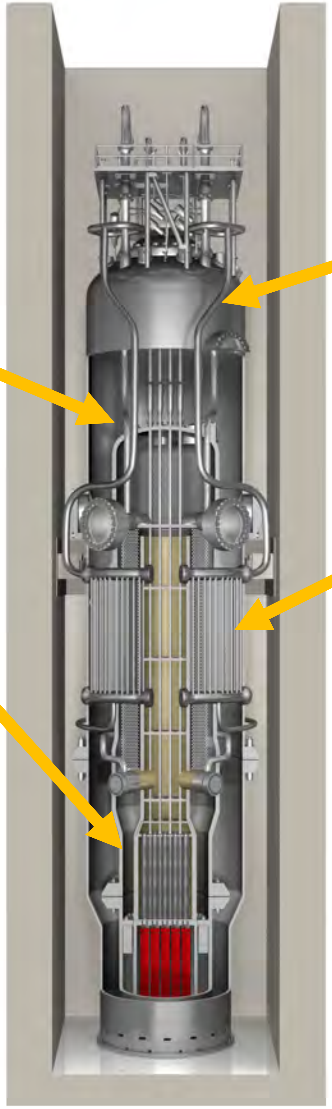
Bioshield

Spent Fuel Pool

Ultimate Heat Sink

*Figures are representative of a typical 6 pack power plant

NuScale Power Module Overview*



Containment Vessel

Emergency Core Cooling System Reactor Vent Valve

Decay Heat Removal System

Emergency Core Cooling System Reactor Recirculation Valve

*Figures are representative of a NuScale Power Module

Event Mitigation Features for Design-Basis Events

- Safe and stable shutdown is achieved without operator actions
  - Automatic reactor trip
  - Containment isolation activates automatically
  - Decay heat removal system passively removes decay heat for 24 hours
  - If AC power supply is not restored by 24 hours, emergency core cooling activates
  - Modules rely on no additional equipment to maintain core cooling
- Spent fuel pool shares common water source with reactor modules, thus only a single volume need be maintained for coping period
- Although not credited for safe-shutdown, system response is assured for 72 hours in the main control room
- All design-basis mitigation features apply to BDBEs providing capability for successful mitigation past large light water reactor design coping periods

Crediting of Safety Features for MBDBE

- NuScale Power Plants implement redundant and reliable safety features that mitigate core damage
  - Immediate plant shutdown and containment isolation at initiation of event
  - Containment vessels are protected by the reactor building from external events
  - No operator actions needed to initiate safe-shutdown

- NuScale Conditions of Use
  - Maintain plant specific design features described in report
  - Provide a thermal analysis to validate proposed coping period
  - Establish a maintenance rule program in accordance with 10 CFR 50.65
  - Establish an emergency plan in application

Summary of Plant Systems Relied Upon During Loss of AC Power (10 CFR 50.155(a))

- Reactor building protects modules from design-basis natural phenomena and houses the ultimate heat sink for passive core cooling
- Control building protects operators from design-basis natural phenomena
- Augmented direct current power system provides continuous DC power for equipment
- Module protection system automatically actuates safe-shutdown functions without need for operator actions
- Plant protection system monitors reactor, refueling, and spent fuel pool conditions, common to all modules
- Safety display and indication system provides nonsafety-related accident monitoring

Summary of Plant Systems Relied Upon During Loss of AC Power (Continued)

- Containment isolation so passive safety systems can reliably initiate safe-shutdown (prevent uncontrolled release, dissipate decay heat)
- Ultimate heat sink provides common water source for maintaining safe-shutdown, assured water make-up line
- Decay heat removal system removes decay heat from shutdown reactor into ultimate heat sink
- Emergency core cooling system circulates reactor coolant into containment to initiate long term passive cooling
- Reactor building and control room building ventilation systems maintain safe atmospheres for respective environments
- Overhead heavy load handling system supports a module during lifts for design-basis natural phenomena
- Communication system used to coordinate operators and responders after a BDBE

Summary of Compliance with Regulatory Requirements

- 10 CFR 50.155(b)(1)
  - Loss of AC power is identical to station blackout, safe and stable shutdown is achieved without operator actions
  - Spent fuel pool shares common water source with reactor modules, thus only a single volume is maintained for coping period
- 10 CFR 50.155(b)(2) and (c)
  - Exterior concrete walls designed for aircraft impacts and similar events
- Training requirements as defined by 10 CFR 50.155(d)
  - No additional training needed
- Spent fuel pool monitoring until final fuel removal from the reactor vessel, per 10 CFR 50.155(e)
  - Spent fuel pool monitoring in common heat sink
  - Instrumentation meets guidance NEI 12-02

Limitations and Conditions (NRC)

- NRC review completed with 24 audit questions, with one docketed response and no RAIs
- NRC review identified site-specific considerations based on plant location
- These considerations are normally part of site selection, and will be documented as part of an application
- Additional Limitations and Conditions state an adopter must:
  - Justify why no pre-planning is needed using the topical report's methodology for 10 CFR 50.155, if not an adopter must identify:
    - Water for ultimate heat sink
    - Necessary motive equipment such as pumps and generators
    - Equipment for debris removal
    - Address sustainment of operators for 72 hours post-event
    - Address how personnel will ascertain plant conditions post-event
    - Address how spent fuel level instrumentation power source will be replaced post-event
  - Provide a fire protection program in accordance with 10 CFR 50.48
  - Provide a training program that includes equipment relied upon in a BDBE

Summary

- The NuScale Power Plant design provides extended coping capability for mitigating BDBEs.
- Operator actions are not necessary to establish a safe-shutdown condition
- Topical report establishes a consistent approach for MBDBE at a NuScale Power Plant to demonstrate compliance to 10 CFR 50.155

Acronyms

AC	Alternating Current
DC	Direct Current
DHRS	Decay Heat Removal System
BDBE	Beyond-Design-Basis Event
NEI	Nuclear Energy Institute
MBDBE	Mitigation of Beyond-Design-Basis Event

Questions?

Presentation to the Advisory Committee on Reactor Safeguards Subcommittee Staff Review of NuScale's Power Plant Design Capability to Mitigate Beyond Design-Basis Events Defined by 10 CFR 50.155

NuScale TR-141299-P, Revision 1

October 1st, 2024
(Open Session)

Non-Proprietary

Acronyms and Definitions

- AC – Alternating Current
- ACRS – Advisory Committee on Reactor Safeguards
- APLB – PRA Licensing Branch B
- APLC – PRA Licensing Branch C
- APOB – PRA Oversight Branch
- BDBE – Beyond Design Basis Events
- BDBEE – Beyond Design Basis External Events
- CFR – Code of Federal Regulations
- CR – Control Room
- DEX – Division of Engineering and External Hazards
- DC – Direct Current
- DNRL – Division of New and Renewed Licenses
- DRA – Division of Risk Assessment
- DSS – Division of Safety Systems
- EDAS – Augmented DC power system
- EEEB – Electrical Engineering Branch
- EMIB – Mechanical Engineering and Inservice Testing Branch
- GDC – General Design Criteria
- L&C – Limitation and Condition
- LOLA – Loss of Large Plant Area due to Explosion or Fire
- MBDBE – Mitigation of Beyond Design Basis Events
- NPM – NuScale Power Module
- NRLB – New Reactor Licensing Branch
- NRR – Office of Nuclear Reactor Regulation
- PRA – Probabilistic Risk Assessment
- SBO – Station Blackout
- SCPB – Containment and Plant Systems Branch
- SNRB – Nuclear Methods Systems and New Reactors Branch
- TR – Topical Report
- UHS – Ultimate Heat Sink

NuScale MBDBE Topical Report Review

Overview

- NuScale submitted Topical Report TR-141299-P, “NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155,” Revision 0 on September 11, 2023, and Revision 1 on June 26, 2024.
- NRC regulatory audit of the TR was performed from December 2023 to March 2024
- 24 audit issues were generated. 11 were resolved via audit response. 13 were resolved via Limitations and Conditions.
- NuScale submitted supplemental information to address one issue raised during the audit
- No RAIs were issued
- Staff completed Topical Report review and issued an advanced safety evaluation to support today’s ACRS Subcommittee meeting

NuScale MBDBE Topical Report Review

Contributors

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NuScale MBDBE Topical Report Review

Sections

- Section 1.0 – Introduction
- **Section 2.0 – Background**
- **Section 3.0 – Technical Evaluation**
  - **3.1 - Plant Baseline Coping Criteria for a Loss of All AC Power**
  - **3.2 - Plant Systems and Responses to a Loss of All AC Power**
  - **3.3 - Safety Functions during a Loss of All AC Power**
  - **3.4 - Capability to respond to a LOLA event, as required by 10 CFR 50.155(b)(2)**
  - **3.5 - Capacity, Capability, and protection of equipment associated with mitigation of events described in the rule, as required by 10 CFR 50.155(c)**
  - **3.6 - Training requirements as defined by 10 CFR 50.155(d)**
  - **3.7 - SFP Monitoring after final fuel removal from the reactor vessel, as required by 10 CFR 50.155(e)**
- **Section 4.0 – Conclusion**
- Section 5.0 – Limitations and Conditions
- Section 6.0 – References

NuScale MBDBE Topical Report Review

Section 2.0 - Background

- 2.1 – Regulatory Requirements and Relevant Regulatory and Industry Guidance
- 2.2 – Summary of Technical Information
  - The TR outlines applicability only to NuScale small modular reactor designs with specific characteristics and Conditions of Use an adopter of the TR must provide
  - The TR contains description of NuScale design capabilities and features including passive safety systems capable of maintaining core cooling, containment, and spent fuel cooling functions and a large reactor pool serving as the UHS
    - TR specifies how these features enable a design to mitigate BDBEs for a specified extended duration without the need for AC power, special equipment or addition guidelines and strategies.

NuScale MBDBE Topical Report Review

3.1 – Plant Baseline Coping Criteria for Loss of All AC Power

- 3.1.1 – Assessment of Electrical Power
  - During the first 72 hours of a loss of all AC power is identical to an SBO and no AC power is relied upon for performing safety functions.
  - The initial conditions and assumptions in NEI 12-06, Revision 4 and RG 1.226, Revision 0, assume that station batteries would remain available following a BDBE since they are considered robust.
  - In the TR, section 3.1.5, "Initial Event Conditions and Assumptions," NuScale states that for the baseline coping capability,

    1) station batteries and associated DC buses remain available for the designed operating time of the station batteries, and

    2) installed electrical distribution system, including inverters and battery chargers, remain available provided they are seismic Category I.
  - Initial assumptions in the TR following a BDBEE are consistent with the NEI 12-06, Revision 4, guidance, endorsed by RG 1.226.

NuScale MBDBE Topical Report Review

3.1 – Plant Baseline Coping Criteria for Loss of All AC Power

- 3.1.1 – Assessment of Electrical Power (cont.)
  - UHS monitoring, including SFP level, is assured for a timeframe in excess of the "7 days post-event" minimum in NEI 12-02.
    - The staff notes that UHS and SFP level instruments can be powered by multiple means.
    - The capacity and capability of such multiple means, are not discussed and therefore, an applicant must satisfy L&C no. 5.2
  - L&C no. 5.2: Address how plant operators will ensure, during the initial coping phase, that the following can be achieved to provide makeup to the UHS:
    - Source of water identified and available in sufficient quantity
    - Necessary motive equipment such as pumps and generators, and the required electrical power/fuel, can be obtained, staged, and implemented
    - Any required debris removal will be accomplished to support placement of equipment and access to site connections

NuScale MBDBE Topical Report Review

3.1 – Plant Baseline Coping Criteria for Loss of All AC Power

- 3.1.2 – Plant Design Capabilities
  - Following a loss of all AC power event, automatic responses of safety-related equipment establish and maintain core cooling, containment, and SFP cooling by placing the reactor modules and spent fuel into a safe, stable, shutdown state with passive cooling.
  - L&C no. 5.1, an adopter must satisfy Conditions of Use in TR Section 1.3, by providing
    - Plant specific design, as described within the report
    - **Plant specific thermal analysis (including configuration of the plant, number of modules, SFP capacity, for all modes of operation)**
    - A maintenance rule program IAW 10 CFR 50.65
    - An emergency plan IAW 10 CFR 50.160 or 50.47(b) and App. E describing communications and coordination with local, state, federal, and tribal agencies.

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NuScale MBDBE Topical Report Review

3.2 – Plant Systems and Responses to a Loss of All AC Power Event

- 10 CFR 50.155 requires that equipment have the capacity and protection to be used to support MBDBE
- The TR section 4.0 lists proposed equipment for a NuScale SMR design
- The TR also lists the protection of this equipment
- The design specific and location specific nature of the MBDBE leads to needed analysis from a COL applicant. The analysis would need to confirm the assumptions in the TR remain true for the proposed plant as well as contain analysis to show that the proposed plant systems would be able to respond to the progression of the event for the proposed period of time.
- L&C no. 5.1 was added to address the need for connection of analysis and the plant system responses for the event.

NuScale MBDBE Topical Report Review

<u>3.3 – Safety Functions during a Loss of All AC Power</u>

- 10 CFR 50.155 states that applicants shall develop, implement and maintain strategies and guidelines to maintain or restore core cooling, containment and spent fuel pool cooling capabilities.
- The TR discusses the strategies for maintaining these functions during the event and how it is accomplished with a NuScale SMR.
- The site-specific nature of hazards and potential site layouts and locations requires that a COL applicant address the potential events at their specified site.
- <u>L&C no. 5.1</u> address the need for an applicant to provide analysis and potential guidelines and strategies dependent on the analysis.

NuScale MBDBE Topical Report Review

3.3.2 – Indefinite maintenance of core cooling, containment, and SFP cooling capabilities

- The regulation requires that offsite assistance and resources be acquired indefinitely or until functional capabilities can be maintained without the need for mitigating strategies.
- The endorsed guidance in NEI 12-06 recognizes that FLEX strategies and offsite resources do not need to be explicitly planned for the period beyond 72 hours.
- TR proposes that no pre-planning for mitigating actions is required because a NuScale SMR design provides BDBEE mitigating capability beyond 72 hours.

NuScale MBDBE Topical Report Review

3.3.2 – Indefinite maintenance of core cooling, containment, and SFP cooling capabilities

- However, the guidance in NEI 12-06 also presumes that initial coping mitigating strategies have been established such that:
  - Staging areas to receive offsite equipment are identified;
  - Means are established to transport equipment to deployment areas;
  - The ability of an offsite organization to provide support has been ensured;
  - Standard FLEX equipment mechanical/electrical connectors that are compatible with site connections are obtained.
- L&Cs nos. 5.2, 5.3, 5.4 and 5.5 address the requirement for long term mitigating capabilities associated with the NuScale SMR design discussed in the TR.

NuScale MBDBE Topical Report Review

3.3.2 – Indefinite maintenance of core cooling, containment, and SFP cooling capabilities

- L&C no. 5.2 specifically addresses inventory makeup to the UHS:
  - Identification of an available and sufficient source of water;
  - Acquisition, staging, and implementation of necessary motive equipment;
  - Provision of debris removal capability to deploy offsite equipment.
- L&C no. 5.3 specifically addresses the assigned control room monitoring function:
  - Sustaining CR operators for the 72-hour period after the start of the BDBEE;
  - Provision of debris removal to allow egress from the control room.
- L&C no. 5.4 specifically addresses the capability of site support personnel to ascertain plant conditions to determine necessary coping requirements once onsite power systems are depleted.
- L&C no. 5.5 specifically addresses long-term support related to SFP level monitoring instrumentation.
- A provision is included in the above L&Cs to allow an applicant to provide a justification that supports not implementing the elements of that L&C.

NuScale MBDBE Topical Report Review

3.4 – Capability to respond to a LOLA event, as required by 10 CFR 50.155(b)(2)

- 10 CFR 50.155(b)(2) requires, in part, in regards to a LOLA, strategies and guidelines must address in a three-phase approach:
  - Phase 1 – Enhanced firefighting capabilities
  - Phase 2 – Measures to mitigate damage to fuel in the SFP, and
  - Phase 3 – Measures to mitigate damage to fuel in the reactor vessel and to minimize radiological release
- The TR follows guidance in NUREG-0800, Temporary Instruction 2515/168, and NEI 06-12. For Phase 1, the TR lists design features that cope with potential LOLA events, but it did not provide plant-specific features to demonstrate adequate LOLA coping capability.
- For Phase 2, the TR provides generic NuScale plant SFP design features that mitigate damage to fuel in the SFP, but it did not provide certain plant-specific details (e.g., minimum SFP water level).

NuScale MBDBE Topical Report Review

3.4 – Capability to respond to a LOLA event, as required by 10 CFR 50.155(b)(2)

- For Phase 3, the TR provides generic NuScale design features that are used to mitigate damage to fuel in the reactor vessel and minimize radiological release by plant key safety functions, but it did not provide plant-specific design features
- L&C no. 5.1 (for Phase 2 and 3):
  - addresses plant-specific design features, including key design features, that would:
    - preclude the need for an enhanced SFP mitigation capability such as a diverse or portable makeup capability to mitigate a LOLA event.
    - maintain key safety functions to mitigate potential fuel damage and radiological release due to a LOLA event.
- L&C no. 5.6 (for Phase 1): addresses the Fire Protection Program, including plant-specific design features and procedures, that would provide assurance of adequate LOLA coping capability.

NuScale MBDBE Topical Report Review

3.5 – Capacity, Capability, and protection of equipment associated with mitigation of events described in the rule, as required by 10 CFR 50.155(c)

- 10 CFR 50.155(c) requires providing the equipment relied on for the mitigation strategies and guidelines have sufficient capacity and capability to perform the intended functions and be reasonably protected from the effects of natural phenomena that are equivalent in magnitude to the phenomena assumed for developing the design basis of the facility.
- The TR provides the equipment relied upon for the mitigation strategies and guidelines. The TR also describes the capacity, capability, and protection of equipment from natural phenomena.
- Subject to satisfaction of L&C no. 5.1, an applicant or licensee adopting the TR for a NuScale design could demonstrate compliance with 10 CFR 50.155(c).

NuScale MBDBE Topical Report Review

3.6 – Training requirements as defined by 10 CFR 50.155(d)

- NEI 12-06, section 11.6 describes the training to be provided to address 10 CFR 50.155(d).
- The TR provided justification why specific training is not needed to address 10 CFR 50.155(d), but did not specifically addressed the required activities related to replacement of SFP level monitoring power supply. Therefore, that staff introduced L&C no. 5.7.
- L&C no. 5.7 requests an adopter of the topical report to confirm that the training program includes the required activities related to replacement of SFP level monitoring power supply.

NuScale MBDBE Topical Report Review

3.7 – SFP Monitoring after final fuel removal from the reactor vessel, as required by 10 CFR 50.155(e)

- The staff has endorsed NEI 12-02, which describes the design criteria for a SFP monitoring system that meets the requirements of 10 CFR 50.155(e).
- The TR discussed the level instruments design criteria and indicated that they meet the guidance of NEI 12-02, but it did not address the training requirements.
- L&C no. 5.7 addresses the training of the operators to make the necessary connections to establish replacement power sources.

NuScale MBDBE Topical Report Review

Section 4.0 – Conclusion

- Based upon its review as discussed above, subject to the limitations and conditions as described in section 5.0 of the SE, the NRC staff concludes that TR-141299-P, Revision 1, provides a reasonable methodology for an applicant or licensee to demonstrate the NuScale plant design capability to mitigate BDBEs as defined by 10 CFR 50.155.

Non-Proprietary

Attendance List for Open NuScale Subcommittee Meeting on October 1, 2024

Michael Snodderly
Thomas Dashiell
Shandeth Walton
Meghan McCloskey (NuScale)
Erwin Laureano (NuScale)
Jim Osborn
James Cordes - Court Reporter
Augi Cardillo (NuScale)
Wendy Reid (NuScale Power)
Ron Ballinger
Vesna B Dimitrijevic
Dave Midlik
Dan Lassiter (NuScale)
Larry Burkhart
Iulia Jianu
River Rohrman (She/Her)
Derek Widmayer
Tyesha Bush
Tim Polich
Angelo Stubbs
Robert White
Sean McCloskey
Robert Martin (He/Him)
Dennis Bley
Sarah Horacek
Thomas Hayden
Gene Eckholt - NuScale
Tammy Skov
Alissa Neuhausen
Gregory Halnon
Sandra Walker
Dominik Muszynski
Joy Jiang
Thinh Dinh
Mahmoud -MJ- Jardaneh