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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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NuSCALE SUBCOMMITTEE

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OPEN SESSION

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WEDNESDAY

MAY 15, 2019

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ROCKVILLE, MARYLAND

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The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2D10, 11545 Rockville Pike, at 8:30 a.m., Matthew W. Sunseri, Michael L. Corradini, and Vesna B. Dimitrijevic, Co-Chairs, presiding.

COMMITTEE MEMBERS:

MICHAEL L. CORRADINI, Co-Chair

VESNA B. DIMITRIJEVIC, Co-Chair

MATTHEW W. SUNSERI, Co-Chair

RONALD G. BALLINGER, Member

DENNIS BLEY, Member

CHARLES H. BROWN, JR. Member

JOSE MARCH-LEUBA, Member

JOY L. REMPE, Member

GORDON R. SKILLMAN, 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 SUNSERI: Good morning. The meeting has now come to order. This is a meeting of the Advisory Committee on Reactor Safeguards, NuScale Subcommittee.

I am Matt Sunseri, Co-Chairperson for today's Subcommittee meeting, along with Mike Corradini and Vesna Dimitrijevic.

Members in attendance today are Dr. Dimitrijevic, Dennis Bley, Jose March-Leuba, Joy Rempe, Matt Sunseri, Michael Corradini, Dick Skillman, Ron Ballinger, and Charlie Brown will be joining us shortly.

He is held up in traffic, but will be here soon. We also have our consultant, Steve Schultz, with us today and did I get everybody? Yes. All right.

Mike Snodderly is the designated federal official for this meeting. The Subcommittee will continue its review of the staff's evaluation of Chapter 19, Probabilistic Risk Assessment and Severe Accident Evaluation and we will review the staff's evaluation of Chapter 21, Multimodal Design Considerations of the NuScale Design Certification

Application.

Today we have members of the NRC staff and NuScale to brief the Subcommittee. The ACRS was established by statute and is governed by the Federal Advisory Committee Act. That means that the Committee can only speak through its published letter reports.

We hold meetings to gather information to support our deliberations. Interested parties who wish to provide comments can contact our office requesting a time after the meeting announcement is published in the Federal Register.

That said, we set aside ten minutes for comments from members of the public attending or listening to our meeting. Written comments are also welcome.

The ACRS Section of the U.S. NRC public website provides our charter, bylaws, letter reports and full transcripts of full and Subcommittee meetings, including slide presented there.

The rules for participation at today's meetings were announced in the Federal Register on May 6th, 2019. The meeting was announced as an open and closed meeting.

We may close the meeting after open portions to discuss proprietary material and presenters can defer questions that should not be answered in the public session to that time.

No written statements or requests for making any oral statements to the Subcommittee have been received from the public concerning this meeting.

A transcript of the meeting is being kept and will be made available as stated in the Register -- in the Federal Register notice. Therefore, we request that participants in this meeting use the microphones located throughout the meeting room when addressing the Subcommittee.

Participants should first identify themselves and speak with sufficient clarity and volume so they can be readily heard. We have a bridge line established for the public to listen in.

To minimize disturbance, the public line will be kept in a listen-in only mode. To avoid disturbances, I request that the attendees now put any electronic devices, like, cell phones in the off or noise-free mode.

So before we begin today's topics, I'd like to pick up on a continuation from yesterday. We had a discussion at the end of the -- yesterday about acquiring information regarding reliability of some components.

And I'm going to ask Mike Snodderly to pick this up and walk us through that issue and where we are as far as getting the information we need. Mike?

MR. SNODDERLY: Thank you. Thank you. Yesterday there was --

(Simultaneous speaking.)

CO-CHAIR SUNSERI: Is your mic on?

(Simultaneous speaking.)

MR. SNODDERLY: Hopefully, well --

PARTICIPANT: Yes. You have to talk straight into it, Mike, and that'll do it.

MR. SNODDERLY: Let's use this one. So there was some discussion about the availability of the NuScale report Passive Safety System Reliability PRA Report.

This report was audited as part of the first audit and it is not on the docket. But there were -- there are four slides that were a part of the October 18th. So yes.

We're going to get those four slides to Dr. Bley and to Dr. Dimitrijevic. Also, because the Committee does not have access to that report at this time, we'd like to request that at the June full Committee that the staff be prepared to provide further details from their notes on what -- on their findings or what they looked at that gave them confidence in the reliability numbers used in that report.

Also, the -- so just wanted to make sure that that was the Committee's understanding of the availability of that report and us -- and going forward. Okay. Thank you.

CO-CHAIR SUNSERI: Michelle, do you have any comments you want to make before we get started?

MICHELLE: I'm very good, thank you.

CO-CHAIR SUNSERI: All right. Now I turn to Sarah Bristol for the start of the presentation today. Sarah?

MS. NORRIS: Good morning, everyone. Welcome back. This is actually Rebecca Norris with NuScale. So today we're going to be covering three sections of Chapter 19.

Starting with Chapter 19.2 or the break with 21 and then 19.3 and 19.5. We have added one person on our presentation team. So I'd like everyone to introduce themselves again for the Members.

MR. RAD: All right. This is Zac Rad, Director of Regulatory Affairs, NuScale Power.

MR. MULLIN: Hi. This is Etienne Mullin.

I'm a PRA analyst.

MS. BRISTOL: Sarah Bristol, supervisor of the PRA group.

MR. GALYEAN: Bill Galyean, NuScale PRA.

MS. NORRIS: So first, we're going to be covering 19.2, Severe Accident Evaluation and with that, I'll let Etienne take it away.

MR. MULLIN: All right. So today we're going to talk about severe accidents briefly on topic. I did not bring as many slides as I did in October and they're pretty high level. So I hope this spurs some discussion.

I'm sure you all won't hesitate to speak up. So in Chapter 19.2, we evaluated severe accidents for the potential to challenge containment integrity and thereby result in a large release per our definition.

As we discussed yesterday, we define a large release as a source term that results in a dose of 200 rem at the site boundary for a stationary individual, standing there for 96 hours, in the worst possible location.

We used the MACCS code, that's MELCOR Accident Consequence Code System, to evaluate the outside consequence for our worst case intact CNV release and that's assuming a certain leak rate from the containment.

And determined from that calculation that there's no way that an intact containment core damage event could possibly result in a large release with orders or magnitude of margin.

MEMBER BLEY: How did you pick the leak rate that you're using?

MR. MULLIN: We have a -- there's a technical specification we create that has to be met at .25.

MEMBER BLEY: Okay.

MR. MULLIN: Yes.

MEMBER BLEY: So we meet the standard?

MR. MULLIN: Yes. And from that result, we decided, as a simplification, to take all of our intact CNV cases, assign them a not-large release and the Level 2 event trees and all of our bypass accidents as a simplification or a large release.

Now that's not to say that we evaluated all of our bypass accidents and calculated doses and determined them to be large releases. It's just a simplification. So, therefore, in our severe accident evaluation --

MEMBER BLEY: I'm sorry. I'm just --

MR. MULLIN: Go ahead.

MEMBER BLEY: -- hanging on that. I remembered some other places where I've heard things like this. So the tech spec leak rate means you can have up to that leak rate and keep operating the facility. So you assume there's no -- essentially assume there's no damage when this falls over?

MR. MULLIN: Yes.

MEMBER BLEY: It's leaking at whatever leak rate it's --

(Simultaneous speaking.)

MR. MULLIN: We're not talking about module drop.

MEMBER BLEY: Okay.

MR. MULLIN: This is it internalize --

MEMBER BLEY: But this is -- okay. And you've -- you check this every refueling or something?

MR. MULLIN: I believe so. Yes.

MEMBER BLEY: Okay.

MR. MULLIN: I think each individual penetration is tested and then they sum them together to get leak rate.

MEMBER BLEY: Thanks.

MR. MULLIN: So we've evaluated the phenomenological challenges to containment integrity that have been evaluated in the industry for years now.

By virtue of using the same fuel, same coolant, similar materials, you know, we were unable to identify any new phenomenological challenges introduced by the NuScale design.

So the challenges that we looked at are steam explosion, in-vessel, ex-vessel, hydrogen combustion, high pressure melt ejection and direct containment heating. That's another ex-vessel challenge --

MEMBER BLEY: I'm sorry. I'm still hanging on that. So even if you have a steam explosion in-vessel, you assume there's no damage to the containment from that?

MR. MULLIN: Well we evaluate the potential for a challenge to containment integrity from a steam explosion.

MEMBER BLEY: Okay. So you have some chance that you do more damage than be allowed? Correct?

MR. MULLIN: Well --

MEMBER BLEY: Are there other accidents for which that's true? That you look for consequential containment damage?

MR. MULLIN: Yes. I mean, we looked at all of these challenges from steam explosion to hydrogen combustion to direct containment heating, melt through.

MEMBER BLEY: And what approach did you use to calculate the probability -- well the chance that you break the containment or break it more than it was leaking?

MR. MULLIN: Well we evaluated all these phenomena. Determined it's --

MEMBER BLEY: So it's a judgment, engineering judgment?

MR. MULLIN: Well calculations are involved.

MR. GALYEAN: Dennis, you're getting ahead of him.

MR. MULLIN: Yes.

MR. GALYEAN: Okay. At this point --

MR. MULLIN: Okay.

MR. GALYEAN: -- he's just saying that if for a nominally intact containment, okay, that --

MEMBER BLEY: So we're going to get to other accidents --

MR. GAYLEAN: There's no way --

(Simultaneous speaking.)

MEMBER BLEY: -- and talk about --

MR. GALYEAN: Yes.

MR. MULLIN: We're going to start talking about --

MEMBER BLEY: When you get there, I'm interested in how you -- what kind of calculations you use to come up with a whole size, if you will, and a probability --

(Simultaneous speaking.)

MR. GALYEAN: And again, we'll get to that.

MEMBER BLEY: Okay.

MR. GALYEAN: We'll get to that.

MEMBER BLEY: I'm happy to wait.

MR. GALYEAN: Okay.

MEMBER BLEY: Don't let me forget.

(Laughter.)

DR. SCHULTZ: One question. On the simplification of the bypass accidents counted as large releases. Is that something you expect to get back to later or is it a simplification that is not to be addressed again?

MR. MULLIN: You can go ahead.

MR. GALYEAN: Yes. It's a simplification, because we can easily meet the NRC acceptance criteria with that simplification. Okay? And so there's no motivation to do detailed calculations when we don't have to, basically.

If the time comes and we need to recapture some margin, so to speak, we certainly have the option to revisit this issue.

But at this point in time, we're simply making the assumption that if there's a bypass or some mechanism that would fail the containment integrity, okay, we're just at this point in time just classifying that as a large release just as a modeling simplification.

There is one exception, which Etienne will get to with regard to a dropped module case, but just to avoid having to do all the calculations for all the different action scenarios that we do have in the PRA, we're simply making the simplification that if it's a bypass or a failure of containment isolation, okay --

DR. SCHULTZ: Uh-huh.

MR. GALYEAN: -- that it automatically goes to a large release. Okay? If the containment is nominally intact, okay, it's not a large release and that's as far as, you know, that's the distinction we're making at this point in time.

DR. SCHULTZ: All right. Thank you.

MR. MULLIN: And to add to that, when it comes to actually evaluating offsite consequences, yet, for the environmental report for emergency planning zone purposes, we are evaluating the actual offset consequences of the accident. We're not just binning them as large release versus not large release.

DR. SCHULTZ: Okay. That gets more to my point. Thank you.

MEMBER BLEY: You'll explain that in more detail later? Or if not, please do a little bit now.

MR. MULLIN: It might come up in conversation. I don't have the slide --

MEMBER BLEY: Okay.

(Simultaneous speaking.)

MEMBER BLEY: It's up in conversation now.

MR. MULLIN: Yes.

MEMBER BLEY: Okay. You say you -- I'm not sure what your statement meant. For every accident, you consider a release fraction or something? What are you doing?

MR. GALYEAN: Yes. We're straying out of the Chapter 19.2 here.

MEMBER BLEY: Not -- where does it show up?

MR. GALYEAN: Etienne talked about the environmental report. Okay? That's a separate application --

MEMBER BLEY: Yes. It is.

MR. GALYEAN: -- it's a separate --

(Simultaneous speaking.)

MEMBER BLEY: We don't review that.

MR. GALYEAN: Okay. And in the environmental report, we do what you call a SAMDA analysis report, you know.

MEMBER BLEY: Right.

MR. GALYEAN: And there, we do actually do detailed dose calculations in order to support the dose benefit -- the cost benefit analysis as part of the SAMDA analysis.

But that's not -- we're not going to talk about here.

MEMBER BLEY: So until --

MR. GALYEAN: It's not a Chapter 19 --

(Simultaneous speaking.)

MEMBER BLEY: So the person who buys one of these from you gets the fuel load. You won't have a Level 3 PRA.

MR. GALYEAN: Right. Yes. Right now, again, we did the calculation for the environmental report.

MEMBER BLEY: Okay.

MR. GALYEAN: That's not what we're going to talk about here.

MEMBER BLEY: Okay.

DR. SCHULTZ: But the second piece, then, Etienne mentioned was the offsite release, the site boundary determination and so forth and that does tie into an open item that the staff has related to the general offsite release calculation. Isn't that right?

PARTICIPANT: Is it?

MS. BRISTOL: Just probably the --

MR. GALYEAN: I'm sorry. Can you repeat the question?

DR. SCHULTZ: The general evaluation of offsite releases and site boundary calculations or determination has to do with the source term -- the general source term documentation that is under review by the staff.

MS. BRISTOL: That's correct.

DR. SCHULTZ: And that piece is an open item that the staff has tied to 19.2.

MS. BRISTOL: That's correct.

DR. SCHULTZ: So what -- hopefully we'll

get to that later being a different meeting, but it does tie in here. So we need to pay attention to that.

MEMBER BLEY: That's set somewhere in the summer.

DR. SCHULTZ: I'm sorry?

MEMBER BLEY: That's sometime in the summer we're supposed to --

CO-CHAIR CORRADINI: Well I think what Steve said is accurate. Staff has received the topical report from NuScale I think revision three --

MEMBER BLEY: Three.

CO-CHAIR CORRADINI: -- or four I think. Yes. Three. And staff is still early in the game of reviewing it. We probably shan't see it until fall.

MEMBER BLEY: Fall. Okay.

MR. RAD: So to be clear, the accident short term topical report is within the scope of the DCA. Emergency planning zone determination is not within the scope of the DCA.

MEMBER BLEY: Right.

MR. RAD: There is a topical report under review, but that is for a COL application stage.

CO-CHAIR CORRADINI: But we're going to -- but to answer Dennis' question, we're going to get the chance to look at that topical report once staff has done its review and issued its questions to the applicant. But the estimate is fall.

PARTICIPANT: Fall.

CO-CHAIR CORRADINI: For the source term.

PARTICIPANT: Yes.

CO-CHAIR CORRADINI: And by then, we'll have our full compliment of folks.

MR. MULLIN: All right. I think we can move on to the next slide. So to kind of support all of our severe accident evaluations, we use the severe accident code MELCOR.

And what MELCOR kind of does for us is it provides the best estimate evaluation of a severe accident progression.

And it also kind of gives us an evaluation of the severity of these severe accident challenges to containment.

These challenges that I listed off -- and MELCOR can evaluate hydrogen combustion, obviously, over pressurization, in-vessel retention, things like that, at least to some degree.

But on top of that, of course, we perform standalone conservative evaluations for all these severe accident challenges to evaluate their credibility or their possibility.

And we use MELCOR to kind of support those evaluations. It provides a physical basis for timing of these events.

It'll give us an idea of how much relocated material we might expect, what kind of materials will relocate, what pressures are as a function of time, temperatures throughout the system, how much hydrogen is produced is an important one.

And these conservative severe accident challenge evaluations all state -- they don't just use kind of our best estimate result from one MELCOR run.

We use kind of limiting values from database simulations and these simulations themselves involve a number of bounding conservative simplifications because we don't want to run at beginning of cycle, middle of cycle, end of cycle.

We've always just run everything at end of cycle to get the maximum decay heat load. We tend not to credit DHRS.

Even though if we did, these events would be -- well if DHRS weren't to work, the probability of these sequences we're evaluating would be below 10 to the negative 15, 10 to the negative 16.

But we assume they don't work and as a

result, their accidents are a lot faster than they really would be.

On top of that, our kind of conservative severe action challenge evaluations, we consider parameters to bound even what we see in MELCOR entirely.

For example, for in-vessel retention, which we'll discuss shortly, we assume the full core relocates as soon as possible, essentially. We assume the whole core is molten.

It's pure UO2. There's no fill-in materials, no steel, no zirconium to dilute the heat flux. We don't credit water in the lower plenum at the time of relocation for in-vessel retention, for example.

So going to the next slide, talk about how we developed the model. I think yesterday we talked about NRELAP5. We used that code to evaluate our success criteria for the Level 1.

It's a more rigorous thermal-hydraulic code than MELCOR. It's validated to test facilities for the NuScale design.

Therefore, part of our goal in MELCOR model development is to match the NRELAP5 characteristics up until core damage, at which point

NRELAP5 can no longer calculate things.

So because of that, we kind of mimic the NRELAP5 model. We've matched all the elevations, the volumes, the flow areas, all of these kind of inputs.

And then we do a pretty rigorous benchmarking of a number of accident sequences up to the point of core damage to show the excellent agreement for the most part.

And then, we move into severe accident space where, of course, we enter a world of uncertainty, but we ensure that we're doing the best we can by modeling appropriate and accurate design characteristics of the NPM and that's -- you get an accurate decay power curve, and all the masses in the core in the right places and locations.

We have the radio and inflight inventory correct from origin calculations and we incorporate modeling best practices that have been around for a while.

We talked to MELCOR co-development staff. We have a good relationship with Sandia National Laboratories. We work with subject matter experts in the industry.

We followed rigorously guidance from state-of-the-art reactor consequence analyses reports.

And, of course, we use the guides and manuals available to us.

DR. SCHULTZ: Etienne --

PARTICIPANT: I'm a --

DR. SCHULTZ: -- does this mean that your modeling, your nodalization that you've developed for your MELCOR model is the same as it is for NRELAP?

MR. MULLIN: I mean, we might have some nodes that are bigger. For example, I mean, MELCOR and RELAP function quite differently in terms of how they handle thermal-hydraulics such that you want different node sizes, essentially.

You can't just copy them one for one. Mainly, our MELCOR would run really slowly if we did that.

DR. SCHULTZ: And the thermal-hydraulic performance -- as you said that you got excellent agreement between your previous evaluations using NRELAP and MELCOR --

MR. MULLIN: Yes.

DR. SCHULTZ: -- convincing yourself that the thermal-hydraulics, which are different in this design than for what MELCOR has been applied to previously. That the modeling is satisfactory for MELCOR, particularly.

MR. MULLIN: Well, I mean, from my point of view, the NRELAP5 code for the new CL design, the NRELAP5 model we have built for it, that came from our safety analysis group that does Chapter 15 analysis, NQA-1 safety related.

It's validated to, you know, we have a test facility it's validated to. You know, I take that as the best I can get as an indication of how our plant's going to perform.

So if I can match that -- those simulations, I think that's the best I can do.

DR. SCHULTZ: Okay. And in matching that with MELCOR, your nodalization is the same as or similar to -- the models that you've developed for MELCOR are similar to it, what you've done, what has been done for -- under the QA program for --

MR. MULLIN: Yes. For the most part. RELAP?

DR. SCHULTZ: -- for the most part?

MR. MULLIN: I might combine two RELAP nodes into one MELCOR node, for example.

DR. SCHULTZ: Are there sensitivity -- nodalization sensitivity studies that you've done? The staff has done one we're going to talk about. That is to say --

MR. MULLIN: Okay.

DR. SCHULTZ: -- they've got a different nodalization with the MELCOR work that they've done. We'll talk about that later, but have you done any nodalization studies, given that you're working with a different sized machine than what MELCOR has been applied to before?

MR. MULLIN: I mean, yes. We've done that over time. I mean, I've built this model in a number of iterations and I've definitely played with nodalization size in various areas. But that's not necessarily been a formalized process.

DR. SCHULTZ: Okay.

MR. BLEY: I'm a little concerned on two points. And I haven't thought of a way out of the things I'm concerned about.

One is given the set of assumptions you made about what you don't consider, this is not in the spirit of a PRA model, except you'd want to look at cases where those things actually were failed and couldn't work.

The reason -- I'm guessing. I don't know. The reason -- if you included those, the results are so low in frequency have to do a lot with the low frequencies we calculate out of the Level 1 PRA of getting into situations where you'd have this problem.

I wonder, don't have a clue, if you did your calculations just conditional on getting into a core damage situation or a challenge situation, your results would look lower or kind of similar to existing reactors.

That's one I worry about. The thing about all these conservatisms in a PRA, you almost want to have a stamp on results saying that.

Because if somebody finds something that goes the other way, they're going to pretend like everything's modeled perfectly and now one thing's worse in calculating high numbers.

So that worries me. The other thing is close to what Steve was talking about and I don't know MELCOR at all.

I've never worked with it. I kind of know what sort of things it's supposed to do. If there's any reason why the assumptions built into MELCOR are not appropriate to this machine, it could change your results and I'm -- have you really studied that?

Is there anything built into the assumptions that are inside of MELCOR that you might not meet?

MR. GALYEAN: Dennis, by the way Bill --

this is Bill Galyean speaking.

MEMBER BLEY: For the front row, you don't have to announce yourself, because your name tag is --

MR. GALYEAN: I want to clarify something you said early on about things we didn't consider. We considered everything pretty much.

I mean, we didn't just dismiss those severe accident challenges that Etienne talked about at the beginning. We analyzed it.

MEMBER BLEY: Uh-huh.

MR. GALYEAN: Okay? And we found that they were simply not credible failure mechanisms for our design. Okay? But we do have analyses to support those conclusions.

Also, with regard to, you know, MELCOR, we worked very closely with Sandia National Laboratories.

MEMBER BLEY: Uh-huh.

MR. GALYEAN: Okay? To modify MELCOR, to accommodate the NuScale design. We paid -- we ended up paying Sandia over a million dollars, okay, for them to modify MELCOR to accommodate our design.

And, again, we worked closely with them. They added some models --

MEMBER BLEY: Yes.

MR. GALYEAN: -- you know, to accommodate the second lower head. They added some models to accommodate our decay heat removal system better.

So there has been a -- there's been a lot of time, effort and money spent, okay, to ensure that MELCOR is, in fact, a high fidelity representation of the NuScale design.

MEMBER BLEY: Yes. Etienne hinted at it or just barely mentioned it. How much benchmarking against your experiments was done?

MR. GALYEAN: You mean the test facility?

MEMBER BLEY: Yes.

MR. GALYEAN: Well, again, the benchmarking was against the NRELAP5 model and results. Okay?

So as Etienne said, NRELAP5 is an NQA-1 code developed by our safety analysis folks. And it is -- it has been benchmarked against the test facility, okay, very closely and --

(Simultaneous speaking.)

MEMBER BLEY: The one code's --

MR. GALYEAN: That's right.

MEMBER BLEY: -- benchmarked against tests and the other one matches. Yes.

MR. GALYEAN: That's exactly right.

MEMBER BLEY: Okay.

MR. GALYEAN: And MELCOR was benchmark against the NRELAP5 code.

MEMBER BLEY: Thank you.

DR. SCHULTZ: Of the MELCOR experts that you worked with in the development and application to the NuScale design, have they been involved in looking at your current results?

The results that you've generated in the MELCOR analyses? Have they still -- have they been involved throughout the process to the end point here?

MR. GALYEAN: I mean, we've hired contractors, consultants that have worked with us for quite a while and have been with us as we develop these results.

DR. SCHULTZ: Okay.

MR. GALYEAN: Yes.

DR. SCHULTZ: Doing the work. So it's continued so that the results that you have achieved with the models that Sandia worked on, have gotten some review?

The way you described it, this is not a QA program that -- this is not under the QA program. So you'd want to have expertise that are -- have done the development also reviewing the results that have been achieved with the models?

MR. GALYEAN: Yes. And I'm sure we'll hear about it later. The NRC has performed an independent assessment.

DR. SCHULTZ: Okay. As they should.

MR. GALYEAN: As they should.

DR. SCHULTZ: Thank you. We'll get into that, too. Thank you.

MR. MULLIN: So let's move on and talk about in-vessel retention and I expect we'll talk about this a fair amount today. This is kind of key to our severe accident management strategy, I guess.

So for the purpose of the rest of this discussion on severe accidents, we're really only going to talk about events where the containment successfully isolated.

We're not talking about bypass events. So for an accident -- for a severe accident, a core damage event in which you successfully isolate the containment, all of the water that leave your RPV, which is necessary to achieve core damage, is going to be in the containment, submerging the RPV.

And we performed conservative analysis to demonstrate that upon a core damage event and a relocation of that core into the lower plenum of the RPV, it will remain there.

And this is based off essentially an evaluation of the heat flux imposed, the maximum heat flux that can be imposed by the core debris on the lower head compared to the maximum or the critical heat flux or maximum amount of heat that you can remove without -- via boiling on the outside of the lower head.

And essentially, we maximize the heat generation rate. We used a bunch of conservative assumptions. We have the full core. The quickest possible relocation times. We have the highest decay heat, end of cycle.

We don't create any upward radiation heat losses. We kind of assumed the worst possible convection situation inside there --

MEMBER BLEY: You're looking at --

MR. MULLIN: -- the core debris.

MEMBER BLEY: -- heat transfer and you're looking at overheating the bottom head. What about when you overheat that whole lower area and it expands?

You look at the stresses on the vessel from that consideration as well?

MR. MULLIN: The expansion of the lower head --

MEMBER BLEY: Yes.

MR. MULLIN: -- and the strains on that? We did do a --

MEMBER BLEY: Connection to the cylinder, you know, blowing up.

MR. MULLIN: We did a structural evaluation as well.

MEMBER BLEY: You did?

MR. MULLIN: Yes. But we focus mostly on this critical heat flux evaluation, which has been done in the past, especially with AP1000 evaluation of in-vessel retention.

MEMBER BLEY: Just --

MR. MULLIN: Go ahead.

MEMBER BLEY: I assume all of those calculations are in engineering reports --

(Simultaneous speaking.)

MR. MULLIN: Absolutely. Yes.

MEMBER BLEY: -- you're don't see.

MR. MULLIN: So we've kind of maximized the amount of heat that you can put into this lower head from the core.

And then, we applied a critical heat flux criterion based on experiments that have been done in the past on hemispheres that do not credit the kind of the high pressure that you're going to have in the containment and the large subcooling that you're going to have in the containment.

So this is one thing that's pretty unique about the NuScale design is that when you have a severe accident, you oxidize all your cladding and a lot of steel.

You're going to generate a bunch of hydrogen. It's going to go into the containment and you're going to get up to one or two -- 250 psi in the containment during severe accident.

That's well below our maximum design pressure of 1,000. I don't know if that's proprietary. And as a result, you're going to get a lot of subcooling in the pool in the NCV.

You're going to have about 100 degrees Kelvin subcooling. So you can actually remove a lot of heat from the lower head without even boiling water.

And when you do start boiling water, you can really boil a lot before you reach the critical heat flux.

However, we didn't credit any of this in our CHF. We just assumed the saturated atmospheric pressure, CHF from experiments, and we still met that with substantial margin.

And, therefore, we concluded that failure of the lower head is unlikely. And, therefore, the accident is essentially stabilized in the RPV.

CO-CHAIR SUNSERI: Unlikely from the standpoint of the analysis is probability zero?

MR. MULLIN: We said it does not occur. Yes.

CO-CHAIR SUNSERI: And these assumptions about the heat that you're talking about, I mean, when you consider that in light of industry standards, this is an extremely conservative approach. Is that a fair statement?

MR. MULLIN: Yes. Yes. And I'd like to kind of talk about why we thought we could do that. You know, our core is a lot smaller than a traditional core.

It's, you know, five percent the size and so that takes advantage of, you know, this cubed square flaw. So for the amount of heat generation you have for a smaller core, you're going to have more surface area.

Furthermore, our power density is less than half of a traditional core and then, we have about five or six times as much primary coolant per unit power.

So it slows down our accidents and we have all this hydrogen in containment that really pressurizes that water and subcools it.

And for all these reasons, we have, you know, it's our opinion that we have quite a lot of margin to CHF and we can conclude that it doesn't occur.

So given that we've stabilized the accident in the RPV, the remaining severe accident challenges, by our analyses, by our evaluation, do not challenge the containment.

Mainly, hydrogen combustion really cannot occur, because of limited oxygen that we've evacuated the containment prior to this even starting to occur.

The containment is just going to be pure hydrogen. So it's hydrogen inerted. Fuel core interactions or steam explosions in the vessel are unlikely.

And if they do occur, are not really energetic enough to induce this supposed alpha mode failure of the containment.

That is, a violent steam explosion would send a slug of water up to the upper head of the RPV and then launches it into the top of the containment, penetrates it.

It's already kind of been dismissed as resolved from the risk perspective by the industry in the '90s, but it still kind of lingers in discussions.

The reasons why it's even less of an issue for our design is we have a small core, lower debris temperatures.

We have a small drop height. So you can't premix as well. It's a shallow pool and relatively high system pressure, which seems to stabilize the steam explosion phenomena.

Furthermore, the big risk in conventional containments is containment over pressure, gradual over pressure. That just can't occur for our design. We have a high-pressure steel CNV.

It's designed for a very limiting LOCA blowdown, which greatly exceeds the maximum pressures we'll see in a severe accident. The CNV is submerged at all times.

In the ultimate heat sink, it provides effective pressure suppression and there's no concrete in the containment to produce more non-condensable gases from a core concrete interaction.

Want to move on? However, an acknowledgement of large uncertainties in severe accident, if the RPV were to fail, if we were to melt through the RPV and corium were to come into the pool in the containment vessel, these various ex-vessel phenomena that are evaluated in conventional designs, are really not credible for our design either.

High pressure melt ejection, which requires a large pressure differential between the reactor vessel and the containment vessel can't occur because we don't have that pressure differential.

Even if the break in the containment is very small and restrictive, just having the water touching the RPV will cool you down and you'll equalize pressures very quickly.

Energetic ex-vessel steam explosions are not likely for similar reasons as in-vessel FCI.

MEMBER BLEY: Can you explain that one a bit? I'm just remembering Hans Fauske's experiments and seeing him poor stuff together and blow up. So why it is the ex-vessel steam explosion can occur?

MR. MULLIN: Well it's a difficult problem not to occur, really in any situation. Maybe assume a hole in the RPV or, you know, corium starting to come out.

It doesn't have enough distance to drop, to pre-mix into little droplets and then trigger and then a shockwave comes up as the steam explosion.

MEMBER BLEY: So it's kind of more of a surface --

MR. MULLIN: It kind of --

MEMBER BLEY: -- heat transfer phenomena in that physical explosive?

MR. MULLIN: Well it's, I mean, a steam explosion occurs as you introduce molten -- very high temperature of molten material into -- you pour it kind of violently into a pool. It hits the pool.

It distributes into little droplets and then, you need a triggering event, which has been hard to identify in a lot of experiments and they usually apply a triggering event that applies a little shockwave.

And then, it collapses all the droplets, which have a little vapor blanket around them. It collapses the vapor blanket.

The droplets solidify and they fragment and they transfer a lot of heat very rapidly to the pool.

And that's a propagating shockwave that sends a slug of water up or just provides an impulse on the sides of the vessel. And we don't really have the geometry for that.

MEMBER BLEY: Okay. Have to think about that a little more, but the size of the pool, yes, you won't get that concentrated slug --

MR. MULLIN: Yes.

MEMBER BLEY: -- you see in the other case. Bill looks like he wants to say something.

MR. GALYEAN: Yes. Again, we're talking -- Etienne talks about dropping it into the pool, into the containment. Right?

I mean, and the volume in the containment that is between the bottom of the reactor pressure vessel lower head and the bottom of the containment lower head, is very limited.

Okay? And so there's just not a lot of volume there to accommodate or to allow this, you know, rapid energy transfer that Etienne's talking about, that would spawn an energetic FCI.

MEMBER BROWN: What FCI?

MR. MULLIN: Fuel coolant interaction or steam explosion. So, you know, further, if you relocate all your core debris into the containment lower head, mind you, you've got water on top of it now and it's poured into water and it's probably kind of plugged up the hold a little bit.

Ignoring all that and you've got a pool of debris in the CNV lower head. For the same reasons that we don't think you melt through the RPV lower head, you won't melt through the CNV lower head.

However, even if that were to fail, you would need to -- the release would need to pass through 60 feet of water and, therefore, it's a small release.

As we've -- to demonstrate our -- as we discussed in the context of drop module, even if you have a release at the bottom of the pool, it cannot be a large release because of the pool scrubbing effect.

And further, even if this upper CNV were to fail from a steam explosion type event --

MEMBER BROWN: Yes. If it goes through the containment vessel, well that just disburses it to the entire pool. Doesn't it?

MR. MULLIN: Yes.

MEMBER BROWN: So, I mean, you effectively contaminate 12 modules if that's what you've done. Don't --

MR. MULLIN: You contaminate the pool, certainly.

MEMBER BROWN: Okay. And there's no chance that based on the volume of water that you could have a severe steam energetic reaction where you could cross it up through the top or anything like that?

I mean, I'm not a thermal-hydraulic guy. I'm just a plain old engineer. So just --

(Laughter.)

MR. MULLIN: No. I don't --

MEMBER BROWN: There's a heck of a lot of energy in that hot stuff coming out through whatever is does and initially it's fairly confined as it goes through from the reactor to the pressure vessel down into the containment.

Then, it leaks through that. A little bit of it then all of a sudden hits it and it just goes. Just seems like that very high temperature fuel and has anybody modeled that to see what that reaction would be?

MR. GALYEAN: I understand that we are being, I want to say wildly speculative here. In the vast majority of core damage accident scenarios that we have modeled using MELCOR and, I mean, the vast majorities, we do not even see fuel temperatures reaching melting temperatures.

Okay? We see core relocation by virtue of failure of the support structures, but we almost never see temperatures reaching melting temperatures --

(Simultaneous speaking.)

MEMBER BROWN: Where you could melt the RPV itself?

MR. GALYEAN: Exactly. Exactly. I mean, we have seen in very limited extreme sensitivity studies, okay, if we tailor the assumptions and the input parameters, yes.

We can achieve melting temperatures of the fuel, but these are very few, very far between, very extreme conditions and so for the sequences and scenarios that we model in the PRA, we simply don't see melting temperatures occurring.

So we're, like I said, we're straying in very extreme speculative space here. So --

MEMBER BROWN: Extremes do happen.

DR. SCHULTZ: So, Bill, that's another way to consider uncertainty. In other words, you -- what you're indicating is that this is a qualitative way to drop down through all the possibilities.

If this, if this, if this, then -- that's one way to handle uncertainties. The other way as you described is you've done substantial number of evaluations, uncertainty evaluations where you've pushed the limit, if you will, pushed toward the limit and found acceptable results.

MEMBER BALLINGER: So what's the distance between the bottom of the pressure vessel and the containment?

MR. MULLIN: Something, like, less than a foot.

MEMBER BALLINGER: Less than a foot? Is there a way, again, being wildly speculative, for creep deformation to close that gap without failing the vessel?

MR. MULLIN: I don't -- I can't answer that. I mean, I would expect if we were actually going to fail the RPV lower head, it would probably swell to maybe come down and meet the containment lower head. I don't really know.

MEMBER BLEY: Well, apparently, he said there's a structural report and I don't know if the staff reviewed that an article last time I got it.

MR. MULLIN: It was part of the evaluation of in-vessel retention. So as I was saying, even if the upper CNV were to fail, from a steam explosion type event, at that time and somewhat later in the accident sequence, your radionuclides have been released and have deposited largely.

Even if you failed the upper CNV in this hypothetical event, it wouldn't result in a large release by our calculations.

At least the instantaneous release of everything airborne in there would not constitute a large release.

So that's kind of it. Our Level 2 insights are this, the core damage events, they're stabilized within the RPV, by our analyses.

And the remaining severe accident phenomena do not challenge the CNV integrity. However, even if we were to fail the RPV and the CNV, the result is not going to be a large release, which confirms our Level 2 event trees.

Thereby, the large release frequency is dominated by containment bypass events. We have a list of kind of key insights here.

There's a number of them. I'm not going to read everything here, but the first one is containment isolation.

As we've discussed before, these containment isolation valves are very important. Containment isolation as a function is very important from a Level 2 standpoint.

As demonstrated in our event trees, if we successfully isolate containment, we don't have a large release.

Furthermore, if we successfully isolate containment, we generally -- we often avoid core damage events altogether by cutting off the flow of the release -- or of the -- with the LOCA outside containment.

You can go to the next slide. We have very good heat removal from the RPV, passive heat removal. We have no insulating material from the RPV because we evacuate the containment.

As a result, for all of our events, which had successful containment isolation, we have water in the containment and we completely submerge the RPV, which helps stabilize all of our severe accidents in the RPV.

To talk about our kind of severe accident challenges specifically, we can't over-pressurize the containment.

Even if we were to assume a RPV over-pressurization event where we fail the DHRS, we fail our RSVs, get up to this ridiculously high RPV pressure and fail it into containment, we've analyzed that.

And even that won't result in a failure of the containment once you equalize the pressures. Hydrogen combustion, it doesn't occur. We don't have enough oxygen in the containment.

In-vessel stream explosions do not occur or do not challenge the containment. The core support plate fails pretty early and for all these reasons we've talked about before.

DR. SCHULTZ: Before you leave this slide, Etienne, the -- one of the staff open items in 19.2 is associated with hydrogen combustion external to the --

MR. MULLIN: Yes.

DR. SCHULTZ: -- to the vessel. You don't mention that here, the staff refers to work that's being -- going to be presented in Chapter 6.

Is that -- I mean, it's still -- that's a piece that's still under review and it's open.

MS. BRISTOL: Yes. We submitted an RAI on that topic and they have an additional supplemental response and reviewing to confirm the results we previously submitted.

CO-CHAIR CORRADINI: But just from a scheduling standpoint, we're not going to hear about that for this letter. Nor are we going to hear about it in June. It's still being evaluated by the staff.

MS. BRISTOL: That's our understanding.

CO-CHAIR CORRADINI: Okay. So that's going to remain an open item coming through --

DR. SCHULTZ: Beyond June. Correct. That's what I wanted.

CO-CHAIR CORRADINI: I wanted to make sure you -- yeah.

DR. SCHULTZ: Okay. That's fine. So both the open items are to be discussed later after now and after June? When I say both, I meant the topical report --

MS. BRISTOL: That's correct.

DR. SCHULTZ: -- coming in November. So --

MS. BRISTOL: Yes.

DR. SCHULTZ: -- that's fine. Thank you.

CO-CHAIR CORRADINI: Where is the bioshield redesign going to appear in second Chapter 3? When we do see the redesign, it'll be in Chapter 3? I'm sorry. I don't remember if it's three or six. I thought three.

DR. SCHULTZ: The staff says six, but --

CO-CHAIR CORRADINI: Says six.

PARTICIPANT: I don't know.

MR. RAD: So I think it affects a couple of areas, actually. So the bioshield redesign will affect discussion in the structural area and I think it also affects the discussion in high energy line break.

PARTICIPANT: Okay.

MR. RAD: But the reality is it'll touch most of the areas where the design of the bioshield is discussed, which are several chapters.

MEMBER SKILLMAN: Etienne, on slide one of five, in the event that the core is not covered, what consideration is given to pyrophoricity?

MR. MULLIN: To what? Sorry.

MEMBER SKILLMAN: Pyrophoricity.

MR. MULLIN: Could you explain that?

MEMBER SKILLMAN: Sure. Core lights itself on fire because of the chemistry of the core materials whether eutectic or whatever, there's sufficient chemical reaction and availability of oxidizing material that the core sets itself on fire.

MR. MULLIN: Well, certainly, oxidation reactions are modeled in our MELCOR simulations. I don't know if that looks like fire, but it releases a lot of heat.

MEMBER REMPE: Am I allowed to make educated just information to help Dick? I know you're coming from the TMI concerns when you mention that, but usually pyrophoric reactions require very small fragmented material. Right?

MEMBER SKILLMAN: I don't know --

MEMBER REMPE: Very small pieces --

MEMBER SKILLMAN: Well just a minute. Right? I don't know. Right or not right. Accurate? We chased this for a good reason.

MEMBER REMPE: The TMI. Yes.

MEMBER SKILLMAN: I'm not trying to be snarly here. Because the design is so different, the core so small, most of the assumptions that you've made regarding core size, DKE, maximizing heat transfer, that type of thing, at least in my mind, makes sense.

So I'm saying what else is out there that might not have been considered and pyrophoricity is something that we gave quite a bit of attention to because of the potential of the added heat that that reaction could contribute.

Particularly when we're headed towards defueling. Dr. Rempe's accurate. It's normally a fine particle interaction where there's moisture, hydrogen and oxygen.

I understand that. But I'm just saying this treatment is so thorough. Is there anything that's not been considered that perhaps is a vulnerability or an uncertainty that might surprise all of us? And pyrophoricity is the one that comes to mind.

MR. MULLIN: Well we have not looked into that but we will. It's pretty interesting.

MEMBER SKILLMAN: Thank you.

MR. MULLIN: Thanks.

PARTICIPANT: Four of five, please.

PARTICIPANT: Okay.

MR. MULLIN: So here we're just kind of talking about some of the other things we've talked about. We skipped the -- let's go to the next one.

I know Sarah yesterday brought up that we have one human action that is risk significant and that shows up in the Level 2.

This is for an event where you have an outside containment break CVCS line break outside containment, you fail to isolate that line.

We can prevent core damage by actuating ECCS and supplying coolant via the CFDS. So the operator action to initiate containment flooding, drain system, emergency coolant injection is -- it comes up as risk significant. Yes.

DR. SCHULTZ: Can you help me understand the last sentence under comment? What is the mathematical limitation that is being described here?

MS. BRISTOL: Yes. Mathematical. I think that was just with respect to Fussell-Vesely being 20 percent of the total large release frequency, not necessarily a limitation.

We provided candidates to the D-RAP panel and so, you know, they could determine if they could -- felt they were risk significant or not.

DR. SCHULTZ: Okay.

MS. BRISTOL: But it was just more, you know, we didn't have human actions we mentioned yesterday for internal events, Level 1, but for Level 2 in order of magnitude a Fussell-Vesely of 20 percent threshold --

DR. SCHULTZ: Okay.

MS. BRISTOL: -- deemed this operator action candidate for consideration.

DR. SCHULTZ: All right.

MEMBER BROWN: In other words, this is a mathematical construct, not a physical construct that you're basing operator action -- satisfactory operation on just because that's the way you've modeled the stuff as opposed to the physical actions?

MS. BRISTOL: Right. Twenty percent of a small large release number.

MEMBER BROWN: Why not 50 percent or ten

percent or five percent?

MS. BRISTOL: As was reviewed in the topical report a couple years ago, we provided --

MEMBER BROWN: I didn't read that.

(Laughter.)

CO-CHAIR CORRADINI: You were here. It was December of '16. We wrote a letter report on the risk significant --

MEMBER BROWN: Okay. All right. I still don't know whether it makes a difference or not. I'd much rather see a physical basis for the operator action being satisfactory based on mathematical construct. That's my opinion.

CO-CHAIR SUNSERI: But in a sense what you're going though, you use the PRA to figure out what the critical steps are and then, the critical steps then have some significance approach it.

And this is one of them and now they know to focus their operator training program. They need to time reactor operator actions, whatever it is to make sure that they do this piece right going forward. Right?

MR. MULLIN: I mean, so that's pretty much it. Here's a list of our COL items, pretty generic.

MEMBER BLEY: Can you go back just a

slide?

MS. BRISTOL: Yes. Overall, they just confirmed that as the design progresses for the combined license phase, we'll continue to look at the PRA and these various inputs into the severe accidents as well as the events we talked about yesterday.

And we'll confirm those for site-specific differences that might arise from a specific site versus the design stage.

MEMBER BLEY: All right. There have been several different approaches proposed in other design certs.

How do you lose the PRA from the designs stage over to the COL stage? You probably say something about that in 19 that I skipped over.

Can you tell me what's going to be required when you actually have a site for the PRA? Will you update the PRA based on site-specific information or do you have some alternative set of criteria that you'll have to meet?

MS. BRISTOL: We -- similarly to DC phase, Reg Guide 1.206 discusses the expectations for PRA application for DC as well as COL and so we'll follow those similar steps and provide a PRA for the site-specific location.

MEMBER BLEY: You will revise the PRA for site-specific location?

MS. BRISTOL: That's correct.

MEMBER BLEY: At the COL stage? That's what I was really asking.

MS. BRISTOL: Correct.

MEMBER BLEY: That's fine.

MS. BRISTOL: And similar to the COL items yesterday, we talked, I know Member Rempe talked about the key assumptions. So that's one example where we're -- we'll verify those.

And to clarify that key assumption, there are the tables in the FSAR. For instance, Level 2 was 191-21 or Level 1 was. Level 2's 28. All of those assumptions will go through at the COL phase.

See if they are any different for a specific versus what we assumed for DCA. We'd have different information for external events or seismic or that that we could incorporate at that time.

If there's testing, if there's additional design information that we didn't have for the DC application that we could apply for the COL application, we would do that.

MEMBER BLEY: Thanks, Sarah.

MS. BRISTOL: And that's the expectation

for a COL PRA.

MEMBER REMPE: So I appreciate you clarifying what you meant by key assumptions. And I was thinking about this.

And so I think I'm allowed to bring this up, but there's a table 19.1-something or other that refers to a lot of assumptions resulting from severe accident phenomena.

And if you actually go through and you have the -- I'm always interested in how the COL applicant will move forward and do they have to justify that many of the assumptions related to severe accident phenomena are indeed as conservative as characterized?

And, again, this is a 19.1 table, not a 19.2 table. So I think I'm legitimate in this because I'm just kind of wondering how -- I'm thinking about ITAAC versus key assumptions.

And some of those key assumptions may be difficult for a COL applicant to say -- they can easily say I have no additional information, because if you don't look, you won't find it.

But I'm wondering if it might be hard to justify you've met that key assumption on some of these cases where -- I mean, as Dennis mentioned earlier, conservative sometimes is in the eye of the beholder and when you're looking at when a release will occur.

And I just am curious about that and how that will progress.

MS. BRISTOL: We're unsure of the actual table, so if can get that table, we can kind of confirm --

(Simultaneous speaking.)

MEMBER REMPE: Okay. I don't have it in front of me now, but I will find it for you.

CO-CHAIR DIMITRIJEVIC: That would be important.

MEMBER REMPE: But it's a table --

PARTICIPANT: And you just sent for it.

MS. BRISTOL: Yes. But it is a table and I'll find it. I had it up earlier. But I had --

MS. BRISTOL: We'll look and get back to you.

MEMBER REMPE: Yes.

MS. BRISTOL: Thank you.

CO-CHAIR SUNSERI: Are you all done with the -- are you done?

DR. SCHULTZ: Well I just have a -- I'm just -- given the comments by Dennis and by Joy, I'm trying to understand what a COL applicant would do under Item 19.2-2?

They've got site-specific information and that's going to cause them to define improvements in the reliability of core and containment heat removal systems.

Can you describe what that -- those conclusions might be? What improvements they might do at a COL stage?

MEMBER BROWN: Does that mean they can change the design?

MS. BRISTOL: No.

MR. RAD: So a COL applicant can absolutely depart from the certified design. That's up to them. Right?

And so if they depart from that design or if they have site-specific information that is outside the assumptions in the PRA as provided in the certified design, then they'll need to provide information in accordance with the COL item.

DR. SCHULTZ: So this is a backstop is what it is?

PARTICIPANT: Exactly.

DR. SCHULTZ: Yes.

MR. RAD: Absolutely.

DR. SCHULTZ: Well it's a handoff of the responsibility of maintaining the PRA from the --

MR. RAD: That's for --

(Simultaneous Speaking.)

DR. SCHULTZ: -- certified design to the operational plan.

MR. RAD: That's correct. That's correct. There are substantial departures that could be taken in theory from the time the design is certified to the time the design is licensed at the application stage.

Both would, of course, involve the NSSS vendor, but that's up to the licensee to make those determinations. Not up to NuScale today.

MEMBER REMPE: And the table of interest is Table 19.1-29, Generic Sources of Level 2 Model Uncertainty.

MS. BRISTOL: Thank you. I'll take a look.

CO-CHAIR SUNSERI: All right. Any members have any other questions for the NuScale team? All right. Well we thank you for your presentation then.

We'll exchange this -- let's see. Yes. We'll go ahead and exchange seats now with the staff and bring up the staff for the 19.2 safety evaluation. Are you going to be --Whenever you're ready, we can go ahead and get started.

MR. CRANSTON: Good morning --

CO-CHAIR SUNSERI: All right.

MR. CRANSTON: -- my name is --

CO-CHAIR SUNSERI: All right. Let's -- we're back in -- go ahead.

MR. CRANSTON: Good morning. My name is Greg Cranston. I'm the senior project manager for the NuScale project and we're continuing today with Chapter 19, as you're aware, with 19.2, .3 and .5.

So at this time, I'd like to just turn over to the staff for presentation. Jason?

MR. SCHAPEROW: Thank you, Greg. My name is Jason Schaperow. I'm a member of the NRC technical staff and I will present the staff's review of severe accident mitigation for the NuScale design. Slide 4, please.

MEMBER BLEY: Jason, do you have any place in the -- I haven't looked at your slides yet, any place in the slides where you going to talk about the audit?

MR. SCHAPEROW: Yes. Actually, the very first slide --

MEMBER BLEY: Okay. Good.

MR. SCHAPEROW: -- actually. Slide 4.

MEMBER BLEY: I won't ask any questions I'll wait for the slide.

MR. SCHAPEROW: Yes. So after hearing the Committee yesterday, I want to try to maybe go a little more detail what we did, certainly in the area of severe accident mitigation.

We began our review in 2017. So roughly two years ago or if we read the FSAR. And then we read a series of detailed engineering reports that NuScale provided.

NuScale provided these reports via website, which we could access from our offices. So basically, we sat for several weeks and we read these reports.

And then we had meetings with NuScale to ask our questions, all as part of this first audit we did in 2017. When we met with NuScale it was -- well we basically went across the street to the NuScale office here in Rockville.

And they had this really nice setup where they -- one wall of the room was basically this big screen and maybe some of the ACRS have experienced this. So basically, you're sitting in a room.

You're not physically there with them, but there's a big screen and you're -- it looks like you're at a big long table and they're at the other end of it. Very impressive. So we asked a lot of questions, whatever questions we had.

So, by going through this reviewing of the reports and asking the questions, that enabled us to kind of come up with what we thought our essential questions were for the review.

And those essential questions we sent to them. The NRC's formal RAI process, which you're probably all familiar with.

MEMBER BLEY: Jason, among those reports, did you folks review the one that was -- talked about -- I think Etienne was talking about, on the structural analysis report on the --

MR. SCHAPEROW: The structural analysis report?

MEMBER BLEY: Yes. He said they're --

MR. SCHAPEROW: For in-vessel retention?

MEMBER BLEY: Yes.

MR. SCHAPEROW: Yes. And I should have -- so --

MEMBER BLEY: If you're going to talk about that separately, that's great.

(Simultaneous speaking.)

MR. SCHAPEROW: I'm not sure I'm give a

lot of detail on what they did exactly for in-vessel retention, because we had some fundamental issues with it. And I'll get into that in a slide or two.

MEMBER BLEY: Okay.

MR. SCHAPEROW: What we actually did do in the area of in-vessel retention.

MEMBER BLEY: Okay. Because I had a question and Ron had one, too. So we'll --

MR. SCHAPEROW: Yes, vice-chairman.

MR. SCHAPEROW: Okay. So we sent them our request for information formally. Roughly, we had one for each issue. For steam explosion, we sent them a request for information.

We did one for in-vessel retention. One for high pressure melt ejection. And we got their responses and -- so we read their responses, but it wasn't quite enough for us.

So we said we'd like to see more, hear more. So we opened up a second audit. This was in 2018 and so we did the same thing.

They put the documents in there that they had developed basically since we asked the questions, these new documents or revised documents.

We read those. We sat down. We met with them again and, you know, we -- at that point, we had pretty much had all our questions answered.

There may have been one or two REIs that were not completely answered by the end of the audit. I think maybe Dr. Bley noticed that.

MEMBER BLEY: Yes.

(Simultaneous speaking.)

MR. CRANSTON: There were two of them. The one on --

MR. SCHAPEROW: The reason -- yes.

(Simultaneous speaking.)

MR. CRANSTON: -- retention, the one on high pressure --

(Simultaneous speaking.)

MR. SCHAPEROW: So the reason was the audit got started before we had finished reviewing all the REI responses. So we weren't able to finish going through those as part of the audit.

So we actually resolved -- finished discussing those with NuScale as -- because the audit was limited. Like, it was, like, a month or a month and a half.

So that ended and then -- so we had a couple of public meetings and closed meetings with them afterwards, more formal, where we discussed their -- some outstanding questions we had on those two

RAIs.

Okay. So another thing that we did as part of our review at severe accident mitigation was we did perform confirmatory analyses. We did MELCOR analyses as we do typically do for each of the design centers.

We also did some in-vessel retention analyses and steam explosion analyses as we did for AP1000 because we had some -- we actually had some very similar issues with NuScale's in-vessel retention analysis as we did with AP1000's.

And, again, I'll go through those as we go through the slide presentation. Slide 5, please. So, of course, any NRC review is guided by the NRC staff guidance.

We have an SRP section that we follow. There really isn't much in there on severe accidents. 19.0 is really aimed at the Level 1 PRA and what 19.0 does is it points to these two SECY papers from the early '90s that lay out the basis for NRC's review of new reactor designs.

This is all in the timeframe of CE80 plus, AP600, ABWR, when we were getting the first new reactor designs in back -- I guess they're not new anymore, back in the early '90s.

(Laughter.)

MR. SCHAPEROW: Of these the kind of the central thing as far as severe accident mitigation goes, is containment performance. That's what we focus on.

And those SECYs and the associated SRMs that lay out the containment performance goals and I will talk about we tried to apply those to NuScale was hard, because NuScale's not exactly like the other designs.

Anyway. Slide 6, please. Okay. So we started out we reviewed the NuScale design, as we typically do, to confirm that they meet the containment performance goals for new reactors.

And I list the goals here. There's two goals. These goals were put into place because the Commission wanted something for mitigation of severe accidents.

The safety goals, the CDF, the LRF, those could all be met on basis of a low CDF. If you have a very good design, you prevent core damage, you can meet all the safety goals.

So we just -- NRC decided we needed something else. So to -- for as a backstop you might say or if you did have a core damage accident, you have to show your containment is robust.

So there were two goals. There was a probabilistic goal and a deterministic goal. And these have been used again through all the design centers.

The probabilistic goal is that if there was a core damage accident, you have to show that the likelihood of containment failure, where bypass is less than .1.

So that's our definition of a robust containment from a probabilistic point of view. From a deterministic point of view, you have to show -- the safety analysis report has to show that the containment will stay intact for 24 hours.

So then they had this other provision, which is what about after 24 hours? Because after 24 hours, the guidance is that it must continue to provide a barrier against uncontrolled release.

So we puzzled -- I puzzled over this myself when I first came from research over to NRO to work on new reactor reviews and it turns out that this was -- this, at least seems to me, I haven't found anything -- absolute documentation, but this is about the ABWR and the event system that it had that after a certain amount of time, the event could lift and you could have a release.

Now that's maybe more of a controlled release, because you have containment venting.

CO-CHAIR CORRADINI: Just for background, did AP1000 address it? Which of the two is that AP1000 address the possibility here for the goal? I can't remember.

MR. SCHAPEROW: So AP1000, to my recollection, did -- they satisfied both of the goals.

CO-CHAIR CORRADINI: They don't have to satisfy both. It's one or the other?

MR. SCHAPEROW: It's both.

PARTICIPANT: Both.

MR. SCHAPEROW: It's both.

PARTICIPANT: Both.

MR. SCHAPEROW: It's both goals. Yes. There actually was -- when we sent the Commission Paper up, the Commissioner's like, well do you mean one or both goals?

So actually, when they wrote back to us in the SRM, we said we talked to the staff and we understand the staff meets both goals. So they have to show both.

And again, a lot of this was, you know, it's post CMI, you know, post all the analysis we had from Mark 1s, you know, looking at all the PRAs and all. What do we want for new reactors?

We'd like to be more robust than the operating reactors and this is what we came up with. And again, this is for -- been applied to all the different reactors.

And I -- this worked out well. We think it's provided a reasonable metric for a safety performance beyond core damage.

So we looked at NuScale's in-vessel retention analysis as Etienne described in some detail here and in previous meetings with ACRS.

We had issues with phenomenological uncertainties. We went back, we took a look at what we had done for AP1000 and, you know, we still had kind of the same uncertainties.

One uncertainty deals with this potential formation of a metallic layer inside of the core debris either on top of it or maybe in the middle somewhere.

And focusing -- conducting the heat outward to the walls of the vessel and really focusing the heat on a very, very small area of the vessel all the way around, like, a little belt.

Another uncertainty involves intermetallic reactions. There are some intermetallic reactions of zirconium. Molten zirconium can react with steel, forming -- and it can give off heat.

It's an excessive thermic reaction there and the third uncertainty dealt with critical heat flux. Now as Etienne pointed out, the experiments that he used were for critical heat flux experiments at atmospheric pressures.

And you can get additional benefit from -- if you're under pressure, you can get better heat transfer. The issue that we had was with geometry.

So to me, the bottom of the containment vessel in NuScale is about a one foot gap. And the tests, I don't think they look quite like that.

I thought it was never a big tank. It was -- there's more room around it. And similarly, under the bottom of the containment, there is a -- there's limited space there under the bottom of the containment as well.

So we had some issues with this. And so if you'll turn to slide 7, this is kind of the punchline.

So we took a look at this again and said, well is there another way we can review the design without just applying the goals as they are stated in the SECY Papers?

And so we came up with alternative criteria. On the probabilistic side, we want to keep that probabilistic goal, but if in-vessel retention is an issue, we're not going to apply this goal.

For the cases where in-vessel retention was an issue, we would apply another goal, no large release, which is really kind of the intent of these containment goals is to provide a large release.

Not that the containment has to stay intact. That's not really -- it's more about keeping the fission products away from the people.

So if we can convince ourselves that the -- we would have a large release, then we could convince ourselves that this goal is satisfied.

We did send up a SECY Paper to the Commission about a week ago on this. It's not been made public.

There's a process where there's about a two-week delay between when the paper's issued and when it's made public for various reasons. I'm not an expert on that.

MEMBER BLEY: But just for clarification, we have a copy of that --

(Simultaneous speaking.)

MR. SCHAPEROW: That's correct. The ACRS -- you were on distribution. We notified ACRS, I think --

MEMBER BLEY: '47, 190047.

MR. SCHAPEROW: Yes. So the ACRS's already -- the Committee already has it, but again, it has to go through this --

MEMBER BLEY: We've all read it. Don't worry.

MR. SCHAPEROW: Bureaucratic process that we have --

MEMBER BLEY: Some of us haven't seen it. Is it in our -- for this meeting?

MR. SCHAPEROW: Yes.

MEMBER BLEY: Okay.

MR. SCHAPEROW: It's not long.

MEMBER BLEY: Missed it.

MR. SCHAPEROW: It's a five-page long paper and I have -- she has a lot of background in it. It goes -- actually goes through a lot of the stuff that Etienne talked about today.

MEMBER BLEY: I think what Jason -- I think it's important the Members read this, because what he's going through briefly here, walks through with background and discussion on how they've come to the thing.

And Vesna asked some questions yesterday, which are directly relevant to how they chose a different approach to largely --

(Simultaneous speaking.)

MR. SCHAPEROW: Yes. I mean, other reactors have the, I believe APR1400 uses CCFP definition. They use -- I think they might have used two.

I know one of them was the large release frequency divided by the CDF. I know one of the two that APR1400 uses that. Again, this whole thing, you know, about large, you know -- so anyway.

So there's a five-page SECY Paper that just went up that provides, again, additional background beyond what I'm going through here in the slides.

Slide 8, please. So anyway. Going back to containment lower head failure, so the possibility that the core debris could leave the bottom of the reactor, going into the containment lower plenum and then, fail the containment lower head, we concluded no large release.

Part of it is due to reactor pool scrubbing, part of it is due to scrub -- to the --

even before the core debris relocates down through these two bottom heads, a lot of the volatile fission products, which are the main risk, the cesium and the iodine, a lot of these have left the fuel and have deposited in the reactor somewhere.

So they're -- even before the bottom of the containment fails, the stuff is -- some of the threat is gone away and, of course, the reactor pool scrubbing helps in addition to that.

With regard to steam explosion in the containment, again, we judge that there's no large release even if there was a steam explosion. Again, due to this deposition of the fission product aerosols in the reactor and the containment.

Slide 9, please. With regard to high pressure melt ejection, we looked at NuScale's MELCOR analyses. They did a wide variety of MELCOR analyses for a wide variety of scenarios.

And we came to the conclusion that there really would not be a significant pressure difference between the reactor and the containment by the time core debris had relocated downward into the lower plenum.

With regard to steam explosion in the reactor vessel, NuScale performed an analysis to show that there was insufficient energy released to fail the containment upper head.

We reviewed NuScale's analysis. We decided that it would be worth our while to perform our own independent assessment.

What we did was we used the approach developed by Theofanous, which is described in a series of papers back from, I believe, the late 1980s.

They're also -- the papers are actually put together in a NUREG. They're actually just one on top of the other, the four papers. They're in NUREG-5030, NUREG/CR-5030.

We wrote a program, a small computer program, using the Theofanous model, together with the inputs for a large LWR. That model is and the inputs are, again, documented in NUREG/CR-5030.

We encoded it and then, we changed some of the inputs to scale it down to NuScale, because NuScale, things are smaller. Geometry's a little different.

And we performed this in-vessel steam explosion analysis using Theofanous method and NuScale's size and we concluded that the energy -- we agreed with NuScale's conclusion that the energy released was insufficient to shoot a slug up through and break the containment upper head.

DR. SCHULTZ: Jason, you -- clearly from what you just described, you concluded that the size features associated with NuScale didn't cause an issue with regard to Theofanous approaches and evaluation either analysis --

MR. SCHAPEROW: Yes.

DR. SCHULTZ: -- or conclusions.

MR. SCHAPEROW: Yes. Well there were some differences. One of the differences in a large light water reactor, there is an additional energy sink, energy loss.

I'm presuming you understand this. I'm sorry. I'm assuming you understand all the -- this particular Theofanous Paper.

So the method is that you -- the core debris falls into the water and releases energy and then as the core goes up through the reactor and -- the chunk of core goes up through the reactor.

It hits different things. Energy is lost at different places along the way. So one of the energy losses for a large LWR is there's a shield plug above the top of the reactor vessel, between the reactor and the containments.

NuScale didn't have that energy loss, but that was one thing we had to -- we changed. Another thing we changed in the analysis was we didn't use the conversion of thermal energy to mechanical energy from the Theofanous step.

We cut that in half because there had been work done since then that shows that the conversion of thermal to mechanical energy would be less.

MEMBER SKILLMAN: Jason, there is a 75-ton shield plug up there. That's a huge massive plug.

PARTICIPANT: That's outside containment.

MR. SCHAPEROW: Outside containment in the --

(Simultaneous speaking.)

MR. SCHAPEROW: -- large LWR, there's a hunk of -- there's a big plug between the reactor and the container. But NuScale, there's really nothing there.

It's an --

(Simultaneous speaking.)

MEMBER SKILLMAN: I understand.

MR. SCHAPEROW: -- there's some piping and things.

MEMBER BLEY: Jason, a quick question. I remember in one of those Theo papers, did you say NUREG/CF?

MR. SCHAPEROW: CR, NUREG/CR-5030.

MEMBER BLEY: I see.

MR. SCHAPEROW: It was a series of four papers that were actually, basically, stapled together and put in a NUREG.

MEMBER BLEY: That's the NUREG.

MR. SCHAPEROW: That's the NUREG. There's --

MEMBER BLEY: Doesn't look like it's on the public website. Okay. Thank you.

MR. SCHAPEROW: Yes. It's pretty old. We'll --

(Simultaneous speaking.)

MEMBER BLEY: All right. Never mind.

MR. SCHAPEROW: Fifty -- that it was a long time ago. But the -- it was a probabilistic analysis with the deterministic stuff woven in.

It was a nice -- that was a very nice piece of work from our -- I wasn't familiar with it actually, until I started working on NuScale, but then I had help from others.

One of the NRC staff members actually worked for Theofanous back then, Hanry Wagage, who was an NRO. He was a post-doc out there at ACSB.

Plus, of course, you know, Hossein Esmaili worked on this with me. He performed the in-vessel steam explosion analysis for AP1000. He's up in research.

So we had a -- I had significant help in this area, because as most of you probably know, my area's more in the fission product area and steam generated tube integrity.

Okay. So bottom line is we performed independent analysis using a different method than NuScale did and we confirmed their conclusion. Slide 10, please.

The topic of severe accident induced steam generated tube failure as I imagine you've heard many presentations on this over the years for different designs.

So NuScale's design, of course, looks quite different than others because everything is in one vessel. You can have a hot leg and a cold leg. So we reviewed their steam generated tube failure analysis.

We asked some questions. They actually -- we did their analysis at one point as a result of our REI.

One issue that came up was -- I realize in an earlier -- I noticed this, too, is that -- so in

NuScale, during a severe accident, the high pressure is on the outside of the tubes because they're primary circuit and primary coolant is on the outside of the tubes and their secondary coolant is on the inside of the tubes.

So an issue came up about there's a statement, I believe it's in the SER that talks about the tubes being less susceptible to failure under compression and under tension.

We didn't really evaluate this. We were satisfied with their analysis on the basis of the thermo-hydraulic conditions just -- it just wasn't that hot and that high pressure.

It was just -- it was more mild thermal-hydraulic conditions. As you can tell from the estimate of the probability of tube ruptures as well.

The other thing that stands out from NuScale as being quite different is that the relief valve that's on their secondary side actually relieves in the containment, which is quite different than large LWRs.

At least the ones that I'm familiar with, the relief valve on the secondary side release into the environment.

So this actually is helpful at least from the point of view of large release frequency because if you have a tube failure, you don't necessarily get a large release.

The tube failure can lead to release into the containment. The only way to get a release outside of the containment for a tube failure, is to have the MSIDs not close or the feed water valves not close.

Slide 11, please. So we also took a look at NuScale's use of the MACCS code as part of their large release evaluation for both 19.1 and 19.2.

One thing we noted was that they're using it at a very closed in distance. This is an issue that's come up, I imagine, at previous ACRS meetings.

I think I may have been here even. The issue is that the modeling is -- MACCS has really been assessed over the years for large LWRs, which maybe they have larger exclusionary boundaries.

Their site is -- the site's bigger. NuScale's site is compact and so the distance from the center of the site and the site battery is only .167 miles.

So we -- again, we asked questions about this and NuScale showed us as part of the audit, second audit, they showed us some comparisons they had done between MACCS and ARCON.

And ARCON is a code which is used for these close in distances. So they showed that in -- for the various cases they ran, that the concentration down wind at .167 miles predicted by MACCS was always higher than the concentrations predicted by ARCON at those distances.

So that satisfied us. Also, as Etienne had described this morning, there are large margins between their large release definition and the doses that they calculated with MACCS.

DR. SCHULTZ: Thank you, Jason. I had a question because the way the SER read, at least the way I read the SER, I was concerned that the staff had done the ARCON evaluation --

MR. SCHAPEROW: No.

DR. SCHULTZ: -- but was the applicant --

MR. SCHAPEROW: Yes.

DR. SCHULTZ: -- demonstrating that for the staff.

MR. SCHAPEROW: Yes.

DR. SCHULTZ: And there was one other issue at about the same place and that was the -- using the large margins. The fact that there are large margins to justify the assumptions here.

If you could take a look at it, because it appeared that that was used to differentiate between the short distances or differences in short distances that were used.

And I think if the right distances aren't used in the calculation, then it ought to be correct, we shouldn't --

MR. SCHAPEROW: The right distances were used. They --

(Simultaneous speaking.)

MR. SCHAPEROW: -- the question is the uncertainty in the prediction of the concentration --

DR. SCHULTZ: Okay.

MR. SCHAPEROW: -- could be -- there could be some uncertainty, but we're not worried about that uncertainty.

That's not an issue for us because, you know, if your predicted dose is one rem and the dose your comparing is 200 rem, we're not concerned about -- that doesn't present an issue for us.

(Simultaneous speaking.)

DR. SCHULTZ: No. That's, again, if the right distances were used, I don't have a concern, but I was --

MR. SCHAPEROW: Yes.

DR. SCHULTZ: -- confused by the way it was written.

MR. SCHAPEROW: No. The correct distances were certainly used.

DR. SCHULTZ: Good. Thank you.

MR. SCHAPEROW: We need to -- maybe we need to clarify the SER.

CO-CHAIR SUNSERI: Yes. Jason, I have a question, too.

MR. SCHAPEROW: Sure.

CO-CHAIR SUNSERI: Since this NuScale definition of large release is kind of a key aspect of the containment acceptability criteria that you mentioned earlier, why did you choose not to independently validate those calculations?

And essentially, it sounds to me like you just accepted a presentation of NuScale's results. Is that -- am I understanding you correctly?

MR. SCHAPEROW: So I guess I'm not sure -- what sort of independent confirmatory analysis are you envisioning?

CO-CHAIR SUNSERI: Well --

MR. SCHAPEROW: I'm not really sure --

CO-CHAIR SUNSERI: -- like, rerun -- run the MACCS calculation using -- inputting your own parameters and assessing kind of independently --

(Simultaneous speaking.)

MR. SCHAPEROW: Yes. We didn't do -- that's correct. We did not do that.

CO-CHAIR SUNSERI: So am I correct in this is a --

MR. SCHAPEROW: I don't know what -- how to answer that other than -- let me think for a second. I didn't identify any issues that would suggest I would need to do something like that.

CO-CHAIR SUNSERI: Okay. So let me back up for a second. Am I correct that the definition of large release is a key criterion as far as assessing the containment acceptability in accordance with the two criterion you have --

(Simultaneous speaking.)

MR. SCHAPEROW: That's correct.

CO-CHAIR SUNSERI: All right.

MR. SCHAPEROW: And this is important.

CO-CHAIR SUNSERI: So that's an important value from a regulatory and safety, public. So what you're saying is we're kind of accepting the NuScale analysis as opposed to independently verifying it?

MR. SCHAPEROW: There was one case --

CO-CHAIR SUNSERI: Am I right?

MR. SCHAPEROW: -- where I did an independent check. It was in the area of if -- I did an independent check on one of NuScale's dose calculations. It was the one for -- during refueling.

If you dropped the upper internals on the core and you had a destructive on the core such that the gap activity was released from each of the fuel rods.

I did a confirmatory calculation on that. I took the NuScale analysis -- and on the NuScale, the one that -- the RADTRAD calculation that was done for the fuel handling accident and I multiplied that times the number of assemblies in the core.

Instead of dropping one assembly, I assumed 37 were dropped and I calculated a dose for that and I got a number that wasn't too different than what NuScale had calculated with MACCS.

But yes. In a sense, I think I was comparing two NuScale calculations together. I believe they were from different code and NuScale could correct me if I'm -- I'm pretty sure that the field handling accident was done with the RADTRAD code.

It's not something we typically do. I don't remember any of our other DCA reviews where we did MACCS calculations for Chapter 19. We have -- we are doing -- I believe we're doing some MACCS confirmatory analysis for the environmental report.

But I really can't speak to that. I don't know if -- well we have our reviewer of the environmental report here in the room if he would like to add something.

MR. PALMROSE: Don Palmrose, senior reactor engineer. I'm the principal technical lead for the environmental review looking at the outside consequences going to the cost benefit analysis.

And we do do our own confirmatory calculations using our own determined staff input values for those things, calculations, to confirm what NuScale has done.

CO-CHAIR SUNSERI: And I presume you didn't see any issues then, with those calculations:

MR. PALMROSE: That review is still under way.

CO-CHAIR SUNSERI: Okay. Thanks.

DR. SCHULTZ: Jason, what about confirmatory calculations related to the accident source term methodology report?

MR. SCHAPEROW: Yes. So first of all, I'll open up broadly with that review's still under way and we just got revision three of the report maybe a month ago.

And we're pouring over it and asking questions, but we have been looking at that report since we got rev one and rev zero.

DR. SCHULTZ: Right.

MR. SCHAPEROW: So from our point of view in our area, the 19.1 and 19.2 and severe accident stuff, we've been looking at their development of source term.

We've been looking at their deposition in the containment. We have done a little bit of independent analysis, which you can find in Chapter 5 of the MELCOR independent confirmatory analysis report, which is cited in --

CO-CHAIR CORRADINI: And we'll talk about that in closed session today.

MR. SCHAPEROW: Okay.

CO-CHAIR CORRADINI: Thank you.

MR. SCHAPEROW: So we do have some information in there, because we, you know, we're very just -- yes. NuScale's unique. It's a different design.

And actually, they're proposing a different source from what everybody else has used for these -- for these, for Chapter 15. So that was of a lot of interest to me, certainly.

Okay. Slide 12, please. Okay. So module drop accidents. We took a look at what NuScale had done for module drop accidents and had basically run MELCOR with the module on its side. And they've done some work.

So we took a look at it and we thought, do we want to build a deck like that? And we said, no. We decided we just didn't.

We thought, well let's focus on pool scrubbing, because we think that's a big deal. So that was what we directed our REI. So we met with them.

We asked questioned. We sent them an REI. We met with them. They sent a response. We met with them. They sent another response. Finally, after a couple of REI responses, we got all our questions answered.

The kind of the crux of the issue for me was that they were using a model for reactor pool scrubbing, which was based on tests done with iodine vapor from back in the early -- this was the paper that goes to the -- they cited a paper from back in 1972.

And, you know, based -- since then, you know, the state of the art is that these releases are aerosols and we made a fuss about that. And so NuScale did some supplemental analysis.

And basically, to demonstrate that the numbers they were using from these earlier studies were okay.

They were conservative as far as decontamination by a large pool of water like this. And slide 13, please.

Finally, as we have done with pretty much each of the DCA reviews, we did an independent confirmatory analysis with MELCOR.

This is one of -- perhaps MELCOR's main -- the main reason we developed it back -- starting way back in the '80s was so we can have a tool for independent confirmatory analysis.

Of course, we've used it for a lot of things since then, a lot of other things beyond confirmatory analysis.

We performed simulations for NuScale for three of the scenarios that NuScale reported in the FSAR. The first scenario was inadvertent opening of a reactor vent valve.

The second was the CVCS line break inside containment. And the third was CVCS break outside containment.

And we also did perform some sensitivity calculations, let's see, as is customary and required, because we learned some things along the way.

The staff simulations, we produced the applicant simulations, but we did see some timing shifts as is, again, not unusual for two different people running a severe accident model.

We do believe that the differences are unlikely to affect the applicant's severe accident mitigation analysis. Slide 14. Thank you.

MS. GRADY: Good morning. This is Anne-Marie -- thank you. Good morning. This is Anne-Marie Grady.

And as Dr. Corradini has already mentioned, in June, we're going to come back to describing our analyses for combustible gas control in Chapter 6 and the exemption requests.

And most of our review on combustible gas control will be discussed then. We'll be discussing the scenarios that were considered, the severe accident scenarios.

We'll be discussing a combustion event in the containment leading to a detonation in the containment, the effect of the detonation event on the containment structure.

Those are all for scenarios that are for intact containments. I'm here just to talk about today one specific event, which is not for intact containment.

And I'm only here discussing it in the context of Chapter 19, because it's going -- it's only being looked at from the point of view not of how it challenged containment integrity, because it doesn't, containment no longer has integrity, but whether or not it would lead to multi-module risk.

Okay. So the scenario that is -- the scenario that was evaluated by NuScale, as a supporting study for their PRA, was CVCS line break outside containment.

It was a multiple failure scenario. The containment -- the reactor tripped. The containment isolated and nothing else worked. No mitigation whatsoever.

No ECCS. No DHRS. No CFDS injection. No operator action. So multiple failure. So that's what they're talking about here.

And in their evaluation, they concluded that within the containment, which is what we look at all the time traditionally, because of 50.44C, but there was no chance of -- there was no hydrogen combustion.

No problem. However, they said, because we now no longer have an intact containment where we have hydrogen leaking out through the broken CVCS line.

And it's now going into the area space under the bioshield and the bioshield has plenty of oxygen.

So they said in their calculation that the conditions for combustion, i.e., the conditions that MELCOR predicts, which are concentrations, hydrogen and oxygen, would support combustion.

And their expectation, this is a qualitative statement now, is that it would be sufficiently low concentration, would get disbursed throughout the reactor building and it would be a non-issue with respect to hydrogen combustion and they stopped there.

So we sent them an RAI on this and asked them to evaluate not just thinking that it was going to be disbursed throughout the reactor building, but evaluate the potential for combustion and/or detonation underneath the space in the bioshield.

And we did a confirmatory -- okay. They responded to that by saying this was a low frequency event, the concentrations would be very low, they didn't see the need to evaluate it further and stopped there.

But we didn't. So we asked our contractor, ERI, to do a confirmatory calculation to see whether or not there would be conditions that would support combustion in the space underneath the bioshield.

They have done that analysis. It's still in draft form. It's almost completely reviewed, but then, NuScale redesigned the bioshield.

So now we had an -- their response as to what took place under the bioshield was based on their old bioshield design.

Now we have a new bioshield design. We have -- there's (Telephonic Interference.) there's a rough --

CO-CHAIR CORRADINI: Somebody is having their speakerphone on. Can you please go to mute, that's on the open line? Thank you.

MS. GRADY: The RAI response that's referenced here, RAI 94.47, is a response that talks about the design, the redesign, the current design today of the bioshield.

It's a proprietary document. So I don't know how much to say. I think I could say at high level, the primary difference is that now the design, in fact, does not have an enclosed space that relied on some action of some panels, but rather, it has fixed vent openings.

That I can say and if we want to talk about the details of it, we can talk about it in closed session.

So that's where we are right now. We are evaluating, using our contractor, whether or not, with the new design, there would -- the hydrogen and oxygen concentration would reach combustion when it's -- if it doesn't, case closed.

If it does, then we have to look to see whether or not it would ever get to the concentration when it's -- of detonation and then, if we have detonation, then we have to look more broadly as to whether or not there could be structural damage such that something could go shooting off and impact one of the other modules.

I'm sure everybody in this room hopes that's not what we're going to find. We're going to be able to screen it out on concentrations, but right now, we are having this evaluated.

We don't have the final response yet and we're looking at it solely from the point of view of multi-module risk. So that's why we're talking about it in this chapter.

And the RAI that's listed here, 94.47, has a lot of the technical details. It doesn't have a drawing of the new bioshield.

However, we have access to that in the electronic reading room and I've seen that and many people -- many reviewers are concerned or interested in the design of the bioshield structurally is, RPAC is, fire protection is.

I mean, many people have looked at it. So it's of great interest and as Zac Rad mentioned, it's discussed in many places in the FSAR.

I don't -- I can't show you a figure if you want to see the new design. But perhaps NuScale could provide one for the closed session.

CO-CHAIR CORRADINI: We can wait, if necessary, until the closed session.

MS. GRADY: Thank you.

CO-CHAIR SUNSERI: Anything else?

MS. GRADY: No.

CO-CHAIR SUNSERI: So your summary?

MS. GRADY: Go ahead.

MR. SCHAPEROW: This is just -- so first one Anne-Marie just talked about, we have -- let me back up. So we have two open ends left, basically, after reviewing 19.2.

The first one is what Anne-Marie talked about with the shield redesign. We want -- we've asked questions about multi-module risk resulting from hydrogen combustion under the shield.

The second one we have not talked about today. This is the -- as part of the accident source from topical report review, NuScale is going to be proposing changes to a Section in 19.2 called Equipment Survivability.

They're going to be giving us new FSAR pages. I think on March -- on May 23rd. So we're going to be getting them, like, in a week or so. So this -- we couldn't close -- as much as we like to close out, we couldn't. So this one is under review and, again, more to come on the accident source term. Okay. That concludes our formal presentation.

CO-CHAIR SUNSERI: All right. Great. Thank you very much. Members, any comments or questions for staff? All right. So it's -- we're way ahead of schedule.

I'll check with -- let me check with NuScale here. Are you guys ready to start the multi-module stuff before lunch?

PARTICIPANT: I think they're looking it over.

PARTICIPANT: NuScale's input.

CO-CHAIR SUNSERI: NuScale Chapter 21. Right? Okay. How about the staff? You guys do your --

MR. CRANSTON: Let's just -- staff is not here and that was going to be after lunch. So --

CO-CHAIR SUNSERI: Well let's back up. So we're now at the stage where we're scheduled for Chapter 21. Does NuScale have presentation for Chapter 21?

PARTICIPANT: Yes.

MS. NORRIS: We will, but they're not -- this is Rebecca Norris of NuScale. We will, but they're not here right now. We are ready for 19.3. If we could do that instead?

CO-CHAIR SUNSERI: Yes. That's fine. So what I would like to do is take a 15-minute break until 25 till and then at 25 till, we convene with whatever we can go with today.

MR. SCHAPEROW: May I interject?

CO-CHAIR SUNSERI: Yes, sir.

MR. SCHAPEROW: I'm sorry. Jason Schaperow. You've mentioned --

CO-CHAIR SUNSERI: Hold on.

CO-CHAIR SUNSERI: Yes, Jason, go ahead.

MR. SCHAPEROW: You mentioned closed session and I'm -- do we -- now I have a break now, because other sections we haven't presented. Do I need to develop something for --

CO-CHAIR SUNSERI: This closed session is scheduled for later this afternoon. So --

MR. SCHAPEROW: If you have certain plans of what you'd like to hear about, that would enable me to --

CO-CHAIR CORRADINI: So I would say for me alone -- we've lost our other curious parties. But for me alone, I think the number of assertions in the MELCOR compilations.

MR. SCHAPEROW: Okay. Thank you.

CO-CHAIR CORRADINI: So for closed session, we want to hear about that at the very least.

MR. SCHAPEROW: All right.

CO-CHAIR CORRADINI: I think Anne-Marie volunteered NuScale who volunteered discussion of the biofield.

(Laughter.)

CO-CHAIR CORRADINI: To the extent that we can understand where they're going with it, that would just be good for the future. But I would think for sure MELCOR.

MR. SCHAPEROW: Thank you.

CO-CHAIR CORRADINI: In the closed session and that's scheduled at the end of the day.

MR. SCHAPEROW: Got it. I'll -- are we going to --

PARTICIPANT: Sorry.

(Simultaneous speaking.)

FEMALE PARTICIPANG: -- question.

CO-CHAIR CORRADINI: They just want to know if we're going to have a closed session. The answer's yes.

MR. SCHAPEROW: Is there anything else you want to hear about at the end of the day? Because I'm going to be out on break for a while because you'll be discussing other sections I wasn't working on.

MEMBER REMPE: So are we, you and me, allowed --

CO-CHAIR CORRADINI: Yes.

MEMBER REMPE: -- to ask questions about --

CO-CHAIR CORRADINI: Yes.

MEMBER REMPE: -- the MELCOR analysis?

CO-CHAIR CORRADINI: Yes.

MEMBER REMPE: Detailed on some of the input assumptions?

CO-CHAIR CORRANDINI: Yes.

MEMBER REMPE: Okay. If I'm allowed to, yes. I'd like to discuss that a lot.

MR. SCHAPEROW: I will be here. Thank you.

CO-CHAIR SUNSERI: All right. So what are we -- all right. I'm only in charge here. So what are we doing?

MR. SCHAPEROW: Break.

PARTICIPANT: Take the break.

CO-CHAIR SUNSERI: What are we doing after break?

MR. SCHAPEROW: They're getting their people here for --

(Simultaneous speaking.)

CO-CHAIR SUNSERI: For 19.3?

(Simultaneous speaking.)

CO-CHAIR SUNSERI: Okay. So at 25 till, we will start with 19.3

PARTICIPANT: They're all it. They're all over on NuScale.

MS. NORRIS: This is Rebecca Norris with NuScale, we have a question. I don't know. Are we still -- okay. We can also do 19.2 closed first, which would be convenient because one of our --

PARTICIPANT: We can't do that.

MS. NORRIS: Just for scheduling?

PARTICIPANT: Are we still on the record?

PARTICIPANT: Well they are discussing schedule. Is it scheduled to be on the record?

CO-CHAIR SUNSERI: No. We're going to go with 19.3 after the break.

MS. NORRIS: Okay.

CO-CHAIR SUNSERI: We're on recess until 25 till.

(Whereupon, the above-entitled matter went off the record at 10:21 a.m. and resumed at 10:40 a.m.)

CO-CHAIR SUNSERI: We're back in session. I apologize for being a little late on that, but we readjusted the schedule. We need a little extra time.

So the way I am told that the actions are going to play out today is, we're going to do 19.3 next, NuScale, followed by staff. Then we'll roll into 21, NuScale and staff. I don't know if we'll get to that before lunch, but it depends on how fast this goes. Then we'll go back to 19.5, NuScale and staff. After that, we'll have the closed session. Everybody on board with that?

All right. So let me turn it over to Rebecca then, for the 19.3.

MS. BRISTOL: Thank you. So again, I'm Rebecca Norris. We still have Zachary Rad, Sarah Bristol, and Bill Galyean up here at the front table. We are going to be presenting on SR Chapter 19.3, as was already stated. This is the Regulatory Treatment of Non-Safety Systems.

In Section 19.3, we evaluated the regulatory treatment of non-safety systems and the various criterion of that, and those from criterion alpha, bravo, charlie, delta, and echo, there were no RTNSS SSCs in the design that met that criteria for the NuScale design.

CO-CHAIR CORRADINI: So I don't -- I want to understand this, because every time we talk about RTNSS I get confused. So let me make sure I've got it right. Is that using the risk-significant measures from your topical report, you've identified things within the PRA that are risk significant. That's correct?

MS. BRISTOL: That's correct.

CO-CHAIR CORRADINI: So what is the screening criteria, then, that it falls into or out of RTNSS, given you had a list on slide -- I've got to find the slide now --

MS. BRISTOL: The candidates from 19.1.

CO-CHAIR CORRADINI: And where you went through in 19.1 -- I'm trying to understand how that list then gets reviewed that either falls into or out of RTNSS. Can you help us, please? Help me?

MS. BRISTOL: Yes. As the RTNSS criteria, charlie, is SSC's functions relied on underpower operation and shutdown conditions to meet condition goals of core damage frequency of less than 1 E minus 4 per reactor year, and large release frequency of less than 1 E minus 6 per year, and this is taking credit for safety-related components.

So on the Level 1 insights and the Level 2 insights we've provided, we had a bullet that said there were no additional components needed to meet those thresholds.

CO-CHAIR CORRADINI: So just walk me through a little slower. So on slide 24, you have an important slide which went through what's important from risk for me from a significance determination.

But because of the definition of what would go into RTNSS, none of those rose to the level that would be RTNSS?

MS. BRISTOL: That is correct. They didn't need the RTNSS criteria. They were still candidates for risk significance, and so as an example, as discussed in FSAR 17.4.3.3, there may be risk- significant components as determined by the D-RAP panel, but they aren't classified as RTNSS components.

They could still be risk significant; for example, a crane is a component that is risk significant, but not RTNSS.

As well as then there could be components that the panel in 17 determined weren't risk significant based on the PRA, but they could still include them in D-RAP.

CO-CHAIR CORRADINI: Okay.

MS. BRISTOL: So for the purposes of RTNSS, it is looking at the components needed to meet the two criteria for core damage frequency 1 E minus 4, and large-release frequency 1 E minus 6.

And if you can solve your PRA with only crediting safety-related components, if you can meet those thresholds, then there are no additional components needed to be classified as RTNSS.

CO-CHAIR CORRADINI: Okay. All right, I think I've got it. Maybe I'm the only one who got it.

MS. BRISTOL: And the additional criterion that the PRA looked at was criterion delta, and that was the containment failure probability than 10 percent, and we met that as well.

CO-CHAIR SUNSERI: Members, any questions? All right. Well, we thank you for 19.3 presentation. Are you done?

MS. BRISTOL: There was also, similar to 19.1 and 19.2, we do have a COL item that confirms that for a COL applicant, they will also look at the PRA and the various criterion for RTNSS and confirm if there is any additional components required for the COL applicant, based on the criterion for that design.

CO-CHAIR SUNSERI: All right. Very well.

MS. BRISTOL: Thank you.

CO-CHAIR SUNSERI: So if we can have the staff come up now?

MR. CRANSTON: I'm Greg Cranston. We're ready to start on 19.3, and I'll just pass it on to Odunayo to commence.

MR. AYEGBUSI: All right, good morning. So the applicant had two slides; I have one slide, if you have any questions.

CO-CHAIR SUNSERI: Well, we'll be sure to take about four hours, then.

MR. AYEGBUSI: So my name is Odunayo Ayegbusi, and I'm a risk and liability analyst in the Office of New Reactors, and I'm here to talk about regulatory treatment of non-safety systems.

So the way I would actually like to start is kind of going back to the question that was asked to the applicant, and that is, this program or process was put together as a result of potential concerns with the performance of passive safety systems and passive design class.

And so the idea behind it was to try to identify non-safety-related SSCs that would be required to support the passive safety systems, given a set of criteria.

So these would be potentially active systems; they're not safety-related, but if they met the criteria, are determined to be, I would say, safety significant to support the passive safety systems.

And so our guidance talks about risk significance. I actually want to stress that when our guidance talks about risk significance in this particular case, it's not talking about just a PRA. It's talking about an overall risk significance.

So when you look at the set of criteria which I will go through, you will see that there are some cases where it's not probabilistic; it's actually deterministic. So I wanted to talk about that up front.

So the other thing I wanted to mention is, there is no regulation for RTNSS. So this RTNSS came out as a result of SECY papers that went up to the Commission that supported doing this, as I mentioned, because of the potential concern or certainty of passive systems.

So it's all policy-driven, and we developed our SRP 19.3 to provide answers to staff on how to review this. There's also guidance for the contents of the application that the applicant uses.

So as I mentioned earlier, there is a set of five criteria, and I was going to go through each one and discuss what we looked at to come to our conclusion about the application.

Actually before I go there, the other major item I wanted to point out was, the way the guidance is set up is, any SSC that is identified as a RTNSS SSC gets placed into D-RAP. And it gets put into the D-RAP, the Design Reliability Assurance Program.

It gets put into that list, and the reason for that is, when you look at our guidance, if it's a RTNSS SSC, similarly with the D-RAP program, such SSCs have to have certain treatments, such as augment equality, QA for non-safety systems, feeding those SSCs into the maintenance rule, creating short-term availability controls for non-safety systems that fit into the RTNSS program. So I wanted to just mention that as well.

So for the first criterion it's completely deterministic. There is some reliance on the PRA, but the first criterion has to do with ATWS, which is the 50.62 rule and the station blackout rule, 50.63 rule.

In that case what we're looking for there is, are there any non-safety active systems that the design would rely on to respond to these events? And in this particular case, the applicant did not identify any.

When we reviewed this, the sections that discuss ATWS and section that discussed station blackout, the staff did not identify any SSCs that would meet the requirements.

As I said, part of that is looking at

Chapter 19, which is the PRA. By looking at the analysis that's done for ATWS, and looking at if that meets the 10 D minus 5 criteria.

So that's criterion alpha. Criterion bravo has to do with any non-safety SSCs, active systems that are requested, post-72 hours, up to seven days. Again, reviewing the respective sections of the design, the staff didn't identify any SSCS.

As the applicant has presented, the ultimate heat sink supports the passive safety-related systems for a period of greater than seven days. So in that case, there were no SSCs identified there.

That criterion, bravo, also has us look at seismic considerations. So we looked at the PRA-based SMA, and in that case there were no non-safety-related SSCs that were identified.

So criterion Charlie is a criterion that utilizes the PRA. This criterion is basically looking at can you perform a sensory study that shows that using the safety systems alone, your results will still be consistent with the Commission's safety goals.

So in this case, if you ran your PRA with no non-safety systems, you'll still meet the Commission's safety goals of 10 to the minus 4 CDF, and 10 to the minus 6 for the LRF.

And in this case, again, the applicant didn't identify any, and when we reviewed the respective sections of Level 1 or Level 2 of the applicant's results, there was no cause for the staff to identify any SSCs that would meet this criteria.

So criterion delta: this has to do with any non-safety-related SSCs that area required to meet the continued performance goals. In addition with that, any non-safety-related SSCs are required to support, to respond to consider a bypass event during a severe accident.

So again, when the staff reviewed sensory studies for Level 1 and Level 2, as I mentioned in regard to that, a focused PRA, the continual performance goals when that continual performance goal is really a 0.1 factor.

And when we reviewed 19.2, 19.2 is severe accidents. This is a discussion about steam generator tube failure. In response to that event, there are no non-safety-related SSCs that are required to respond.

So again, this staff didn't identify any SSCs that the applicant should have added to this list.

The final criterion, echo, has to do with

SSCs that rely on preventing significant adverse system interactions between safety systems and non-safety systems. So in this case we looked at Section 6.2 of the SR. We looked at passive SSC design, and we looked at potential isolation from non-safety systems, and what we found was the applicant identified that the pressurized heaters are a potential area for an adverse interaction.

However, the pressurized heaters are in the reactor vessel, so when you look at that event, there is, in consideration of adverse issues, there are systems in place that will prevent the heaters becoming uncovered, and that's done by a safety system. So again, to prevent that scenario no non-safety-related system is required.

Lastly, the ultimate heat sink is cooled by non-safety systems, but as I said earlier, the ultimate heat sink has been shown to provide passive cooling for greater than seven days.

And then for the cooling systems, there are no connections to the ultimate heat sink pool at levels that would challenge the potential cooling of the pool for the passive systems.

Lastly, I wanted to say that the applicant, as our guidance has required, has identified a COL Action Item for a COL applicant to perform a review of the DC RTNSS and then develop any potential site-specific RTNSS SSCs and add that to that list. So that's adequately captured in the DCA.

So in conclusion, as the slide says, based on the staff's evaluation, we found the applicant's application reasonable and adequate.

CO-CHAIR SUNSERI: Good discussion. Just one question: in one of the first criterion you mentioned was dealing with the performance of the plant during a station blackout.

And I know it's been a long time since we've reviewed this, but we did review a chapter on highly-reliable DC electrical system, I think. So that system, if you consider it in the scope of a station blackout response still screamed out as not needing to be -- I presume since it's a highly-reliable system, it's not a safety system, right? So it would be a candidate for this criterion, but it screamed out.

MR. AYEGBUSI: So for all of the five criteria we have, for each one of them, the question you're asking is, do you need a non-safety-related SSC?

To respond to this event, given the plant design for the criterion you mentioned, the station blackout, the highly-reliable DC system, battery system, is not required in response to a station blackout. So the answer in this case would be no.

CO-CHAIR SUNSERI: Fair enough, thank you. Any other questions from members?

CO-CHAIR DIMITRIJEVIC: Actually, I have two: one is an observation and one is sort -- even though I understand how this happened, I'm a little concerned about.

One observation is that when you this was designed there was the 50.69 maybe wasn't for the federal -- you know.

So when 50.69 is in operation, you have a different categories of the safety and non-safety systems which are very significant, it may happen that these two things are in opposition to each other.

Because the one program, not many applicants came with 50.69, and I understand why, because the PRA are not in the stage to be applicable.

But if anybody applies 50.69, you will have a situation the 50.69 takes component out of QA program and all a bunch of regulations, and this problem puts them back. That's an interesting thing which may happen.

Maybe the future will be lucky, and the people actually using 50.69, then it will be an interesting controversy between those two -- maybe. I mean, some of those things are not -- even 50.69 uses an expert panel, I don't think some of those criteria were used.

The other thing is, we had the exemption from the GDC 27, the criticality, and that was a long time into the accident, which we analyzed, and I remember that, because I wrote exemptions from all that down there.

And I was wondering now, because the charging season was considered. But that's not a part of the ATWS analysis; however, it's a part of the exemption request.

So it's not in the PRA, but it is asking for exemptions based on the technicality of criticality of a long time after the shutdown, so I don't know if that seven-day area where the charging system would be considered, which is not of safety systems. I'm just wondering about it.

CO-CHAIR SUNSERI: I think we have somebody coming to respond to that.

MR. SCHMIDT: This is Jeff Schmidt from Reactor Systems. So on the return to power and the exemption for PDC 27 or GDC 27 technically, the replacement of PDC 27, where we're going with that from Chapter 15 sampling, and I think this will help inform you beyond 72 hours as well, is that what we're looking at is a potential return to power, and then we're evaluating whether the staff goals are met at that point.

So if the staff goals are met at 72 hours and longer, then we don't consider it to be a safety concern.

CO-CHAIR SUNSERI: All right. Anybody else? No? Okay.

CO-CHAIR CORRADINI: He just had backup.

CO-CHAIR SUNSERI: All right.

CO-CHAIR DIMITRIJEVIC: I have to think about it.

CO-CHAIR SUNSERI: Okay. All right. Ayo, go ahead.

MR. AYEGBUSI: Just going back to the observation on 50.69 and the interaction with this program, this program is not a PRA-based program. RTNSS is really looking for non-safety-related SSCs that will respond to very specific, deterministic, and probabilistic events; well, deterministic and probabilistic areas.

So I think --

CO-CHAIR DIMITRIJEVIC: It's sort of an old-fashioned program, because you're just looking at all the division between safety and non-safety. But let's say, I hope, in the near future that the division will be absolute, and we will have four categories.

So now if somebody comes with those four categories, then the question is, how compatible will those divisions will be? That's just my -- let's say maybe somebody in the COL phase will decide to apply 50.69, which will make perfect sense from my perspective, because a lot can be saved in the procurement of the components and things like that.

So let's say that some of the COL applicant decides to apply 50.69. It will be interesting how they interact with this program.

MR. AYEGBUSI: So I guess my point there is, if you look at something like ATWS or a station blackout, our review there is looking at the deterministic aspect.

CO-CHAIR DIMITRIJEVIC: What about the expert panels in the 50.69? You covered that, but some of those, I noticed, would not be covered. So it will be interesting to -- to merge all of these 50.69

-- treatment of the non-safety systems to look into how do they interact with each other and how -- 50.69 should actually be involved in all of this.

MR. AYEGBUSI: Okay. Understood.

CO-CHAIR SUNSERI: Is that it? Anybody else? Well, thank you all for your presentation, and at this point we will transition back to NuScale for Chapter 21.

So this part of the presentation is on the schedule for about an hour. We will go all the way through this section, and then we'll break for lunch when they're finished, and then we'll take staff after lunch.

(Pause.)

CO-CHAIR SUNSERI: It looks like Sarah brought some new faces with you.

CO-CHAIR CORRADINI: It's not the usual suspects.

CO-CHAIR SUNSERI: We may need to have introductions. We don't have name cards for --

CO-CHAIR CORRADINI: We have one.

CO-CHAIR SUNSERI: So for the recorder, we will need to state your name so he can write that down and notes who's speaking when you make your presentation. Sarah, whenever you're ready.

So since this is a new crowd, let's go through the procedure. You have a microphone; there's a button that says Push right near the edge of it. Not the light; the edge of the speaker. So push that, the green light comes on. Use it only while you're talking and turn it off when you're not, because of the background noise.

MS. JOERGENSEN: My name is Nadja Jorgensen. I am a licensing specialist with NuScale Power. We're here today to present Chapter 21, which is multimodule design considerations.

With me I have Zach Rad, director of regulatory affairs; Chris Maxwell; he's from our plant startup group. We have J.J. Arthur from NuScale engineering, and Sarah Bristol, whom you are very familiar with. I'll turn it over to Chris.

MR. MAXWELL: Hi, Chris Maxwell, plant startup. With 12 identical modules and a separate reactor in each module, it will be operated independent of the state of construction or operating conditions of those other modules. And given that we have shared systems between the modules, the NuScale Power plant design meets the definition of a modular design in accordance with 10 CFR 52.1.

MEMBER REMPE: So when I saw this quote on your slide the other day, I just don't think the folks who wrote 10 CFR 52.1 were thinking about, you might have all 12 of the modules in a swimming pool, with one of them swimming by the others to go for refueling.

So despite the fact that you meet the definition, I just don't think that the people who wrote that had envisioned that when they started it. It's just a comment. So some of my questions may be based on that.

MEMBER BROWN: You've got to be careful with the words co-located on the same site. This is different from co-located on the same site. That's kind of a stretch. I had the same comment as Joy did when I read that.

There are acres between co-location on currently actives, and here they're eyeball to eyeball. That's kind of a stretch, saying you meet that. Not that we're not going to do it, but that statement is kind of out of sorts.

MR. MAXWELL: Well, with the modular design, 10 CFR 52.47(c)(3) requires for modular designs an evaluation of the modular operating configurations to consider common or shared system, interface requirements, system interactions, and restrictions during construction startup.

We'll start with discussing the evaluation of the shared systems. When designing those shared system, only the systems which allow for safe operation of one module independent the operation of any of the other modules is permitted.

As a result, the ultimate heat sink is the only safety-related system common to multiple modules. Or, stated another way, with the exception of the ultimate heat sink, the safety-related systems are module-specific, and they're functionally independent of share systems and other NPMs.

So for each of our modules there exists an independent emergency core cooling system, a containment system, a decay heat removal system, module protection system, and a set of demineralized water isolation valves, safety in valves, and chemical environmental control system.

MEMBER MARCH-LEUBA: How do you treat AC power?

MR. MAXWELL: AC power was evaluated for multimodule conditions. We'll see here a list of the systems; it was evaluated.

MEMBER MARCH-LEUBA: But it's shared among the modules, right?

MR. MAXWELL: Shared, but not safety-related; that's correct. We have no safety-related electrical power.

MEMBER BLEY: Well, AC electrical power.

MR. MAXWELL: No safety-related AC or DC power.

MEMBER BLEY: I thought there was --

CO-CHAIR CORRADINI: You wrote the letter.

MEMBER BLEY: Yes, I did, but I thought there was one DC.

CO-CHAIR SUNSERI: You have some 1E stuff, right? For --

CO-CHAIR CORRADINI: It was non-class 1E.

MR. MAXWELL: We have no class 1E power. We have class 1E module protection system, but not power supply.

Because the ultimate heat sink supports multiple modules, accordingly it was designed with multimodule demands in mind. Regarding multimodule operations, the ultimate heat sink has sufficient capacity to remove the heat from one module that's experiencing a design-basis accident, while simultaneously removing heat from the other 11 modules while undergoing a controlled shutdown and cooldown.

In the completion of that shutdown and

cooldown, long-term cooling will be provided by safety-related ultimate heat sink and the module-specific safety-related systems, again, without reliance on any other shared systems or any operator action.

So an evaluation of each shared system that has the potential for adverse system interaction --

MEMBER BROWN: Go back. Excuse me; under the DBA with 11 shutdown and cooldown, once they get there, you said they will be maintained using their explicit systems. You don't need any shared systems at that point, once they're to that circuit? Did I understand that correctly?

MR. MAXWELL: Yes, sir. The one shared system is the ultimate heat sink.

MEMBER BROWN: I understand, the big tank of water, but all the rest -- there's other shared systems when you're in normal operation, none of those are required once you have this DBA and the other ones are being maintained shutdown?

MR. MAXWELL: That's correct.

MEMBER BROWN: Okay. Thank you.

CO-CHAIR DIMITRIJEVIC: Well, how about non-DBA, you add? Does it have the chemical capacity to do this with LOCA?

MR. MAXWELL: Yes, it's designed for -- in Chapter 15 space for accident on all 12 modules has sufficient capacity.

CO-CHAIR DIMITRIJEVIC: Okay. So even if all 12 modules are inactive, then it has enough capacity to remove the --

MR. MAXWELL: That's correct.

CO-CHAIR DIMITRIJEVIC: Okay.

MR. MAXWELL: Yes, not to be misled by that statement, specifically talking about operating conditions with a module experience is a design-basis accident, and the idea is that the 11 modules could be undergoing normal shutdown and cooldown with the ultimate heat sink capacity.

And then once those modules are shut down and cooled down, normal long-term cooling can exist with the ultimate heat sink and the module-specific safety-related systems.

CO-CHAIR DIMITRIJEVIC: Okay.

MR. MAXWELL: So again, the evaluation of each shared system that has the potential for an adverse system interaction or an undesirable multimodule interaction was conducted.

The types of systems that were evaluated include the cooling water systems, radioactive waste systems, building HVAC systems.

MEMBER REMPE: Before you leave that one, just so I understand: for a system like the reactor component cooling water system which -- there's two of them, and each one supports six modules.

MR. MAXWELL: That's correct.

MEMBER REMPE: So if you have -- and again, if you have a failure in it, does that mean you have to shut down all six of those modules?

MR. MAXWELL: I'll answer your question in two slides, if I may.

MEMBER REMPE: Oh, okay.

MR. MAXWELL: And maybe more than two, but I've got a direct answer to that question.

MEMBER REMPE: Okay.

MR. MAXWELL: So in addition to these systems, we have support systems are evaluated. All of the electrical distribution and power systems, and the com and I&C systems all evaluated.

CO-CHAIR SUNSERI: I don't see the -- where's the system for drawing the vacuum on these containments at? Is that up there?

MR. MAXWELL: The containment acquisition system is module-specific, not shared.

CO-CHAIR SUNSERI: Oh, okay.

MR. MAXWELL: The result of the evaluations demonstrate that the shared system operation does not result in any adverse system interaction such as a loss of a safety function or a design-basis event and a simultaneous degradation of a safety function, or a design-basis event and simultaneous degradation of critical operator information, or a design-basis event and a requirement for an operator action outside the control room.

That's a summary of the evaluations of those systems.

One category of systems would be those shared systems that serve one NPM at a time. And for those systems, they are equipped with isolation features that prevent a direct module-to-module interaction during normal operation.

That's to say that those systems, while they are capable of supplying each of the modules or six modules, they are designed to be in service with only one module at a time.

So for an adverse multimodular interaction to occur as a result of a failure in that shared system, one of two conditions would have to exist. The first would be that the shared system was placed in an abnormal lineup prior to operation outside of the normal operating procedures.

The second would be that multiple concurrent failures would have to occur within either the system or the support system.

MEMBER BLEY: Are there interlocks or anything that help prevent operators from doing that?

MR. MAXWELL: Not all interlocks have been defined at this stage of the design.

MEMBER BLEY: Do you plan to have interlocks, or are you counting on your operating procedures?

MR. MAXWELL: I guess I can't answer that question at this time. I'm not sure what direction plant ops plans to go with.

MEMBER BLEY: Okay.

CO-CHAIR CORRADINI: But I guess to answer Dennis' question, it will either be operator procedures or interlocks, but you don't know which is which, or which ones would be in which category?

MR. MAXWELL: That's correct.

MEMBER BLEY: If it's strictly admin, then it's going to end up a line drawn some time. It's just going to happen.

CO-CHAIR DIMITRIJEVIC: I'd like to ask

you about these reactor pool cooling system. What is that system?

MR. MAXWELL: It typically takes a suction from the spent fuel -- I'm sorry -- the reactor pool, goes through the heat exchanger, cooled by cycling water system and returns back to the -- it goes through demineralizers for cleanup if necessary and returns the water back to the ultimate heat sink.

CO-CHAIR DIMITRIJEVIC: So there is another head sink on this heat exchangers, where it's cooled, right?

MR. MAXWELL: Site cooling water system, right. It has cooling towers associated with it. Water circulated from those cooling towers --

CO-CHAIR DIMITRIJEVIC: What will be consequence of losing that system?

MR. MAXWELL: Of site cooling water?

CO-CHAIR DIMITRIJEVIC: Of that -- yes, the reactor pool cooling system.

MR. MAXWELL: The gradual heatup of the ultimate heat sink.

CO-CHAIR DIMITRIJEVIC: So how long will it take for this to be a problem?

MR. MAXWELL: We also have another system, the spent fuel pool cooling system, and the reactor pool and the spent fuel cooling system communicate with on another during normal operations. So you're still removing heat when you have the spent fuel cooling system in service.

CO-CHAIR CORRADINI: So you'd need a failure of both to actually cause the pool to heat up? I think that's where she's going with that.

MR. RAD: I think the answer to that is going to be conditional, but we'll see what sort of answer we can find for your away team.

MEMBER BLEY: Chris, your second bullet there makes sense, but I don't think, on the next slide, you tell us what you do if you have adverse multimodule interactions.

MR. MAXWELL: Well, the point of the slide is that --

MEMBER BLEY: You don't expect them?

MR. MAXWELL: That's correct.

MEMBER BLEY: But it could happen.

MR. MAXWELL: Operators would respond to the conditions that were generated as a result of the adverse interaction, and procedures and indicators that would provide them --

MEMBER BLEY: Probably some set of abnormal procedures will exist.

MR. MAXWELL: Yes, sir.

MEMBER BLEY: Okay. Those are all going to have to be in place for a COL?

MR. RAD: So the question is, what do you mean by COL? Do you mean at the application stage, or do you mean prior to loading fuel and starting up a module?

MEMBER BLEY: Well, what I mean, first, is before the COL applicant gets its license.

MR. RAD: Generally, the answer to that is no. Usually in Chapter 13, if you look at applications, there's a short description of what the procedures will be, and then a commitment and milestone to implement them prior to a particular phase. It's usually post-licensing, though.

MEMBER BLEY: Yes. Then tomorrow, we'll get into --

CO-CHAIR CORRADINI: ITAAC.

MEMBER BLEY: ITAAC. My mind goes blank on acronyms. So likely to be accurate ITAACs on those?

MR. MAXWELL: No ITAAC.

MEMBER BLEY: No ITAAC? Just a commitment?

MR. MAXWELL: That's correct; not an

ITAAC.

CO-CHAIR CORRADINI: Keep on going.

MR. MAXWELL: Yes, sir. So in a separate category is design includes reactor cooling water system, RCCW. It's the only shared system that directly interfaces with multiple modules and is also designed to simultaneously support more than one NPM at a time.

CO-CHAIR CORRADINI: And there's one per six?

MR. MAXWELL: That's correct, two subsystems for the entire -- if there's a failure in a leak in a cooler or a heat exchanger, either in the system itself or support system, the RCCW system is designed to be able to completely isolate that and not impact the other five modules.

It's also designed with sufficient capacity such that a single component failure doesn't result in loss of cooling to more than one module at a time.

MEMBER REMPE: Okay. So my question is, you're saying that even though it's there to support six modules, if it fails, five of those six modules can still keep plugging along and chugging along?

Because the other RCCW system can't

support those five modules, is my understanding. You're not allowing any sort of crossties. And so this isn't an essential item for the modules, the other five, that can be serviced by the RCCW system to be serviced by it? They can function safely without it; is that true?

Commonwealth Safely, yes; however, not continue to operate.

MEMBER REMPE: Oh, you do have to shut them down?

MR. MAXWELL: That's correct.

MEMBER REMPE: Okay. So are there other shared systems where one supports six modules, that if it goes out, you're going to have to shut six modules? Or if there's one that supports four modules?

This is more of an investment protection, but it could tie to safety, because it seems like you may have a lot of modules that can't continue operating because of these shared systems, and you're going to be seeing a lot of shutdowns that people have not really thought about because of the way this plant is configured.

CO-CHAIR DIMITRIJEVIC: Like a circulating water system?

MR. MAXWELL: Circulating clear water

system; a complete loss of one -- so the answer to the question is, it depends on how far back you zoom from the module itself. By that I mean, right circulating water, if it cools the condensers on the balance of plant side, if they've experienced a complete loss of one subset of circulating clean water system, you would have to shut down those six modules.

MEMBER BLEY: You say it's the only shared system that can interface with modules, but I guess that means excluding electric power.

MR. MAXWELL: It's the only one that directly interfaces with the six modules; in this case, the interface requirement being a direct piped connection between the shared system and the module itself.

MEMBER BLEY: So it's the only fluid system that can --

MR. MAXWELL: That's true.

MEMBER BLEY: Go ahead.

MEMBER SKILLMAN: What about ventilation?

MR. MAXWELL: Ventilation is a spatial coupling between the modules, but again, not a direct interface with the modules themselves. Like a malfunction in the -- and the important distinction there is that a malfunction in an HVAC system affects each of the modules individually, but doesn't result in an adverse system interaction, module to module. You have to respond to that condition for each module independently.

CO-CHAIR SUNSERI: So I think, just pulling the string -- and we'll do more on Joy's question -- so if you lose component cooling water, you have six modules that are going to be shutting down, right?

MR. MAXWELL: That's correct.

CO-CHAIR SUNSERI: Your control room staffing is not large enough to deal directly with six modules, so you're going to have some of these going into DRS cooling, and then some of them in a monitoring mode? So that's where the safety nexus is, right? The distraction or the work demand on the operating crew to be able to handle six simultaneous shutdowns, right? Have you looked at that safety implication?

MR. MAXWELL: We have. We did a staffing plan validation and an in-grade system verification, which I was lucky enough to be a part of. We were challenged with multimodule interactions, including the loss of RCCW.

One important point is that this isn't a complete loss of RCCW; it doesn't require an immediate action or an immediate response. The necessary response time is measured in tens of minutes, approximately 40 minutes, because of the systems that are cooled by RCCW.

It controls our control rod drive mechanisms, cools the wetdown cooler for CVCS, and various sample coolers.

So the driver of the shutdown is the loss of cooling to the control rod drive mechanism, and the staffing was adequate to respond to six modules, to take the actions to trip the units, isolate containment, initiate the decay heat removal system, passively place six modules in cooling, so safe shutdown.

CO-CHAIR CORRADINI: So to answer Matt's question differently, if you lost component cooling there would be procedures such that modules one through six would sequentially be shut down in a normal procedural fashion such that there's more than enough time to do it for all six before any one of them has a critical function that needs to be cooled; am I understanding this correctly?

MR. MAXWELL: Yes, sir, that's correct, and you won't challenge a safety function on a lost

RCCW. There won't be a safety or a critical safety function challenge by the loss of this support system.

CO-CHAIR DIMITRIJEVIC: But it would be for loss of circulating collecting water. You will challenge your decay heat removal system.

MR. MAXWELL: Not safety related. We still have the decay heat removal system and our ultimate heat sink to provide the safety-related at the decay heat removal. So there's no challenge to a safety function.

CO-CHAIR DIMITRIJEVIC: Okay. But it's challenged because it has to operate. I don't know how you interpret challenge. I thought that you say challenge, that means that it's not required. But it is -- so did you analyze the similar situation for loss of feed water to six units?

CO-CHAIR CORRADINI: I'm looking at his definition. The way the applicant has defined the definition, this is the only direct connection.

But circulating water is not a direction; nonetheless, if it fails for six units --

MR. MAXWELL: That's correct.

CO-CHAIR CORRADINI: It might require an orderly shutdown.

CO-CHAIR DIMITRIJEVIC: Yes, and it may

require a little more human action than the operator. I mean, I understand you have a danger in pass or fail, more things are happening, so operators have to be ready. That's happening on six units.

MR. MAXWELL: I think an important point to make is that a loss of the RCCW system or the circulating water system, neither of those require an operator action to maintain the safety functions.

The plant is designed to respond if you reach actuation set point for the module protection system to trip the reactor, isolate containment, and initiate decay heat removal system.

So the safety functions are maintained. These are just the support systems to in support of operation, not of safety.

MEMBER REMPE: So we've heard that the procedures aren't going to be developed until the COL applicant is engaged, et cetera, et cetera.

But there are a lot of these support systems that will require that you have some procedures to say, Okay, if you lose system X, just like this RCCW system, you've got to trip, in a safe, orderly manner, X number of modules, depending on what they are.

Is there a list somewhere? I see you have a list in the DCA 21 about initiating this triggered by systems, and how many systems support how many modules, but is there a list somewhere that will be turned over to the COL applicant about all the different systems that are going to require that X number of modules be shut down?

Am I making myself clear, what I'm trying to ask here?

MR. RAD: I think so. We'll be present of the COL application development as an entity.

MEMBER REMPE: So I've heard that statement before, but we are talking about a design certification. And historically in the U.S., once you have a certified design, it may be hard, but another design developer could come in and take that certified design and go forward.

So when the agency certifies your design, we need to know what we're certifying. So that's why I'm asking, is that information available as part of the design certification?

MR. RAD: We can look and see if it's part of the certified design or the approved information in Tier 2, Part 2, in answer to your question.

MEMBER REMPE: Okay. Thank you.

CO-CHAIR SUNSERI: So I'll leave it alone after this, but I'm just going to make a statement just for you to consider.

I heard many times your group and other groups come in here and say, Well, you know, at the time of event, we don't have to take any immediate actions.

But having been in a control room, I know the operators -- it's against human nature not to do anything, and there will be things that they have to do. They will be monitoring systems and all that stuff.

And I know from being in a control room and from all the training and background that I've had, I know they are at their best when they do that, because this is what they've been trained for.

However, on the flip side of that is the most error-likely situation you can put them in. It's high workload; it's a high-stakes environment, so I just ask you to consider that more -- not as casually as I've been hearing it stated to us, well, we just hands off and watch this, or let's not do anything, really.

MR. RAD: I think this is a good topic for the onsite visit, because we've put a tremendous amount of work into it. This is one of our longest-running programs. It's a tremendous investment for us, and I think our program for HFE and all the way out to the ISV program that was just discussed, I think it would be very important for you guys to come out and witness directly the work that we've put into this, because it's not something that we've not taken seriously. Like I said, it's one of our longest-running programs.

MR. MAXWELL: On that same note, I mentioned the staffing plan validation that I was a part of prior. I was previously a licensed SRO at a commercial facility, as were the other individuals during the staffing plan validation.

And I'll say that I had a modicum of skepticism myself, having been an operating control room. But I think the point that was just made is that then, when you experience it and realize that there are no immediate actions, there isn't that pressure for the operators to take action because of the design, that there wasn't a sense of time pressure. You just didn't feel the same pressure that I was used to, coming from a traditional commercial design.

CO-CHAIR DIMITRIJEVIC: Same example for common initiators, like loss of offset power, which effect is the same situation? You don't have to worry about supplying batteries, starting generators, doing things like that?

MR. MAXWELL: Yes. The design as such doesn't require additional water, any operator action, or electrical power, AC or DC, to maintain the key safety functions.

So in those scenarios that we were put under, we didn't know what scenario was going to be run on us with our standard staffing, and we had a loss of all AC power.

We experienced loss of circulating water, loss of RCCW, multimodule, modules in ATWS with other modules undergoing events, and just because of the element that I spoke of, the design and the design of the HSI, the interface, the information available to the operators removed that pressure, because you could see the automatic response of the systems and that the safety functions were occurring without any operator action as designed.

CO-CHAIR DIMITRIJEVIC: So how long would your batteries last without AC?

MR. MAXWELL: Well, in our highly reliable DC power system we have 72-hour batteries and 24-hour batteries, and they have fully-redundant backups as well, so --

CO-CHAIR DIMITRIJEVIC: And you don't need to shed some loads and everything to preserve them?

MR. MAXWELL: No. Designed to maintain a minimum of 72 hours with the loads that are on them at the event initiation.

MR. MAXWELL: The next element that was evaluated was concerns during construction startup, and during the construction startup, phases that occur prior to the initial module fuel load, the shared module-specific systems that are in the reactor building, the control building, and the rad waste building will be substantially completed, again, prior to that initial fuel load.

The construction method and the phased expansion of the modules provides the assurance that the operating configuration is not materially different from that assumed in our safety analysis, and that the independence of each module's safety-related system is maintained.

MEMBER BROWN: Can you backtrack that slide? I think one of the difficulties of understanding how the --

CO-CHAIR SUNSERI: Do you have your green light on?

MEMBER BROWN: Thank you. I had it on; she told me to turn it off, my mother hen over here.

One of the difficulties I had was, there's no integrated architecture that you're showing how all the shared systems are integrated across all 12 modules, and therefore how they are isolated, for instance. The description of isolation is vague, if non-existent.

So it's just a little bit -- the words are fine, but in terms of agreeing with the concept, which I don't have any problem with, but how it's done, at least in a conceptual manner, it is not -- you just state in Chapter 21 that they're going to do this.

Have I misstated that anywhere? I don't remember seeing a giant, 12-module full facility operation showing an architecture that shows how all these shared systems are set up and how they will be available for isolation, if necessary, from module to module.

MR. MAXWELL: Right. That drawing does not exist in Chapter 21.

MEMBER BROWN: That's what I thought. Okay. And if I'm trying to get my mind around it, or to agree with it, while conceptually I agree with it, it's kind of hard to see that there are adequate means of showing that you've got reliable ability to isolate and to not have non-adverse reactions on more than one, other than the RCCW system, based on their plant configuration. That just doesn't exist. So the words are nice, but the implementation is non-existent.

MEMBER BLEY: It might be, from what they said earlier, Charlie, it might be just direction to the operator --

MEMBER BROWN: And that may not be -- that's not necessarily a good conclusion to come to. When you have multiple and complex interactions between 12 different reactors -- I can only speak to that as I was a Naval nuclear guy, and the Enterprise had eight reactor plants on it.

MEMBER BLEY: I can speak real directly to that.

MEMBER BROWN: I know you can speak real directly to that, and we fought like the devil to be try to ensure that those systems, that could be cross-connected, whether they be electrical or fluid systems -- and Dennis had direct relations to it in his operations -- was very complex, relative to how you do it, and you've got a similar situation with 12, and two dozen different systems that you're integrating on this.

MEMBER BLEY: And if they're not normally cross-connected as they inadvertently or on purpose get cross-connected someday, you might not have the right to balance as you would expect, and water can move really fast.

MEMBER BROWN: So I've got real difficulty with fundamentally agreeing with Chapter 21 the way it's presently configured. I have no idea what we'll do. That's a personal opinion; that's not a committee opinion, by the way.

MR. MAXWELL: That information is available in the equipment drawings and the P&IDs, detailed P&IDs that have the valves and show those isolations.

MEMBER BROWN: But it's like -- pardon?

CO-CHAIR CORRADINI: I was going to say, I think all Charlie is saying is, If I go to Chapter 21 in isolation, there's nothing there to graphically explain what you're describing.

MEMBER BROWN: It's like telling me the module protection system is independent, four channels, does all these nifty things, but I don't have an architecture that demonstrates that. There's no architecture that demonstrates that, and this is very complex in terms of overall plant operations.

That's just my thought process.

MEMBER BLEY: It would be very helpful to see the P&IDs, but --

MEMBER BROWN: That's not enough. I know, but if you look at those, and you try to stack all those together individually one at a time, you just don't get --- it's difficult -- I'm supposed to get all my drawing tables out and start drawing all these things and showing them all at the same time? I didn't think so.

I've said my piece; I'll stop there, and you can go on.

CO-CHAIR SUNSERI: All right, continue, please.

MEMBER SKILLMAN: Before you change that slide, let me ask this. Several meetings ago, one of the lines in questioning had to do with how the build out is occurring. Let's for instance say you've got two, or three or four modules in operation and you're bringing number five into the plant. When number five arrives from the manufacturer, is number five fully tested and ready to be inserted into its operations bay? Or does number five find a place in the pool where there is additional testing to confirm that five is fit for duty before it is moved to its operating bay?

MR. MAXWELL: The plan is that module five is placed in the module five operating bay. It's connected and then, as part of the initial test program -- there's a set of pre-operational tests that must be performed to verify that -- and ITAAC that must completed prior to loading fuel for module five.

MEMBER SKILLMAN: Thank you.

CO-CHAIR CORRADINI: Actually, the way you said it makes it -- I guess I hadn't thought about it that way. But you can -- I have just a couple slides, so somewhere in there you want to answer it. So, to satisfy this pre-operational procedure -- pre-operational testing, and to satisfy ITAACs, it's a module-by-module satisfaction?

MR. MAXWELL: There's a -- we will discuss it in more detail with Chapter 14.

CO-CHAIR CORRADINI: Why don't we just wait? Let's just wait. I can bring it up tomorrow.

MR. MAXWELL: Okay. Next slide, please.

MEMBER BLEY: I am still back where Charlie was, and I have a question for Sarah. Did the PRA consider -- or why shouldn't it have considered -- the possibility of shared systems being operable with more than one unit at a time? Could that have led to some initiating event, added to the likelihood of an initiating event, or complicated things in any way?

(No audible response.)

MEMBER BLEY: You can think about it till the next time we see you, if you want.

MS. BRISTOL: Okay.

(Simultaneous speaking.)

MEMBER BLEY: I am interested in your --

CO-CHAIR CORRADINI: I'm not understanding the question. Help me.

MEMBER BLEY: Okay. You're running the plant --

CO-CHAIR CORRADINI: Yes.

MEMBER BLEY: You open the wrong valves and now you've got one of these shared systems or more connected to more than one unit. Which isn't supposed to happen. Can that to some event? Or increase the likelihood of some event that should have been modeled in a particular -

CO-CHAIR CORRADINI: The question is, is that sort of initiator considered.

MEMBER BLEY: Yes. And if not, why -- why didn't it need to be considered? Or, its impact on some initiating event frequency.

MS. BRISTOL: And so, as Chris had

mentioned, with examples of circ water or component cooling water, we do look at a general transient. You know, and so we have the initiating event frequency of about one a year of these numerous trips that could impact the modules. And so, if that was a shared system that failed, that would be captured in the single-module PRA as a general transient. And that's how the plant would respond. We didn't find anything specific with the loss of one of these support systems that would impact or present a different event sequence.

CO-CHAIR DIMITRIJEVIC: You also have a loss of support sequence --

PARTICIPANT: Vesna, do you have your mic on?

PARTICIPANT: Your mic is off.

MS. BRISTOL: For the loss of support system initiating event frequency, we looked at the loss of responding systems. So the loss of CVCS or the loss of instrument air in order to respond to the event. The loss of a shared system impact on a module to create the initiating event was captured as a general transient. The loss of support system initiator was the loss of a support system needing to respond to the initiating event.

MEMBER DIMITRIJEVIC: And loss of feed water?

MS. BRISTOL: That is correct.

MEMBER DIMITRIJEVIC: So do you have actually fault trees for initiating events for some of those systems?

MS. BRISTOL: We didn't calculate any -- outside of ECC -- as a spurious ECCS initiator, we didn't have any support system faults redeveloped initiators.

MEMBER DIMITRIJEVIC: It was just based on genetic data?

MS. BRISTOL: Generic data, yes.

(Pause.)

MR. MAXWELL: The analysis, though, of the shared systems that we talked about previously was conducted for six systems -- or, six modules that were talked about, subsystems -- or 12 modules for the support systems that support all 12. And as a result it continues to be bounding as you install subsequent NPMs. As a result there's restrictions in the operating configuration, or interface requirements, aren't necessary to ensure that the operating NPMs continue to operate safely during the installation testing or startup of the subsequent NPMs. And that's

-- that concludes the Chapter 12 presentation.

CO-CHAIR CORRADINI: Thank you. Members, any other comments?

MEMBER MARCH-LEUBA: So, different topic, but since we have a qualified reactor operator here, you lose power to your reactor and all the valves close and open, depending on which direction, and you go into DHRS. What mode are you in? In operator action? You're definitely not in mode 1 anymore. Let me remind you, mode 1 is operation, mode 2 is hot shutdown and mode 3 is safe shutdown, transition --

MR. MAXWELL: Assuming 100-percent power, the event initiation you would be in mode 2, heading towards mode 3, safe shutdown.

MEMBER MARCH-LEUBA: Right. So you will need to establish a boron concentration consistent with safe shut-down before you transition from 2 to 3, correct?

MR. MAXWELL: No, the module operates with the required boron concentration --

MEMBER MARCH-LEUBA: For hot shutdown.

MR. MAXWELL: It's safe -- for safe shutdown. We establish safe shutdown without additional boron.

MEMBER MARCH-LEUBA: You operate a safe

shutdown boron?

MR. MAXWELL: That's correct.

MEMBER BLEY: But not cold shutdown, then?

MEMBER MARCH-LEUBA: No, not cold, but safe -- safe shutdown -

CO-CHAIR CORRADINI: Is a temperature defined quantity.

MEMBER MARCH-LEUBA: Yes.

MR. MAXWELL: I am sorry, what's?

CO-CHAIR CORRADINI: No, I am just clarifying what I thought was --- when you say safe shutdown, it is not cold shutdown. It is at a specified temperature, which I cannot remember.

MR. MAXWELL: Four hundred and twenty degrees. That's correct. That's safe shutdown.

MEMBER MARCH-LEUBA: Safe shutdown is below -- less than 420.

MR. MAXWELL: That's correct.

MEMBER MARCH-LEUBA: And that -- it has no lower limit.

MR. MAXWELL: That's correct.

MEMBER MARCH-LEUBA: So it could go to 40 degrees Fahrenheit.

MR. MAXWELL: That's correct.

MEMBER MARCH-LEUBA: And you operate to a normal operation without boron concentration?

MR. MAXWELL: That's correct.

MEMBER MARCH-LEUBA: I don't believe you.

(Laughter.)

MEMBER MARCH-LEUBA: This is a very calm, honest statement coming from this microphone. I don't think that is correct, but we will come back to it.

CO-CHAIR DIMITRIJEVIC: MY question is -- I mean, I am not sure -- when I was reading first time this chapter, I was expecting also more than just shared system and I was expecting -- because you have this other consideration which will be like, you know, module drop during refueling affecting other modules -- or, accident in one module affecting other modules. How come that's not part of this section?

MR. MAXWELL: As far as an accident in one module impacting other modules, that is part of the interfacing evaluation to the point that one module can't affect -- there isn't a viable interface between the two that causes a multi-module interaction between these shared systems.

CO-CHAIR DIMITRIJEVIC: Well, they're just considering this hydrogen explosion impact, so I mean, that is consideration for -- I mean, you know, it's not -- well, you may conclude that there is no problem, but there should be discussion somewhere. That's what I was thinking. Why is the discussion -- like, that not the part of this session? That's only my question. I mean, I don't -- module drop impact on other two modules, that's all part of the module design. So I was thinking, those things will nicely belong to this session to cover all the issues with the multi-module design.

MR. RAD: So, when we wrote this application, we invented this discussion. And we decided to limit it to the scope given here and put those sort of severe accidents in Chapter 19, which were previously discussed. That's all. It was just a matter of deciding which --

(Simultaneous speaking.)

CO-CHAIR DIMITRIJEVIC: All right, because you know somebody would check this and say, okay, what is the -- what type of issue I can see with multi-modules and the person saying multi-module design consideration, and then there is -- not all considerations are here. That's my point. So. I mean, if you wanted to do, you should make it like really to cover all the issues experienced with the -- analyzes -- I mean, only know ones from 19. Maybe they're in the other chapters, also, bad issues -

MR. RAD: It is already somewhat redundant in that GDC-5 is discussed with each of the systems. And so it's -- in this case, already redundant in a lot of cases to a lot of the system descriptions. And so it -- in some ways, it's just a summary of the discussions and the system descriptions that are provided throughout the FSAR today. So, I understand what you're saying though. I understand the comment.

CO-CHAIR DIMITRIJEVIC: Especially when you had a peer review that told you you guys are making first step for the industry in the multi-module, so -- I know it's not easy to be first, but we appreciate --

(Laughter.)

CO-CHAIR SUNSERI: Any other comments from members?

MEMBER MARCH-LEUBA: Yes, just a comment to Zach. When we go there in July -- to your place, I will bring this issue of the -- of the temperature after shutdown and boron concentration with you guys. If you can be ready for it. I will ask it.

CO-CHAIR SUNSERI: I understand.

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: I don't think the answer we have today is --

CO-CHAIR SUNSERI: Yes, and I think -- and I -- just kind of being a -- listening from the side here. I don't think you all are connecting. So maybe if you could write down the question, so they clearly --

(Simultaneous speaking.)

MR. RAD: I think it would be helpful for us to answer rather than off the top of someone's head -- to answer specifically, you know, given the appropriate information right in front of us.

MEMBER MARCH-LEUBA: We were told that the Core Operating Limits Report, the COLR, hasn't been developed yet. And I suspect that the COLR will have some boron concentration that you must achieve to have such safe shutdown based on calculations you provide in Chapter 4 and I would like to hear more about that once we are there.

MR. RAD: That's great. Yes, I believe the tech specs say in accordance with the Core Operating Limits Report.

MEMBER MARCH-LEUBA: We don't have a -- you don't have a COLR. No, the tech spec is in accordance with the COLR because it says thou shall satisfy the COLR. The question is, what is in the COLR?

(Simultaneous speaking.)

CO-CHAIR SUNSERI: Let's coordinate through Snodderly so that when we show up in --

(Simultaneous speaking.)

CO-CHAIR SUNSERI: -- they'll know what we are talking about.

CO-CHAIR DIMITRIJEVIC: And you are going to tell us a little more about this reactor or pooling -- pool-cooling system, right? You said that they --

CO-CHAIR CORRADINI: We have it all -- it's all planned as part of the visit.

CO-CHAIR DIMITRIJEVIC: It's plan as a visit -- it's not going to be discussed in closed session or anything?

CO-CHAIR CORRADINI: What are we talking about?

CO-CHAIR DIMITRIJEVIC: Reactor pool cooling system, which I ask in the beginning of this.

CO-CHAIR CORRADINI: We can talk about it in closed session if you would like to.

MS. JOERGENSEN: So I have -- I just have one comment. So you asked if you lost site cooling, what the impact on the -- on the UHS was.

CO-CHAIR DIMITRIJEVIC: Uh-huh.

MS. JOERGENSEN: And from engineering,

they said the pool temperature would rise very slowly at a rate less than a degree per hour, which would give the operator sufficient time to respond.

CO-CHAIR DIMITRIJEVIC: That's -- this is not a safety system reactor pool --

(Simultaneous speaking.)

CO-CHAIR DIMITRIJEVIC: Raised one degree. So 72 degrees for your mission time. Right? Something to think about.

(Simultaneous speaking.)

CO-CHAIR DIMITRIJEVIC: So in accident condition, that's not necessary, you're saying.

MR. MAXWELL: That's correct. It's not -- reactor pool cooling system is not required in accident condition.

CO-CHAIR DIMITRIJEVIC: All right.

MR. MAXWELL: Also, just to point out that as that -- as pool temperature escalated, there's tech spec limits on ultimate heat sink temperature that will direct the operators to conduct controlled shutdowns, which they'd use the normal heat sink for prior to reaching any safety limits.

CO-CHAIR DIMITRIJEVIC: What do you call normal heat sink?

MR. MAXWELL: Condenser.

CO-CHAIR DIMITRIJEVIC: Oh, condenser, okay.

CO-CHAIR SUNSERI: Any other member comments or questions?

(No audible response.)

CO-CHAIR SUNSERI: All right, at this point we are going to recess for lunch. We will reconvene at 1:00 and we will begin with the staff's presentation on Chapter 21. There is about an hour-and-a-half of technical presentations remaining in the agenda today, and then we will go into a public comments session followed by the closed session. So please plan appropriately for the after-lunch time. Thank you. You're recessed.

(Whereupon, the above-entitled matter went off the record at 11:54 a.m. and resumed at 1:00 p.m.)

CO-CHAIR SUNSERI: All right, 1:00, let's reconvene the session here on Chapter 21, multi-module design with a review by the staff. And I will turn it over to Mr. Chowdhury.

MR. CHOWDHURY: Yes, good afternoon. My name is Prosanta Chowdhury. I am the NRO project manager for Chapter 21. I have been with the agency for about 15 years -- 11 of which I have been working in the NRO as a project manager. Prior to that I was employed by the State of Louisiana for their radiation protection program involving nuclear power plants surrounding the state, and I worked there for 18-and-a-half years.

I have a master's degree in electrical engineering, and one in nuclear engineering. With that, this is Chapter 21 of NuScale Design Certification Application that the agency is reviewing right now. And Chapter 21 is multi-module design considerations. We have had the applicant's presentation before lunch today.

Before I go into my slides, right next to me is Marie Pohida, one of the key reviewers of multi-module interactions. I do have other staff present in the audience and I will mention their names in one of my slides and the specific questions related to those areas that they have reviewed. They will be able to answer those questions. I request them to come up to the microphone and state their names and affiliation, and then answer questions or make comments.

The technical staff that were involved in reviewing aspects of Chapter 21, Multi-Module Design Considerations, are Hanry Wagage, and I hope Hanry is in the audience. If not, he will be here soon. Joseph Ashcraft -- yes, Joseph is in the background, in the back. Nan Chien and Chang Li -- Nan is there? Okay. Sheila Ray, electrical. Angelo Stubbs, Ryan Nolan and Michelle Hart and Marie Pohida is next to me. Please don't ask me why Marie is sitting here and nobody else.

CO-CHAIR CORRADINI: It says Chapter 19?

MR. CHOWDHURY: Chapter 19, yes. So, anyway, the lead project manager is Greg Cranton, he is sitting here. And he will be helping us answer any questions there from the project management side. And I am the chapter project manager. Next is -- I believe you have seen the three-page safety evaluation report on Chapter 21. And it is really not an evaluation of Chapter 21 -- Applicant's Chapter 21 in SER 20 -- Chapter 21 by the staff. It's -- it merely points to where some of the shared systems have been evaluated by the -- by the staff, and findings have been documented. I would like to point out that Chapter 15 has not been issued yet. But there are a couple of sections in Chapter 15 the staff has confirmed that they have addressed some of the aspects of the design that's pertinent to Chapter 21.

So I have listed here 23 shared systems that the staff has evaluated -- in parentheses, in blue, are the number of NuScale power modules that are handled in those systems. And the failure of shared systems that are not safety related is considered in Chapter 15. Once again, Chapter 15 has not been issued yet. Chapter -- Section 15.0.0 contains the staff's review of categorization and classification of design basis events. 15.0.3 is radiological consequences of design basis accidents. That's why the staff's review is documented. And then, 19.1.4.9, Evaluation of Multimodule Risk, where staff discuss the multimodule risk, including internal and external events.

So the two concluding points, as far as Chapter 21 SE goes, is that the staff used standard review plan as well as prior applicable design-specific legal standards, DSRS, and then they used these guidance to -- to conduct their review and document their findings -- in those sections that I have already mentioned. And a list of reactor names. That's all the slides I have. And with that I will open this for any questions from members of the committee.

CO-CHAIR SUNSERI: So I have -- this is going to be an easy one for you. So on page 3 -- can you go back to page 3?

(Pause.)

CO-CHAIR SUNSERI: This is -- I love the table, actually. It clearly defines, you know, how many systems, or essentially, modules. But I have a question. So in a couple of cases you say six each for two independent subsystems. Can you clarify that nomenclature? Because in other cases you just say six nuclear power modules. What do you mean by six each for two independent subsystem? Like it is containment flooding and drain number 5.

MR. CHOWDHURY: Yes. Okay, so I am -- Angela, did you -- or, Hanry Wagage is going to explain the nomenclature there.

MR. WAGAGE: My name is Hanry Wagage. What it means it that there are two different systems. Each system has six units.

CO-CHAIR SUNSERI: Okay, so it's really just -- term number three is the same? Is that --

MR. WAGAGE: Number three --

(Pause.)

MR. CHIEN: Nan Chien from NRC. There is no information from NuScale, but my understanding is one system for turbine. There is one turbine building. There is only one turbine building.

CO-CHAIR CORRADINI: I think the question we're asking is, is the interpretation of six each of two independent subsystems the same as what you say under number 3?

MR. CHIEN: No, six

CO-CHAIR SUNSERI: So six should be 12 for number 3. Okay.

(Simultaneous speaking.)

MR. CHOWDHURY: But there are two turbine buildings.

CO-CHAIR DIMITRIJEVIC: So it's the same HVAC system.

CO-CHAIR SUNSERI: Oh, yes. Right. Okay.

MR. CHOWDHURY: There are two turbine buildings.

CO-CHAIR DIMITRIJEVIC: So there are two independent systems, right? Turbine HVAC systems?

PARTICIPANT: Yes.

MR. CRANSTON: This is Gary Cranston. There are two turbine buildings. Each turbine building serves six modules.

CO-CHAIR DIMITRIJEVIC: And has independent HVAC system.

MR. CRANSTON: They're independent. The two different buildings are independent -- one on each side of the reactor building.

PARTICIPANT: They haven't answered your

question.

CO-CHAIR SUNSERI: No, no, I understand. I understand that, though.

MR. MAXWELL: Number 3 should be six each for two independent subsystems as well.

CO-CHAIR SUNSERI: Okay, but I understand the distinction they're trying to make now. Just all the rest of it is in one building. This happens to be two, right? Okay. I don't have any other questions.

CO-CHAIR DIMITRIJEVIC: I have a question. It's not directly maybe connected to this, but I just noticed that we never discussed this control room habitability system. Then was that -- this loss of that which will result in that control room evacuation?

(Pause.)

MR. CHOWDHURY: Hanry?

CO-CHAIR DIMITRIJEVIC: And how was that treated? Would -- obviously the operators would trip before they evacuate control room, right? Even they may not be needed.

MR. CHOWDHURY: I am looking at the Chapters 6. Staff.

CO-CHAIR CORRADINI: You're asking NuScale?

CO-CHAIR DIMITRIJEVIC: Well, somebody.

(Pause.)

MR. MAXWELL: Again, Chris Maxwell from NuScale. A note, a loss of control room habitability system wouldn't require evacuation of the control room. It's rather a loss of all electrical power or a event -- toxic gas event -- would initiate control room habitability system, compressed air, pressurizing the control room to maintain the -- the environment within the control room suitable for habitation.

CO-CHAIR DIMITRIJEVIC: So loss of that system would require -- is that what you say?

MR. MAXWELL: No, not -- normal operating conditions, if the control room habitability system was rendered inoperable, then normal control room HVAC system would maintain humidity and temperature in the control room for the operators.

CO-CHAIR DIMITRIJEVIC: Okay. Which is number 6 versus number 8, if that is -- there are two redundant system, you say.

MEMBER BLEY: Well, no the -- the -- number 6 one is -- only comes into play if you have a problem with the normal HVAC, or you have contaminants come in.

MR. MAXWELL: Reversed. Number 6 is the

normal operating system. Number 8 --

(Simultaneous speaking.)

MEMBER BLEY: Oh, I am sorry. Yes.

MR. MAXWELL: Eight comes into play if we have toxic gas --

(Simultaneous speaking.)

CO-CHAIR DIMITRIJEVIC: Toxic releases. Okay, so let's discuss number 6. If they lost ventilation, would that require evacuation of the control room?

MR. MAXWELL: No, it would not require a control room evacuation.

CO-CHAIR CORRADINI: I think what he is saying is, 6 is normal operation. If that is compromised, then 8 would be activated to keep the control room -

CO-CHAIR DIMITRIJEVIC: I thought 8 is only if they have a toxic releases somewhere.

MR. MAXWELL: Normally the control habitability system -- number 8 is in a stand-by lineup with isolation valves presenting the compressed air from being released into the control room. So the normal control room HVAC system maintains the normal environment. It is only upon that loss of power, which renders the normal control room HVAC system inoperable -- or, on a toxic gas or fire -- that the valves in the control room habitability system then open and supply the compressed air into the control room.

MEMBER BLEY: How long does that last?

MR. MAXWELL: A minimum 72 hours.

MEMBER BLEY: That's a lot.

(Simultaneous speaking.)

CO-CHAIR SUNSERI: Okay, good. Anybody else?

(No audible response.)

CO-CHAIR SUNSERI: All right, well thank you for the presentation.

PARTICIPANT: Thank you.

CO-CHAIR SUNSERI: Now we are going to transition back to NuScale where we will discuss Chapter 19.5, Adequacy as Design Features in Functional Capability Identified and Described for Withstanding Aircraft Impacts. Open session.

(Pause.)

MR. BRYAN: Good afternoon. My name is Marty Bryan. I am the Licensing Project Manager for NuScale for 6 and 19.5, and today we are going to discuss adequacy of design features and functional capability identified and described for withstanding aircraft impacts. Our presenter will be Amber Berger.

MS. BERGER: Hello, I am Amber Berger, a civil structural engineer with NuScale Power.

CO-CHAIR SUNSERI: Could you pull that just a little bit closer to you?

MS. BERGER: Sure. So quick introduction and a background of what we will present today. So the NuScale Power Plant has been designed for the potential effects of a beyond-design basis large commercial aircraft impact, following the guidance of 10 CFR -- Title 10 CFR 50, 150-A, which requires that the reactor core remains cooled, or the containment remains intact -- and that the spent fuel pool -- or, the spent fuel remains cool, or that the spent fool integrity is maintained.

We have followed design-specific assessment requirements of Reg Guide 1.217, which is the guidance for beyond-design basis aircraft impact. And that endorses NEI 07-13, which is the methodology for aircraft impact assessment for a new plant design. We have followed NEI 07 guidance, taking no exceptions from the methodologies. And just as a note, the aircraft impact has actually informed the design of our plant, such as requiring five-foot thick walls for the exterior concrete walls of the reactor building.

The scope of our assessment --

CO-CHAIR CORRADINI: So just to make sure, the outside reactor walls are treated as potentially a -- the equivalent of a missile shield, as a robust barrier?

MS. BERGER: Yes.

CO-CHAIR CORRADINI: Okay. Thank you.

MS. BERGER: The reactor building has been assessed for three different areas of postulated aircraft impact. One is the physical damage of the airplane actually hitting the reactor building, exterior walls, and the roof. We also assessed for shock damage, which is the shock-induced vibrations on the structure systems and components housed within the reactor building, and we've assessed for fire damage, which is the aviation fed -- fuel-fed fire from aircraft impact.

The methodology, as I mentioned, is taken from NEI 07-13. The reactor building is our structure of concern, so the reactor building is what we've assessed for the aircraft impact. The reactor building houses nuclear power modules, the NuScale power modules, houses the ultimate heat sink and it houses the spent fuel pool. So, all of these systems inside the reactor building are what we're trying to protect.

The impact locations of the aircraft hitting the reactor building were screened based on NEI 07 guidance. The rad waste building, which is a building directly to the west of the reactor building, is considered as an intervening structure for the structural portion of our assessment. We did not -- conservatively, we did not consider it an intervening structure for the shock or the fire assessment. No credit is taken for the control building or the turbine generator buildings, which, on the next slide, we show a schematic view of our site plant, showing you the -- a site layout.

The building right in the middle, the long, rectangular gray building, that's the reactor building. That's the building of concern for us -- the structure of concern. To the west of that -- so, north is up and to the left. The building -- the small gray building low and to the left of the reactor building is the rad waste building. It's a heavily reinforced concrete building directly adjacent, about 30 feet from the reactor building. That's the building that we've used as an intervening structure for a structural assessment.

MEMBER BLEY: Can you -- whoever's got the mouse, can you point at that one?

MEMBER MARCH-LEUBA: Yeah, you --

MS. BERGER: Yeah.

PARTICIPANT: It's the lower gray box.

MEMBER MARCH-LEUBA: Amber, you can get the mouse and point.

MS. BERGER: Okay.

MEMBER MARCH-LEUBA: It's long enough.

MS. BERGER: Sure. That's great.

PARTICIPANT: Can you see it there?

MEMBER MARCH-LEUBA: Yeah.

MS. BERGER: Okay. Thank you. Okay. So, I'll start from the beginning. The reactor building is the long, skinny gray building.

MEMBER BLEY: You said that was heavily reinforced. The same as the reactor building, or?

MS. BERGER: Not quite as heavily reinforced, no.

MEMBER BLEY: How thick's the --

MS. BERGER: The control building walls are three-foot thick.

MEMBER BLEY: Three foot?

MS. BERGER: Yeah. With two layers of number 11, approximately. No shear tie reinforcing.

MEMBER BLEY: You look at the ceiling too, the roofs?

MS. BERGER: Not for aircraft impact, no.

MEMBER BLEY: Because the angle has got to be two --

MS. BERGER: Yes. Yep.

(Simultaneous speaking.)

CO-CHAIR CORRADINI: Could you say that louder? I didn't understand.

MEMBER BLEY: I asked about the roof and I asked why it didn't have to be considered. And the answer was because the angle is going to be a glancing blow.

CO-CHAIR CORRADINI: So, that -- the requirement is the roof does not need to be fortified in a similar manner as the walls?

MS. BERGER: Yes. Flat roofs.

MEMBER BROWN: That's the control room building, right? You're talking about the control room building right now?

MS. BERGER: Yeah. I believe so. The flat roofs --

MEMBER BLEY: Well, I was first going to talk about that, then I was going to ask about the reactor building that's --

MS. BERGER: Got it. Yes.

MEMBER BLEY: -- on the roof.

MS. BERGER: Yep.

MEMBER BLEY: Same thing?

MS. BERGER: Well, so, on the reactor building, we do have some sloped portions of the roof that are not completely horizontal. And we did assess the sloped portions of the roof for aircraft impact.

MEMBER BLEY: Okay. How thick's the roof on the reactor building?

MS. BERGER: It's four-feet thick.

MEMBER BLEY: Okay. So, that's pretty hefty. Okay.

CO-CHAIR CORRADINI: So, this is an impact barrier, but yet not pressure-retaining.

MS. BERGER: Pressure-retaining for what type of --

CO-CHAIR CORRADINI: Internal pressure. It's a leaky, strong building.

MS. BERGER: Well, we prevent fire from getting into the reactor building -- any pressurized fire from getting into the reactor building -- the main area of the reactor building.

CO-CHAIR CORRADINI: Okay.

MS. BERGER: And so, it doesn't need to be pressure-retaining for an aircraft impact perspective.

CO-CHAIR CORRADINI: Okay. Because the reason I'm asking like that is the next question is, if it's pressure-retaining, does it have a leak rate specification?

MS. BERGER: We don't need it to be pressure-retaining.

CO-CHAIR CORRADINI: Okay.

MS. BERGER: Yeah.

CO-CHAIR CORRADINI: Okay.

MS. BERGER: So it's not.

CO-CHAIR CORRADINI: So, how do you prove that what's outside can't get inside?

MS. BERGER: This is sort of later on in the slides. We can start right now, if you'd like.

CO-CHAIR CORRADINI: That's fine. We'll wait. No, no. You can tell us to wait.

MS. BERGER: Okay. So I'll continue -- actually, go back to the previous slide, please. Just to finish out this slide, the turbine generator buildings are the long, larger brown buildings north and south of the reactor building. Those will be steel buildings. We did not credit those as intervening structures, so we ignored those in our assessment. And also, to the --

MEMBER BLEY: So, the plane can come

directly in from the sides there and hit the reactor building.

MS. BERGER: Yes. Yep.

MEMBER BLEY: You took no credit for any --

MEMBER BROWN: What did she say?

(Simultaneous speaking.)

MEMBER BROWN: -- buildings.

CO-CHAIR CORRADINI: The yellow buildings aren't credited.

MEMBER BLEY: It's like they aren't there.

MEMBER BROWN: Okay. For aircraft, it's -- I got it. Thank you.

MS. BERGER: Got it.

CO-CHAIR SUNSERI: So, let me just pause for just a second. I don't know where the line between sensitive information and non-sensitive information is on this topic is, but don't let us ask you questions to expose sensitive information and --

MS. BERGER: Appreciate it. Thank you.

CO-CHAIR SUNSERI: Okay. Thank you.

CO-CHAIR CORRADINI: But we leave it to you to tell us that it's not appropriate.

MS. BERGER: Okay. Will do. Thanks.

CO-CHAIR CORRADINI: So, now I have a

question.

MS. BERGER: Okay.

CO-CHAIR CORRADINI: So, because of the low profile of the design, does that essentially affect how Dennis was asking the question about angle of attack, or angle of --

(Simultaneous speaking.)

MS. BERGER: The low profile of the reactor building? The height of the reactor building, essentially?

PARTICIPANT: Smaller target.

MS. BERGER: Well, we've assessed the walls of the reactor building for aircraft impact strikes along all four sides of the reactor building.

CO-CHAIR CORRADINI: But the fact that it's a low profile, you don't have to worry about it coming in, in unusual orientations.

MS. BERGER: No. We follow the NEI guidance.

CO-CHAIR CORRADINI: Okay.

MS. BERGER: So --

CO-CHAIR CORRADINI: Okay.

MS. BERGER: Yeah.

CO-CHAIR CORRADINI: Okay. Then I'll ask

it this way. Does the NEI guidance have a height of the building aspect too, so that if I have a low height I have to follow one set of procedures for analysis versus a higher height?

MS. BERGER: There's no specific stipulation in the guidance for that.

CO-CHAIR CORRADINI: Okay. I couldn't remember.

MS. BERGER: We've assessed the building above grade. All of the building above grade is susceptible to aircraft impact.

CO-CHAIR CORRADINI: Okay. Thank you.

MS. BERGER: Yeah.

MEMBER MARCH-LEUBA: And as a mechanical engineer, would have been cheaper to dig 20 foot deeper and not have any vertical walls?

MS. BERGER: Yeah. We looked into that, actually. We did an aircraft impact design decision paper and we considered that.

MEMBER MARCH-LEUBA: If there is no vertical walls, you don't have to do the assessment, right?

MS. BERGER: Sure. Yes. But then your entire building is low grade. Yeah. There's advantages and disadvantages, and we did do those.

MEMBER MARCH-LEUBA: And this is cheaper, 4-foot reinforced concrete?

MS. BERGER: Yes.

MEMBER MARCH-LEUBA: I thought digging was cheap.

PARTICIPANT: No.

MS. BERGER: It's the operational impacts that are more expensive. Yeah.

MEMBER BLEY: And just so you don't forget, we covered the control building --

MS. BERGER: Yep.

MEMBER BLEY: -- the two turbine halls, and the you got two more buildings at the back end, there.

MS. BERGER: Well, so, actually, the control building is this brown building that's sort of behind the reactor building that's hard to see in this graphic. The control building, again, is -- we did not use it as an intervening structure.

MEMBER BLEY: Okay.

MS. BERGER: Just pointing that out. And then adjacent to that's the central utility building, not considered in our analysis.

MEMBER BLEY: Oh, that's not considered either.

MS. BERGER: Yeah. Yeah. So, really, only -- we only considered intervening structure on one side of the building, that's the red waste building here on the west, and we only considered that for the structural assessment. Next slide.

So, the assessment results for physical damage, we assessed local damage of the aircraft hitting the exterior walls of the reactor building and the sloped portion of the roof and found there's no perforation of the aircraft and no scabbing effects that would cause --

MEMBER BLEY: How detailed an analysis did you do on that?

MS. BERGER: For structures?

MEMBER BLEY: Yeah.

MS. BERGER: Let's see here. Going down. Is it in this slide? Oh, second -- next slide. We did a detailed finite element analysis model, considered --

MEMBER BLEY: So not even the rotors get through the building.

MS. BERGER: No.

MEMBER BLEY: And you were right at scabbing, so don't forget that. Tell us about that.

MS. BERGER: Talk about the scabbing?

MEMBER BLEY: Yeah.

MS. BERGER: We had done an assessment for scabbing and found that it doesn't cause an issue to meet the 10 C.F.R. guidance or requirements. So, that was the local assessment. We also did the global assessments using our detailed finite element model, using -- and using the NRC-specific force-time history.

The reactor building external walls prevent physical damage from entering the reactor building, so we completely keep the plan out of the building proper. There are no internal missiles for secondary impact, which would be a scabbing issue. We have no impact to the containment boundary or the containments' boundaries. Our spent fuel --

MEMBER BLEY: There's nothing on any walls that could get knocked loose or be sufficient?

MS. BERGER: That's part of the shock assessment.

MEMBER BLEY: Okay. I'll wait for that.

MS. BERGER: Thank you. And our spent fuel pool is protected as it's inside the reactor building and it's completely below grade, so there's no issues -- physical damage to the spent fuel pool. And then, also, the reactor building crane trolley, it cannot be dislodged from its rails.

MEMBER SKILLMAN: Is that because of the seismic clamps that hold that bridge and trolley in place?

MS. BERGER: Not entirely. It's really just the -- we've done -- we did an analysis to determine what the displacement on the rails would be from the aircraft hitting the exterior of the reactor building. And that displacement geometrically is less than the length of the bridge rails.

MEMBER SKILLMAN: Thank you.

MS. BERGER: Yeah. And pertaining to the shock damage, so the aircraft impact causes short-duration, high-acceleration, high-frequency vibrations, and this would be the sort of damage that would cause equipment to malfunction inside the reactor building, cause equipment to fall off walls.

For the core cooling, we do an at-power and shutdown scenarios. That was considered in our assessment. For our plant, there's no active equipment that's required for -- to successfully keep the cores cool.

MEMBER BLEY: But would some of the equipment be damages?

MS. BERGER: There's one strike out of 17 strikes around the reactor building. There is one strike on the west side of the building where the rad waste building is. Taking no credit for the rad waste building as in intervening structure, some of the shock from that impact does reach into the room where the core cooling equipment is on the 50-foot elevation, two floors below grade. And when the shock footprint touches on that equipment, we assume that it loses its function.

MEMBER BLEY: Okay.

MS. BERGER: It loses its ability to perform that function. And then the general jolt for our core cooling is the adequate heat removal shown for all strikes. The cores are -- remain cool based on our passive designs.

MEMBER SKILLMAN: For that portion of the analysis, is the equipment so fragile that, even with that shock loading, it would fail? Or do you just simply assume that it fails?

MS. BERGER: We assume -- it's part of the guidance you assume that it fails. There's two levels of shock. There's near shock, which is -- mechanical is more susceptible to the near shock. And if near shock -- or, sorry. Vice versa.

CO-CHAIR SUNSERI: There's somebody on the phone line that is open. Could you please mute your phone line?

MEMBER SKILLMAN: You were saying, two levels of shock?

MS. BERGER: Yeah. Two levels of shock, and I said it wrong the first time. I apologize. The far shock is effective to mechanical equipment. If the far shock profile has the potential to reach that equipment, the assumption in the guidance is you lose the capacity of -- or, the capabilities of that system.

MEMBER SKILLMAN: Thank you.

MS. BERGER: And then, additionally, spent fuel for the shock damage, the spent fuel pool integrity is maintained for all strikes. The shock profiles don't actually even reach the spent fuel pool. And next slide.

So, fire damage. The results for the fire damage assessment, we've designed and located three-hour fire barriers and three-hour 5 psi fire barriers to prevent the propagation of fire into the reactor building. We've also designed and located 5 psi fast acting dampers inside of the reactor building HVAC air intakes where the louvers are on the exterior of the building. Those prevent those -- blast dampers prevent pressurized fire from getting into the reactor building.

MEMBER BLEY: What activates them?

MS. BERGER: Pardon?

MEMBER BLEY: What activates them?

MS. BERGER: The pressure.

MEMBER BLEY: The pressure from the blast? Okay.

MS. BERGER: Yes. Yeah. We also have on the exterior of the building, at all exterior wall penetrations, we've placed concrete shrouds -- five-foot thick, heavily-reinforced concrete shrouds, to protect pipe penetrations, HVAC penetrations, prevents physical damage from getting inside those -- into those -- through those penetrations, and it also prevents fire from propagating into the reactor building.

MEMBER BLEY: I don't remember. At each of the turbine halls, the main steam lines, do they go underground and then come up in the building?

MS. BERGER: They don't. They come out the level at-grade, and then they go straight across that grade, yes. Well, they don't go straight across. They come out of the building and then they have to go down and then across to get underneath the shrouds that we have located.

MEMBER BLEY: Okay. So they could be ripped up by this thing.

MS. BERGER: The main seam lines?

MEMBER BLEY: Yeah.

MS. BERGER: Yeah. Past the shrouds. Past the exterior of the reactor building, yeah. And then the last bullet for the fire damage assessment is, on that -- on all four corners of the reactor building, we have personnel doors. Mostly are for emergency egress. Fire can get inside of those personnel doors.

But then we have stairwells at each door and five -- three-hour fire barriers so that unpressurized fire can get into stairwells. It propagates up and down, vertically, through the building, but it stays contained within the stairwell, so it never gets into the reactor building main area. And so, we keep fire out of the fuel -- the spent fuel pool area and the modules area -- the operating module bay.

And then another part of our assessment is that required -- operator action is required prior to the impact of an aircraft, and that is part of the methodology of any --

MEMBER BLEY: You must have some kind of equipment hatches somewhere for letting -- bringing in new equipment.

MS. BERGER: We have the west end of the building and adjacent -- between the reactor building and rad waste building, we have a barrier, approximately, 20-foot-by-20-foot door that has a door that is made of -- it's steel-concrete composite and it's been designed for the physical damage of an aircraft impact, and it will be sealed to prevent pressurized fire from entering the reactor building, too. But that's the largest -- other than the personnel doors --

MEMBER BLEY: What's the longest it sits open if you're moving things in and out?

MS. BERGER: I don't know the answer to that. I'm sorry --

MEMBER BLEY: So, it could be a substantial period of vulnerability.

MS. BERGER: No. I wouldn't expect so because it's -- that door also separates the rad boundary, right? So, we don't want that door to be open -- it would not be open for normal operations or anything. We need that --

MEMBER BLEY: So, if you needed large

equipment and you shut down all the reactors before you brought in --

MR. RAD: So, there's a period of time between when the plant is warned of an impending impact that operator actions occur and the plant is placed in certain conditions. And we would expect that this would be one of the actions that would be taken, would be to close this door.

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: I'm confused. Who warns you? How do you know it's coming?

MR. RAD: They.

MEMBER MARCH-LEUBA: Who warns you that an --

MR. RAD: Okay. Right. So, that's a matter of plant operations, right? So, the Federal Government warns you that there's an impending aircraft impact. But that's --

PARTICIPANT: Like the FAA?

MR. RAD: Yeah. Yeah.

MEMBER MARCH-LEUBA: The airport tower?

CO-CHAIR SUNSERI: There's a protocol. There's an industry protocol.

PARTICIPANT: Yeah.

MEMBER BLEY: Unless you're the first one.

Go ahead.

MS. BERGER: Okay. Next slide. In conclusion, the design and functional capabilities of our plant provide adequate protection to public health and safety. NuScale Plant meets 10 C.F.R. 5150 regulations by both maintaining containment integrity and maintaining core cooling capability. Only one of these is required to meet the guidance. We also maintain spent fuel pool integrity.

And then, for most postulated aircraft impact strikes, spend fuel pool -- active spend fuel pool cooling is maintained, meeting all four of the 10 C.F.R. requirements.

MEMBER BLEY: Which implies there's some of these strikes that wipe out your cooling and fuel pool.

MS. BERGER: Just that one strike. One out of 17 strikes affects the spent fuel pool cooling equipment.

MR. CHOWDHURY: From shock that we don't take credit for that equipment.

MEMBER BLEY: Okay.

MS. BERGER: Yeah.

MEMBER BLEY: But you would still have the main pool cooling --

MS. BERGER: Still have -- well, we still have our ultimate heat sink and we still have --

MEMBER BLEY: That's what I meant.

MS. BERGER: Yeah. Yeah. And that is the rest of our slides and acronyms. Appreciate any questions or clarifications.

CO-CHAIR SUNSERI: Members, any additional questions?

MS. BERGER: Great.

CO-CHAIR SUNSERI: Thank you.

MS. BERGER: Appreciate it. Thank you.

CO-CHAIR SUNSERI: We'll transition for the staff to come up with their safety evaluations. Whenever you're ready.

MR. CRANSTON: Good afternoon. I'm Greg --

CO-CHAIR SUNSERI: Your microphone.

MR. CRANSTON: I think I blew mine out.

CO-CHAIR SUNSERI: Hey, I mess it up.

MR. CRANSTON: Good afternoon. I'm Greg Cranston, project manager for the NuScale project. We're here for the continuation of Chapter 19 with 19.5, and Ryan is going to make the presentation.

MR. NOLAN: All right. Thanks, Greg. My name's Ryan Nolan. I'll be giving the staff's presentation on our Safety Evaluation Section 19.5, Aircraft Impact Assessment. You can go to the next slide. The 51.50 requires a design-specific aircraft impact assessment and incorporation of design features to ensure that the reactor core remains cool or containment remains intact, and spend fuel pool cooling or spent fuel pool integrity is maintained.

The introduction of the Safety Evaluation tries to step through what it is we do in the licensing review and what we're making a finding on. And in this case, we're looking at -- that the Applicant identified design features, described design features, and stated how those design features meet the assessment. The assessment is not submitted to us. We don't review the assessment or the adequacy of the design features as part of our licensing review. We just ensure -- it's more of a 51.50 Bravo is the content of application. That's what our licensing review is on in 19.5. And so --

CO-CHAIR CORRADINI: So you're reviewing that they followed procedures.

MR. NOLAN: We review to ensure that they've identified the design features in the licensing basis document in 19.5, that they've described those design features.

CO-CHAIR CORRADINI: All right.

MR. NOLAN: So, as far as the design features, they're typically broken down into three areas: fire protection, structural, and systems. For example, fire protection, they would identify -- they do identify which three-hour walls, 5-psi barriers are credited as design features.

The structure, I think the main one here is the exterior reactor building walls, that they're identified as key design features. You know, dimensions, rebar size, things like that. And then as far as systems, they would identify any of the systems that the assessment relies on to meet the requirements. That would be decay heat removal system, ability to trip the reactor, things like that.

As far as how they meet the requirements, you know, we ensure that the systems or SSCs that were identified as key design features, that they are linked to the requirements. And in this case, NuScale has committed to meeting the requirements by ensuring the reactor core remains cooled and containment remains intact, and then spent fuel pool integrity is maintained. We review to ensure that the assessment looked at shock damage, that they followed the appropriate guidance, and that the assessment was performed by qualified individuals.

There's one confirmatory item, and this was more of just a clarification on the use of instrumentation and monitoring, and whether it was a key design feature, or if it was just there as a monitoring aspect of it. And so, this is just to identify that there is a confirmatory item. We reviewed the RAI. We found it acceptable. And so, we'll close it out when we see your Provision 3 of the FSAR.

CO-CHAIR CORRADINI: I don't think I understand the RAI. Can you help me again?

MR. NOLAN: Sure. A lot of our Phase 1 RAIs were more clarification on the description of systems and are they considered a key design feature or is it kind of just there as part of normal operations. And the clarification that we received from NuScale is that the decay heat removal system doesn't rely on instrumentation. So, it's a clear, passive system. Doesn't rely on any external power. The instrumentation is there just for monitoring.

CO-CHAIR CORRADINI: Okay. Thank you.

MR. NOLAN: It's not needed to meet the core cooling requirement.

CO-CHAIR CORRADINI: Okay. I got it.

MR. NOLAN: So, in conclusion, we found that the Applicant adequately identified and described the design features and that it's reasonably formulated and performed by qualified individuals.

CO-CHAIR SUNSERI: That it?

MR. NOLAN: Yeah.

CO-CHAIR SUNSERI: Members, questions? All right. Well, thank you very much for the presentation. At this point, we have completed all the prepared presentations -- or, we've gotten through all the prepared presentations, so now I'll open up the floor to public comments. We'll first start with the members of the public that are in the room. And while we are asking for their comments, we will also open the phone line.

So, any members of the public in the room that would care to make a statement? Okay. We'll turn to the phone line now. Any members of the public on the phone line that would care to make a statement? Now is the opportunity. And if there is a member out there, they could just acknowledge that the phone line was open, we would appreciate that.

PARTICIPANT: Phone line is open.

CO-CHAIR SUNSERI: So, no comments to be had?

MR. SNODDERLY: If there are no comments, we're going to close the public line at this time.

CO-CHAIR SUNSERI: Okay. All right. So, the public comments are done.

Subcommittee, let's go around the table now for the open portion of this meeting and hear from individual members on any comments you'd like to make. We'll start with Vesna.

CO-CHAIR DIMITRIJEVIC: I don't have any additional comments now.

CO-CHAIR SUNSERI: Dennis?

MEMBER BLEY: Yeah. I'm still -- just continuing from yesterday, but, you know, the big things that are sitting there for me area the ECCS valves and the passive system performance. Excuse me. I've heard the phrase, investment protection issue, from a number of people. And just a caution, you know? We've seen a lot of times when these things we call investment protection have led to events that have occurred, they quickly morph into safety concerns and everybody gets spun up.

The other thing is, if I'm -- I sell you a plant and we scuff up -- we get the plant in a situation that makes you shut down all your reactors, the executives of that utility aren't going to be very happy with their vendor friend. So, thinking hard about those things to make sure they're very unlikely, yeah, makes -- probably worthwhile.

MEMBER MARCH-LEUBA: No further comments.

CO-CHAIR SUNSERI: Charlie?

MEMBER BROWN: No further comments other than what I had earlier.

CO-CHAIR SUNSERI: Okay. Joy?

MEMBER REMPE: So, I -- again, I believe today we talked about a concern about exactly which shared systems failures could require modules to be shut down because they've lost that share system. And I believe the Applicant did agree to look into whether they had to provide such a list and whether it is part of the Tier 2 required documents.

And so, I'm interested in seeing that answer. And maybe -- again, if it's not required, maybe that's something where we have a gap and it should be required, and it's something for us to think about, so I'd like to have that follow-up.

And then, I think the comments I had yesterday, one of them got addressed. But the question I had about a list of what other instrumentation would give the operators an indicator that -- or, would we, not the operators, excuse me, would cause the module protection system to generate a signal if the -- if you didn't have the pressurized water level, and I still would like to hear that answer, too. And so, I think that's it.

MEMBER BROWN: Matt, when you're done.

CO-CHAIR SUNSERI: Thank you. I was just taking a note here --

(Simultaneous speaking.)

MEMBER BROWN: Yeah. I just want to backtrack for a minute. I'm not sure how I covered this during the multi-module design consideration relative to integration of and the lack of an architecture in anything that showed that. One of the things -- or, two of the things, actually, that I kind of expected when I ready Chapter 21 was a discussion of how abhorrent behavior or abnormal behavior, such as interlocks that would be to ensure certain things.

You couldn't start the system in another plant without turning it off in the previous -- whatever's required, as well as an annunciations and/or alarms or warnings that you might say is something inadvertently energizes itself in another one of the modules. And that's my concern on the whole chapter is that it's sparse on how those shared systems are integrated into the overall control and operations in the overall facility in the plant instead of just operators are going to control everything and it's going to be just fine. I have -- very skeptical on that point.

That's all. I just wanted to amplify that -- that's what I meant by architectures. How they were integrated, how you interlock them, what warnings and indications and monitoring you provide that give operators a heads up that, hey, they're heading into some situation or condition that they're not aware of.

CO-CHAIR SUNSERI: Okay.

MEMBER BROWN: That's all I have.

CO-CHAIR SUNSERI: Yep. Thank you, Charlie. Mike?

CO-CHAIR CORRADINI: I want to ask Charlie. So, you're saying it doesn't exist or you couldn't find it in that chapter?

MEMBER BROWN: It doesn't exist. He said there's no diagram. And if you read the chapter, it doesn't talk about specifics of how you ensure that those systems are alerted, alarmed, interlocked, or what have you. At least, I didn't find it.

CO-CHAIR SUNSERI: Well, he said it exists in discrete drawing --

(Simultaneous speaking.)

MEMBER BROWN: It's some type of P&IDs of some kind, if you go take 27 drawings and try to stitch them all together yourself in the backroom.

CO-CHAIR CORRADINI: But what I'm trying to understand is this a recommendation that you want something added to the DCA, or?

MEMBER BROWN: I think the -- my personal opinion, if -- this is my personal opinion. If I had been doing this, I would've provided an architecture diagram in Chapter 21 identifying how interlocks and other alarms would be used to ensure operating -- one control room, three operators, you got to give them some operation instead of having it.

And that's just based on experience of watching at least one aircraft carrier trying to operate eight reactor plants and steam systems, and some of it with various means of cross-connecting. So, that's what I would've expected to see in the chapter, and it was absent.

MEMBER BLEY: As to Matt's point, you know, you know, they said they do have, you know, P&IDs, which we haven't seen, which would show you the valves one might use. They've said -- they haven't decided whether they're going to control that administratively or by interlocks --

(Simultaneous speaking.)

CO-CHAIR CORRADINI: For the various shared systems.

MEMBER BROWN: For the shared systems, yeah.

MEMBER BLEY: And I -- you know, from Charlie's point, I don't think we can insist we can insist they have that in the DAC.

PARTICIPANT: That's fine.

MEMBER BLEY: But we might want to raise it as something to watch out for later at COL stage.

MEMBER BROWN: Okay. Yeah. I'll recognize that we can't dictate what they put in the DAC.

CO-CHAIR CORRADINI: Well, you can try. But I just want to sure I understood what you're asking.

MEMBER BROWN: I've never been hesitant. We can discuss that later.

CO-CHAIR CORRADINI: So, for the full committee meeting, I think we need -- three's only two open items as far as I remember for 19.2. I think it would benefit the members of the committee that aren't here today to hear about those open items and their path to resolution. I think one is relative to the bioshield redesign, which affects a number of things. But I think at least from the standpoint of hydrogen control, this is something that the full committee ought to hear.

And the second thing is, relative to their path forward, relative to, I think it was, how the source term is affected. So, I think those two things ought to be in the full committee meeting as part of the presentation to the -- to us and the other members. That's it.

CO-CHAIR SUNSERI: Dick?

MEMBER SKILLMAN: The concerns that I have about the load drop I already expressed yesterday, so I don't think it is constructive to waste our precious resources now. But that's where the bulk of my concerns lie. Thank you.

CO-CHAIR SUNSERI: Thanks. Ron?

MEMBER BALLINGER: No additional questions.

CO-CHAIR SUNSERI: okay Before I go to Steve, I'm going to come back to Joy.

MEMBER REMPE: Oh, I just wanted to follow-up about Mike's comment about open items. And one thing I forgot to mention today was there's an open item, also, about the electrical penetrations and their effect on source term, and I think that's one I'd like to hear about, too.

CO-CHAIR SUNSERI: And before I close, I'll go to Steve with any comments you want to make.

DR. SCHULTZ: Just a general comment on the aspects of Chapter 19. And it is a general comment. And that is, each of the presentations tend to start off discussion about how best estimate evaluation is going to be done because we're talking about probabilistic risk assessment approach. But somewhere along the line, each of the discussions, we get into handling uncertainty or doing evaluations. And we get to the difference between the goal, the margin, and get into general discussions about how we have plenty of margin, so we don't have to worry about the details of the specific calculation.

And I think looking toward -- if you will, looking toward the long-term, the future, it's going to be very important before all of this gets wrapped up. The best-estimate evaluation is very, very clear. And the uncertainties are more clear than what they are today, and that a clear engineering evaluation can be demonstrated so that the margins are really known, not qualitatively expressed.

CO-CHAIR SUNSERI: Okay. Very good.

CO-CHAIR DIMITRIJEVIC: Matt?

CO-CHAIR SUNSERI: Yes, Ves. When you asked me first time, you -- I told you I have any additional comments on 19.5. Since we are summarizing to this --

CO-CHAIR SUNSERI: Yeah. If I wasn't clear on that, I apologize. Yeah. The whole --

PARTICIPANT: The day.

CO-CHAIR SUNSERI: The day, yeah.

CO-CHAIR DIMITRIJEVIC: So I can -- I mean, I summarized yesterday, but I had --

(Simultaneous speaking.)

CO-CHAIR SUNSERI: And you can have another -- yeah. This is like bridge. We keep bidding, you know? Go ahead.

CO-CHAIR CORRADINI: Go ahead.

PARTICIPANT: What did she just say?

CO-CHAIR DIMITRIJEVIC: I just said --

(Simultaneous speaking.)

PARTICIPANT: It's like bridge. We just keep bidding.

CO-CHAIR DIMITRIJEVIC: That's all right. The thing is that what I still have left on the -- and, I mean, I will bring several things up on the alpha committee. First I have these comments on the SER itself. There's the three statements where they're technically questionable, and one of them is just technically wrong, which I would like to see change before we --

The other thing is that we have -- since then, I learn more about, you know -- even today, I learn more about the certain aspects of the ECCS valves and passive heat removal, and analyzing single units. So one of my concerns was that the analysis of the PRA single unit is not taking in account that there is multiple units and that multiple units could be affected.

I don't think that's going to be my big concern about what I heard today. How much time do I have and how did answer? But that's definitely true. This just analyzes the single units and presence of other units, most not consider in the sum of the aspects of the PRA. So, I don't think that will make difference. It would not be my -- I bought it up yesterday, but today I don't think it's going to be my big concern.

My bigger concern is that I don't think the sensitivity and uncertainties are completely done, and I'm especially concerned about uncertainties in the passive heat removal because it's very small, and uncertainty in the total result is very small. That would not affect conclusions, you know? Even if uncertainty was done, you know, be the less optimism. It will not change the conclusion. However, these should be showing high uncertainly. That's something you would expect that we didn't see it. It's unusual, so I may make comment on that. I'm not sure how to face it.

My other thing is that things which are extremely important from the safety valve, the ECCS values, there was some non-conservative assumptions made, which I learned today. And that those non-conservative assumptions, again, are not going to impact the results in the sense that --

CO-CHAIR CORRADINI: Can you remind us what you mean by the non-conservative assumptions?

CO-CHAIR DIMITRIJEVIC: Well, you know, we were talking about the trip valve versus the valve itself. And the trip valve has a much higher fail probability than that was assumed about ECCS valve fail probability. However, the trip valve is combined with these assumptions that there is some probability the valve will open on delta low delta P and it doesn't need signal and trip valve.

But that's assumption and pilot valve was assume, and it wasn't consider in common cause. It could have a impact on the total result. It's a big difference in the failure rate. So there is some non-conservative modeling, which would not impact results, you know, because results are so low.

CO-CHAIR SUNSERI: Yeah. I thought -- but I thought I heard them say, though, that -- or, like, the pilot valve versus the main valve, that there were different industry-accepted frequencies for those events that --

(Simultaneous speaking.)

CO-CHAIR DIMITRIJEVIC: Yeah. It's the trip valve, which never had frequency. It was much higher frequency. It was -- that's considered always necessary because it was, as we will say, PRA ended. So, they say, okay, this trip valve fails, there is still probability the valve will open on the low delta P. And because they don't know what the probabilities, it was assumed that it's 0.1.

So, now this valve was put in common cause group, and that's a high probability. However, it's multiplied by this 0.1 for two valves, which is 10 to minus 2, lowering impact of that valve significantly.

CO-CHAIR CORRADINI: But I think -- if I

might just -- this is what I -- I think we're repeating what we want to see for the full committee. But I think what I'd ask is, we want to understand better how they got to a number, which you view is non-conservative, so that we hear in the full committee with a better explanation.

CO-CHAIR DIMITRIJEVIC: Yeah. I understood that today. I got information.

CO-CHAIR CORRADINI: Okay.

CO-CHAIR DIMITRIJEVIC: So, I have it, information, and I'm really not comfortable with this. But the question is, so, okay. So, maybe they're not conservative, but the 10 to minus 9 and 10 to minus 4, we are talking hundred thousand -- you know, five orders of magnitude.

So, the question is -- I mean, you know, I will point, but this is one of the things which I'm still uncomfortable. There are a lot of things I'm uncomfortable, but you do the sensitivities analysis and you only concentrate on one parameter, then you don't know how combination of your different parameters. So, you say, okay. I will fail or, you know, I will increase common cause.

But what if that goes in combination with increasing this failure raise. There is no combination of sensitivity cases, and some of them can be combined and they show different results. That's on my major commons which I will, do, do, do, write down, and try to break up --

(Simultaneous speaking.)

CO-CHAIR CORRADINI: But I think the first one you brought up -- I want to make sure we're on the same page. The first one you brought up, though, I -- yesterday, at least, my recommendation is that, for the full committee, the Applicant and the staff have got to at least explain their view of how -- what they find is acceptable relative to the reliability and the effect of degradation. I think we've got to have an exposition of this.

The actual valve qualification is later. But to answer your questions about what's non-conservative, we've got to have that -- at least discuss with, again, the full committee, just so we're all on the same page with what we sense we hear.

CO-CHAIR DIMITRIJEVIC: Okay.

CO-CHAIR CORRADINI: Okay?

CO-CHAIR DIMITRIJEVIC: Well, I -- you know, the question is also, how do we feel about making comments which will not affect the main conclusions. I mean -- you know?

CO-CHAIR CORRADINI: So, let me reverse it. Part of the reason I'm asking these questions is I want to make sure that we give the right guidance to the staff and the Applicant for the full committee. So, for example, I wouldn't look forward -- just to get out of the box, I wouldn't look forward for anything in 19.3 to 19.5. I'd rather they concentrate their time on the open items in 19.1, the open items in 19.2, these questions about valve reliability estimates for their things, versus just, essentially, going through details that aren't necessary.

CO-CHAIR DIMITRIJEVIC: Well, you know, for that one the other question is also this passive, which I want to say, they gave us the slice, which we saw in October, which says, okay, this is probability that we have inadequate heat removal. And they say, okay, we did the Monte Carlo, blah, blah, blah, and we came out with these two numbers -- and I run factories, too. That's mean, A, we almost know exactly what that number is because this is very small --

(Simultaneous speaking.)

CO-CHAIR CORRADINI: Are we in closed session? You're not talking about anything we need to be in closed session about, are we?

CO-CHAIR DIMITRIJEVIC: Are we in -- no.

CO-CHAIR SUNSERI: It doesn't say.

CO-CHAIR DIMITRIJEVIC: No.

CO-CHAIR CORRADINI: Okay.

CO-CHAIR DIMITRIJEVIC: I didn't say --

CO-CHAIR SUNSERI: Well, what's it say that it's on?

CO-CHAIR CORRADINI: Yeah. We're closed. We're closed.

(Simultaneous speaking.)

CO-CHAIR CORRADINI: So, why don't we wait --

CO-CHAIR DIMITRIJEVIC: Well, I didn't not to use any information which I did not use in the open session.

CO-CHAIR CORRADINI: Why don't we wait on discussion?

CO-CHAIR DIMITRIJEVIC: All right.

CO-CHAIR SUNSERI: All right. Very good. Well, so I think -- Jose? You got any -- we're going to wrap this up. So, I think we had a very good session today. I mean, the presentations informed our thinking quite a bit. And, obviously, we have some things to consider as we formulate our final -- our thoughts on what -- you know, areas impact reasonable assurance or not as far as safety goes. And we look forward to the discussing these items further in the full committee meeting. So, thanks to everyone that made a presentation today. And at this point in time, we're going to go out of the open session and into the closed session. Well, I suggest we take a 15-minute break till 2:15, and then we'll start with the closed session.

CO-CHAIR CORRADINI: And then we rely on the Applicant and the staff to tell us who's allowed in the room.

CO-CHAIR SUNSERI: That's correct. All right. We're at recess until 2:15.

(Whereupon, the above-entitled matter went off the record at 1:59 p.m.)

May 09, 2019 Docket No. 52-048

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

SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Subcommittee Presentation: NuScale FSAR Chapter 19, Probabilistic Risk Assessment and Severe Accident Evaluation," PM-0519-65372, 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 May 14 and 15, 2019. The materials support NuScale's presentation of Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation," of the NuScale Design Certification Application.

Enclosure 1 is the nonproprietary presentation entitled "ACRS Subcommittee Presentation: NuScale FSAR Chapter 19, Probabilistic Risk Assessment and Severe Accident Evaluation," PM-0519-65372, Revision 0.

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

If you have any questions, please contact Rebecca Norris at 541-452-7539 or at rnorris@nuscalepower.com.

Sincerely,

Zackary W. Rad
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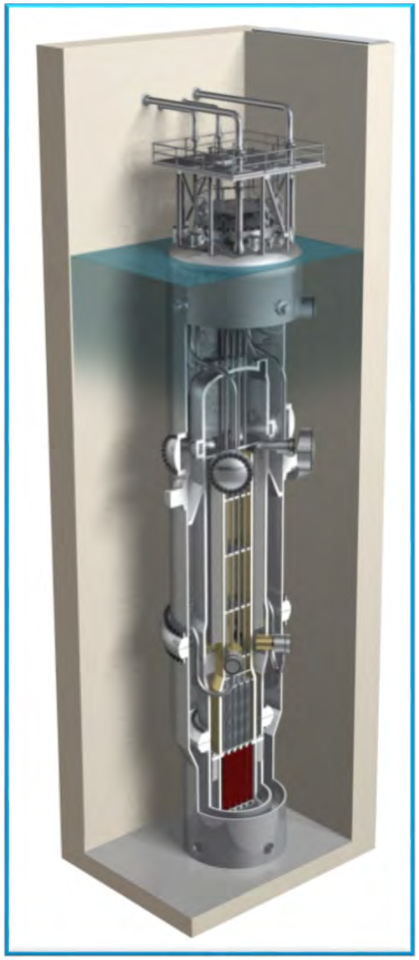
Enclosure 1: "ACRS Subcommittee Presentation: NuScale FSAR Chapter 19, Probabilistic Risk Assessment and Severe Accident Evaluation," PM-0519-65372, Revision 0

Enclosure 1:

"ACRS Subcommittee Presentation: NuScale FSAR Chapter 19, Probabilistic Risk Assessment and Severe Accident Evaluation," PM-0519-65372, Revision 0

ACRS Subcommittee Presentation:



NuScale FSAR

Chapter 19

Probabilistic Risk Assessment and Severe Accident Evaluation

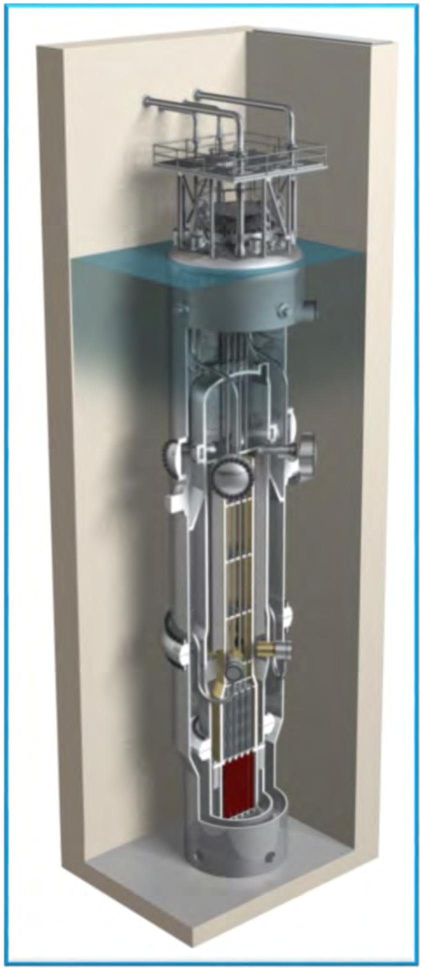
May 14 and 15, 2019

PM-0519-65372
Revision: 0

Chapter 19

Section	Title	Comment
19.0	Probabilistic Risk Assessment and Severe Accident Evaluation	Overview
19.1	Probabilistic Risk Assessment	Level 1, 2
19.2	Severe Accident Evaluation	Thermal hydraulic & phenomenological analyses
19.3	Regulatory Treatment of Nonsafety Systems	No RTNSS SSCs
19.4	Strategies and Guidance to Address Loss of Large Areas of the Plant due to Explosions and Fires	Addressed in Chapter 20
19.5	Adequacy of Design Features and Functional Capabilities Identified and Described for Withstanding Aircraft Impacts	Overview

ACRS Subcommittee Presentation:



NuScale FSAR Chapter 19.0 and19.1

Probabilistic Risk Assessment

May 14, 2019

PM-0519-65372
Revision: 0

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Section 19.0: Probabilistic Risk Assessment and Severe Accident Evaluation

- Developed in accordance with applicable regulations, regulatory guidance, and industry standards
- Performed for a single module
- Considered all modes of operation for both internal and external initiating events
- Provides risk insights including those related to risk-significant systems, components, human actions, relevant programs (e.g., RTNSS, SAMDA), and multiple module risk
- PRA demonstrates that the NuScale design exceeds NRC safety goals with significant margin

Section 19.1: Probabilistic Risk Assessment

- Objective: to assess risks associated with all modes and all hazards for a single NuScale Power Module (NPM)
- Level-1 (CDF) and Level-2 (LRF)
  - Full power, internal events (FP-IE)
  - Low power and shutdown (LPSD)
    - Include crane failure
  - Internal fire
  - Internal flood
  - External flood
  - High winds
  - Seismic margins assessment (PRA based)

PRA Quality Process

- NuScale PRA quality procedure
  - Follows guidance provided in NRC Regulatory Guide 1.174
- NuScale PRA follows guidance provided by
  - ASME/ANS PRA standard
  - NRC Regulatory Guide 1.200 and Interim Staff Guidance 028
- Each PRA notebook reviewed for conformance with PRA standard
  - Self-assessment documented by notebook authors
  - Self-assessment independently reviewed/verified by outside consultants
- PRA reviewed by outside, independent expert panel

PRA Expert Peer Review Group

- Separate and independent from PRA standard self-assessment reviewers
- Expert review group members:
  - George Apostolakis (chairman)
  - Mark Cunningham
  - Rick Grantom
  - Dave Moore
  - Per Peterson

Expert Panel Findings

- Review group authored a final report
  - No major concerns or objections
    - Minor points that were raised include
      - » NuScale multi-module risk approach represents an important “first step” in advancing the state-of-the-art
      - » There are more detailed and sophisticated HRA methods available compared to what was done in the NuScale PRA
      - » The terms CDF and LRF are tied to current large reactors and use of these terms in the NuScale design may be misleading

Independent Self Assessment

- External review of the NuScale PRA self-assessment against the high level and supporting requirements of the ASME PRA Standard
- In general, there was agreement, and in fact, in some cases, a higher capability category than identified was believed to be met. However, there were also some instances of a lack of concurrence, and possible enhancements were provided
- NuScale was able to incorporate those recommendations into the design certification PRA

Initiating Event Analysis

- Multiple sources of input used to identify potential initiating events (IEs)
  - NuScale design-specific master logic diagram
  - NuScale design-specific simplified system-level failure modes and effects analysis (FMEA)
  - Traditional lists of PRA initiating events
  - Continuous focus (over the years of NuScale design and PRA development) on identifying potential initiating events and hazards

Full Power Internal Initiating Events

- CVCS LOCA (injection line) inside containment vessel (CNV)
- CVCS LOCA (injection line) outside CNV
- CVCS LOCA (discharge line) outside CNV
- Spurious opening of ECCS valve
- Loss of DC power
- Loss of offsite power
- Steam generator tube failure
- LOCA (other) inside CNV
- Secondary-side line break (i.e., feedwater or main steam)
- General reactor trip
- Loss of support system (e.g., instrument air, AC power bus)

Accident Sequence Analysis

- Initiating events and subsequent plant responses evaluated
- Key safety functions identified
  - Fuel assembly heat removal, reactivity control, containment integrity
- End states of the accident sequences defined
  - Level-1: core damage frequency (CDF)
  - Level-2: large release frequency (LRF)
- Event trees constructed for each of the initiating events associated with system successes or failures to accomplish the applicable safety functions

Success Criteria

- The Level 1 PRA overall success criterion is the prevention of core damage, defined by maintaining a peak cladding temperature less than 2,200 degrees Fahrenheit
  - This is demonstrated for a 72 hour mission time
- System success criteria is determined by the minimum system availability required to prevent core damage
- The Level 1 success criteria evaluation is built upon a comprehensive simulation suite of more than 40 unique accident sequences
- The Level 2 success criterion is large release defined as a source term resulting in acute whole body 200 rem dose to the maximally exposed individual stationary at the reactor site boundary for 96 hours

Success Criteria

- PRA success criteria simulations use NuScale's safety-related NRELAP5 code with an input model that starts with NuScale's safety-related input model
  - The PRA simulations augment the safety-related input model with additional nonsafety-related models for beyond-design-basis phenomena
    - Chemical and volume control system (CVCS) and containment flooding and drain system (CFDS) models
    - Multi-dimensional core thermal hydraulic and neutronic models are used to simulate complex beyond design basis transients such as ATWS

Human Reliability Analysis

- Human actions are not credited in the evaluation of design basis events
  - Human actions only relevant to beyond design basis analyses
- Human error probabilities for beyond design basis events based on methodologies provided in NUREG/CR-4772 and NUREG/CR-6883
  - Latent human errors and recovery actions
- As a modeling convenience, when quantifying the PRA model, the bounding human error probability of the complete set of post-initiator human failure events, is used for all independently modeled post-initiator human failure events
- Risk significant human action candidates input to D-RAP
  - Operator fails to initiate CFDS injection
  - Operator fails to initiate CVCS injection

Post-initiator HEPs in PRA Quantification

- Post-initiator final human error probability (HEP) values range from 4E-3 to 2E-5
  - Time available (based on bounding scenarios) for human actions range from 30 minutes to 2 hours
- To simplify the quantification of the PRA model, bounding value of the set of HEPs used to quantify all post-initiator HEPs

Event	Description	Value	EF
HEP01	Human error probability for first HFE in cutset	4.0E-03	10
HEP02	Human error probability for second HFE in cutset	1.5E-01	3
HEP03	Human error probability for third HFE in cutset	0.5	-

NuScale PRA Human Errors Modeled (Pre-Initiator)

Name	Description
CFDS--HFE-0001A-UTM-N	Operator misaligns MDP 0001A CFDS train A manual valves during test and maintenance
CFDS--HFE-0002A-UTM-N	Operator misaligns MDP 0001B CFDS train B manual valves during test and maintenance
CVCS--HFE-0001A-UTM-N	Operator misaligns MDP 0002A CVCS train A manual valves during test and maintenance
CVCS--HFE-0002A-UTM-N	Operator misaligns MDP 0002B CVCS train B manual valves during test and maintenance
EHVS--HFE-0001A-UTM-N	Operator misaligns CTG 0003X EHVS combustion turbine generator during test and maintenance
ELVS--HFE-0001A-UTM-N	Operator misaligns DGN 0001X ELVS standby diesel generator during test and maintenance
ELVS--HFE-0002A-UTM-N	Operator misaligns DGN 0002X ELVS standby diesel generator during test and maintenance
MPS---HFE-0001A-UTM-S	Operator miscalibrates safety function modules during test and maintenance

NuScale PRA Human Errors Modeled (Post-Initiator)

Name	Description	Context
CFDS--HFE-0001C-FOP-N	Operator fails to unisolate and initiate CFDS injection	Used for LOCA-OC (2 IEs), SGTFs, and transients (1 IE)
CVCS--HFE-0001C-FOP-N	Operator fails to unisolate and initiate CVCS injection	Used for LOCA-IC (3 IEs), LOCA-OC (letdown) (1 IE), transients (1 IE) and secondary steam line break (1 IE) upon failure of ECCS, and SGTFs
CVCS--HFE-0002C-FOP-N	Operator fails to locally unisolate and initiate CVCS injection	Local unisolation due to lack of control from a partial loss of DC power
ECCS--HFE-0001C-FTO-N	Operator fails to open ECCS valves	Backup action to MPS autofunction failure
EHVS--HFE-0001C-FTS-N	Operator fails to start/load combustion turbine generator	Backup local action to control room initiation failure during loss of offsite power
ELVS--HFE-0001C-FTS-N	Operator fails to start/load backup diesel generator	Backup local action to control room initiation failure during loss of offsite power

Data Sources

- Industry information (e.g., NUREG/CR-6928, LERs) where applicable
  - Common cause failure (CCF) modeling based NUREG/CR-5497
- Design-specific analyses
  - Passive safety system reliability (i.e., ECCS, DHRS)
  - Unique events (e.g., steam generator tube failure)

Quantification

- Quantification of the PRA model was performed with the SAPHIRE code
  - Including CCF models, failure data correlations and uncertainty analyses
- Using the ASME/ANS PRA Standard convergence criterion, a truncation value of 1E-15 per module year was used for the CDF

Uncertainty Analyses

- Addressed using both quantitative uncertainty analyses and sensitivity studies
  - SAPHIRE PRA code has capability for propagating parametric uncertainties
  - Sometimes augmented using sensitivity studies (e.g., SGTF)
  - Thermal hydraulic analyses typically use bounding inputs
- Uncertainty addressed in all modes and all hazards of single module PRA
  - Multi-module risk quantification uses conservative, bounding estimates

Parametric Uncertainty

- The data parameters include initiating event frequencies, component failure probabilities, CCF events and their alpha factors, and human error probabilities
  - Initiating event frequencies that rely on generic industry data were assigned an expanded uncertainty distribution (i.e., lognormal error factor = 10)
- SAPHIRE has the built-in ability to perform an uncertainty analysis
  - Includes correlating failure probabilities
- After cutsets were generated in SAPHIRE, an uncertainty analyses was performed using the Latin Hypercube uncertainty sampling methodology.

Importance

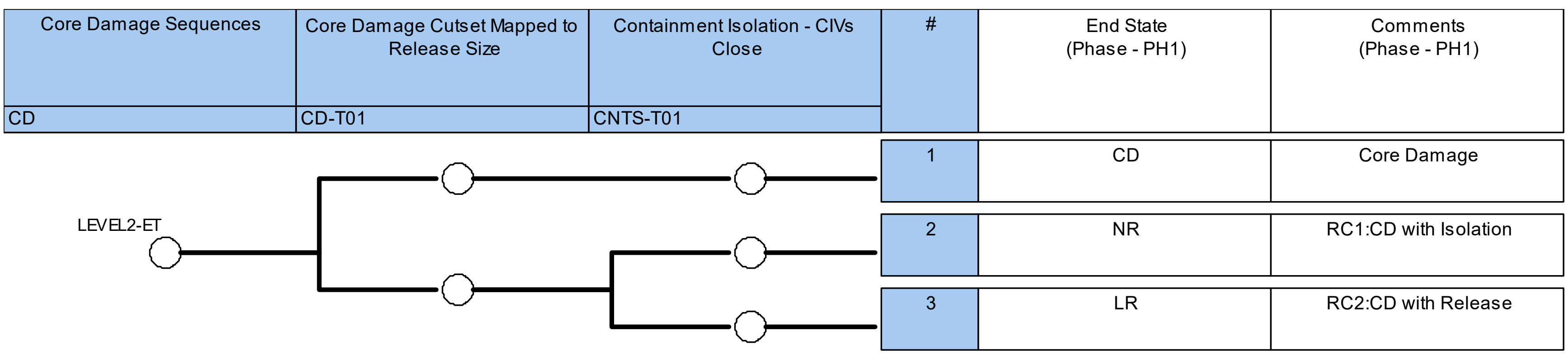
- Systems
  - CNTS (containment isolation valves), ECCS, MPS, and UHS
- Components
  - ECCS RVVs and RRVs
  - DHRS actuation valves
  - RSVs
  - CVCS and CES containment isolation valves
  - Combustion turbine generator
- Other events and initiators (FV>20%)
  - RBC, LOCA inside CNV, LOCA outside CNV, LOOP, internal fires, internal flood
- Human actions (FV>20%)
  - CVCS actuation and CFDS actuation (Level 2 and LPSD)

Sensitivity Studies

Parameter	Parameter Change	CDF Result	LRF Result
Base Case		***2.7E-10***	***1.7E-11***
HEP	All HEPs set to FALSE	2.0E-10	1.0E-11
HEP	All HEPs set to TRUE	3.2E-8	2.8E-9
CCF	All CCFs set to FALSE	5.4E-12	1.2E-12
CCF	All CCFs set to max value of 0.002	4.2E-6	3.7E-8
LOOP-IE	LOOP frequency set to 1 per year (base = 3.1E-2 per year)	2.2E-9	1.7E-11
LOCA-IC-IE	LOCA inside CNV frequency increased 1 order of magnitude	3.4E-10	1.7E-11
SGTF-IE	SGTF frequency increased to generic value	2.8E-10	2.2E-11
ECCS & DHRS PSSR	ECCS and DHRS passive heat transfer failure increased 1 order of magnitude	3.2E-10	1.7E-11
I&C sensors	Failure probability of sensors was increased an order of magnitude	2.8E-10	1.7E-11

Level 2 Methodology

- Analysis indicates that the only applicable containment vessel (CNV) failure mechanisms are containment bypass events and failure of containment isolation
- No bridge trees or Level 1 plant damage state binning
  - Level 2 event tree is directly linked to the Level 1 event trees



External Hazards

- External events are evaluated using Level 1 PRA model and the following methodologies
  - Internal fire: NUREG/CR-6850
  - Internal flood: Part 3 of ASME/ANS RA-Sa-2009
  - External flood: Part 8 of ASME/ANS RA-Sa-2009
  - High winds: Part 7 of ASME/ANS RA-Sa-2009
  - Seismic margin assessment: Part 5 of ASME/ANS RA-Sa-2009

Seismic Risk Evaluation

- NuScale performed a PRA-based seismic margin assessment (SMA)
- Design-specific fragility calculations were performed for SSCs that contribute to the seismic margin
  - Consulted with Simpson, Gumpertz & Heger, Rizzo Associates, and Stevenson and Associates
- Generic capacities with design-specific response factors were used for other SSCs
- DC/COL-ISG-020 seismic margin goal: high confidence of low probability of failure (HCLPF) value of 1.67 times the certified seismic design response spectra (CSDRS)
  - Corresponds to 0.84g peak ground acceleration (PGA)

SMA Methodology

- PRA-based SMA uses internal event logic, seismically-induced initiators, and maps seismic failures to random failures
- HCLPF: high confidence (95%) of low probability (5%) of failure
  - HCLPF can also be interpreted as a 1% probability of failure at the mean (or best-estimate) confidence level (i.e., at the HCLPF PGA there is a 1% probability of core damage)
  - Evaluated at the sequence level using min-max criteria
- Seismic margin determined by those seismic failures that would result in a conditional core damage probability of greater than 1%
- Structural fragilities evaluated for those SSCs that contact the module, are located above the module, or where collapse might damage the module (which is assumed to result in core damage)

Seismic Risk Evaluation

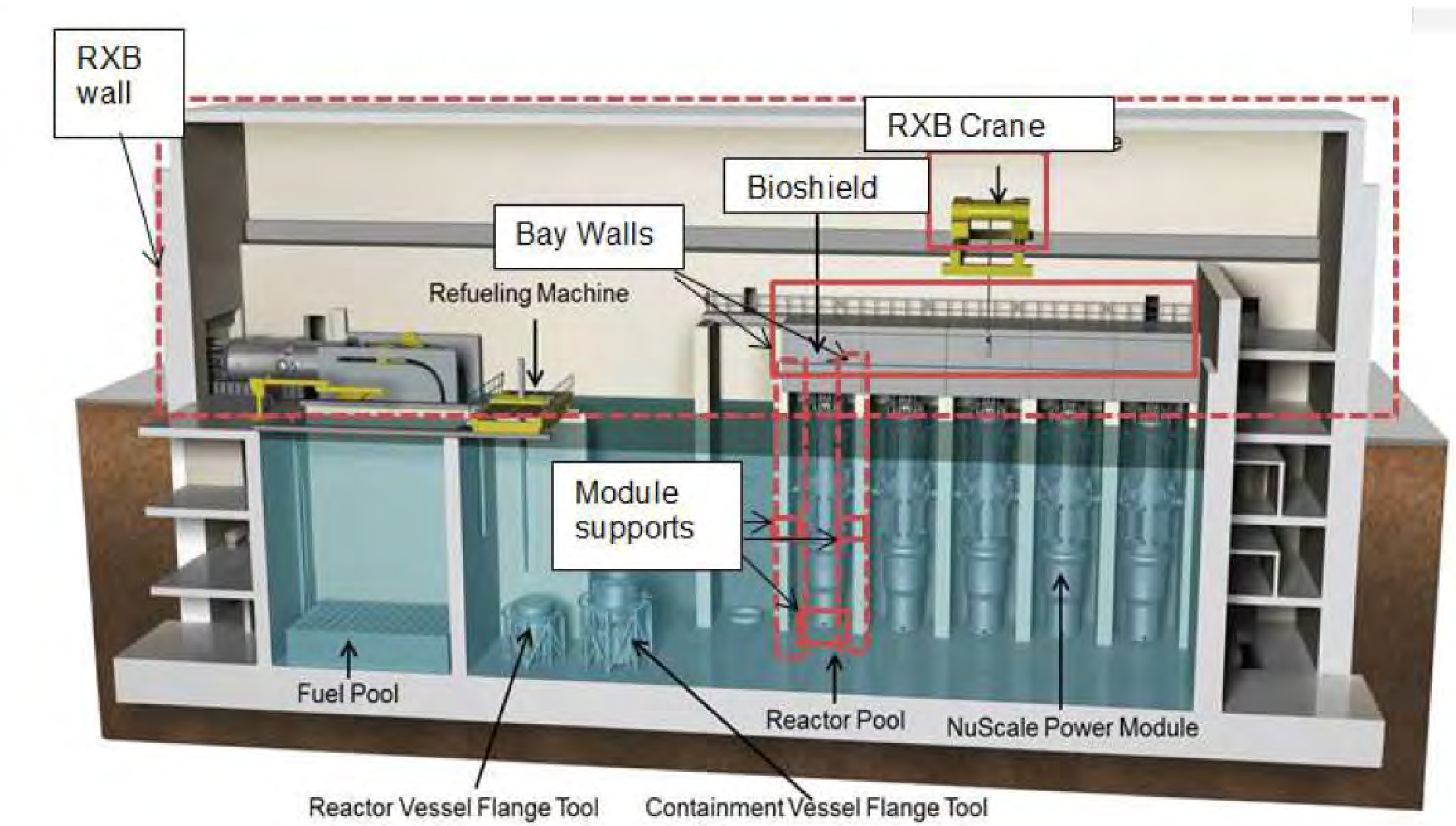
Seismic plant response

- Induced initiator event trees
  - Structural failures
  - LOCAs
  - Loss of offsite power
- Seismic failure mapping
  - No pre-screening (all PRA cutsets included in the SMA)
  - Evaluated at 14 ground motion levels ranging from 0.05g to 3.5g

SMA results

- Plant level HCLPF: 0.88g
- Structural failures dominate
  - Crane
  - Exterior walls
  - Bay walls
  - Module supports
- At lower PGAs, LOOP combined with random failures dominate results
- Negligible seismic risk from low power and shutdown states

Select NuScale SMA Structures



Fragility Calculation Parameters

- Design calculations for demand/capacity (D/C) ratio inputs
  - Uses bounding, conservative values
  - For fragility purposes, design calculates are adjusted to median-centered values, uncertainties quantified
- Structural response factor variables
  - Ground motion response
  - Damping
  - Modeling
  - Mode combination
  - Time history simulation
  - Foundation-structure interaction
  - Earthquake component combination

Fragility Calculation Parameters

- Capacity variables
  - Strength
  - Ductility
- Earthquake scale factor (ESF)
  - Used in wall calculations, where capacity changes with demand
  - Ratio by which the seismic demand must increase for overall demand to equal capacity
    - Static demand + ESF * seismic demand = static capacity +/- ESF * dynamic capacity (sign is dependent on load in compression / tension)
  - Used to calculated median capacity A_m

Low Power and Shutdown

- Potential initiating events are those considered for full power and those unique to LPSD
  - Reduced inventory (drain down) events not applicable
    - No reduced inventory operations in the NuScale design
- Evaluated external events shown to be not important
- Dropped module event most significant CDF contributor
  - Relatively high level of conservatism embedded in analysis

Dropped Module Evaluation

- Drop probability developed based on conceptual reactor building crane design
- Core damage conservatively assumed for dropped module
  - For a horizontal module the core partially uncovers
  - Containment assumed to fail in a manner that prevents pool water incursion but allows radionuclide release
- Maximum radiological release much less than large release due to pool scrubbing effect
- Up to two operating modules theoretically could be struck by free-falling module, potentially inducing LOCA or transient in struck module

Postulated Dropped Module Impacts

- Potential damage to the decay heat removal system (DHRS) because the heat exchangers are located external to the containment and face central pool channel
  - Likelihood is an insignificant contributor to the modeled frequency of secondary side line break initiating event
- Potential damage to the chemical and volume control system (CVCS) piping where the piping penetrates the bay wall as a result of movement of the struck module
  - Likelihood is an insignificant contributor to the modeled frequency of the CVCS pipe break outside containment initiating event
- Considering the probability of a load drop, the contribution of a potential module drop to the initiating event frequencies of an operating module is judged to be negligible both in absolute terms and in comparison to the frequency of a randomly occurring initiating events

Multiple Module Evaluation

- Each NPM comprises a separate, independent RPV and CNV, and is serviced by separate, independent safety systems
- Systematic evaluation performed per SRP 19.0
- Single module PRA with bounding multi-module adjustment factors (MMAF) applied to each and every basic and initiating event
  - MMAF value of 1.0 for SSCs shared amongst multiple modules and plant wide initiating events (e.g., LOOP)
  - MMAF values from 0.1 to 0.3 for SSCs with potential coupling mechanisms between modules (e.g., potential for common cause failures)
  - Smallest applied MMAF of 0.01 to events that would nominally be considered independent (e.g., pipe failures)

Level 1 Insights

- NuScale design exceeds NRC core damage frequency safety goal with significant margin
  - Full power internal event CDF 3.0E-10/mcyr
    - External initiator CDFs: 1.0E-09 to 6.1E-11/mcyr
  - LPSD CDF dominated by module drop event: 8.8E-08/mcyr
  - Focused PRA CDF (no credit for nonsafety-related systems): 3.1E-06/mcyr
    - Approximately equivalent to a long-term station blackout with no recovery of ac power
  - Multiple module CDF factor: 0.13

Level 2 Insights

- NuScale design exceeds NRC large release frequency safety goal with significant margin
  - Full power internal event LRF 2.3E-11/mcyr
    - External initiator LRFs: 4.3E-11 to <1E-15/mcyr
  - Module drop event does not result in large release
  - Focused PRA LRF (no credit for nonsafety-related systems): 1.6E-07/mcyr
    - Approximately equivalent to a long-term station blackout with no recovery of ac power
  - Multiple module LRF factor: 0.01

Level 1 Key Insights (1 of 2)

Design Feature/Insight	Comment
Failure to scram events (ATWS) do not lead directly to core damage.	Core characteristics result in ATWS power levels that are comparable to decay heat levels. Heat transfer from the containment vessel (CNV) to reactor pool is adequate to prevent core damage and most ATWS sequences require approximately the same system success criteria as non-ATWS events.
Passive heat removal capability is sufficient to prevent core damage if a reactor safety valve (RSV) cycles.	RSV cycling transfers adequate RCS water to the CNV to allow heat transfer through the RPV to the CNV and ultimately to reactor pool to remove decay heat.
Post-accident heat removal through steam generators or decay heat removal system (DHRS) is unnecessary if RSVs cycle.	The steam generators and DHRS provide effective heat removal paths to prevent core damage, but are unnecessary if RSV cycling allows heat transfer to reactor pool. Passive, fail-safe DHRS provides a natural circulation closed loop system that does not require pumps, power, or additional water.
Passive, fail-safe emergency core cooling system (ECCS) functions to preserve RCS inventory, which is sufficient to allow core cooling without RCS makeup from external source.	The ECCS consists of 5 valves that fail-safe on a loss of power and provides a natural circulation path through the core and CNV, thus providing heat transfer to the reactor pool. The closed-loop system does not need additional inventory.

Level 1 Key Insights (2 of 2)

Design Feature/Insight	Comment
Containment isolation preserves RCS inventory for core cooling without external makeup.	Containment isolation eliminates the potential for breaks outside of containment to result in loss of RCS inventory. For breaks inside of containment, containment isolation is not necessary to support passive core cooling and heat removal.
Passive, fail-safe safety systems (ECCS, DHRS, RSVs) include redundancy and do not need support systems, including electric power or operator actions.	Safety-related mitigating systems are fail-safe on loss of power and do not require supporting systems such as lube oil, air or HVAC to function. No single failure results in a loss of system function.
There are no risk significant, post-initiator human actions associated with the full-power PRA.	No operator actions, including backup and recovery actions, are risk significant to the CDF because of passive system reliability and fail-safe system design.
Risk significant structures, systems and components (SSCs) for external events are largely the same as those found risk significant for internal events.	The module response to external events is comparable to the response to internal event due to the passive features of the design and independence from support systems such as power. Additional systems and components have been identified as risk significant for external events due to a conservative evaluation.
Active systems providing makeup inventory to the RPV are not risk significant.	Inventory addition is possible by the active systems chemical and volume control system (CVCS) and containment flooding and drain system (CFDS). Due to the reliability of the passive safety systems, the active systems providing this backup function were found not to be risk significant.

Section 19.1 COL Items

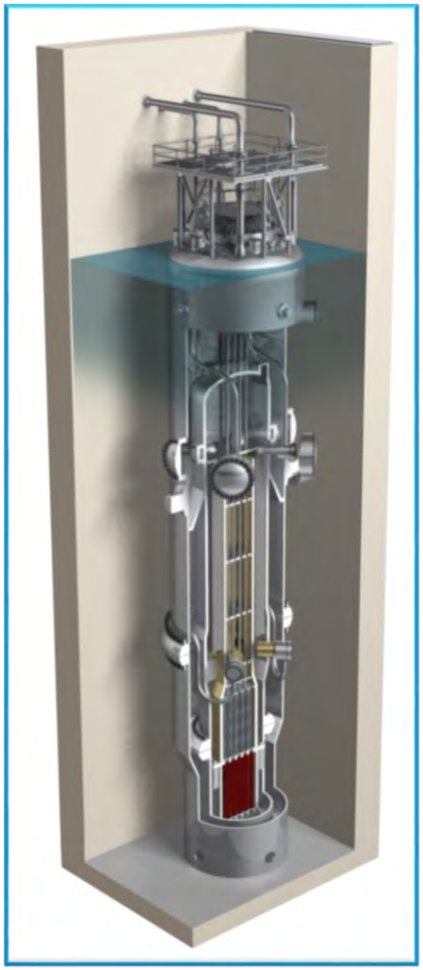
Item Number	Description
COL Item 19.1-1	A COL applicant that references the NuScale Power Plant design certification will identify and describe the use of the probabilistic risk assessment in support of licensee programs being implemented during the COL application phase.
COL Item 19.1-2	A COL applicant that references the NuScale Power Plant design certification will identify and describe specific risk-informed applications being implemented during the COL application phase.
COL Item 19.1-3	A COL applicant that references the NuScale Power Plant design certification will specify and describe the use of the probabilistic risk assessment in support of licensee programs during the construction phase (from issuance of the COL up to initial fuel loading).
COL Item 19.1-4	A COL applicant that references the NuScale Power Plant design certification will specify and describe risk-informed applications during the construction phase (from issuance of the COL up to initial fuel loading).

Section 19.1 COL Items

Item Number	Description
COL Item 19.1-5	A COL applicant that references the NuScale Power Plant design certification will specify and describe the use of the probabilistic risk assessment in support of licensee programs during the operational phase (from initial fuel loading through commercial operation).
COL Item 19.1-6	A COL applicant that references the NuScale Power Plant design certification will specify and describe risk-informed applications during the operational phase (from initial fuel loading through commercial operation).
COL Item 19.1-7	A COL applicant that references the NuScale Power Plant design certification will evaluate site-specific external event hazards (e.g., liquefaction, slope failure), screen those for risk-significance, and evaluate the risk associated with external hazards that are not bounded by the design certification.
COL Item 19.1-8	A COL applicant that references the NuScale Power Plant design certification will confirm the validity of the "key assumptions" and data used in the design certification application probabilistic risk assessment (PRA) and modify, as necessary, for applicability to the as-built, as-operated PRA.

NuScale Nonproprietary

ACRS Subcommittee Presentation:



NuScale FSAR Chapter 19.2

Severe Accident Evaluation

May 15, 2019

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Revision: 0

Template #: 0000-21727-F01 R5

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Section 19.2: Severe Accident Evaluation

- Accident sequences resulting in core damage are evaluated in the Level 2 PRA for potential to challenge containment vessel (CNV) integrity and result in a large radionuclide release
  - Large release defined as source term resulting in acute whole body 200 rem dose to the maximally exposed individual stationary at the reactor site boundary for 96 hours
  - MACCS off-site consequence calculations demonstrate that sequences with intact CNV are not large release
  - CNV bypass accidents counted as large release (simplification for convenience)
- Potential challenges to CNV integrity identified from SRP, PRA standard, and NUREGs
- There are no unique phenomenological challenges that are introduced by the NuScale design

Use of MELCOR

- Provides a best estimate evaluation of severe accident challenges to CNV
- Informs conservative evaluations of severe accident challenges
  - Provides a physical basis for parameters
    - Timing of core damage, core relocation
    - Quantity of relocated material, composition of relocated material
    - System pressures, temperatures, quantity of hydrogen produced
  - Evaluations use limiting values from database of simulations that each involve bounding/conservative simplifications
    - End of cycle decay heat load
    - DHRS not credited to slow down accident progression
  - Evaluations also consider parameters that bound all results observed from database of simulations
    - 100% of fuel UO_2 relocates at first observed relocation time from database
    - Assume debris is molten, pure UO_2 composed of no filler materials (e.g., steel, zirconium)
    - No credit for water in lower plenum at time of relocation

MELCOR Model Development

- Thermal-hydraulics modeling developed from NRELAP5 model
  - Matching elevations, volumes, flow areas, frictional losses, heat structure material, surface area, thickness, heated diameters, etc
- Benchmarking of steady-state operation and transients demonstrate reasonable to excellent agreement with NRELAP5
  - Goal is to approximately match NRELAP5 accident simulation to the point of core damage and then extend simulation into severe accident space
- Severe accident modeling based on appropriate and accurate modeling of NPM design characteristics
  - Decay power curve, core component masses and locations, radionuclide inventory, core flow geometry
- Incorporates modeling best practices from
  - MELCOR code development staff and industry leading subject matter experts
  - State-of-the-Art Reactor Consequence Analyses (SOARCA) reports
  - MELCOR guides, manuals, assessments

In-Vessel Retention (IVR)

- Conservative analysis demonstrates that RPV lower head integrity is maintained if core debris relocates to lower plenum
- Maximum heat flux remains below critical heat flux (CHF) on exterior surface
  - Heat generation rate based on conservative assumptions/inputs (e.g., 100% core UO_2 - no upward radiation heat losses)
  - Assumed CHF threshold conservatively does not credit high absolute pressure and large subcooling in CNV
- With effective external vessel cooling, the lower head remains intact and the severe accident progression is stabilized in RPV

Severe Accident Phenomena

CNV integrity not challenged by severe accident phenomena

- Hydrogen combustion not challenging due to limited oxygen concentration
- In-vessel fuel-coolant interactions (FCI) (i.e., steam explosion) are not sufficiently energetic to induce alpha mode failure due to factors including:
  - Small core size, low debris temperatures, small drop height, shallow pool, relatively high system pressure
- Containment overpressure does not occur
  - High pressure steel CNV designed for most limiting LOCA blowdown which exceeds maximum severe accident pressures
  - Submergence of CNV in UHS provides highly effective pressure suppression
  - No concrete interactions to generate non-condensable gases

Consideration of Uncertainty

- If IVR in RPV fails
  - High pressure melt ejection (leading to direct containment heating) does not occur because there is no driving pressure differential
  - Energetic ex-vessel FCI not likely for similar reasons as in-vessel FCI
  - Debris relocated to CNV would be retained by CNV lower head
    - Effective external cooling of CNV by reactor pool
- If lower CNV fails
  - Pool scrubbing minimizes release
- If upper CNV fails
  - Instantaneous release of entire airborne radionuclide inventory in module at time of postulated CNV failure would not constitute a large release

Level 2 Insights

- Core damage events are stabilized within the RPV
- Severe accident phenomena do not challenge CNV integrity
- Large release does not occur even if RPV and CNV are postulated to fail
- The large release frequency is dominated by containment bypass events

Level 2 Key Insights (1 of 5)

Design Feature/Insight	Comment
Containment Isolation	
The primary purpose of CNTS is to retain primary coolant inventory within the CNV. With primary coolant inventory maintained in the RPV or CNV, cooling of core debris is ensured.	If coolant remains primarily within the RPV, then the core is covered. If the core is not covered in the RPV then sufficient primary coolant is in the CNV to submerge the outside of the lower RPV and establish conductive heat removal from the core debris to the coolant in the CNV through the RPV wall.
CNTS terminates releases through penetrations leading outside containment.	Containment penetrations through which releases are assumed to occur that dominate risk include those that bypass containment such as CVCS (injection and discharge) and paths through the steam generator tubes (main steam and feedwater piping). Isolation of normally open valves in these penetrations prevents releases from bypassing containment.

Level 2 Key Insights (2 of 5)

Design Feature/Insight	Comment
Passive Heat Removal	
The RPV has no insulating material and passive heat removal capability from the RPV to the CNV is sufficient to prevent core debris from penetrating the reactor vessel.	Retaining primary coolant in the containment results in collection of sufficient RCS water in the CNV to allow heat transfer through RPV to CNV and ultimately UHS to remove heat generated in the fuel regardless of its location.
The CNV is uninsulated and passive heat removal capability from the CNV to the UHS is sufficient to prevent the containment from pressurizing and or core debris from penetrating the containment	

Level 2 Key Insights (3 of 5)

Design Feature/Insight	Comment	
Severe Accident Containment Challenges (1 of 2)		
Primary coolant system overpressure failure cannot lead to overpressurization of containment (i.e., loss of decay heat removal through the steam generators plus failure of the RSVs to open).	Addition of water to the containment from external sources (CFDS) results in submergence of the reactor vessel and establishes passive heat removal through the containment wall to the reactor pool. Even if containment flooding is not successful, the RPV failure mode is such that containment ultimate capacity would not be exceeded.	
Hydrogen combustion is not likely as the containment is normally evacuated.	There is very little oxygen available (oxygen generated from radiolysis is only a long-term issue) and containment is steam inerted under severe accident conditions. In addition, conservative AICC analyses predict containment pressures that do not exceed the design pressure.	
In-vessel steam explosions are not likely due to core support design and volume of lower vessel head.	Core support failure is expected before the fuel has a chance to become molten. With the core uncovered there is little water in the bottom of the RPV with which core debris can interact.	
HPME cannot occur	Submergence of the lower RPV establishes passive heat removal and prevents core debris from exiting the RPV. No ex-vessel challenges occur if the core remains within the vessel.	With passive heat removal from the reactor to containment established, the reactor is depressurized even if core debris is postulated to exit the vessel.

Level 2 Key Insights (4 of 5)

Design Feature/Insight	Comment
Severe Accident Containment Challenges (2 of 2)	
Ex-vessel steam explosion does not occur with a submerged RPV.	Submergence of the lower RPV establishes passive heat removal and prevents core debris from exiting the RPV. No ex-vessel challenges occur if the core remains within the vessel.
Overpressure of containment due to non-condensable gas generation is not applicable to the NuScale design.	There is no concrete in the containment with which the core debris could interact and generate non-condensable gases.
Basemat penetration is not applicable to the NuScale design.	There is no basemat making up the containment boundary. This issue is addressed as a part of considering protection against contact of core debris with the containment wall.

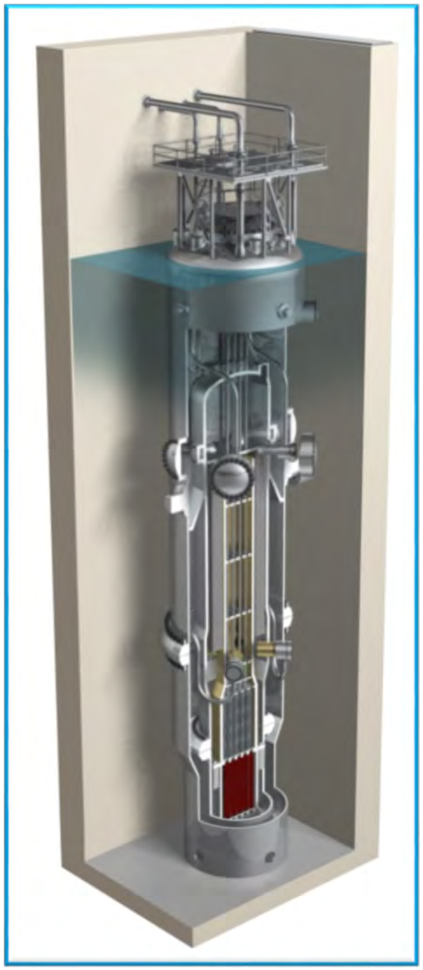
Level 2 Key Insights (5 of 5)

Design Feature/Insight	Comment
Support Systems, Human Action, External Events	
Support systems are not needed for safety-related system functions (i.e., containment isolation) important to the Level 2 PRA.	Safety-related mitigating systems are fail-safe on loss of power and do not require supporting systems such as lube oil, instrument air, or HVAC to function.
With one exception, there are no risk significant, post-accident human actions associated with the full-power internal events Level 2 PRA. The exception is alignment of CFDS during accident sequences in which isolation of a broken CVCS line outside containment fails, ECCS is successful but coolant inventory in containment needs replenishment in order to maintain natural circulation between CNV and the RPV.	Operator actions, including backup and recovery actions, are not significant to the Level 2 analysis because of passive system reliability and fail-safe system design. The operator action to align CFDS during a CVCS break outside containment meets the risk significance thresholds because of a mathematical limitation of the calculation of the Fussell-Vesely measure of importance
Risk significant SSC for external events are largely the same as those found risk significant for internal events	The module response to external events is comparable to the response to internal event due to the passive features of the design which are not affected by the external events and plant systems that are protected against external event challenges.

Section 19.2 COL Items

Item Number	Description
COL Item 19.2-1	A COL applicant that references the NuScale Power Plant design certification will develop severe accident management guidelines and other administrative controls to define the response to beyond-design-basis events.
COL Item 19.2-2	A COL applicant that references the NuScale Power Plant design certification will use the site-specific probabilistic risk assessment to evaluate and identify improvements in the reliability of core and containment heat removal systems as specified by 10 CFR 50.34(f)(1)(i).
COL Item 19.2-3	A COL applicant that references the NuScale Power Plant design certification will evaluate severe accident mitigation design alternatives screened as “not required for design certification application.”

ACRS Subcommittee Presentation:



NuScale FSAR Chapter 19.3

Regulatory Treatment of Nonsafety Systems

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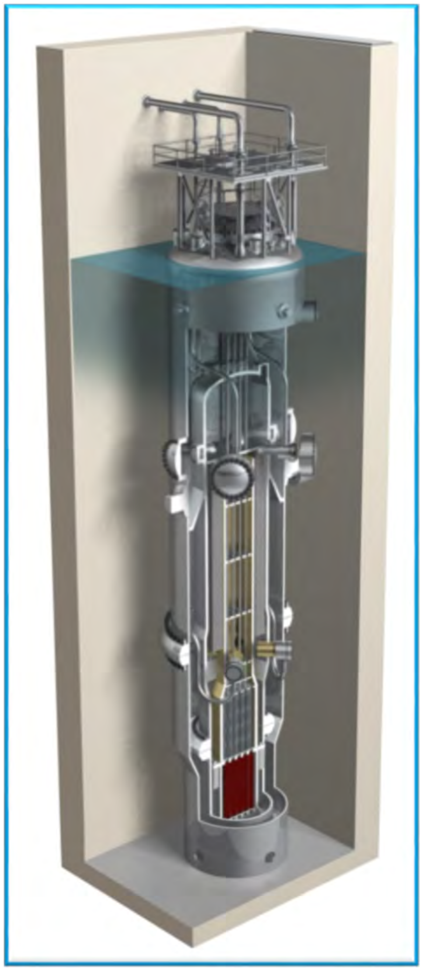
Section 19.3

- There are no RTNSS SSCs in the NuScale design
  - None of the five RTNSS criteria were met by any NuScale SSC
- RTNSS is also discussed in FSAR 17.4.3.3

Section 19.3 COL Item

Item Number	Description
COL Item 19.3-1	A COL applicant that references the NuScale Power Plant design certification will identify site-specific regulatory treatment of nonsafety systems (RTNSS) structures, systems, and components and applicable RTNSS process controls.

ACRS Subcommittee Presentation:



NuScale FSAR Chapter 19.5

Adequacy of Design Features and Functional Capabilities Identified and Described for Withstanding Aircraft Impacts

May 15, 2019

PM-0519-65372
Revision: 0

Template #: 0000-21727-F01 R5

Presentation Team

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Introduction and Background

- Plant design for potential effects of beyond design basis large commercial aircraft impact [10 CFR 50.150(a)]
  - The reactor core remains cooled, or the containment remains intact
  - Spent fuel cooling or spent fuel pool integrity is maintained
- Design-specific impact assessment per RG 1.217, which endorses NEI 07-13
- NEI 07-13 methods followed with no exceptions
- Aircraft impact informed the plant design

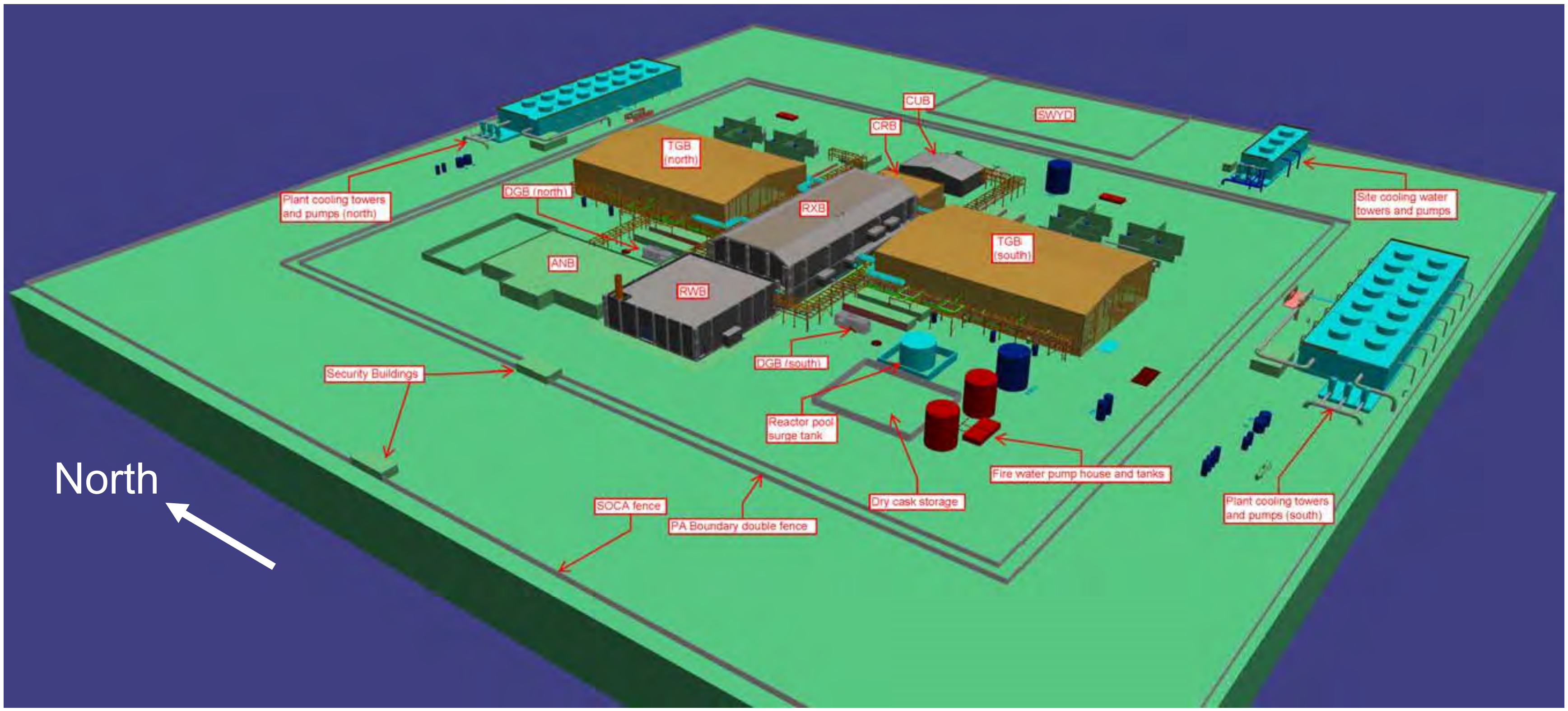
Assessment Scope

- Reactor Building assessed for effects in three areas for postulated aircraft impact
  - Physical damage
  - Shock damage from shock-induced vibration on structures, systems, and components
  - Fire damage from aviation fuel-fed fire

Assessment Methodology

- NEI 07-13
- Reactor Building is structure of concern
  - NuScale Power Modules
  - Ultimate heat sink
  - Spent fuel pool
- Impact locations
  - Screening by NEI 07-13
  - Radioactive Waste Building (RWB) is “intervening structure” to mitigate physical damage to RXB, conservatively do not credit RWB in shock assessment
  - No credit taken for CRB or TGB

NuScale Site Plan



NuScale DCA Tier 2 Figure 1.2-1 Conceptual Site Layout

Assessment Results

- Physical damage
  - Local assessment per NEI formulas for perforation and scabbing
  - Global response performed using detailed finite element models and NRC specified force-time history
  - RXB external walls prevent physical damage from entering RXB
  - No internal missiles for secondary impact
  - No impact on containment boundary
  - Spent fuel pool protected inside RXB below grade
  - Reactor Building crane trolley cannot be dislodged

Assessment Results (cont'd)

- Shock damage
  - Aircraft impact causes short duration, high acceleration, high frequency vibration
  - Core cooling
    - At-power and shutdown scenarios considered
    - No active equipment required for success
    - Adequate heat removal is shown for all strikes
  - Spent fuel
    - SFP integrity maintained for all strikes

Assessment Results (cont'd)

- Fire damage
  - Design and location of 3-hr fire barriers and 3-hr, 5-psid fire barriers prevent propagation of fire into RXB
  - Design and location of 5-psid, fast-acting blast dampers at RXB HVAC key design feature
  - Concrete shrouds protect exterior wall pipe and HVAC penetrations from physical damage and prevent fire propagation into the RXB
  - Fire that enters through external personnel doors at grade level does not propagate beyond stairwells
- All required operator actions occur prior to impact

Assessment Conclusions

- Design and functional capabilities provide adequate protection of public health and safety
- NuScale plant meets 10 CFR 50.150 regulation
  - Maintain containment integrity AND core cooling capability (only required to meet one)
  - Maintain SFP integrity
- For most postulated aircraft impact strikes, spent fuel pool cooling maintained, meeting all four CFR requirements

Acronyms (1 of 3)

- ATWS anticipated transient without scram
- BDB beyond design basis
- CCF common cause failure
- CD core damage
- CDF core damage frequency
- CES containment evacuation system
- CFDS containment flooding and drain system
- CFR Code of Federal Regulations
- CHF critical heat flux
- CIV containment isolation valve
- CNV containment vessel
- CNTS containment system
- COL combined license
- CRB Control Building
- CVCS chemical and volume control system
- CSDRS certified seismic design response spectra
- CTG combustion turbine generator
- D/C demand/capacity
- DGN diesel generator
- DHRS decay heat removal system
- ECCS emergency core cooling system
- EHVS 13.8 kV and switchyard system
- ELVS low voltage AC electrical distribution system
- ESF earthquake scale factor

Acronyms (2 of 3)

- FCI fuel-coolant interaction
- FMEA failure modes and effects analysis
- FP-IE full power, internal event
- FSAR Final Safety Analysis Report
- HCLPF high confidence of low probability of failure
- HEP human error probability
- HPME high pressure melt ejection
- HVAC heating ventilation and air conditioning
- IE initiating event
- IVR in-vessel retention
- LOCA loss of coolant accident
- LOOP loss of offsite power
- LPSD low power and shutdown
- LR large release
- LRF large release frequency
- mcyr module critical year
- MDP motor driven pump
- MMAF multi-module adjustment factor
- MPS module protection system
- NEI Nuclear Energy Institute
- NPM NuScale Power Module
- NR no release
- NRC Nuclear Regulatory Commission
- PGA peak ground acceleration
- PRA probabilistic risk assessment

Acronyms (3 of 3)

- RBC reactor building crane
- RCS reactor coolant system
- RG Regulatory Guide
- RPV reactor pressure vessel
- RSV reactor safety valve
- RTNSS regulatory treatment of nonsafety systems
- RVV reactor vent valve
- RWB Radioactive Waste Building
- RXB Reactor Building
- SAMDA severe accident mitigation design alternative
- SAPHIRE Systems Analysis Programs for Hands-on Integrated Reliability Evaluations
- SGTF steam generator tube failure
- SMA seismic margin assessment
- SOARCA State-of-the-Art Reactor Consequence Analysis
- SSC structures, systems, and components
- SFP spent fuel pool
- SRP Standard Review Plan
- TGB Turbine Generator Building
- UHS ultimate heat sink

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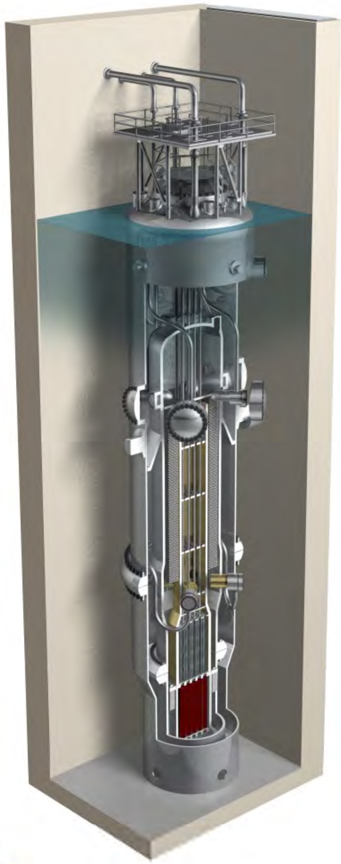

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Safety Evaluation with Open Items: Chapter 19 “Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors”

NuScale Design Certification Application

ACRS Subcommittee Meeting
May 14-15, 2019

Agenda

- Presentation Topic for May 14, 2019:
  - Section 19.1, Probabilistic Risk Assessment

- Presentation Topics for May 15, 2019:
  - Section 19.2, Severe Accident Evaluation
  - Section 19.3, Regulatory Treatment of Nonsafety Systems for Passive Advanced Light Water Reactors
  - Section 19.5, Adequacy of Design Features and Functional Capabilities Identified and Described for Withstanding Aircraft Impacts

- Presentation Topic Included in Chapter 20 (for future discussion):
  - Section 19.4, Strategies and Guidance To Address Loss of Large Areas of the Plant Because of Explosions and Fires

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Section 19.2
Severe Accident Evaluation

Staff's Review - Overview

- SER is based on DCA Revision 2

- Staff conducted two regulatory audits

- Staff conducted independent confirmatory analyses

Overview of Review Guidance

- SRP Section 19.0 – acceptance criteria for PRA and severe accident evaluation

- SECY-90-016 and SECY-93-087, and the associated SRMs – containment performance goals

Severe Accident Mitigation

- Containment performance goals for new reactors
  - Probabilistic: CCFP < 0.1
  - Deterministic: stay intact for 24 hours, after 24 hours provide barrier against uncontrolled release

- Uncertainties associated with in-vessel retention prevent confirming the NuScale design meets the Commission containment performance goals

Severe Accident Mitigation

Alternative criteria

- Probabilistic: CCFP < 0.1
  - Applied to in-vessel steam explosion and containment bypass
- Deterministic: no large release
  - Applied to accidents for which demonstration of in-vessel retention is inconclusive

Severe Accident Mitigation

- Containment lower head failure
  - No large release, due to aerosol deposition in reactor and containment and reactor pool scrubbing

- Steam explosion in containment
  - No large release, due to aerosol deposition in reactor and containment

Severe Accident Mitigation

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- High pressure melt ejection
  – NuScale's unique design results in depressurization before lower head challenged

- Steam explosion in the reactor vessel
  – Insufficient energy release to fail containment upper head
  – Staff performed independent confirmatory analysis

Severe Accident Mitigation

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- Severe accident-induced steam generator tube failure
  - Staff did not evaluate applicant's assumption that tubes are less susceptible to failure under compression than under tension
    - Low rate of creep under postulated accident conditions
  - Small impact on large release frequency
    - Steam generator relief valve relieves into containment
    - Tube failure leads to release only if the main steam or feedwater isolation valves fail to close

Severe Accident Mitigation

- NuScale used MACCS to show the dose at the site boundary (0.167 miles) is less than NuScale's large release definition
  – Benchmarked MACCS plume modeling against ARCON which is valid for close-in distances
  – Large margins between the large release definition and the doses calculated using MACCS

Severe Accident Mitigation

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- Reactor pool scrubbing for module drop accidents
  - Pool scrubbing reduces the release to a fraction of a large release
  - Supplemental analysis showed that NuScale's pool scrubbing model is conservative

Severe Accident Mitigation

- Staff's independent confirmatory analysis
  - Staff compared its independent MELCOR simulation results to the applicant's
  - Staff's simulations generally reproduced the applicant's simulations but usually with timing shifts
  - Differences unlikely to affect the applicant's analysis of severe accident mitigation

Multi-module Severe Accident Risk

- Hydrogen generation and control
  - Core damage sequence of an un-isolated CVCS line break outside containment could lead to a hydrogen combustion event under the bio-shield
  - Staff is reviewing the redesigned bio-shield
  - Staff is evaluating the potential impact on the multi-module severe accident risk
  - Staff is tracking this as an open item (RAI 9447, Question 03.11-19)

Summary

Severe Accident Evaluation

- Due to the open items, the staff is unable to make a finding on the applicant's severe accident evaluation in DCA Part 2, Tier 2, Section 19.2
  - RAI 9447
  - Accident Source Term Topical Report Review

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Section 19.3
Regulatory Treatment of Non-safety Systems

Regulatory Treatment of Non-safety Systems (RTNSS)

- Review evaluated NuScale's determination that there are no RTNSS SSCs

- Staff found the evaluation reasonable and acceptable

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Section 19.5
Adequacy of Design Features and Functional Capabilities Identified and Described for Withstanding Aircraft Impacts

Technical Topics

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Overview

- **Staff Verified**:
  - Adequacy of Key Design Feature Descriptions
    - Fire Protection
    - Structures
    - Systems
  - How Key Design Features met 10 CFR Part 50.150(a)(1)

    (i) The reactor cores remain cooled and containment remains intact,

    (ii) The spent fuel pool integrity is maintained
  - Shock Damage Description
  - Reasonably Formulated Assessment of Aircraft Impact
  - Assessment Performed by Qualified Individuals

One Confirmatory Item

RAI 9241, Question 19.5-23: related to the availability of monitoring functions to ensure that the identified design features are performing as expected following the impact of a large commercial aircraft.

Technical Topics

Conclusions:

- Section 19.5 meets 10 CFR Part 50.150 because the applicant:
  - Adequately identified and described the key design features and functional capabilities credited to meet 10 CFR Part 50.150
  - Adequately described how the key design features meet the assessment requirements in 10 CFR Part 50.150(a)(1)
  - Developed a reasonably formulated assessment performed by qualified individuals

Abbreviations And Acronyms

- **CCFP –** conditional containment failure probability
- **CVCS** – chemical and volume control system
- **DCA –** design certification application
- **NRC –** Nuclear Regulatory Commission
- **PRA –** probabilistic risk assessment
- **RAI** – request for additional information
- **RTNSS –** regulatory treatment of nonsafety systems
- **SSCs –** structures, systems, and components

Questions/comments from members of the public before the closed session starts?

May 9, 2019 Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Subcommittee Presentation: NuScale FSAR Chapter 21, Multi-Module Design Considerations," PM-0519-65533, 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 May 15, 2019. The materials support NuScale's presentation of Chapter 21, "Multi-Module Design Considerations," of the NuScale Design Certification Application.

Enclosure 1 is the nonproprietary presentation entitled "ACRS Subcommittee Presentation: NuScale FSAR Chapter 21, Multi-Module Design Considerations," PM-0519-65533, Revision 0.

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

If you have any questions, please contact Nadja Joergensen at 541-452-7338 or at njoergensen@nuscalepower.com.

Sincerely,

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Enclosure 1: "ACRS Subcommittee Presentation: NuScale FSAR Chapter 21, Multi-Module Design Considerations," PM-0519-65533, Revision 0

Enclosure 1:

“ACRS Subcommittee Presentation: NuScale FSAR Chapter 21, Multi-Module Design Considerations,” PM-0519-65533, Revision 0

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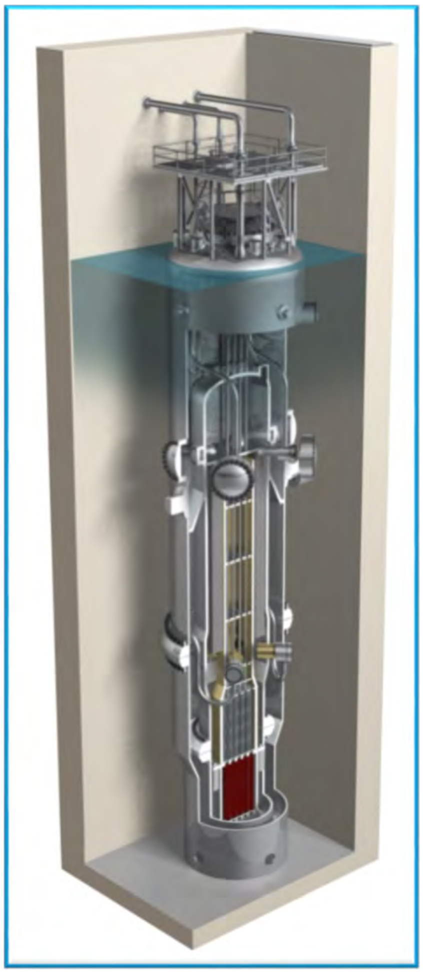
ACRS Subcommittee Presentation



NuScale FSAR

Chapter 21

Multi-Module Design Considerations

May 15, 2019

PM-0519-65533
Revision: 0

Template #: 0000-21727-F01 R5

Presenters

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Senior Reactor Operator 4

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Modular Designs

- Per 10 CFR 52.1, modular design means

  A nuclear power station that consists of two or more essentially identical nuclear reactors (modules) and each module is a separate nuclear reactor capable of being operated independent of the state of completion or operating condition of any other module co-located on the same site, even though the nuclear power station may have some shared or common systems.

- The NuScale Power Plant design is consistent with this definition.

Modular Designs

- 10 CFR 52.47(c)(3) requires evaluation of module operating configurations, considering:
  - Common (shared) systems
  - Interface requirements
  - System interactions
  - Restrictions during construction and startup

Shared Systems

- The NuScale Power Plant is designed such that each NPM can be safely operated independent of other NPMs.
- With the exception of the ultimate heat sink (UHS), safety-related systems are module-specific and functionally independent of shared systems and other NPMs.
  - Emergency core cooling system
  - Containment system
  - Decay heat removal system
  - Module protection system
  - Demineralized water isolation valves

Shared Systems

- The UHS has sufficient capacity to remove heat from one NPM experiencing a DBA, while simultaneously removing heat from 11 NPMs in shutdown and cooldown.
- Long-term heat removal is provided by the UHS and the module-specific safety-related systems without reliance on other shared systems or operator action.

Shared Systems

- Module heatup system
- Boron addition system
- Containment flooding and drain system
- Reactor component cooling water system
- Process sampling system
- Circulating water system
- Auxiliary boiler system
- Site cooling water system
- Nitrogen distribution system
- Demineralized water system
- Fire protection system
- Fire detection system
- Fuel handling equipment
- Module assembly equipment
- Instrument air system
- Gaseous radioactive waste system
- Liquid radioactive waste system
- Normal control room HVAC system
- Reactor building HVAC system
- Control room habitability system

Shared Systems

- Chilled water system
- Ultimate heat sink
- Reactor pool cooling system
- Pool surge control system
- Pool cleanup system
- Pool leak detection system
- Spent fuel pool cooling system
- 13.8 kV and switchyard system
- Medium voltage AC electrical distribution system
- Low voltage AC electrical distribution system
- Highly reliable DC power system
- Normal DC power system
- Safety display and indication system
- Plant protection system
- Plant control system

System Interactions

- The shared systems that have potential for an adverse system interaction or an undesirable multi-module interaction were evaluated.
- The evaluations demonstrate that shared system operation does not result in adverse system interactions, such as
  - a loss of a safety-related function,
  - a DBE and a simultaneous degradation of a safety-related function,
  - a DBE and simultaneous degradation of critical operator information, or
  - a DBE and a requirement for operator actions outside the control room.

Interface Requirements

- Shared systems that serve one NPM at a time are equipped with isolation features that prevent a direct module-to-module interface during normal operation.

- For an adverse multi-module interaction to occur as a result of a failure associated with a shared system that serves one NPM at a time:
  - Abnormal lineups
  - Multiple concurrent failures

Interface Requirements

- The reactor component cooling water system (RCCWS) is the only shared system that directly interfaces with multiple NPMs and is also designed to simultaneously support more than one NPM at a time.

- The RCCWS is designed such that no single failure can cause the loss of RCCWS heat removal from more than one NPM.

Construction and Startup

- During the construction phase that occurs prior to the initial NPM fuel load, the shared and module-specific systems within the Reactor Building (RXB), Control Building (CRB), and Radioactive Waste Building (RWB) are substantially completed with the exception of the installation of additional NPMs.
- The construction method and the phased expansion of NPMs provide assurance that
  - the operating configuration is not materially different than that assumed in the safety analysis and
  - that the independence of NPM safety-related systems is maintained.

Construction and Startup

- In addition, the analysis of shared system interactions continues to apply to the operating NPMs during installation of subsequent NPMs.
- Consequently, restrictions in operating configurations or interface requirements are not necessary to ensure the safe operation of operating NPMs during installation, testing, or startup of subsequent NPMs.

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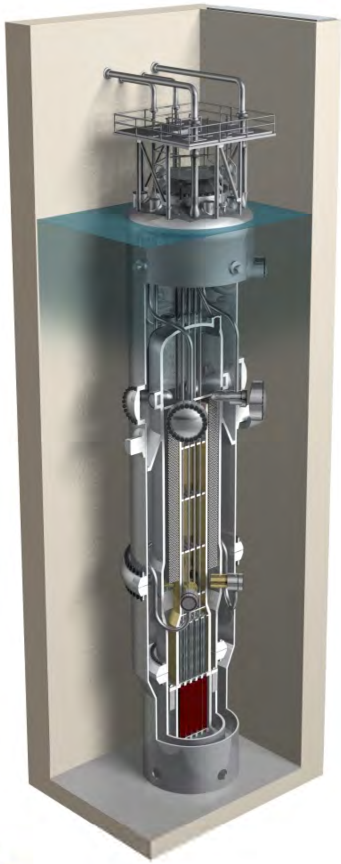

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Safety Evaluation with Open Items: Chapter 21, Multi-Module Design Considerations

NuScale Design Certification Application Review

ACRS Subcommittee Meeting

May 15, 2019

NRC Staff Review Team

- **Technical Staff**
  - Hanry Wagage – FSAR Tier 2, Chapters 6 and 9
  - Joseph Ashcraft – FSAR Tier 2, Chapter 7
  - Nan Chien, Chang Li – FSAR Tier 2, Chapter 9
  - Sheila Ray – FSAR Tier 2, Chapter 8
  - Angelo Stubbs – FSAR Tier 2, Chapters 9 and 10
  - Ryan Nolan, Michelle Hart – FSAR Tier 2, Chapter 15
  - Marie Pohida – FSAR Tier 2, Chapter 19
- **Project Managers**
  - Gregory Cranston – Lead Project Manager
  - Prosanta Chowdhury – Chapter Project Manager

NuScale Standard Design Shared Systems Evaluated by NRC Staff (SE Section) *(No. NPMs Supported)*

1. RWB heating, ventilation, and air conditioning (HVAC) system (9.4.3) *(12 NPMs)*
2. Diesel generator building HVAC system (9.4.4) *(12 NPMs)*
3. Turbine building HVAC system (9.4.4) *(6 NPMs)*
4. Annex building HVAC system (9.4.2) *(12 NPMs)*
5. Containment flooding and drain system (9.3.6) *(6 each for two independent subsystems)*
6. Normal control room HVAC system (9.4.1) *(12 NPMs)*
7. Reactor Building (RXB) HVAC system (9.4.2) *(12 NPMs)*
8. Control room habitability system (6.4) *(12 NPMs)*
9. Boron addition system (9.3.4) *(12 NPMs)*
10. Reactor component cooling water system (9.2.2) *(6 each for two independent subsystems)*
11. Circulating water system (10.4.5) *(6 each for two independent subsystems)*
12. Site cooling water system (9.2.7) *(12 NPMs)*

NuScale Standard Design Shared Systems Evaluated by NRC Staff (SE Section) *(No. NPMs Supported)*

13. Demineralized water system (9.2.3) *(12 NPMs)*
14. Auxiliary boiler system (10.4.10) *(12 NPMs)*
15. Potable water systems (9.2.4) *(12 NPMs)*
16. Ultimate Heat Sink (9.2.5) *(12 NPMs)*
17. 13.8 KV and switchyard system, medium voltage AC electrical distribution system (EMVS), low voltage AC electrical distribution system (ELVS) (8.3.1) *(12 NPMs)*
18. Highly reliable DC power system (EDSS) common (EDSS-C) (8.3.2) *(12 NPMs)*
19. Normal DC power system (EDNS) (8.3.2) *(12 NPMs)*
20. Safety display and indication system (SDIS) (7.2.11) *(12 NPMs)*
21. Plant Protection System (PPS) (7.2.11) *(12 NPMs)*
22. Plant Control System (PCS) (7.2.11) *(12 NPMs)*
23. Utility water system (9.2.9) *(12 NPMs)*

NuScale Standard Design Shared Systems Evaluated by NRC Staff

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- The failure of shared systems that are not safety-related is considered within the NuScale transient and accident analyses and is evaluated in SER Chapter 15, “Transient and Accident Analysis.”
- SER Section 15.0.0, “Classification and Key Assumptions,” contains the staff’s review of the categorization and classification of Design Basis Events (DBEs).
- SER Section 15.0.3, “Radiological Consequences of Design Basis Accidents,” contains staff’s review of the radiological consequences of DBEs.
- The staff discussed multi-module risk including internal and external events in SER Section 19.1.4.9, “Evaluation of Multimodule Risk.”

Chapter 21 SE Conclusion

- Consistent with the SRP and, DSRS if applicable, the staff reviewed information presented in the DCA on the design and operation of the aforementioned shared systems.
- As applicable to NuScale multi-module design considerations and assertions, the staff has documented findings and conclusions in the SER sections cited in the previous slides.

Acronyms

DBE: Design Basis Event
DCA: Design Certification Application
DSRS: Design Specific Review Standards
HVAC: Heating, Ventilation, and Air Conditioning
ITAAC: Inspections, Tests, Analyses, and Acceptance Criteria
NPM: NuScale Power Module
NRO: US NRC Office of New Reactors
NRR: US NRC Office of Nuclear Reactor Regulation
PCS: Plant Control System
PPS: Plant Protection System
RWB: Radioactive Waste Building
RXB: Reactor Building
SDIS: Safety Display and Indication System
SER: Safety Evaluation Report
SRP: Standard Review Plan