Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title:	Advisory Committee on Reactor Safeguards NuScale Subcommittee
Docket Number:	(n/a)
Location:	Rockville, Maryland
Date:	Tuesday, May 14, 2019

Work Order No.: NRC-0328 Pages 1-377

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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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TUESDAY

MAY 14, 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., Michael L. Corradini and Vesna B. Dimitrijevic, Co-Chairs, presiding.

COMMITTEE MEMBERS:

MICHAEL L. CORRADINI, Co-Chair

VESNA B. DIMITRIJEVIC, 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
MATTHEW W. SUNSERI, Member

ACRS CONSULTANT:

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

MIKE SNODDERLY

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P R O C E E D I N G S

8:26 a.m.

CO-CHAIR CORRADINI: The meeting will come to order.

This is a meeting of the Advisory Committee on Reactor Safeguards, NuScale Subcommittee. My name is Mike Corradini. I am the Co-Chair with Dr. Dimitrijevic for today's meeting.

Members in attendance are Ron Ballinger, Gordon Skillman, Matt Sunseri, Joy Rempe, Jose March-Leuba, Dennis Bley, Dr. Dimitrijevic, and myself. And I think we will have Harold Ray joining us by phone.

And Mike Snodderly is the Designated Federal Official for this meeting.

The Subcommittee will review the staff's evaluation of Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation," of the NuScale's Design Certification Application. Today we have members of the NRC staff and NuScale to present to the Subcommittee.

The ACRS was established by statute and is governed by the Federal Advisory Committee Act, or FACA. That means the Committee can only speak through its published letter reports. We hold meetings to gather information to support our deliberations, and interested parties who wish to provide comments can contact our office requesting time after the meeting announcement is published in The Federal Register.

That said, we set aside 10 minutes for comments for members of the public attending or listening to our meetings, and written comments are also welcome.

I forgot to mention we have our esteemed consultant, Dr. Schultz, with us. I apologize. I looked past you.

The ACRS section of the U.S. NRC's public website provides our Charter, Bylaws, letter reports, and full transcripts of all full and subcommittee meetings, including slides presented here.

The rules for participation in today's meeting are announced in The Federal Register notice dated on May 6th, 2019. The meeting was announced as a previous open/closed meeting. We may close the meeting after the open portion to discuss proprietary materials and presenters can defer questions that should be asked, or should be answered, in the private session, in the closed session.

I'll just go off-script and note that, if we start asking something of NuScale or the staff, you guys have to keep us disciplined, and we'll hold it off until the closed session.

No written statement or request for making an oral statement to the Subcommittee has 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 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 the meeting. To minimize disturbances, the public line is being kept in listen-in-only mode. And to avoid disturbances, I request that all members of the Committee and attendees put their electronic devices on mute or noise-free mode, so we don't get interrupted by noises.

I will note for myself that I participated in a PIRT for Severe Accident Phenomena with NuScale in 2009, and we've put that on the record in the past when we met in March and October.

MEMBER REMPE: And I also need to acknowledge that I participated not only in that PIRT, but I performed some other activities for the Applicant back in around that timeframe.

CO-CHAIR CORRADINI: So, I'm going to turn it over to Dr. Dimitrijevic to start us off.

CO-CHAIR DIMITRIJEVIC: This is our third meeting, right?

CO-CHAIR CORRADINI: Her mic is on. I think you need to be louder.

CO-CHAIR DIMITRIJEVIC: I need to be louder? That was never said to me.

(Laughter.)

All right.

CO-CHAIR CORRADINI: There we go.

CO-CHAIR DIMITRIJEVIC: So, you presented the PRA which shows that you're meeting safety goals with the large margins and that your risk is practically zero, because we don't want to really discuss the numbers, although it's 10 to the minus 9.

So, what I heard, that we can concentrate today -- even I saw some of the slides that we saw before -- and as we go through them, we can concentrate on the important issues and, important, staff, how sure we are that that's true, that your risk is practically zero.

So, for example, important assumptions

that impact on the risk, the uncertainties, sensitivities, uncertainty, multi-module issues which are characteristic for your plant, that's why they are interesting and important for us. So, just keep that in mind when you are giving this presentation, and we can just assess it. Did I say "uncertainty" three times? That was my goal, to make sure that we do address those. Okay? So, please.

Thank you.

MS. NORRIS: Thank you, Vesna.

Good morning, everyone.

I'm Rebecca Norris with NuScale. As has been stated, we are here to present Chapter 19 for the ACRS Subcommittee presentation. This is on "Probabilistic Risk Assessment and Severe Accident Evaluation". The packet that you have includes Chapter 19 items for both today and tomorrow. As you know, if you're looking up an acronym, it will be in the very back of the packet. We just made one acronym list.

Today, we will be presenting 19.0 and 19.1, "Probabilistic Risk Assessment and Severe Accident Evaluation," and then, the general overview. So, this is the presentation for both of those.

The presentation team for today is myself, on the bottom, Rebecca Norris. I am Supervisor in Licensing.

Go ahead. You're going to introduce yourselves?

MS. BRISTOL: Sarah Bristol, Supervisor of the PRA group at NuScale Power.

MR. MULLIN: Etienne Mullin. I'm an Analyst in the PRA group.

MR. GALYEAN: I'm Bill Galyean, a consultant to the PRA group.

CO-CHAIR CORRADINI: Just one thing. We have such a terrific system here, you have to speak up because our mics are of various qualities.

MS. NORRIS: And with that, we can go ahead and get started.

MS. BRISTOL: Okay. An overview of 19.0. This really just goes over the general regulations that were looked at in describing the PRA, both the PRA and severe accidents. We note that we performed the PRA for a single module, and we looked at all modes of operations and both internal and external events. The PRA demonstrates that NuScale design exceeds those safety goals with significant margin, as described further in Chapters 19.1 and 2.

In Section 19.1 of the PRA, the objective was to assess the risks associated with all modes and all hazards for a single module. And that is what is involved in that section. We looked at multi-module risks separately using a systematic process, which we will discuss further in the presentation. But, overall, we looked at Level 1, core damage frequency, and we looked at level 2, large release frequency. And we did that for full power internal events, low power and shutdown, and we did crane failure analysis. We did internal fires, floods, external floods, high winds. And we did a PRA-based seismic margin assessment.

The quality process we used, being a Design Certification, ISG 28 came out right in the middle of the time that we were doing our PRA. We've been involved with the Standard Committee work, and from that work, out came this ISG 28. And so, what we did was we still looked at the ASME/ANS PRA standard, and we looked at all the supporting requirements. And we did individual self-assessments for each element of the standard, and we evaluated those independently in each notebook, if you will, for the PRA we looked at the standard, and when the standard directed -- or Reg Guide 1200, as well as the ISG and the standard, is what we used to assess the quality for the PRA.

We had both an independent review of those independent self-assessments from outside consultants and we also had an independent expert panel.

MEMBER BLEY: Sarah?

MS. BRISTOL: Yes?

MEMBER BLEY: You make a big point that you didn't have a peer review. Those sound like peer reviews to me. Can you explain the difference?

MS. BRISTOL: Well, from our perspective, there was really no significant difference in what we did and a peer review. The only difference was that we didn't officially have a team with a leader come in and evaluate each one of those under the NEI guidance of what a peer review is. But we finished our PRA --

MEMBER BLEY: Did you review all the things that are called out in the standard for a peer review?

MS. BRISTOL: Yes, we did, and we evaluated those, and we said -- we met and there's a slide in a couple that discussed that. In general, the self-assessment we did compared to the consultants, we exceeded even some of the points of the ASME/ANS standard.

And then, there were a few that we couldn't meet, walkdowns data, calculations we didn't have at the time. And so, from our perspective, we did as much as we could without an official peer review team.

MEMBER BLEY: Okay.

MEMBER BALLINGER: With respect to this self-assessment documented by notebook authors, I'm going to be kind of a pain in the rear end about this. Your claim is that the plant is extremely safe and the probabilities of certain events are very, very, very low. Fine.

Along with that claim goes a responsibility, at least in my mind, that you have to be very careful about and very deliberate on determining the uncertainties. So, when you say "self-assessment," did you folks establish, independent of what I call "a murder board," within the company or outside the company, where the charge of that board was to find out the mistakes? In other words, not verify the calculations, but take an outside view and say, how can I make this fail? So, did you guys do that?

MS. BRISTOL: And so, what we did was we had an expert peer review group come in. And so, on this panel, we had Dr. Apostolakis be the Chair of that committee. Mark Cunningham was on there, Rick

Grantom, Dave Moore, Per Peterson. And they all evaluated our design, not necessarily from the standard model, but just what is in your PRA and just from a high level.

Next slide, please.

MEMBER BALLINGER: Okay, but I guess that not too many of these folks have ever actually put a calculator in front of them and done the calculation in a very long time.

(Laughter.)

My issue is, did you guys find some people that have done calculations that are independent, that actually could go and find out where the errors were, or discover paths or cutsets, or whatever, that they hadn't thought about?

MS. BRISTOL: And, yes, that was the next consulting group that we had -- next slide -- was they did look at all of those supporting requirements, and those were individuals that had performed different analyses in the industry recently for items from security, operating plants, different events. So, they have been applying the PRA recently and are familiar with the standard. And those individuals evaluated our PRA and looked at each of the individual requirements, and if they thought we met, if they thought we didn't, if they thought there was a gap that we could close between DCA and COL. And so, we did have both that expert side as well as independent individuals currently in the industry applying PRA practices.

MEMBER MARCH-LEUBA: Don't go so fast.

MS. BRISTOL: Yes.

MEMBER MARCH-LEUBA: Following up on that, my concern with PRA analysis -- and you know I don't like it as much -- is not with the math or with the reviews. It's with the input data. This reactor has a lot of first-of-a-kind components, divisions of the protection system, all of those valves around the containment; they're one of a kind. They haven't even been built yet. Okay? How do we get any confidence that the input data for failure rates that you assigned for this particular valve is acceptable or conservative? Because not having built one and not having operated these in the field, the uncertainty is tremendous on the input data. Please.

MR. GALYEAN: So, this is Bill Galyean.

There's a couple of aspects of the question you just asked. I mean, on one side, some of the components that are in the NuScale design are standard nuclear power plant components. And there, we used the generic industry operating data to generate failure rates. Our expectation is that the performance of the components in the NuScale design will be no worse than what has been experienced in the industry.

However, there are some components in the NuScale design that are relatively unique. Okay? And there, we have done detailed kind of piece-part analyses of these components, where we dissect it down into, as I said, individual piece-part. We come up with estimates for each, failure rate estimates for each individual piece-part, combine them into an overall failure rate.

Again, the expectation is not that we precisely determine what the failure rate is, but that we come up with a failure rate that we believe is conservative. Okay? And there, we do account for uncertainty. We do put uncertainty bands on the failure probabilities, the failure rates, and combine them.

An example of that is particular on the initiating event data that we use. Although we do use the industry experience, we deliberately expanded the uncertainty bands on the failure, on the rates predicted were generated in the industry. And so, by expanding the uncertainty bands, we feel like we encompass what the expectation is for a NuScale design.

Getting back to the quality process and the peer review, again, it was kind of a two-pronged process. We did have the expert panel, the George Apostolakis panel, come in to do a high-level overview, to look at what we were doing globally and identify, or at least point to, different aspects that they thought we should be focusing more attention on. And, in particular, they were very keen on looking at what we did for multi-modules, for example, and a few other things.

Sarah already alluded to the self-assessment that we did. Now, in the self-assessment, of course, we used the ASME/ANS PRA standard which goes through each of the individual supporting requirements on what's expected for a PRA. And NuScale staff evaluated each one of those, and then, we had a separate pair of external consultants come in and review the NuScale self-assessment. And they made some comments about how we evaluated each supporting requirement. In some cases, they thought we were optimistic; in other cases, they thought we were conservative in how we evaluated the supporting requirement match.

So, NuScale has been very keen on assuring that the PRA is a high-quality product, and we have devoted a lot of time and effort, through the expert panel and through the self-assessment, to uncover any potential deficiency that may have existed.

CO-CHAIR CORRADINI: Maybe this is for later, maybe it's Chapter 3, but I'm interested in the connection with -- let's just take the IAB as an example or the RVV or the RRV. Those are, from my perspective, somewhat unique. And were those the ones that you did more of what you call a piece-part analysis?

MR. GALYEAN: Exactly.

CO-CHAIR CORRADINI: Okay.

MR. GALYEAN: Exactly.

CO-CHAIR CORRADINI: So, how does that, then, feed into potential testing going into a first module or any new module being constructed, such that you're -- what are you looking for in terms of testing -- and maybe this is not here; maybe this is in 14 or in 3 -- relative to assuring that the reliability is of some level in comparison to what you're estimating?

MR. GALYEAN: Yes, that's something that we can't answer from the perspective of the PRA. I mean, certainly the expectation is that the vendor will be doing testing of the completed valve.

MEMBER BLEY: Well, can I interrupt you there? We've had other cases where we've had the design certs with unique parts. And when they came to this point, they had run extensive tests and were able to report those tests, the results of them. I'm not quite sure why it's not your responsibility.

MR. GALYEAN: No, I mean, the PRA group at NuScale. I mean, NuScale will certainly, there will certainly be some testing done. In fact, there already has been testing of the prototype --

MEMBER BLEY: Have you given a prioritized list of what things need to be tested and why?

MR. GALYEAN: We have not coordinated with the design folks on that aspect.

CO-CHAIR CORRADINI: So, to get to my question, I should wait until 14 or 3? That's what you're really telling me?

MR. GALYEAN: Yes.

CO-CHAIR CORRADINI: Okay. I just want to make sure that what you estimate is not -- I'll pick a number -- must less than what one is testing to see a level of performance; that's all. I'm just looking for the connection. Because I think Jose's question is a fair one, which is it's really the input reliability that tends to drive the result.

MEMBER BLEY: One more question in this area. You mentioned this group of components where you had to do piece-part analysis, if you will. Are you going to have a slide that shows what those were?

MR. GALYEAN: I believe we do have a slide that alludes to that. It's just a single bullet item that --

MEMBER BLEY: How many of those kind of things were there?

MR. GALYEAN: Well, the ECCS valves for one. I'm trying to think.

CO-CHAIR CORRADINI: Those are the three that I thought were unique.

MR. GALYEAN: Three?

CO-CHAIR CORRADINI: Well, RRV, RVV.

MR. GALYEAN: Oh, yes, I was just using collectively the ECCS valves.

MEMBER SKILLMAN: And there's a crane and there's some other stuff in there that's very unique.

MR. GALYEAN: Yes, certainly the crane, yes. We did an evaluation.

MEMBER SKILLMAN: I'd like to weigh into this. I'd like to go back to slide 6, please. As I review Chapter 19, the question that is overarching in my mind is, to what extent has the PRA, and those who reviewed the PRA, fully comprehended that this plant is aquatic? This might as well be a Jules Verne plant. Everything is underwater. And this is the only plant I've ever experienced in my over 50 years that is 100 percent aquatic.

And that has some great advantages in terms of source term, in terms of decay heat removal. Those of us who have been around water and the aquatic environment know that that environment creates challenges that are unique in terms of chemistry, material degradation, operability, inspectability, those types of things. So, unless the people who were reviewing this had that aquatic lens adjusted for their review, it would seem to me that the uncertainties are extraordinary.

MR. GALYEAN: Well, none of the components that were modeled in the PRA are underwater. Okay?

MEMBER SKILLMAN: Oh, they are.

MR. GALYEAN: They're not submerged. They're not -- you know, I mean, they're protected. They're not -- I don't know how to say it. I mean, certainly the containment vessel is underwater, but things like the valves, the containment isolation valves, the ECCS valves, they're not maintained underwater.

But, certainly, the reviewers that we had, both in the Apostolakis panel and in the review of the self-assessment, they were all familiar with the design. And, of course, I mean, they had many of the questions that you all are asking, things about the crane and the movement and the reliability of the components.

So, I hope that -- I don't know what else to say. I mean, the valves are not maintained underwater.

MEMBER MARCH-LEUBA: Yes, what you are saying is that containment is in a vacuum. So, most of the active components are in a vacuum. My concern with being underwater is that they are inaccessible, that they're difficult to work with, difficult to maintain.

MR. GALYEAN: Most of the equipment that needs to be maintained, like the containment isolation valves, they're at the top of the module and there's a maintenance platform that allows access to them. Most of that will take place during refueling, when the upper module is in the dry dock.

MEMBER SKILLMAN: Okay. Let me ask one

more on this slide. The reason I asked you to put this up is because you identify low power in shutdown, including crane failure. When you're in the refueling mode and you're moving that module -- and the module is listed at 762 tons in some of your documents; it's 730 in a couple of others -- how does one make the connection between low-power shutdown, an event involving the module, and an adjacent module?

MR. GALYEAN: We've looked at that. We've looked at the potential for crane failure resulting in a module being dropped off of the crane and possibly impacting an operating module. We've assessed the likelihood of that. And then, we've done a qualitative assessment as to what the potential impact might be on the operating module.

The impact to an operating module is insignificant compared to the hazards that are already accounted for in the internal event initiating events. And, of course, we have done a separate analysis on the crane failure and the potential for a dropped module.

MEMBER SKILLMAN: Okay. Bill, I understand your sermonette about the equipment basically being in a protected environment inside the containment. I got that message and I understood it when I asked the question.

But I want to continue to project my concern. Here you have 12 of these large machines largely underwater for 60 years. I would suggest that that is an operating environment or a physical environment that is unlike any we've dealt with before. And it would seem to me that there are some challenges that we really haven't tumbled to because of that unique situation. And that's all I'm trying to communicate.

MR. GALYEAN: Fair enough.

MEMBER SKILLMAN: Thank you.

MEMBER BROWN: I wanted to make one other along the uncertainty comment that Dennis brought up.

PARTICIPANT: Can anyone else on this line hear what's going on at the ACRS meeting? I haven't had any audio for this meeting yet, and I'm wondering if they're running a little late.

CO-CHAIR CORRADINI: Why don't we just hold a minute so you can clarify? We can hold for a minute or two.

MEMBER MARCH-LEUBA: We need to hold. We need to hold because they're not hearing it.

CO-CHAIR CORRADINI: Yes.

MEMBER MARCH-LEUBA: It's a public

meeting.

(Whereupon, the above-entitled matter went off the record at 8:53 a.m. and resumed at 8:56 a.m.)

CO-CHAIR CORRADINI: All right, let's keep on going.

MEMBER BLEY: Okay, Bill, I'm going to get you with one more.

CO-CHAIR CORRADINI: Oh, wait a minute. I think Charlie was in the middle of something.

MEMBER BLEY: Oh, I forgot Charlie was talking.

MEMBER BROWN: That's all right. Everybody forgets me. I almost forgot my question, but not quite.

Along the line of, excuse me, uncertainties, if I recall properly, the pressurizer level is one of the trip or trigger functions for a lot of the operations relative to the module protection system and some other type of stuff, and it's presently just supposed to be a radar-based system if I'm not mistaken. Isn't that correct? Did I get that right?

(Simultaneous speaking.)

MEMBER REMPE: That's what they told us -- That was --

MEMBER BROWN: Pardon?

(Off-mic comments.)

MEMBER REMPE: That was actually presented by NuScale and acknowledged by NuScale in a public meeting last month, okay?

MEMBER BROWN: Good.

(Off-mic comments.)

MEMBER REMPE: Right, okay. Call it an unknown.

MEMBER BROWN: Well, this unknown system that nobody knows how it operates, it hasn't ever been in a system like this before in a saturated steam 600-pound or whatever the pressure level is, excuse me, high temperature or high pressure environment. The frothing level of the level that you're trying to measure is a significant factor in trying to get an accurate measurement.

And if you look up, which I did, some of the products that are available, they try to institute some fairly sophisticated algorithms to evaluate the frothing level and get the errors down, and if you don't do that, you can have errors in that signal of upwards of 20 or 30 percent, and that's an uncertainty relative to your level and how that operates with all the rest of your systems.

How -- I mean, that's another system that is totally unrelated, but yet how do you factor in or even determine an uncertainty because you haven't developed it, you haven't applied it, and you don't even know how it's going to perform?

MR. GALYEAN: Yeah, fair enough. You know, the technology for the sensors, you know, might be somewhat unique, and we don't --

MEMBER BLEY: Mike?

MR. GALYEAN: We don't --

MEMBER BLEY: Should we hold this for a closed session or --

MEMBER REMPE: Yeah, NuScale can hold --

MEMBER BLEY: I'm a little hesitant.

MEMBER REMPE: -- this to a closed session if you'd prefer.

MR. GALYEAN: Yeah, I'm not going to talk about the technology, okay? All I'm going to say, all I want to say is that these will be safety related. They will be evaluated or proven to be safety-related sensors through whatever means necessary.

Additionally, typically the equipment that needs to be actuated automatically, the safety-related equipment, for example, the ECCS and the DHRS, Decay Heat Removal System and Emergency Core Cooling System, will be actuated -- will rely on multiple sensors, multiple physical sensors, whether they're level sensors, pressure sensors, level in the reactor pressure vessel, level in the containment vessel, pressure. There are typically multiple means for actuating these safety-related systems.

In the PRA, we only take credit for either one or two sensors, okay, to actuate these systems. We don't take credit for all of the diverse means for actuating these systems, okay, and we view that as a conservatism in the way we model system operation.

And so although we don't explicitly account for uncertainty that might be produced by the technology, we do incorporate conservatism in the way we model the actuation of these systems.

MEMBER REMPE: So, Bill --

CO-CHAIR CORRADINI: Hold on a second. I think you want to try to reestablish the public line, so let's hold and see if we can get this done. Do you want to test it? Can you test it?

PARTICIPANT: Who is on the line?

MS. FIELDS: This is Sarah Fields, a member of the pubic.

MR. SNODDERLY: Okay, thank you, Sarah. Okay, Sarah, yeah, we're going to leave this line open. Please just be in listen mode, and that would be greatly appreciated. We want to make sure you can hear, okay?

MS. FIELDS: Okay.

MR. SNODDERLY: Thank you. Sarah, can you still hear us?

MS. FIELDS: Yes.

MR. SNODDERLY: Okay, thank you.

CO-CHAIR CORRADINI: Okay.

MEMBER REMPE: So we're going to rely on diverse sensors. It may be possible. That's what industry does with boiling water reactors and pressurized water reactors.

That requires a lot of operator guidance and a lot of -- and it's not acknowledged, this diverse, like you're going to use the flux detectors to help you decide when the water level is to a certain height.

That's not documented in any of the DCA document sections that I remember reviewing, and a lot of times when we ask about such guidance, it's been cut off to the COL applicant stage.

Is there some place that -- I know the sensor technology is in ITAAC for this water level, but I've not seen anything about this guidance in using diverse sensors and all of that, and where is that documented and how do we know that the actual COL applicant will be able to instruct the operators to do this?

MR. GALYEAN: I'm not quite such which -- I'm talking about automatic actuation in the module protection system.

MEMBER REMPE: Okay, so is that actually encoded in if the flux detectors have some sort of change? I mean, I've not seen any of that documented in the DCA.

MR. GALYEAN: I can't tell you off the top of my head which sensors or which parameters, you know, would actuate the various systems. It depends on things like the initiating event, you know, whether it's a transient, or a LOCA, or, you know, even a loss of offsite power or whatever. Different physical parameters will trigger the ECCS and DHRS system. Etienne, did you want to --

MR. MULLIN: I mean, yeah, like one, if you have a LOCA inside containment, for example, you could have a low pressurizer level, low RPV pressure or signal, a high CNV pressure signal. That will all go off within --

MEMBER REMPE: But the documentation

usually reflects that the triggers are the pressurizer water level as Charlie indicated. I have not seen other sensors that would be put into the protection system. Have you, Charlie?

MEMBER MARCH-LEUBA: You know, I'm sorry to disappoint you, but we saw it two weeks ago on a slide.

MEMBER REMPE: Yeah, but the water level --

MEMBER MARCH-LEUBA: At least it was -- yeah, the water level --

MEMBER REMPE: Okay, and that's the DHRS.

MEMBER MARCH-LEUBA: No, no, the LOCA pressure level triggers a secondary isolation which will then eventually create the high steam pressure, which will trigger DHRS.

MEMBER REMPE: But we're not talking about DHRS in this case.

MEMBER MARCH-LEUBA: These four signals trigger the DHRS, high pressurized pressure, high and hot temperature, high stream pressure in the primary or loss essay voltage.

MEMBER REMPE: But there are other -- the DHRS is one aspect, but there are other things --

MEMBER MARCH-LEUBA: Many more.

MEMBER REMPE: Right, that the pressurizer level triggers, and the PRA --

MEMBER MARCH-LEUBA: Pressurizer level?

MEMBER REMPE: I thought -- is there not some systems that depend solely on the pressurizer level?

PARTICIPANT: No.

CO-CHAIR CORRADINI: I think the answer is no.

MEMBER REMPE: Okay.

PARTICIPANT: Right.

MEMBER REMPE: Then I have another question if we can change the topic. The expert panel, the review group that reviewed the, the folks who can still punch a calculator as I've identified them, we only know they can punch a calculator. What level of industry folks are they and can you even give names? Are they industry folks? Who are they?

MR. GALYEAN: Dave Blanchard and Wes Brinsfield. They do a lot of consulting for the industry. They've both been in the industry doing PRA for a long time. In fact, Dave Blanchard, I think, worked on the Big Rock Point PRA, and he's been involved in the PRA business all --

MEMBER REMPE: Okay.

MR. GALYEAN: -- all along.

MEMBER REMPE: Okay, and the, on my package, it's slide nine, but I think it's a different number. Oh, there is it. It's up there right now, this no major concerns or objections.

Aren't there some assumptions also that the expert panel made, like the treatment of multi-modules be reviewed by the greater PRA community, and that's -- they had certain assumptions to come up with this no major concerns or objections. It maybe makes the slide a little too positive.

MR. GALYEAN: Yeah, well, I think that was Apostolakis' comment, and he was more being a proponent for expanded use of PRA in general. Doing multi-module assessments is obviously new to the PRA community.

I know there has been some work on doing multi-unit PRAs, but, you know, he was just -- he was very complimentary of the approach we took, but he felt that it warranted a broader, how do I say, engagement with the industry just to propagate the methods that we developed more than anything else, so he was just trying to be PRA promotional as much as anything else.

MEMBER REMPE: Okay, so perhaps there was a difference of opinion is what you're trying to convey, the majority opinion of the group?

MR. GALYEAN: No, I think the whole group was on board with that. They liked what we did and they just thought that we should share it with the rest of the PRA community basically.

MEMBER REMPE: Okay, I guess what I had reviewed in the summary was a little different take. That we assumed this to come up with these conclusions is what I recall the quote, or perhaps I've misremembered or something?

MR. GALYEAN: I --

PARTICIPANT: Yeah, I don't know, sorry.

MR. GALYEAN: I can't remember --

MEMBER REMPE: Okay.

MR. GALYEAN: I can't remember that, so.

DR. SCHULTZ: Bill, in the sequence you presented today with regard to the overall QA program and review of the, or quality program and review of the PRA, the sequences as you've described, the expert panel saw this relatively early compared to the other groups?

MR. GALYEAN: We had the expert panel engaged at multiple places --

DR. SCHULTZ: But --

MR. GALYEAN: -- in the development of the PRA.

DR. SCHULTZ: Did they --

MR. GALYEAN: I think the --

DR. SCHULTZ: Did they wrap it up? Did they wrap --

MR. GALYEAN: That's right.

DR. SCHULTZ: -- their review up as a result of the improvements that you've described here or did they finish their job before the final reviews by the other teams?

MR. GALYEAN: I don't recall the exact timeline. It seemed to me the expert panel that we had was fairly close to the end of the PRA development, and of course the self-assessment and the external review of our self-assessment was also fairly late in the process. I think the expert panel actually finished up after this self-assessment.

DR. SCHULTZ: That's what I was getting to or wondering about. All right, I appreciate that.

MR. GALYEAN: Yeah.

DR. SCHULTZ: And the level of effort of the expert panel and the other teams, how would you describe that?

MR. GALYEAN: As I said, the expert panel came and visited with us a couple of times over maybe a two-year period, and we had a couple of telephone conference calls. We certainly sent them various reports that we generated. So all in all, each member, again, I'm just trying to recall, maybe spent a couple of man months looking at what we've done, and so.

DR. SCHULTZ: Thank you.

MEMBER REMPE: So to beat a dead horse, on page 19.1.97 of the open DCA, it says the CVS -- the MPS -- the safety-related MPS generates a CVCS isolation signal if the high pressurizer level set point is exceeded.

Other places, when you talk about the pressurizer water level, you do say pressurizer water level or some other signal, but here, you don't. Is it just you omitted this in the DCA?

MR. GALYEAN: I can't answer that question. I'd have to go back and research that. I don't recall.

CO-CHAIR CORRADINI: Why don't we try to do that offline then?

MR. GALYEAN: Okay.

MEMBER REMPE: Thank you.

MEMBER SKILLMAN: I'd like to ask on that last set of cards, the terms CDF and LRF tied to large reactors' current use may be misleading a NuScale design. I can interpret that two different ways.

I can interpret that to mean that if you're kind of set on the current 10 to the minus four and 10 to the minus six and you look at NuScale, you find that use of the current metric doesn't quite fit, but I can also interpret that to mean a 10 to the minus eight to the 10 to the minus 10 on NuScale should be met with great skepticism, and so I wonder what was really intended by that comment?

MR. GALYEAN: The comment is -- there -- remember in the NuScale, the NuScale cores are small, okay? A NuScale core is only about five percent the size of a large light water reactor, okay, five percent.

If you take all 12 NuScale cores and pile them up into one big core, you are still only half the size of a large light water reactor.

And so when you compare core damage frequencies, you're comparing a core damage event in this tiny NuScale five percent core to a core damage event in a large light water reactor with a core 20 times as large, okay.

I mean, how is that a rational comparison, okay. That's what that comment is intended to convey.

MEMBER SKILLMAN: I can understand that logic, but I come from a background where even an 80-megawatt reactor out of control poses a huge concern, and so while I try to understand the practical perspective that you've just communicated, I've lived the life where a much smaller core can evacuate a city.

So I am one member on the ACRS that is on the one hand supportive of the plumbing, but cautious on your conclusions on your PRA.

MR. GALYEAN: Well, all I can say is that we're talking about, you know, 21st century technology here, okay. I mean, we have a lot of information or a lot of knowledge that we've gained over the years of, you know, 50 years of nuclear power operation with material science, with the development of safety systems, digital, you know, fiber optics.

You know, each of these five percent cores, okay, are contained in their own reactor pressure vessel. Each one is contained in its own high pressure containment vessel. They're all submerged in the ultimate heat sink, all contained in a seismic Class I reactor building.

You know, the design, the NuScale design is incredibly safe, albeit unique, but the uniqueness is what makes it safe.

CO-CHAIR CORRADINI: So let me, I think I'm going to move us on if it's all right because I think you understand Member Skillman's position, but let me put it in a direction, I think, that can take us forward, which is what you're really saying is, "I don't just look at the CDF. I essentially look at the release fractions and the source term."

Is that going to be discussed within 19 or are we going to wait for that for the topical report? Because I want to make sure because there are going to be questions about the source term relative to its application. Are we going to talk about it today?

PARTICIPANT: It wasn't -- we --

MS. BRISTOL: No, not specifically.

CO-CHAIR CORRADINI: Nor tomorrow?

MS. BRISTOL: Correct.

CO-CHAIR CORRADINI: Okay, but eventually we want to talk about it?

MS. BRISTOL: Yes.

CO-CHAIR CORRADINI: Okay, because I think, if I understand where Member Skillman is coming from, I think that's the issue.

MEMBER SKILLMAN: Avogadro's number is a

big number, 6.02 times 10 to the 23. I don't care how big your core is. That is a big number and a little bit goes a long, long way. I lived that life at TMI-2.

MEMBER BLEY: I'm stuck on two things. One will be easy for you to answer, so I'll ask that first. You have a section in the report on passive system uncertainty that raises some very important issues, I think. Do you have slides on that or are you going to talk about that?

MS. BRISTOL: We don't have specific slides in this. We talked about that quite a bit in October. We can answer some questions. We have a couple of bullets, but no specific slides.

MEMBER BLEY: You've raised the issues. Have you tried to quantify them?

MS. BRISTOL: In the passive safety system reliability?

MEMBER BLEY: Yes, yes, yes.

MS. BRISTOL: Yes, we did quantify those, and that was that whole analysis was to develop those values for reliability of the emergency core cooling system and DHRS, Decay Heat Removal System, passive reliability.

MEMBER BLEY: Okay, we haven't gotten to

chapter six yet. We're going to get there, so the ECCS valves are in there, but it won't talk anything about test frequencies, I don't think. What did --

CO-CHAIR CORRADINI: I was just going to help you. If you look, I think, in one of our notes from the staff, I think that actually resides in chapter 396, what you're looking for.

MEMBER BLEY: Valve test frequencies?

CO-CHAIR CORRADINI: I think so.

MEMBER BLEY: What did you -- did you assume the same thing in the PRA?

MR. GALYEAN: Assume what?

MEMBER BLEY: The testing frequencies for the ECCS valves. So if you're doing the PRA on the equipment, you've got to consider the possibility --

MR. GALYEAN: Right, I believe we just assumed they would be tested during refueling.

MEMBER BLEY: Okay, once a year?

MR. GALYEAN: Yeah.

MEMBER BLEY: How does that compare with --

MR. GALYEAN: Once every two years.

MEMBER BLEY: Oh, that's right, once every two years. How does that compare with the test frequency for similar valves, not the same valves, in the current plans?

The reason I'm bringing this up is valves, especially valves that rely on springs or other passive modes of operating them, when they sit for a long time, don't work as well as they did before they sat for a long time, and yours are going to sit for a longer time than others.

I had some experience a long time ago with safety valves that sat for several years before they got tested, and we finally got around to testing, none lifted anywhere near where they were supposed to.

So these things sitting here for a long time ought to have some kind of degradation and their common cause impact ought to be affected by that.

And most test programs wouldn't look at this, but I think this is an area where it's essentially when you get a test program that it looks at these kind of, I'll call them short-term aging effects because these are really important valves and they're in a unique place for testability, so I wanted to raise that with you.

I wasn't here for that discussion on the passive degradation. When you quantified the effects for passive systems, did you make assumptions about how much degradation you would have due to any of the aging issues that could affect thermohydraulic systems with a fairly delicate balance?

MR. GALYEAN: We treated it as an uncertainty.

MEMBER BLEY: And how did you judge the worst end of the uncertainty bound?

MR. GALYEAN: I can't recall off the top of my head. I mean, certainly for the passive system reliability, we did a series of RELAP runs to identify the major dependent variables for the success of heat transfer, okay.

And once we identified those, we developed uncertainty bands on each one, and then we did Monte Carlo simulations where we sampled from those uncertainty bands and generated --

MEMBER BLEY: Where did the uncertainty bands come from?

MR. GALYEAN: Well, again, I'd have to go back and look, you know, at how those limits were determined.

MEMBER BLEY: Certainly not from any tests anywhere or were there tests --

MR. GALYEAN: No, I --

MEMBER BLEY: -- far away that you --

MR. GALYEAN: I would guess it was simple engineering judgment that --

MEMBER BLEY: By people who probably --

MR. GALYEAN: -- developed a high --

MEMBER BLEY: -- never saw this phenomena.

MR. GALYEAN: -- you know, a high and a low value for these physical, for these --

MEMBER BLEY: In the past, on the other design certs we've looked at, we've recommended strongly that when, should there be somebody who ever builds this, when they are approaching fuel load and are supposed to have a really complete PRA, that these issues be thoroughly addressed, and I don't think they have yet.

And at least from my way of thinking, these valves and the possible short-term aging of them and long-term aging within the systems for the passive effects are things that could upset these wonderfully low numbers we calculated.

CO-CHAIR DIMITRIJEVIC: Well, you know we are still in the period of consideration and review, but you can see what are the subjects we are most interested in. So what Dennis just brought up is you have these two events which are the ECCS and DHRS, which both are failure of passive heat removal, right?

Their failure probability is slightly

smaller than valves itself. I think DHRS is 40 minus six and ECCS is 20 minus seven. So, I mean, I don't know how you made those guesses. Maybe it's based on the thermohydraulic, but that don't factor on those multi, very small. It's just two. So you obviously don't think there is a huge uncertainty associated with this.

So that's what is really -- when I look here, we talk uncertainty, uncertainty, uncertainty, but when we look in the results, we don't see uncertainty. You know, your idle factors and total numbers are also very small. Your mean value is almost identical to your point estimates.

Somehow this uncertainty discussion didn't show up when you present the total results, and input data also don't show these uncertainty bands. So, I mean, it's very difficult to put decisive tests to determine the answers to the bands.

That's not easy for the PRA, but I think those two events which you tried to incorporate, there is some attempt to put this uncertainty, but then you make them very certain.

MR. GALYEAN: Well, I guess I would argue that just the presence of those events in the PRA model represents a treatment of uncertainty, okay.

Those events represent -- okay, given that the system operates the way it's supposed to, that the valves open and, you know, you've got flow going, despite that fact, heat transfer to the ultimate heat sink is insufficient.

CO-CHAIR DIMITRIJEVIC: Right.

MR. GALYEAN: Okay, that's what those events represent.

CO-CHAIR DIMITRIJEVIC: They allow the passive --

MR. GALYEAN: That's right.

CO-CHAIR DIMITRIJEVIC: -- and all of these other things which you need for the passive heat removal.

CO-CHAIR CORRADINI: Just you guys are talking as two subject matter experts at each other. Let me make sure I'm following this. What you're asking, Vesna, is that there are --

What you're saying, Bill, is that there are certain events that essentially take into account partial actuation of passive systems that cover what is the uncertainty that Vesna and Dennis are concerned about? Am I misunderstanding?

MR. GALYEAN: Not partial actuation. I mean, it's that these systems in the NuScale design, ECCS and the DHRS, rely on natural circulation, right, for heat transfer, and that these events that we're talking about represent the failure of gravity, right, I mean, that the natural circulation fails to transfer heat from inside the reactor pressure vessel to the ultimate heat sink.

CO-CHAIR CORRADINI: But if I might just push the point, Dennis is -- I'm just trying to link it. Dennis' point is that I might have, if I understand it, I've got a valve. It's supposed to open upon a signal, but it's sitting there and it's, I'll use the word corroded just for want.

It's somehow aged and it doesn't open 100 percent. It opens five percent or something like that. That essentially then would affect natural circulation or pressure-driven flows. Am I -- I think that's where Dennis was going.

CO-CHAIR DIMITRIJEVIC: The both, actually.

CO-CHAIR CORRADINI: Both?

CO-CHAIR DIMITRIJEVIC: Both, either the valve failure rate or the passive heat removal failure rate --

CO-CHAIR CORRADINI: Right.

CO-CHAIR DIMITRIJEVIC: -- given the valve operated right.

CO-CHAIR CORRADINI: And your point back is that you've, I'm not going to use the word bounded, but you've considered it by some sort of bounding scenario?

MR. GALYEAN: Well, we did look at the potential impact of partial opening of these valves. I mean, we did a number of sensitivity studies which I don't think -- you know, I don't know how we documented, but we --

CO-CHAIR CORRADINI: Well, that's going to be my next question, is where do I look for that?

MR. GALYEAN: Yeah, it would be in a supporting PRA engineering report. It won't be in the --

CO-CHAIR CORRADINI: Okay.

MR. GALYEAN: -- you know, DCA or the FSAR.

MEMBER BLEY: And those were not made available here, but during the -- we can ask the staff about this too because they did audits --

MR. GALYEAN: Right.

MEMBER BLEY: -- and we looked at some of those --

MR. GALYEAN: Okay.

MEMBER BLEY: -- reports on the electronic system, but I don't know if they ever had hard copies of them, but I'm going to -- sorry.

(Simultaneous speaking.)

MEMBER BLEY: We're not talking about failure of gravity. We're talking about increasing friction, something like that, that has happened in many places in the past.

And to me, graded for identifying the issues, running a million thermohydraulic runs if you haven't covered the range of things that could happen doesn't help you. It sounds good, but it doesn't help, so understanding how you set those bounds over the life of the plant, trying to think this through because this is a plant that has to work as well later.

You might not have all of the valves open. You might have some of them open. They could be partially open and then it doesn't take as much of a fouling problem to mess up the thermohydraulics. So those are things that are just crucial to believe in the results.

MR. GALYEAN: Agreed, we did. The passive system reliability work was documented, again, in a PRA engineering report, and that, I do believe that was one of the reports that was made available to the staff for audit.

CO-CHAIR CORRADINI: Okay.

MEMBER MARCH-LEUBA: I have a question too while we're talking about DHRS. DHRS has three, at least three failure modes. One is the hardware failure if the valves don't open, if they had been arresting for a while, and then you have non-condensables, which are supposed to be prevented by the level on the DHRS line.

But I believe that's a minor actuation and there's a human error probability, and also the overfill possibility, which is prevented during the startup by establishing some vacuum and, you know, more operation. You could consider leaving that. Have you considered those other failures?

MR. GALYEAN: Those were part of the parameters that were buried in the PSSR analysis. What we call the passive safety system reliability analysis looked at both DHRS and ECCS as almost two separate analyses.

MEMBER MARCH-LEUBA: Right, but do you have non-condensables --

MR. GALYEAN: Non-consdensables, the volume.

MEMBER MARCH-LEUBA: It completely takes the DHRS to zero.

MR. GALYEAN: That's right. You know --

MEMBER MARCH-LEUBA: Can you assume that with certain probability?

MR. GALYEAN: We used it as a parameter that we varied. We put an uncertainty on it.

MEMBER MARCH-LEUBA: No, I mean, if the operator fail, I mean, if that pressure, the level sensor on the DHRS line fails to identify that nitrogen is building up there, when you turn it on, it won't work. I mean, it's not that it would be degraded. It won't work, zero.

MR. GALYEAN: Well, for the most part, you know, we certainly assume that the systems operate the way that they're designed, okay, and --

MEMBER MARCH-LEUBA: Yeah, but if it's an operator action, PRA --

MR. GALYEAN: It's not an operator action. That's not an operator action, okay. That's an automatic actuation.

MEMBER MARCH-LEUBA: Even though the operator relies on automated action by a digital protection system which may or may not happen with a certain probability. Was that considered on the PRA?

MR. GALYEAN: Well, certainly we account for the fact that the automatic actuation fails to operate. I mean, that has a certain failure probability associated with it, and so whether that fails completely or it fails in a timely fashion and results in a failure as you talk about is, you know, it's still a failure, so that --

MEMBER MARCH-LEUBA: Is one of the inciting events is --

MR. GALYEAN: That's right.

MEMBER MARCH-LEUBA: -- I build up non-condensables and didn't notice?

MR. GALYEAN: It's not explicitly identified in that fashion. It's identified as a failure of the actuation system to operate as designed.

MEMBER MARCH-LEUBA: And if it fails DHRS to zero?

MR. GALYEAN: That's right. If it fails, the DHRS fails, right.

MEMBER MARCH-LEUBA: Okay, different subject, but more dear to my heart, with the failure probabilities, the input data, let's assume that you run all of your best estimates for these valves and you come up with a failure probability of 10 to the minus four. I'll give you that.

And then you start to operate in the reactor, and within three months, the valves start leaking, meaning it failed. Does that invalidate your PRA and you are supposed to go and recalculate your probabilities because you obviously missed it?

I mean, if you're assuming that your probability of failure is 10 to the minus four, and now either the valve starts leaking or you run a test and it didn't work properly, and you caught it.

The probability is not 10 to the minus four because you are testing two or three times. Your probability is more like 50 percent, so that invalidates your PRA and you have to redo it?

MR. GALYEAN: Well, it's certainly standard practice in the PRA industry to maintain the PRAs through the operating life of the plant, and there are requirements that the PRAs be updated at least every two years, and certainly there is the NRC reactor oversight process that monitors the performance of equipment modeled in the PRA.

MEMBER MARCH-LEUBA: My point is every two years, you would be testing these valves. If one fails, did you change the input data from 10 to the minus four to 0.5?

MR. GALYEAN: It would be --

MEMBER MARCH-LEUBA: Or do you keep saying, oh, it really wasn't 10 to the minus four? It was a fluke test?

MR. GALYEAN: No, certainly there are techniques for updating generic data with operating experience data, the Bayesian techniques for example. It wouldn't change to 0.5. It would be changed to something in between.

The expectation certainly is that that failure mode would be addressed in some fashion and the valve repaired. No one wants to have repeated failures occurring for safety-related equipment.

MEMBER MARCH-LEUBA: Yeah, I'm looking at the NUREG, NUREG CR 70.37. That's the industry performance of relief valves and more stuff, and the valve failure probability based on licensee event reports was already there. It's all over the place.

I mean, there are some types of valves that fail with a 15 percent probability and those are the pilot operator valves which are more like the ones you've got, and they're, all of these are 10 to the minus four and 10 to the minus five.

So I just wanted to put on the record that I don't share your confidence that you know the input data for your analysis that way, and I would like to see some commitment or some requirement from the staff that if during testing, the testing invalidates your input data, you have to do something.

MS. BRISTOL: Right, moving onto the initiating events. We, as discussed earlier, we can go over these various topics we talked about in October, but the initiating event analysis was, we looked at various inputs to identify the potential initiating events that could affect a NuScale module.

It was a deliberate effort to be complete and comprehensive. We developed a NuScale specific master logic diagram to look at the various impacts that could affect a module. We looked at FMEAs for all of the systems that could impact and cause potential plant upsets.

We looked through various applicants and traditionals lists of those initiating events. And continuously over the past seven years, we've continued to look at the design as it's developed and incorporated those insights into developing these initiating event --

MEMBER BLEY: Can I ask you a question about the magic logic there, magic?

(Laughter.)

MEMBER BLEY: This is a language issue. It seems to me on a lot of these, you talk about losing control and then things go either up or down. On heat removal, you talk about insufficient heat removal, but you have at least four events that are increased heat removal, and I assume that's intentional and that didn't mean just insufficient heat removal. It meant upsets in heat removal.

Is that true or do you think that inadvertent turbine control valve opening is insufficient heat removal? You haven't looked at this in a long time.

MS. BRISTOL: Yeah, sorry.

MEMBER BLEY: Look at it later.

MS. BRISTOL: Okay, I will do that.

MEMBER BLEY: And it's a matter, I think it's just a matter of language, but there are a number of those kind of things --

MS. BRISTOL: Okay, thank you.

MEMBER BLEY: -- in that logic diagram. I think it's reasonably complete, but not exactly as stated.

MS. BRISTOL: Okay, thank you. If you look at the next slide, we go through, these are the initiating events that were evaluated in the PRA. We looked at injection line breaks inside the vessel and outside of the vessel.

We also looked at the CVCS, chemical volume control system discharge line outside of the containment vessel and the significance of these events was injection capability through CVCS. That's our main injection source to get inventory into the reactor pressure vessel.

We looked at spurious ECCS valve actuations. We looked at loss of power, both DC and offsite power. We looked at two failures, secondary line breaks, both in the steam and feedwater, as well as the DHRS. We looked at other LOCAs inside containment, whether that be a safety valve lift, a pressurizer heater over pressurization event.

We looked at just general reactor trips and we looked at loss of support system, instrument air, power, so what impacts to the support systems maybe CVCS or CFDS would be impacted to respond to an event --

MEMBER BLEY: You --

MS. BRISTOL: -- in that loss of support system.

MEMBER BLEY: You talk about how you were able to collapse this to about 10, and I remember in some of the LOCA discussions, you described other LOCAs that would be like the ones you've had, so they were adequate models.

When you came up with the initiating event frequency, did you get the initiating event frequency for the specific event or did you accumulate it for all of the things that you've thrown in that same category?

MS. BRISTOL: We evaluated, for example, the CVCS injection line was the piping in that system, and so a break in that pipe. For the general reactor trip, we looked at all of the various impacts that could cause a trip, loss of circ water, manual trips, loss of support system as we say, instrument error. So we tried to incorporate the various failures of the systems that would support that initiating event frequency.

Then we, again, moved on. For the accident sequence analysis, we looked at those initiating events and identified the various conditions that a module could experience in response to those initiating events, what systems would respond.

We identified the key safety functions, that heat removal, reactivity control, containment integrity, what responses to those initiating events needed to occur in order to get us to an end state, and those end states we evaluated were core damage as well as large release frequency, and so it was the success or failure of those sequences that we evaluated.

MEMBER BLEY: Sarah, I had to go back and find my notes.

MS. BRISTOL: Okay.

MEMBER BLEY: There's a place where you talk about loss of coolant accidents, and you talk about for a number and you describe some additional smaller RPV penetrations. The staff finds that the plant response can be expected to be similar to or bounded by what you explicitly modeled in the CVCS line break because they have similar mitigation.

What I was asking you is did you combine the initiating event frequencies of all of those multiple LOCAs into the CVCS line break LOCA?

MS. BRISTOL: Yes, we did.

MEMBER BLEY: Okay, thank you.

MS. BRISTOL: Yeah. For success criteria, for the level one, we used core damage frequency, and the core damage then was defined by 2,200 degree Fahrenheit peak cladding temperature, and we demonstrated that over 72 hours. For --

MEMBER MARCH-LEUBA: The criteria for one node reaching 2,200 or a significant number of pins?

MR. MULLIN: Anywhere, anywhere in the core, yeah.

MEMBER MARCH-LEUBA: And do you want to stick with that? That's extending the -- it's going to bite you later if -- it's a bad idea to do it because -- I keep pushing. I want to find an event that will get you 2,200 in one node and it won't be of any consequence, but if you stick to that, that is bad for you and it's not necessary. I mean, I think it's a bad idea.

MR. GALYEAN: Remember, this is, we're talking about beyond design basis accidents here, and so no matter what kind of definition we employ for core damage, you can always postulate failures that would get you there, and that's all we're doing here is just defining what we mean.

When we talk about a core damage frequency, we're saying here is the frequency of reaching this state in the core, okay. And again, we are limited to beyond design basis accidents here, or not limited, but we are talking about beyond design basis accidents here.

MEMBER MARCH-LEUBA: I'm trying to help you here.

MR. GALYEAN: I appreciate that. I appreciate that.

MEMBER MARCH-LEUBA: This might come back and bite you.

MR. GALYEAN: Yeah, and --

MEMBER REMPE: But he is just telling --

MEMBER MARCH-LEUBA: It's unnecessary.

MEMBER REMPE: He is just saying what they did. I mean, even if he wanted to change his mind, right, not today.

MEMBER MARCH-LEUBA: I'm just saying. The same way I tell you when you do something wrong, it's wrong. I think this is too conservative. In operating reactors, we accept one percent failure of the fuel and it's during normal operation.

MS. BRISTOL: Okay, and the level two success criterion we used as the official definition, and we'll talk about maybe a little bit different when we get to the containment event tree, but we defined it as the source term resulting in acute whole body 200 rem dose at the site boundary for an individual standing there for 96 hours.

As we go through the event trees, we did various simulations with, various TH simulations with RELAP, and we used our initial safety-related code and model, and then we modified it to meet the PRA needs, and that included adding in chemical and volume control system and containment flooding and drain system models to the code, as well as we developed various core models as we were able to simulate beyond design basis transients using our PRA codes for that, including ATWS as well as other --

CO-CHAIR CORRADINI: So let me ask, does that require a software development or just simply very clever user modeling with NRELAP?

MR. MULLIN: No, these are just input models.

CO-CHAIR CORRADINI: Okay, fine. I guessed that. I just wanted to make sure.

MEMBER SKILLMAN: Sarah, why isn't the containment evacuation system on that first bullet?

MR. MULLIN: It's not used to mitigate any accident.

MEMBER SKILLMAN: Is the assumption that you have a preexisting vacuum that ensures, or either ensures or doesn't ensure heat transfer?

MR. MULLIN: I mean, in the model, in our NRELAP5 model, there's a flow path that pulls a vacuum continuously during the event until you isolate that flow path, until the transient starts to actually --

CO-CHAIR CORRADINI: But to follow on Dick's point, you evaluated both -- I seem to remember in previous discussions we had in the previous meetings, you evaluated both ways, one where it works and one where it doesn't work. That is you actually have some air pressure --

MR. MULLIN: Yeah, we evaluated the failure to isolate that --

CO-CHAIR CORRADINI: Okay, that's what --

MR. MULLIN: -- line.

CO-CHAIR CORRADINI: Whatever the right way of saying it is.

MR. MULLIN: Yeah, I think the purpose of adding the CVCS and CFDS models on this slide is these are used to mitigate coolant injection during events, and that's not done in the design basis analyses.

MEMBER SKILLMAN: So without including the evacuation system, are you able to depend on heat transfer to the pool?

MR. MULLIN: Without the containment of -- without a vacuum?

MEMBER SKILLMAN: Yeah.

MR. MULLIN: Yeah.

MR. GALYEAN: Can I -- I mean, in the model, we assume an initial condition of what? The pressure in the containment is one PSI approximately?

MR. MULLIN: I will be less than that actually, but --

MR. GALYEAN: I know, but what did we model in the --

MR. MULLIN: That might be proprietary.

MR. GALYEAN: Oh, okay, yeah, fair enough. So, now, we did model -- you know, how do I say this?

(Simultaneous speaking.)

CO-CHAIR CORRADINI: Can I try something?

MR. GALYEAN: We had an initial condition in containment, okay.

MR. MULLIN: Yeah, you certainly don't require a vacuum in containment to have successful heat transfer.

MR. GALYEAN: Yeah, that's --

MR. MULLIN: It's just a normal operations thing to avoid heat losses, and corrosion, and stuff like that.

CO-CHAIR CORRADINI: Thank you.

MEMBER MARCH-LEUBA: If you use vacuum in the containment, you transfer more heat to the UHS, correct?

MR. MULLIN: Yeah.

MEMBER MARCH-LEUBA: So it is a conservative assumption to assume vacuum. If you assume there is steam in it or anything other than vacuum, you're transferring more heat.

MEMBER SKILLMAN: No, if you have more vacuum, the heat's in the reactor vessel. If you have less vacuum, you have degradability transferred to the pool.

CO-CHAIR CORRADINI: It's not that straightforward. I think what Mr. Mullin said is correct. They tried it both ways. There is public data from Oregon State prior to them developing their system for NuScale that's published in reports as to the effect of vacuum and no vacuum. That's a way you might want to answer that to help everybody look something up that's in the public.

MR. MULLIN: Sure, I mean, there's plenty -- the containment provides plenty of heat transfer whether you have non-condensables in there or not.

MS. BRISTOL: Okay, the next slide, for the human reliability analysis, just to confirm and clarify, there are no human actions credited in the design basis event, as design basis events, but we do have them to support beyond design basis events in the

PRA.

And so we were using methodologies previously used in the industry, both ASEP as well as SPRA-H, and we used those methodologies to develop blatant human errors as well as the recovery actions in the PRA.

CO-CHAIR CORRADINI: I'm assuming that the experts will ask a question about this, but a simple question for me is you have the EOPs, right? This goes beyond.

These are operator actions that they may take in spite of or, and if they enter into an essentially beyond design basis event, so it doesn't follow strictly the emergency operating procedures. Am I correct in that?

MS. BRISTOL: We don't have EOPs developed yet.

MR. GALYEAN: Yeah, there's operating -- there are --

CO-CHAIR CORRADINI: If I've asked it wrong, clarify for me, please.

MR. GALYEAN: Yeah, I don't think NuScale uses the term emergency operating procedures, okay.

MEMBER BLEY: We've been told they don't have any yet. That's what we've been told.

MS. BRISTOL: That's correct.

CO-CHAIR CORRADINI: Okay.

MS. BRISTOL: And so, but we have worked with operations to model these operator actions in the simulator.

CO-CHAIR CORRADINI: Oh, okay.

MS. BRISTOL: And so they have modeled both of these actually risk significant human action candidates, the operator injecting with the containment flooding and drain system, as well as the CVCS. They have modeled that and they have tested on those, you know, risk significant operator actions.

CO-CHAIR CORRADINI: All right, thank you.

MEMBER MARCH-LEUBA: But in the case of the CVCS, it's a very complex system. There are 12 of them and there's probably 50 valves on it and I don't know how many pumps.

The probability of it being misaligned when you try to make it work by a human following an event that trips all 12 reactors and you're trying to think of what is going on is fairly high.

I don't think -- I'm going back to -- my question is what do you use for your -- for input data?

MS. BRISTOL: Right.

MEMBER MARCH-LEUBA: And if you're having a bad day, that CVCS is very complex. I mean, count the number of pumps on the lineup. It's very complex.

MS. BRISTOL: And we looked at the valve lineups for these various actions we modeled the potential for spurious failures as well as, you know, pumps failing to start, the operator failing in error.

MEMBER MARCH-LEUBA: And if you put all of those conservatively, you come up with a failure probability for 120 percent.

MS. BRISTOL: That was not the failure probability we calculated, but we did model the, you know, potential failures of the injection in both of those lines.

MEMBER BLEY: I take it by latent errors, you're including things like Jose asked about valves being misaligned?

MS. BRISTOL: Correct, and so if we go to the next slide, as we quantified these events, as you'll see in the next two slides after this, both those latent errors as well as the recovery actions. We calculated them and came up with a range from about 4E minus three to 2E minus five based on the different events.

And we looked at the potential to

unisolate and flood with the CVCS and CFDS. We looked at the potential to start diesels, combustion turbine generator, as well as the operator failing to push the buttons in the control room, and so we looked at this range of different operator actions and we ended up bounding the values and did some post-processing as we evaluated our models.

So what we did was rather than assigning these small probabilities to each event that would come up in a cutset, we applied a human error probability of one in the first.

The first cutset was set to 40 minus three. Then if there was a second operator action in that same cutset, we assigned that 1.5E minus one, and then if there was a third, 0.5.

And so we were able to capture dependency that way as well as put a conservative bound on what's the most limiting conservative human action that we calculate in doing our HRA, and we just assigned that to the first event whether that was the most limiting calculated value being unisolating or flooding or whether that was an operator going out into the field and an operator pushing the button to actuate ECCS.

MEMBER MARCH-LEUBA: So for people that don't speak PRA in their sleep, you're saying that the human error probability of actually doing the right thing is only four in 1,000 trials?

MS. BRISTOL: Yes.

MEMBER MARCH-LEUBA: And this is a system that uses computer procedures on a non-safety grade computer that is programmed by a couple of graduate students working at Oregon State.

I'm just going back. I'm acting like I'm a salesman. I'm repeating my topic over and over again. I find the number very small, very small.

MR. GALYEAN: Well, in the limiting case, the operators have a minimum of 30 minutes to execute this action, and these actions have actually been tested out in our control room simulator, okay. So the operators have run through these simulations where they have created an upset condition.

In fact, they have used our sequences from the PRA and mimicked those in the control room simulator, and then they had the operators respond with executing these particular operator actions.

So this isn't something we just pulled out of the air. We've actually tested these in our control room simulator.

MEMBER BLEY: Bill, Jose mentioned the large number of valves in this distributed system.

Are all of those valves readable in the current control room design and can they all be operated from --

MR. GALYEAN: Well, of course they don't all need to be operated to --

MEMBER BLEY: I didn't ask if they need to be --

MR. GALYEAN: -- function. They're --

MEMBER BLEY: -- operated. I asked if you can see their position. That's what I meant to ask.

MS. BRISTOL: We'd have to look at the drawing to see which ones have indication. Numerous ones do have indication. Most of them fail open or fail in the way that the flow path remains open.

Outside of that, the demineralized water isolation valves are the ones that would isolate on a plain trip, you know, but the system is aligned. For CVCS, the system is aligned to the boron admission system nominally.

MEMBER MARCH-LEUBA: Excuse me, and this is where the multi-module comes into play for human factors. Did you consider the multi-module when you tested this thing? Because when it rains, it pours.

The day you would want to have an accident is when we have some unusual event that hits operators in something they did not train in the simulator. If it's an event they have trained in the simulator, they'll do it right, absolutely, not for 10 to the minus three or 10 to the minus six probability failure.

It's those events that they didn't train for and there's two operators, 12 modules. The two operators are working modules A and B, and then module F in the meantime is doing something unusual and they're not looking at it. Here is where multi-module analysis would really help. I mean, if we're going to have a failure, it would be that.

MR. GALYEAN: We did look at operator actions in our multi-module assessment, yes, and I think we'll probably get to that at some --

MS. BRISTOL: In a couple seconds.

MR. GALYEAN: In a couple seconds.

MEMBER MARCH-LEUBA: The concern --

CO-CHAIR CORRADINI: Can I just ask his question a little differently? So when you did a simulation with watching one module have to survive an upset, did you then have all of the various modules being up and being simulated such that one might be then confounded by something occurring on another? I think that's kind of what you're asking.

MEMBER MARCH-LEUBA: It's an external event that following a seismic, a hurricane is coming. It's going to be something like that, that a fix on 12 and number 1 valve is sticking, and you don't see it. That's what we worry about.

And even then, the reactor's so good that it would probably survive it. It's just, I'm concerned that the numbers that you calculated are not realistic. That you are off by several hours. Remind me to, because you didn't consider all these common cause failures. A failure of computer system to tell you to do the right thing, and there is something we brought up in a previous meeting, and I want to repeat it again.

The computer alarm system, a non-operator action, is an excellent idea. It really gives you better performance than if you don't have it. But you have to train your operators to assume it fails. To verify with different means that what the computer is telling you to do is the right thing to do.

So yes, because you see over there that module 7 is green, don't assume it's green. Go over there and check it out. It's just, we're facing death.

MS. BRISTOL: Thank you. As you had mentioned earlier here, the failures, the latent failures that may occur that we've modeled in the PRA, typically CFDS, CBCS, so the valves will misalign during, you know, as we've been discussing, in those systems prior to it being actuated, so during test and maintenance those valves were misaligned.

Misalignments within the generators, that combustion turbine generator, the diesels, or missed toleration of MPS. We model that as well, as a potential to be a human error.

In the response to the initiating events, we

have operator failing to unisolate and initiate, injection from both the CFDS or the CVCS, and CVCS can be unisolated from the control room or locally, depending on the initiating events, and then we have modeled operators failing to start the diesel or the CTG, as well as push the button for ECCS. So you see that ECCS didn't actuate, they push the button and that's an action from the control room.

Data. So, as we've been discussing, we've been using industry data from the NUREG 69.28, we've looked at LERs, common cause failures are modeled based on the NUREG 54.97 and as we've mentioned, this is generic data for a plant with no operating experience, the expectation. As well we have COL items. As information becomes available we update our data to be consistent with the information available at the time we're doing the PRA.

The expectation is that when a plant is operating, they will update as well. There's an expectation for every other refueling outage to update the data to be consistent with the current industry themes.

We've also done the design-specific analyses, the passive safety system reliability, a PSSR, as we've been discussing earlier. We did ECCS and DHRS. We looked at those, to include those failures into the model beyond just the valve failures or the condenser failures of those systems. We looked at potential TH failures that could apply.

We also looked at unique events, tube failures. We did an analysis on the tube failures and how to, in a severe accident what's the potential for an induced steam generator tube failure, so we did design specific analyses on those components as well.

And then on top of that, we did sensitivities for the data that we weren't sure about, or maybe a little, where there was more uncertainty, we did sensitivities to support those, that data that we used.

We quantified the PRA model with using SAPHIRE code. In that code we included the common cause failure modes and there was correlations in uncertainty analysis, and so we utilized that in SAPHIRE. We ended up calculating a truncation value of 1E-15 using the ASME and SPRA standard guidance that related to the convergence criterion.

So when we looked at uncertainty, we looked at both quantitative using SAPHIRE as well as we did sensitivity studies. On these event, PSSR events, MPS, we looked at different sensitivities, orders of magnitude, different calculations, to address any uncertainties within those new events, that new data, that we didn't have operating experience to support.

Quantitatively, we used the SAPHIRE code to propagate the parametric uncertainties when we calculated the mean as well as the point estimates for the CDF and LRF. Sometimes we used augmented sensitivity studies, so for the initiating event frequencies, where we did something a little bit different than what was in the industry. We evaluated what the industry data would look like with our model, and we calculated and compared that against our internal events core damage frequency and large release frequency.

CO-CHAIR DIMITRIJEVIC: Did you have any combined sensitivity studies? I mean, you only present the 1 factor. You never try to combine how that would look like if more people practice.

MS. BRISTOL: That's correct. We didn't look at numerous ---

CO-CHAIR DIMITRIJEVIC: In these studies, what did you find, both for the standard and this too?

MS. BRISTOL: So, we have a slide in a couple, and we can go over those various sensitivity studies.

CO-CHAIR DIMITRIJEVIC: Okay.

MS. BRISTOL: The various data parameters we looked at for initiating events, they weren't, again they were data from the industry, and if we weren't, we looked at various error bands on those. We typically use an error factor of 10. Due to the fact of a new design and with those initiating events from the industry effective, the new scale module, the same way.

So we captured that uncertainty in the log normal error factor, and then after the cutsets were generated in SAPHIRE, we did an uncertainty analysis using Latin Hypercube and came up with the point estimate and the means for that evaluation.

Coming out of the quantification, we were able to look at the importance as well as of the various components and systems in the PRA, and we compared these against the risk criterion that we presented in the topical report a few years ago, and so we evaluated we had --

CO-CHAIR DIMITRIJEVIC: Can you take us to the slide? Sorry I missed this, because I was concentrating on something else. When you said that SAPHIRE can build that built-in ability to perform and sets the analyses which include correlating failure probability.

MS. BRISTOL: The various components, say valves or pumps, would all have the same correlation class and so they were correlated when we did the uncertainty evaluation using SAPHIRE.

CO-CHAIR DIMITRIJEVIC: So you define correlation groups in SAPHIRE.

MS. BRISTOL: Right.

(Simultaneous speaking.)

CO-CHAIR DIMITRIJEVIC: How did you choose those correlation groups based, like, on common cause, or how did you choose what the correlation would design?

MS. BRISTOL: Yeah, the similar design. So the valves, if they were HOVs or MOVs or check valves, or motor-driven pumps --

CO-CHAIR DIMITRIJEVIC: Well, usually base correlation is showing a difference within your point estimate and your mean value. Given your point estimate, your mean value should be usually identical. That means that if you don't correlate anything in your point estimate, your mean values are the same.

So here, even the single events don't have the impact and is all common cause, so that would be the result that you are mean valuing your point estimates throughout almost the same, or I wasn't sure that you correlated. That was my question in uncertainty analyses.

So because there is not too many independent, you know, the state of knowledge, relationships with companies back when you had that sampling dating of certainty, you know, your two valves had the same uncertainty -- maybe billions of uncertainty multiplying them with that in the state of knowledge increasing your probability of failure.

But if you go through the common cause and you just have, you know, quote unquote factors like that number you don't see that. So that was one of my concerns, especially here when you're defending on very small number but identical elements. So I, you know, that was one of my concerns, this correlation type.

It will be interesting to see how that impacts. Will that be part of them and what assets have a possibility to be impacted by this.

MS. BRISTOL: Anybody else? These are the events we provided the D-RAP panel as well as just had to show that the systems that were important were in fact safety-related systems coming out of the PRA, containment isolation valves as part of the containment system, emergency core cooling system, module protection system, ultimate heat sync, these are important in the PRA as well as safety related.

The components that had the elevated risk profiles were the reactor vent valves and the reactor recirculation valves, the actuation valves of the decay heat removal system, the safety relief valves and the CVCS and CES isolation valves, those are required to open when we inject and then the combustion turbine generator was also right a Basically the Fussell-Vesely threshold.

Other events and initiators, we also look at initiating events as well as human actions. The crane was important based on our PRA, LOCA inside containment, and outside containment, loss of outside power, fires, floods, and the various hazards, we looked at all the initiating events and provided those inputs.

And as we've discussed, the human actions being the ability to inject into the vessel were both risk-important.

MEMBER BLEY: These are all in terms of contribution to core damage treatments.

MS. BRISTOL: That was a clarification that I wanted to make.

MEMBER BLEY: Did you do RAWs as well?

MS. BRISTOL: We did.

MEMBER BLEY: Do you have a slide on those?

MS. BRISTOL: The systems, we didn't explicitly say on this slide whether it was the RAW or the Fussell-Veseley. It was provided in the SR, but the systems on here would be based on RAWs, not on Fussell-Veseley. And just to clarify, the human actions aren't, it's based on level 2 and low-powering shutdown, not the internal events --

MEMBER BLEY: In the committee, the jargon is, what's achievement worth? And that means if the component fails, it's guaranteed to fail, how much increase in core damage frequency do you get, rather than the other one that's telling you what fraction of the core damage frequency is due to it.

So those systems up there were the ones that had high RAWs, if that makes sense.

MS. BRISTOL: And that the human actions are related to LRF and low-powering shutdowns, and in the low-powering shutdown model, so not the core damage frequency.

CO-CHAIR DIMITRIJEVIC: So this Fussell-Vessell is contribution to what?

MS. BRISTOL: To large-release frequency. Level 2, the operator action associated with Level 2 was in the containment flooding and drain system, operator action was part of Level 2, and then low-powering shutdown with this CVCS --

CO-CHAIR DIMITRIJEVIC: So they don't show as important if you just look in the core damage report.

MS. BRISTOL: That's right. Of a Level 1.

MEMBER BLEY: The interesting thing about the RAW contribution, it's where you would expect ECCS but that sum is, if we got the failure probability's wrong, reliability values wrong on these valves, it's a big deal It's much worse than we thought it was.

MR. BARBOUR: Say that again, Dennis, please, I want to make sure I understand that.

MEMBER BLEY: If the likelihood of failure of those systems up there is substantially higher than we thought it was going in, there would be a big change in core damage treatments.

MR. BARBOUR: If that's what we see.

MEMBER BLEY: That's what you see up there in the top line.

CO-CHAIR DIMITRIJEVIC: It goes to 2E-5, I saw that somewhere. I think it comes to the ultimate heat sync and ECCS have the same values because basically you cannot mitigate any LOCA events without that, so it basically comes to your LOCA frequency.

MEMBER MARCH-LEUBA: So it's failure to open when on demand?

CO-CHAIR DIMITRIJEVIC: If it's assumed it always fails to open, then your core damage frequency will be something ---

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: IAB, inadvertent actuation block, failure will not continue to ---

(Simultaneous speaking.)

MR. GALYEAN: Okay, the IAB is a kind of a special case ---

MR. BARBOUR: Help us with what that means.

(Simultaneous speaking.)

MR. GALYEAN: Obviously we spent a lot of time looking at the performance of the IAB in the context of the PRA and in the context of the thermal-hydraulic simulations we did to support the PRA. Of course the function of the IAB is to prevent spurious actuation of the ECCS system. And so it depends on what kind of failure mode you're talking about, right? Does the IAB fail to inhibit initial action--

MEMBER MARCH-LEUBA: I'm more concerned of the IAB remains closed, prevents it from opening. It does its job too well if it prevents it from opening, even though it needs to open. Have you considered that?

MR. GALYEAN: Yes. We do consider that in the PRA.

MEMBER MARCH-LEUBA: With the proper sequences?

(Simultaneous speaking.)

MR. GALYEAN: But I think you made your point. I think what, I want to make sure I understand, that the whole community understands your point by the RAW and the Fussell-Veseley, is you're assuming a unity failure and then looking at its jump-in probability.

MS. BRISTOL: That's right.

MEMBER BLEY: No. A change in probability of the core damage --

MR. BARBOUR: I'm sorry. I'm sorry. I said that wrong.

CO-CHAIR DIMITRIJEVIC: How do you, do you have a basic event for them?

MR. GALYEAN: For the IAB?

CO-CHAIR DIMITRIJEVIC: Yes.

MR. GALYEAN: No. Only in the, for the ECCS valve failing to operate the way it's designed.

CO-CHAIR DIMITRIJEVIC: So if ECCS valve fails to open, you have analyzed this which concludes that the block valves --

MR. GALYEAN: The IAB?

CO-CHAIR DIMITRIJEVIC: Yes.

MR. GALYEAN: Right. We did kind of an off-line piece-part analysis of the ECCS valve, which included the IAB, and then we took the result of that analysis and plugged it into the ECCS valve basic event in the PRA model.

MEMBER BLEY: Doesn't that affect the common cause failure of those valves?

MR. GALYEAN: Well, we have common cause failure of the ECCS valves.

MEMBER BLEY: Well, I mean if you did the IAB separately and just put it into the failure rate of the valve, that doesn't put its failure into the common cause contribution.

CO-CHAIR DIMITRIJEVIC: And the other factors.

MR. GALYEAN: Again, I ---

MEMBER BLEY: You didn't think it affected it.

MR. GALYEAN: Well, only in the sense that if the IAB affects the failure of the ECCS valve, then the common cause failure model in the PRA looks at common cause failure of the ECCS valves due to the same failure mechanism. I mean, the common cause failure modeling, it's a parametric model, right? It's not a mechanistic model. It doesn't look at all the different ways the --

(Simultaneous speaking.)

MEMBER BLEY: I agree with that. But there's only one IAB, right?

MR. GALYEAN: No. Each valve has its own.

(Simultaneous speaking.)

MEMBER BLEY: Okay. I forgot that.

MR. BARBOUR: I view it almost like they're together.

MS. BRISTOL: It's an internal --

MR. GALYEAN: It's embedded in the valve.

MR. BARBOUR: It's an integral part of the valve.

MEMBER BLEY: Yeah, okay.

CO-CHAIR DIMITRIJEVIC: What you say is common cause is true. But it's also true that this is completely new, so we basically don't know anything about common cause. So if you looking at failure mechanisms and conclude that those failure mechanisms are always applicable to both valves, now you can actually conclude your own common causes going to be the band, there is no way the band can fail at the same cause doesn't apply to another.

So then you, I mean that will be, because you are very sensitive to the common cause. It's not on your sensitivity standard, so in your other assumptions that, when you did the analyzing of failure modes, maybe you should include this common cause there too. What is the, why would we choose some common cause parameters, but not other?

Your -- basically fails to open for the added is 2.52E-6 common cause data, which is already a low number, right? It's a 2E million, and this is one of the most important components, right, so, it's interesting, I mean I have to go back to this passive system analyses, maybe you have there a break and if this fails to open, based on standby failure rate, or on the demand failure rate? Wherever you got that. Is it based on standby?

MS. BRISTOL: It's based on demand. So when the system demands it. And also to note --

CO-CHAIR DIMITRIJEVIC: It does include this 2ES, that's intervallic in the fails to open number. If you are using standby failure then you will have to test intervally. If you are using just the demand rate that you took from some data, based on whatever you found, something like, I guess that BWR safety valve, so something you just took the demand rate, not the standby.

MS. BRISTOL: And also in the PRA we model the potential to open at low pressures. And so, indifferent of the IAB position, the valves can open with a low DP. And so that's also captured in the PRA. That at low pressures, these valves will open indifferent of the IAB position.

CO-CHAIR DIMITRIJEVIC: And you assume it's point 1, correct?

MS. BRISTOL: Correct.

CO-CHAIR DIMITRIJEVIC: Probability to open.

MS. BRISTOL: Yes. So that's also captured, so if it opens or closes, the valves will open with the pressure between the RPVNC is low.

CO-CHAIR DIMITRIJEVIC: That's in a specific sequence. I did not find it in any cutsets or things there.

MS. BRISTOL: No, it's, I shouldn't say no, but it's in the fault tree model as a potential to actuate PCCS valves to open is the low DP.

MEMBER BLEY: I just want to make a comment. You may not like it. If you use the PRA along the way to make decisions about the design of this plant, when the PRA finds that something's kind of crucial, like the ECCS valves and they're new, seems to me a recommendation out of the PRA to the rest of the project about what kind of testing's necessary would have been very appropriate. Still would be.

And to me, that kind of test ought to include multiple valves so that we see if we have common cause problems that we don't know about yet.

MS. BRISTOL: Thank you, yes. And there's ongoing testing plans with the design group that we have been in conversation with and not at this point in the design, but as first of a kind it goes on, we have been talking to them with respect to testing potential.

MEMBER BLEY: Okay, good. And I hope you'll consider in this area, because I think your expert panel is raised, as we've raised, and you must have thought hard about it. The only way out is going to be to get enough testing to be comfortable with the results. Or let Mother Nature teach us about it later.

MEMBER MARCH-LEUBA: Before just that, what's a HEP? H-E-P?

MS. BRISTOL: Human error probability. Those are the human actions. And so what we did for various sensitivities, here's a list of some of them, we set all of the human actions to always succeed, and we looked at the impact on the core damage in the large release. As you can see, our base case PRA values for the core damage frequency, 2.70-10, large release frequency 1.7E-11, we were able to see and evaluate the significance of the various sensitivities to that base CDF and LRF results.

If we failed, if all of the operator actions failed, you could see the elevated core damage as a result as well as a large release frequency, MEMBER MARCH-LEUBA: Is that because the CDF is controlled by something else, like the module drop?

MS. BRISTOL: These are internal events --

(Simultaneous speaking.)

MS. BRISTOL: With respect to the Level 1 internal events. The module drop is not included.

CO-CHAIR DIMITRIJEVIC: Those valves, ECCS valves.

MEMBER MARCH-LEUBA: ECCS valves failing is for control, so everything you put into has nothing to do with it?

CO-CHAIR DIMITRIJEVIC: Not too much.

MEMBER MARCH-LEUBA: Is that what you're saying?

MEMBER BLEY: I don't know if you have to show these to anybody else at another time, but if you use success and failure instead of --

MS. BRISTOL: True and false.

(Simultaneous speaking.)

MEMBER BLEY: That would be helpful to most people.

MS. BRISTOL: Thank you. We then looked at common cause failures and as we discussed earlier, you can see here the common cause failures is the impact to the PRA.

CO-CHAIR DIMITRIJEVIC: Explain to us, what did you set to one, common cause events or factors?

MS. BRISTOL: The events.

CO-CHAIR DIMITRIJEVIC: So, total events, including a failure, right? For example, that anomaly, ECCS 2-3, that was set to one.

MS. BRISTOL: Correct.

MR. GALYEAN: 2 times semi.

MS. BRISTOL: Sorry. 2E-3. They were set to, yeah.

CO-CHAIR DIMITRIJEVIC: By failures you mean success.

MR. GALYEAN: The probability.

MS. BRISTOL: Yes. They were set to success or the failures were set to 2E-3. The EPRI guidance --

MEMBER MARCH-LEUBA: So you're saying the probability of the two are VV or RSV, the two ECCS valves failing is 2.10-3. The two of them at the same time is 2.10-3. In my mind, if you designed the valve wrong, it won't fail as another one is about to go, wouldn't you say? I mean, I keep repeating, I'm trying to sell cars, what is this 2.10-3 come from? And happens if it was 2.10-2?

MEMBER BLEY: You can see that pretty

easily from the common cause failures up there. The core damage frequency jumped all the way to 2D-. So E-6. It's essentially, as you make that number bigger, as it goes up to the challenge frequency.

MEMBER MARCH-LEUBA: Yes, of course, then if these valves are supposed to be very highly reliable, if they fail, it was of course because they failed. And the other one has the same course.

CO-CHAIR DIMITRIJEVIC: It might.

(Simultaneous speaking.)

MR. GALYEAN: Just to be clear, okay, just to be clear, that's not, that 2x20-3 is not a conditional probability. It is the independent probability of the set of valves failing. All failing.

MEMBER BLEY: Oh, thank you.

CO-CHAIR DIMITRIJEVIC: I just asked you but I am thinking about how do you do that in the model? You went, because you have a, you know, everything is broken on the 2, 3, 4, did you only, which ones did you set to E-3?

MS. BRISTOL: It quantified basic events in the model.

CO-CHAIR DIMITRIJEVIC: Every common cause, basic event. That's a message to 2, 3, 4, every common cause basically.

MR. GALYEAN: That's correct.

MEMBER BLEY: And so for Jose's statement, it's not just the 2 valve, any set of multiple valves that have a common cause failure are all being failed on this --

MS. BRISTOL: Pumps, diesels, all of the multiple component groups in the --

MEMBER BLEY: There is no redundancy left.

MEMBER MARCH-LEUBA: Yeah, but there was a control of ECCS valves, right? And what's the failure for, do you remember, of one ECCS valve failing?

MR. GALYEAN: I don't remember off the top of my head.

MEMBER MARCH-LEUBA: Is it 2.10-3 also? Or is it higher?

MR. GALYEAN: I think it's lower than that. I think it's, yeah.

CO-CHAIR DIMITRIJEVIC: It's 5.88E-5.

MEMBER MARCH-LEUBA: E-5? Failure of one of those valves?

MEMBER BLEY: For a fail open valve.

MR. GALYEAN: These are failsafe.

CO-CHAIR DIMITRIJEVIC: And the tree valve is 3.8D-4. This is main valve, is 5.88E-5. Tree valve is 3.8D-4.

MEMBER BLEY: In the very beginning, you said our components ought to be at least as good as what's out there in the industry, so when you found a failure rate for fail-open, spring-operated valves, was that this number or did you say that we're a little better than that?

CO-CHAIR DIMITRIJEVIC: Like BWR SOV, which you said you used, is that the number for --

MR. GALYEAN: We used the data as --

MEMBER BLEY: From the NUREG.

MR. GALYEAN: From the NUREG,

MEMBER BLEY: Okay. So that's data for all of this kind of valve that's out in the current fleet.

MS. BRISTOL: That's correct.

CO-CHAIR DIMITRIJEVIC: Yes. That's why they have 5.88 as the rate to -- (Simultaneous speaking.) We don't know anything, how did we figure out to do that --

MEMBER BLEY: We used to tell people, teach people not to do that.

MR. BARBOUR: So we're into discussion, you have one more slide, I think that then leads us into external hazards. Do you want to take that slide now, or take a break?

CO-CHAIR DIMITRIJEVIC: We can take break, because that's the Level 2 methodology.

MR. BARBOUR: It leads us back to external hazards, that's the thing. I think all they're doing is defining how they transition in this slide.

MS. BRISTOL: That's right. We can do the next slide. The Level 2 methodology, as we mentioned, we look at containment isolation failures and that's what we look at in the PRA model, SAPHIRE model, leading to large release isn't necessarily the dose at the site boundary but it's just the fact that containment is unisolated.

We don't use bridge trees in all of the end states that end in core damage in the Level 1 event trees then just transition directly to the containment event tree and those all then get directly linked.

MR. BARBOUR: Tell me to this one now, I just want to raise it, I'm not sure if you'll put many event trees up there later or not. You have a lot of cases where you have leaks from one tree to another, and some of them, like for -- Some of them are very simple. I'm not sure why in the world you did that and it's really hard to chase from one to the other. The computer can do it easily, but I have trouble finding that next link. Don't talk about it now, but if we get into some of them, tell us why you did that.

MS. BRISTOL: Okay.

MR. BARBOUR: I think this is a good time, because they're going to switch to a topic of external hazards next. So why don't we take a break until quarter-of?

(Whereupon, the above-entitled matter went off the record at 10:27 a.m. and resumed at 10:42 a.m.)

CO-CHAIR CORRADINI: Let's reassemble so we can get started. Sarah, do you want to start us off again, please.

MS. BRISTOL: I'll go ahead and we'll talk about internal events at a high level. We evaluated internal fires, floods, external floods, high winds, just consistent with the various parts of the standard, and looked at the requirements of those and evaluated those.

And we also did a seismic margin assessment and looked at Part 5 of that standard.

CO-CHAIR CORRADINI: And that's what we're going to hear about now?

MS. BRISTOL: That's the next slide, yes.

MR. GALYEAN: So, as Sarah just said, NuScale did a PRA base seismic margin assessment, or

SMA.

In the context of that, we did perform some design-specific fragility calculations. We actually employed a set of three different consulting companies to assist us in developing these design-specific fragility calculations.

Now, we had the consultants focus on those, mostly, structures that we felt would be the dominant drivers of the SMA results.

For many of the other components in the design, like valves and pumps and things, we typically use generic capacities that we obtain from the literature and then modified them using the in-structure response factors that were -- that we obtained from our seismic design folks -- or the design folks when they did their seismic analysis.

Per the Interim Staff Guidance 20, they -- the figure of merit -- or the metric, I guess I should say, for the acceptance criteria, is that the high-confidence-of-low-probability-of-failure out of the SMA should be at least 1.67 times the design basis earthquake; or, you know, to be more precise, the certified seismic design response spectra.

And in NuScale's case, that means our acceptance criteria is that our HCLPF needs to be at least 0.84g or higher peak ground acceleration.

So, the seismic margins assessment is basically layered on top of the full-power internal event PRA logic.

The fault trees -- basically, what you -- what happens is all the basic events in the PRA model now get an additional failure mode of seismic failure attached to them.

And so, what you then have, basically, is a model that includes the random, independent failures org'd with the seismic failure, okay. So, a component can fail either randomly, or it can fail by virtue of the seismic event, okay.

As I said, the result that comes out of the seismic margins assessment is the high-confidence-of-low-probability-of-failure.

And basically, it just means that your -- you have a 95 percent confidence that the probability of failure is no greater than 5 percent.

An alternate way of looking at that, though, is that you have your best estimate confidence level that the conditional failure probability is no greater than one percent, all right. So, those two definitions are essentially equivalent, okay.

MEMBER BLEY: Can I slip in a question?

MR. GALYEAN: Yeah.

MEMBER BLEY: In the staff audit, they ask you guys a question and I haven't seen the answer to it yet.

You did what, in the past, I think, was called an NRC seismic margins. There was also an EPRI seismic margins approach that did a pathways to success approach.

They ask you if both safety-related and nonsafety-related equipment was included in the model for the seismic margins.

So, I'll ask you the question, because I didn't find the answer to it.

MR. GALYEAN: Okay. I guess I'm -- yes, I mean, we included both safety-related and nonsafety-related --

MEMBER BLEY: Okay. In the margins --

MR. GALYEAN: Yes.

MEMBER BLEY: Okay.

MR. GALYEAN: Yes.

MEMBER BLEY: That's all.

MR. GALYEAN: So, the margins assessment, then, I mean, basically what we're doing in the margins assessment is determining those seismic failures that would result in a conditional core damage probability of greater than one percent.

And translated, you know, what is the postulated peak ground acceleration that would produce a conditional core damage probability of one percent?

And that's basically what the margins assessment does. It determines what is that peak ground acceleration.

Included in that assessment, we looked at structural failures. And those comprise, basically, the reactor building walls, the crane, the module supports, that type of thing, those that would touch the module directly, okay.

Next slide, please.

DR. SCHULTZ: Bill, before you leave that one, when you say "we looked at the structural fragilities" and you mentioned on the previous slide the component fragilities, is that something that the consultants have done?

Who did what here --

MR. GALYEAN: Yeah.

DR. SCHULTZ: -- in terms of the evaluation specifically of the fragilities?

MR. GALYEAN: Yeah. For the most part, the consultants looked at the structural fragilities, okay, the fragilities of the major structures.

And NuScale did the component fragilities using the generic fragilities modified by the in-structure response spectra that came out of the NuScale design -- came from the NuScale design folks.

DR. SCHULTZ: So, the evaluation of the new valves -- or the design valves, for example, that fragility was determined by NuScale?

MR. GALYEAN: Well, again, I say for the most part.

DR. SCHULTZ: That's what I want to find out.

MR. GALYEAN: I'd have to check.

DR. SCHULTZ: Okay.

MR. GALYEAN: I do believe that the ECCS valves and the DHR valves were evaluated by the consultants.

CO-CHAIR DIMITRIJEVIC: And how did you incorporate (speaking off mic) --

MR. GALYEAN: Not directly into the fault trees. Those were handled at a higher level at the event tree level, and we'll get to it.

I think in the next slide -- let's see. Yeah, this is good.

MEMBER BLEY: I want to --

MR. GALYEAN: Basically, the structural

failures -- I'm sorry, did you --

MEMBER BLEY: I'm sorry, go ahead. You're going ahead with this slide.

Back on what Steve asked you, I'm -- there must be a separate engineering report on --

MR. GALYEAN: Right.

MEMBER BLEY: -- the evaluation of those valves. That would be interesting. We'll ask the staff if they've had a chance to look at that.

The reason I ask, is because it ought not be the same people who were looking at your structures, it ought to be people who really know the guts of how the valve is put together and usually that's not the same kind of people.

MR. GALYEAN: We -- again, we had the three different consultants. And, in some cases, we gave the same component or structure to more than one consultant.

And obviously, the consultants all provided their final reports to NuScale and those were, you know, we do have those, as well as our own fragility calculation reports.

MEMBER BLEY: Okay. And those aren't things we can see, at least not --

MR. GALYEAN: If you do an audit. I mean

--

MEMBER BLEY: Okay.

MR. GALYEAN: So, the seismic response, I talked about the structural failures handled at a high level.

Basically, the assumption was made that if the -- one of these major structures failed, whether it's the crane or the reactor building walls or the bay walls or whatever, that just automatically went to core damage, okay.

There was no potential for mitigation or recovery or anything like that. It just went right to core damage.

If the plant survived the structures, then it looked at the major -- well, LOCAs, for example. And then if it survived the LOCAs, then it assumed the loss of offsite power.

We did no pre-screening of the internal event PRA results. The seismic model overlay comprised the entire, you know, logic of the full-power initiating internal events.

And we evaluated the model for 14 different ground motions ranging from 0.5g to 3-1/2 g. And the end result was that at 0.88g is where we determined that there would be the one percent chance of core damage.

And that was dominated by the structural failures, you know, the crane, the exterior walls, the bay walls and module supports.

Again, we simply assume that if failure occurred -- and, in the case of the crane, it was the seismic restraints on the trolley of the crane that hold it to the rails that rotate.

If those structural seismic restraints were to fail, we'd simply assume the crane falls on top of the modules and it goes to core damage.

In the case of the exterior walls, it was if there was a displacement crack occurred, we simply assumed the walls collapse and it goes right to core damage.

MEMBER BLEY: So, for all the structures, you assumed it went to core damage?

MR. GALYEAN: That's right. That's right.

MEMBER BLEY: Okay.

CO-CHAIR DIMITRIJEVIC: And that will apply to all modules?

MR. GALYEAN: That's right, yeah. I mean, it was just -- we didn't differentiate 1 versus 12. It just -- we figured, you know, we're just calculating the HCLPF at this point. We're not calculating core damage frequency.

MEMBER MARCH-LEUBA: Did the seismic analysis evaluate internal components?

I'm always worried about the control rods with a very long drive on top of them.

MR. GALYEAN: Yeah. In our case, of course, our control rods -- our fuel is only half height. So, the control rods, of course, are half height and --

MEMBER MARCH-LEUBA: No, I'm talking about the driver, the spaghetti that was on top of it to hold them and activate them.

If it misaligns by two degrees after the seismic event, it won't go in and you will say that it doesn't need to go that much anyway because how much is of no concern, right?

How about additional LOCAs in the CVCS lines? Because the way I see it, is you have the containment, which is a very heavy equipment, on the vessel, which is a very heavy equipment, two flanges and a two-inch pipe in between.

If anything moves in there, the -- at what "g" level do you break the CVCS lines?

MR. GALYEAN: Yeah, I can't answer that question off the top of my head. As I said --

MEMBER MARCH-LEUBA: Doesn't take much.

MR. GALYEAN: Yeah.

MEMBER BLEY: You looked at that?

MR. GALYEAN: Oh, yes.

CO-CHAIR CORRADINI: Can we get back, then, to that question if you have -- can we get some response to that if you have it?

MR. GALYEAN: Sure.

MEMBER MARCH-LEUBA: Because you have two heavy elements --

MR. GALYEAN: Right.

MEMBER MARCH-LEUBA: -- connected by a two-inch line in which --

MEMBER BLEY: I don't think you answered Jose on the reactor internals question. I know many years ago that was looked at in great detail on existing PWRs.

Did you do fragility analysis on the vessel internals and the control rod drive?

MR. GALYEAN: I can't answer. I don't recall exactly the details of that.

MEMBER BLEY: We'd like to hear if you did or not --

MR. GALYEAN: Okay.

MEMBER BLEY: -- or if you took it from

somewhere else.

I didn't ask it earlier, but you did -- I looked through the crane stuff here pretty well.

You did the crane in the Level 1, right?

MR. GALYEAN: Right.

MEMBER BLEY: When you used failure of the crane, the probability of failure of the crane, did you also look at the accouterments, the connection mechanisms, the temporary things that are brought to make connections and the operator -- the hooking up of those?

Is there an operator involved in any of the hookups on the crane lifts?

MR. GALYEAN: No.

MEMBER BLEY: Not at all?

MR. GALYEAN: Well, I think there's an auxiliary crane attached to the main bridge crane that's used for incidental things, but that doesn't really factor into, you know, what we did here.

MEMBER BLEY: Okay. But the bridge crane, there's no connections that operators have to monitor and manually hook up?

MR. GALYEAN: I -- I mean, there's a design-specific -- what we call a module lifting adaptor.

MEMBER BLEY: Uh-huh.

MR. GALYEAN: That connects the modules to the crane.

MEMBER BLEY: So, they have to put that in place?

MR. GALYEAN: Pardon me?

MEMBER BLEY: That has to be connected to the crane before the lifts?

MR. GALYEAN: No, it's --

MEMBER BLEY: A permanent part?

MR. GALYEAN: Yeah, a permanent part of the crane.

MEMBER BLEY: Okay.

MR. GALYEAN: Next slide.

This is just an illustration to show the major structures that were evaluated, the reactor building walls, the crane, the bioshield, the bay walls and the module supports.

CO-CHAIR CORRADINI: Is the bioshield strictly a radiation shield or does it have some structural integrity?

MR. GALYEAN: I believe it's just a bioshield. I don't believe there's any structural mission to it or function to it.

CO-CHAIR CORRADINI: So, if shaken or

impacted, it would just fall off. I'm asking the question -- I'm trying to understand what structural integrity is. That's what I'm interested in.

MR. GALYEAN: I mean, we did -- that was part of the fragility -- I mean --

CO-CHAIR CORRADINI: Oh, it was?

MR. GALYEAN: It was part of the fragility evaluation.

CO-CHAIR CORRADINI: So, if I enter into a seismic event that's of a sufficient magnitude, it would shake and fall into the --

MR. GALYEAN: On top of the module.

CO-CHAIR CORRADINI: On top of the module, which then would lead to --

MR. GALYEAN: Core damage.

MEMBER BLEY: Assumed core damage.

MR. GALYEAN: Right.

CO-CHAIR CORRADINI: So, let me ask the question a different way.

So, now I do that, all 12 would then undergo -- you know what I'm asking? Because if I have a sufficient seismic event, all 12 are being affected by it simultaneously.

MR. GALYEAN: Yeah.

CO-CHAIR CORRADINI: Do you understand my question?

MR. GALYEAN: That's an issue with the current state of the art in seismic analysis whether seismically induced failures like this are fully correlated, you know, or not.

CO-CHAIR CORRADINI: Oh, we address that in a different manner.

MR. GALYEAN: Yeah.

CO-CHAIR CORRADINI: Okay. I remember this.

MEMBER BLEY: Well, you know, you've got -- but you've kind of got six and six.

Aren't those the bioshields across --

MR. GALYEAN: But, again, we're not --

MEMBER BLEY: -- six of them pretty much hooked together?

MR. GALYEAN: Yeah. Again, we're not calculating core damage frequency here. We're just calculating high-confidence-of-low-probability-to-failure.

CO-CHAIR CORRADINI: So, you're just basically saying if I pass a threshold, it fails to go to core damage.

MR. GALYEAN: That's right.

CO-CHAIR CORRADINI: So -- well, okay.

All right. Thank you.

MEMBER REMPE: So, I recall seeing somewhere that there's some changes in the bioshield that are being discussed or are happening.

Could you elaborate, and did that affect this analysis or not?

MR. GALYEAN: Again, I can't comment on that, you know. All I can say is all the bioshields are independent, you know. They're not connected to each other.

MEMBER REMPE: Can you elaborate, but you can't say because you want to talk about it in the closed meeting?

What is the change to the bioshield and does it affect --

MS. BRISTOL: It's still under -- there are RAIs still out there.

MEMBER REMPE: So, you have not decided what the change is.

MS. BRISTOL: Correct. The staff has not evaluated the responses to the bioshield design.

CO-CHAIR CORRADINI: But from the standpoint to help Joy, I think it's connected to 19.2 and hydrogen distribution.

MR. GALYEAN: That's right.

MS. BRISTOL: There is no --

MEMBER REMPE: There's a different material or something or something different in the design, is what I'm asking.

MEMBER SKILLMAN: I think polyethylene is removed. Replaced with concrete.

MEMBER REMPE: Okay.

MEMBER SKILLMAN: And then I think another portion of the shield curtain had polyethylene added to it.

So, I think they were -- the material changes had to do with concrete, steel and polyethylene.

MEMBER REMPE: That wouldn't affect your analysis of the --

MR. GALYEAN: Yeah. I guess I would rather not talk about design details.

MEMBER REMPE: Okay. But maybe in the closed session or something.

CO-CHAIR DIMITRIJEVIC: But in the previous slide, something --

(Simultaneous speaking)

CO-CHAIR DIMITRIJEVIC: -- you know, when you say crane study was least --

MR. GALYEAN: I said -- okay. That's not

one of the structural failures that drives the HCLPF.

CO-CHAIR DIMITRIJEVIC: It's not?

MR. GALYEAN: It's not. It was evaluated, you know. There was a fragility calculation made, but it's not one of the drivers of the HCLPF.

CO-CHAIR DIMITRIJEVIC: Okay. Even it was assumed also that it could lead to core damage.

MR. GALYEAN: Right.

CO-CHAIR DIMITRIJEVIC: All right.

MEMBER MARCH-LEUBA: When I look at -- are you done?

CO-CHAIR DIMITRIJEVIC: Yes.

MEMBER MARCH-LEUBA: When I look at this figure, I -- this is the component that worries me and I direct you to it, see where it says "refueling machine" and there's a yellow thing?

Move your eyes to the left and there is a gray wall, which is adjacent to the refueling pool -- the spent fuel pool.

Can that wall fall on the spent fuel and has that been evaluated?

I mean, when I'm looking at this, I'm thinking refueling. That's the first thing that points to me is that wall is going to go boom on top of all your spent fuel.

And if it has not been evaluated, should it be?

MEMBER BLEY: Well, you included the wall, right?

MR. GALYEAN: The wall that separates the fuel pool from the bulk of the reactor pool.

MEMBER BLEY: Oh.

MEMBER MARCH-LEUBA: No, no, no, I'm talking --

MEMBER BLEY: He's talking about the high wall.

MEMBER BLEY: the one labeled "reactor building wall"?

MEMBER MARCH-LEUBA: This one right here.

MEMBER BLEY: Oh, that wall.

MEMBER MARCH-LEUBA: This one. I mean, the -- if -- I'm sure that's not the seismically designed wall, it's probably just masonry.

And if you don't worry about the sign that says seismic, you can send out lot of bricks on top of the spent fuel. That's what my eyes are telling me.

MEMBER BLEY: Do you have masonry walls around the inside of this here -- unreinforced masonry walls?

MR. GALYEAN: I don't believe so.

MEMBER BLEY: That's what he said. I wouldn't have thought and -- we used to have a lot of them, but we got rid of most of them.

MR. GALYEAN: You know, that's the dry dock area and I don't know how much detail there is on the dry dock.

MEMBER MARCH-LEUBA: But under sufficiently horizontal g-force, can part of the wall fall on top of the spent fuel? That's my question.

CO-CHAIR DIMITRIJEVIC: Is your concern about spent fuel pool?

MEMBER MARCH-LEUBA: Yes.

CO-CHAIR DIMITRIJEVIC: Okay. But that's not -- spent fuel pool is not part of the PRA here -- or is your concern about impact of the total pool?

MEMBER MARCH-LEUBA: If you drop that wall on top of the spent fuel pool --

CO-CHAIR DIMITRIJEVIC: Right.

MEMBER MARCH-LEUBA: -- it's going to be bad.

CO-CHAIR DIMITRIJEVIC: It's going to be bad, but this is -- I mean, the -- but we are not doing -- you know, the spent fuel pool is --

MEMBER MARCH-LEUBA: I was just asking -- I'm sure it's not seismic category 1, but this is strong enough to be seismic category 1.

CO-CHAIR CORRADINI: Okay. So, since we've raised it, let me make sure I understand.

So, was -- were all of these part of the structural analysis, or not, that he's mentioning or do you have to check on that?

MR. GALYEAN: The dry dock area was not part of the structural analysis, if that's the question.

CO-CHAIR CORRADINI: Okay. That's your question.

MEMBER BLEY: You had a full Level 1 model, you had fragilities, you had some kind of generic hazard curve and you almost had to do a seismic PRA to come up with your HCLPF following this.

Why didn't you?

MR. GALYEAN: Well, our hazard curve, of course, did not have frequencies on it, right. They were just postulated ground motions.

MEMBER BLEY: So, it was just to set some kind of bound on what you expect?

MR. GALYEAN: That's right. It was just ground motion segregated into --

MEMBER BLEY: Okay.

MR. GALYEAN: -- those 14 --

MEMBER BLEY: So, you left that to the COL people to deal with?

MR. GALYEAN: Right. Yeah. So, we did not have --

MEMBER BLEY: Okay.

MR. GALYEAN: -- a full hazard curve.

MEMBER BLEY: Okay.

MR. GALYEAN: So, the fragility calculations, you know, for the most part, you know, we used bounding conservative values.

These are just some details of, like I said, the fragility calculations that we thought that one member who's not here might be interested in.

CO-CHAIR CORRADINI: He's with us in spirit.

MR. GALYEAN: Okay. So, you know, just the factors that went into the structural response, you know, damping and modeling and its various factors that account for not only the magnitude of the motion, but the frequency of the ground vibration.

So, next slide, please.

MEMBER BLEY: When you did this, did you accumulate all the component and structural -- no, I don't think you did -- fragilities into an overall fragility curve and then apply the hazard, or did you have a plant fragility curve and a structures -- separate structures fragilities and apply the hazard to both of them and look at them independently?

MR. GALYEAN: No, it was -- we modeled the fragilities individually, if that -- in the model.

MEMBER BLEY: Even the components?

MR. GALYEAN: That's right. Even the components.

MEMBER BLEY: Okay. So, when you come up with a HCLPF, it's the highest HCLPF for the set of things you were looking at?

MR. GALYEAN: Right. I mean --

MEMBER BLEY: Because you have a different one for every --

MR. GALYEAN: Yeah. People refer to it as, like, a min/max approach.

MEMBER BLEY: Yeah. Okay.

MR. GALYEAN: You know, each action sequence comprises multiple cut sets. And so, the maximum HCLPF of a cut set -- of each cut set is combined, and then the minimum of that set is the sequence.

MEMBER BLEY: Okay. And although Pete's not here, this HCLPF is -- I remember back when people came up with it, but it's a figment of a model.

You don't believe the tail is that far out. In fact, essentially, that place where there's a HCLPF, nothing is going to fail.

It would be like a truck driving past, but it lets you do this kind of bounding analysis to get an idea of what's going on.

MR. GALYEAN: Yeah. So, again, more details on the fragility calculations and what went into the structural response. So, I'm not going to get into those.

MEMBER BLEY: Okay.

MR. GALYEAN: Low-power shutdown, you know, we did the full -- we started with the full -- a list of initiating events from the full-power internal events PRA.

Obviously, there is unique features in the NuScale design. There's no reduced inventory configuration during refueling. Everything is done underwater.

We looked at -- we did look at all the external event hazards during low-power shutdown. Obviously, the dropped module came out as the most significant core damage frequency contributor.

I think, in my opinion, that's attributable to the conservatism in the crane model.

The crane design is still being refined and so a lot of the details on things like the control system are not final.

We made a lot of conservative assumptions with respect to how the crane is operated.

MEMBER BLEY: Dick, I don't want to put you on the spot, but you've raised the issue before about whether there are or should be some kind of physical barriers to prevent a dropped cask from hitting another one.

Do you want to pursue that at all here? Because the PRA should have thought of that.

MEMBER SKILLMAN: I don't know that it's constructed to do so. I've made my concerns known.

CO-CHAIR CORRADINI: But I do think they want to -- I think where Dennis is coming from, which I think would be beneficial, is at least to go through your concern about minimizing interaction.

Isn't that -- I think that's where Dennis was going.

MEMBER SKILLMAN: Couple different things --

MEMBER BLEY: Exactly. And I thought Dick can do it better than I can.

MEMBER SKILLMAN: Yeah. If I look at

NUREG-0612, asset failures are about 15 percent. Now, this is a dated document. I acknowledge that.

My concern is a wire rope failure. It might have 12 or 15 sheathes and, under that condition, the load intensity is relative minor, but it just takes a single snap -- asset failure to drop.

I was admonished by one of your colleagues, you know, this is not too different than removing a reactor vessel head or a steam drum, and that's not exactly accurate.

It's rare that you move 750 tons. You do in construction, but you don't after live operation, and you certainly don't over a live core. And here, you're moving the whole core, so my concerns are really maybe twofold or threefold.

The reactor building liner is one-quarter of an inch throughout the entire reactor building; it is seismical in structure; it's QA-1, for good reason, it is the envelope; but it's not much thicker than the inner liner of the wheel wells in your car, if it's a plastic liner.

And that is going to see at Module No. 11-12 if there's -- I don't know if it's 1, 2, 3, 4, 5, 6, but the 2 bays that are closest to the refueling pit will see 720 module passings in 60 years of a 762-

ton machine using the 762-ton number.

My belief, based on operating experience, is the likelihood of bumping or scuffing the floor is fairly high. I think the floor ought to be armored on your key load path.

Independent from that, I think that the trolley and bridge coming out of the module bay should have some barrier to ensure that, in the transit, for any module, the module is so restricted it can only fall in one direction, and that direction has been fully analyzed, but it cannot hit adjacent modules.

It either has bumpers or has a chaffing design, but something assures that the module cannot fall in any direction that would bump into an adjacent module.

So, those are the concerns I have. The floor, the fragility of the floor. I'm told, hey, look, quarter-of-an-inch, they got them all -- all plants have quarter-of-an-inch. If you drop, all you're doing is maybe puncturing the membrane, but the concrete's really the load-bearing surface.

I understand that. Concrete is 20 feet thick, but I've also been in plants where just a nick in the floor drives the operating crew crazy trying to find a leak.

And I honestly believe that this design would be well-served by the ability for the NSSS vendor to say, we consider a module drop, but it's confined.

It can only fall in a certain geometry and we know, for that failure, whether it's banging up the decay heat removal system, whether it's chaffing the walls, the 11 other operating modules are safe.

I would say that's an argument that I think really makes sense.

MR. GALYEAN: Fair enough.

I would just clarify that there are -- the crane design has 2 wire ropes. Each is 100 percent capacity, okay.

So, a failure of a single wire rope would not result in a dropped module. Again, there are two wire ropes, you know.

Your points about nicks in the liner, I think, are, you know, reasonable. I view that as an asset protection issue that the operator would make a decision on -- the owner/operator.

We think we've done a pretty comprehensive job at identifying the potential for crane drop, for crane failure and module drop and what the potential impacts are.

Again, you know, the owner/operator certainly has a prerogative to make changes as they see fit for asset protection, and I think that's reasonable.

MEMBER BLEY: But the way you did it, as I understand it, is you came up with a frequency of drop and then assumed you'd have core damage if that happened.

MR. GALYEAN: That's correct.

MEMBER BLEY: That's the extent of the analysis.

MR. GALYEAN: For the dropped module.

MEMBER BLEY: Okay.

MR. GALYEAN: We have done other -- we have taken other looks at, you know, what the potential scenario might entail --

MEMBER BLEY: Uh-huh.

MR. GALYEAN: -- okay, but I think --

MEMBER BLEY: Somewhere I saw qualitative pictures of where it might hit and it could hit other modules and --

MEMBER SKILLMAN: Yeah. We had a presentation -- it's been three years, and I've got those in here somewhere -- and it showed three different geometries where a module could impact one or two others and either fell or slid or -- I'm kind of retaining that information even though it's a couple years old.

MEMBER BLEY: Now, just for me, I know there are other loads, but when you lift the module, how high off of the bottom is it?

MR. GALYEAN: Well, in the initial lift --

MEMBER BLEY: Yes.

MR. GALYEAN: -- okay, it's lifting up out of the operating bay.

MEMBER BLEY: So, it's all the way up.

MR. GALYEAN: And so -- and then it moves out of the -- you know, out of the operating bay and then is lowered to where it's just a foot off the floor.

MEMBER BLEY: Just a foot.

MR. GALYEAN: And so, for the majority of the travel it's moving basically a foot off of the floor.

MEMBER BLEY: Did you do any analysis given the maximum velocity of travel and being a foot off the floor, what would happen to it?

I would think it would land upright and sit there, but it might not. It sounds like something easy to analyze.

MR. GALYEAN: We did consider that and --

MEMBER BLEY: Uh-huh.

MR. GALYEAN: -- frankly, we thought it was more for the context -- in the context of the FSAR in the design certification, frankly, we thought it was more trouble than it was worth.

MEMBER BLEY: Seems like a "back of the envelope" calculation, but maybe not.

MR. GALYEAN: Yeah. It's more complicated than that.

CO-CHAIR CORRADINI: And we may not believe they're back of the envelope.

MEMBER BLEY: We may not believe they're computer code.

MEMBER BALLINGER: I think it's over 20 feet, right, the lift? Part of the --

MEMBER BLEY: Oh, I wasn't talking about the initial lift.

MEMBER BALLINGER: Oh, okay.

MEMBER BLEY: From the initial lift, yeah, lots of stuff can happen. But once -- before it starts traveling --

MR. GALYEAN: Yeah. I guess I don't want to get into too much detail on --

MEMBER SKILLMAN: This is the essence of

my concern: The thought is it's moving so slowly in its hoist distances, 12 or 14 inches, so what. And I'm just not in "so what" for a machine this big. It's 762 tons.

And it might be that one wire rope parts, the crane clamps as it should, the load settles, but, for whatever reason, it begins to dip and I'm imaging an event that I probably shouldn't be imaging.

But this, to me, is the real life and you get into these situations and you say, golly, if we had just had a bumper here or something there, that would not have happened.

The assumption is the module falls. If there is a release, it's well underwater. It's under 55 feet of water.

There is no offsite dose -- I got that -- but I will tell you the actions to recover it are going to be extraordinary.

And so, it just seems to me that if we have only one module in the entire plant and that module fails, I'm almost at "so what."

If I've got 11 modules operating at 160 megawatts thermal and I drop one because it's gone, you know, it's 24-month fuel cycle, I'm taking it to refueling, I've got at least one module problem in terms of failure and now I have 11 that are at power.

And it seems to me that that is an operating environment. I don't know if it's 1, 2, 3, 4, 5, 6 and 6A, but that is definitely an operating scheme that is different than any other plant, that I'm aware of. I just --

CO-CHAIR DIMITRIJEVIC: Yes. So, we want to concentrate on this because that is something very specific to your design, these module moves and things like that.

Especially because you have a very high CDF, you know, those 10 to minus 7, which leads to nowhere, you know, hang in there.

And there's obviously some, you know, the administrative decision was decided that that's all right. That CDF shouldn't have been counted because releases is negligible.

I see in your shutdown assumptions in your PRA in the Table 19.1-71, says that during the ABC leave (phonetic) the module is kept below the height that would damage the ultimate heat sink if dropped.

So, does that mean they assume that this assumption is hundred percent true? I mean, is there -- also, you have a lot of scenarios that leads to load of 10 to minus 14.

Is there difference between 0 and 10 to minus 14? So, why we don't count the CDF?

Is the safety goal say the core damage frequency should be below 10 to minus 4 and, you know, only for the vents which could lead to some potential release.

So, my main concern is how -- what type of failure modes do you consider for ultimate heat sink in general?

What's your failure probability to ultimate heat sink?

Is there any, or is zero?

MR. GALYEAN: There are a lot of questions in there.

CO-CHAIR DIMITRIJEVIC: Yes.

MR. GALYEAN: First of all, you know, a lot of these questions, again, in my mind, are more asset protection questions.

And the -- of course the design certification is focused on is the public health and safety -- are the public reasonably protected.

We do report the core damage frequency associated with module drop, you know, crane failure, module drop.

So, we keep it separate so as not to hide or obscure the risk insights that might be gained from the different portions of the PRA, you know. We want to make sure that we're not overlooking anything when we look at risk insights.

As far as failure of the ultimate heat sink, we have done a number of internal sensitivity studies looking at, well, if the ultimate heat sink is damaged and potentially it gets drained away, what is the safety impact on the other modules. We have done these studies.

In one case, we simply open up the hole in the bottom of the reactor pool and drain out the water and do a thermal hydraulic simulation using MELCOR to see what the impact would be on the operating modules.

I mean, we have done these analyses to satisfy our own -- to educate ourselves on what the impact would be.

These are not part of the FSAR because, again, there's no credible mechanism that we could identify that would damage the ultimate heat sink in this fashion.

CO-CHAIR CORRADINI: Could I just stop you there to make sure I understand?

Your point is that if you did the seismic margin analysis, which struck me as the way in which

I could get damage, as long as I satisfy that, it's a small enough probability.

Am I understanding it correctly?

MR. GALYEAN: No. Okay. Understand that the plant is going to be built on an engineered site, right. And so, the only -- how do I say this?

There is no credible mechanism for creating a drain in the ultimate heat sink of sufficient size that would threaten the integrity of the -- of 12 modules.

CO-CHAIR CORRADINI: And it's not determined by a seismic -- it's determined by what can drain by some sort of assumption of various sensitivity calculations?

MR. GALYEAN: That's right. Again, we just postulate a hole in the bottom of the reactor pool and drain the water out, and we look at what the impact would be on the operating modules.

CO-CHAIR CORRADINI: Okay.

MR. GALYEAN: There's no credible mechanism that would create a hole like that.

CO-CHAIR DIMITRIJEVIC: But if you have a hole, then you have a huge impact on the module.

MR. GALYEAN: Again, the hole that I'm talking about is one that would not result in core damage.

MEMBER BLEY: As long as everything is operating normally.

CO-CHAIR DIMITRIJEVIC: Yeah. As long as you don't have these LOCAs on the heated module.

MEMBER BLEY: To my question --

MR. GALYEAN: Again, we're talking about an -- again, there's no credible mechanism that would get us there.

MEMBER BLEY: And Mike's question -- we're calculating numbers like 10 to the minus 9, 10 to the minus 14, these crazy little numbers.

There's an earthquake at most sites at much higher frequency than that that would be totally devastating.

I mean, they don't hit a cutoff and you never get a bigger earthquake. Something at 10 to the minus 5, 10 to the minus 6, 10 to the minus 7 can be massive.

MR. GALYEAN: Fair enough.

MEMBER BLEY: And that's not --

MR. GALYEAN: But we're not doing a seismic PRA here.

MEMBER BLEY: And that's not incredible --

MR. GALYEAN: We're not doing a --

MEMBER BLEY: -- which you are implying.

MR. GALYEAN: -- seismic PRA here, right.

MEMBER BLEY: Is that incredible?

MR. GALYEAN: We're doing seismic margins assessment.

MEMBER BLEY: It's a lot higher frequency than what we're calculating for.

MR. GALYEAN: Fair enough. I'm not going to argue with that. I mean, who can predict the results of a peak ground acceleration of 3g, for example. No one.

I mean, we are talking about an earthquake the size that has never been experienced, to anyone's knowledge, in the history of the world, right.

I mean, you want to postulate events that have never happened. I mean, that --

MEMBER MARCH-LEUBA: That's the definition of 10 to the minus 5 and 10 to the minus 11.

MEMBER BLEY: That's right.

MEMBER MARCH-LEUBA: And one thing you can postulate is there's a hole that you didn't even know suddenly the core goes like this.

The right side of the pool drops a couple of meters from the left side of the pool. That's more likely that SVG (phonetic).

Again, probability to 10 to the minus 6, 10 to the minus 7, nothing to minus 11.

MEMBER BLEY: Go ahead. I just wanted to note there are some -- I don't know what kind of hole you get or anything like that, but there are big earthquakes that are very unlikely, but not as unlikely as some things we're actually calculating.

MR. GALYEAN: I'm sure there are lots of things -- supervolcanoes, for example. Asteroid impacts, you know, there's all kinds of things that can happen, but I don't know that --

MEMBER BLEY: Meteor strikes are more likely.

MR. GALYEAN: -- I don't know that analyzing these things adds value to what we're trying to accomplish here.

CO-CHAIR DIMITRIJEVIC: Well, okay, because we now are mixing apples and oranges. We are not going to talk about seismic event happening during 72 hours after the, you know, module drop or something like that.

So, if you ever do that, your seismic PRA, I mean, (unintelligible) although the results with the small numbers and then we are in completely different ball park.

How I started the question, is that you have here assumption the module will never reach the height which can damage the pool.

That means that you have some height in mind?

MR. GALYEAN: That's right.

CO-CHAIR DIMITRIJEVIC: So, you have some height where actually pool can be damaged.

MR. GALYEAN: That's right.

CO-CHAIR DIMITRIJEVIC: My question was how.

MR. GALYEAN: We have a design study that was performed that determined what the maximum height of a lift would be that would damage the pool integrity.

CO-CHAIR DIMITRIJEVIC: So, what does that mean, "damage pool"?

Does that mean in making ultimate heat sink --

MR. GALYEAN: No. It means causing a crack in the concrete.

CO-CHAIR DIMITRIJEVIC: Well, does ultimate heat sink not fail to perform its functions, right; is what you are saying?

MR. GALYEAN: That's correct -- well,

again, in the context of the PRA, the -- you know, the size of the holes -- or the draining of the pool that we have looked at doesn't compare -- you know, is far more catastrophic than what would happen if the pool concrete was damaged.

I mean, we postulate just an open hole that goes out into a vacuum. No back pressure, anything, just open up a hole, a drain in the bottom of the pool and just drain out the water.

CO-CHAIR DIMITRIJEVIC: So, in your mind, it's okay to have here -- just because you claim the latch release probability given this is zero, it's okay to exclude these sort of event but then we have a 10 to minus 14 for extent of flood we are going to include that CDF.

MR. GALYEAN: I guess I don't -- you know, I take exception to "excluded." I mean, we do report it. We report --

CO-CHAIR DIMITRIJEVIC: Yea, but it's not counted in CDF.

MR. GALYEAN: We do report core damage frequency.

CO-CHAIR DIMITRIJEVIC: Yes, but your total core damage frequency doesn't include this core damage frequency.

MR. GALYEAN: We don't report a total core damage frequency anywhere. We just report core damage frequency for each individual hazard.

CO-CHAIR DIMITRIJEVIC: Okay. So -- okay. Let's go back.

You mean you don't report the core damage frequency for low-power shutdown or --

MR. GALYEAN: Yes, we do. We report each individual hazard --

MEMBER BLEY: But you don't aggregate

MR. GALYEAN: We don't aggregate.

MEMBER BLEY: We can aggregate.

MR. GALYEAN: You can aggregate.

CO-CHAIR DIMITRIJEVIC: Wait a second. You have this really significant report where you use your CDF to make argument about significant determinations.

MR. GALYEAN: And we evaluate every hazard individually in the context of that risk significance determination.

And that's why we have some that show up coming out of the low-power shutdown, we have some coming out of the external hazard, you know, as risk significant.

If a component or system or operator

action shows up as risk significant in any of those evaluations, then it's added to the list of risk significance.

CO-CHAIR CORRADINI: So, in my mind, one could think of a table that has various categories of internal events, low-power shutdown, module drop, whatever, and then, on here, all the systems and where they appear as risk significant or not.

MR. GALYEAN: That's correct.

CO-CHAIR CORRADINI: That's what you're saying?

MR. GALYEAN: That's correct.

CO-CHAIR CORRADINI: Does such a table exist?

MS. BRISTOL: Those are in the FSAR. So, there's one for Level 1, there's one for --

CO-CHAIR CORRADINI: Oh, they're individual, but no --

MS. BRISTOL: -- Level 2 and then there's external events.

CO-CHAIR CORRADINI: Okay.

MS. BRISTOL: And so, all of those -- and then at the end of the set of tables there's also a table of all the different hazards, full power, low power, and then --

CO-CHAIR DIMITRIJEVIC: But you have it here some importance. We just look in the one slide, the importances, right?

MS. BRISTOL: Right.

CO-CHAIR DIMITRIJEVIC: And that --

MS. BRISTOL: And we took those from all of the various tables.

CO-CHAIR DIMITRIJEVIC: And so, the ladder dropping event would be there, right?

MS. BRISTOL: So, the crane is on there. So, the --

CO-CHAIR DIMITRIJEVIC: Okay. Crane is --

MS. BRISTOL: The reactor building crane, sorry, RBC, under the other events is greater than, you know -- so, that one is captured as --

(Simultaneous speaking)

MS. BRISTOL: Yeah, since it's evaluated a little differently than just, say, a ECCS valve.

CO-CHAIR DIMITRIJEVIC: Okay. All right.

MS. BRISTOL: So, that is captured on there.

CO-CHAIR DIMITRIJEVIC: All right. Because I saw that in some tables that you presented in our previous meetings, you said, well, logically they are not applicable and we should maybe say "negligible," like, for others, you know, things like that instead of "not applicable," because it implies is not possible, which is not correct, right?

So, okay. All right. So, I will -- with what you said in mind. I have to go look back in the sum of the conclusions because this initiating event of that -- of the dropping module consist of the multiple -- it probably has some human actions inside and things like that, right?

MS. BRISTOL: The dropped module looks at, yeah, numerous potential failures for the crane.

CO-CHAIR DIMITRIJEVIC: Right. But those are not -- like, those human actions are not part of the human actions importances and things like that?

MS. BRISTOL: That's correct. We looked at the crane as somewhat of a supercomponent and any potential failure of the crane we just designated the crane as a risk-significant component candidate.

CO-CHAIR DIMITRIJEVIC: Okay.

MEMBER BLEY: Okay. And, actually, I guess your last table, 19.1-80 --

MS. BRISTOL: Correct. That's the one --

MEMBER BLEY: -- it might not sum them up, but it has --

MS. BRISTOL: Yes.

MEMBER BLEY: -- all the contributions on the one table. So, you can see them there.

MR. GALYEAN: Right.

MEMBER BLEY: I forgot what else I wanted -- what I really wanted to ask you, so go ahead. I'll remember in a minute.

MR. GALYEAN: Okay. So, the dropped module, we assumed that if the crane failed and a module went horizontal, okay -- well, part of the core would uncover and that it would then go to core damage.

And then we did -- we simply assumed that the containment would fail by virtue of the impact with the pool floor. And we did a -- evaluated the radiological release that came out of it and it was small. Okay.

MEMBER BLEY: You showed us that analysis --

MR. GALYEAN: Right.

MEMBER BLEY: -- and I don't remember just when --

MR. GALYEAN: This is the one exception to the simplification we made in the full-power internal events where we simply said if there was a bypass or a LOCA outside containment, that automatically resulted in a large release.

This is the one scenario -- one sequence or scenario that we explicitly evaluated the release and the potential for dose at the site boundary to compare to our definition of a large release.

And in this particular case, it was much smaller than a large release.

MEMBER MARCH-LEUBA: So, without going into too much details, you're assuming that the module drops --

MR. GALYEAN: Right.

MEMBER MARCH-LEUBA: -- and then, by some miracle, a gas forms on half of it or 10 percent of it, and then the fuel that is uncovered would then melt.

MR. GALYEAN: Right.

MEMBER MARCH-LEUBA: Why does the gas -- because you're moving it completely filled.

MR. GALYEAN: No. The module is not completely filled when --

MEMBER MARCH-LEUBA: Why not?

MR. GALYEAN: -- moved.

MEMBER MARCH-LEUBA: If it's your primary -- your primary contributor to CDF, why not require to move it full and cold? Why not?

MR. GALYEAN: Well, there's a couple of reasons. One, is it adds weight to the module for the crane -- that the crane has to lift.

And for another, there is equipment at the top of the module inside the containment that the designers don't want to be submerged.

MEMBER MARCH-LEUBA: If it tips, it's going to be submerged.

MR. GALYEAN: Well, certainly, but that's not the plan.

MEMBER BLEY: Bill, you chastised me about asking you about earthquakes bigger than have been recorded in the history of the world.

In your set of hazards, what's the biggest one you have?

MR. GALYEAN: In the HCLPF evaluation, we went up to 3.5.

MEMBER BLEY: 3.5g. You did go up to 3.5.

MR. GALYEAN: That's right.

MEMBER BLEY: Okay. Thank you.

DR. SCHULTZ: Bill, on this slide, a radiological release evaluation stops with the third bullet -- in other words, the fourth bullet -- where you could potentially induce LOCAs or transients. You didn't go further and say the consequence is there.

MR. GALYEAN: Yeah. We just looked at the potential for a crane failure and a module drop to impact an operating module.

DR. SCHULTZ: Right.

MR. GALYEAN: So, we then looked at the likelihood -- or not the likelihood, but the probability of inducing an upset condition in an operating module by virtue of crane failure or module drop.

DR. SCHULTZ: Right.

MR. GALYEAN: That then has a, you know, potential for inducing an upset condition in an operating module.

And then we took that likelihood and compared it to the likelihood of those upset conditions already modeled in the PRA, in the full-power internal events PRA, okay.

And those -- and the crane failure-induced contribution was orders of magnitude lower than the independent likelihood of those events already modeled in the full-power internal events PRA.

CO-CHAIR CORRADINI: So, your point is they're bounded?

MR. GALYEAN: That's right.

CO-CHAIR DIMITRIJEVIC: You consider this to be just LOCA or LOCA with failure of containment insulation?

MR. GALYEAN: LOCA.

CO-CHAIR DIMITRIJEVIC: Just LOCA?

MR. GALYEAN: That's right.

CO-CHAIR DIMITRIJEVIC: So, there is no --

MR. GALYEAN: A LOCA outside containment.

CO-CHAIR DIMITRIJEVIC: LOCA outside of containment, which is the same bypass of containment.

MR. GALYEAN: Right.

CO-CHAIR DIMITRIJEVIC: And so, since the frequency of that is 1E minus seven, is much less than your other LOCA outside the containment --

MR. GALYEAN: That's correct.

CO-CHAIR DIMITRIJEVIC: -- which was --

MR. GALYEAN: That's correct.

CO-CHAIR DIMITRIJEVIC: I forgot what it was, but I can look in there. So, basically they consider that what happen is LOCA outside containment. And then that component is internal LOCA outside containment.

MR. GALYEAN: Yeah.

CO-CHAIR CORRADINI: And later, we'll hear about that from -- you had a grouping of these that you analyzed. Later, meaning tomorrow.

MR. GALYEAN: Okay. That's right. Next slide, please.

Now, as the -- now, as part of our multi-module evaluation, our multi-module evaluation comprised two parts.

We did a qualitative evaluation where we looked at all the systems in a NuScale plant and basically did a hazards assessment.

If that system failed, what would be the impact on the operating modules? And I said that's a qualitative evaluation. We looked at that and documented that.

We also did a quantitative evaluation. We started with the results of the full-power internal event assessment, PRA, and we -- and for each basic event in the full-power internal events PRA model, for each basic event, we applied what we called a multi-module adjustment factor, okay, where we said, if this event happens in one module, what's the conditional probability that it could occur -- simultaneously appear in another module, in two or more modules?

And that's what we refer to as these multi-module adjustment factors. These are conditional probabilities given that the failure occurs in one module, what's the conditional probability that it could also occur in another module.

MEMBER BLEY: And these are judgment based.

MR. GALYEAN: These are engineering -- but they are very -- in my opinion, I mean, they are extremely conservative in the sense that every single basic event in the full-power initiating event model is accounted for with at least a one percent probability.

So, think of, for example, a pipe break. Okay. Even if you have a -- or a LOCA inside containment, you know, just a relief valve opening or, you know, something, a CVCS pipe breaks inside containment.

We say, given it occurred in the first module, there's a one percent chance it could simultaneously occur in a second module, okay, just to see what the impact would be.

And so, using a minimum value of one percent and a maximum value of a hundred percent, we applied these adjustment factors to every basic event in the PRA.

CO-CHAIR CORRADINI: Can you help me with the second one, the 0.1 to 0.3 one? I don't think I understand where you -- how you assigned those.

MR. GALYEAN: These -- okay. These are events that -- you can think of the one percent conditional probability as applying to events you would, on the surface of it, think of it as completely independent. Okay.

And then there's another class of events that you could look at and say they are completely dependent.

For example, a loss of offsite power. It affects all 12 modules, right? That would be a hundred percent.

Then there's an intermediate class of events, maybe operator actions, maybe thermal hydraulic conditions that maybe there's a common cause failure in one module.

And given that a common cause failure occurs in one module, there's a chance that that same common cause failure could occur in a second module.

And that's the purpose of these intermediate values of 10 percent and 30 percent, okay, is to account for these things that maybe have some dependency, okay, but less than complete dependency, but more than completely independent, okay.

And so, there, we went and applied these values to those basic events.

CO-CHAIR DIMITRIJEVIC: Well, we had this discussion last time when we had -- and there was some issue which I had with some of those things.

Let's discuss shared system. Give me some good example of shared system like a --

MS. BRISTOL: Diesel.

MR. GALYEAN: Yeah. Instrument air, service water, you know, electric power.

CO-CHAIR DIMITRIJEVIC: How about the important one like this --

MEMBER SKILLMAN: Nuke service. Nuke service is the important one.

CO-CHAIR DIMITRIJEVIC: Which one?

MEMBER SKILLMAN: Nuclear service cooling water.

MR. GALYEAN: We call it circulating water.

MEMBER SKILLMAN: Circulating water.

MR. GALYEAN: Circulating water, but --

CO-CHAIR DIMITRIJEVIC: But from the PRA --

MS. BRISTOL: Right.

CO-CHAIR DIMITRIJEVIC: -- but something which has a high importance in the PRA, but it's -- is this combustion tubing common?

MR. GALYEAN: Yes.

MS. BRISTOL: Yes.

CO-CHAIR DIMITRIJEVIC: What does that mean?

There's only one?

MS. BRISTOL: Correct.

MR. GALYEAN: That's correct.

CO-CHAIR DIMITRIJEVIC: All right. So, this is what I want to tell you is my problem. You have, let's say, 10 units -- 12 or whatever -- and you have one turbine, right? You have loss of offsite power to all the units.

You can only use it for one, right?

MR. GALYEAN: I'm sorry?

CO-CHAIR DIMITRIJEVIC: You can only use that to backup the power to one module.

MR. GALYEAN: No. No.

CO-CHAIR DIMITRIJEVIC: You can use it to all modules?

MR. GALYEAN: Yes. It is sized to supply the entire site.

CO-CHAIR DIMITRIJEVIC: Okay.

MR. GALYEAN: And same with the diesel generators.

CO-CHAIR DIMITRIJEVIC: Okay. So, you can supply all the modules with that and you have operator actions to do that, right?

Operator action to do this -- whether it supplies one unit or supply 12 unit, it has to have different probability, right?

MS. BRISTOL: It's to start the generator indifferent of how many units that that generator is supporting.

So, the action modeled in the PRA is to start the CTG, combustion turbine generator.

CO-CHAIR DIMITRIJEVIC: And he doesn't have anything to do module-specific just as the diesel generator, the load, everything goes automatically on the different modules?

MS. BRISTOL: That's how the design is currently modeled.

CO-CHAIR DIMITRIJEVIC: Okay. Well, then my concern is mostly about these human actions because I think that human actions will be different in the multi-modules that affect the various uses when the single modules ---

MR. GALYEAN: Yeah. In this particular

context here on the multi-module evaluation, which is not on this slide, is the human actions were multiplied by a factor of ten to increase the failure probability. A factor of ten.

CO-CHAIR DIMITRIJEVIC: For second unit, because you actually calculate --

MR. GALYEAN: Well, for --

MS. BRISTOL: Any greater than 1. 2 to 12.

CO-CHAIR DIMITRIJEVIC: Okay. Well, that's why we are calculating CDF based on the one and not based on two. That's my question.

Why is -- because you are calculating two, but you are multiplying human actions only for -- other than one, and you are using one as your base CDF.

Why you are not using two as your base CDF? You understand which I'm asking? Because you're going to have one unit where the human action will be 40 minus 3. And you're going to have 11 units where human action is going to be 80 minus 3, but you are using 40 minus 3 as your base case.

MR. GALYEAN: We -- in essence, we assumed complete dependence on human action. We said, you know, if the human action is performed, it's performed for everything or nothing. And we take that base human error probability and multiply it by ten.

So, what you saw before, the 4 times 10 to the minus 3, now becomes a 4 times 10 to the minus 2 and --

CO-CHAIR DIMITRIJEVIC: And I think you should use that as base case. That's my main comment. Because what's happening, you are --

MR. GALYEAN: In this context of the multi-module evaluation, that is the base.

CO-CHAIR DIMITRIJEVIC: No, no, because you are calculating probability to fail 2 or more out of 12. That's what you are calculating.

MR. GALYEAN: That's right.

CO-CHAIR DIMITRIJEVIC: But you are not taking in account where you're calculating 1 for your base calculation there, you are calculating 1 of 12. Not just 1, 1 of 12.

So, therefore, you should choose that 1 of 12 should be representing the worst situation for things like that.

Let's say the human action -- this is why I was trying to explain this. Let's say that you have a -- it's much better to understand what I mean. Let's say that you have something which can only supply six units and not more than six units.

Therefore, you know, you can put it to one for the second unit, but then for the first you are crediting it, and that first thing shouldn't be your base case.

Your base case should be the one which is problematic.

MR. GALYEAN: In this particular -- in the context of the multi-module evaluation, we did what you suggest.

We took the first module human error probability, multiplied it by ten, okay, and then assumed all the others were probability of one, okay. They were completely correlated, okay, so --

CO-CHAIR DIMITRIJEVIC: But you don't apply that in your base case. We are discussing all the time base case, yes.

MR. GALYEAN: In the multi-module evaluation, that's -- it is the probability for the first module, okay.

CO-CHAIR DIMITRIJEVIC: Yeah.

MR. GALYEAN: That probability is multiplied by ten.

CO-CHAIR DIMITRIJEVIC: So, why don't you use that as a base case? That's my question.

MR. GALYEAN: We talked about the human reliability assessment and we picked the value that we did, you know.

It was the limiting value of all the values we calculated. I mean, we have a basis for why we picked it. I don't know what else to say.

CO-CHAIR DIMITRIJEVIC: Okay. I will think about that and present that in a letter so you better understand.

What I have a problem with this multi-modules is when you talk multi-modules, you say 1 -- 2 or more -- 2 or more, but you don't keep in account that you're calculating 1 out of 12 as a base case, even you're just doing one module.

That module cannot be done as the -- in some -- the one thing which sits there independent of everything else, right? That module is part of unity.

So, you are calculating 1 out of 12, right, or you, in your mind, you're just doing 1? That's a very different question.

Are you calculating 1 out of 12 or you are just calculating 1? You understand the differences?

MR. GALYEAN: I think I do. But I think there's still a misunderstanding going on about what we do.

CO-CHAIR DIMITRIJEVIC: Okay. So, what I want to say, let's say the situation, like the loss of, loss of power that he has to do multiple actions on the multiple modules. Obviously that's a much more complex action then if you had one module which losses also power, and he's only concentrate on that.

MR. GALYEAN: Agreed.

CO-CHAIR DIMITRIJEVIC: So, that 4E minus 3 represents that. That he's dealing with multiple modules doing all of those steps.

You cannot represent --

MR. GALYEAN: That's not what we did.

CO-CHAIR DIMITRIJEVIC: I know. But why not?

MR. GALYEAN: Because we -- because we believe that the complication imposed by having a multi-module event would disrupt the response to the first module.

So we increased the failure probability of the first module by a factor of 10. And then we simply just --

CO-CHAIR DIMITRIJEVIC: To the 4E minus 3, right?

MR. GALYEAN: That became 4E minus 2.

CO-CHAIR DIMITRIJEVIC: Two, right.

MR. GALYEAN: For the first module. And then we assumed that all the related actions were completely dependent.

That if he failed the first one, he would fail them all.

CO-CHAIR DIMITRIJEVIC: Buy why don't you do this in your basic study? Because in the basic study that always happens.

There is no loss of offsite power where he had to deal with only one module. He always deals with multiples.

So why don't you use this 40E minus 2 in your basic study?

MR. GALYEAN: Because in basic --

CO-CHAIR DIMITRIJEVIC: For those situations where he has to deal with the multi-module problem.

MR. GALYEAN: In the basic study we're assuming there is a plant upset, how do I say, a complication with only one module. Okay?

I don't know, yeah.

CO-CHAIR DIMITRIJEVIC: But that's a --

MEMBER BLEY: But then loss of offsite power is not an appropriate initiator then. Because that can't happen, unless you're only running one module.

CO-CHAIR DIMITRIJEVIC: Or a million others cannot happen.

MEMBER BLEY: Yeah.

CO-CHAIR DIMITRIJEVIC: Extended flood, fire.

MEMBER BLEY: Yeah. All of those.

CO-CHAIR DIMITRIJEVIC: Yeah.

MR. GALYEAN: Well, obviously a loss of offsite power by itself is not a concern. Right? I mean, you have to have something else going on.

Other failures going on for it to show up in as a safety hazard.

MEMBER BLEY: But it still means you're, where you're normally really focused on one or just the overview of all the modules, now you have a reactor trip on one, but on 11 others as well.

So it puts your three operators in a different mode then you're analyzing them for all the really internal events that are separate that are independent events, so.

MR. GALYEAN: I understand.

CO-CHAIR DIMITRIJEVIC: That's exactly. Because you did your base case as you just have -- and maybe that's okay, because Joy always brings up how are you going to be loading modules?

Maybe somebody will just decide to have the one module. But your base, this is just for somebody who decides to just have one module. Other then this drop.

MR. GALYEAN: Effectively, yeah. That's true.

CO-CHAIR DIMITRIJEVIC: Yeah. Yeah.

MEMBER BLEY: Human actions have come up here. So I'm going to ask a question I didn't ask earlier.

Back when we did the human engineering chapter, it -- or the referenced technical reports to it, talked a lot about how we considered there, errors of commission and errors or omission.

And it says over and over again we thought about errors of commission. And then when you get to the list of events they have, there aren't any.

Did you guys look at errors of commission?

MR. GALYEAN: We looked for them. The only place that we actually --

MEMBER BLEY: I don't think in the report you tell us much about how you look for them, do you? I might have missed it.

MS. BRISTOL: They were looked at. And it was described in the actual supporting technical theory document in Journal Three.

MEMBER BLEY: Okay. But not a technical report that we have access to?

MS. BRISTOL: Correct.

MEMBER BLEY: So there's a lot of this stuff we -- you got -- can we take a minute and do that?

Tell us how you looked for them.

MS. BRISTOL: We looked at --

MEMBER BLEY: I know lots of ways to look for them.

MS. BRISTOL: We looked at the various events that operator -- the model PRA human actions. And if they were performed at a different time, would they put the plant in an upset condition?

For instance, if they operated CFDS at power, where would the impact on the module be? Or any of the actions that they were trained to perform, if they performed them in an inappropriate time, would that contribute to a plant upset?

And how would the module respond?

MEMBER BLEY: And you did this qualitatively? You built a list and said, we're not going to look at these because they don't affect it.

Are there errors of commission in the PRA?

MR. GALYEAN: Only in the way we modeled the crane, the reactor building crane.

MEMBER BLEY: Okay. And kind of sort of in the way you did this multi-module evaluation. There's some effective allowance for them, I think.

When you throw these factors on the errors, you could make that argument. Another way to look at them is for each event tree, each sequence model, it's a given I have this.

Are there any things people could do for a variety of reasons that could change the likelihood of any of my top events? Did you do something like that?

MS. BRISTOL: Qualitatively that is what we did.

MEMBER BLEY: Okay. That is what you did.

MS. BRISTOL: We looked at those, and -- yes.

MEMBER BLEY: Okay. And the kind of things people can do, or as you said, they're the kind of things they have in their procedures, but it's not in there.

There's also the kind of things, and you don't have a history here, but there will be some kind of informal knowledge about how things work that builds up over time and gives the operators a rationale to do things that aren't called for. Because they don't omit or commit. They respond to the best of their ability.

I don't know. I got a feeling you didn't really look for that kind of thing. What could lead people to do things you weren't expecting them to do?

Turn something off or turn something on. Or cross connect something. It happens. Anyway. It's a thing that we'll look for some more.

CO-CHAIR DIMITRIJEVIC: Well, like a controlling crane speed could be one of the good examples, you know.

MEMBER BLEY: If it's done manually, or can be done manually. Yeah.

CO-CHAIR DIMITRIJEVIC: Yeah. Yeah.

MEMBER BLEY: That was all. You gave me a sense.

MS. BRISTOL: Overall, you know, we discussed in sites the -- as we've continued to discuss the NuScale design exceeds the core damage frequency safety goal with a significant margin. Being an internal event CDF of 3 minus 10.

CO-CHAIR CORRADINI: So, I know you want

to get to the end, and I'm sorry. But, I was listening to Vesna trying to express her worry. And your answer to her relative to these ones where it connects across it.

If you were to do what she's suggesting in terms of re-base lining a human action for one, as a community of 12, how much would that change that?

Or is your sensitivity where you took in multiplying by a factor of 10 bounding that? Am I making sense?

MR GALYEAN: Somewhat.

CO-CHAIR CORRADINI: I'm trying to get a feeling for what her worry is. Because what I heard from her question and your answer was, she would have done it a different way.

And your answer is, well, you know, we looked at it, and I'll misrepresent, as one individually. But when they were connected for multi-module, you did effectively do kind of a worst case sensitivity.

Does that worst case sensitivity bound these insights?

MR. GALYEAN: Okay. I --

CO-CHAIR CORRADINI: I was kind of understanding --

MR. GALYEAN: I just first, I want to try and make clear what we're talking about. And remember for the single module, full power internal events' PRA, --

CO-CHAIR CORRADINI: Yeah.

MR. GALYEAN: Okay. We did do a sensitivity study where we set all human actions to always fail. Okay.

CO-CHAIR CORRADINI: that was in the summary we got back a year ago.

MR. GALYEAN: And earlier just this morning.

CO-CHAIR CORRADINI: Right. And early. Right. Okay.

MR. GALYEAN: I mean, we did that sensitivity study, and we showed you the results in the slides.

MEMBER BLEY: But that's kind of an extreme application.

CO-CHAIR CORRADINI: Of her question.

CO-CHAIR DIMITRIJEVIC: Yeah. 3.3 minus 8, I think was. Is there any shared system which doesn't have a capacity to supply all modules?

MEMBER MARCH-LEUBA: Yes. Boron addition system. And you can only boron one module while supplying a little bit to the others.

I'm sure there are more. But this is non-safety grade.

MS. BRISTOL: The containment flooding and drain system, there's two subsystems. One operates six modules.

There's two systems. One --

CO-CHAIR DIMITRIJEVIC: And each operates each one?

MS. BRISTOL: Correct. Central.

CO-CHAIR DIMITRIJEVIC: Yeah. Because that will also, we will see in the impact of this one that is 12. You know, also in this shared system.

Because we cannot have an initiator which creates the, you know, you -- we have a new initiator which will call for the, you know boration. But that's not common to the unit, yeah?

MEMBER MARCH-LEUBA: I don't think it's even is included in the PRA at all.

CO-CHAIR DIMITRIJEVIC: Not for the action even?

MEMBER MARCH-LEUBA: I don't know. You tell me, is BAS on the PRA?

MS. BRISTOL: Boron addition is not modeled explicitly in the PRA at all.

MEMBER MARCH-LEUBA: Yeah. Because it's not needed for any event.

MR. GALYEAN: Right.

CO-CHAIR DIMITRIJEVIC: Oh, okay. All right.

MS. BRISTOL: And apparently Table 19.1, Tech 76 shows all the shared system broke down by common modules each time.

CO-CHAIR DIMITRIJEVIC: In the 76?

MS. BRISTOL: In the 76.

CO-CHAIR DIMITRIJEVIC: Okay.

CO-CHAIR CORRADINI: Page 278.

MEMBER REMPE: So I didn't hear the final part of your answer, Bill, to Mike's question about how important would this be. I know you said you did the sensitivity for a single unit where you assumed all the human actions failed.

But, what's the bottom line to the whole question for the multi-modules?

MS. BRISTOL: Well, again we did not do sensitivity studies on the multi-module quantification. Okay.

Again, what we did in the multi-module quantification was, we took the base case human error probability and multiplied it by a factor of 10 to account for the complication of the operators having to deal with multiple modules.

I mean, that was -- that was the multi-module adjustment factor that we did. And so I don't --

CO-CHAIR CORRADINI: Let me ask you, I think I understand in terms of bounded by sensitivities calculation. The difference toward a different distinction typically --

MR. GALYEAN: Right.

CO-CHAIR CORRADINI: Sorry, that you did it in a different manner to bound it.

CO-CHAIR DIMITRIJEVIC: But they have to reach agreement with the reviewers. In the beginning when you started these that that reviewer will do it that way, right?

That you were just looking in one module like it's independent from that.

MR. GALYEAN: That's right.

CO-CHAIR DIMITRIJEVIC: And that's acceptable.

MEMBER BLEY: And the thing that Vesna brought up, it's very legitimate. It is four these multi-unit initiating events of which there is a handful, the way it's modeled is not the way it's carried out.

Because you don't have a single unit operation. Yeah, you always have more than one here. And how big an effect that is, that's something that has to be thought about carefully.

But it shouldn't be any worse than the sensitivity study they ran.

CO-CHAIR CORRADINI: That's what I was saying.

CO-CHAIR DIMITRIJEVIC: When it comes to human error.

MEMBER BLEY: Right.

MS. BRISTOL: And so the other various analysis in the associated core damage frequencies is we discussed the low power and shut down was dominated by the conservatively performed module drop event.

We did a focused PRA where we only credited the safety-related components in the PRA to support D-RAP. And that was below the threshold for those criteria.

And the multi-module CDF factor, as we discussed, was .13. And so that was -- while we didn't do, you know, a full multi-module, that was the impact of applying those multi-module adjustment factors to the core damage frequency.

MEMBER BLEY: So, lest we get overconfident here, we haven't done a level three PRA, which is where the multi-module effects would show up. And you don't know if this is a bound on --

MR. GALYEAN: Okay.

MEMBER BLEY: That or not, until it gets looked at. And it will have to be looked at, at some point.

MR. GALYEAN: What will?

MEMBER BLEY: Level three PRA.

MR. GALYEAN: Well, I mean, we have done --

MEMBER BLEY: Before fuel load.

MR. GALYEAN: We have done --

MEMBER BLEY: Not for you guys, before fuel load.

MR. GALYEAN: We have done dose calculations for the site boundary.

MEMBER BLEY: Okay.

MR. GALYEAN: Okay. And for example the dropped module, you know, that was a dose calculation.

MEMBER BLEY: Um-hum.

MR. GALYEAN: We've also done dose calculations to support the environmental report. Okay.

And you know, all I can say is that even if you took those dose calculations and multiplied them by a factor of 20, we're still way below the large release definition.

MS. BRISTOL: From the level 2 perspective, we looked at the large release frequency from the various hazards. And we were well below the safety goal of the large release frequency.

As we've discussed, module drop is blow -- is underwater. Didn't impact the large release frequency.

And we did the focused PRA for LRF as well. And was below the criteria for that for RTNSS purposes. And evaluated the multi-module factor.

As requested, we tried to include these insights at a high level from the FSAR. And more so if there was anything of interest or to discuss.

But, you know, our design is a very passively safe design. And there are a lot of components and elements of that design that contributes to these low values that we're seeing for core damage frequency and large release frequency.

And that give us confidence that the numbers, you know, while they are values, we do have supporting design features that support those low values.

For instance, you know, as we've mentioned previously, failure to scram events don't directly lead to core damage. You know, they progress similar to trip events.

MEMBER MARCH-LEUBA: And do you have a calculation to support that statement?

MS. BRISTOL: We do, yes.

MEMBER MARCH-LEUBA: Has anybody seen it? Have you seen it? Because I've been asking for it for a long time.

And the first time I'm seeing something from the staff, which I'll drill this afternoon, which I got this morning.

MS. BRISTOL: Okay.

MEMBER MARCH-LEUBA: So --

MS. BRISTOL: So we can add that.

MEMBER MARCH-LEUBA: I find it hard to believe that that was a bounding calculation.

MS. BRISTOL: Understood.

MEMBER MARCH-LEUBA: And stay tuned for this afternoon.

MS. BRISTOL: Okay. And so as we note, the cycling reactor safety valve provides enough inventory to provide that coolant flow path from the

RPV to the CNV that cools the core adequately.

As Bill mentioned, these cores are much smaller then current industry cores. And so having that water available for heat transfer goes a low way in our success criteria runs that we have performed.

Our safety systems are fail safe.

CO-CHAIR CORRADINI: So, just a clarification. But if I cycle and freeze open, I turn into essentially in containment LOCA.

MS. BRISTOL: Correct. And we analyze that as well.

CO-CHAIR CORRADINI: Okay. That's what I want to make sure.

MS. BRISTOL: Yes. The safety systems passively fail safe. The ECCS functions to preserve that inventory within the containment, and allows the core cooling without additional inventory.

We then in PRA look at the beyond design basis where we need that inventory. But from just the containment being isolated, ECCS functioning, there's no additional inventory needed.

We have talked about the lack of need for power or operator actions because these -- the containment isolations fail closed. The ECCS valves fail open. The HRS actuates. All without electrical power and operator action.

MEMBER BLEY: I have a question. And we haven't gotten to six. But I'm not sure if I remember seeing it in there.

The ECCS valves, not the reliefs, they are fail open, spring operated. I assume it's spring operated. Are they held shut by air? Or some other fluid or mechanism?

Are they design?

MR. GALYEAN: I don't know how much detail we want to get into the design of the valves.

MEMBER BLEY: Then let's do it later. Because I really want to ask you about something there.

CO-CHAIR CORRADINI: Okay. We'll hold this for a closed session.

MS. BRISTOL: And then we go into the C-well items. Just overall completeness of the PRA as we've discussed, you know, there are a lot of open items and assumptions that are in the PRA.

And we acknowledge that at the design phase. There's a lot of information that isn't available. There's testing. There's operation experience. There's walk downs. All of those things that we can't do at this stage.

We've attempted to capture its assumptions. And we have items so that we will go back and address those items at the next phase, being a COL phase.

And so we have a lot of reminders to go back and check various critical assumptions that we make in the PRA.

MEMBER BALLINGER: I have sort of a metallurgical question, I guess. And that is, are some of these items that you've identified as needing to be done later on, are they -- do they have the possibility of being a very big ticket item which would change your results?

MS. BRISTOL: There's always that potential if there's something. But as you've seen in item eight, you know, the applicant, the COL applicant, you know, will confirm the validity of the key assumptions in the data use.

You know, and so as Bill mentioned, if there's testing, or if there's additional information that gets applied to the PRA, as we've discussed, we believe our design, you know, the way it's functioning, there aren't.

MEMBER BALLINGER: But somebody must have had the discussion around the table at Starbucks, which one of these, if they happened, would screw us up big time?

MEMBER BLEY: Well, they had that in those numbers they showed us on the Risk Achievement Worth.

MEMBER BALLINGER: Yeah. Yeah. But those are --

MEMBER BLEY: But you may have others you've talked about.

MEMBER BALLINGER: That's what I mean.

MS. BRISTOL: Well the ECCS valve is an example. You know, we -- they're important and we'll evaluate them again you know, if they're -- when additional information becomes available.

But they'll still be important. And they'll still be safety related.

MEMBER BALLINGER: But I would think that those are not, even those are not that bad, because you would then redesign it. Presumably.

Are there things that you can't design yourself out of on this list that would cause real problems? I'm assuming you can make a valve work.

MS. BRISTOL: Nothing that we can think of, --

MEMBER BALLINGER: Okay.

MS. BRISTOL: Or that we've evaluated to

date. So we don't.

(Off mic comment)

MEMBER BALLINGER: Huh?

MEMBER BLEY: I said in principle Ron, you could make a plan.

CO-CHAIR DIMITRIJEVIC: Well, to Ron's discussion, I would like to say not with the question that you guys are meeting safety goals. And whatever we question, and whatever uncertainty means produced, you're not going to go above the 10 to minus 4 or 10 to minus 6, so.

The reason we ask so many questions is because when you're coming with claiming that basically you have a zero risk, and that not too many things are important, like our job is to make sure that not too many things are important.

Because if we -- if some of those uncertainties and sensitivities bring something else that's important, you know, there is the program, the wrap, things like that, that that should be check procedure.

You know, you will hear, we are here to identify is there vulnerabilities. Is there something which is essential to keep you where you say you are?

And it also the difference is, are you a

10 to minus 7 or are you a 10 to minus 9 is also different in concluding what is important and things like that.

So, it's not every, you know, that's why we have to clean all of these to make sure. I mean, for me that means that I have to feel comfortable that we did not miss something that, you know, which is like, you know, the crane movement which should have a procedures check or some, you know, just something with testing.

I notice that you wrote out that you assume the test is staggered. Well, but then I concluded that you assume everything is tested during the refueling.

So, I mean, I don't know what does that assumption mean? I find a lot of things which makes me think, you know, about things like that.

So, this is why I feel that you're dreaming. We just want to feel secure ourselves.

MS. BRISTOL: With respect to the testing, the -- we say non-staggered to be conservative.

CO-CHAIR DIMITRIJEVIC: Yeah.

MS. BRISTOL: And only for instance, DHRS and ECCS. The other things can be tested. We don't assume that all testing is done at refueling outages.

Only those items for PCCS.

CO-CHAIR DIMITRIJEVIC: For the CVCS and the --

MS. BRISTOL: Yeah. Those can all be tested. And we have testing maintenance events for CVCS, demineralized water.

CO-CHAIR DIMITRIJEVIC: Well, CVCS is operating systems. So, I mean --

MS. BRISTOL: Right.

MEMBER BALLINGER: But to put things in perspective, these numbers in the words of that great Greek philosopher George Apostolakis, this cost is two times ten to the ninth year's old.

CO-CHAIR DIMITRIJEVIC: Yeah. I know. We will have to have dinosaurs building a million of those reactors to get there I feel.

Well, then one of the things is also, in your review that George Apostolakis panel said that this is interesting first step in multi-module. But they say first step.

So, we have to be a little more. You cannot just stay on this first step. We have to identify other issues associated with multi-module models to make a little dent in this, you know, multi-module system.

MEMBER REMPE: Could you put that previous slide back up where it had about the key assumptions? So that COL item is beyond just Section 19.1. Right?

Or is it just key assumptions as identified in Section 19.1?

MS. BRISTOL: That is associated with the PRA key assumptions that we make.

MEMBER REMPE: And just the PRA 19.1. So, if there are other key assumptions that you can't just identify using risk achievement work, I thought that that COL item actually pertained to not just frequency, but also consequence evaluations too.

MS. BRISTOL: That's correct. All of -- there's tables and tables in the FSAR. I don't --

MEMBER REMPE: Right.

MS. BRISTOL: Know that they were back. Yeah, all of those key assumptions.

MEMBER REMPE: Okay. So then how do you decide what's key and not key? Is it some -- is there some sort of -- to kind of ask what Ron's asking in a different way, is there some sort of a process that tells you that the key assumptions are those that are needed to provide reasonable assurance for adequate protection of safety and health of the public?

And how do you complete that assump --

that evaluation to come up with the key assumptions?

MR. GALYEAN: It might be more appropriate to say identified assumptions.

MS. BRISTOL: Yeah. Assumptions. Yeah.

MR. GALYEAN: You know, rather than key.

MEMBER REMPE: So then all assumptions.

MS. BRISTOL: Yes.

MR. GALYEAN: All assumptions. That's right. There's no particular significance attached to the work key other then these are the assumptions that we have explicitly identified --

MEMBER REMPE: Okay.

MR. GALYEAN: In the -- for the PRA.

MEMBER REMPE: Yeah. Thanks.

MEMBER SKILLMAN: Bill, I'd like to go back to a term that you used an hour ago, maybe two. How do you make the distinction between what is asset protection, and what is really operating experience proven safety protection?

MR. GALYEAN: Well, obviously the objective of PRA and Section 19 is to establish that the public health is protected here. Right?

And in order to achieve that, the NRC staff has identified these safety goals. And so the objective of the PRA is to show that we conform to the expectation in the context of the safety goals for core damage frequency and large release frequency.

I don't know if that, you know, answers your question. If there's something there that affects the calculation of core damage frequency or large release frequency, then it's more than just asset protection. Okay.

But if it does not affect the core damage frequency or the large release frequency, then that's what I classify as asset protection.

MEMBER SKILLMAN: Yeah. I guess I would just respond that brings to my mind the question of vigor and thoroughness in the model.

And if the review teams have genuinely used a thick magnifying glass and concluded this model represents a credible set of assumptions. And I can concur with your theorem.

Okay. Thank you.

MEMBER MARCH-LEUBA: So, less then ten minutes to go of course. What tool did NuScale use to calculate this obvious transient? You've seen the calculation, right?

Was it RELAP 5?

MS. BRISTOL: Yes.

MEMBER MARCH-LEUBA: With point kinetics?

MR. GALYEAN: Point kinetics and nodule kinetics as well.

MEMBER MARCH-LEUBA: With nodule kinetics or with point?

MR. GALYEAN: Both.

MS. BRISTOL: Both.

MEMBER MARCH-LEUBA: With both. And both survived?

MR. GALYEAN: Yeah.

MEMBER MARCH-LEUBA: Using minus 5 PCM for a high? Which is the --

MR. GALYEAN: I'm not sure of that.

CO-CHAIR CORRADINI: Before we get into this, if we can take -- we're going to take it up in the closed session. Right?

MR. GALYEAN: Sure.

CO-CHAIR CORRADINI: So, my suggestion is, let's take it up in closed session.

MR. GALYEAN: Okay.

CO-CHAIR CORRADINI: Because he's going to start with one question. And there will be ten more coming.

And I just have a funny feeling we're going to get into things that are more precise and proprietary.

MS. BRISTOL: Okay.

MR. GALYEAN: But just briefly, I mean we have a long history of looking at ATWS. I mean, when we first started looking at ATWS we used MELCOR using our developed model, which is a point, you know, uses a point kinetics model, I think, for the core.

MEMBER MARCH-LEUBA: But let's talk within the closed session.

MR. GALYEAN: Okay.

MEMBER MARCH-LEUBA: So we can talk numbers.

MR. GALYEAN: Okay.

MS. NORRIS: There were two questions brought up earlier that we just got answers to. I'd like to bring them up again.

CO-CHAIR CORRADINI: Good.

MS. NORRIS: So the first one was on the bioshield redesign affecting the PRA hydrogen analysis. So we did confirm that they will be minimizing the hydrogen concentration underneath the bioshield.

And this is discussed in our RAI response.

MEMBER REMPE: So again, my question was, yeah, I know that there is something that it will affect other phenomena. But what I want to know is, will it affect the structural evaluation you did?

Or are you going to reevaluate the structural integrity of it? Because you've made a change in the design.

And again, you don't have to get into the details of the design. But, just a definition.

MS. BRISTOL: Right. In our initial analysis we provided that change in design does not impact our previous analysis.

MEMBER REMPE: That's what I wanted to hear.

MS. BRISTOL: Yes.

MEMBER REMPE: Thank you.

MS. BRISTOL: You're welcome.

MS. NORRIS: And the second question was on the SMA analyzing of the different components. So, we did analyze the control rods. They were shown to not lead to core failure without additional random other failures.

And also specifically the steel piping with CVCS was actually not included, due to steel piping ductility assumptions. So, you asked about the control rods and the --

MEMBER MARCH-LEUBA: Yeah. So the one thing is, the control rods may fail, but even if they are failing, it doesn't make any difference.

MS. BRISTOL: Correct.

MEMBER MARCH-LEUBA: Is that what I heard you say?

MS. BRISTOL: Any additional failures.

MEMBER MARCH-LEUBA: Yeah. It turns into an otherwise. And you're telling me it's okay. Which we'll talk later.

MS. BRISTOL: Okay.

MEMBER MARCH-LEUBA: The CVCS pipe, you say is ductile and it won't fail you. And if it -- if you have high rates?

MS. NORRIS: Yes. Yes, and we do have RSME on the phone for that if you'd like more details on it, I believe.

MEMBER MARCH-LEUBA: No, I'm not -- the guy that knows how to do this operation is not here.

MS. NORRIS: Right.

MEMBER MARCH-LEUBA: So, I (whistle).

MS. NORRIS: But yes, that is the assumption anyhow.

MEMBER MARCH-LEUBA: I'm looking at breaks, but.

CO-CHAIR CORRADINI: Any other questions from the members?

MS. NORRIS: I believe Tom has one.

CO-CHAIR CORRADINI: Oh, Tom?

MR. BERGMAN: Tom Bergman --

CO-CHAIR CORRADINI: I think you're going to have to get closer. Or tap it. Is it working? No.

MR. BERGMAN: Now is it working?

CO-CHAIR CORRADINI: Okay. Now it is.

MR. BERGMAN: Tom Bergman with NuScale. I did get the answer. I was out of date on our status of our proprietary. The use of radar technology is not proprietary.

So, if you had questions on that, that didn't get answered, you can ask them now in the public session.

MEMBER REMPE: Well, I had asked earlier even, without worrying about what the mysterious sensor is, a question about the high pressurizer level setpoint that we could generate by itself according to the text on 19.1-97.

And it does not say there's a back up sensor that would give you that isolation signal for the CVCS. And again, you don't have to answer it now.

But I just -- that was a question.

MS. BRISTOL: Understood. And I think

what was discussed earlier, is while there are numerous sensors for all of -- for the various setpoints in the safety analysis.

And it's analytical and it's in the PRA we just model one. And so while we only discussed one of them, there are various different setpoints and sensors that would trigger, you know, a containment isolation, a reactor trip, so addresses.

MEMBER REMPE: So, it's Section 19 in the DCA. Every other place I looked at it, when it talks about the DHRS, it talks about the backup sensor signals.

Give me an idea of what is the back up for these CVCS isolation signals. And you don't have to do it today. But just sometime let know.

MS. BRISTOL: Okay.

MEMBER REMPE: Thank you.

CO-CHAIR CORRADINI: Other questions by the members?

(No response)

CO-CHAIR CORRADINI: Okay. Let's take a break for lunch. We'll be back at 1:30.

(Whereupon, the above-entitled matter went off the record at 12:24 p.m. and resumed at 1:27 p.m.)

CO-CHAIR CORRADINI: Okay, we'll come back into session and we're now going to hear from the staff. Greg, are you the one that's going to lead us off, or is it Alissa?

MR. CRANSTON: I'm just going to -- good morning. I'm Greg Cranston. I'm the lead project manager for the NuScale project, and I'm here on behalf of Rani Franovich, who's the chapter PM for this particular chapter.

A presentation on Chapter 19 will occur over the next two days, with today's discussion focusing on 19.1 PRA. And I just want to note that Section 19, formulated to loss of large area due to -- with the plant due to explosion and fires, is part of Chapter 20, which will be presented at a later date.

So, with that, I'd like to turn over to our initial presenter, Alissa, for this presentation.

CO-CHAIR CORRADINI: Okay, good.

MS. NEUHAUSEN: Good afternoon. My name is Alissa Neuhausen. I'm a technical reviewer in the PRA and Severe Accident --

CO-CHAIR CORRADINI: And bring it close. You're --

MS. NEUHAUSEN: This one's really far. Okay, I'll just start over. My name is Alissa Neuhausen. I'm a technical reviewer in the PRA and

Severe Accidents Branch.

This morning we're going to talk about the full scope of the PRA. We'll start with that, our Internal Events Level 1. We'll touch on Level 2, the power shutdown external events.

And then, I know this morning Alice came up. We do have some slides in the open session. We're going to go ahead and present those, even if we need to push that discussion mostly to the closed session.

So, I'm going to start with the staff's review approach. And then, I'll present some of the external events towards the end.

CO-CHAIR CORRADINI: And, you guys, make sure your green light's on and you talk loud for our -- yeah. Thank you.

MS. NEUHAUSEN: All right. This is slide 4. Okay, so the staff's Chapter 19 SE is based on Revision 2 of the DCA. We issued 31 RAIs, which contains about 59 questions. And that was for Sections 19.1, 19.2 and 19.3.

As part of the review, staff conducted two regulatory audits. These provided access to the PRA notebooks.

The first audit took place April to August of 2017, and staff reviewed NuScale documents which included over 50 of those notebooks supporting the PRA, and we asked 31 questions to clarify information in the DCA. Those were the formal questions that are included as part of the audit summary.

The staff sampled the notebooks. Those notebooks included both the self-assessment and the external review of the self-assessment that NuScale performed.

So, as a result of the audit staff was able to resolve some of the questions, and then issued RAIs based on some of the others.

So, staff determined that an extensive number of calculations and auxiliary studies support the description and results of the PRA that were reported in the SR, and that the scope and level of detail is generally consistent with the expectations of the NRC. The expectations documents are in NRC Red Guide 1.206 and SRP 19.0.

And then, the second audit occurred from March to April 2018, and we evaluated and examined documents to support those RAIs that haven't been resolved.

And then, also early in the review, staff acquired the Enhanced Safety Focus Review approach --

that's shorthand known as ESFRA -- to support integrated decision-making and increase the focus on safety for effectiveness and efficiency of the review. Next slide.

CO-CHAIR CORRADINI: When you did the audits, they were a week long? Because you said one in April of '17, one in March of '18.

MS. NEUHAUSEN: No. The first --

CO-CHAIR CORRADINI: Or much longer?

MS. NEUHAUSEN: The first audit I think was two months or four months, and the second audit was one month.

CO-CHAIR CORRADINI: And you went to the local offices, or out to the Pacific Northwest?

MS. NEUHAUSEN: The local offices.

CO-CHAIR CORRADINI: Local offices. Okay. So, in town.

MS. NEUHAUSEN: Yep.

CO-CHAIR CORRADINI: Okay. Thank you.

MS. NEUHAUSEN: So, during the review, staff focused on the quality, completeness and consistency of the information in the DCA to ensure that the results, conclusions and insights obtained from the PRA are valid.

The staff focused on the purpose of the

PRA, and specifically at the DCA stage. Some of those include like the determining the risk insights, provide information about risk contributors and defense and deaths, the inputs to operational programs in that some of those were incorporated into the design.

So, at the DCA stage many aspects of the PRA rely on key assumptions, and those are documented in tables throughout the Chapter 19 FSAR. And then, it's the responsibility of the COL applicant to confirm that those results are valid.

For uses of the PRA beyond those considered for DC purposes, the applicant or licensee will need to demonstrate acceptability in accordance with that intended use.

And so, for all the PRA topics, staff focused on ensuring that the appropriate key assumptions were included in those FSAR tables, and the review focus was guided by the commission's goals for core damage frequency, large release frequency, conditional containment failure probability, and PRA insights.

The staff used their review guidance provided in SRP 19.0, which includes the acceptance criteria for PRA and severe accidents, and the guidance provided in DC/COL-ISG-28, which is endorsed by reg guide 1.200, and addresses the use of the PRA standard.

NuScale committed to using that ASME/ANS PRA standard as endorsed by that reg guide and modified by the ISG, which is one way to assess the technical acceptability of the PRA at the DCA stage.

So, for the rest of the presentation, staff will cover specific topics of interest to NuScale that is related to NuScale-specific attribute, and I'll turn it over to Ayo.

MR. AYEGBUSI: All right. Good afternoon. My name -- sorry.

CO-CHAIR CORRADINI: Nice and loud.

MR. AYEGBUSI: That's always hard. Can you hear me? All right, good afternoon. My name is Ayo Ayegbusi. I'm a risk and reliability analyst in the Office of New Reactors.

So, over the next four slides my goal here is really to discuss topics or aspects of the PRA that the committee has shown some interest in, or that we had some further interactions beyond just what we saw in the DCA or during the audit, and we thought were interesting to highlight to the committee.

So, the first one had to do with data.

Right? So, the staff reviewed the applicants' data analysis, and for a large portion of the failure probabilities that we used, the applicant relied on generic data that the agency puts out.

However, for some, for unique components such as the ECCS valves, the applicant developed its own failure probability for those values. And so, the staff reviewed the reasonability of those failure probabilities and the assumptions that went into them.

MEMBER BLEY: Did you go through -- was this done during the audit? Did you go through the detailed engineering reports on those valves?

MR. AYEGBUSI: Actually, we cannot speak to that because the person who did the review and the audit is not here.

MEMBER BLEY: Was there only person did the audit?

MR. AYEGBUSI: On this particular area, yes.

MEMBER BLEY: Nobody who was involved in the audit can address that?

PARTICIPANT: Can you find out?

MEMBER BLEY: We'd like to hear back about that some other time, then.

MR. AYEGBUSI: Okay.

CO-CHAIR CORRADINI: I don't know if you were in the room or not. They did an audit in '17 and an audit in '18, a few months each time.

MEMBER BLEY: If you read the reports, it was a continuation of the same audit, but there were different questions.

MR. AYEGBUSI: So, you would like just to double-check --

MEMBER BLEY: I want to know what was looked at to decide if these things are reasonable or not.

MR. AYEGBUSI: Okay. We do -- I mean, as a result of the audit we do put out an audit report --

MEMBER BLEY: I got the audit report. It tells me the questions you asked. It doesn't tell what you found out.

MR. AYEGBUSI: So, typically with audit reports, we raise the topics that we discuss with the applicant. We don't typically make an assessment in audit reports.

MEMBER BLEY: You didn't. That's true, you didn't. I can't read the audit report and find the answer to my question.

MR. AYEGBUSI: So -- but I think your question is, you'd like to know if during the audit we looked at the engineering reports, engineering design of the valves.

MEMBER BLEY: Yes. And whatever was done by whoever did it to come up with the failure rates if they decided to do this.

CO-CHAIR CORRADINI: And to put it more succinctly, the applicant suggested that for things that were unusual -- we'll just call them the ECCS -- as an example, the ECCS valves that they called the piece parts analysis that developed the failure rate for the valve as a whole, the question is, did somebody from the staff look at that and determine that it was reasonable, or had RAIs, or there was a re-analysis. Is that close, Dennis?

MEMBER BLEY: Yeah. I want to know the basis for the staff's evaluation.

MR. AYEGBUSI: So, I mean --

MEMBER BLEY: And I don't just want to hear that we looked at it and it was reasonable.

MR. AYEGBUSI: Okay. So, I can give you my -- the insights that I have from discussions with the individual who did the audit. Right? So -- because this particular set of failure rates for the ECCS valves were definitely something that we wanted to take a look at.

In this case, the person who did the audit looked at the applicant's assessment -- right? -- looked at the inputs going into the applicant's assessment, and I can only speak overall to how we would look at that. Right?

Typically, what we'd look at is, if you look at the industry generic data for valves -- right? -- you typically would see -- they're typically on the order of ten to minus three, ten to minus 4. Right?

So, when we look at what the applicant did from a reasonability standpoint, what we're looking for, we look to see if the final failure rate for the valve is somewhere in that ballpark, given what we have --

MEMBER BLEY: I think the ECCS valves are something like five times minus 5?

CO-CHAIR DIMITRIJEVIC: 5.90 minus 5.

MEMBER BLEY: That's not like ten to the minus 3. So, why is that a reasonable result?

MR. AYEGBUSI: So, again, in this case, there are many things we look at. So, we would look at the valve --

MEMBER BLEY: You can't tell me what was exactly --

MR. AYEGBUSI: That is correct.

MEMBER BLEY: I want to hear what was looked at and why it was decided that this is reasonable.

MR. AYEGBUSI: Okay.

MEMBER BLEY: So, I -- keep going --

MR. AYEGBUSI: Okay.

MEMBER BLEY: -- in generality.

CO-CHAIR CORRADINI: And just -- I'm sorry. Just so you know where we're coming from, so at the end of today we're going to come up with suggestion at the June meeting what you want to emphasize in the presentation. This you want to emphasize.

MR. AYEGBUSI: Okay.

DR. SCHULTZ: The conclusion on the slide seems to indicate that in the evaluation of the component reliability or probability, could have been dismissed because there was so much margin between the CDF goals and that which was calculated. And we prefer not to hear that. We prefer to know what was determined associated with the failure rate calculations and whether they were valid.

MR. AYEGBUSI: Okay. Our intent was not to make it seem that way. Our intent was to say that because this is a new design -- the whole plant itself is a completely new design. These unique components are new designs that have no previous operating experience. Right?

As I was trying to explain earlier, a couple of ways we, as a group, we evaluate reasonability of such things as to look at how it compares with similar components that have operating experience, but also big-picture-wise, to look at how, you know, using these failure rates, how the results compare with commission goals, and also looking at the sensitivity studies that were performed, and looking at how those compare with the commission goals.

And so, all we're saying here is when we go through that process, is the process we in our group typically do, when we go through that process we still find that the results are still favorable when you compare them to the commission goals. That's the point of this slide.

DR. SCHULTZ: Thank you.

MR. AYEGBUSI: Yes, sir.

MEMBER MARCH-LEUBA: The language only on the first point says that the things are reasonable for the DCA stage. Was that in play that you were going to revisit this at the COL stage?

MR. AYEGBUSI: So --

MEMBER MARCH-LEUBA: So, once it's approved, it's going to be -- nobody's going to touch it.

MR. AYEGBUSI: So, particularly with the PRA, our expectation is if there is any additional information that the next revision of the PRA would consider any new information. Right?

So, if you needed to update some of the failure rates -- right? -- our expectation is at least that would be considered. Right? We can't say for a fact that that would become the new failure rate when you consider any new information.

MEMBER MARCH-LEUBA: You expect the applicant to initiate that change.

MR. AYEGBUSI: The COL applicants.

MEMBER MARCH-LEUBA: The COL applicants.

MR. AYEGBUSI: Yes, sir. I think Ian wanted to say something.

MR. JUNG: Yeah. So, I just want to share the information that I have on audit notes. Ian Jung, Senior Reliability and Risk Analyst.

So, those staff participate in the audit left actually detailed notes. It's a proprietary information here. So, I'll share some information through a different channel.

Specifically here, it talks about the staff auditing data analysis ECCS system notebooks, PRA analysis of the ECCS. So, there are multiple pages of detailed information what staff looked at. That's one information.

And also, we've been working with the mechanical engineering branch folks. And actually, Alissa and I are currently a part of the audit team participating. And Tom Scarborough, another member, actually doing the audit of the ECCS actually valve testing that's going to happen pretty soon.

So, we are following up on that to make sure that there's no significant delta between the submissions and the system develop --

CO-CHAIR CORRADINI: Can you repeat what you said? You said that you're going to be part of the ECCS testing that'll be what, I'm sorry?

MR. JUNG: There's an audit ongoing right now on ECCS system valves. There's actually onsite valve testing that's going -- I think that's going to take place pretty soon.

MEMBER BLEY: The test program's defined?

MR. JUNG: You're asking -- I've seen the audited plan defines the system testing, the plan and the outcome, and the detailed information behind that. So, I think you're going to hear some of this information through another chapter.

MS. NEUHAUSEN: The answer, too, is that the MEB staff is reviewing that testing plan leading up to the onsite testing that they're going to be auditing.

MEMBER BLEY: We're reviewing Chapter 6 next month. You're going to be ready to talk about it then?

CO-CHAIR CORRADINI: I think it's in Chapter 39-something -- 396 -- but they're together next month. Am I -- you guys have to correct me, but I think somewhere in the notes that I can't find at the very instant here -- I think it's 396 -- is where valve testing or valve certification is part of.

MS. NEUHAUSEN: Yeah.

CO-CHAIR CORRADINI: I'll check.

MR. JUNG: Okay.

CO-CHAIR CORRADINI: But to answer your question, we're going to ask, if we can't get it this month, we're going to get it next month at the subcommittee meeting. And Chapter 3 is on Tuesday, June the --

PARTICIPANT: Eighteenth.

CO-CHAIR CORRADINI: -- 18th. Thank you.

PARTICIPANT: Okay.

CO-CHAIR CORRADINI: Does that help?

MEMBER BLEY: It does a little. And I guess I -- I hope that the PRA review group that's talking to the people who are going to be watching the testing to make sure your concerns are being addressed by the people who are going to oversee the testing. You're part of the team --

MS. NEUHAUSEN: Yes. So, I'm part of the -- I'm not overseeing the testing but I'm part of the audit team.

MEMBER BLEY: The audit team of the testing.

MS. NEUHAUSEN: Yes.

MEMBER BLEY: Okay.

MEMBER MARCH-LEUBA: My concern with respect to the testing is that extreme reliability on the order of the ten to minus 5 is claimed for this complex valves, one of a kind. This distinction is important. I mean, if you're in testing else once, you cannot claim the ten to the minus 5 reliability. Just keep that in mind.

I mean, unless you run another 100,000. So, we need to have a problem that validates the --

this is a very high reliability for a one-of-a-kind, complex system. And I'm no expert on frequency of failure, but it's hard to believe.

MR. AYEGBUSI: Thank you. Understood.

CO-CHAIR DIMITRIJEVIC: I also have a couple of comments.

CO-CHAIR CORRADINI: Microphone.

CO-CHAIR DIMITRIJEVIC: Sorry. This table 19.1-9 in PRA with the comparison to be modified generic data. But they don't really give a generic mean values for the half of the table.

I assume -- they said not applicable, but I assume that's because they started something and it was modified so much that they considered not applicable.

However, I think that this table will be very useful to include these generic data. For example, for this hydraulically-operated ECCS, which they base on the BWR, because they give a generic source. They give data, too, put a star if it's significantly modified, but they should know where it starts.

So, I will complete this table it misses in the six of those -- misses this generic data.

The second bullet there is not true.

Sensitivity studies were not done on the component failure rates. Sensitivity studies were done on the failure of the passive heat transfer. That's completely different.

CO-CHAIR CORRADINI: Say that again, please?

CO-CHAIR DIMITRIJEVIC: Sensitivity studies were not done on component failure rates. They're done on this event, which says the -- how is it called -- passive heat transfer failure. That's a different -- you know there is a valve.

And then, there is a passive heat failure, and the sensitivity studies are not done on component failure rates. What you said in the second bullet is not true.

MR. AYEGBUSI: Okay. Yes, I understand what you're saying on components, specifically on components.

CO-CHAIR DIMITRIJEVIC: On component failure rates.

MR. AYEGBUSI: Yes.

CO-CHAIR DIMITRIJEVIC: There's nothing one in sensitivity on that.

MR. AYEGBUSI: Correct. Well, I guess it depends on what you're referring to, because the sensitivity done with common cause failure of components. I don't know if that --

CO-CHAIR DIMITRIJEVIC: Yes, but that sensitivity only shows that that's important. So, we cannot conclude like that.

CO-CHAIR CORRADINI: Vesna, can I ask the applicant -- all right, I want to make sure, because you're looking at slide 25 from the applicant's thing where it had the sensitivity studies.

CO-CHAIR DIMITRIJEVIC: Yes.

CO-CHAIR CORRADINI: Is how Vesna described it correct? Or you also did component sensitivities? Is anybody from NuScale can help us here?

CO-CHAIR DIMITRIJEVIC: Well, if they did that, it's not presented anywhere.

CO-CHAIR CORRADINI: Well, I know. But I wanted to get clear to it. Yeah, they're going to turn it on. It take a while to energize. It's failsafe. Just keep on hitting it.

PARTICIPANT: There's no switch?

CO-CHAIR CORRADINI: No, no. They have to do it in the control room, I think. There you go.

MR. GALYEAN: Okay. Well, as we --

CO-CHAIR CORRADINI: You are?

MR. GALYEAN: I'm sorry. This is Bill Galyean, NuScale PRA Group. We did do the sensitivity studies on, for example, the component -- the common cause failure rates. Okay? The results, of course, of our PRA are dominated by common cause failure.

So, you talk about the five times ten to minus 5 failure probability for the ECCS valves. Remember that we scaled -- or the sensitivity study that we did on common cause failures, we changed the probability of the common cause failures to two times ten to minus 3.

So, indirectly, that's a sensitivity study on the failure rate, because the combined failure of -- the common cause failure of -- the common cause failure groups of the ECCS valves, both of the failing, was set to two times ten to minus 3.

CO-CHAIR DIMITRIJEVIC: I understand. But this is still not the same. You did the common cause so everything -- and it was put to two minus 3, is not change of the failure rate because exchange are both common cause factor rates even worse change, because otherwise, it's 2.5 E minus 6, you are changing to two E minus 3.

I know you're conservative, but I don't really still don't know how sensitive you are to this failure rate. I don't that if you make really conservative assumption, you still meeting safety goal, and I never doubt that. I just said that doesn't show us how sensitive are you to the failure rates of the valve.

CO-CHAIR CORRADINI: Of the valve specifically.

CO-CHAIR DIMITRIJEVIC: Yes.

MR. GALYEAN: Granted. There are lots of components and details --

CO-CHAIR DIMITRIJEVIC: Right. It's all --

MR. GALYEAN: -- that we could do sensitivity studies on.

CO-CHAIR DIMITRIJEVIC: -- these are -- I mean, really, on common cause of change here --

CO-CHAIR CORRADINI: Okay.

CO-CHAIR DIMITRIJEVIC: -- the big change. So --

CO-CHAIR CORRADINI: Okay. Go ahead.

MR. AYEGBUSI: Yeah. I mean, I agree with you, we could probably clarify the slide better.

CO-CHAIR CORRADINI: Keep on going. If there's quiet, keep on going.

MR. AYEGBUSI: All right. So, next slide

please? All right, so this is another hot topic earlier today. So, the staff reviewed the applicant's passive system reliability evaluation. And, you know, our assessment was documented in the safety evaluation report.

But mainly the staff was looking at the uncertainty around the passive systems that the applicant relies on, the DHR system and ECCS systems.

And as a result of our assessment, we raised some questions with applicant, one of which, a good example, had to do with the non-condensable gases, volume and distribution in the passive system.

The subsequence of that interaction with the applicant, the applicant -- I would say significant, but changed a significant portion of their discussion, the release of that, in the FSAR.

And upon review of the revision of the DCA in this particular area, the staff has determined that the applicant's passive system, the liability evaluation, was reasonable, and identified areas of potential challenges to the passive systems adequately. Next slide, please.

So, in our SER the staff documented one open item, which had to do with RAI 8840, question 19-2. In that RAI, what the staff identified was for LOCA's inside containment the applicant did not specifically identify an assumption that -- and those specific events that containment isolation would not be necessary for the safety systems to actuate and function, and get the plant to a safe and stable condition.

So, the staff engaged the applicant on this particular issue, and through subsequent supplemental responses, the applicant and audits and independent evaluation -- or independent calculations done on our part, we were able to find what the applicant assumed reasonable.

And so, we're awaiting the supplemental response that would capture this assumption in the supplemental response and the subsequent revision to the DCA.

CO-CHAIR CORRADINI: I don't -- I'm sorry. Do you understand this?

MEMBER MARCH-LEUBA: I was going to ask, what is the problem with not isolating --

CO-CHAIR CORRADINI: Thank you.

MR. AYEGBUSI: So, the typical prior response is when you have an event -- right? -- you have containment isolation. That bottles up containment. Right?

And when the ECCS actuation valves -- when the ECCS valves actuates -- right? -- allows for that passive cooling of the core through the vessel into the containment, into the ultimate heat sync. Right?

So, when we looked at the event trees specifically for LOCA's inside containment, the LOCA's outside containment isolation was shown to be necessary.

But for LOCA's inside containment, the applicant did not question if containment isolation was necessary to prevent core damage.

MEMBER MARCH-LEUBA: Did they perform a calculation with and without isolation, or did you perform a calculation? I cannot see how the isolation of -- if you don't really work inside containment, you're depressurized.

Whether the valves outside are open or closed, it changes the pressure a little bit, but not much.

MR. AYEGBUSI: I don't think I -- so, I think --

MEMBER MARCH-LEUBA: Do you have a calculation, a TRACE calculation, with and without isolation for an inside containment LOCA that's just a difference? Because I'm doing it in my head and I don't see much difference.

MR. AYEGBUSI: So, obviously for -- I don't have both calculations with me, but obviously for Chapter 15 -- right? -- you have to assume that you have containment -- you have to go with the plant design, and that's containment isolation. Right?

MEMBER MARCH-LEUBA: So, what you're saying, in the notices of record they assume isolation, and it's up to that 19 they didn't require it? Is that what you're saying?

MR. AYEGBUSI: I guess what I'm saying is the plant design is to have containment isolation. Right? So, in Chapter 15 you would. And that would support the first part of your question.

MEMBER MARCH-LEUBA: Right.

MR. AYEGBUSI: Right? The second part of your question is, in Chapter 19, after multiple interactions with the applicant, they said containment isolation's not necessary -- right? -- for LOCA's inside containment.

MEMBER MARCH-LEUBA: And the obvious path for that is you issue another ISA. Show me a calculation without isolation, and show me it's okay.

MR. AYEGBUSI: Correct. That's what we want --

MEMBER MARCH-LEUBA: Show me the result.

MR. AYEGBUSI: We went through that iteration, audited their calculation, and we had our own staff perform our own calculations, and we got to the point where we could reasonably conclude that what the applicant did was adequate.

MEMBER MARCH-LEUBA: I'm just saying it's fastest to us either to run the calculation --

MR. AYEGBUSI: Well, we did that. We had our internal people run the calculation.

MEMBER MARCH-LEUBA: It's much faster to run TRACE or RELAP than to talk about it.

MR. AYEGBUSI: Well, so, the real -- I mean, the number one concern, though, was the PRA's supposed to reflect the plant design. Right? So, that was the number one concern there.

MEMBER MARCH-LEUBA: That's good.

MR. AYEGBUSI: Not necessarily just the thermal-hydraulic response.

MEMBER MARCH-LEUBA: My claim is, that raises to an RAI that you issue earlier in the review, because there's a hole in the review.

MS. NEUHAUSEN: That is the case. We issued this early in the review and it's just there were subsequent questions.

MEMBER MARCH-LEUBA: I'm concerned this is still an open item.

MR. AYEGBUSI: Oh. Well, so, as I said --

MEMBER MARCH-LEUBA: It's easy to close. Very easy to close.

MEMBER BLEY: Or better yet, why is it still open?

MEMBER MARCH-LEUBA: Yeah.

MR. AYEGBUSI: So, I would say it's still open because obviously, like you said, we perform our independent calculation, and it took a while to understand their inputs, and kind of -- and then change around our inputs to ensure we're working along the same path, basically.

DR. SCHULTZ: And the SER says it's still open here.

MR. AYEGBUSI: Correct.

DR. SCHULTZ: So, there are differences that have not yet been explained between what has been submitted in response to the RAI? Or you just haven't had the time to put everything together to close the open item?

MR. AYEGBUSI: So, well, where we are as of today is the supplemental response that was -- a supplemental response was sent in and we need to review that, and then close out the open item. Right?

And that supplemental response was as a result of a public meeting we had with applicant where they explained their position and we found their position reasonable, or acceptable.

DR. SCHULTZ: It was close but not complete, in terms of the conclusion.

MR. AYEGBUSI: I would say we're close -- I would say at this point we're ready to close out this open item.

DR. SCHULTZ: Okay, thank you.

MR. AYEGBUSI: Any other questions?

CO-CHAIR CORRADINI: Keep on going.

MR. AYEGBUSI: All right.

MS. NEUHAUSEN: Next slide.

MR. AYEGBUSI: Next slide. Okay, so this is another hot topic that came up that we were told about that the committee was interested in. So, for ATWS the staff reviewed the applicant's ATWS discussion. And that's actually done -- and our SER's done in several places. Chapter 7 of the SER documents the staff approval of the ATWS exemption from 5062.

And then, Chapter 15 also touches on the overall conclusion of compliance to that. However, Chapter 19 documents the --

MEMBER MARCH-LEUBA: If I remember correctly, Chapter 7 -- I mean, or -- of the other ten to the minus 5?

MR. AYEGBUSI: That's correct.

MEMBER MARCH-LEUBA: All right. That would not drop you below the goal. If ATWS would fail the core and the probability to failure to scram is ten to the minus 5, you are at the limit, meaning you're still within the non-acceptable ban.

So, you have to combine the probability of failure rates -- the probability of failure to scram is ten to minus 5, and then you have to have additional failures to have core damage.

MR. AYEGBUSI: Correct.

MEMBER MARCH-LEUBA: For which you have to an on-the-record calculation that shows there's no failures if you fail to scram.

MR. AYEGBUSI: Correct.

MEMBER MARCH-LEUBA: But then, in Chapter 15, they have a paragraph that says, we are not going to license because the probability extends to the minus, I don't know, 43. An original number.

There is one of those logical holes in there. Somebody needs to do what Pete has done, which is analyze it and demonstrate that you don't fail with an ATWS, and you have to put it on record. You cannot just say it. And I haven't seen NuScale's calculation anywhere.

MR. AYEGBUSI: Okay.

CO-CHAIR CORRADINI: Don't agree with him unless you really agree with him. I don't know what the statement says up there. Is the statement saying that because of the MPS the failure to scram, to insert rods as one in ten to minus 5th and that's acceptable to the staff? Is that what that statement says? I'm still not sure what it says in front of me.

MR. AYEGBUSI: So, just to be clear, what we're really doing here is setting the stage for Pete to present. And all we're really saying is there's some discussion of ATWS in Chapter 7, Chapter 15, and Chapter 19. Right?

And the specific calculations that Pete -- well, analysis that Pete has done, he's going to present that in the next slide.

MS. NEUHAUSEN: Yeah. And I think this -- Ian, correct me if I'm wrong, that his commissioned CDF goal is 20 to the minus 5 per year --

MEMBER MARCH-LEUBA: Can you put your microphone closer?

MS. NEUHAUSEN: Closer? There was that SECY paper that we referenced in the SE that documented some goal for ATWS specifically.

MR. JUNG: Yeah. Ian Jung. So, around 1983, after the Salem ATWS event, that there was a whole series of conversations between the commission and the staff, and this particular the SECY paper, although it's not like current -- the SECY paper is like 700-something pages long, presentations back and forth and discussion of it.

But the whole basis for the ATWS rule, the purpose of the ATWS rule, is to reduce the risk from ATWS to be below one minus 5, based on the studies on the existing plants, PRAs, and all that.

Most of the plants were 20 minus 5-ish. So -- and some of them are higher. The commission wanted that ATWS to be below 20 minus 5. That is a policy acceptance goal. So, during the Chapter 7 review, Mark Caruso, who has retired since then, was involved in looking at ATWS risk portion of the exemption request to confirm that.

MEMBER MARCH-LEUBA: But administratively, the ATWS rule is a rule that imposes a number of design criteria that you must have in your reactor. Like you have to have an alternate means of rod insertion, and I don't remember them all right now.

And because -- is NuScale asking for an exception to that? To the rule? Is that what you're saying? They need that exception because they meet the goal that the commission had when they wrote the rule, but they don't really meet the rule?

MS. NEUHAUSEN: Yeah. So, that's what we were saying, is that this is kind of the administrative side to say Chapter 7 is what's documenting the ATWS exemption request.

But then, what we reviewed was, in Chapter 19, the beyond-design basis. And then, Pete's going to talk about his evaluation. And then, Chapter 15 makes the overall conclusion. So, this was really just trying to point you to the right chapters where we make different conclusions.

MEMBER BLEY: It just says what the criteria is, and that Pete's going to tell us about whether they made it or not.

CO-CHAIR CORRADINI: But maybe you guys don't understand. I'm still confused.

MEMBER BLEY: I think we need to wait for Pete here.

CO-CHAIR CORRADINI: Well -- but I want to make sure what the -- what is the exemption request?

The request is an exemption from the ATWS rule as stated?

MS. NEUHAUSEN: It was from a portion of the ATWS rule, and I have to get back to you. I didn't review --

CO-CHAIR CORRADINI: Okay. All right, I think that's what's, at least, confusing me. Maybe it's confusing him, too.

MEMBER MARCH-LEUBA: I know that they have an ATWS rule exception from it. I think we're going to get a clarification.

MR. GILMER: Jim Gilmer, NRO Reactor Systems. The clarification is, NuScale's only asking for exemption from the automatic turbine trip portion of 5062, not exemption from the ATWS rule entirely.

In pre-application discussions, there was back-and-forth on whether or not they also need auxiliary feedwater automatic initiation.

CO-CHAIR CORRADINI: Okay.

MR. GILMER: And there was something called the gap letter and the staff response to that, which we can provide to you.

CO-CHAIR CORRADINI: So, they're not -- make sure, Jim, they're not looking for an exemption to the rule, they're looking for a specific exemption from some of the actions.

MR. GILMER: From a portion of the rule. The 5062 C1, which is the automatic turbine trip. And staff has agreed with NuScale's position on the need for auxiliary feedwater. Basically, the decay heat removal system performs the equivalent function.

CO-CHAIR CORRADINI: Okay. Okay.

MR. GILMER: So, they don't need to ask for an exemption from the aux feed initiation.

MEMBER BLEY: My memory entirely -- when we did Chapter 7, we had a fairly extensive talk about the turbine trip and it seemed to me there was something we were waiting for when we got to some other chapter. Remember?

MR. GILMER: No, I'm just -- I'd have to look at the notes.

MEMBER BROWN: I just pulled it back up and yeah, it's very explicit. In 7.1.6 they talked about an exemption from the diverse turbine trip --

CO-CHAIR CORRADINI: Right.

MEMBER BROWN: -- capability. And I --

MEMBER BLEY: I was just looking at that.

MEMBER BROWN: So, my memory's getting jogged. I'd have to go back and -- I don't remember talking about heat in the transcript. I'd have to go back and look.

MEMBER BLEY: We did. But I thought there was something reserved until later.

MEMBER BROWN: I'll go back and look at the transcript.

CO-CHAIR CORRADINI: Thank you.

MEMBER BLEY: I think we need to do that.

MS. NEUHAUSEN: We can turn it over to --

MEMBER BROWN: But that was the only part of it. It was just the discussion. And I remember that discussion.

CO-CHAIR CORRADINI: That was -- okay, so that's a lot more specific. That helps.

MS. NEUHAUSEN: Okay. Pete will present the research ATWS.

DR. YARSKY: Hello. I'm Dr. Peter Yarsky from the Office of Research and I'm here to discuss confirmatory calculations that we performed with TRACE for ATWS scenarios for NuScale.

In performing these calculations, the key figures of merit that we considered were the peak reactor vessel pressure, which is to confirm the integrity of the RPV.

Additionally, we looked at the collapsed liquid level in the riser to ensure that that liquid level was above the top of active fuel, and that's to confirm the core coolability. These are figures of merit that we used in our analysis.

We go to the next slide. The base case ATWS that we evaluated is initiated by a loss of AC power. And this leads to an immediate turbine and feedwater system trip, at which point we assumed that the module protection system fails to insert the control rods.

And for our analysis we assume that the control rods remain withdrawn through the entire event.

MEMBER BROWN: So, you mean you assumed failure of all divisions of the module protection system.

DR. YARSKY: Right.

MEMBER BROWN: And they do not initiate a trip.

DR. YARSKY: Right. So, we assumed that the module protection system --

MEMBER BROWN: -- I got it. All right.

DR. YARSKY: -- there's no diverse protection --

MEMBER BROWN: I just want to make sure I had your words down exactly.

MEMBER MARCH-LEUBA: It's either that or there is a mechanical failure that presents the rods from going in.

MEMBER BLEY: All the rods.

MEMBER MARCH-LEUBA: All the rods.

DR. YARSKY: All the rods. From the standpoint of the TRACE calculation, the control rods remain withdrawn from the full transcript.

In response to the event, the RPV pressure increases due to the loss of heat sync, and the reactor cooling system heats up. This higher RPV pressure will initiate the actuation of the key heat removal system.

As RPV pressure continues to increase beyond that during the ATWS event, this would initiate opening of the reactor safety valve. We analyzed --

MEMBER BLEY: You didn't -- yeah, you didn't consider turbine bypass at all either. Everything's bottled up.

DR. YARSKY: Right. So, if the way we simulate the event is the turbine will trip, and then we don't simulate any kind of turbine bypass. Instead, we would simulate the actuation of the DHRS valves to open.

MEMBER MARCH-LEUBA: Yeah. Plus, you've

assumed loss of AC power, so there is no feedwater -- there is no steam flow to go through a bypass.

DR. YARSKY: Yeah. So, the turbine bypass valves were to open and DHRS was unavailable, there would be no feedwater to supply any kind of liquid to remove heat from the primary side.

MEMBER MARCH-LEUBA: There wouldn't be any steam to go to the bypass.

DR. YARSKY: Exactly. So, seismic analysis has performed a number of sensitivity studies. And what we looked at were different key scenarios. Beyond the base case, we looked at -- we did perform calculations of both end-of-cycle and beginning-of-cycle kinetics parameters.

We also performed a case where we assumed that the RSV-1 valve was out of service. We then considered a scenario where both RSV-1s --

MEMBER BLEY: I don't remember, what's RSV-1?

DR. YARSKY: So, there's two reactor safety valves that are at the --

MEMBER BLEY: Yeah, this is one of them.

DR. YARSKY: -- on the pressurizer. So, RSV-1 is the lower pressure --

MEMBER BLEY: Thanks.

DR. YARSKY: -- RSV valve. Yeah, so they have lift and set pressures, and they're in different bands. RSV-1 is the lower pressure one.

We also performed a case where we assumed that RSV-1 is out of service, but in addition, DHRS is out of service.

MEMBER MARCH-LEUBA: All the interest.

DR. YARSKY: All the interest.

MEMBER MARCH-LEUBA: On both of them?

DR. YARSKY: Right. So, there's no DHRS at all. And lastly, we did a sensitivity calculation which is like the base case, but we have artificially reduced the steam generator heat transfer until we achieve a very high initial RCS temperature.

MEMBER MARCH-LEUBA: Okay. On the PRA case you still have one RSV working. Correct?

DR. YARSKY: Right. So, RSV-2 is in service, RSV-1 is out of service. So, we go to the next slide.

Just to summarize our findings in these cases, the base case, the BOC case --

MEMBER MARCH-LEUBA: Sorry. Let me -- you're going to show us your backup slides on the closed session. Right?

DR. YARSKY: If you would like to see the detailed calculation results, we can discuss them in the closed session.

MEMBER MARCH-LEUBA: Yeah. I will reserve my questions for the closed session.

CO-CHAIR CORRADINI: I told them you might do that.

MEMBER MARCH-LEUBA: Clairvoyant.

DR. YARSKY: Yeah. So, in the closed session we can discuss our results in more detail. But I wanted to present a summary of our key findings.

The base case, the BOC case, and the 1-RSV case that we've analyzed, are largely quite similar. And what they all demonstrate are large margins to the RPV, the pressure limit.

And in the long-term, we find that the reactor power comes at the balance with the DHRS heat removal in conjunction with a little bit of heat removal through the CNV, or the containment vessel, and that the long-term level remains well above the top active fields. So, therefore, we find that the RPV integrity is maintained, and core coolability is maintained.

For the PRA 1-RSV case, well, this is only one RSV in service and no DHRS. We find that there's still large margin to the peak RPV pressure, and in the long-term some inventory builds up in the containment vessel. And this provides a heat removal pathway.

And so, the heat generated in the core is balanced by the heat removal through the CNV, and we find that the level remains well above the top of the active fuel.

PARTICIPANT: And for the second bullet, the DHRS is also out of service.

DR. YARSKY: Right. Correct. The DHRS is out of service for what I'm calling the PRA-1 RSV.

CO-CHAIR CORRADINI: So, I'm only dumping heat through essentially -- through the containment.

DR. YARSKY: Correct.

MEMBER MARCH-LEUBA: You're venting entropy through the steam that goes into containment.

DR. YARSKY: Yep.

MEMBER MARCH-LEUBA: I guess you're losing more entropy by condensation of the steam by that. But done by --

CO-CHAIR CORRADINI: It all goes up as water eventually.

DR. YARSKY: Early on, you'll be venting from the RPV into the containment vessel. So, you're burping steam through the RSVs.

MEMBER MARCH-LEUBA: Mm-hmm.

DR. YARSKY: However, because you don't have the DHRS available, the RCS does reach a really high temperature. When it's at that really high temperature, the reactor shuts down, so it's in a subcritical state. So, you're still on decay heat but just at a very high temperature.

Once you build up a decent amount of fluid inventory in the CNV, you're actually able to convect about a decay heat-level worth of heat through the containment. And then, RSV cycling stops.

So, you eventually reach a point where you're at high pressure and you're hot, but you're below the RSV lift pressure.

So, we can show more of those results, and specifically like what's going on during the --

MEMBER MARCH-LEUBA: Let me ask on the opposition and learn the details. Do you have time to calculate what the moderator temperature coefficient is on your simulation?

DR. YARSKY: We provide a table of the moderator temperature coefficient as a function of the reactor in a hydraulic state when that's interpolating. But I don't have that value.

MEMBER MARCH-LEUBA: Is that document

somewhere? I haven't seen it.

DR. YARSKY: Yeah.

MEMBER MARCH-LEUBA: I know it's -- I'm asking you where is it.

DR. YARSKY: The values come from the design calculation --

MEMBER MARCH-LEUBA: Oh, no. Where can I find that document to look at it?

DR. YARSKY: The document would be the parts calculation notebook, which is the source for the kinetics.

MEMBER MARCH-LEUBA: I would like to be able to look at it before the full committee.

DR. YARSKY: I'll work with the staff and ask to give you access to all of our calculation notebooks.

CO-CHAIR CORRADINI: It's an open book for you. Mike? Mike, Mike.

CO-CHAIR CORRADINI: He's writing.

DR. YARSKY: We'll have to figure out the logistics.

MEMBER MARCH-LEUBA: So, you did calculate an equivalent MTC for your BOC calculation.

DR. YARSKY: Correct.

MEMBER MARCH-LEUBA: Now, we're talking --

I, and obviously this is open session. This is what I think the rules are. If you want to do a best-estimate cycle-specific calculation, you are perfectly welcome to run a best-estimate, cycle-specific calculation.

If you want to do a one-of-a-kind FSAR calculation that demonstrates that you don't have an issue, you have to use your corporate limit report bound in numbers.

And if you're doing a genetic -- that's what I think the rules are. You either do a bounding calculation, or you repeat the calculation for a recycle. And you're welcome to do either of the two. And I think you're thinking an imaginary review cycle, and call it bounding. That's what you're doing now.

DR. YARSKY: Well, I think that the question of reload licensing is very separate from what research today.

MEMBER MARCH-LEUBA: But -- well, not you. I'm thinking ahead. Do you understand the logic? If you use a cycle-specific MTC -- moderate temperature coefficient -- that's completely acceptable. If the cycle is specific, you have to do the calculation and recycle.

If you want to use that calculation to say

I will never have a problem anymore, but in my COLR -- cooperator limits report -- I'm going to allow them to see to be much higher, indeed, they're allowed to be plus-6, PCM per Fahrenheit, there is a disconnect. So, if --

CO-CHAIR CORRADINI: I think he's trying to -- I think Member March-Leuba is trying to say to you is, you took a best-estimate and --

DR. YARSKY: Correct, yeah. The way I would characterize the TRACE calculation is, it's best-estimate based on equilibrium cycle.

MEMBER MARCH-LEUBA: All right.

DR. YARSKY: Right. As opposed to a COLR-limiting value or generic-limiting value.

MEMBER MARCH-LEUBA: Or a bounding -- I mean, the FSAR in Chapter 4 I believe, there's a figure of what is the maximum MTC that you can have. At least negative. And it shows a curve, and it was calculated using some type of procedure. That's the one one should use for a bounding calculation.

MR. SCHMIDT: This is Jeff Schmidt from Reactor Systems. So, what Pete did is a Chapter 19 event. For something like you're describing, which is say a 15.8 event, if they have a diverse actuation system, there's no reload analysis that's necessary.

They've shown that the probability is so low with the diverse actuation system, that you don't need to do a calculation, like on a reload basis.

MEMBER MARCH-LEUBA: I thought the failure to scram priority extends to the minus 5. You get to the ten to the minus 11 by adding additional failure.

MR. SCHMIDT: Yeah. I don't know the details of the probability. But I think what was done in Chapter 7 was basically show that the MPS effectively meets the diverse actuation system requirement.

And once you make that requirement, there's no Chapter 15 analysis that's performed.

MEMBER MARCH-LEUBA: Even for the FSAR?

MR. SCHMIDT: Even from the FSAR.

DR. YARSKY: Right. So, if you look at 15.8, there'll be no calculation of say peak RCS pressure.

MEMBER MARCH-LEUBA: I know there is nothing on 15.8.

MR. SCHMIDT: Yeah. So, I'm confirming that there is nothing in 15.8.

DR. YARSKY: Right. So, if we were to re-perform the TRACE analysis but were to credit the diverse actuation system, then there would be control rod insert. The event would look in many ways very substantially similar to --

MEMBER MARCH-LEUBA: -- Assuming it was. But they're not American reactors. You perform numerous calculations with ARI that keep all the rods out.

DR. YARSKY: Well, for the BWRs, that would be the -- whether or not you credit the ARI.

MEMBER MARCH-LEUBA: Right. So, in operating reactors, we don't credit that independent actuation of the rods.

DR. YARSKY: So like, we'd need to swing a crowbar to separate out like how the PWRs are treated differently than the BWRs in this respect. But in the Chapter 15 analysis for the BWRs, there would be an analysis that would show failure of the rods to insert following the RPS signal, but would credit the insert of the rods with ARI. And then, there'd be a supplemental analysis assuming also the failure of the ARI.

CO-CHAIR CORRADINI: Remind me what the ARI is.

DR. YARSKY: Alternate rod insert.

CO-CHAIR CORRADINI: Oh okay, fine. Thank you.

DR. YARSKY: Right. But if -- and these TRACE calculations we were to credit the diverse actuation system, then that credit would lead to control rod insertion, and these TRACE calculations assume no control rod insertion.

MEMBER MARCH-LEUBA: Yeah. That would be an ATWS with rod insertion.

DR. YARSKY: Right.

CO-CHAIR CORRADINI: Keep on going.

MEMBER BROWN: Do you want the answer on the ATWS thing?

CO-CHAIR CORRADINI: No.

MEMBER BROWN: Not now?

CO-CHAIR CORRADINI: Not now. Let's wait until we go into closed session anyway. I would like --

MEMBER BROWN: Okay. It was done in open before, in our previous meeting. But that's okay. It can wait.

MEMBER MARCH-LEUBA: If that is going to be we don't need to do the calculation, why do we want to?

MEMBER BROWN: Well, you deferred. The bottom line is -- I'm going to be short -- his final comment on that was, in the Chapter 7, was we'll wait to see it in Chapter 15 and 19. So, we made no decision and our letter did not address it.

CO-CHAIR CORRADINI: But I think, just so we're on the same page, I think I understand what Jeff is saying, based on the definition of what is allowed to be credited and not credited for Chapter 15.

But I kind of want to get to finishing Chapter 19 for the moment.

DR. YARSKY: And just -- my last bullet that I want to discuss here is for what we're calling the SGHT case. We find that there's still large margin-to-peak RPV pressure limits, and that the liquid level in this case drops further.

It reaches about the top of the riser in this case, but it still maintains a significant collapsed liquid level above the top of active fuel.

MEMBER BLEY: PRA ought to consider best-estimate conditions and all others, with their likelihood of being what's going on at the time of an event.

This calculation, as I understood what you said, is strictly best-estimate. So, we don't -- and it's a confirmatory calculation. We don't know what things look like at their worst possible time, or at their best possible time, which ought to be part of the PRA calculation.

They ought to look at not just best-estimate conditions, but all of them.

CO-CHAIR CORRADINI: I think he did -- unless I misunderstand, they did a best-estimate with uncertainty?

DR. YARSKY: No. This is just strictly best-estimate. I think that is one of the --

MEMBER BLEY: -- But sensitivities. Four sensitivities off the best estimate.

DR. YARSKY: Correct, but not treatment of uncertainty.

MEMBER BLEY: Okay.

DR. YARSKY: But if you were mentioning, let's say point in time, we did consider different points in cycle.

MEMBER BLEY: Oh, you did? Okay.

DR. YARSKY: Correct. But that would primarily affect the nuclear parameters and the assumptions regarding the core. But are there other factors? When you said, in time, I just -- I'm not sure I fully understand.

MEMBER BLEY: No, I meant the PRA has to consider events occurring at random points in time.

DR. YARSKY: Okay. Yeah, here, we just

considered beginning a cycle and ending a cycle.

That concludes what I had to present on the TRACE confirmatory calculations.

CO-CHAIR CORRADINI: Until we go closed.

DR. YARSKY: Until we go closed.

PARTICIPANT: Conclusion?

MS. POHIDA: Good afternoon. I'm Marie Pohida. I'm the senior PRA analyst and NRO, and I'm going to be talking about Level 2 and module drop.

CO-CHAIR CORRADINI: Great.

MEMBER BLEY: May I ask you a question right at the outset? I was reading both your inspection -- I'm sorry, your audit reports. And the second audit, they don't tell us quite how, but they told us most all of the events, RAIs, that were looked at were either closed or no real outstanding issues, except for two.

And one of those is about corium retention in the reactor pressure vessel, and the other is the potential for high-pressure melt injection, both of which talk about due to phenomenological uncertainties.

This staff is continuing to evaluate these. Are you going to talk about those, too? Because I didn't find anything more about it and --

MS. POHIDA: No. That will be discussed tomorrow in our 19.2 discussion.

MEMBER BLEY: 19.2. Okay.

MS. POHIDA: Yeah. But thank you for providing a good lead-in to my slide.

MEMBER BLEY: You're welcome.

MS. POHIDA: I appreciate that. The containment event tree is very simple. There's basically two end states. There's leakage from an isolated containment, and release from an unisolated containment.

Severe accident phenomenon, other than a severe accident induced entire tube rupture, was screened from the containment event tree. So, if the RPV fails or the CNV fails due to corium, a large release does not occur due to pool scrubbing. But the details of our assessment are going to be discussed tomorrow in the 19.2 discussion. And also, hydrogen detonation is also addressed in Chapter 6 of the SER.

CO-CHAIR CORRADINI: So, just a question.

MS. POHIDA: Sure.

CO-CHAIR CORRADINI: My impression is that we're also then going to get --

MS. POHIDA: Thank you.

CO-CHAIR CORRADINI: -- some audit -- I'm going to call it audit calculations -- independent calculations by the staff tomorrow.

MS. POHIDA: Yes.

CO-CHAIR CORRADINI: Okay. Thank you.

MS. POHIDA: Next slide, please. Okay. Now, I'm going to be talking single module drop, as to differentiate between multi-module drop.

Okay, I audited the reactor-building crane PRA notebook. There was a calculated drop probability and it's dominated by operator errors of commission. It's over-speed, you know, over-rays, over-travel, and a failure of instrumentation, the inner locks or the switches, to provide a safety stop.

Then, I went to review the NUREGs and load drops. I went and reviewed the EPRI PRA and cask drops, and recent events, to further evaluate the drop probability.

So, when I went and reviewed NUREG 1774 and looked at operating data from 1980 to 2002, and when you're looking at load drops were greater than 30 tons, there was an estimated drop probability of 5b minus 5 quick demand.

And the events that went into that drop probability, they were rigging failures. They weren't crane failures, they were rigging failures.

The EPRI PRA for both the cask drops, they reported a drop probability in the order of E to minus 6 per lift. And then, I went to look at the details of the STATOR drop, the 525-ton STATOR drop at ANO, and that resulted from a temporary hoist assembly. The calculation was not reviewed or load tested.

MEMBER BLEY: We don't have, or at least I haven't seen detailed information on, the design of the crane. I think they told us that's coming at some point --

MS. POHIDA: That information was given to us during the audit.

MEMBER BLEY: Oh, it was? So, there is -- you've seen the design of the crane.

MS. POHIDA: I saw the PRA. I audited the PRA of the crane.

MEMBER BLEY: What I heard this morning, and I might not be getting this right, is that there are no rig devices that are our usual source of failure, but there's some kind of coupling mechanism that doesn't depend on people going out and hitching up the crane. And I don't have a clue of how that works.

MS. POHIDA: The nuclear power module is

transported from the operating bed --

MEMBER BLEY: How it gets hooked.

MS. POHIDA: Yes.

MEMBER BLEY: To the crane.

MS. POHIDA: The module lift adapter.

MEMBER BLEY: Yeah.

MS. POHIDA: And that is a permanent feature. You know, a permanently designed feature for using -- for moving the module.

CO-CHAIR CORRADINI: I think what he's asking is, how does that permanent feature get attached to a particular module?

MEMBER BLEY: Yes. Is there a --

CO-CHAIR CORRADINI: Did somebody walk up there and kind of put A to B?

MEMBER BLEY: Does it screw itself on or how does it hook up?

MS. POHIDA: I'm going to defer to NuScale for those type of details.

CO-CHAIR CORRADINI: Okay.

MEMBER BLEY: The last time we talked about it they said there was no design.

CO-CHAIR CORRADINI: If it's in closed session tell us.

MR. GALYEAN: Yes, we can talk about this in the closed session.

MEMBER BLEY: Thanks, Bill.

MR. GALYEAN: We have, we'll get some pictures that you can take a look at.

CO-CHAIR CORRADINI: Thank you.

MS. POHIDA: Thank you.

MEMBER BLEY: Thanks a lot.

MS. POHIDA: Okay, my last bullet on the slide is, NuScale committed to the guidance that's used by operating plants, and that's NUREG-0554, supplemented by ASME NOG-1, for single failure proof of the crane. And that's consistent with operating plants.

And I just wanted to note that, in DCA Section, Chapter 9.1.5, there is a table that documents the max speeds and lift heights of the reactor building crane.

Can I go to the next slide please?

MEMBER SKILLMAN: Are those --

MEMBER BLEY: Leads me to the question, oh, go ahead.

MEMBER SKILLMAN: Are those the parameters, Marie, that you used? Those heights and lift speeds.

You said you did the audit?

MS. POHIDA: Yes.

MEMBER SKILLMAN: Are the heights and speeds from the table in Chapter 9, the ones that you used for that audit?

MS. POHIDA: Let me see, I don't know how to describe this. What I will say, is the top cutsets are dominated by an operator of commission, followed by a failure of a limit switch or an interlock that is assumed to cause a module lock. And that's about what I can say.

CO-CHAIR DIMITRIJEVIC: The cutsets for the crane failure, for the module drop?

MS. POHIDA: I'm treating the crane and the module lift as a lift fixture. As an integrated lift mechanism.

CO-CHAIR CORRADINI: So, some combination of an operator of commission plus a limit, some sort of safety latch that doesn't latch?

MS. POHIDA: Yes.

MEMBER SKILLMAN: I was going after momentum.

MS. POHIDA: I'm sorry?

MEMBER SKILLMAN: I was going after the issue of momentum. Speed and height. The amount of energy tacked to the floor or to a lateral bumper of some sort.

So, when you said you did the audit, and you referred to the table that has the bridge trolley and the height, I was asking whether or not those are the numbers that you used in the audit. And I hear you said, no, not quite, it would be something else.

MS. POHIDA: From what I saw in the reactor building here, there wasn't direct correlation between those cutsets and the speed limits, if you will, that are documented in Chapter 9. Does that help?

MEMBER SKILLMAN: If you used those speed limits, that helps a lot.

MS. POHIDA: Yes.

MEMBER SKILLMAN: You did?

MS. POHIDA: I beg your pardon?

MEMBER SKILLMAN: You did use those speed limits?

MS. POHIDA: It helped to substantiate the drop probability those speed limits were documented in Chapter 9 of the DCA. Does that answer your question?

MEMBER SKILLMAN: No. And I'm confused.

MS. POHIDA: Okay.

MEMBER SKILLMAN: We don't have to debate it here.

MS. POHIDA: Okay.

MEMBER SKILLMAN: I don't understand.

MR. GALYEAN: Can I chime in and ask that we defer some of these details to the closed session?

MEMBER SKILLMAN: Sure. Yes. I'm not trying to be a bulldog here, I'm trying to understand. You've got a 762 ton load, how fast is it going, how high was it and what did you evaluate to determine everything is okay? That's a big load.

MS. POHIDA: I understand.

MEMBER BLEY: Marie, this probably, I'm not being aimed at you, but you and everybody else, when I read the audit reports it mostly said, during the course of the audits we reviewed documents and had discussions with them.

In past design certs, we, on the Committee, at least some of us have had the opportunity to review the event trees as, I'm sorry, the fault tress as well as the event trees, and to ask the Applicant to manipulate the PRA models so we can see importance of things.

During the audit, did you or some of the other people, especially in level, in the internal events activities, actually get a chance to do that?

Watch the PRA model being manipulated and changes and making sure by looking at some of the fault trees that you were convinced things were modeled well? And I don't know who to direct that to.

MS. POHIDA: Oh, for the reactor building crane notebook, I just did an inspection of the dominate cutsets.

MEMBER BLEY: Okay. But no looking at the actual computer model of the PRA?

MS. NEUHAUSEN: Yes, we didn't look at the actual.

MS. POHIDA: No.

MEMBER BLEY: Nobody did?

MS. NEUHAUSEN: No.

MEMBER BLEY: Every? Okay.

MS. POHIDA: All right. The risk significance of the reactor building crane did result in additional ITAACs. So there was, added was a rated load test of the nuclear power module lifting fixture and the module lift adapter.

And an inspection of the as-built nuclear power module lifting fixture and the model lift adapter.

There were also changes to the key assumptions table to state that the interlocks and limited switches will be functional during module movement.

This drop probability needs to be reevaluated for risk informed decision making, but the analysis did meet what our needs for SRP Chapter 19 and ISG-028, that we recognize that this is risk significant and that additional ITAACs were added and additional detail was added to the PRA assumptions.

MEMBER BLEY: You said something, you said it needs to reevaluated. you mean just these, you need to look at the ITAACs and see that they're meet or that you need to do something more on the analysis of the drop frequency?

MS. POHIDA: In the future, if this PRA is used for an application that's risk informed, then this module drop probability will need to be reevaluated.

MEMBER BLEY: I'm curious --

MS. POHIDA: We got --

MEMBER BLEY: -- as to what needs to be reevaluated and why. You've seen the design now, so what needs to be reinforced or added to in the future?

MS. POHIDA: As I learned from the audit, the design is not finalized.

MEMBER BLEY: Okay. But it's more finalized then I guess we had heard earlier, so.

MS. POHIDA: I beg your pardon?

MEMBER BLEY: At least there was a design that you got to look at?

MS. POHIDA: There was a design that I reviewed.

MEMBER BLEY: So it could be new and that's what needs --

MS. POHIDA: As I understand it, it was not finalized and it's being --

MEMBER BLEY: Okay.

MS. POHIDA: -- it's evolving.

MEMBER BLEY: Fair enough.

MS. POHIDA: May I continue? Okay.

CO-CHAIR CORRADINI: Please do.

MEMBER SKILLMAN: Do you want to?

(Laughter.)

MS. POHIDA: Yes, I do. I do.

MEMBER SKILLMAN: Let's get it done.

MS. POHIDA: I want to get to multi-module drop. Okay.

For external events, once again, I'm talking about single module drop. It's given a loss of AC power, and that could be either from external flooding or a high wind event.

There are redundant reactor building

breaks which will set and stop the motion. Each break is rated to hold the maximum allowable crane load. And it is assumed that this load can remain suspended until AC power is restored.

So, on that basis, we believe that this analysis is consistent with our guidance in ISG-028.

CO-CHAIR DIMITRIJEVIC: I'd like to ask you something. The calculate, they have a flow stream which calculates probability of the drop, which then was changing initial event sequence by calculating number of the movements through the air, right?

MS. POHIDA: Yes.

CO-CHAIR DIMITRIJEVIC: And what was that time? What was the mission time for that probability?

MS. POHIDA: You mean the transit time?

CO-CHAIR DIMITRIJEVIC: Well, you have a fault tree, which you analyze. What was the mission time in that fault tree?

MS. POHIDA: I think I need to defer this discussion because I can, to the closed session.

CO-CHAIR DIMITRIJEVIC: All right.

MS. POHIDA: If you're looking at duration of time, the module is in transit.

CO-CHAIR DIMITRIJEVIC: Yes.

MS. POHIDA: And for the three module drop scenarios, single module drop scenarios that were evaluated in the DCA. So if I may defer to the closed session I'd appreciate that.

CO-CHAIR DIMITRIJEVIC: Okay. All right.

MS. POHIDA: Okay. Thank you. All right, next slide.

Now I'm going to multi-module risk. And with our SRP, the applicant doesn't need to quantify a CDF and a LRF. But the Staff needs to look at the module drop assessment and ensure that there is no vulnerabilities from a multi-module event that is greater than an accident happening in a multi-unit site.

We also need to look at to ensure that there's no significant operator errors that could lead to a multi-module core damage event. Okay.

All right, so we believe that the applicant used a systematic process to evaluate multi-module risk. And I'm looking at internal events now.

And we believe, the assumptions on the multi-module factors, they are based on engineering judgment, but the design relies on an independent module specific safety related system that prevents and mitigate core damage.

In external events, they are evaluated

qualitatively.

Next slide. Okay, now I'd like to continue with multi-module drop. Okay.

CO-CHAIR DIMITRIJEVIC: Do you mean, it's not going to be multi-module drop because --

MEMBER SKILLMAN: Microphone.

CO-CHAIR DIMITRIJEVIC: -- more than one.

CO-CHAIR CORRADINI: No, she --

MS. POHIDA: Yes. As the module is being moved, I'm trying to be careful that I don't trip into proprietary space here, as the module is being moved from its bay, in the operating bay to the refueling, if it's dropped it can impact up to two operating modules.

CO-CHAIR DIMITRIJEVIC: Right. So you mean impact module drop. You said multi-module drop.

MS. POHIDA: You drop a module that's being removed, that's being moved, excuse me, for refueling, and it strikes an operating module.

CO-CHAIR DIMITRIJEVIC: Right. So it's impact of multi-modules.

MS. POHIDA: Okay.

MEMBER BROWN: Can you back a slide?

MS. POHIDA: Oh, I'm sorry.

MEMBER BROWN: Could you go back a slide?

MS. POHIDA: Yes.

MEMBER BROWN: Multi-module risk, there's a whole another briefing on Chapter 21 --

MS. POHIDA: Yes.

MEMBER BROWN: -- tomorrow. So, it just seems redundant relative to, you said go away, I mean, there's not a problem.

MEMBER BLEY: It's kind of content free compared to what we're hearing today --

MEMBER BROWN: Yes.

CO-CHAIR CORRADINI: So let's --

MEMBER BROWN: Well, I don't know. I guess it's got more slides as opposed to five bullets, or four bullets.

MEMBER BLEY: We talked about multi-module risk earlier today.

MEMBER BROWN: I know, I didn't bring it up then and --

MS. POHIDA: Oh, I'm sorry. Yes, the results of the Applicant's analysis was brought up this morning.

MEMBER BROWN: Yes, I understand that, but that was a summary. But then there's still another briefing on it tomorrow.

CO-CHAIR CORRADINI: But that one goes

beyond just this, that goes for --

MEMBER BROWN: Multi-module design considerations.

CO-CHAIR CORRADINI: Right. Which involves construction --

MEMBER BROWN: Not just drops.

CO-CHAIR CORRADINI: Not just drops.

MEMBER BROWN: This is just drops?

CO-CHAIR CORRADINI: Correct.

MEMBER BROWN: Okay, then I'll shut up until tomorrow. They're just going to get ready, that's all.

MS. POHIDA: Oh, I believe, we're talking about multi-module risk, the overview. What I just covered was the module adjustment factors that the Applicant used to come up with a multi-module CDF and LRF for internal events. And external events were evaluated qualitatively.

MEMBER BROWN: Good enough. So tomorrow was going to be addressing shared systems and all that other type of stuff, correct?

MS. POHIDA: Yes. Yes, that's the plan.

MEMBER BROWN: Okay, I'll wait.

MS. POHIDA: Thank you. Okay. All right. In DCA, in Revision 1, it's stated in Chapter 19.1.7, that if a module that's being moved for refueling drops on an operating module near the top, it could damage DHRS piping or the heat exchangers.

In Revision 2, there was an addition, and it states is, that additional pipe breaks may occur that could lead to a CVCS line break outside of containment.

So, we asked and RAI, and the RAI is, once again, it's to make sure our risk insights are complete, is what pipes are assumed to fail, which is CVCS, DHRS and the cavity flood and drain system.

And I guess more importantly is that if you have a strike to an operating module that's sufficient to cause pipe breaks, is the capability of the containment isolation valves to close compromised. And so, we are evaluating this event. This RAI is still under Staff evaluation.

CO-CHAIR CORRADINI: So, can I ask the question a little bit differently?

MS. POHIDA: Thank you.

CO-CHAIR CORRADINI: What we heard from the Applicant prior to that was there were other accident scenarios that in some sense bounded the damage from other approaches. You want to make sure that so that they analyze essentially what would occur with the module drop? That's what I hear you saying.

MS. POHIDA: We want to make sure that those conclusions, we can confirm those conclusions.

CO-CHAIR CORRADINI: Okay. Got it.

MS. POHIDA: Okay. All right, if I may continue, I'm going to talk about other external hazards.

For external flooding, the DCA states that there are no flooding penetrations, such as flood doors, that are credited in the analysis, so therefore no flooding penetrations were found to be risk significant.

For the high winds' assessment, the Staff verified that all important accident mitigation features are housed within a seismic Cat 1 structure, the reactor building structure. And thus, are protected from the effects of high winds.

MEMBER BLEY: I'm a little confused by the first statement, which is theirs and I didn't ask them about it.

MS. POHIDA: Oh, okay.

MEMBER BLEY: Since there are no flood doors, we don't have to worry about flooding. Well, flood doors are designed to keep the water out.

There are other doors, and are they all

sufficiently above grade that we don't have to worry about ingress of water from those or from ventilation systems?

I don't know where they're going to plant one of these things but.

MS. POHIDA: I'm going to have to defer to the Applicant on those specifics.

MEMBER BLEY: Okay. I didn't ask them because we didn't get to this level of detail.

MS. POHIDA: I asked an RAI on this, on the status of anything, if the operators are required to do anything for an external. You know, any doors, penetrations, anything needed to change state.

And as a result, there was an addition to the DCA that no flooding penetrations, external flooding penetrations were found to be risk significant. But I defer to the Applicant if there is more detail that's needed.

MEMBER BLEY: I apologize for not asking that this morning, but if you guys can respond to it. It just snapped for me there are no flood doors. Well, flood doors are designed to keep the water out.

A lot of plants have sand bags they put up to keep the water out. You have doors somewhere to get into this dang on thing and how do you know those are not under any risk of flooding or have ventilation ducts, that sort of thing?

MR. GALYEAN: We treat flooding as basically a loss of all AC power, okay. So we made certain assumptions --

MEMBER BLEY: So as the water comes in, it may short things out and lead you to something approximating a loss of offsite power?

MR. GALYEAN: Exactly. Exactly.

CO-CHAIR DIMITRIJEVIC: You haven't --

MR. GALYEAN: We simply assume that if a flood occurs, a beyond design basis flood, that it simply results in a loss of all electrical --

MEMBER BLEY: And you did some --

CO-CHAIR DIMITRIJEVIC: With point one, you have a factor there, which I asked last time, the point one, in ten percent cases you assume results in loss of loss of power. External flood.

MR. GALYEAN: Right.

MEMBER BLEY: So ten percent of external --

MR. GALYEAN: But it's --

MEMBER BLEY: -- floods or the floods of both --

MR. GALYEAN: The flood is --

MEMBER BLEY: -- design flood?

MR. GALYEAN: I think the design basis flooding. Ten percent probability that given you have a flood that exceeds the design basis that results in loss of all AC power.

MEMBER BLEY: Are the doors, the main access ways and equipment access areas, are those well above grade or where do they sit?

MR. GALYEAN: Obviously, we don't have a site.

MEMBER BLEY: If you, if they were up a little high, then ten percent of the floods above design basis floods is kind of reasonable. If they're right at grade level and design flood is anywhere near grade level, there's not such a good assumption, I think. But that's where we are.

MR. GALYEAN: Right.

MEMBER BLEY: It's an assumption --

MR. GALYEAN: That's right, it's an assumption.

MEMBER BLEY: -- and the COL is going to have to take a look at that.

MR. GALYEAN: That's right.

MEMBER BLEY: And see if it's reasonable for them.

MR. GALYEAN: Once there's a site that can be evaluated for --

MEMBER BLEY: Okay. So this ten percent thing is --

MR. GALYEAN: -- floods.

MEMBER BLEY: -- just an artifice for now?

MR. GALYEAN: That's right.

MEMBER BLEY: Okay. Okay, for now.

MS. POHIDA: Well, that concludes my presentation, are there any more questions?

MEMBER BLEY: Could you finish the high wind?

MEMBER SKILLMAN: I thought she did.

MS. POHIDA: Oh, yes, I am missing a conclusion here. So, for both external flooding and the high winds analysis, we found the analysis to be consistent with our Staff guidance, with our guidance in ISG-028 and SRP Chapter 19. Thank you.

MEMBER BLEY: So I may as well have fun with this. I don't know of anybody whose included it in their PRA, but I don't know why not.

Some of the existing plants, and you usually think of things like a large dry containment as being almost a Faraday cage with all the steel in it, but some of them have had lightning be brought inside through some penetration can really do some bizarre things inside the containment.

Did you, any chance you looked at, is that a possibility here?

MS. POHIDA: That was not part of my review, no. The effect of lightning.

MEMBER BLEY: Yes.

MS. POHIDA: Yes.

MEMBER BLEY: You know, for just an aside, for some weapons, bunkers, you see those things that the Military has, they thought they had Faraday cages, but they had a metal support or something else that extended inside and outside of containment, and if the lightning hit that, it can kind of bypass the Faraday cage and get inside. And once it gets inside it just jumps all around and burns stuff. And that's interesting.

At least one of our large dry containments has had that same kind of event with some pretty spectacular damage inside. You can design around it.

But personally, I don't know anybody whose looked hard at that, but it's an interesting failure mode.

MS. POHIDA: Would you like us to follow-up on that?

CO-CHAIR CORRADINI: He's trying to have fun with you.

MS. POHIDA: Okay.

(Laughter.)

MEMBER BLEY: Well, I'm a little bit trying to have fun with you but it's a fairly serious thing and it's been absent from most of our PRAs.

MS. POHIDA: Okay.

MEMBER BLEY: Maybe all of them. And I don't why it's absent. Except maybe most of us think, yes, it's a Faraday cage, nothing can get inside.

There's a lot been learned about lightning in the last 20, 30 years that wasn't known before. So you can give it a little thought and come back to us the next time around --

MS. POHIDA: Okay.

MEMBER BLEY: -- that would be interesting.

MS. NEUHAUSEN: All right. There's a still a couple more slides, so, next slide.

Okay. This is at power and internal fire and internal flood. So for the internal fire PRA, we focus on the assumptions used in the FPRA and the consistency with methods in NUREG-CR 6850.

And so we found that because this is the

DCA, many tests were omitted or simplified analyses were used. So, for example, specifics of cable routing, ignition sources, target locations weren't known.

And so, we still now reviewed what the assumptions were and made sure that the assumptions were included in these tables. And found FPRA sufficiently consistent with the SRP and ISG.

Similarly, for the internal flood, a lot of design details are unknown. Staff considered that the design is less dependent on active systems and that the mitigating functions of the active systems aren't credited for flood in the reactor building.

Next slide.

MEMBER REMPE: Could you move the microphone a little closer to you?

MS. NEUHAUSEN: I can.

MEMBER REMPE: Some of us are old and hard hearing.

MS. NEUHAUSEN: Sorry. I have two in front of me, so.

And then the PRA based seismic margins. For the PRA based SMA, we focused on the review of the scope of SSCs included in the fragility evaluation and the analysis methods that were used to determine the seismic fragility.

All of the SSCs were included in the fragility evaluation. There were two methods, conservative deterministic failure margins and separation of variables, which are endorsed by the SRP were used for the seismic fragility for the PRA critical SSCs.

And then the component boundary includes all failure mechanisms affecting component functions. So, Staff found that the plant-level HCLPF capacity demonstrated adequate margin in accordance with the SECY 9387 in the SRP.

Next slide. So, due to the open items mentioned earlier, Staff hasn't made a finding on the acceptability of the PRA for NuScale's Chapter 19 yet. And we can take any more questions.

MEMBER BLEY: Marie just told us about one of the open items. Is it a short list? Can you give us the other ones?

MS. NEUHAUSEN: Yes, we already talked about both of them. So, Marie talked about one and then OI spoke to the, it was the containment.

MEMBER BLEY: And that's all there are?

MS. NEUHAUSEN: Yes. For 19.1 there's just those two.

MEMBER BLEY: Okay.

CO-CHAIR DIMITRIJEVIC: I have a couple questions on something that you stated in SSC. That which I have a problem with the sum of the statement.

But boundaries related to Level 2, are you going to discuss the, are we done with the, all right, let me just ask you this --

MS. NEUHAUSEN: That's our whole presentation.

CO-CHAIR DIMITRIJEVIC: What?

MS. NEUHAUSEN: That's our whole presentation.

CO-CHAIR DIMITRIJEVIC: That's your whole presentation? You're done with the --

MS. NEUHAUSEN: There's the closed session on the ATWS, but --

CO-CHAIR DIMITRIJEVIC: Yes, okay. So let me just ask you, on the Page 19.16, in the end of the third paragraph says, the LOCA inset containment initiating event end loop are also embedded by very significant initiating event because they meet the least achievement mode to failure.

The initiating events are not ran by these achievements, they're only ran by Fussell-Vesely, so I don't know how that found the place there because, least achievement doesn't make sense for an initiating event.

MS. NEUHAUSEN: We'll take it back and make sure that it's correct.

CO-CHAIR DIMITRIJEVIC: All right. Achievement means what you will put in frequency to one, it just doesn't make sense.

The other thing, which I have on the Page 19.22, I didn't understand these statements. It says in the first paragraph, 19.1457 said, Applicant defines CCFP as the ratio LRF to CDF to the solvent certainties regarding potential failure of the RPV and CNV bottom half. What does that mean?

I mean, isn't the CCFP definition, I mean, what does this statement mean?

MR. AYEGBUSI: I'm sorry, what page is that?

CO-CHAIR DIMITRIJEVIC: The Page is 22 in the first section, 19.1457.

MR. AYEGBUSI: 19.1457.

CO-CHAIR DIMITRIJEVIC: Yes.

MS. POHIDA: May I answer that?

MR. AYEGBUSI: Go ahead.

MS. POHIDA: CCFP is Conditional Containment Failure Probability. And what I, I wrote that, what I'm referring to in there is when, remember, containment event tree is really simple.

CO-CHAIR DIMITRIJEVIC: Okay.

MS. POHIDA: Its, you know, all severe accident phenomena, other than severe accident induced, steam generator tube rupture was screened. Okay.

And so, that tree was, if you're looking at the screened phenomena and you're looking at its likelihood of causing containment failure, that will be discussed tomorrow. The uncertainty regarding severe accident phenomena --

CO-CHAIR DIMITRIJEVIC: Okay, but I'm going --

MS. POHIDA: -- as it impacts containment failure and RPV failure.

CO-CHAIR DIMITRIJEVIC: Okay, but it's not the CCFP failure or review of the LRF to the CDF, how does this resolve essentially?

I just don't understand what, I mean, you're not trying to bring in the definition of CCFP because that's also sort of controversial issue, right?

MS. POHIDA: Oh, okay. I'm sorry, I believe I misspoke.

Yes. We have the commission goals for new reactors, and we have a subsidiary goal of having a conditional containment failure probability of .1.

CO-CHAIR DIMITRIJEVIC: Right.

MS. POHIDA: But in this, but for this application, instead of looking at because the Staff is going to explain tomorrow in the 19.2 discussion, there's uncertainty with a severe accident phenomenology.

What we're talking about here is using, that definition is not conditional containment failure probability, it's meant to be the ratio of the LRF to the CDF.

CO-CHAIR DIMITRIJEVIC: Well, what do you consider to be condition of containment failure probability?

MS. POHIDA: I'm sorry?

CO-CHAIR DIMITRIJEVIC: How do you define conditional containment --

CO-CHAIR CORRADINI: They don't.

CO-CHAIR DIMITRIJEVIC: -- failure probability?

MS. POHIDA: For this application it's --

CO-CHAIR DIMITRIJEVIC: Just in general, how do you define it?

MS. POHIDA: Oh. I'm going to take an example for shut down, that's typically what I do. It's the likelihood of an operator failing to isolate containment.

CO-CHAIR DIMITRIJEVIC: Go ahead. I know every isolation failure is considered LRF --

MS. POHIDA: Yes.

CO-CHAIR DIMITRIJEVIC: -- but I just discussed, okay, even if you consider CCFP always to be LRF of a CDF, or there is some other definition and they're using this as a, to cover for something.

I doubt, I mean, we can consider CC, containment failure probability to be annually, right? It doesn't have to be large release.

So let's say that we arguable hear the containment failure probability is just large release, right?

MS. POHIDA: Okay.

CO-CHAIR DIMITRIJEVIC: I assume that you can reach this agreement before --

MS. POHIDA: I think I'm going to defer to Jason on this.

CO-CHAIR CORRADINI: Yes, I was going to say, I think the Staff is going to help her. James.

MR. SCHAPEROW: So, different, oh, Jason

Schaperow of NRO Staff. Different applicants, over the years, have actually used different definitions for what they consider conditional containment failure probability.

For example, ABWR actually used two definitions. They use one which is actually given a core damage accident, the actual chance that the containment fails. Like there's an actual hole in it or a tear or something.

But they also used a dose base definition that if you have a severe accident that dose at the site boundary is more than 25 rem.

So NuScale's application uses a definition of containment failure, given a core damage, definition of containment failure that we have a large release. And they gave their large release definition this morning.

Tomorrow morning we'll go into more detail why that works for the NuScale design review. Because of some uncertainties with regard to in-vessel retention of core debris sitting in the lower plenum of the reactor.

CO-CHAIR DIMITRIJEVIC: So, to make a, to be clear, that's not what I mean. I don't want to bring this definition of the CCFP.

Let's assume they're using large release to CDF, right, that's fine with me. This is the least conservative definition of containment failure probability.

So I don't understand this sentence that says, this ratio resolve uncertainties. Why would the least conservative definition of CCFP resolve any uncertainties? That's what I don't understand.

So this is resolve uncertainties is which I have a problem.

MS. HAYES: This is Michelle Hayes. I'm with the staff as well. We're going to look at -- I understand what you're saying. We're going to look at that. I think it's a grammatical thing.

MEMBER BLEY: I have two kind of quick things. One is for 30 years LERF was L-E-R-F and now it's become LRF which leaves me confused half the time.

The other is I'm going to have to look at this some more. The staff in their audits reviewed many of NuScale's engineering reports. One we talked about earlier on the ECCS valves and I think you're going to talk more about that in closed session.

Another that we heard about this morning I'm not sure. I couldn't find in the audit report if you had reviewed this or not. There's an engineering report on passive system uncertainties. And I don't know if I got that right. PSSR, PSSR, passive safety system reliability. That's the other one. But I know I would like to hear more about it. If the staff has anyone who can talk about that second one, I'd be very interested.

There are probably others that we'll ask about later. I don't think they rise to the importance of these two, but you can expect at least one member is going to ask again, maybe at the full committee meeting, then sometime between then and when we come around in the next phase we can hear something about those. That's that. PSSR.

CO-CHAIR CORRADINI: Anything else?

MEMBER MARCH-LEUBA: Yes, I have some.

MR. AYEGBUSI: First of all, so I guess, when you say engineering reports, that was confusing earlier. I think a lot of those refer to NuScale's PRA notebooks, right?

MEMBER BLEY: Some might have been notebooks and others were like ERPO 60 dash something else rev something and I understood that those were some kind of engineering reports. There were engineering reports on selection of initiating events and a bunch of other things that were discussed this morning. And I think I saw that a number of those who are in things people listed as being reviewed on the electronic system during your --

MR. AYEGBUSI: So we discussed the passive safety system reliability evaluation earlier today and we talked about our review and what we found and the iterations with the applicant.

Did you have specific questions that you would like us to address?

MEMBER BLEY: No, I want to know what was in your internal report on your audit of that report.

MR. AYEGBUSI: Internal report?

MEMBER BLEY: Well, we heard earlier that on at least the ECCS valves somebody back here was looking through the audit report by the guy who did the audit and had a lot of detail on what they looked at and --

MS. NEUHAUSEN: Yes, those are just personal notes, not any sort of internal report.

MEMBER BLEY: Well, they're still a sort of internal report. It had information on your audit for heaven's sake. So that's what I'm getting at and many of those items that you looked at on the audit were based on, at least when I read the audit report were based on your reviewing and discussing with the applicant their detailed engineering calculations for various technical topics.

So those two areas I would like to hear some more details on and they're probably going to be a couple of others where under other conditions I'd just get to look at those reports and read them myself and see if I had any problems.

MS. POHIDA: May I ask, did you want more details of the inputs to the study?

MEMBER BLEY: I want to understand why the staff felt that the approach NuScale has taken to developing failure rates for the ECCS valves and the process they use for considering the uncertainties in passive systems are adequate. And I don't want to hear that we read them and thought they were adequate. I'd like to hear exactly technically why. And the real reason is because I haven't had a chance to read the reports. I don't know what's in them. I can't make my own judgment about them, so I'd like to more thoroughly understand what issues the staff pursued and why you have been happy with those right now, those two issues.

CO-CHAIR CORRADINI: Jose.

MEMBER MARCH-LEUBA: Yes, I wanted to save you some time and I'm going to say for the record that I don't need a closed session for ATWS. I've seen the data that Peter is going to show and the rest are not not interested in it.

But I would like for the staff to confirm that I understand their position so -- the microphone is there.

MEMBER BLEY: Well, I really think your assumption is that nobody else is interested is wrong.

(Laughter.)

And I think that if he would get into some of that would be very useful.

CO-CHAIR CORRADINI: Yes, why don't we defer Jeff to when we see some numbers. I want to see some graphs.

MEMBER MARCH-LEUBA: It's important but the numbers make no difference. The staff position is that 10 CFR 5062, the ATWS rule, is a capability-based rule. It's not a performance-based rule. Thou shalt have ARI, turbine trip, and a number of things. And once you have those capabilities implementing your plan, you satisfy the rule. You don't need to do any analysis. Is that your position?

MR. SCHMIDT: This is Jeff Schmidt from Reactor Systems again.

So there are multiple parts of that rule like we're talking about where they did take an exemption to the turbine trip.

MEMBER MARCH-LEUBA: Sure.

MR. SCHMIDT: So that's --

MEMBER MARCH-LEUBA: But your evaluation --

MR. SCHMIDT: But the 15.8 evaluation you're referring to?

MEMBER MARCH-LEUBA: Right. So I was going to argue with Pete that he's done a cycle specific ATWS calculation when he should have done a bounding ATWS calculation for the FSAR, Chapter 15.8 on 19. But it's not required because they satisfy the rule with a coolant design and they don't have to do an analysis.

MR. SCHMIDT: That's right. So they satisfy the rule as written except for the exemption which --

MEMBER MARCH-LEUBA: Correct.

MR. SCHMIDT: Yes, that's correct.

MEMBER MARCH-LEUBA: And the fact that we have -- they have run some calculations with -- I haven't seen and Pete has run some calculations which I have seen that show that for cycle-specific numbers you're safe. It's icing on the cake. But it's not required to add additional conservatism and bounding assumptions.

MR. SCHMIDT: I think it's still informative and I haven't read Pete's report in a while at this point because he did it quite a while ago, but I think it's still very representative of what the module would do even on a reload basis.

So you're right. MTC does change with the reload. Right and MTC has a reasonably big effect on the response.

MEMBER MARCH-LEUBA: Okay.

MR. SCHMIDT: But you know, what I heard Pete say and maybe Pete can speak to it, too, I think is that the relief capability of the safety relief valve is pretty substantial and that protects your over pressure from the RCS being over pressure. So it allows the system to basically come to equilibrium with a heat decay removal system.

MEMBER MARCH-LEUBA: Okay, go back, sit down. We will have a closed session and we will talk about it.

MR. SCHMIDT: Okay.

CO-CHAIR DIMITRIJEVIC: I have one comment here also, Mr. Chair, which is very important and I don't really know how I want to treat this from here.

On the Chapter 9 in the SER on page 19.15, you said the following: The uncertainty base on the CDF reported by applicant accounts for only parameter uncertainty not model uncertainties. Therefore, the staff finds that a design state uncertainty could be very large. Okay? You are right. It should be very large. That's not what applicant reported. Applicant reported very actually small uncertainty and if you presented the numerical data with 95 and 5 percent, you will see because there is only base unremittant parameter uncertainty.

So my question is even with the latch potential uncertainty, velocity estimated a flight barely initiating the same effort to reduce or eliminate the contribution to CDF found in the previous PRA. So my question is, first, do you offer even the table you presented in the beginning you can only use mean values. That doesn't give uncertainty range. And if you are expecting large uncertainty and you didn't see large uncertainty, why didn't you question that?

MS. NEUHAUSEN: I think that what the staff was trying to convey here is just that even with large uncertainties and we're measuring this against the Commission's safety goal. And so NuScale reported their uncertainties. We did our review, but there's really -- we can't ask any questions. And so even though there may be -- even if there is more uncertainty than is shown in those -- what they've included, it's still not going to reach the safety goal as we're measuring that against.

CO-CHAIR DIMITRIJEVIC: That's why I don't really know. I just want to say you say here large uncertainty, but it's not true. You could say surprisingly we saw very low uncertainties with all of these uncertainties. Something is -- I'm not sure really where this leads because you would assume as they go through the collar, this uncertainty will reduce, but you don't see it. I mean you cannot really go back to do the more complete uncertainty or you don't want to question others factors of the sum of those elements which you know are very questionable.

CO-CHAIR CORRADINI: There's somebody -- staff wants to say --

MS. HAYES: This is Michelle Hayes. And I think that paragraph that you're referring to is we're trying to say there's more uncertainties than they included in their parametric uncertainties and that even given model uncertainties that haven't been identified yet, there's still a large margin.

Your point about not going further into the 90.55 percent, we can take that back and come back to you on that. I think it has to do with the large margin and that our finding is that it meets the Commission goals. We're not certifying every number that they provide. We're making sure they follow the process and they meet the Commission goals and we did that with -- while recognizing there's a large margin even given this extreme CCF sensitivity study.

CO-CHAIR CORRADINI: I think she's agreeing with you.

CO-CHAIR DIMITRIJEVIC: I think you don't want to say we have a zero risk and we know this very well. Yes.

MS. HAYES: We are not saying that, but we are saying it's the same as the safety goals and we know that very well.

CO-CHAIR DIMITRIJEVIC: All right. Okay. We meet the safety goals and we know that very well, that's true.

MS. HAYES: That's what we're trying to say.

CO-CHAIR CORRADINI: Other comments by the committee? I want to get public comments if we could before we go into closed session.

Okay, so as we open up the phones lines and people can speak from the phone, is anybody in the room that wants to make a comment? If not, I have my wanted helper here, Mr. Snodderly.

MR. SNODDERLY: Is there anyone on the public line? Good afternoon, Marvin. Would you like to make a comment?

MR. LEWIS: Not at this time. I'm torn. Yes, I would like to make a comment.

MR. SNODDERLY: Please do.

MR. LEWIS: Respectfully. Respectfully. I'm trying to be most respectful, but I'm having a very difficult time of it.

The reason is this. You're talking probabilistic. Yes, there is such history as probability. However, I must remind you Three Mile Island, 1979; Fukushima, a few years ago. That day when those accidents happened, probabilistically it was a hundred percent because it did happen and you refuse to see that there is a glitch in your probabilistic assessment and analysis where it actually happens, the accident actually happens and has happened. Thank you.

MR. SNODDERLY: So Marvin, I appreciate that comment and I encourage you on June 18th and 19th, we'll be reviewing Chapter 6 and 15 that deal more with the deterministic in a given design-basis accident and it complements the probabilistic. So hopefully, you'll be able to join us for that and again, we appreciate your comment.

Are there any other -- is there anyone else on the public line who would like to make a comment?

MS. FIELDS: Yes. This is Sarah Fields. One of my comments is it was very hard to hear some of the speakers. There's that difficulty. And then in one of the discussions appeared to be disconnected from the slide presentations that were sent out. So it was -- you kind of get lost because you think that the NRC staff is reading from something, but whatever they're reading from is not part of the slide presentation and for someone who is not a technical person, it's very easy to get lost and I appreciate that you will be putting a transcript up as soon as possible. It would be very useful to get a handle on what the full discussion was about.

MR. SNODDERLY: So Sarah, this is Mike Snodderly. I appreciate your comment and you are right. We could -- we'll make an announcement to make sure you are aware of what slide set we're on. So I understand because I sent you I think three sets of slides, so it was probably hard for you to figure out what slide set we were on.

MS. FIELDS: Not only the sets, but particularly where you think you're on the right page and you are, and then it goes to discussion, but the discussion isn't exactly what's on the next page. There's just so much more coming from the staff that is not really reflected on the slides. I mean I know they're hard to put together and they have a lot to say and the discussion will go this way and that. So the transcript will be important. Thank you.

MR. SNODDERLY: You're welcome. Thank you for your time.

Are there any other comments from people on the line?

CO-CHAIR CORRADINI: Okay, thank you very much. Let's close the public line and we're going to take a break and go into closed session. You want to do that? I'm sorry.

MEMBER BLEY: There's nobody there.

CO-CHAIR CORRADINI: Why don't we take a break and we'll come back. Fifteen minute break at 25 of.

MR. SNODDERLY: And we'll be in proprietary closed session so we close this line, and clear --- make sure. So anyone that returns afterwards, you'll have to be a member of NuScale or you have to be a member of the staff or a need to know and there will be no members of the public.

(Whereupon, the above-entitled matter went off the record at 3:17 p.m.)

May 09, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Presentation 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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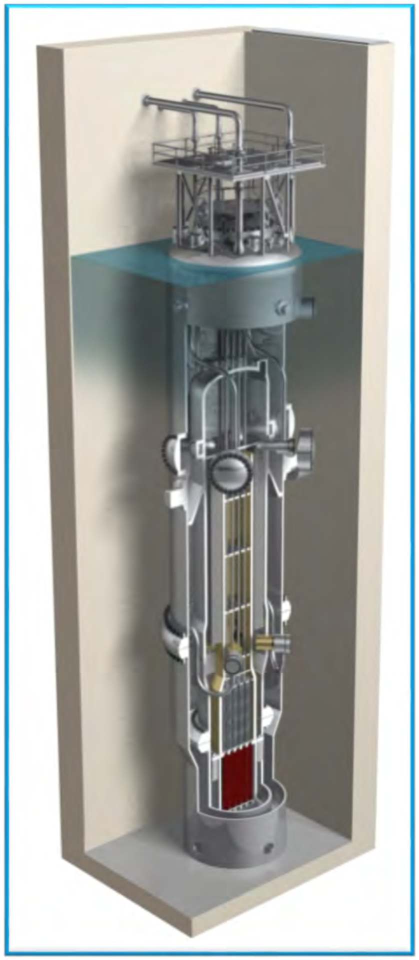
Enclosure 1: "ACRS Subcommittee Presentation: NuScale FSAR Chapter 19, Probabilistic Risk Assessment and Severe Accident Evaluation," PM-0519-65372, Revision 0

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

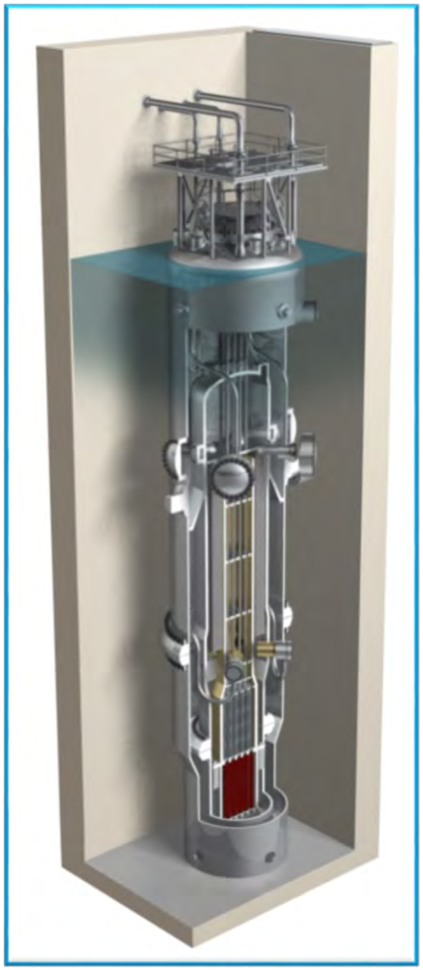
May 14 and 15, 2019

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

Presentation Team

Sarah Bristol

Supervisor, Probabilistic Risk Assessment

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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

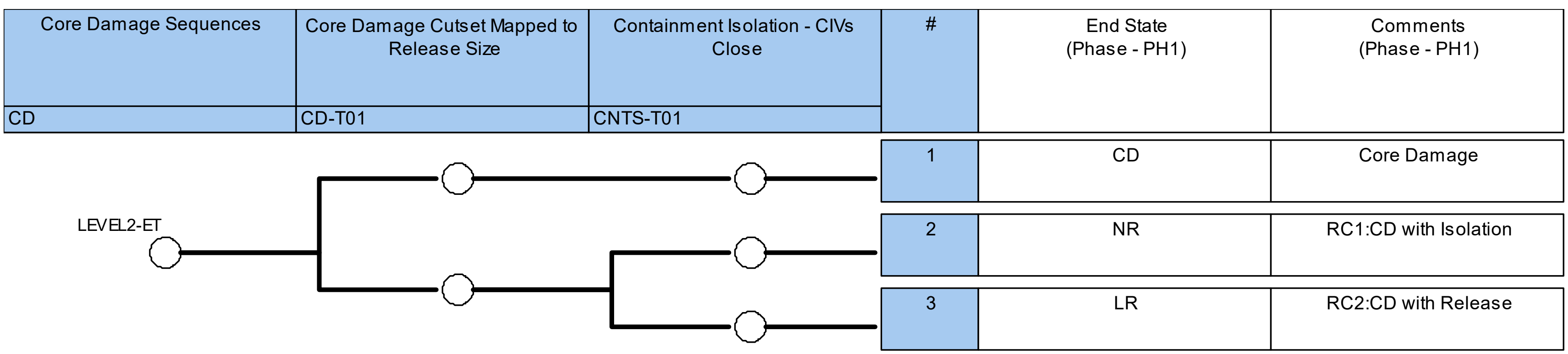
- Systems
  - CNTS (containment isolation valves), ECCS, MPS, and UHS
- Components
  - ECCS RVVs and RRVs
  - DHRS actuation valves
  - RSVs
  - CVCS and CES containment isolation valves
  - Combustion turbine generator
- Other events and initiators (FV>20%)
  - RBC, LOCA inside CNV, LOCA outside CNV, 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

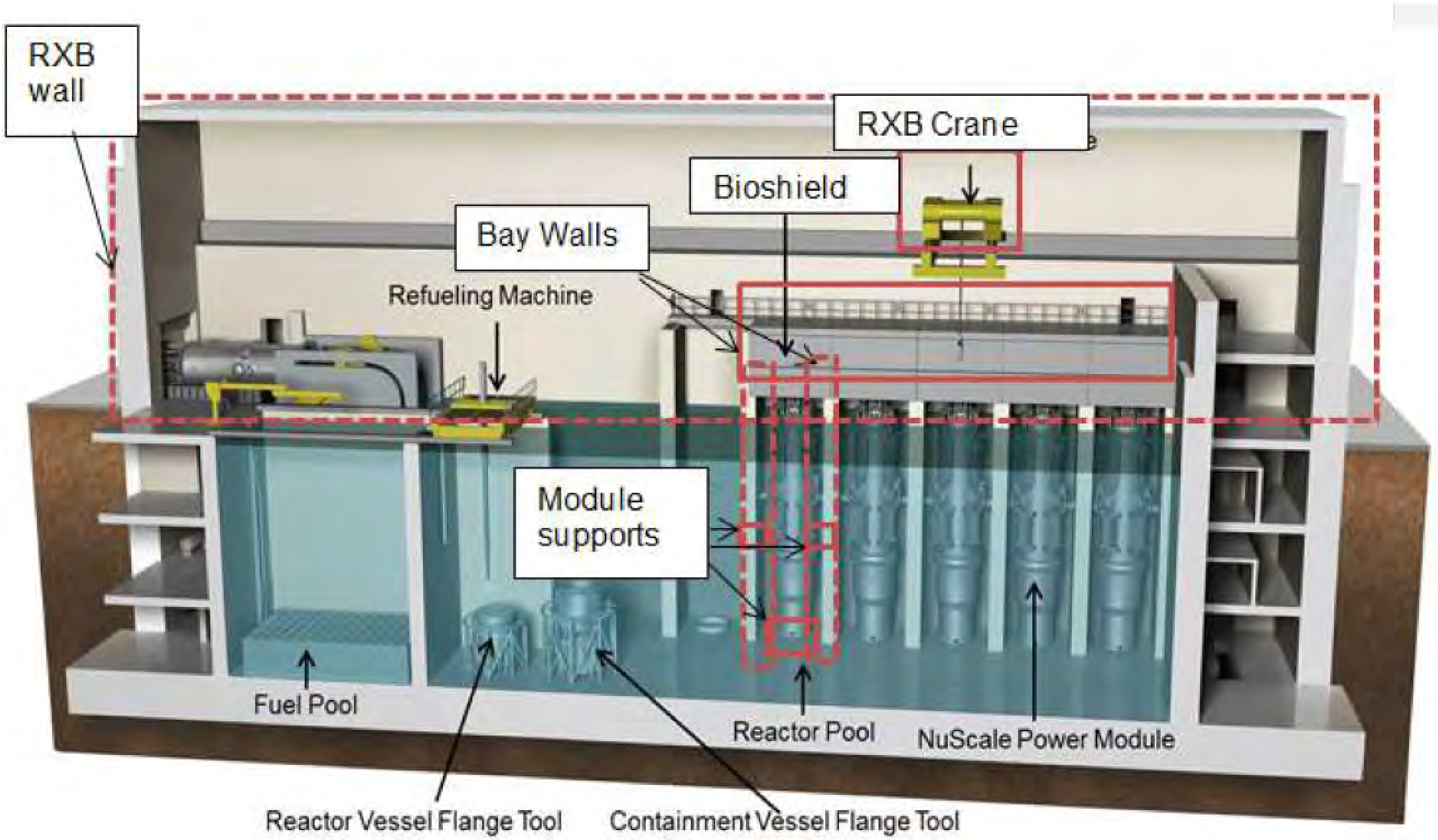
Seismic plant response

- 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.

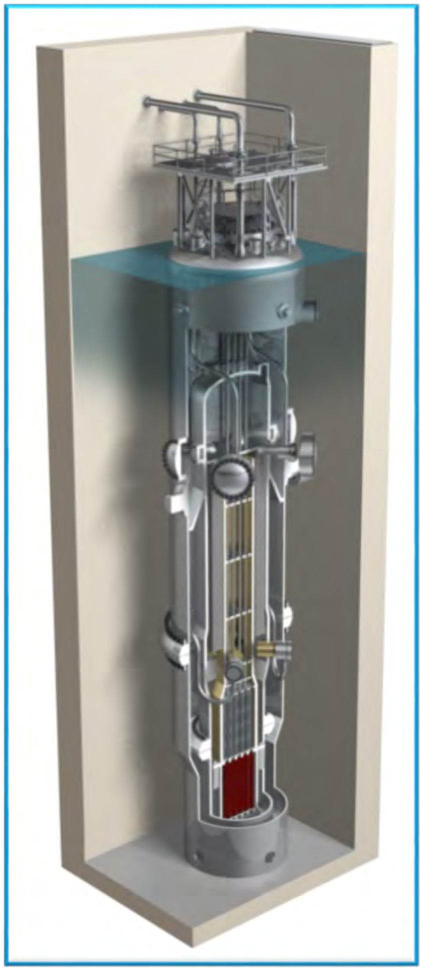
Section 19.1 COL Items

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.

ACRS Subcommittee Presentation:



NuScale FSAR Chapter 19.2

Severe Accident Evaluation

May 15, 2019

PM-0519-65372
Revision: 0

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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.

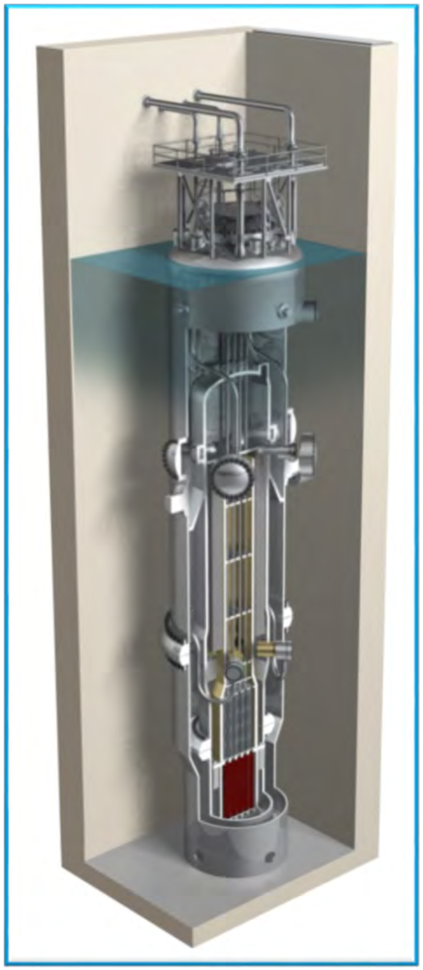
Level 2 Key Insights (5 of 5)

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

May 15, 2019

PM-0519-65372
Revision: 0

Template #: 0000-21727-F01 R5

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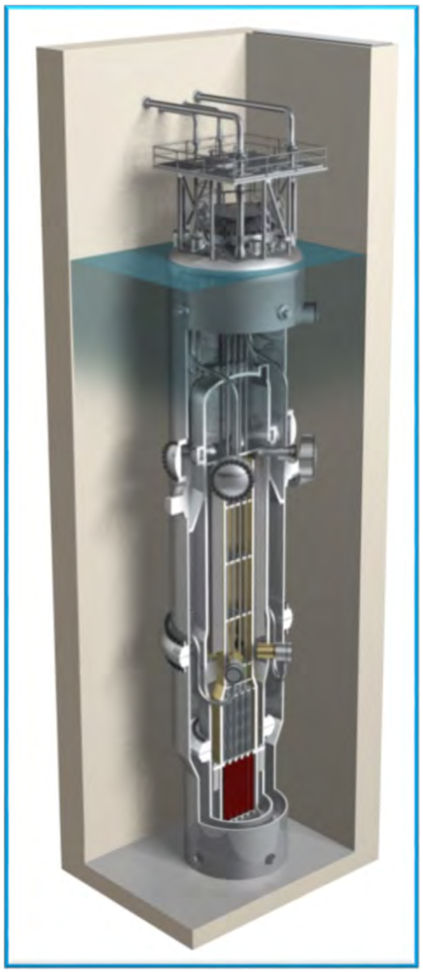
Section 19.3

- 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

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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

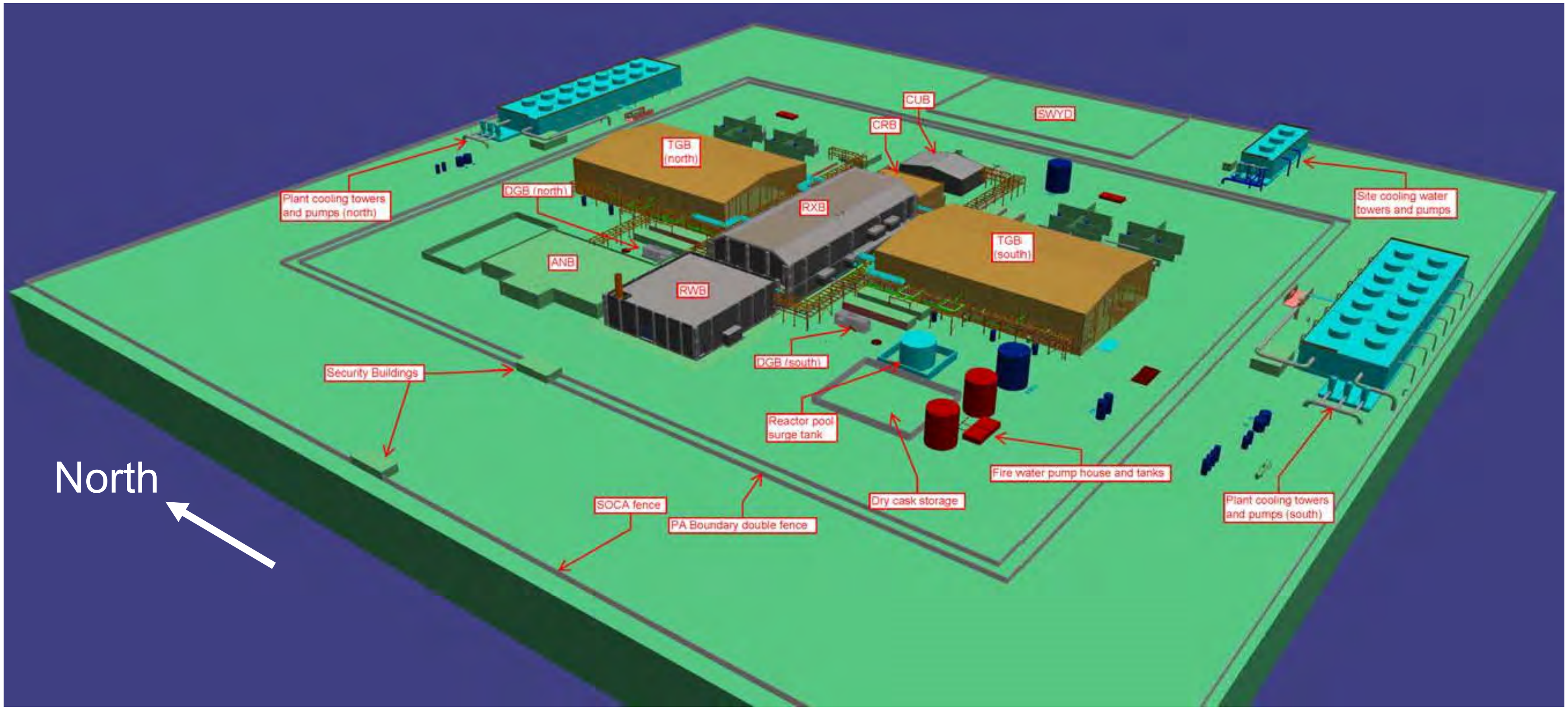
Assessment Scope

- 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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United States Nuclear Regulatory Commission

Protecting People and the Environment

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

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Section 19.1
Probabilistic Risk Assessment

Staff's Review - Overview

- SER is based on DCA Revision 2
- Staff conducted two regulatory audits
- Applied the enhanced safety focused review approach to:
  - Support integrated decision making
  - Increase focus on safety for effectiveness and efficiency of the review

Staff's Review - Overview (Continued)

- Reviewed the quality, completeness, and consistency of the information in the DCA
- Paid an increased attention to key assumptions
- Review guided by the Commission goals for CDF, LRF, conditional containment failure probability, and PRA insights

Overview of Review Guidance

- SRP 19.0 – Staff guidance, including acceptance criteria, for PRA and severe accident evaluation

- DC/COL-ISG-28 – Guidance on the use of ASME/ANS PRA Standard (as endorsed by RG 1.200) for a DCA

At-Power Internal Events Level 1 PRA

Data Sources

- Basic event data
  - PWR generic failure probabilities are reasonable for the DCA stage
- NuScale unique components
  - Failure rates and probabilities are key assumptions
  - Sensitivity studies using conservative assumptions for component failure rates show results that compare favorably with the Commission's CDF and LRF goals

At-Power Internal Events Level 1 PRA

Passive System Reliability Evaluation

- Staff audited the inputs for NRELAP5 thermal-hydraulic simulations

- Applicant adequately considered effect of passive system reliability for the DHRS and ECCS consistent with the level of detail at the DCA stage

At-Power Internal Events Level 1 PRA

Event Tree – LOCA inside containment

- NuScale assumes containment isolation is not necessary for LOCAs inside containment (Open Item: RAI 8840, Question 19-2)

- Staff audited NuScale's analysis

- Staff is evaluating the NuScale response

At-Power Internal Events Level 1 PRA

ATWS Sequences

- SER Chapter 7 documents staff approval of ATWS exemption request for 10 CFR 50.62(c)(1), acceptability of the module protection system design, and the reduction of ATWS risk below the Commission CDF goal of 1E-5 per year

Non-Proprietary

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Office of Nuclear Regulatory Research TRACE Confirmatory Analysis of Anticipated Transients without SCRAM for NuScale Power Module

NuScale Design Certification Application

ACRS Subcommittee Meeting

May 15, 2019

TRACE Confirmatory Calculations

- The staff performed confirmatory calculations for ATWS using TRACE.
- Key figures of merit:
  - Reactor pressure vessel (RPV) pressure to confirm RPV integrity, and
  - Riser collapsed liquid water level above the top of active fuel (TAF) to confirm core coolability.

Base Case Scenario

Evaluated Event Progression for Margins to RPV Pressure and Level Criteria

- Initiated by LOAC which leads to immediate turbine and feedwater trips.
- Control rods fail to insert and remain withdrawn for the full transient.
- RPV pressure increases as reactor coolant system heats up.
- High RPV pressure triggers decay heat removal system (DHRS) actuation.
- Higher RPV pressure initiates reactor safety valve (RSV) cycling.

Key Alternate Scenarios Analyzed

Staff Analyzed Several Cases in Addition to the Base Case

- BOC: Like the base case but with BOC kinetics parameters.
- 1RSV: Like the base case but with RSV-1 out-of-service (OOS).
- PRA-1RSV: Like BOC case but with RSV-1 OOS and DHRS OOS.
- SGHT: Like base case but RCS initial temperature is much higher.

TRACE Results

Demonstrated Large Margins in All Cases

- Base/BOC/1RSV Cases: Large RPV peak pressure margin. In the long term, reactor power is balanced by DHRS at low levels and riser level remains well above the top of active fuel (TAF).
- PRA-1RSV Case: Large RPV peak pressure margin. In the long term containment (CNV) inventory increases and core power is balanced by CNV heat removal. Level remains well above TAF.
- SGHT Case: Large RPV peak pressure margin. In the long term, RPV level drops to top of the riser but well above TAF. Core power balanced by DHRS and CNV heat removal.

At-Power Internal Events Level 2 PRA

- Two Containment Event Tree end states
- Other severe accident phenomena are addressed in the presentation on Section 19.2

LPSD Internal Events Level 1 PRA

Single Module Drop

- Staff audited Reactor Building Crane PRA notebook
- Staff reviewed NUREGs on load drops, EPRI PRA on cask drops, and recent events to evaluate drop probability
- NuScale committed to: NUREG-0554 as supplemented by ASME NOG-1 - single failure proof – consistent with operating plants

LPSD Internal Events Level 1 PRA

Single Module Drop (continued)

- Risk significance of Reactor Building Crane resulted in additional ITAACs:
  - Rated load test and inspection of NuScale power module lifting fixture and module lifting adapter
- Analysis consistent with SRP Section 19.0 and DC/COL-ISG-028

LPSD External Events

Single Module Drop

- Reactor Building Crane design adequately considers loss of AC power due to external flooding and high winds

- Module drop analysis consistent with SRP Section 19.0 and DC/COL-ISG-028

Multi-module Risk

Overview

- Systematic process is used to evaluate multi-module risk
- Approach relies on assumptions made based on engineering judgement
- Design relies on independent, module-specific safety-related systems to prevent and mitigate core damage
  - DHRS, ECCS, CIVs
- Impact of external events is addressed qualitatively

Multi-module Risk
Module Drop Event

- Module dropped during refueling can impact up to two operating modules
  (Open Item: RAI 9659, Question 19-39)

- Staff is evaluating NuScale's analysis of this event

External Flooding and High Winds Analyses

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- For the external flooding PRA, staff finds the applicant's approach reasonable

- For the high winds analysis, staff verified that all important accident mitigation features are housed within seismic Category I reactor building structure and are protected from the effects of high winds.

- Staff finds these external hazard analyses sufficiently consistent with DC/COL-ISG-028 and SRP Section 19.0.

At-Power Internal Fire and Internal Flood PRAs

- Staff focused its review on the appropriateness of assumptions used to address incomplete aspects of the design and operating procedures

- Staff finds:
  - the estimated risk is reasonable for the DCA stage
  - the internal fire and internal flood PRAs for at-power and LPSD operations are sufficiently consistent with DC/COL-ISG-028 and SRP Section 19.0

PRA-based SMA

- Staff focused its review on the scope of SSCs included in the fragility evaluation and the analysis methods used to determine seismic fragility

- Staff finds the plant-level HCLPF capacity demonstrates adequate margin in accordance with SRP Section 19.0

Summary
PRA

- Due to the open items, the staff cannot make a finding on the PRA description in DCA Part 2, Tier 2, Sections 19.0 and 19.1
  - RAIs 8840 and 9659

Abbreviations and Acronyms

- **AC** – alternating current
- **ANS –** American Nuclear Society
- **ASME –** American Society of Mechanical Engineers
- **ATWS** – anticipated transient without scram
- **CDF –** core damage frequency
- **CIV** – containment isolation valve
- **COL –** combined license
- **DC –** design certification
- **DCA** – design certification application
- **DHRS** – decay heat removal system
- **ECCS** – emergency core cooling system
- **HCLPF –**high confidence low probability of failure
- **ISG** – Interim Staff Guidance
- **ITAAC –** inspections, tests, analyses, and acceptance criteria
- **LOCA** – loss-of-coolant accident
- **LPSD –** low power and shutdown
- **LRF** – large release frequency
- **PRA –** probabilistic risk assessment
- **PWR** – Pressurized water reactor
- **RAI** – request for additional information
- **RG** – Regulatory Guide
- **SER** – safety evaluation report
- **SMA** – seismic margin assessment
- **SRP –** standard review plan
- **SSCs** – structures, systems, and components

Questions/comments from members of the public before the closed session starts?