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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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NuSCALE SUBCOMMITTEE

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WEDNESDAY

FEBRUARY 19, 2020

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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., Walter L. Kirchner and Jose March-Leuba, Co-Chairs, presiding.

COMMITTEE MEMBERS:

WALTER L. KIRCHNER, Co-Chair

JOSE MARCH-LEUBA, Co-Chair

RONALD G. BALLINGER, Member

DENNIS BLEY, Member

CHARLES H. BROWN, JR., Member

VESNA B. DIMITRIJEVIC, Member

DAVID PETTI, Member

JOY L. REMPE, Member

PETER RICCARDELLA, Member

MATTHEW W. SUNSERI, Member

ACRS CONSULTANTS:

MICHAEL L. CORRADINI

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

CHRISTOPHER BROWN

MIKE SNODDERLY

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P R O C E E D I N G S

8:30 a.m.

CO-CHAIR KIRCHNER: This meeting will now come to order. Good morning.

This is a meeting of the Advisory Committee on Reactor Safeguards NuScale Subcommittee. I am Walt Kirchner, Chairman of the NuScale Subcommittee. My colleague, Jose March-Leuba, will co-chair this meeting with me today.

Members in attendance today are Ron Ballinger, David Petti, Joy Rempe, Jose March-Leuba, Pete Riccardella, Charles Brown, Dennis Bley, Vesna Dimitrijevic. Sorry. And we also have --

PARTICIPANT: That's easy for you to say.

CO-CHAIR KIRCHNER: Yeah. We have two distinguished consultants with us today, our former chairman, Mike Corradini, and Stephen Schultz. Mike Snodderly and Christopher Brown are the designated federal officials for this meeting.

The subcommittee will be briefed on loss-of-coolant accident evaluation model, non-loss-of-coolant accident analysis methodology, and rod ejection accident methodology. Today we have members of the NRC staff and NuScale to brief the subcommittee.

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

The committee can only speak through its published letter reports. We hold meetings to gather information and perform preparatory work that will support our deliberations at a full committee meeting.

The rules for participation in all ACRS meetings were announced in the Federal Register on June 13, 2019.

The ACRS section of the U.S. NRC public website provides our charter, bylaws, agendas, letter reports, and full transcripts of all full and subcommittee meetings, including slides presented there. The meeting notice and agenda for this meeting were posted there.

Portions of this meeting can be closed as needed to protect proprietary information pursuant to 5 USC 552(b)(c)(4).

As stated in the Federal Register notice and in the public meeting notice posted to the website, members of the public who desire to provide written or oral input to this subcommittee may do so

and should contact the designated federal official five days prior to the meeting as practicable.

We have also set aside ten minutes for comments from members of the public attending or listening to our meetings. We have not received written comments or requests for time to make oral statements for members of the public regarding today's meeting.

A transcript of the meeting is being kept and will be made available on the ACRS section of the U.S. NRC public website.

We request that participants in this meeting please use the microphones located throughout the meeting room when addressing the subcommittee. Participants should first identify themselves and speak with enough volume and clarity so that they can be readily heard.

A telephone bridge line has been established for the public to listen to the meeting. To minimize disturbance, the public line will be kept in a listen-in-only mode.

To avoid further disturbances, I request that all attendees put their electronic devices, like cell phones, in the off or noise-free mode.

And now we'll proceed with the meeting.

I'll just make one quick comment.

Given the sensitive nature of a lot of the information, which requires us to access it only through NRC laptops and approved devices, not getting the slide information in advance causes problems, because for many of us if we're traveling, we can't use our NRC laptops while we're traveling. And so I'd just make that observation for the record.

And now I'll turn to my co-chair, Jose March-Leuba.

CO-CHAIR MARCH-LEUBA: So thanks, Walt. And following up on that, we have a meeting March 2nd. If we could get their slides by Friday instead of Sunday night, it would be greatly appreciated.

CO-CHAIR KIRCHNER: Yes.

CO-CHAIR MARCH-LEUBA: All right. So, that said, we have an unusual, not unheard of, but unusual situation. We have a subcommittee meeting in which we are reviewing three topical reports. We typically only have one subcommittee for every topical report.

So we are going to go through all of the open session for all three topical reports. And then we'll move into closed session for the rest of them. Okay.

We know that is in the agenda. I just wanted to make sure you understand that when we finish the review of the first one in the open session, we have not finished the review. We will pick up this afternoon.

On that topic, we have a reserved time tomorrow morning to follow up, continue whatever we don't cover in the closed session. By 4:00 or 4:30, we'll see where we are. We may not have to meet tomorrow. It depends on how quiet the members are. And my bet will be we'll be here tomorrow, but just so you know.

Another thing that makes this situation unusual is these topical reports describe the methodology that has been used to analyze the NuScale reactor, Chapter 15 and Chapter 19 and all the other.

And that was completed, the review from the staff was completed on December of last year. And we are now reviewing these topical reports at the ACRS level. I wanted to make sure that the staff is, to give time to the staff to address this situation, which is clearly not desirable.

And my point is we have a good excuse this time. But this should not become a precedent that every single design certification that comes here

waits until everything is done to certify, to approve the methodology, because if we find even a minor detail on this modification on your methodology, the only recourse is for us to tell you to rerun all your Chapter 15 analysis again.

So you're putting the ACRS in a very difficult situation because it is a big step that -- I mean, for any comment that we make on your methodology, it has to raise to a tremendous significance to make you do a lot of work and a big delay. Okay.

So right now this is a member's comment. And I hope to convince my colleagues to write in the letter that this, while acceptable this time, if we find it acceptable this time, it should not make a precedence for future certifications. Just for the members keep that in mind, because I will propose that to be in the letter during our full committee meeting.

Another question I got from Chris was we will make comments about what we want to hear in the full committee meeting, which is scheduled for March, at the end of all the presentations.

So we will have comments on all three topical reports, so take notes if you want. What you want to see in the topical report full committee

meeting next month, take notes and we will cover that at the end.

MEMBER KIRCHNER: And with that, we'll turn to Becky.

MS. PATTON: Thank you. I'd just like to start by thanking the committee for their time reviewing these topical reports.

As you're aware, these are three important methodology topical reports for the NuScale design. These are referenced in the design certification application, FSAR Section 15.0.2.

These topical reports were accepted for review by the staff back on April 27, 2017. And the staff had accepted the NuScale DCA for review on March 23, 2017.

The three methodologies were reviewed in parallel with the staff's Chapter 15 review, although this is not the ideal situation, that being when the methodology is reviewed and approved in advance of the methodology being applied to the design. However, the staff did work with NuScale to accommodate this situation.

At the Chapter 15 Phase 2 ACRS meetings last year, the methodologies were open items, although at the time most of the methodology issues had

actually reach resolution. Some, like the update of NRELAP Code to Version 1.4, were ongoing.

In the fall of last year, the technical issues with the topical reports were all closed. But the safety evaluation reports were in various stages of editing, writing, and concurrence.

So what we decided was to ensure a close focus on closing all of the technical issues with the design, including those on the design certification application, the office prioritized closure of all technical issues above writing, editing, and concurrence.

All technical issues were closed prior to the office issuing the Chapter 15 Phase 4 SER in mid-December. Since the technical issues with the methodologies were closed, the office strategy was to defer the final writing, editing, and concurrence for the topical report SERs until January.

In the Phase 4 SERs, with the, the issuance of the methodology SERs is left as confirmatory items because the topical report SERs hadn't been issued. The approved versions of the topical reports are going to be referenced as part of the Phase 6 SER.

So I just wanted to lay that out as kind

of the process and the strategy that we had used and why these are being presented now.

CO-CHAIR MARCH-LEUBA: So are you implying that all the technical review had been completed --

MS. PATTON: That's correct.

CO-CHAIR MARCH-LEUBA: -- prior to the issuing of SER?

MS. PATTON: That's correct. Before we issued the Phase 4 SER for Chapter 15, all of the technical issues, including those on the methodology, had been closed.

CO-CHAIR MARCH-LEUBA: And the reason for the delay is a technicality and going through the lawyers, all the good that you have to go --

MS. PATTON: Right.

CO-CHAIR MARCH-LEUBA: -- SER.

MS. PATTON: Yeah, and right. And there were some portions where the staff hadn't finished the writing yet, but they had, you know, accepted the RAI responses and the closing of the technical issues.

CO-CHAIR MARCH-LEUBA: Okay. So, again, one member's opinion, I think this was an unacceptable approach for this time. I'm just worried that this becomes the precedent, and everybody is going to be doing the same thing over and over again.

We should be at least, tell the new submittals, not NuScale, but new reactors that our expectation is to finish the review before we do the review of the Chapter 15, 19, and others.

MS. PATTON: We understand your comment.

CO-CHAIR KIRCHNER: Okay. Now we'll turn to NuScale. Matthew, do you want to start? Please, go ahead.

MR. PRESSON: Thank you very much. I'm Matthew Presson, licensing project manager with NuScale. And do I need to get --

CO-CHAIR MARCH-LEUBA: The court reporter can't hear you. Can you? Yeah, so it's -- the room might be a little low.

(Off-microphone comments.)

MR. PRESSON: Yeah. And as you've discussed, we will be going over several topical reports today. We will be starting with the rod ejection accident methodology. And up here is Kenny Anderson, our nuclear fuels analyst.

There's a quick agenda, and there we go. Just to give a quick overview and kind of scope our discussions today, we wanted to provide this slide to kind of show you where it all fits together.

So we have NRELAP5, which is developed

from RELAP5-3D. A lot of that discussion of that development is in the LOCA method, at least it's been on more and non-LOCA method. And we also have our control rod ejection methods.

So those are the -- up at the top in the image, we have NRELAP5. We have our topical report, LOCA, in blue. In brown beneath that, we have our control rod ejection. And then in purple, we have the non-LOCA methods.

So, kind of branching off from that, we have our technical reports, which go into Chapter 15 which will be discussed two weeks from now. But this is kind of the basis of --

CO-CHAIR MARCH-LEUBA: For those of us that have short memory retention --

MR. PRESSON: Yeah.

CO-CHAIR MARCH-LEUBA: -- can you describe the difference between a topical and a technical report?

MR. PRESSON: Yeah, so the topical reports are stand-alone licensing documents. We'd submit them. They provide a method, for example, for LOCA, how we would handle a loss-of-coolant accident. The way that it is applied within the FSAR, we will have FSAR-specific events.

And then for some of our other branching analyses, such as containment response, we have a technical report within the FSAR, as we did not have a separate method going through a separate approval process for it as these topical reports --

CO-CHAIR MARCH-LEUBA: The technical report does not have an SER associated with it.

MR. PRESSON: Correct.

CO-CHAIR MARCH-LEUBA: The SER for Chapter 15 or whatever it applies is the one that approves the technical report --

MR. PRESSON: Correct.

CO-CHAIR MARCH-LEUBA: -- as if it was a chapter of the FSAR.

MR. PRESSON: Yeah, the technical reports are essentially part of the chapter.

CO-CHAIR MARCH-LEUBA: Call it an appendix of the SER.

MR. PRESSON: Yes.

CO-CHAIR MARCH-LEUBA: Okay.

MR. PRESSON: So, you're going into some of the more technical details. So these topical reports are stand-alone. They are separate approvals. We do use those methods in the FSAR. But we get to, the ACRS gets to review and offer their comments on

those separately here at this meeting. So --

CO-CHAIR MARCH-LEUBA: Okay.

MR. PRESSON: All right. And with that, I will pass this over to Kenny.

MR. ANDERSON: Good morning. Thank you. I'm Kenny Anderson from NuScale. I will be talking a brief overview of the rod ejection event in this open session.

So rod ejection is very unique. And that's why it has its own topical. And the rest of the day we'll be talking about the other Chapter 15 events.

And rod ejection is one that is really dominated by the nuclear physics. And so I have kind of a comparison table to kind of stress how or illustrate how unique it is. And that's why we chose to focus in one topical.

The goal of doing the analysis is to satisfy essentially General Design Criteria 28. And we'll talk about that.

Yeah, and then the other thing, too, is that it does have very unique acceptance criteria. This is a comparison of the acceptance criteria and whether it's unique to other Chapter 15 events. And that's the last column.

And the ones that say no, those are the same as other Chapter 15 events, for example, coolant, high coolant system pressure or critical heat flux.

The ones that are yes, those are specifically related to energy deposition in the fuel. And that's to protect the fuel from either dispersal or at core coolability.

And then we'll go a little bit more on these, on the different fuel criteria. So, over time, the acceptance criteria for fuel has evolved. And we just wanted to kind of explain where we are with that.

And this first figure is from NUREG-800. And that's the criteria we use currently in our topical report. And you can kind of, and the dashed lines, you can kind of get a feel of where NuScale is and related to their, to our corrosion rate. So we fit within that box.

In the future, Draft Guide 1327, the acceptance criteria, we do expect to change. And you can see it's, they look pretty similar figures. The key difference is on the X-axis. There's -- the units are different. It's hydrogen uptake. And as that becomes law, we expect to have the methodology and support that as well.

CO-CHAIR MARCH-LEUBA: Kenny, on the left

figure, the dotted line, the NuScale criteria, is that a fixed criteria or it's a goal that you are --

MR. ANDERSON: That's our, that's what we propose to be our criteria in the topical report. So, instead of using the more complicated shape, we say a max of 75 calories per gram. And then that way we don't do an exposure census to ensure that we're within the more complicated shape.

CO-CHAIR MARCH-LEUBA: And can you give us a preview of the Chapter 15 results? Where do you fit? Where does the real NuScale fit in that criteria? Can you point with the mouse more or less where --

MR. ANDERSON: Sure. Yeah, just in this box here is where we fit. And we'll see --

CO-CHAIR MARCH-LEUBA: Yeah, but how close to the, how close are you to limits?

MR. ANDERSON: There's a lot of margin. And in the closed session, we have exact numbers and --

CO-CHAIR MARCH-LEUBA: Oh, okay.

MR. ANDERSON: -- the exact margin.

DR. CORRADINI: That's why we don't see it here.

MR. ANDERSON: Yes.

DR. CORRADINI: Okay. That's what I was waiting for you to say. Thank you.

MEMBER PETTI: Just a question, and maybe you'll do it in the closed session. What would the dotted line --

CO-CHAIR MARCH-LEUBA: Do you have the microphone?

MEMBER PETTI: What would the dotted lines look like on the Draft Reg Guide?

MR. ANDERSON: Yeah, they would, we would still have the 75 here. And then M5 is our current fuel. And it has very low corrosion. So I would expect it to be in the less than the 50 over here on DPM.

DR. CORRADINI: But I guess since you brought this slide up, I will ask the question and then somebody else who understands materials better than I. But now there's a disconnect. When you showed an M5 reserve load, there is not a one-to-one -- the one-to-one correspondence between Zircaloy uptake of oxygen and hydrogen have -- there is a disconnect.

So does hydrogen pick-up now dominate the issue, and that's why you're changing the X-axis on the right-hand side?

MR. ANDERSON: Yes. So that was a -- it's more than a unit conversion. It's a --

DR. CORRADINI: Yeah, yeah, yeah.

MR. ANDERSON: It's the physics behind it.

DR. CORRADINI: Yeah, I understand that. But then --

MR. ANDERSON: Right. Yeah, but it's generally correlated to exposure.

DR. CORRADINI: Okay. And so the exposure that you kind of wiggled around is related to how much burn up?

MR. ANDERSON: In our current design roughly --

DR. CORRADINI: If you're allowed to say here.

MR. ANDERSON: Yeah, that's fine. Roughly 30 to 40 of the gigawatt --

DR. CORRADINI: Okay.

MR. ANDERSON: -- metric ton.

DR. CORRADINI: All right. Thank you.

DR. SCHULTZ: Kenny, how does that -- on the left-hand graph, you have the maximum ratio per the corrosion limit. How has the corrosion limit been set? And how do you monitor it? How do you propose to monitor it, the corrosion?

MR. ANDERSON: Yes. We do have an analysis from our fuel vendor, Framatome, that does perform that calculation. They have a limit of 100 microns, thank you. And our analysis shows that, under best estimate normal operating conditions, we're roughly at 30 in those units.

DR. SCHULTZ: And you'll monitor that with cycle by cycle inspection of the fuel?

MR. ANDERSON: Yes, that's -- I'd have to defer to my colleagues to -- but we can come back with that exact answer.

DR. SCHULTZ: That would be great. Thank you.

CO-CHAIR MARCH-LEUBA: And one thing I read somewhere, and I'm going to say something that it's not supposed to say. I'm not an expert on this.

Because we run in natural circulation and very low flow velocities, the heat transfer coefficient out of the clad is higher than normal in the operating fleet. And therefore, the clad is operating hotter than normal in the operating fleet.

Does the higher temperature of the clad affect this? And do we have sufficient data to go over that operating condition?

MR. ANDERSON: Yeah, that's a good

question. I'd want to defer to the --

CO-CHAIR MARCH-LEUBA: Please keep us on this.

MR. ANDERSON: Yeah.

CO-CHAIR MARCH-LEUBA: We'll come, I'll make a note to come back this afternoon.

MR. ANDERSON: Okay. Sounds good.

DR. SCHULTZ: From the documentation, Kenny, it said that you don't have the information related to hydrogen pick-up at this point in time. Does --

MR. ANDERSON: Yes, well, we --

DR. SCHULTZ: Does Areva have it? Or --

MR. ANDERSON: Well, yeah --

DR. SCHULTZ: -- they must be looking at it for their own future work.

MR. ANDERSON: Sorry to interrupt you. Yes, we're working on, with Areva or Framatome, our partner, to have an official method for calculating it. So it's, we can do it.

DR. SCHULTZ: You haven't gotten there yet.

MR. ANDERSON: We just haven't proven it yet. And we're working to do that.

DR. SCHULTZ: Do you anticipate that your

criterion box is going to be appropriately set below the curve on the left, right-hand side?

MR. ANDERSON: Yes, for many reasons, including the low corrosion performance of M5.

DR. SCHULTZ: All right. Thank you.

MR. ANDERSON: Okay. So, to analyze the event, unlike the other Chapter 15 events, we have a multi, a unique multi-step process that is a -- we usually rely on the nuclear physics code as the, as that comparison table showed. And this is an event dominated by nuclear physics.

So that SIMULATE5 is a steady state nuclear physics analysis that sets the initial conditions up to make sure we have the full spectrum of conditions.

We then pass that to the transient kinetics code, SIMULATE-3K. And that calculates the power pulse as a function of time, which is kind of the workhorse or the main item in this event that then feeds the other analysis paths.

So, as we'll talk about, the one analysis path is to NRELAP5 to do the system pressurization. Then in the middle path, we pass the pressure pulse or the power pulse to end the conditions calculated by RELAP5 to our subchannel analysis code, VIPRE, to do

the official CHF evaluation.

And then there's also a path at the bottom where, to calculate the fuel enthalpy criteria. We do essentially a hand calculation that calculates and checks those various energies deposited in the fuel.

MEMBER REMPE: Can --

MR. ANDERSON: And I'll talk about each one of those.

MEMBER REMPE: I'm sorry to interrupt you. But I forgot at the beginning of this session that I was supposed to declare that I might have some conflict of interest. And I'm going to have to limit my discussion on this topic. Okay? Thank you.

MEMBER BLEY: Are you going to talk about the actual physical event that starts the rod ejection?

MR. ANDERSON: No, unless, not really unless you'd like to talk about it.

MEMBER BLEY: Well --

MR. ANDERSON: It's generally a postulated.

MEMBER BLEY: What I want to ask you is it's postulated to happen like that, right?

MR. ANDERSON: Yeah, effectively the -- we would --

MEMBER BLEY: What if it's a little slower?

MR. ANDERSON: Then it's generally --

MEMBER BLEY: Is it better if it's slower?

MR. ANDERSON: Yes, yes.

MEMBER BLEY: In all cases?

MR. ANDERSON: Yes.

MEMBER BLEY: You've looked enough to be confident of that.

MR. ANDERSON: Yes.

MEMBER BLEY: Okay.

MR. ANDERSON: What makes this event unique from all the other Chapter 15 events is its, how rapid it is. And so the key is understanding to make sure you have that power pulse, and then is that power pulse high enough to basically create a water hammer, which would break the, or it would violate the criteria of reactor coolant pressurization.

And then the other aspects of it is, is that power pulse high enough to deposit a lot of energy in the fuel and have the fuel either fail and have that released radiation that would violate the limit, or would it damage the fuel such that it was no longer coolable.

So the, yeah, so the focus of this event

is, is it rapid and qualifying that. And then other events, if it's just slower, we'll talk about where we can see that. It's basically benign.

CO-CHAIR KIRCHNER: Follow up on that, so just for the record. When you do the RELAP analysis, you don't take any credit for the rupture of the control rod drive mechanism, right? You just, you essentially are doing the analysis as if the system is intact to get your peak pressure.

MR. ANDERSON: Yeah, yes.

CO-CHAIR MARCH-LEUBA: And while we have you interrupted, on the non-LOCA methodology, which also uses RELAP on VIPRE, we have some criteria, a screening criteria, for RELAP to determine whether we have to go on VIPRE or not. Here you always run VIPRE, correct? You don't have a screening criteria?

MR. ANDERSON: We use the criteria, because we run a full spectrum of cases. So we're looking at hundreds of cases. And only a subset are actually run in VIPRE explicitly, because they're screened by RELAP.

CO-CHAIR MARCH-LEUBA: Is that proprietary, or can you tell us what the screening criteria is?

MR. ANDERSON: No, it's not proprietary.

It's generally we have a correlation in RELAP that we test. And the non-LOCA topical talks about that. But it's basically we benchmark the RELAP evaluations calculation of CHF to VIPRE and then, and proven to ourselves that that's a good indicator. But all the --

CO-CHAIR MARCH-LEUBA: I want to keep it at a non-proprietary level. But the RELAP CHF correlation is way off.

MR. ANDERSON: Well, there's multiple ones depending on the problem to be solved.

CO-CHAIR MARCH-LEUBA: So you're not using the one from the non-LOCA? Are you using a more accurate one here?

MR. ANDERSON: Well, there's several CHF correlations in the RELAP model we use. And this one is one that is an approximater for our NSP correlation.

DR. CORRADINI: We can wait.

MR. ANDERSON: Okay. And --

DR. CORRADINI: You can tell us to wait. You don't have to be nice to us. We can wait to the proprietary if you want.

MR. ANDERSON: Yeah, well, the non-LOCA topical, in particular, will talk --

DR. CORRADINI: Okay.

MR. ANDERSON: -- more about that.

CO-CHAIR MARCH-LEUBA: Okay. We'll wait for the non-LOCA or the proprietary, and I'll ask you again.

MS. McCLOSKEY: Megan McCloskey, NuScale. Yes, we can discuss this in the non-LOCA topical report presentation a little bit later this morning.

I would also clarify that the screening that we do with NRELAP5 in the, as part of the non-LOCA topical report, is only to indicate the trend of which cases are more limiting for CHF. and the margin assessment is all done with VIPRE and the approved NSP4 CHF correlation.

DR. CORRADINI: So, since you're up there, can you clarify? The way I look at this is there's three potential criteria. You then have what I'll call a factor of safety. If you approach your factor of safety, if any one of these three, you tag it as something that's got to be looked at more closely.

In other words, if MCFR is --

MS. McCLOSKEY: I would defer to Kenny --

DR. CORRADINI: Just pick a number. If MCFR is within a factor of, if CHF is within a factor of two of what you see, then you tag it so that you

look through all three criteria and only redo them with VIPRE once you get close enough.

MR. ANDERSON: Yes, so the, for the -- we kind of analyze each acceptance criteria uniquely.

DR. CORRADINI: Oh, okay.

MR. ANDERSON: So, for the system response, we calculate, or for the pressurization of the system, we just use RELAP. And we make sure that we've got the limiting case.

For the CHF evaluation, we send basically up to hundreds of RELAP cases to pick out a family of what appears to be the limiting.

DR. CORRADINI: Okay. Fine.

MR. ANDERSON: And we want to be robust. So we don't want to have it too small of a group.

DR. CORRADINI: Okay.

MR. ANDERSON: And then, in the closed session, you'll see some results of that screening process. There's ten candidates. And then you can see how the VIPRE actually says, oh, this is the one that's the most limiting.

DR. CORRADINI: Okay. Thank you.

MR. ANDERSON: Yeah.

DR. CORRADINI: Thank you very much.

CO-CHAIR MARCH-LEUBA: Yeah, I just wanted

to say it in the open session. We need it 21st century. How much does it cost to run VIPRE? I mean, why not run them all properly and know what -- you're supposed to do a whole bunch of scenarios for control rod ejection, I mean, number one, number two, number -- just run them all and you know.

I mean, I just don't see the -- in 1970, we had to do it with a slide rule. And yes, definitely you had to do approximations. I don't see the reason for doing approximations now. But anyway, that's just recreational complaining.

MR. ANDERSON: So the first main step is this, the steady state initiation of a nuclear physics code. And here, we're focused on setting up the full spectrum and making sure we have all the possible equipment configurations at all the possible times in life. And then we also do with this determine the worst rod stuck out, it's part of the fundamental assumption.

Then the workhorse, the star of this whole show is the transient kinetics. We, as part of our work and to prove the methodologies is applicable, this design and the code used is we benchmarked this SPERT-III and the NEACRP computational benchmark.

DR. CORRADINI: So let me ask about SPERT-

III is back in the 60s, is it not?

MR. ANDERSON: I believe the 60s or 70s, I'm --

DR. CORRADINI: So those are integral measurements. So you're benchmarking off of integral measurements and you're getting detailed measurements from the computational benchmark? Because SPERT didn't measure anything other than overall power increase based, they didn't do any sort of spatial analysis that you can benchmark.

MR. ANDERSON: Correct. To my knowledge, correct, we benchmarked the overall power pulse.

DR. CORRADINI: Fine.

MR. ANDERSON: And shape and worths.

DR. CORRADINI: Thank you.

MR. ANDERSON: Yes. Then also in this part of the analysis, we applied uncertainties in the, we do reference a topical, the nuclear analysis topical. Generally that presents our method for calculating the uncertainties. And then each time we apply this method, we do ensure that our uncertainties that we're using are valid.

DR. SCHULTZ: Kenny, what you present here are your, what you call your major uncertainties? Thought there was a table with several other

uncertainties, which are applied.

MR. ANDERSON: Correct, these are just the uncertainties for this portion of the analysis.

DR. SCHULTZ: I see.

MR. ANDERSON: And then at the, in one of the slides there's a stack-up that shows all the uncertainties and where they're treated.

CO-CHAIR KIRCHNER: While that's up, which of those uncertainties did you see had the most effect? I have my own idea which one it is, but could you go through those delayed neutron fractions? Rod worth's obvious, Doppler is the, what's terminating the event. And --

MR. ANDERSON: Yeah, I don't have a specific quantification. Generally, delayed neutron fraction, the static worth, and the Doppler coefficients are extremely significant. But we could -- I don't have a exact quantification.

CO-CHAIR KIRCHNER: Okay.

CO-CHAIR MARCH-LEUBA: And this is in the event that you have to evaluate the power, because that typically is the one that becomes worse. And those conditions you can have such high boron concentration that day, the moderator temperature coefficient can be positive. Is that correct? Or do

you always consider it negative?

MR. ANDERSON: Well, we, as we'll see in the closed session, we have some example results. And we'll explain why for our design it's actually middle powers is the most limiting. And so I can --

CO-CHAIR MARCH-LEUBA: It may not be, if this has serial power with a positive boron coefficient.

MR. ANDERSON: I guess I'd have to --

CO-CHAIR MARCH-LEUBA: If your boron coefficient were to be positive, very positive, that --

MR. ANDERSON: I'll have to check. I don't think that that situation is possible.

CO-CHAIR MARCH-LEUBA: We'll wait, I'll wait. I'll make another note for the closed session.

MR. ANDERSON: Okay.

CO-CHAIR MARCH-LEUBA: I see that I started this morning that we may have many time for tomorrow.

MEMBER PETTI: Question, there's burnable poisons in the design?

MR. ANDERSON: Yeah, we use gadolinium in the current design.

The next step, or a key step, is the CHF

evaluation using the subchannel code VIPRE. And we rely on our subchannel topical, but there are a few deviations from that. This is the unique event, it is very, the power pulse is very quick. It's on the order of milliseconds.

And so making sure that the finite difference code is being solved correctly requires smaller time steps. And then with the current number that involves ensuring that the axial nodalization is appropriate. So we do a set of sensitivity cases to make sure that we are using the code appropriately.

DR. SCHULTZ: Kenny, in that regard, you've done these things prior to doing the analysis to demonstrate that your analyses is identifying?

MR. ANDERSON: Yeah, I would say the answer's both.

DR. SCHULTZ: You're not doing these sensitivities for every evaluation that you do.

MR. ANDERSON: The answer's both. We do a set of sensitivities to make sure we understand and the methods is good and appropriate. But because of the unique nature of this event, we do specific sensitivities to ensure that what we expect to happen in terms of the codes --

DR. SCHULTZ: If you've done it not only

in general to understand it, but when you have a limiting case evaluation, you're also going through the sensitivities to demonstrate that you've got, you've nailed it computationally.

MR. ANDERSON: Yes.

DR. SCHULTZ: Okay, thank you.

CO-CHAIR MARCH-LEUBA: Can, remind me again because we saw this two years ago. When you do the RELAP calculation, you do a full core, one, every single channel you calculate the power. When you do the VIPRE calculation with those boundary conditions, how many rods do you model?

MR. ANDERSON: In this case, it's 52 rods.

CO-CHAIR MARCH-LEUBA: So you do only a fraction of each bundle.

MR. ANDERSON: Yeah, the bundle of, or the assembly where the actual, we postulate the rod to be ejected, every rod in that assembly is fully detailed. And then as we get further away, there's progressive lumping, such that the furthest away is --

CO-CHAIR MARCH-LEUBA: I remember the maps when we saw. Roughly there are 52 representative regions, and one to one for the rods that are closer to the control rod? Or do you use pin power reconstruction to identify three rods in the fuel

bundle is the worst, or something like that?

MR. ANDERSON: The transient kinetics does do pin power reconstruction, and we do use that to inform the power distribution that we use --

CO-CHAIR MARCH-LEUBA: So you run RELAP, you get the average power of the bundle. Using pin power reconstruction, you get the power in every rod. And that, then you confirm that the distribution around the control is adequate. Is that more or less what you're doing?

MR. ANDERSON: More or less. But yeah then, we get the full power distribution from SIMULATE-3K, and then we, yeah, then like you said we apply that in the VIPRE.

CO-CHAIR MARCH-LEUBA: Thank you.

MR. ANDERSON: As we talked about, one of the unique features of this event is the fuel criteria. And all -- most of them use an indicator of calories per gram to help indicate either fuel failure or core coolability.

And to satisfy that, we do a very conservative adiabatic heat-up hand calculation, where we essentially put all the energy into the pallet and calculate the calories per gram as a result of the power pulse calculated from SIMULATE-3k.

DR. SCHULTZ: This is the calories per gram as well as fuel melting is from this? And the fuel melt evaluation is also with this analysis?

MR. ANDERSON: Yes, we do both the calories per gram and the fuel centerline melt.

DR. SCHULTZ: These are the conservatisms that are applied to the evaluation, I'll call it, of fuel melt.

MR. ANDERSON: Correct.

DR. SCHULTZ: Conservative in that regard. Thank you.

MR. ANDERSON: Okay, the next part we can talk about is the system model. And there's two parts, so this is part one. This is the part where we evaluate CHF, and as we talked about previously, this is primarily to, well, to screen the cases for VIPRE.

And as part of that it also looks at the boundary conditions that's happening in the whole, the whole module, the inlet flow and the exit pressure to pass those to VIPRE to ensure VIPRE has all that information.

The SIMULATE-3K doesn't do the, doesn't do a boundary. It assumes boundary conditions, it doesn't calculate that. But the event is quite rapid, and that's so it justified that.

The other part, there's another aspect that RELAP is used for, and that's for the system pressurization. For us, the, part of the intent of GDC 28 is to prevent the very high integral energy that the, vaporizes the steam and can create a water hammer. But with our very low integrated powers, what actually happens is the system in the classical peak, the power pulse type event, the energies are just too low.

And so you don't see a system pressurization response. So we do have a more traditional non-LOCA type analysis to have a pulse that's just high enough to be below, to increase the power, but it's below the high power trip set point. And then eventually the system will heat up and pressurize, and then the system will trip on the high pressure. And then we show that that does stay below the high pressure acceptance criteria.

DR. CORRADINI: You're going to say that all over again, you've lost me. I thought I'd got you, but then as you started proceeding you lost me. Can you try that one more time?

MR. ANDERSON: Yes, there's the classical rod ejection, which, you know, GDC 28 is really designed to protect, is there's this extreme power

pulse. And that extreme power pulse can vaporize the coolant.

DR. CORRADINI: Right.

MR. ANDERSON: And we've, we want to protect against. But if the power pulse is so low that there's not that much energy to increase the system coolant, then that's not actually the limiting pressurization.

The limiting pressurization looks like a traditional non-LOCA event, where you increase the power just enough to be below the set point and have it sit there and slowly pressurize the system. And then eventually the system pressure is high enough to hit the system pressure set point.

MEMBER PETTI: So it's like a transient overpower.

MR. ANDERSON: Yes.

MEMBER PETTI: In some of the other reactors.

MR. ANDERSON: Yes.

MEMBER PETTI: And let me just, so I think I know, why is the worth so low?

MR. ANDERSON: It's the static, it's part of the design, the static worths are just low.

CO-CHAIR KIRCHNER: Well, you're running,

you're running at power and the rods are pretty much all out. So when you eject a rod that's already well withdrawn, you're just not putting that much worth into the system. And so you just get a pressurization as a result.

MEMBER PETTI: I figured it's the low initial stored energy of this. These are half the linear power of a large light-water reactor. And all this burnable poison is holding down any excess of water in it, right?

DR. CORRADINI: So you're searching for -- I guess I was -- the way Dave explained it, is that correct? Because that's how I was guessing is the reason.

MR. ANDERSON: Yes, that's --

MEMBER PETTI: From a non-physicist view --

MR. ANDERSON: No, yeah, that's exactly right. Yeah, if you look at a reactor design very different from ours, like SL1, it had a very --

DR. CORRADINI: Bad example.

MR. ANDERSON: Well, it had a very high, it had very high static worths, and that's basically the opposite of us. So we have very low static worths, and a key input that is where we have the

power and searching limits.

DR. CORRADINI: And we'll see some representative numbers in closed session.

MR. ANDERSON: Yes.

DR. CORRADINI: Okay.

DR. SCHULTZ: Kenny, in the draft Reg Guide 1327, the guide sets limits also for the potential for pellet clad interaction looking at differential pressure between system and internal pressure of the fuel. Do you do evaluations of that, of the pressure versus the internal pressure of the fuel to see whether you match up with the assumptions that are in the reg guide -- draft reg guide, excuse me?

MR. ANDERSON: Oh, I was going to say we, as we saw in that first table, we do, those are some of the criteria, the differential pressure criteria, where we used the calories per gram as an indicator.

DR. SCHULTZ: So you're looking at it to match up with the calorie-per-gram levels that are chosen for that draft reg guide. You're not doing the calculation of differential pressure to determine whether you --

MR. ANDERSON: Correct, yeah, we --

DR. SCHULTZ: Whether there's an issue or

problem there.

MR. ANDERSON: We use, we calculated the calories per gram, and we used that as an indicator for the internal pressure and the differential pressure.

DR. SCHULTZ: Yeah, I'm just curious as to whether, when you do all these calculations, whether you see situations where that differential pressure criteria would in fact come into play.

MR. ANDERSON: And generally no. What limits, as we'll see in the closed session, what limits us is the CHF evaluation.

DR. SCHULTZ: Okay, that's what I was looking for. Thank you.

MEMBER PETTI: So a typical -- I'm sorry. That's all right, no, you go ahead.

MEMBER BLEY: Okay, go back two, one or two slides where you have the picture. There, that one. When we make these things up and then analyze in great detail the neutronics and other parts of what's going on, what about the kind of larger, gross things? Exactly where do we assume the break occurs that allows the rod to be ejected?

MR. ANDERSON: Will it be at the --

MEMBER BLEY: Can you show it on there?

MR. ANDERSON: Oh, sure. So the, this is the reactor vessel.

MEMBER BLEY: And those are the mechanisms sticking out --

MR. ANDERSON: Yeah, and then so then there would be a failure --

MEMBER BLEY: Right around there.

MR. ANDERSON: Yeah, and then a --

MEMBER BLEY: Okay. So somehow that happens and it falls off.

MR. ANDERSON: It was postulated --

MEMBER BLEY: And where's the rod go?

MR. ANDERSON: For this analysis, it --

MEMBER BLEY: Disappears.

MR. ANDERSON: It disappears, yes, thank you.

MEMBER BLEY: So where's the rod go, and can it -- what happens when it hits the top of the reactor vessel and maybe the containment? I mean, we're doing all this fancy analysis on a really hypothetical event, and yet the --

MEMBER BALLINGER: The rod is about 30 or 40 feet long.

MEMBER BROWN: Like a spear.

MEMBER BALLINGER: And the pressure

boundary is way up there.

MEMBER BLEY: It ain't way up in this one. I mean, it's way up in other plants.

MR. ANDERSON: Yeah, as part of the, and we can bring in our ASME guys, well we --

MEMBER BLEY: It just goes away.

MR. ANDERSON: Well, in general it's not an ASME event that the, we don't use j-groove welds or anything like that. And so they, we don't analyze turbine missiles -- or I mean, not turbine, missiles.

MEMBER BLEY: No, you don't.

MR. ANDERSON: Yeah, and the reason we don't do that is because the ASME, you know, rules and regulations for that. But we can get more detail on that.

MEMBER RICCARDELLA: You say you don't use j-groove welds in the top bay?

MR. ANDERSON: Correct.

MEMBER RICCARDELLA: How are those nozzles attached, full pen loop?

MR. ANDERSON: I'd have to -- yeah, it's part of the integral to the -- well, I'd have to get back on that, the exact.

CO-CHAIR KIRCHNER: But, or the ends are bolted flange connections.

PARTICIPANT: That's what it looks like.

MEMBER RICCARDELLA: But the attachment, yeah, but it's, that's bolted at the other end of the penetration. There's got to be an attachment to the vessel head, right. We --

MR. ANDERSON: Yeah, I'm just going to make sure I bring back the best people to talk about that.

MEMBER RICCARDELLA: Yeah, but I have a more general question, is aren't there other ways to eject a rod other than a pressure boundary failure? I mean, I thought like in a regular PWR, you've got these mechanisms that can fail, or in a BWR you can drop the rod without necessarily failing the pressure boundary.

MR. ANDERSON: Yeah, in --

MEMBER BALLINGER: These are fine motion control rods, I think. And so the failure, when you lose power it clamps them on. They can't move. So it's sort of like the navy.

MEMBER RICCARDELLA: Yeah, but if a --

MR. ANDERSON: One of the things to remember, too, is it's not necessarily how much the rod moves from beginning to end, it's how quickly the rod moves. And it's the quickness of the rod moving

when you involve the delayed neutron fraction that makes the dynamic worth so important.

MEMBER RICCARDELLA: But that being the case then, doesn't this involve a LOCA? If you break the penetration --

MR. ANDERSON: Yes, it does.

MEMBER RICCARDELLA: Oh, so your analysis does assume that a LOCA, that there is a LOCA.

CO-CHAIR KIRCHNER: But they don't take credit for the leakage.

PARTICIPANT: It evaporates. The control rod disappears --

PARTICIPANT: It disappears.

PARTICIPANT: In a certain amount of time.

DR. CORRADINI: It's a bounding calculation.

MEMBER PETTI: Yeah, it's a bounding calculation.

MEMBER RICCARDELLA: I understand, but it seems to me there might be ways that the teeth on the mechanism break or something, and you eject a rod without necessarily failing the penetration.

MEMBER PETTI: We could ask our risk experts. I don't think this is a risk-dominant event, you just don't see it.

CO-CHAIR KIRCHNER: No, and in NuScale's favor in this design, the extension is quite substantial and the weight. So there's no way for the rod to disappear, short of having something like a failure of the control rod drive.

MEMBER BROWN: Then why couldn't, I mean, you've got a vacuum in the containment vessel or partial vacuum. You've got lots of pressure in the reactor pressure vessel. You eject a rod that seems I've got considerable forces driving it up. Why couldn't it penetrate the containment vessel?

It's not even addressed. In fact, as the SER from the staff says --

DR. CORRADINI: I wouldn't want to address it.

MEMBER BROWN: How would you want it addressed?

DR. CORRADINI: Why would you want to address it?

MEMBER BROWN: I don't know.

DR. CORRADINI: It's a bounding calculation to show that from a reactivity and insertion standpoint, you have no problem.

MEMBER BROWN: I'm talking about the LOCA --

DR. CORRADINI: If you get mechanistic, you're never going to get close to this.

MEMBER BROWN: I'm talking about the LOCA situation that this SER even says you could cause a LOCA with this. And you're, all the, how do you cool it after that? All the coolant, just said that disappears from the ability to cool the core, if you look at their pictures. And it's not covered in the LOCA, it's not covered in the LOCA TR.

MR. ANDERSON: Well, I think it is as part of the LOCA methodology.

MEMBER BROWN: I just looked, I looked up rod ejection and it's not in there.

MR. ANDERSON: And we --

MEMBER BROWN: Doesn't mention it.

MR. ANDERSON: Yeah. We could have the, as part of the, or we could do it anytime you want, but we could have the LOCA people speak to that.

MS. McCLOSKEY: Megan McCloskey, NuScale again. The control rod ejection's not part of the LOCA topical report, as you identified. It was considered with respect to the containment pressure response analysis. But in the NuScale design, the liquid space breaks are much more limiting compared to the vapor space breaks, which would be from the top of

the reactor pressure vessel.

And the amount of energy that's added due to the power pulse is small compared to the thermal energy of the fluid initially in the RCS that drives the containment pressurization response. So it's also non-limiting for peak pressure temperature.

MEMBER BROWN: That part I understand, I guess my question is really related to the fact why isn't the velocity of this ejection that it's being driven from the core, 2000, whatever the pressure is down in the reactor pressure vessel, the vacuum up there, but why can't it drive itself right through the containment vessel. And now I've got a complete breach of the containment --

MR. ANDERSON: But we can provide a --

MEMBER BROWN: And that's not discussed in any of the documents that I could find.

MR. ANDERSON: Yeah, we can provide an answer for the missile, the missile question.

MEMBER RICCARDELLA: That's what it is, it's a missile question.

CO-CHAIR MARCH-LEUBA: You can and -- you can or you will?

MR. ANDERSON: We will, sorry. I can't at this moment articulate it.

DR. CORRADINI: I just think we're going down a rabbit hole that we don't want to go down.

MEMBER BROWN: I don't agree with that, Mike.

DR. CORRADINI: Well, I do --

MEMBER BROWN: I mean, you've got the CHF --

DR. CORRADINI: That's just the consultant's opinion, it's a rabbit hole.

MEMBER BROWN: But then you've got, can you breach, actually breach the containment vessel. Why isn't that something to think about?

DR. CORRADINI: Then you'd have to do a mechanistic acceleration, which is going to be small because the depth, the distance between the time where it is and to where it's going to hit is so small you probably wouldn't hit it for, with a few meters a second. Therefore, it probably won't attack, it won't hurt the containment at all.

MEMBER BROWN: Huge differential pressure.

DR. CORRADINI: But if you make it mechanistic, Charlie, then the power pulse itself will be much smaller than what they're assuming. It's a no nevermind in either case, but if you want to make a consistent calculation with what Dennis I think is

driving towards, Dennis started it, which is --

MEMBER BROWN: If he hadn't I would have, so.

DR. CORRADINI: Is that you would probably have a very small power rise, very small power rise.

MEMBER RICCARDELLA: Yeah, but forget about the power run, just hole in the containment. Just having a LOCA that -- and then having something that pierces the containment is a concern, independent of the power surge.

MEMBER BROWN: Exactly. To me it looks like you're looking at the trees.

MEMBER RICCARDELLA: No, I would suspect you could do the analysis and you show, you could show that that's not going to penetrate.

MEMBER BROWN: Well, that would be fine. I just --

MR. ANDERSON: We have considered it and we do have an answer, I just, I don't know, and I'll have to work on my colleagues to get that answer for you.

CO-CHAIR MARCH-LEUBA: So let me go back to my question. When will we see that analysis?

MR. ANDERSON: Oh, we have an answer, we'll have an answer for you today.

CO-CHAIR MARCH-LEUBA: So you have somebody in Corvallis working on it? My people will work with your people.

MEMBER RICCARDELLA: But you're saying there's no other way -- I guess it doesn't really matter. You're saying there's no other way to eject the control rod other than to violate the pressure boundary. And the fact that you didn't assume you violate the pressure boundary is conservative for the purposes of this analysis, is that what you're saying?

MR. ANDERSON: Yes, generally, we postulate this as a bounding method for evaluating the GDC 28 criteria.

CO-CHAIR MARCH-LEUBA: So what I hear the members say is that from the point of view of neutronics and the power peak, you're likely conservative. I mean, we cannot, we don't see where you are non-conservative. However, there is a possibility of a missile inside the containment that you are going to tell us this afternoon why it's not a problem.

MR. ANDERSON: Correct.

DR. CORRADINI: For this one person, if you're going to make them do this, then you're going to have to ask the question is does it leak, or does

it actually sever. Because it strikes me if I start worrying about a control rod drive mechanism opening up, I would expect it to leak, not sever and shoot out.

MEMBER RICCARDELLA: Well, we came pretty close to severing it at Davis-Besse, so I mean.

MEMBER BALLINGER: Except for the fact that now we have all new materials and a lower temperature.

MEMBER RICCARDELLA: I understand all that, but --

MEMBER BALLINGER: And a few other minor details.

MEMBER RICCARDELLA: But the other question I'd like answered is what are those attachments of the control rod drive nozzles to the head if they're not j-groove welds?

MR. PRESSON: Those are full penetration welded.

MEMBER RICCARDELLA: Full penetration welded, okay, thank you. That should solve the problem.

MEMBER BALLINGER: That's another minor detail.

CO-CHAIR MARCH-LEUBA: But, Mike, what I

understand from them is they have evaluated two limiting cases. One is the rod moves at speed and the issue of power pulse. The other one is the rod heats the containments. We're going to see that this afternoon.

MR. ANDERSON: We'll provide the evaluation for the missile.

CO-CHAIR MARCH-LEUBA: So you've done two complete independent bounding analysis and both of them are good.

MR. ANDERSON: Correct.

CO-CHAIR MARCH-LEUBA: And none of the two are credible within some --

MEMBER BROWN: I'm not arguing credibility, it's just a rod ejection, we're required to evaluate it, it has a certain end result, can that end result be a problem? And that's all.

CO-CHAIR MARCH-LEUBA: I'm going to write it down for my closed session. Ask Charlie to ask the question again.

MR. ANDERSON: So in summary this is the overall approach to analyzing the event, and which we then provide input to the Chapter 15. And in the closed session there'll be example inputs and output values and show how the uncertainties stack up.

CO-CHAIR KIRCHNER: Okay, any other questions on rod ejection in the open session? We've got notes for the closed session.

MR. BRISTOL: This is Ben Bristol with NuScale. The RAI 9647, we specifically address the difference between the postulated event for GDC 28 and the CRDM, the postulated CRDM failure, which is not a design-basis failure.

So the dynamic effects of that event are not part of the design basis, up to and including considerations of if the shaft were to accelerate to the containment head.

PARTICIPANT: What does that mean, it's not part of the design basis?

MR. BRISTOL: It is not part of the design basis. So the, a failure of the CRDM pressure boundary is part of the RPV, is considered part of the RPV pressure boundary per the ASME specifications.

MEMBER RICCARDELLA: But you're saying you don't postulate that as a potential LOCA.

MR. BRISTOL: That's correct. The dynamic effects of the failure of the CRDM is not considered as part of the design, design basis.

MEMBER BROWN: You don't have to worry about it penetrating the containment, in other words.

MR. BRISTOL: That's correct. It is designed to ensure that it will not fail as part of the reactor pressure boundary.

CO-CHAIR MARCH-LEUBA: Wait, wait, wait, how is it designed and you haven't analyzed it? I mean, you're saying it's not a design-basis event. I'm not going to bother with analysis of this missile inside my container.

MR. BRISTOL: So the components are designed per the ASME specification so that they do not need to be considered as postulated failures.

MR. PRESSON: So that's part of the full weld --

MEMBER RICCARDELLA: Are you saying because it's a vessel, not a pipe? I mean we --

MR. BRISTOL: That's correct.

MEMBER RICCARDELLA: Design piping to ASME code as well. We postulate failures of piping. This mechanism up to that flange that attaches the, it seems more like a pipe to me than a vessel. But.

DR. CORRADINI: My suggestion is wait till closed session.

CO-CHAIR MARCH-LEUBA: I think the thing is wait until Chapter 15 when they talk about LOCAs.

CO-CHAIR KIRCHNER: When they get to

LOCAs, I think they'll be able to demonstrate that a failure of that part of the pressure boundary is not the limiting case for them.

CO-CHAIR MARCH-LEUBA: Yeah, but that, there's a difference between local depressurization concerns and the concern is the missile.

PARTICIPANT: And the missile's a separate topic, yeah.

MEMBER BROWN: This is a totally different design than all the rest of them, the containment that's just sitting right there adjacent with everything else. It's a very closely coupled system, so I don't understand why you make the comment that it's not part, you don't have to consider it as part of the design.

It's like telling me I don't have to worry my reactor protection system because I design it not to fail. Jesus, you know, when am I going to go work on that one?

CO-CHAIR MARCH-LEUBA: That's what they tell me when I complain about ATWS. It's not going to happen.

MEMBER RICCARDELLA: Historically, in nuclear design, we've made the assumption that pipes can fail, and we design for LOCAs that are failures of

pipes. And, but the reactor vessel itself, we draw the line there and we say that, we're not going to postulate failure of the reactor vessel.

But to me, it's skinning the cat pretty thin to say well, I got this long tube that sticks out of the vessel, but we don't have to assume that fails because it's part of the vessel. I mean, that's a very legalistic interpretation of something that I think was beyond the intent of that. But we can talk about it more later.

CO-CHAIR MARCH-LEUBA: But let me be -- but the summary of the rod ejection accident, rod ejection is considered here as a neutronic power peaking effect. So we have to take a note to address this issue. And just because we're talking about the methodology for LOCA, that doesn't include missile this afternoon. We need to address it for March 2 when we're talking Chapter 15 about --

MS. PATTON: I'd just like to point out there's -- I can give you guys a couple of references where this is addressed in the DCA SER. So you'll find the missile section with a discussion of this in Section 3.5.1.1 and then a discussion with respect to return to power.

This is also addressed. It was a UOI.

That's in 15.0.6.4.3. But you're right. It's a appropriate for the DCA section, not the methodologies.

CO-CHAIR MARCH-LEUBA: Do you understand the issue that the members are having?

MS. PATTON: Yeah.

CO-CHAIR MARCH-LEUBA: We are worried that because there's an initiation then in which the they are all fails and that causes a failure of the containment.

MS. PATTON: That's correct. What I am telling you is what NuScale said is correct and you'll find sections in the SE where it writes up --

CO-CHAIR MARCH-LEUBA: Well, 3.5.1.1 should tell me something about --

MS. PATTON: 3.5.1.1 will give you the missile information and then there's also --

CO-CHAIR MARCH-LEUBA: It's not going to tell me what they told me, that I don't want to consider it because it's --

MS. PATTON: It's going to tell you that it's -- that it's not considered as a -- as a failure.

CO-CHAIR MARCH-LEUBA: So we may have to revisit our Chapter 3 letter. You disagree?

MS. PATTON: Well, you can look at the

write-up and it's in there from what you guys saw previously. So --

MR. PRESSON: And the RAI 9647 does ask you pretty much this exact question. So having that question and that response.

CO-CHAIR MARCH-LEUBA: Yeah. The question -- can anybody --

PARTICIPANT: What's the RAI?

MR. PRESSON: 9647.

CO-CHAIR MARCH-LEUBA: Chris, can you get us a copy of the RAI 9647?

Has anybody analyzed this missile? I mean, the way that Dr. Rena (phonetic) says there is no way this missile has enough time to accelerate to break anything.

DR. CORRADINI: Well, why don't you ask the question of current light water reactors required to analyze what you're asking? I think the answer -- the answer is no.

CO-CHAIR MARCH-LEUBA: I am asking is this reactor safe or not?

MEMBER RICCARDELLA: Containment is 50 feet away.

DR. CORRADINI: Gravity is not going to matter 50 feet. Once -- if I accelerate this thing it

could be 50 feet away or five feet away. It's going to be the same impact in terms of kinetic energy.

CO-CHAIR MARCH-LEUBA: Well, my point is Mike's hand calculation says that it's not going to fail anything. But have you guys done anything to it or you just said we don't have to do it? It seems to me that you don't have to do it.

MR. PRESSON: I'll have to get back. So --

CO-CHAIR MARCH-LEUBA: If you can think about it for the closed session and tell us.

MR. PRESSON: Yeah.

CO-CHAIR KIRCHNER: Okay. At this point then, Matthew, then are you and Kenny going to do the next presentation or you have to change out people?

MR. PRESSON: I have to change out some people so --

CO-CHAIR KIRCHNER: Okay. Well, why don't you go ahead and do that and --

CO-CHAIR MARCH-LEUBA: Let me remind you that you guys are responsible for your own name tags. Are you going to have one tomorrow?

CO-CHAIR KIRCHNER: Okay. Thank you, Kenny.

CO-CHAIR MARCH-LEUBA: And Walt was

reminding me that we are a little behind so -- and it's our fault. So it was our fault.

MR. PRESSON: All right. So we are now here to discuss our loss of coolant accidents evaluation model methodology. I am sensing today will be Dr. Pravin Sawant, supervisor of code validation methods, Dr. Selim Kuran, who is our thermal hydraulic analyst, and we left off Ben Bristol, our supervisor within hydraulic methods. So but he will presenting as well.

Start and pass it off to Pravin.

DR. SAWANT: So in this open session on LOCA evaluation model topical report we will just try to provide a overview of the methodology. Basically our objective will be to show, like, how methodology follows Reg. Guide 1.203 guidance.

We will visit each element of Reg. Guide 1.203 and provide a overview of how methodology is addressing those elements and more specificities will be eventually discussed in closed session.

So next slide. Yeah. So progression of LOCA in our design is simpler in part because of innate features of the design. With all the RCS complements contained within RPV the potential for number of breaks and the size of breaks is limited,

and the coolant is simply captured in containment, cooled, and returned back to the RPV, and the phenomena -- familiar phenomena that are in all to model LOCA progression are well known and that makes the modeling LOCA straightforward.

We are following, as I said, Reg. Guide 1.203 guidance and the methodology is in compliance with the applicable continuities in 10 CFR 5046 Appendix K. So --

CO-CHAIR KIRCHNER: Pravin, can I ask you to speak out a little louder and pull your microphone closer. Yeah.

DR. SAWANT: Okay.

CO-CHAIR KIRCHNER: You've made an assumption that --

MEMBER BLEY: That microphone is not turned on. You can turn it on by pushing -- nearer to you. There you go.

CO-CHAIR KIRCHNER: It should have a green light.

DR. SAWANT: Okay.

CO-CHAIR KIRCHNER: Can we go back one slide? You made a -- you postulated something. You're correct. Your design has less extended piping runs and such. But you went -- you kind of said it's

simpler to analyze but yet the third sub-bullet, the CNB heat transfer, that actually is an area that's not simpler. So we will come back and visit -- revisit that with you.

DR. SAWANT: Yes. So yeah, it's -- we are calling it simpler because what we saw was, like, the heat transfer mechanisms and all are well understood -- that it's been condensation, conduction, and convection to be -- convection to cool on external surface. These are the -- these mechanisms are understood from first principles.

CO-CHAIR KIRCHNER: Yes, but let me say it this way. Taking NRELAP5 and ringing it up to do that analysis is not straightforward and the code wasn't designed for that. So it does actually challenge you in the application of the code to analyzing these phenomena. I just want to make that observation.

DR. SAWANT: And we have --

CO-CHAIR MARCH-LEUBA: Let me help you a moment. We have -- you have performed a number of verification tests that has a benchmark in RELAP and that confirmed the condensation correlation, which I think is proprietary -- the name of it. But there is a condensation correlation in RELAP that has been verified against data. Is that correct?

DR. SAWANT: Yes. So we have NuScale's experimental data to confirm that. We have to be careful. We have to model the same heat transfers. We will talk about this.

CO-CHAIR MARCH-LEUBA: And so they agree with you all that that was a difficult problem and that's why they spend a lot of money on tests.

CO-CHAIR KIRCHNER: Yeah. Yeah. You're making something simpler than it actually was for you. You had to put a fair amount of effort into both your methodology and your benchmark experiments that you ran out in Corvallis.

DR. SAWANT: We do have extensive database that we will talk about and when you look at the report and this difference of the phenomena we will see that these are not new and in time I will brief you. So we will talk about that in more detail in closed sessions.

So, yeah, 10 CFR 50.46 provides the acceptance criteria policies as designed in light water reactors and we are all aware of these acceptance criteria.

What we saw that in our design peak clad temperature is this will be the normal operating clad temperature on the steady state and core during LOCA

doesn't experience ACHIP (phonetic) and it remains covered and we don't see clad heat up.

So we ended up using the much conservative acceptance criteria for LOCA, which are listed there as the sub-bullets for the last bullet that core remains covered and we show that by comparing collapsed level.

We show that by comparing collapsed level show that -- showing that collapsed level we miss our protective wheel, minimum critical heat flux ratio remain several or -- stage (phonetic) of our limit and we also had additional acceptance criteria that continued pressure and temperature in its developed design limit because containment is an integral part of ECCS, although I want to mention that there is a separate methodology to calculate peak containment pressure and temperature.

CO-CHAIR MARCH-LEUBA: Yeah. I asked you cover, and because we got all these slides yesterday I read it overnight. I asked you to cover the difference between the different CHF correlations that we are using all over different methodologies.

You're referring here to the CHFR limit or the RELAP LOCA analysis.

DR. SAWANT: LOCA analysis.

CO-CHAIR MARCH-LEUBA: Yes. Will we -- will you guys be covering that or is it the staff that is going to be covering the difference in CHF?

DR. SAWANT: Part of LOCA closed session presentation we will talk about the correlation that is in the LOCA methodology.

CO-CHAIR MARCH-LEUBA: So let me to do it myself, okay, because if anybody is reading the record and goes back in time to July or June of 2019, we left this rule with me saying that there is no way that these CHF correlations are possibly correct because there is a significant difference between the ones predicted by the non-LOCA, ones predicted by LOCA and then finally ones predicted by rolodiction (phonetic) are also different, and when we went to Corvallis we had a presentation and we -- told that indeed they are different, and for LOCA there is not a CHF correlation but there's a critical power correlation, and that's a difference in numbers and that's why they gave a factor of two difference between one and the other.

So I just wanted to put in the record that as far as I am concerned I am satisfied on whatever we left on back in June has been -- has been resolved and needs to be on the record that we don't have that discrepancy any longer.

DR. SAWANT: Understood.

CO-CHAIR MARCH-LEUBA: So you are talking about the CHFR -- that you are -- the CHF correlation disagrees with the power correlation and using the LOCA specific for the NuScale fuel and applicable to the RELAP average bundle conditions.

DR. SAWANT: Yes, for LOCA -- and this slide shows a roadmap of the methodology. Again -- we do satisfy conservative requirements of Appendix K and these elements are basically from Reg. Guide 1.203 with Element 1 that talks about establishing the requirements for the evaluation model which is generally done by developing for it.

So we present are working LOCA LTR Chapter 4. We will talk about the phenomena in a little more detail in closed session.

Element 2 is, basically, a development of assessment base that involves development of your separated and integrated case assessment in this and it's been documented in LTR Chapter 7, and Element 3 is a development of actual mathematical model.

We talk about that in several chapters in LTR we would like to visit this mathematical model in more detail in closed session.

And finally, Element 4, which is adequacy

evaluation of evaluation model, which is there we go through top-down and bottom-up evaluation to assure the applicability of the evaluation model to predict these high-ranked phenomena.

So, again, like in open session I recall the overview but there is some more details about scaling in distortion analysis in closed session.

DR. SCHULTZ: Pravin, you draw this as if it's direct from top to bottom, that is, in terms of information flow. Was it not iterative? In other words, you did the PIRT but I presume you came back to that as you did Elements 2, 3 and 4.

DR. SAWANT: Yes -- like, we were talking in closed session. Like, there is some phenomena that we had a different understanding when we started developing methodology and when you finish the analysis. So we will talk about those kind of things.

DR. SCHULTZ: Thank you.

DR. SAWANT: This slide just gives very intense safety systems. I think almost everybody probably is aware of the -- how safety systems -- NuScale safety systems work.

ECCS basically consists of three reactor vent valves located at the top of the RPV and two reactor recirculation valves located in the liquid

part of the RPV on lower relation and when ECCS activates all these valves open and the primary function of these ECCS is to depressurize the RPV so that we can inject the coolant that is collected in the containment back into the RPV and develop it.

And the actuation signal for ECCS is a high containment level. It also gets actuated on loss of AC power. But there is a timer, like, 24 hours. And there is a feature called inadvertent actuation block which is there to prevent the inadvertent opening of any of these releases as well when the reactor pressure is very high, and these blocks they release the ECCS once the differential pressure between RPV and CNV goes below some certain set point.

And model prediction system function is, like, basically, to make sure that reactor scrams and steam generator containments are isolated and safety systems are activated. This can be ECCS and DHRS.

Decay heat removal system consists of blueprints each designed to remove 100 percent decay heat and just wanted to note that we are not crediting this additional system in our LOCA analysis -- LOCA evaluation model and we will show that that -- that that approach is conservative for LOCA analysis.

CO-CHAIR MARCH-LEUBA: You go back --

probably the question is for Ben, or maybe you know. I was asked before earlier and I didn't remember. What action of the model protection system triggers the DHRS passive cooling -- turning it on?

I mean, the -- I know loss of power will do -- will be one. But if not loss of power, what triggers it? Is it high containment level or --

MR. BRISTOL: So the containment will isolate on high containment pressure after the break occurs and then DHR.

CO-CHAIR MARCH-LEUBA: High pressure?

MR. BRISTOL: High containment --

CO-CHAIR MARCH-LEUBA: Oh, high containment pressure.

MR. BRISTOL: Yeah, high containment pressure so which is still well below atmospheric conditions. So we reached that actuation very quickly under --

(Simultaneous speaking.)

CO-CHAIR MARCH-LEUBA: Under the DHRS automatic critical -- I mean, DHR is activated by high pressure containment?

MR. BRISTOL: So the secondary isolates on high containment pressure and then the DHR will actuate on high secondary pressure after the isolation

of the secondary.

CO-CHAIR MARCH-LEUBA: I knew that. I just -- somebody asked me and I couldn't remember. It's convoluted but we will include this.

DR. SAWANT: So Element 1 of the Reg. Guide 1.203 which is establishing the requirements for the evaluation model as a first step you basically develop the PIRT and PIRT processes are quite well established.

Basically, we divide the NPM into different systems and complements so that we can identify high-ranked phenomena in each phase of the LOCA progression and this PIRT development was done by organizing the panel of experts -- known experts in thermal -- area.

DR. CORRADINI: Can you remind us who those are?

DR. SAWANT: Analysis presentation we will a list of those experts.

DR. CORRADINI: Oh, okay. Fine.

DR. SAWANT: So I will defer that.

DR. CORRADINI: Okay. Thank you.

DR. SAWANT: There is a long list so --

MEMBER BLEY: Page 42.

DR. SAWANT: Yeah.

MEMBER BLEY: Page 42 of the topical gives a list of 16 of them. Yeah.

DR. SAWANT: So because summaries with our acceptance criteria for the LOCA methodology and we identified the phenomena and ranked them for their importance based on their impact on the PR submerator (phonetic) into three different importance levels and we also identified knowledge level, like, based on the -- how well they are understood and what kind of assessments they have at the level.

So this next slide basically shows the -- how we break down our system into different components for development of PIRT and then also a typical progression of LOCA in this design.

This is -- break progression where we show the depressurization of RPV and pressurized from the CNV Basically, on the left side you can see that RPV is divided into low plenum core, a riser for plenum and a downcomer and pressurizer to identify the phenomena. And on the right side you can see that Phase 1 is basically the blowdown phase where RPV depressurizes and CNV pressurizes.

Once ECCS opens on ECCS, based on CNV level activation, there is significant depressurization of RPV and significant pressurization

of CNV.

That is the initiation of Phase 1(b) which ends when there is a recirculation established from containment to -- back to the RPV because initially when ECCS opens there is a flow from RPV to CNV through all the walls, like RVV -- RRVs, but once pressure is equalized and there is enough high-risk straticate (phonetic) in the containment flow in the ECCS and that's the end of our Phase 1(b).

So we considered, like, a Phase 1(a) and 1(b) as a part of LOCA analysis and Phase 2, which is the long-term cooling analysis, which is -- covers the long-term cooling analysis methodology.

Element 2 of the Reg. Guide 1.203 is like establishing the assessment base. As I said earlier, this involves, like, developing separate and integrative assessments. But before we talk about the assessment base, I'd just like to review some background on our mathematical model that we have selected.

Our thermal -- system thermal hydraulic code that we have selected for LOCA analysis is NRELAP5. It's developed from the RELAP5-3D as a baseline for it and I have some sub-bullets there that gives basic information of RELAP5-3D, which is I think

well known thermal-hydraulic code.

We have established a code configuration control as it relates all the development to convert RELAP5-3D. The NRELAP was done under NuScale's NQA-1 program and these -- we'll talk about some of these modifications in closed session.

But these modifications that we did in RELAP-3D are mainly related to model NuScale-specific components and some modifications so that we can implement Appendix K models and corrections that were reported as part of RELAP5-3D program.

So the next slide shows just overview of our assessment base -- basically, database. We have -- as we will see in closed session we have extensive coverage of all of our high-ranked phenomena.

There are six NuScale-specific integral effect tests that were performed in NIST facility. There are two separate effect tests that also were performed in a NIST facility. Four of the separate effect tests were -- these are also NuScale-specific.

Facilities includes helical coil simulator test facilities, SIET, and CHF test facilities, KATHY and Stern. So there is an extensive set of NuScale-specific data for evaluation of evaluation models.

We do have a comparison against nine

legacy separate effect test data. These are the datas that we are offering from the existing literature because we found that they are applicable for our design. So our database includes those nine tests also.

Next slide. Since this NIST facility is our -- one of the primary source of separate effect as relating to -- effect data there's just a little bit more datas of the design features of the facility.

This facility has all the important components of the NPM are reprinted here. RPV has basically a lower plenum core riser, downcomer pressurizer, represented as a integral part of reactive pressure vessel.

RPV is located outside of the containment for scaling reasons and it's connected to the containment by ECCS walls and break -- by ECCS lines and break lines.

Containment -- that you see that new vessel is connected to the reactor pool, which is represented by green vessel there by a rectangular heat transfer plate, which it scales the NPM active containment heat transfer area, and there is this shell -- for the containment that is there to contain the pressure. So I just want to highlight that that

heat transfer plate is the scale of the area that connects the containment to the pool.

And, yeah, the scaling -- a little bit scaling information, scale it's a power rolling (phonetic) scaling used for designing the facility. It's various height, various volume scale. Pressure temperatures are one-to-one scale and time scale is also the same, one to one.

CO-CHAIR KIRCHNER: Could you, with the mouse, point, Matthew, perhaps to where the containment is on that -- on that figure. On the right. Yeah, it's there. But now go over to your diagram. Yeah, there.

DR. SAWANT: Okay. This is -- yeah.

CO-CHAIR KIRCHNER: So that's the containment there. So is the gap in the containment scaled or is that prototypical in terms of -- where I am getting at is the distance between the reactor vessel and the containment wall. Is that prototypical or scaled?

DR. SAWANT: The volume is scaled.

CO-CHAIR KIRCHNER: No, I understand the volume is scaled. But is the gap -- the width, is that prototypical or is it scaled as well?

DR. SAWANT: So the gap in this limited

heat transfer plate that connects the --

CO-CHAIR KIRCHNER: No. No.

DR. CORRADINI: He's asking -- he's asking -- look at the blue. He's asking you've got a semicircle -- a semi-hemisphere or a semi-cylinder with the blue. He's asking the width of the blue, is it scaled or not. The blue which is the containment -- is the width of it scaled? The top -- the height I understand is --

DR. SAWANT: Yeah. Width of the heat transfer plate that connects the --

DR. CORRADINI: Not the plate.

CO-CHAIR KIRCHNER: Not the plate, but the actual --

DR. CORRADINI: Volume.

CO-CHAIR KIRCHNER: The annulus. Yeah, what simulates the annulus. What I am getting at is that you would have a turbulent mixing as you have a blowdown.

DR. SAWANT: Okay.

CO-CHAIR KIRCHNER: But that mixing, in effect, on the condensation and heat transfer is going to be to a certain extent impacted by how wide or that annulus -- you know, in the real -- in your actual NuScale power module that gap is like -- like this and

something -- I am just guesstimating. But how did you scale that? Is that kind of the same width or --

DR. KURAN: Well, this is Selim Kuran from NuScale.

The issue is that maintaining the required volume scale along with the heat transfer area as well as internal length scale, which is where we do hydrotherameter (phonetic) is very difficult issue at the same time.

CO-CHAIR KIRCHNER: Yes.

DR. KURAN: So you prioritize what you are scaling for. Obviously, volume comes first, and the same importance heat transfer area. So we are not scaling the internal length scale that would define multi-dimensional phenomena in this facility.

CO-CHAIR KIRCHNER: So it was a leading question because I am thinking of the impact on the condensation heat transfer based on turbulence during the blowdown and subsequent phases in that gap, and whether that is characteristic in terms of the phenomena with the actual NPM. You see where I am going with this?

DR. SAWANT: Yes, I think we can address that in closed session.

CO-CHAIR KIRCHNER: Okay.

DR. KURAN: Well, given the curvature of the surface in a prototypical geometry and how that is similar to, you know, flat plate, I think for that film condensation on a flat surface we are not off.

CO-CHAIR KIRCHNER: Yeah. No, I understand that, you know, because the diameter of the actual module containment vessel is so large you can effectively treat it as a flat plate, in effect.

But the gap becomes a factor in how the turbulence, how the phenomena, plays out during the blowdown and that would affect, you know, the condensation rate and that would affect how the pool that accumulates is heated or sub-cooled. You see where -- that is what I am asking.

DR. SAWANT: Yes, and then this distributed distribution of mixture in containment. While it is not high-ranked phenomena, we will talk about that in closed session.

CO-CHAIR KIRCHNER: In the closed session. Okay.

MR. BRISTOL: And in the assessments in the closed session you'll see that the containment level increases characteristic within time to the NPM.

CO-CHAIR KIRCHNER: So they are -- that's the volume part is --

MR. BRISTOL: That's right.

CO-CHAIR KIRCHNER: -- is consistent. Yeah.

MR. BRISTOL: Right. So the accumulation and the removal of that surface area is characteristically scaled.

CO-CHAIR KIRCHNER: Okay.

MEMBER BLEY: I have a much simpler question. When you did the PIRT you then went after the things that were of high importance and limited knowledge, I guess. You had four knowledge scales -- 1, 2, 3, 4, with 4 being we know this real well and you had a lot of 2 and 3 level knowledge, which is partial uncertainty, and no 1's that I saw, after you've done your experiments you have all of those up to 4's? Did you evaluate where you stand after you did all the experiments?

DR. SAWANT: So we -- so based on our assessment base and assessment of correlations and models, we can finally conclude that we have enough information to say that our evaluation model is applicable to model those phenomena

So yeah, we had final conclusions showing that's applicable.

MEMBER BLEY: You're well-bounded now.

Your knowledge is good in all of these areas that were --

DR. SAWANT: Yes.

MEMBER BLEY: -- questionable before. Okay.

DR. CORRADINI: So can I follow on Dennis's question?

You had three PIRT panels over six years.

DR. SAWANT: Yes.

DR. CORRADINI: A couple of them were the same.

DR. SAWANT: They were --

DR. CORRADINI: Of people. So did something drastically change after the three times that you did it? Or, to put it differently, why did you do it three times?

DR. SAWANT: So there were evaluation of the design --

DR. CORRADINI: So these are different designs they were looking at?

DR. SAWANT: Some minor changes to the design like how ECCS is functioning or it was changed from 2008 to 2015 and we basically captured those design evaluations.

DR. CORRADINI: Okay. By these --

DR. SAWANT: Yes, on the PIRT panels. We heard three PIRT panels over this period of 2008 to 2015 and each time we went through the same process to make sure that PIRT is consistent with the design.

DR. CORRADINI: But these experiments were done after the 2015 PIRT panel.

DR. SAWANT: Yes. Mm-hmm.

DR. CORRADINI: So the relevant one, if I were to look at one versus the other, would be the 2015 because the design was more stable by that time and the knowledge gaps -- the places where you needed to do something were based off of that.

DR. SAWANT: Yes. All the experiments that we -- we are going to talk about NIST -- NuScale's experiments that were conducted after 2005.

DR. CORRADINI: Okay. Fine.

DR. SAWANT: Except, like Stern CHF tests, they were performed earlier. But we will talk about those tests.

DR. CORRADINI: That's fine. I just want to make sure I understood the logic.

MEMBER BALLINGER: So the PIRT ranking table that Dennis has talked about you got heavy duty instrumentation in your experiment. After the experiments were there any surprises? Because

surprise sometimes is actually more beneficial than confirmation bias, if you want to call it.

DR. SAWANT: Mm-hmm. Not huge surprises. But there were some lessons learned that we implemented eventually in the evolution model development.

MEMBER BALLINGER: You're going to tell us about those?

DR. SAWANT: Yes. Yes.

CO-CHAIR MARCH-LEUBA: While we are wasting time, can you go back to slide seven?

MEMBER BALLINGER: While we are what?

CO-CHAIR MARCH-LEUBA: Wasting time. We are behind schedule. But we have all of tomorrow morning --

MEMBER BLEY: Well, you're not focused on angels on a pin.

CO-CHAIR MARCH-LEUBA: Okay. I am looking at the PIRT and I don't see any -- that the 16 experts identified volume distribution in the vessel and containment as one of the important characteristics of this. Is that correct?

DR. SAWANT: The PIRT was specific for LOCA, thermal hydraulic --

CO-CHAIR MARCH-LEUBA: This -- that's why

I asked you to go to level -- to seven you can see here. If you see under this condition where the water levels are, the core is boiling off concentrating boron inside.

The stimulator is condensing the still water and the still -- the steam is going into containment. So you have a high concentration of boron in the red region and a very low concentration of boron in the downcomer.

Eventually, you are going to flood the containment to recover your levels so you can refuel and you are going to restart the circulation and flush that cold unborated water from the downcomer. Was that identified in the -- in your PIRT?

DR. SAWANT: Not in the LOCA PIRT. It's part of --

CO-CHAIR MARCH-LEUBA: This is my favorite topic, right.

DR. SAWANT: It's part of long-term cooling analysis and I think --

CO-CHAIR MARCH-LEUBA: It was not identified in the long-term cooling analysis here because I have that data -- important because I asked for it. It was not. So somebody is going to address my favorite topic, right. Not here, not now, but

eventually.

MR. SCHMIDT: This is Jeff Schmidt from Reactor Systems. Yes, that'll be in the Chapter 15.

CO-CHAIR MARCH-LEUBA: I want to say in the -- I am going to suggest that the PIRT was slightly defective in the sense that it did not identify a very significant physical mechanism that's happening here.

And the problem I see is that both NuScale engineers and the staff reviewers we are putting our sight shields from -- like a donkey and looking specifically at very specific problems and are not seeing the whole thing.

And we are taking advantage of things that happened in operating reactors and we have -- where we have a lot of experience and saying that could not possibly happen in my reactor.

Therefore, I don't have to analyze it when, obviously, this reactor has particular behavior that does not occur in operating reactors like the rod ejection could create a missile that actively heats containment.

And I've been reviewing the references that the staff gave us and Section 351.1 says, no, it can't happen, and the response to RAI 92 -- 9647,

which Chris has been nice enough to give me, says it cannot happen so we are not going to analyze it.

Okay. I see a problem in the -- both the design and the review of another reactor in which we take all the advantages of the existing reactors, because if they don't do it we don't have to do it, and then all these advantages we say that doesn't apply to me either.

So I am complaining about it and I see two examples that came out this morning, and inside the vessel missiles that can be generated in a heated containment, which I agree with Mike, it is not going to do anything but you didn't analyze it. You said it's not possible.

I see the boron dilution event, the PIRT. They didn't even identify it. So I see deficiencies. Okay.

My five cents.

CO-CHAIR KIRCHNER: And just as a reminder, we are in a subcommittee meeting and you're hearing individual opinions, and we will speak to you through a letter report after a full committee meeting. So just keep these in context. Thank you. Thank you, Jose.

MR. BRISTOL: And I'll speak to a little

bit of context. There's several things that we have considered and licensing strategy comes into play when we are considering how we want to submit a particular safety evaluation. The dynamic effects of CRDM failure was considered within NuScale and it was determined the most appropriate strategy for addressing those dynamic effects was the strategy that you see in the topical or in the FSAR itself and, I mean, the -- I think a very simple evaluation of the momentum versus the pressure and the length to the top of the CRDM or to the top of containment would lead us to think that that's probably a fairly benign transience.

CO-CHAIR MARCH-LEUBA: It's not that I agree with you but a back-of-the-envelope calculation that is placed in the record or at least kept as an internal document in Corvallis would help your argument that yes, we did consider it. We looked at it and the maximum velocity was only so many meters per second that cannot do anything.

MR. BRISTOL: Sure. And, again, not all of that analysis that we have considered throughout ends up as part of our licensing submittal basis.

CO-CHAIR MARCH-LEUBA: All right. But your documentation FSAR says inside missiles it cannot

possibly happen.

MR. BRISTOL: That's correct.

MR. PRESSON: Well, at least, in terms of the -7th. But yes.

MEMBER REMPE: Your response says a simple calculation could show. It would provide more confidence if you'd said we did a simple calculation and found. Is that -- which happened?

MR. BRISTOL: We had considered a variety of different strategies and the strategy with all of the rest of the considerations specific to the dynamic effects of the CRDM failure was to go the route that we have submitted as part of the FSAR.

MEMBER REMPE: Well, that's one thing what you submit for your licensing strategy. But what I am asking did you do a calculation -- has it been done?

MR. BRISTOL: We have -- we have considered the surface areas in the DPs. I cannot say that we have specifically documented the acceleration effects and the -- and those dynamic effects because of the analysis strategy that we selected.

CO-CHAIR MARCH-LEUBA: So that sounds to me like no, we did not do our calculations. Let me --

MR. BRISTOL: And just to address the

long-term, the long-term cooling does consider the effects of boron transport. We identified within NuScale the limitation there was the accumulation of boron.

Now, the recovery actions we will discuss in two weeks.

CO-CHAIR MARCH-LEUBA: I cannot wait to see what the recovery actions are going to say because I don't see how you can recover.

MR. BRISTOL: Sure.

CO-CHAIR MARCH-LEUBA: And we will talk about it, but you're going to hear -- next month you're going to hear a lot about a two-space bin getting out of the tube. Once you separate the boron you cannot put it back in the tube. Okay. Let's keep going.

DR. SAWANT: So Element 3 of Reg. Guide 1.203 builds the development of a mathematical model. So, again, in closed session we will talk about the analysis that we did and justifications of how they selected nodalization, time slips, also biases that we are implementing or initial and boundary conditions and treatment of set points and trips.

So we will cover this information, again, in closed session. Other -- this element also

presents this local break spectrum calculations where we look at the -- several break sizes and locations, probability and single failures, and we will talk about those results in closed session.

And we also did several sensitivity calculations that we are required by Appendix K but also the sensitivity calculations that -- phenomena-specific to guide us on particular aspects of the methodology, and also to establish our conservative biases. So we will, again, like, introduce those calculations in closed session.

Then, finally, this Element 4, which is the applicability evaluation where we assess the accuracy of evaluation model with the bottom-up and top-down analysis. So, basically, in bottom-up analysis we look at every high-ranked phenomena and models and correlations needed to simulate those high-ranked phenomena.

So we identify important parameters for those models and correlations and what is the required range for those parameters in NPM local application so that we can evaluate the models and correlations for their applicability to all those range. So we look at the validity of the models and correlations, their assessments against external data that is reported as

well as any specific assessment that we have performed.

We also look at scalability of those models and correlations to our NPM design. So all of this information is summarized in LOCA LTR Table 8-1 and we will revisit some of these aspects in closed sessions.

For the top-down evaluation, we basically look at the field equations and numerics and its applicability to model LOCA operation. But a major part of top-down evaluation is to show the applicability of NIST integral effect test data through a simulator LOCA in NPM through scaling and distortion analysis. So we will talk about that aspect a little bit more later in closed session.

And, basically, we show that -- how the data -- the IET data is applicable and why NRELAP is still able to model the LOCA in the NPMs.

So, again, these are the conclusions. We will expand on these conclusions in closed session or we -- so the evaluation model has multiple layers of conservatisms in the addition to Appendix K, so we will talk about those conservatisms in closed session.

LOCA methodology that we are presenting we have developed here is cycle independent. Again, like, we

will talk about that in closed session. It's supported by extensive assessment database and so sensitive to calculations and follow Reg. Guide 1.203 guidance and what we have showed is that core doesn't experience CHF and it remains covered and there is no clad heat-up and continuing pressure, again, preserving its design limit. So that ends, like, my open statement.

CO-CHAIR KIRCHNER: Okay.

DR. SAWANT: But we helped, like, this -- opening up a reactor vent valve is also presented as part of LOCA LTR Appendix B, and extension of LOCA evaluation model to analyze that event. You will recall that event. So --

MR. BRISTOL: Thanks, Pravin.

So, just quickly, the purpose of the Appendix B is to describe the evaluation or the extension of the LOCA evaluation model for what's traditionally analyzed more like a non-LOCA event, i.e., an inadvertent opening of a reactor valve. In the NuScale design that is not an event that is re-isolated. Therefore, phenomenologically, it's more consistent with the LOCA event progression.

However, because of the event categorization there are some minor differences in how

we do the evaluation. So as part of the appendix there's a description as to how -- the events including the applicability of the LOCA PIRT and assessment basis, and then the applicability evaluation that Pravin just covered. However, the evaluation model is slightly different.

Next slide. So those differences are in order to be a little bit more consistent with our AOO evaluation techniques and specifically that's related to the data set that was used to assess the CHF correlation limit as well as the way that the power profiles are applied. Otherwise, the evaluation model nodalization containment modeling, et cetera, are consistent with the LOCA EM.

Next slide. So in conclusion, the IORV extension of the LOCA EM is justified because of the similarity of the transient production and the phenomena.

Again, there's some minor differences in the way that CHF is evaluated that's more consistent with an AOO event. The transient conclusions, the CHF occurs very quickly. It's driven by flow and pressure transient effects.

Reactor trip occurs nearly instantaneously in this event because of the depressurization rates

and the collapse level remains well above the top of active fuel for the event.

MEMBER PETTI: I have a question. Why is it classified as an AOO? Is it based on probability? Do you think that once in the light fuel plant you're going to inadvertently open this valve?

MR. BRISTOL: We do not expect to inadvertently open one of the valves in the life of the plant. It was a conservative characterization and that's consistent with what we evaluate as being industry precedent for a 15661 type of event.

MEMBER PETTI: Okay. A 1561 --

MR. BRISTOL: So that's the inadvertent opening of a reactor safety valve. In this case because the reactor safety valves and reactor vent valves are fairly similar, even though the ECCS analyses, as we know, have a very low likelihood of spuriously opening. It's a postulated event and we characterize it similarly.

MEMBER PETTI: Thank you.

CO-CHAIR MARCH-LEUBA: We have heard -- let me use two minutes. We have heard that the IEV or the RVVs or RRVs especially have some tolerances on their block/no block activation pressures and, therefore, it is very important to use the proper

staggering of opening of valves in the analysis.

Is that -- it becomes part of your methodology that we are approving today in this report or is that something that you consider once?

MR. BRISTOL: That would be the implementation of the SR. So the implementations calculations as part of the EM development we evaluate sensitivities to a range.

But the actual specifics of the range and the change that was made to the range is part of the Chapter 15 analysis and not part of the top core.

CO-CHAIR MARCH-LEUBA: So that would a question for the staff, which I'll ask you later on is if we found that the staggering -- the time of opening of the valves made a significant difference why wasn't it a limitation or a condition that for implementation of this EM the valve shall use the most conservative valve opening times according to specifications?

I mean, if you can do it between -- let me use some number parameters -- 500 and 700 psi. You should use either 500 or 700 on these -- and in the proper order.

MR. BRISTOL: So the methodology evaluates the IAV and identifies it as being important for particularly the minimum level scenarios, and then

specifies that as a sensitivity that's performed on a regular basis as part of the Chapter 15 event analysis.

CO-CHAIR MARCH-LEUBA: If we already know that opening -- if you have an IORV of an RRV, opening the second IORV first and then the RVV is worse. So the methodology should specify that that's what you need to do.

MR. BRISTOL: So that's worse in respect to peak containment pressure as we identified. So that would be covered as part of the technical report for containment pressure analysis.

CO-CHAIR MARCH-LEUBA: And for the next six months if you run the analysis you are going to do it that way, correct?

MR. BRISTOL: For the containment peak pressure analysis, yes. Yeah.

CO-CHAIR MARCH-LEUBA: But we are approving this topical report for the next 60 years. When you guys are working somewhere else and the new engineer comes in, if that is not specified in the methodology he may forget.

So, I mean, this is something that -- it should be a condition. If we know that it affects the results, it should be a condition in the SER. It

should have been in your topical report as your methodology and -- okay, I am just putting it in here that there might be a discussion of it as a condition to your SER.

CO-CHAIR KIRCHNER: I would disagree with you, Jose. If the methodology is robust enough to look at different valve sequences and their opening, then the limitation comes based on the containment pressure analysis or other places like Chapter 15, not in the methodology report.

CO-CHAIR MARCH-LEUBA: You know, it comes on the methodology. These type of limitations come always in the topical report.

It tells you whenever you use this methodology this is the order of opening valves that you have to use. You know, otherwise, if you -- if you use a different opening order you get no conservative results.

CO-CHAIR KIRCHNER: Well, you look at the various sequences of valves opening and you then come up with your peak containment pressure.

CO-CHAIR MARCH-LEUBA: Or the -- or the methodology. Look at the spectrum of opening times, which would be the best thing to do. But it has to be in the methodology. Otherwise, I want to run one.

And my concern -- I mean, these guys are going to do it right.

My concern is 60 years from now when the new guy comes in what is he going to do and that's why we do approved methods so you follow them and nobody has to question.

Yes. Okay. That will be for the staff.

DR. SCHULTZ: And what is the technical basis for removing the 15 percent stored energy bias that -- in this particular evaluation?

MR. BRISTOL: So we apply more of a cycle-specific fuel condition and so the LOCA methodology was developed considering a broad array of potential stored energy conditions based off of the FSAR core design.

DR. SCHULTZ: And the methodology is to be cycle specific in this particular evaluation?

MR. BRISTOL: And the IORV would be more cycle-specific. Correct.

DR. SCHULTZ: Thank you.

CO-CHAIR KIRCHNER: Any more questions from the members? Okay. I am going to propose that we take a break and return at quarter to 11:00 on the clock on the wall. And so we are recessed.

(Whereupon, the above-entitled matter went

off the record at 11:28 a.m. and resumed at 11:43 a.m.)

CO-CHAIR KIRCHNER: We're back in session.

We're going to turn back to NuScale staff again. And now we're going on to the non-Loss of Coolant Accident methodology.

So, please, Matthew.

MR. PRESSON: Thank you.

CO-CHAIR KIRCHNER: And introduce your colleagues.

MR. PRESSON: Yes. Again, I'm Matthew Presson, Licensing Project Manager for NuScale. We'll be discussing the non-Loss of Coolant Accident in this presentation.

And presenting this will be Ben Bristol, Supervisor of System Thermal Hydraulics; Megan McCloskey, our thermal hydraulics analyst; myself, and Paul Infanger will be available if there are any specific questions that come up, so.

I will pass it over to you all.

MS. McCLOSKEY: Thanks, Matthew.

Today in the open, in the open session we'll cover the scope of the topical report, starting with the non-LOCA events and the acceptance criteria and which of the acceptance criteria that are

addressed as part of the non-LOCA system analysis in the topical report, as well as those that are addressed by other methodologies that are part of separate review and approval processes. The interface with those methodologies is part of the non-LOCA topical report.

I have one slide on the factors of the NuScale power module design that really control the margins to the acceptance criteria for the non-LOCA event.

And then we'll move into an overview of the development of the non-LOCA EM starting with the PIRT and the GAP analysis done to identify phenomena that were specific to or more important to the non-LOCA system analysis compared to the LOCA EM and the RELAP, NRELAP5 assessment plan for non-LOCA. And then, finally, review the general event analysis methodology and the scope of the event-specific methodology that's identified in the topical report.

So first, what in-scope versus out-of-scope of this topical report were focused on the system transient analysis of non-LOCA events using NRELAP5. And the EM is applicable for the MTM plant design during the short-term transient progressing -- progression with DHRS cooling while the level in the

compression vessel remains above the top of the riser and natural circulation is maintained.

We describe the interface between the system transient analysis and the downstream sub-channel and accident radiological analysis, but the details of those EMs are part of separate topical reports.

And as we had already started to discuss today, we have NuScale topical report for control rod ejection and LOCA and valve opening events. So, those are not here.

We discussed the containment pressure temperature analysis technical report in as part of the Chapter 15 review with this subcommittee back in June of last year.

And, as I mentioned, the EM applies to the short-term transient progression where the riser remains covered. And that longer-term transient progression with DHRS that may result in a return-to-power event or riser uncovering is excluded from the scope.

In terms of the applicable initiating events they are summarized here. The events are classified as anticipated operational occurrences, unless they're identified as infrequent events for

postulated accidents. They're categorized into one of five categories, depending on the impact to the reactor module from the specific -- from the initiating event. So, what we have here is a subset of the Chapter 15 event that we have had some previous discussions on.

There are two NuScale-specific events that are covered here. One of those is a cool-down event, the loss of containment vacuum or containment flooding. And then one heat-up event is inadvertent operation of the DHRS.

MEMBER MARCH-LEUBA: And this is one of the good points of the standard review plan. I mean, they approach Chapter 15 real well by dividing into these five categories. Makes you think about it. And it's pretty comprehensive.

Really it's one of the, I will call it one of the few places where the SRPs will be helpful because it tells you what to do.

But do you think of any specific picture? Because here's what developed based on the experience with operating reactors. We have 50 years' experience, 60 years' experience with operating reactors and we know what can happen to them. Do you consider anything that is not an operating reactor?

NuScale has special control rods or special valves or special --

MS. McCLOSKEY: When we went through this, the identification of design basis events for the NuScale power module design we considered the systems that are specific, we considered the NuScale system and components and what effects those failures could have on the modules. And what we found through that systematic review process was that our scope of initiating events look a lot like a typical PWR.

And then we have event progressions that maybe that are somewhat un -- that are unique to the NuScale design.

The long-term cooling with ECCS event progression is a little bit unique because we, with the boiling condenser condensing design of the ECCS in containment will tend to redistribute flowing into the core. And we don't have a mechanism as part of the safety system design to require that operators take action to prevent accumulation, further accumulation of water in the core, while we treat it as part of the recovery processes but we don't have a hot rod switchover time.

So, we also considered the extended operation of DHRS. And during the development process

what we found there was that with respect to events that have -- with respect to margin to acceptance criteria our limiting conditions are in the short-term part of the event progression. And so that's part of why the focus of this topical report is on the short-term event progression.

MEMBER MARCH-LEUBA: Okay, thanks.

DR. CORRADINI: So, but I guess Jose's question I thought you were going to answer it by saying the two italicized things are the two things that make this unique, everything else is pretty much -- So, are you coming back to those specifically and we're going to talk about those later? Or should I ask now?

MS. McCLOSKEY: Go ahead and ask now because I'm --

DR. CORRADINI: Well, I'm kind of curious about how the loss of containment vacuum and/or containment flooding would be initiated.

MS. McCLOSKEY: Well, loss of containment vacuum since during normal operation the containment operates in sub-atmospheric conditions --

DR. CORRADINI: Right.

MS. McCLOSKEY: -- the loss of containment vacuum may occur due to a failure in the containment

evacuation system. But maintaining that sub-atmospheric condition during normal operation, containment flooding is postulated.

DR. CORRADINI: But they're separate, they're separate events. It's a common system.

But what I was trying to get at is I wouldn't have one necessarily concurrent with the other.

MS. McCLOSKEY: Correct.

DR. CORRADINI: Okay.

MS. McCLOSKEY: Correct.

DR. CORRADINI: All right.

MS. McCLOSKEY: We kind of lumped them into the same event analysis. But the causes are different.

DR. CORRADINI: Okay.

MS. McCLOSKEY: And there's specific calculations that are addressed are different RELAPS.

DR. CORRADINI: Okay. Right. And the inadvertent operation of DHRS is just early, early switchover to DHRS?

MS. McCLOSKEY: There is a couple, there's different permutations of that event. It may be inadvertent opening of one of the DHRS actuation valves. It may be inadvertent actuation of one train

or two trains.

And we've also considered inadvertent secondary side isolation of one train or two trains as part of that event.

DR. CORRADINI: Okay, thank you.

MS. McCLOSKEY: Then this slide shows the non-LOCA event acceptance criteria. And these are really by just the range of acceptance criteria that are defined in the SRP and in the NuScale DSRS for AOOs and postulated accidents. We've also included the infrequent events acceptance criteria, which is consistent with the radiological acceptance criteria that's discussed in the SRP.

The criteria in bold are those, the maximum RCS pressure and the maximum steam generator pressure that are addressed by system analysis.

And we also consider, we also evaluate the escalation criteria. We demonstrate that an AOO doesn't postulate an accident or constitute a loss of functionality.

DR. CORRADINI: So, so I know that you guys have told us this, but I forget. Infrequent event versus accident, I'm looking at all the criteria. They're essentially the same. So, what am I missing as to why the differentiation?

MS. McCLOSKEY: Well, radiological dose criteria are different --

DR. CORRADINI: Ah, okay. Excuse me.

MS. McCLOSKEY: -- in a small fraction of the accidents. That's critical.

DR. CORRADINI: Okay, thank you.

MS. McCLOSKEY: So this slide, again, shows the general approach of the NuScale methodologies to demonstrate that the range of acceptance criteria for the non-LOCA analyses are met. We take the plant design in combination with some performance into the NRELAP5 model of the plant and evaluate the pressure and primary and secondary pressure limits, and demonstrate that we face stabilized condition.

Boundary conditions are provided to the downstream VIPRE analysis or for the radiological dose analysis, which are, and the methodologies for those are in separate topical reports.

And this slide gives a very high-level overview of the characteristics of the design that governed margin to acceptance criteria for the non-LOCA transient. For minimum critical heat flux ratio there are key features during the design where we have an overall low power density. And as a result of that

general operation it's the combination of high temperature, high power, high pressure conditions that result in the limiting NCHFR conditions during reactivity insertion events as compared to more rapid power overshoot reactivity insertion events.

Our primary pressure is protected by the reactor safety valve lift. There are two safety valves on the reactor pressure vessel, and they're non-LOCA analysis for the maximum lift pressure and the minimum capacity of the first valve to maximize the transient pressure calculation.

For secondary site pressure the design pressure is equal to the primary site pressure. And the secondary pressure keeps after DHRS actuation of the boiling condenser mode coolant for the DHRS is established. So, it's really limited by the primary site temperature conditions.

And then in terms of radiological release, the module protection system is the one to rapidly automatically isolate the source of the release based on the measured conditions.

And, finally, in terms of establishing stable condition, the module protection system is designed to actually isolate at least one steam generator sufficient for it to support the DHSR

operation and support due to an event.

MEMBER MARCH-LEUBA: Some are complaining it doesn't affect the non-LOCA methodology, but on radiological releases we still want to hear more about the procedure to de-isolate containment at 72 hours to make sure of the hot oxygen content. So, we are expecting to hear more about that beginning in the full committee next month.

And then the last one, establishing a safe, stable condition, I keep hearing from people that a return to power is not a safe, stable condition. Maybe safe but certainly not stable.

So, sure, I mean, we're using here nomenclature. And I'm complaining about semantics. But that makes it different than other reactors to have a weakened return to power. And that was just justification for complaining.

MS. McCLOSKEY: The non-LOCA evaluation model was developed to perform a conservative analysis following the intent of Reg Guide 1.203, and applying a graded approach to the evaluation model development and assessment process.

As the first element of the EMDAP includes establishing the applicable transience and the acceptance criteria, which we've just discussed. And

then what the transience and acceptance criteria identify we still develop the non-LOCA PIRT. This was done towards the end of 2015, beginning of 20 -- sorry, end of 2014, beginning of 2015 timeframe, so the design had progressed fairly far along in the development. And we were able to take advantage of that when evaluating the PIRT.

Then we did a GAP analysis that examined the high-ranked phenomena identified in the non-LOCA PIRT as compared to the LOCA PIRT and the assessment plan for the NRELAP5 assessment plan as part of LOCA. Considering that overlap we determined that a graded approach to the EMDAP was reasonable, specifically in that the LOCA topical report describes the NRELAP5 code and the validation basis for LOCA since the premise of the graded approach was that that topical report would be approved. And the non-LOCA topical report leverages that code description and much of the validation basis described without repeating that information.

And so that the results of the GAP analysis allowed us to focus on the additional NRELAP5 assessment and sensitivity calculations to focus on non-LOCA phenomena events predominantly in the steam generator and the DHRS validation.

I'll have some additional about that in the closed session tomorrow morning.

Finally, in the non-LOCA EM suitably conservative emission boundary conditions are applied for several parameters to bound the plant response.

And we'll talk as well about sensitivity calculations to demonstrate the fact there's control of margins to acceptance.

This slide summarizes the PIRT developments, since the non-LOCA events are categorized into different event types. We have several of them to consider.

What we did for the PIRT development was to fairly generically characterize the phases of the non-LOCA transient progression.

Phase 1 is the pre-tripped transient where the RCS was still near full power conditions or higher power conditions, RCS flows are high. The DHRS hasn't been actuated yet. The figures of merit during this phase are critical heat flux ratio and RCS pressure.

Phase 2 is the post-trip transition from reactor trip and DHRS actuation while the DHRS is started but cooling is not fully effective. So, the module is in a transitional state of flow and power level is reactor trips.

MEMBER PETTI: He's having trouble hearing you I think, our transcriber. So, make sure you're --

MS. McCLOSKEY: Okay. I'll try to speak a little more directly into the microphone here. Thank you.

And then you'll see CHF, critical heat flux ratio was a figure of merit throughout all of the phases. RCS pressure and secondary pressure were also figures of merit here as the DHRS gets actuated and starts to pressurize the secondary system. For events that results in breaks inside the containment on the main steam line and the feedwater line. We also considered containment pressure.

And then those three reflect stable, natural circulation under with DHRS cooling fluid only effective, and we're at low flow levels, and decay heat power levels.

Under this condition, at least in this phase the RCS mixture level was considered important because with sufficient DHRS cooling and the right initial mass inventory in the RCS, the DHRS coolant can result in riser uncovery and interruption of natural circulation. And at the end of Phase 3, that's where we find the end of Phase 3 in the PIRT, and the cooling could result in reactivity insertion,

and therefore we also considered some sub-criticality during this phase.

MEMBER BLEY: In your report I don't recall that you gave the names of the people who participated in this PIRT. Did you?

MS. McCLOSKEY: We did not include that in the --

MEMBER BLEY: You did in the other report, in the LOCA report.

MS. McCLOSKEY: -- in the non-LOCA topical report, right.

MEMBER BLEY: Yes. Any reason? Same people or was it always your people?

MS. McCLOSKEY: Different people. Predominantly NuScale internal experts familiar with the plant design and the transient response. It was also reviewed by some folks with broader industry experience outside of Nu -- from outside of NuScale.

MEMBER BLEY: Any reason you didn't identify them this time? It seems odd to me, that's all.

MS. McCLOSKEY: Didn't consider it necessary to include in the topical report in this space.

MEMBER MARCH-LEUBA: You can ask for an

RAI. Well, we cannot ask for it. But we can ask for a supplement to this presentation --

MEMBER BLEY: Right.

MEMBER MARCH-LEUBA: -- to give us a list if you feel it's important.

MEMBER BLEY: It just feels funny. You're counting on experts and you don't say who they are.

MS. McCLOSKEY: Yes.

MEMBER MARCH-LEUBA: In your opinion, Megan, was this intentional or was this an oversight that you didn't think it was important?

MS. McCLOSKEY: I think that the PIRT development was quite thorough and detailed. And that that's evidenced in the summary of the PIRT that's described in the topical report, and didn't rely necessarily on the specific individuals involved.

MEMBER MARCH-LEUBA: What I was telling Matt during the break is it's easier to get a scoping calculation done than justify why you didn't do it.

MS. McCLOSKEY: Uh-huh.

MEMBER MARCH-LEUBA: How much work will it take to provide us, send us a list this afternoon and you give it to us?

MS. McCLOSKEY: I have the, I have the PIRT and I can show you the, the committee and the who

reviewed it.

MEMBER MARCH-LEUBA: It's easier to get it done than to justify why you didn't do it. I mean, it will take you, what, 10 minutes? So, and you can put it on the record this afternoon when we're in the closed, closed session.

MS. McCLOSKEY: Right.

So, after the PIRT was developed, and again this was a fairly comprehensive and integrated PIRT that covered the different ranges of event types and the phases, and it was -- and comprehensive in terms of the range of phenomena addressed that were focused on the non-LOCA events themselves and not specifically for the system thermal hydraulic response.

So, after the PIRT was developed we did a GAP analysis to determine how we would address the high-ranked phenomena identified. And that can be -- the way we addressed it falls very broadly into four categories. There are phenomena where the validation performed as part of the NRELAP5 assessments for LOCA is considered efficient -- sufficient and applicable.

We identified where additional validation or benchmarks for non-LOCA were warranted, primarily in the steam generator and DHRS regions.

Several phenomena were focused on or are addressed by the downstream sub-channel analysis.

And then other areas we bounded the issues of conservative input.

So, here it is identified from the GAP analysis, again related to DHRS modeling of heat transfer. The topical report describes validation against the KAIST test and the separate effects test done at NIST for full DHRS heat removal, as well as plant sensitivity calculations we'll discuss further in the closed session some more for steam generator modeling and heat transfer, the validation based on the large-scale testing done at the CX facility for the secondary side. Heat transfer and pressure drop in the helical coil steam generators. And the fluid heated test as well.

We did a reactivity event responses, more important, can be more important in some of the non-LOCA events. We also did a benchmark against the RETRAN-3D code.

And, finally, we looked at the integral response with testing done at the NIST facility.

MEMBER MARCH-LEUBA: On the steam generator heat transfer coefficient, DS-2 -- you know, my favorite topic is on the secondaries, so I could

spend a whole week running TRACE calculations on reviewed and verified on containment body only.

But what I'm seeing is that the power transfer from primary to secondary installations oscillates a lot in a single -- on each two. Right? And RELAPS 1300 might be something different.

But the other it wasn't seemed to change much, if anything it increases. Did you guys even try to -- because now the design we have on the table says we don't care if the secondary oscillates. Indeed, we expect it to oscillate. However, did you keep this coefficient like you mentioned in DS-2 agrees with what I'm calculating with TRACE? Or did it change a lot during oscillations? Did you even attempt to measure it?

I'm sure you measured it in the steady state.

MS. McCLOSKEY: My recollection of what we looked at as part of the stability topical report was that there's a relatively small change in the overall heat transfer coefficient that we observed based on the DS-2 analysis results.

DR. CORRADINI: That was the Appendix A?

MS. McCLOSKEY: I believe so.

MEMBER MARCH-LEUBA: But if you assume

that the flow oscillations are uncorrelated two to two, when you do the other, you get the square root of N, and then when it is, well, don't assume it isn't, and it does not transfer to the core.

MS. McCLOSKEY: Right.

MEMBER MARCH-LEUBA: But the heat transfer itself, if you put it to oscillate, it goes like that. You can see it. You can plug the power in TRACE and it oscillates just as much as the flow does. Behind that, so the whole oscillation doesn't change that much.

MS. McCLOSKEY: And we have some additional discussion in the closed session about the effect of the primary to secondary side heat transfer in the steam generator and the effect on the initial safety conditions as well as the non-LOCA transient progression. And what we observed in the range of sensitivity calculations is that it affects where your steam generator is operating initially, but doesn't have a significant impact on your margins to acceptance criteria.

MEMBER MARCH-LEUBA: My point is if the design says it is going to be unstable and going to be oscillating like crazy, your methodology should assume that. And I don't think it does.

MR. PRESSON: And I would clarify that we don't -- our design doesn't present itself as oscillating on the secondary side like crazy. But we do join that to a DWO. It's part of our assessment to protect against that if it occurs, but the intent is not for that to happen.

MEMBER MARCH-LEUBA: Yeah, but I haven't seen any evidence that that's --

MR. PRESSON: Yeah. To be hopefully resolved.

CO-CHAIR KIRCHNER: I think what works to their favor for most of these transients is that, you're right, even though there is a lot of oscillation it's masked because you've got so much thermal inertia on that side.

MEMBER MARCH-LEUBA: On -- yeah.

CO-CHAIR KIRCHNER: So, individual tubes having oscillating heat transfer properties is a concern from fatigue, and wear, and vibration, and other matters but probably for these analyses the large volume in the primary system just will mask it. And it won't be a critical factor.

MEMBER MARCH-LEUBA: The first thing they do is trip the reactor and isolate the secondary. So, it really doesn't have an impact on the AOOs. And we

have so much margin.

But, again, it's easier to have done it and say, see, it doesn't have it, than say, no, I don't want to do it. Again, that's philosophy.

And again, with the DHRS same problem. You did a lot of analysis with non-condense -- the biggest problem with DHRS, there are two problems with DHRS operating: non-condensables that block the steam from going from one sector to the other; or over filling.

Did any of the tests that you benchmarked against have non-condensables in them? Because over filling is something very different.

MS. McCLOSKEY: Non-condensables were not specifically introduced into the tests as an objective of the tests that we validated against.

In the DHRS design there is a level switch that is -- that will indicate what if there is sufficient stimulation of non-condensables towards the top of the DHRS that it would expect to be inoperable. So that will be recognized --

MEMBER MARCH-LEUBA: The key purpose of the DHRS you assume your non-condensables are to limit to each level, in that you have non-condensables just above the trip. You should assume that; right?

MS. McCLOSKEY: What we, what we did --

MEMBER MARCH-LEUBA: I know I've seen some sensitivity to condensables.

MS. McCLOSKEY: That was done as part of the DHRS thermal hydraulic analysis and as part of the design development. What we see in non-LOCA transient events is you buy that DHRS heat transfer plus or minus 30 percent is not having a significant impact on our margins to acceptance criteria. So, we haven't specifically initialized with non-condensables in the calculations.

MEMBER MARCH-LEUBA: Just put in my concerns. It can fail, too. It's just highly unlikely to.

MR. BRISTOL: Yeah. And just to build off of the thermal hydraulic analysis demonstrated that for the period of time of the non-LOCA short-term transient a\progression the system pressures are much too high that that mass concentration is negligible. Therefore, it is not specifically added as part of the modeling techniques in the non-LOCA EM.

MEMBER MARCH-LEUBA: Well, the AOOs do not assume the worst expected conditions for non-condensables?

MR. BRISTOL: We don't explicitly model

non-condensables in that loop for the AOO event analysis.

MEMBER MARCH-LEUBA: Yeah, I'm not saying you model there. But assume the heat transfer calculation of DHRS assuming the worst tech spec non-condensables possible. That's what tech specs are for. I mean, you have to demonstrate that you have -- I can only have this much non-condensables, and that's what I'm assuming on my, on my analysis.

MR. BRISTOL: Yeah. I think I'd answer that with we've done some pretty extensive sensitivities looking at the influence of pool temperature and the initial steam generator mass on the DHR performance, and generally concluded that the DHR performance isn't very sensitive for figures of merit.

So, we stopped. We stopped once we'd reached that conclusion.

MEMBER MARCH-LEUBA: Okay.

MS. McCLOSKEY: So, our overall conclusion with the work done to support the applicability is that the code with the MTM system model is applicable for calculations of the MTM non-LOCA system response.

The overall non-LOCA transient analysis process is described in the topical report Section 4.

And this is fairly straightforward and typical.

We developed a plant-based model in the NRELAP5 that includes geometry, the control systems, protection systems. That base model is adapted as necessary for the specific event analysis model, the malfunction that's causing the initiating event, and the desired initial conditions survives the transient response.

We've taken, we perform our steady state and transient analysis calculations in the steady state calculations. The topical report identifies a list of various steady state conditions. And those are prioritized depending on the transients of interest. And the results are compared to the intended design conditions, and the transients are performed as restarts.

We evaluate the results of the transient analysis calculations to confirm our margins to acceptance criteria for the pressure, primary and secondary pressure. And confirm we have an appropriate run time to demonstrate a safe, stabilized condition is achieved.

And then we identify cases, as applicable, for downstream sub-channel or radiological analysis. And in the sub-channel analysis the conservative bias

directions are to maximize reactor power as well as core exit pressure and core inlet temperature. And that's based on the conservative directions for the NuScale-specific CHF correlation as well as minimum RCS flow rate, which is typical.

As was mentioned earlier in the discussion about the topical report, we can use the NRELAP5 CHF calculation from a dummy hot rod model of a screening tool to reevaluate it to identify there weren't any cases, to evaluate downstream analysis.

And for events such as the reactivity event, uncontrolled rod withdrawal, single rod withdrawal, where our margin is more limited in CHF, we'll analyze more sub-channel analysis cases. And typically all of the sub-channel, all of the cases get sent to the sub-channel analysis to confirm we've captured the limiting CHF result.

Similarly for radiological analysis, we identified cases for maximum release and iodine spiking time.

MEMBER MARCH-LEUBA: Okay. Point five, second bullet, the word "may" is giving me a heartache. I mean, when you have a topical report to approve, the purpose was that you were going to place it in the licensing basis as a reference that can be

used for future design licensing safety analysis reports, so for cycle 21 they want to use this Dash-A methodology and submit it to the staff for review or for the approval.

And all the staff has to do is ask themselves did they use the Dash-A methodology to calculate the set points? That's how the system is.

I'm complaining that the methodology is not descriptive enough when you say it may or may not be used. It is used.

MS. McCLOSKEY: Yes. What we intended to indicate there was that for some events like a, like the increase in inventory events or CBCS malfunction that has a very limited impact on the overall module core condition, we may use -- we'll use the screening tool to identify one, maybe one limiting case against them for sub-channel analysis.

And that's sufficient to demonstrate for that event that it's not limiting with respect to CHF margins.

For other events, like reactivity events, we'll do more analysis into the sub-channels.

MEMBER MARCH-LEUBA: The problem I have is that the CHF correlation that you guys chose to implement for the non-LOCA calculation, it's

incredibly biased, I mean, it's grossly non-conservative. And we're not talking 10 percent here. Right, when we want to do assessment I can talk the number, but it is bad, bad, bad.

The only thing that saves you is that it seems to follow the same phase as the good correlations. But it's incredibly non-conservative.

MS. McCLOSKEY: And we're not using it to demonstrate margin to the CHF acceptance criteria. It's only to identify the trend.

MEMBER MARCH-LEUBA: So what I would like to see as a condition on the staff will see, it specifies you identify the top N by transience. And those N's you analyze VIPRE where N can be 1, 2, 5, 12. And we agree on what is reasonable. Or one from each of the categories or something like that.

To be an approved methodology it has to be well defined, and I don't think it is well defined. The way you have it is trust that it will work for Cycle 21.

And when the staff is seeing that section of this report all they need to do is check -- they tell me they use the Dash A approved methodology, I don't have to review it. And if the Dash A approved methodology is wishy-washy then I cannot do that.

And the staff doesn't want to be reviewing your analysis 21 years from now.

So that, I will bring that to the piece up here and we'll have to suffer with it. But I think that prescriptive it's easier for you, it's easier for them, it's easier for everybody instead of flexibility. Flexibility is bad.

And, of course, myself, I would have run every single VIPRE case because CPUs are cheap. And you should have it to alternators, so all you have to do is push a button and everything comes out. But that's in my -- I give my thoughts.

Please continue.

CO-CHAIR KIRCHNER: Let me ask one question since on that same bullet how much conservatism do you pile into this hot rod, this "dummy hot rod"? Do you load up all the biases, whether it's stored energy or initial conditions or whatever, and then use it, may use it, or use it as a screening tool? Or is it an average rod?

Hot rod leads me to believe that you put probably all of the EM biases into that particular analysis before you use it as a screening tool.

MS. McCLOSKEY: The hot rod is separate, a separate part of the RELAP model that has no

feedback to the system thermal hydraulic response. I would have to look up the specific biases that go into that.

MR. BRISTOL: Yeah, so, we applied the radial peaking typically, but we didn't necessarily go through an effort of tuning it so that it matched the actual margin calculations better because once we'd demonstrated that the trends were sufficient then that's where we, that's where we stopped in terms of the development of that model.

In the cases like Megan was alluding to, the reactivity events, the power changes as a function of the -- or the power distribution changes as a function of the initial condition. And there's some specific sub-channel techniques for how they address that. We don't try to adopt any of that and, therefore, that's a place where there's a lot more cases that are sent for sub-channel analysis.

MS. McCLOSKEY: All right. In the topical report, Section 7.1 discusses methodology that's generally applicable to all of the events in terms of basic conditions that are considered and prioritized and met. The treatments of plant controls, that if a non-safety related plant control system is a benefit to the transients but not credited in the non-LOCA

analysis, loss of power, the analyses consider whether power available or loss of AC, or loss of AC and DC power is more limiting for the acceptance criteria. Or the single failure of a safety-related component in the analyses. This is the section that discussed the founding reactivity input, predominantly the moderator temperature coefficient, the Doppler temperature coefficient, as well as the decay heat pump contribution and SCRAM worth.

And it discusses other parameters such as the valve characteristics and the modular protection system and the limits and response times.

No operator actions are required to achieve the safety functions, so that's identified in this section as well.

And then in the event-specific methodology subsections of 7.2 for each event, there's a description of the initiation and progression. We have put those criteria of interest which identify which criteria would be more challenged in this event and, therefore, more sensitivity calculations are done for that criteria compared to other events.

So, cool-down events or activity events, NHR is of interest. It's not really challenged in the heat-up events, and those would focus on the primary

pressure and secondary pressure response.

The limiting single failures and lower loss of power scenarios for the events are identified, or the need to submit a calculation is identified for the event as well. And similarly, in the initial condition biases and conservatism there's a table for each event of whether the initial condition is biased or if it's sampled in the analysis.

And examples are presented in Section 8.

So, overall the evaluation model was developed on a graded approach following the guidance in Reg Guide 1.203. It is applicable only to the MTM type plant design with natural percolation and helical coil steam generator and coil pressurizer meets the NRELAP5 code as we've been discussing to demonstrate primary and secondary pressure criteria are met, and that a safe, stabilized condition is achieved, and provides boundary conditions to downstream sub-channel and radiological analysis.

MEMBER MARCH-LEUBA: Thank you very much. I think we have reached the end of the open presentation from NuScale. You can go and enjoy being part of the audience in the back while we make the staff --

CO-CHAIR KIRCHNER: Well, we're at, we're

at 11:30. We're scheduled to have lunch at 11:30. Let's start with the staff after lunch.

So, we'll take a recess until 12:30 on this clock.

MEMBER MARCH-LEUBA: Which is now correct.

CO-CHAIR KIRCHNER: Yeah. And we're, just to calibrate everyone, we're about one presentation behind in terms of time.

So, with that, we're recessed.

(Whereupon, the above-entitled matter went off the record at 11:26 a.m. and resumed at 12:30 p.m.)

CO-CHAIR KIRCHNER: We're going to recommence our meeting and turn to Chris Van Wert from the staff. Please go ahead, Chris.

MR. VAN WERT: All right. Thank you very much. Good, I guess, afternoon now. So, yes, this is the staff's presentation. It's an open session presentation. Hopefully it will answer a few of the questions that you brought up this morning. And with that I will go ahead and get started.

Again, I'm Chris Van Wert. The reviewers were myself and Jeff Schmidt. Becky is our branch chief.

Just as a quick overview Rev. 1 of the

topical report was submitted just a few months ago in November. We do have a full committee scheduled for March 5th. And if there are no changes, we will be working on completing the dash A version of this topical report soon, but in the very least it will be done by Phase 6 of the DCA.

All right. As a high overview of what the staff's review included, we looked at -- we evaluated the analysis criteria, also the code suite that was used within the analysis methodology, the evaluation of the plant and cycle assumptions that were used, and also the rod ejection accident analysis methodology itself.

I do think it's important to just remind everyone that this does not include the actual licensing basis analysis that is contained in the DCA itself in Section 15.4.8, but they do include an extremely similar example calculation in Section 6 of the topical report, but it is not the licensing basis itself. In addition of course to the review of the materials provided we did audit some of the calculations that were used to support their application.

To talk a little bit about the analysis criteria, I want to give you a brief overview of what

they did and our evaluation of it.

For the most part they followed the SRP Appendix B -- sorry, SRP Section 4.2, Appendix B Guidelines guidance. The first item, reactor coolant system pressure, they kept with that guidance and used maximum RCS pressure must be less than or equal to 120 percent design pressure.

For fuel cladding failure they did make a couple of modifications, but they were in a conservative direction. The first criteria is in regards to high temperature at zero power conditions. They set their limit to 120 calories per gram for all cladding differential pressures whereas the SRP allows for some increased enthalpy values for smaller differential pressures. So it took the most limiting one and stuck with that for everything.

For intermediate and full power conditions they assumed failure for heat flux -- or local heat flux exceeds the CHF limits. It's also worth pointing out that whereas the SRP would allow you to fail that rod and use that as part of your fission gas inventory they do not allow it to move forward in the core design if they have a failure there.

CO-CHAIR MARCH-LEUBA: Hey, Chris --

MR. VAN WERT: Yes?

CO-CHAIR MARCH-LEUBA: -- the figures -- the one -- the slide they showed us this morning, the one that has two figures --

MR. VAN WERT: I'm coming to it.

CO-CHAIR MARCH-LEUBA: All right.

(Laughter.)

MR. VAN WERT: Two more slides and I'll --

CO-CHAIR MARCH-LEUBA: It's inconsistent with your numbers there, you know that?

MR. VAN WERT: Yes, and we were discussing that right before lunch. And I'll explain --

CO-CHAIR MARCH-LEUBA: Did somebody change their mind?

MR. VAN WERT: What was that?

CO-CHAIR MARCH-LEUBA: Did somebody change their mind?

MR. VAN WERT: There was an inconsistency which I will explain --

CO-CHAIR MARCH-LEUBA: Okay.

MR. VAN WERT: -- two slides from now.

CO-CHAIR MARCH-LEUBA: I'll patiently await --

(Laughter.)

MEMBER BLEY: We're still in open session.

CO-CHAIR MARCH-LEUBA: That was open

session.

MR. VAN WERT: Yes. Yes. Well, yes.

MEMBER BLEY: Yes.

MR. VAN WERT: But I did confirm with NuScale that I can mention certain numbers that are listed as non-probable within the report.

The PCMI failure threshold, which is what you're referring to, in Section 2.3.2 of their topical report they request the threshold as the figure B-1 from the SRP. That's what we initially did our review towards and that's going to lead to the clarification here in a second. And also failure is assumed at the fuel temperature anywhere, and the fuel pellet exceeds incipient fuel melting conditions.

To finish off some of the criteria, core coolability, they stuck with the same as the SRP. You must remain below 230 cals per gram maximum. And then fuel melt is not allowed.

For fission product, since they do not allow a core design that results in clad failure due to melting enthalpy increase or exceeding MCHFR limits, and also by incorporating some of the more limiting differential pressure-based limits, they preclude the need to follow the guidance that's provided in the SRP 4.2, Appendix B regarding fission

gas release. So they're a little bit more conservative in that perspective.

CO-CHAIR MARCH-LEUBA: This methodology, once approved, is going to be applicable to power uprates --

MR. VAN WERT: Correct.

CO-CHAIR MARCH-LEUBA: -- which may or may not come sooner or later.

MR. VAN WERT: Right.

CO-CHAIR MARCH-LEUBA: They will be frozen to this -- I mean these are more restrictive criteria than you would need.

MR. VAN WERT: Right. Right.

CO-CHAIR MARCH-LEUBA: It's easier to prove.

MR. VAN WERT: Yes, so the criteria -- we're approving these criteria. They're consistent with our guidance or more restrictive.

CO-CHAIR MARCH-LEUBA: Yes.

MR. VAN WERT: They do these analyses on a cycle-by-cycle basis, and we'll get a little bit more into this discrepancy here and you can see --

CO-CHAIR MARCH-LEUBA: I'm not talking the discrepancy.

MR. VAN WERT: Oh, okay. About the no-

fuel failures?

CO-CHAIR MARCH-LEUBA: Yes.

MR. VAN WERT: Yes, that's their criteria. So they cannot fail it and then --

CO-CHAIR MARCH-LEUBA: Are they married to this criteria for power uprates?

MR. VAN WERT: Until Rev. 1 comes in of the -- or Rev. 2 of the topical report comes in --

CO-CHAIR MARCH-LEUBA: Unless --

MR. VAN WERT: -- addressing that issue, they are married to it.

CO-CHAIR KIRCHNER: And by then the DG should be --

MR. VAN WERT: Pardon?

CO-CHAIR KIRCHNER: If they come back for power uprates, by then I expect the Draft Guide, your last bullet --

MR. VAN WERT: Correct. Correct.

CO-CHAIR KIRCHNER: -- will have been released.

MR. VAN WERT: Yes. So that leads into the discussion that we are aware that there is a draft guidance that's been in the works for a number of years now. It has been out in the public for comment multiple times, however, at the time the submittal of

the application the topical report, the guidance in-house in play was the SRP 4.2, Appendix B.

With that being said, we knew that there were going to be some questions. So there was a question this morning regarding kind of comparing apples to oranges when you have a corrosion-based limit figure and then a hydrogen-based figure.

Luckily I was able to talk with Tim Drzewiecki who is a colleague of mine. I don't see him in the audience right now, but he was willing to make a MATLAB script and using some oxidation limits that are calculated for NuScale and then the freed hydrogen from the zirc water reaction and then the hydrogen pickup information developed in -- or presented in -- I think it's also a Draft Guide, 4.224 or 6. One of those two. You kind of do a translation and put the SRP figure B-1 into the newer parameters.

I also want to highlight again that the blue figure here, which is from the draft guidance, that is literally draft and I do not want to stand behind it in saying that this is what would eventually become the final Reg Guide, but it just give you a flavor for trying to get as close to an apples-to-apples comparison as we could.

MEMBER BLEY: Just a staff procedural

question.

MR. VAN WERT: Yes.

MEMBER BLEY: Once the Draft Guide is a Reg Guide, the SRP's guidance to the staff on how to review the Reg Guide is guidance to applicants on how to kind of meet the regulation. Do you immediately make a change to the SRP or does the staff guy have to try to figure out which one to apply?

MR. VAN WERT: I mean immediate as immediate as the NRC moves, but yes, it would be something that would have to be updated as part of SRP 4.2. Yes.

MEMBER BLEY: Okay. But that might take years, yes.

MR. VAN WERT: Yes. I'm not saying the timeline, but it would be eventually done, yes. I think it would be done relatively quickly because that would be a pretty glaring discrepancy.

So with that in mind, if we were to look at this and -- yes, if you happen to have the topical report in front of you, you can look at tables 6-2 and 6-3, and that provides some of the analysis results for the sample equilibrium cycle analysis. Again, that's not the analysis of record, but that will give you an idea of where the numbers kind of fall. And I

think the maximum change in enthalpy was around 29 cals per gram. So I think there was a question earlier about how far within that dotted box they were. They're well within it.

Now as far as the question goes why was that dotted box on there versus me talking about the SRP line on here, the topical report in Section 2.3.2 requests the figure. They basically copied the figure from B-1, put it in their report, but it's the same figure. So that's I think figure 5-2. One or two?

PARTICIPANT: Two.

MR. VAN WERT: Two? Okay. So they request that, but then when you go to Section 5.3, they discuss how their maximum oxide calculated for their plant is significantly lower, and they kind of put a little bit of margin on top of that. And then they ran it up and came out to be around 75 cals per gram and said, okay, we're going to just live within this box so that it's very clear. It doesn't matter if there's a revision to the Reg Guide that comes out later. They will be in good standing for meeting all the future requirements.

I would not be surprised if later if they come in with a revision. They could change things, but as long as it lived within our guidance we are

fine. We can't impose a stricter limit without some heavy justification.

Any further questions on this figure?

(No response.)

MR. VAN WERT: Okay.

MEMBER BLEY: Just for the record we are aware of the status of DG-1327, so we will probably after this summer timeframe we look to schedule Mr. Clifford to come before us and present that.

MR. VAN WERT: Right.

PARTICIPANT: If he's still employed.

MEMBER BLEY: If he's still -- yes, well, part of that, we want to catch him before he goes with his body of knowledge, so to speak. Thank you.

MR. VAN WERT: All right. So a lot of the codes were discussed this morning that were used in the SPERT. In general as far as the valuation goes the analysis is based on mostly the Studsvik code suite, CASMOS5, SIMULATE5 for the core physics and the initialization, SIMULATE-3K for the 3D nodal kinetics, NRELAP5 for the transient system response, and VIPRE for the sub-channel analysis.

Most of these: CASMOS5, SIMULATE5, NRELAP5 and VIPRE, have been reviewed and approved in a separate topical report, however SIMULATE-3K, the

validation for that code was supplied within this topical report. So our review did include that.

We looked at their validation which was basically two parts. They validated against external data, the SPERT III experiments. They also validated against a benchmark from the NEACRP control rod injection problem, and the staff did it acceptable.

DR. SCHULTZ: Chris?

MR. VAN WERT: Yes.

DR. SCHULTZ: This is not the first time that the staff's reviewed SIMULATE-3K, correct? And other licensees --

MR. VAN WERT: Correct, but it was not --

DR. SCHULTZ: -- have utilized it?

MR. VAN WERT: Yes, other licensees. It was not --

DR. SCHULTZ: Right.

MR. VAN WERT: -- a stand-alone Studsvik product, so they couldn't just refer to an approved topical.

DR. SCHULTZ: Got you. I see. So it came in on --

MR. VAN WERT: Yes.

DR. SCHULTZ: -- a separate document, a different document?

MR. VAN WERT: Right, there's an operating plant or owner that --

DR. SCHULTZ: Yes.

MR. VAN WERT: -- for their fleet. The have seen that.

DR. SCHULTZ: I recall similar approaches in terms of the validation in the --

MR. VAN WERT: Right.

DR. SCHULTZ: -- application.

CO-CHAIR KIRCHNER: So, Chris, just as a follow-up on Steve's point, as a process question. So you accepted SIMULATE-3K as part of this TR. Does that mean SIMULATE-3K is approved as --

MR. VAN WERT: It's approved for use in NuScale's rod injection accident methodology.

CO-CHAIR KIRCHNER: And that's the limit of the approval?

MR. VAN WERT: Yes, and it --

CO-CHAIR KIRCHNER: So it's not broader? They couldn't come back say and say we want to use SIMULATE-3K for some other purposes or --

MR. VAN WERT: Correct. Correct.

CO-CHAIR KIRCHNER: Okay.

MR. VAN WERT: Now I don't want to --

MR. SCHMIDT: But they could come back

asking for approval for --

(Simultaneous speaking.)

MR. VAN WERT: They could always come back and request it --

CO-CHAIR KIRCHNER: Yes.

(Simultaneous speaking.)

MR. SCHMIDT: -- as a new application, but this is only applicable to NuScale rod ejection.

CO-CHAIR KIRCHNER: But I guess SIMULATE-3K does not have generic approval?

MR. VAN WERT: Correct.

CO-CHAIR KIRCHNER: All right.

MR. VAN WERT: Yes, I would normally expect Studsvik as a vendor to come in with the topical report and then try to sell that off to potential customers, but I have not heard of any potential plans of that coming in anywhere in the near future.

CO-CHAIR MARCH-LEUBA: And while we're interrupting you, this applicant talked about uncertainties in parameters.

MR. VAN WERT: Yes.

CO-CHAIR MARCH-LEUBA: And we have the opportunity of as part of the ACRS research quality review process -- which we review a couple of NUREGs

from research. We reviewed one on uncertainty of nuclear parameters. And the conclusion from the NUREG is that there is a tremendously high and unexpected uncertainty on beta effective. And the numbers published there based on the scale calculations are 20 to 100 percent error, which even the authors, when we confronted them, said that was probably an error on the -- in the ENDRB data.

PARTICIPANT: Or processing error.

CO-CHAIR MARCH-LEUBA: Or processing. I mean how confident are we that the beta effective that CASMOS produces is not 25 percent off?

MR. SCHMIDT: I mean we have been using that beta effective like from CASMOS3 to 5 for a long period of time. I don't know if I know the uncertainty associated with that parameter. I mean usually you look at beginning cycle and end of cycle type values, but I don't recall the uncertainty applied to beta effective here. This is what, table --

CO-CHAIR MARCH-LEUBA: Yes, back when I worked in ORNL they paid me to do an analysis of uncertainty, of how to generate cross-section for TRACE-PARCS. And I wasn't qualified to do the job, so I just passed the money to somebody else. And in that

NUREG, which I can give you a copy of, we found out if you do the weighting of it affected with fission power versus weighting with the flux, there was like a 15-20 percent error.

So the uncertainty was the effect issue. We should maybe -- I mean it's not 100 percent because we'll never be able to match a SPERT.

MR. SCHMIDT: Right. Yes.

MR. VAN WERT: And that's probably --

CO-CHAIR MARCH-LEUBA: But this is large.

MR. VAN WERT: I would expect that that would be in the Nuclear Analysis Codes and Methods Qualification Topical.

CO-CHAIR MARCH-LEUBA: Yes.

MR. SCHMIDT: The actual beta effective?

MR. VAN WERT: Yes.

MR. SCHMIDT: That's in the --

(Simultaneous speaking.)

MR. VAN WERT: Yes, I was trying to --

MR. SCHMIDT: Yes.

MR. VAN WERT: -- quickly look it up and I didn't see it. But since CASMOS5 I think was covered under that separate topical report, I would imagine that's where it is.

CO-CHAIR MARCH-LEUBA: It is one of those

parameters where it cannot possibly be that large, as EMDAP says, but other evaluations kind of point in the line that maybe. And it has a tremendous impact on this answer. I mean, you know what I mean? The direct impact, real fast. So I'm just putting it out there.

MR. VAN WERT: All right. Thank you.

CO-CHAIR MARCH-LEUBA: And I can -- we can find NUREG numbers to put them on the record. Let me look for them and I'll --

PARTICIPANT: I have it.

CO-CHAIR MARCH-LEUBA: Oh, you have them?

PARTICIPANT: Yes, they're within my notes.

MR. VAN WERT: You want me to continue? Are we looking up the --

CO-CHAIR MARCH-LEUBA: Continue. We'll wait. Get a break and put it on the record.

MR. VAN WERT: Okay. So the staff also reviewed the plant and cycle assumptions used in the analysis are within the methodology. We determined that the methodology included ranges in power, time of cycle, core power that averaged a wide range of operating conditions, and we captured the most limiting condition for NuScale. The staff also agreed

with the assumptions associated with the automatic system response of non-safety systems and that they were conservative and that the methodology for the timing of loss of AC power conservatively biases the RCS pressure evaluation.

And as far as the REA methodology, the staff reviewed the analysis methodology including steady-state initialization, the dynamic core response, the dynamic system response, sub-channel heat flux and the adiabatic heat-up fuel response. In that review the staff's review included the methodology by which information is passed between the codes, the application of the uncertainties and the modeling assumptions used for the inputs and handling of the reactor trips. We concluded that the methodology for calculating the system response, sub-channel and fuel response analyses were conservative and acceptable for demonstrating compliance with the acceptance criteria.

And in conclusion, overall for the staff's review we agreed with them that the REA methodology either follows or is more conservative than the staff guidance and that the methodology accounts for various potential operating conditions in time and life and conservatively addresses the uncertainties and plant

conditions. Therefore, the staff finds that the use of this topical report is acceptable for evaluating REA for the NuScale plant design.

CO-CHAIR MARCH-LEUBA: Is there something you can say in open session about the time in cycle where this becomes worse? Because we mentioned in open session this morning it wasn't at zero power, but it was more -- something else.

MR. VAN WERT: But which part becomes worse?

MR. SCHMIDT: Like what time in life? Like conditions?

CO-CHAIR MARCH-LEUBA: Yes. We can wait for closed.

MR. SCHMIDT: Yes, why don't we wait for NuScale? Yes, because I think they're going to present --

CO-CHAIR MARCH-LEUBA: Yes.

MR. SCHMIDT: -- analysis results.

MR. VAN WERT: They're presenting numbers later, so --

CO-CHAIR MARCH-LEUBA: But you have reviewed that and you --

MR. SCHMIDT: Yes, there was a whole -- there's a whole table on the top of the report that

looks at powers, times in lifes, EOC, MOC, EOC.

CO-CHAIR MARCH-LEUBA: And you're happy with their --

MR. SCHMIDT: Yes.

MR. VAN WERT: Yes.

MR. SCHMIDT: Yes, it was a full spectrum.

CO-CHAIR MARCH-LEUBA: Yes, I wanted to make sure in the public record.

MR. SCHMIDT: Yes.

MR. VAN WERT: Yes.

CO-CHAIR MARCH-LEUBA: You're happy with it?

MR. VAN WERT: Yes. Yes, we --

MR. SCHMIDT: That was the main focus.

CO-CHAIR MARCH-LEUBA: And in the closed record we'll have more fun with the numbers.

MR. SCHMIDT: Okay.

MR. VAN WERT: All right. Sounds great.

All right. Any other questions?

CO-CHAIR MARCH-LEUBA: Just wanted to put on the record since I was talking --

MR. VAN WERT: Yes.

CO-CHAIR MARCH-LEUBA: -- I found the NUREG report. There are two: NUREG-CR-7249 and NUREG-CR-7164. Those two address the beta effective

uncertainty and are coming from different points of view, but it's something that for -- maybe not now, but for future reviews you guys need to consider because it has the potential of being bad.

MR. VAN WERT: Yes, thank you.

CO-CHAIR KIRCHNER: Well, maybe a different way to say it without being pejorative is that certain concepts will be much more dependent on beta effective for their safe operation and shutdown. And so that's why we flagged it as a concern. And that will be reflected in our report to RES as part of our biennial review.

MEMBER REMPE: It's not voted on yet, so --

CO-CHAIR KIRCHNER: Not voted on, so --

(Laughter.)

CO-CHAIR KIRCHNER: We expect will be in -- as part of our letter report.

DR. SCHULTZ: Chris?

MR. VAN WERT: Yes?

DR. SCHULTZ: Just to go back to the basis of the staff approval, the discussion or the approach that NuScale has taken to assign uncertainties throughout their methodology and then the way in which they combine those uncertainties in order to create a

bounding evaluation, is that -- are those all specifically identified as part of their methodology that you have approved?

MR. VAN WERT: So some of this is actually coming from the Nuclear Methods Topical Report. So you're right that it is approved as part of REA, but that's because it's also referencing an approved topical report.

DR. SCHULTZ: It even goes further than that in terms of their overall methodological approach?

MR. VAN WERT: Correct.

DR. SCHULTZ: Thank you. That helps.

MR. VAN WERT: Thank you. I did my best to try to put you on schedule, or at least a little closer to schedule, but --

CO-CHAIR KIRCHNER: Thank you, Chris. Yes.

Members, any further questions of Chris or Jeff at this point? Because if not, we will move onto the LOCA evaluation model and the staff's review of the applicant's TR.

(No response.)

CO-CHAIR MARCH-LEUBA: And the same goes for you guys. Chris doesn't want to hold onto your

name tags, so save them for tomorrow.

MEMBER REMPE: So, Walt and Jose, while they're coming up I'm not sure if other members noticed it, but yesterday while we were getting all the new slides I noticed that the staff SE had changed. And I had earlier versions, and so I had reviewed those. But like the LOCA one had an increase of three pages. The non-LOCA had an increase of seven pages.

PARTICIPANT: Yes, I can explain.

MEMBER REMPE: Yes, I'd like some explanation on the record, because that seems a bit for editorial changes and I -- did anyone have a chance to really carefully compare what caused that increase in three pages and seven pages?

CO-CHAIR KIRCHNER: Well, maybe it's a question that we can direct to the staff. I can't answer that, but the agreement that we had to prepare for this meeting was that the staff would give us as near a final draft as possible mid-last month. And as to the actual extent of the changes --

MEMBER REMPE: I'd like to know what they are.

CO-CHAIR KIRCHNER: -- we can ask the staff to address that.

MEMBER REMPE: Yes, because I --

CO-CHAIR KIRCHNER: And maybe Becky can help us here.

MS. PATTON: Yes, right. So I will tell you that a couple of the changes that were made added an amazing number of pages when you did it. We had to add -- anywhere where we were referencing a figure number they wanted the full title of the figure in every single circumstance. So when you actually run a compare; and I have some file compare versions on my computer if anybody wants to see them, that added a huge amount. Every time we said we audited anything, they wanted the full ADAMS reference in there.

Other than that there were a few times where maybe the lawyers wanted a little bit more of a basis explanation, and so there was -- there were maybe a couple of times where there were a few sentences added, but there weren't like substantive technical changes in any of these. It was a large amount of changes due to editorial formatting things.

MEMBER REMPE: Okay. So I'm a trusting soul on this. The fact that you're saying on the record that there were no substantial changes I'll buy off on it, but I just was surprised at the length of the --

(Simultaneous speaking.)

MS. PATTON: Yes, right, that's correct. And, I mean, there are some explanatory type of basis statements that were added in various places. Like I said, I can -- we can make a compare file version available if anybody wants it.

MEMBER REMPE: I doubt any of the other members had a chance to go through the updated versions and so I just wanted to make sure we discussed it here today. So thank you.

CO-CHAIR MARCH-LEUBA: Since you offer, send us a compare.

MS. PATTON: Okay. Sure.

CO-CHAIR KIRCHNER: All right. So we're going to turn to the staff's review of the LOCA evaluation method.

And I'll turn to you, Carl, right? Are you heading up or --

DR. LU: It will be Shanlai Lu from NRR staff, Division of Safety Systems, and Carl is going to run the slides for me. Actually he can also answer questions. We have been working together.

This is one of the most heavily invested -- of the topical report from both NuScale and the staff side. So we are dividing this presentation to

two parts: The first part will be high-level summary in -- for the non-public -- the public session. And then we are going to have also proprietary sessions later, so in details.

So for this particular part -- and then I'm going to cover the -- the review team is here. And you can see that because it's a -- really a lot of effort from NuScale side, we started from -- almost 15 years ago started having the concept, developed this idea of the reactor design. So we had a lot of folks join together to form a team and then at different time people spent time on that.

So from NRC side we have Carl Thurston and Dr. Peter Lien from Office of Research, Dr. Weidong Wang actually from ACRS staff running the TRACE code in support of Office of Research, Ron Harrington from Office of Research, myself, Mr. Antonio Barrett. He worked on the CHF. Tim worked also on CHF. Syed Haider, because it's really related to the containment, the ECCS system. So he worked on this one, too. And then we have our expert supposed to be on the public line.

Do we have a bridge line open now? I don't know. Is bridge here?

CO-CHAIR KIRCHNER: Well, it's muted.

DR. LU: Okay. Yes, oh, it's muted? Okay.

All right. So we have Dr. Leonard Ward. He retired from the NRC and he has been a LOCA expert. And Dr. Don Rowe, a thermal hydraulic expert from IEEO previously. Now they all work for NuMark. And then Bert Dunn from Areva. He was the lead principal engineer on LOCA before he retired and joined the NuMark. And then we also have Upendra from Brookhaven National Lab. And then -- on the scaling. So that's the review team. And we had quite a lot of input from the -- our consultants and the national lab experts.

CO-CHAIR KIRCHNER: Let me interrupt you just for a moment --

DR. LU: Sure.

CO-CHAIR KIRCHNER: -- and for the record just say that Matt Sunseri, Chairman of the ACRS Committee, has joined us. Okay.

DR. LU: Okay. All right.

CO-CHAIR KIRCHNER: Please proceed.

DR. LU: So let's take a quick look of the design features and the scope. And actually NuScale provided more detail. They include the DHRS, but as part of a topical report since the LOCA methodology did not take the credit of the DHRS, so I simply this

ECCS system for the benefit of understanding what we are dealing with. So this is very, very simple, the simplest ECCS system I have ever seen from the PWR design.

Three valves on top and then two valves -- the reactor recirculation valves at the bottom. And you can see from the hydraulics perspective that's pretty much three valves into the containment. And the containment functions as part of ECCS. That's the features that I think -- I think when NuScale was presenting this part of the topical report, they talked about this one.

And then one of the ACRS asked questions, members asked questions. What's the -- really the big change or difference? I think that's the major difference. So our review actually also focus on the major difference. Especially in the RELAP5 code has been used so much extensively. And I think a couple years ago we just gave the Subcommittee a presentation on APR-1400, a large break LOCA. It's also RELAP5.

So let's turn to the next page. Okay. So what's the scope? And the NuScale asked for the approval for methodology to analyze NuScale LOCA scenario. That's number one. That's primary goal.

And after we interacted with them with

Revision 0 and then we added on one RAI which led to the change of the appendix, addition of Appendix B to provide a methodology to analyze the IORV. Okay?

And the methodology by itself and also the testing of the -- the NIST testing. And that supported the peak containment pressure analysis, so the RELAP5 code plus the methodology and then the base integral effect test supported the peak containment pressure analysis, LOCA analysis methodology too and the long-term cooling analysis. Although the approval of this -- they ask us to approve is not the approval for all those analysis and the methodology, but the base. For example, the geometry, the base code version are approved as part of this topical report. That was the application scope.

Okay. Next slide. So with such a large chunk of information coming in, so we decided to take the approach to expedite the review, but at the same time keep the sufficient adapt (phonetic) as much as we can within the scope and the time and the resources we have.

Okay. The initial part is the pre-application three years even before they submitted the topical report. And we were engaged and as a part of pre-application and we -- several staff -- several of

us visited on site and we checked their NIST test facility. And after we took on this topical report we decided to review and they wait -- launch an initial site audit. And then we have almost 10 people from us and then I think from NuScale side they had almost 20. So we met one -- for one week, expedited the review. And then we have -- after that we had conducted continuous audits until even this month. We finally have a closeout of the audit to finish this one and to finish SER.

So because of extensive audit and we reduced the number of RAIs. Actually we have 13. That's historically low number. But each RAI has its own questions, so total of 45 issues there, the RAI questions there under 13 RAIs. And through the audit we resolve 210 audit issues. That's a lot of effort there to get the issues resolved.

CO-CHAIR MARCH-LEUBA: So are you saying that you will use the number of RAIs by just packing the issues into -- like more issues into a single RAI, or do you actually resolve the issues in the audits and you have to ask the --

DR. LU: Well, both actually. Both. Actually some of them -- because we are -- I think that's the office instruction to raise the bar of how

you're going to launch, how much merit you have to launch an RAI. And so we make sure that each RAI really indeed has significant safety implications there. And also a lot of issues we resolve as part of 210 audit issues. Some of them actually evolve to be an RAI. So later -- so we issue -- we said, okay, we have to document this one. So that goes into the RAI.

CO-CHAIR MARCH-LEUBA: Yes, my experience; and let's just put it on the record, is that audits are very, very valuable because you can ask an informal questions- and get an informal answers that explain to the reviewer- where they're going. And maybe you have to follow up with an RAI to document it.

DR. LU: Yes.

CO-CHAIR MARCH-LEUBA: But this is a documentation RAI that is just paperwork.

DR. LU: That's right.

CO-CHAIR MARCH-LEUBA: So I really would like to encourage the -- like the fact that the 210:245 ratio on the audit to RAI is very good.

DR. LU: Thank you.

CO-CHAIR MARCH-LEUBA: I mean we should even try to get it better for the next reactor.

DR. LU: Yes.

MEMBER BLEY: Those audit issues aren't documented in the audit reports?

DR. LU: Yes, we have -- actually I think there were issues the two audit reports on LOCA side.

MEMBER BLEY: But they are in the audit reports?

CO-CHAIR MARCH-LEUBA: But my experience is often RAI --

MEMBER BLEY: There's probably more detail in the RAI?

CO-CHAIR MARCH-LEUBA: Yes, often an RAI comes from the staff because the staff does not understand the question.

MEMBER BLEY: Yes. No, I agree on that.

CO-CHAIR MARCH-LEUBA: They don't know what they're asking. So when you do it informally you get answers. They say, oh, yes, you're right. I don't have to issue this one. Ninety percent of the issues are like that.

DR. SCHULTZ: Just for the public record could you give some dates associated with this? You say it's been going on for a long time, but dates of when -- you said pre-engagement was three years before the submittal and then these audits took place. Can you just throw in a few annual -- year dates so we can

get an appreciation for the time scale?

DR. LU: Oh, okay. For the pre-application I remember the first visit we had was back to 2014.

DR. SCHULTZ: Yes.

DR. LU: All right? The initial on-site visit was three years ago.

DR. SCHULTZ: Three years ago?

DR. LU: Three years ago, yes.

DR. SCHULTZ: Okay.

DR. LU: And then we had two phases of continuing audits until actually several weeks ago.

DR. SCHULTZ: All right.

DR. LU: So the final closeout. We had it at the meeting on the RAI.

DR. SCHULTZ: Oh, the peak of activity --

DR. LU: The peak is actually the first year. And then -- but I'm going to talk about what's the major issues. And then so the major issues actually caused the -- a lot of iteration in the additional testing.

DR. SCHULTZ: I just wanted to get on the record the general timeframe that demonstrates what was done.

DR. LU: Yes. Thank you.

DR. SCHULTZ: Thank you.

DR. LU: All right. So those are the issues and the number of issues. There are a number of RAIs. But to really focus the staff's review -- and we performed our own NRELAP5 calculation, sensitivity calculating confirmatory analysis and TRACE. Our staff from Office of Research and -- had developed this TRACE and also checked their -- independently checked the separate effects test and also integral effect test to make sure that TRACE can also do the job, too.

CO-CHAIR KIRCHNER: Clarification: The staff did sensitivity analyses with NRELAP5 --

DR. LU: Yes.

CO-CHAIR KIRCHNER: -- or with RELAP5?

DR. LU: NRELAP5.

CO-CHAIR KIRCHNER: NRELAP5?

DR. LU: So we obtained the -- we normally -- as part of the code topical report review we normally get -- obtain the source code and then debug and make sure that it was okay. Then we -- this time we got the executable and we also had access to the source code from their site. So we were able to dive into. But we did a lot of confirmatory analysis sensitivity cases. Actually Carl is the -- our expert

on that one.

MR. THURSTON: Yes, we do have RELAP5 also.

DR. LU: We also have RELAP5-3D. We have 3D, right.

(Simultaneous speaking.)

MR. THURSTON: -- some cases with RELAP5.

DR. LU: Yes.

MR. THURSTON: That's right.

DR. SCHULTZ: Carl or Shanlai, with the TRACE analysis did you also do sensitivity analyses?

DR. LU: Yes, we did.

DR. SCHULTZ: Matching up what NuScale had done?

DR. LU: Right.

MR. THURSTON: Yes, that's what we --

(Simultaneous speaking.)

DR. LU: I think that will be the --

DR. SCHULTZ: Good. Go ahead.

MR. THURSTON: No, I'm done.

DR. LU: Oh, you're done? Okay.

So we did. And then we -- especially when we performed the NRELAP5 sensitivity calculations -- and then we want to make sure it was okay. So we gave our research staff and say -- actually Dr. Peter Lien.

And he's going to present a lot of those comparisons here including the sensitivity cases we ran initially with RELAP5 and later with TRACE.

DR. SCHULTZ: That's good. Thank you.

DR. LU: Okay. All right. I'm going to talk about the major issues we identified and resolved. So at this point -- and then the -- for LOCA Topical Report we resolved all the issues here and then in later side within a certain review area.

Okay. Next slide. Okay. I'm going to go through this very quickly because I think the Committee has been very familiar with 10 CFR 50.46.

And the peak cladding temperature, that's acceptance criteria: 2,220 degree Fahrenheit and localized maximum fuel cladding oxidation, 17 percent. And the whole core fuel cladding oxidation with hydrogen generation, that's one percent.

And the green is actually for -- is NuScale decided to use Appendix K. And Appendix K does give a lot of -- give them a lot of leverage to simplify the -- to identify the limiting case because they simply have so much margin in terms of PCT. Therefore, they can afford that one. That were -- that both simplify their calculation and also our review.

And however Appendix K, that was a specific -- a lot of specific features. So through the pre-application stage they ask for the exemption. So not all of them were actually needed, because for example, the post center of heat transfer because they never uncover the core. And no flood/refill issue, the pump model. There's no pump, so therefore it's -- becomes very simple. And then that part of process we approve that part, approach right before the topical report was submitted to us.

And so with that so much margin because the core -- if you have a figure, they presented early design -- you know, design certification presentation and the core, they wind up at total volume of water inside the RPV. It's -- this reactor is a hot -- has a list of the ratio. That's means the -- much more water than power. So with that margin they can afford the different figure of merit, not like operating fleet.

So maintain containment, integrate, provide water back to the RPV. And so let me see. Okay. Yes. So the issue here is because once you have the break and have LOCA -- and the limiting scenario here is the -- whether the containment can maintain to have its own integrity to protect the --

provide the cooling. So that's part of ECCS. So that's the figure of merit of the containment integrity.

And for IORV maintain the critical heat flux ratio to prevent significant clad heat-up and the collapsed water level above the core. This just very similar to the paths of the -- like the BWR. There's so much water they have. They can say, okay, as long as I have the collapsed water level above the core, it's fine. So that's the figure of merit, and we agree with that. And that's the baseline level review.

Okay. That's criteria.

CO-CHAIR KIRCHNER: Can I -- just though going back to one point. Did you also look at -- you've got collapsed water above the top of the core. Did you also look at whether they have saturation conditions above the core?

DR. LU: Yes, actually for most of time the -- it's saturated.

CO-CHAIR KIRCHNER: Yes. Yes.

DR. LU: Yes, it's saturated. There is a bubble coming out. So the reason the -- actually if you -- yes, that's a good point. Actually because of it's saturated and then the bubble is coming out of

the core -- and that's a two-phase level. It's way above the collapsed water level.

CO-CHAIR KIRCHNER: Right.

DR. LU: So from that perspective using a collapsible level, it's very conservative. And because your two-phase level is way above, and then it's saturated there anyway.

CO-CHAIR KIRCHNER: Yes.

DR. LU: Okay. Any other questions after this one? No? All right.

Okay. For IORV, because they added at Appendix B and then declared this IORV case as anticipated, operational occurrence and AOO. So therefore, it's the SAFDL, typical SAFDL criteria into CHF. So they maintain the margin to critical heat flux. And then as long it's greater than the CHFR limit, then it should be fine. So that's not the typical LOCA topical report we -- you're able to see, but that's the new item they added over there. So that's the figure of merit IORV.

Next slide.

CO-CHAIR MARCH-LEUBA: Go back. Can you go back to the previous slide on -- and point the second -- the number -- well, the second bullet. I thought that will require SAFDLs to be applied to IORV

is because there was the possibility of return to power afterwards.

DR. LU: Okay.

CO-CHAIR MARCH-LEUBA: Is it because they chose to characterize it with a higher frequency or the return to the power?

DR. LU: Okay. Let me address this because that's a good question, and that's the reason -- if you go back to the scope of this topical report, this one only covers short-term blowdown to long-term -- to the point that it have the return to -- you have a sufficient return flow to maintain the core coverage. So anything beyond that point -- our approval, we are approving the way they are using RELAP5 code, but the actual application of the methodology to address return to power and then peak containment pressure, it's not covered by this topical report.

CO-CHAIR MARCH-LEUBA: Yes, but the figure of merit, if you allow return to power at 72 hours you cannot damage the fuel from the first 10 minutes.

DR. LU: Right.

CO-CHAIR MARCH-LEUBA: So it changes the figure of -- the acceptance criteria.

DR. LU: That's true. And then that's --

I think that that's related to the long-term cooling power. I think we are going to cover that. That's for sure.

CO-CHAIR MARCH-LEUBA: But --

DR. LU: Yes, agree.

CO-CHAIR MARCH-LEUBA: So maybe it's both. Is the fact that they chose to make an IORV a high-probability event or high-frequency event --

DR. LU: Yes.

CO-CHAIR MARCH-LEUBA: -- and also the fact that we allow return to power if there is no operator action forces you to have a much more strict criteria for LOCA than you will have operating a reactor?

DR. LU: That's right. If you get to that scenario. Yes, agree. Yes, that's the case. Yes. That's true.

CO-CHAIR MARCH-LEUBA: Yes.

DR. LU: Any other question?

(No response.)

DR. LU: All right. Next slide. Okay. NuScale already presented it at the -- NuScale develop actually a very good extensive PIRT because part of the design in the past 15 years and then their initial PIRT panel was formed back to 2008. And then they

came up with 21 phenomena important to capture the LOCA evaluation model. So it's not only -- it's -- it does not cover any other event, but just focus on LOCA. And the -- we are very glad that they decide to select the RELAP-3D as the base code. And then from that base code they develop their own NRELAP5 code to model to capture this LOCA analysis.

And they develop quite a lot of integral test specifically tailored to capture the features of NuScale design. And the good news with using RELAP5-3D is because it has had already extensive separate effects test, and plus like integral test before. So that's the good news from their perspective. And from PIRT perspective we can go to the next slide.

CO-CHAIR MARCH-LEUBA: Stay here on the PIRT.

DR. LU: Sure.

CO-CHAIR MARCH-LEUBA: Back to my favorite topic of boron redistribution and boron dilution, did the PIRT panel identify boron redistribution as an issue that's important?

DR. LU: No. Actually, I can answer that one. As particular part we reviewed that part. The focus of -- at that time back to 2008, I think even initially when they -- I think Dr. Jose Reyes

developed this concept. Initially it was a class project for his graduate students. They were thinking about how to mitigate the LOCA itself. So the -- any long-term cooling related to any boron dilution, it was not part of the PIRT panel's original task, so they did not really address that one. And --

CO-CHAIR MARCH-LEUBA: But if --

DR. LU: Yes.

CO-CHAIR MARCH-LEUBA: It is the PIRT panel responsibility to look at the transient and see if you have reach a safe stable condition. And if you're in a condition in which you're diluting the boron, you have not yet reach a safe stable condition when they choose the stop. So just by arbitrarily stopping it at one hour doesn't save them for -- they should have thought of it.

DR. LU: Right.

CO-CHAIR MARCH-LEUBA: And they didn't.

DR. LU: I think if NuScale started again, they probably can definitely ask their PIRT panel to do that. And specifically for this topical report as a historical -- as it is right now what we have is leading up to that, unfortunately.

CO-CHAIR MARCH-LEUBA: And I have reviewed the long-term cooling technical report which has a

PIRT on it and does not have boron dilution on it either. So --

MEMBER BLEY: Yes, I want to kind of support what Jose is saying here. Even if you define the conditions a PIRT panel is supposedly looking at --

DR. LU: Right.

MEMBER BLEY: -- it's on them. It's their responsibility to see if you've over-specified the case and there are some important issues that go beyond what they've been specifically asked to look at. Otherwise, it just doesn't work.

CO-CHAIR MARCH-LEUBA: Yes, and going back to Charlie's topic this morning, the PIRT also didn't identify the internal missiles --

DR. LU: Oh, okay.

CO-CHAIR MARCH-LEUBA: -- as an initiating event even though they -- through probabilistic or frequency of occurrence they can be discarded, but the PIRT should have identified that there is a special condition of the NPM design in which -- whereas in a BWR the containment is maybe 100 feet away and is five feet of concrete and there are three walls and floors in between the core and the containment.

DR. LU: Right.

CO-CHAIR MARCH-LEUBA: Here the containment is only a couple of feet away and it's a straight path for the control rod and it's only yea thick of steel.

So in my opinion the PIRT panel should have looked at the whole problem of LOCA and identified special characteristics of this particular design that are not in the operating fleet. And I'm putting this on the record just to make sure that we don't make a precedent on this. For new reactors when they submit it to the staff and they have a PIRT --

DR. LU: Right.

CO-CHAIR MARCH-LEUBA: -- they don't have to see which parts of the operating reactor apply to me. But you also have to look at why is my design different than the operating reactor and what problems can I have? So and that should be one of the lesson learned for the future reviews that when you do your PIRTs, you should be looking at it from that point of view.

DR. LU: Yes. Okay. So I think I understand where you guys come from. And actually related to the control rod housing failure potential on the design. So that's related to that part, right?

CO-CHAIR MARCH-LEUBA: We know that the

internal missile is not an issue.

DR. LU: Right.

CO-CHAIR MARCH-LEUBA: But my gut feeling is not an approved or a method for analysis, but my gut feeling tells me first is very low-priority then that anything will keep the containment. And if it hits, it doesn't have time to accelerate.

DR. LU: Yes.

CO-CHAIR MARCH-LEUBA: So but it was not analyzed.

DR. LU: Right.

CO-CHAIR MARCH-LEUBA: It was not even identified.

DR. LU: Yes. Well, I think you're correct that the PIRT does not identify this. So, okay. Yes, I see the problem. So you're correct that they did not and then staff did. And we actually launched RAIs. So that's the reason. And we launched RAIs about the specific control rod housing failure because there when this particular rod ejection went as part of it, we had a Regulatory Guide when this was developed and we were trying to figure out what can be wrong with this one if the PIRT panel did not really capture this one but we captured that one.

And I just want to give you a high-level

summary. We don't think it's a problem, the reason is the following. Okay. We asked them that RAI and their answer is, oh, they design everything according to ASME code, so therefore it's not design-basis. And we disagree. And, but is that the really problem and they are not? I think Carl, our expert -- actually we did a lot of analysis trying to address in case. What's wrong with this wrong? It was not ignore. It was not captured.

So assuming there is a rod ejection, there is a follow-on the -- controlled housing failure, which can be a low-frequency event, but with Davis-Besse; everybody heard about that, it's the -- because of the corrosion. So it's just one cycle away from rod ejection. So it's not something that much that can be treated without any basis.

So we launch an RAI, and the RAI came back then that we disagree, but we close that RAI because we did our extensive -- our own analysis. Our own analysis, number one, is the -- by itself because controlled rod housing we have a control rod nozzle, a control rod driver nozzle. And then that's the particular welding point with the control rod driver system. There is a welding point. That's the place that is suspect -- that it's the place potentially

have -- may have the break or may have a failure.

If that pops up, the control rod will be ejected like a missile, right?

So from a reactor system perspective we did dive into that point, into that. And so the good news here, the nozzle diameter is less than the RVV, the throat area. So they have a bounding case from thermal hydraulics perspective, bound to this LOCA scenario.

And then Carl did additional -- quite a lot of sensitivity cases trying to see if we extended the rod -- because if you have a rod ejection, you don't really right away shut down, you have a power spike. So we have a simulated power spike based on a rod ejection calculation assuming there's no void effect, which is very conservative. And then we add that to the core. Do we have a problem with peak containment pressure? We don't. So we close this RAI because we agree with them that it's not -- should not be a design-basis issue. It's because we don't believe that the failure of that particular control rod housing where you resulting in more damaging event than the limiting case they already analyzed from a reactor assessment perspective.

And then related to the missile, the rod

ejection and the missile, in fact that's the position taken by the staff from Chapter 3. So that's the part -- probably if you guys have a problem, you may want to talk to Chapter 3 guys. But unfortunately I agree.

CO-CHAIR MARCH-LEUBA: I hate when you guys say that.

(Laughter.)

CO-CHAIR MARCH-LEUBA: Okay? Because when you say it's not my problem, then I'm here, so who did it? Okay? So try not to say that.

DR. LU: We will pass along that -- your comments.

MEMBER RICCARDELLA: I went back after the discussion this morning and looked at the Chapter 3, both the DCA and the SER for Chapter 3, and the SER for this particular 3.5.1.1, it goes into a probabilistic argument. It says if the combined probability of the nozzle rupture -- it's the probability of hitting a safety-related structure and it -- probability of damaging the safety-related structure. So it's all less than 10 to the minus 7th and it's considered acceptable.

And so I don't know if anybody's done those risk -- probabilistic calculations, but that was the basis. It's not the basis that, oh, it's part of

a vessel, therefore it can't fail. So I'm okay with that consideration.

MEMBER BROWN: Did it address the containment or just the reactor pressure vessel?

MEMBER RICCARDELLA: Both. It's a combination of three probabilities they say. If the probability of the rupture is less than 10 to the minus 7th, it's okay. You don't have to consider it. If the rupture --

MEMBER BROWN: So it's not going to become a missile?

MEMBER RICCARDELLA: Well, the probability of -- yes.

MEMBER BLEY: It's less likely than that.

MEMBER RICCARDELLA: If it's less likely than that. If it's -- but if the probability of that plus the probability of it hitting the safety-related structure --

MEMBER BLEY: The containment.

MEMBER RICCARDELLA: -- the combined probability is less than 10 to the minus 7th, it's okay.

And thirdly, if the probability of it rupturing, it hitting a safety-related structure and damaging the safety-related structure is less than 10

to the minus 7th. So I think P1 times P3 is probably less than 10 to the minus 7th, but it's -- that was the basis for the staff acceptance of that Chapter --

MEMBER DIMITRIJEVIC: So for all three probabilities the same application, same limit applies, 10 -- less than 10 to the minus 7th?

MEMBER RICCARDELLA: Ten to the minus seventh, yes.

MEMBER DIMITRIJEVIC: Well, what --

MEMBER RICCARDELLA: And I guess that's the same as what's currently applied to operating -- I mean, that's out of a SRP.

MEMBER DIMITRIJEVIC: Well, it's currently, but would that obviously with, you know, a large release frequency like 10 to the minus 11 it doesn't make sense to apply this criteria.

MEMBER RICCARDELLA: Right. I think --

MEMBER DIMITRIJEVIC: Because the LOCA, the charging LOCA, I check, currently, which results in containment bypass dominates large release, but is right currently one minus nine challenging frequency. So it doesn't -- if this is a real event which can cause the bypass of containment in small LOCA, I mean it would be much -- 10 to minus 7 is a hundred times higher than 10 to the minus 9.

CO-CHAIR MARCH-LEUBA: So maybe we should have that discussion with the Chapter 19 guys when they're here next month.

MEMBER DIMITRIJEVIC: No, the Chapter 19 -- yes, you have a specific picture with Chapter 19 that's takes care, but everything you guys miss in Chapter 15, and that's now it --

CO-CHAIR MARCH-LEUBA: No, Chapter 15 tells you if the control rod breaks and hits the containment, you get a LOCA, but it bypasses containment. And now you have to tell me what the frequency of that is. And that comes from 19. Well, what we're thinking is that would be bad. If you have a rod ejection --

MEMBER DIMITRIJEVIC: Well, why does it go back to this 10 to minus 7 criteria in Reg -- because we discuss that unfortunately when we are discussing this. We have a criteria in regulation which doesn't apply for these low-frequency things. And this is the big thing. Because we saw this in many cases. This is not the first case we see this. You cannot apply the same criteria when you have a completely different risk measures. We saw the selection of the scenarios for the source terms and we are seeing it again here, that this is not applicable. So somebody should be

taking care how does this apply. And I don't think that should be NuScale.

MS. PATTON: I just need to correct something that was said earlier. It's not only in Chapter 3 where we say that this isn't considered. It's not in the design-basis. It's also in 15.

So what the LOCA Team is saying is correct in terms of we ran a bunch of sensitivities and all that to look at it. That was when we were trying to determine at that time whether we needed to consider it within the design-basis.

However, that decision is documented in 1506 in the return to power section, one of the references I gave you earlier, because it was also being looked at as a potential return to power scenario. So we did determined that that was not within the design-basis. Although we do have a lot of sensitivities, that's solely from a LOCA perspective. It would have been all right, but it's not considered even within 15. It's in the design-basis.

CO-CHAIR MARCH-LEUBA: When you're doing 1506 and doing that evaluation, how do you determine what the frequency of the event is? So it's not a design-basis event.

MS. PATTON: Right, but that's -- the

basis for that, it was like a regulatory basis basically that -- and one of them was the -- part of it was based on the Chapter 3 evaluation.

CO-CHAIR MARCH-LEUBA: Yes.

MS. PATTON: Also looking back at the GDCs, the historical implementation of those, right? And the rod ejection is considered a short-term type of event.

CO-CHAIR MARCH-LEUBA: But it is because in operating reactors the containment is a mile away and there are three floors of concrete in between it and the containment and the containment is five feet of concrete.

DR. CORRADINI: No, that's not true for all containments. There's a lot of free-standing steel shell containments that are out there and operating. So that's not true.

MS. PATTON: We actually researched the entire regulatory history of the whole thing and that whole event with the attorneys and looked at this in a good amount of detail and determined that it was not going to fall within the design-basis within the regulations.

CO-CHAIR MARCH-LEUBA: It would have been so much more satisfying to have made a simple missile

calculation showing that it cannot break the containment, which again -- I mean, there is clearly -- the rod doesn't have time to accelerate. But, okay.

DR. LU: Okay. So next slide.

All right. Now let me talk about the significant issues we identified and we resolved. Just give a heads up. Resolved.

But the first, No. 1, that when they initially submitted this one they did not have a flange. So they just welded the valves with the RPV. So later they change it to the flange. So between the valve and then RPV there is a flange. So possibly there -- has that been considered as part of a PIRT, considered potentially that either the nozzle welding point or the flange may become a break? And I think that part has been resolved. They declared this one as exclusion zone in the RPV. The unit containment they -- outside of RPV of course. And then so therefore there would be no more break location considered as a design-basis for this one. And Chapter 3 reviewer agreed with their special treatment to declare this the break exclusion zone.

And so that's the part resolved. We identified this issue as part of a PIRT and then our

expert from previous Areva retired, Bert Dunn. And also from NuMark, they identified this issue very good issue. And then so we resolved that this point and the Chapter 3 reviewer accepted this claim that they are excluded from any potential break assumption location, break location assumption.

CO-CHAIR KIRCHNER: So, Shanlai, just to summarize. So you're saying -- if I remember correctly, this is a bolted flange for the RRV.

DR. LU: It's a bolted flange. Yes, that's right.

CO-CHAIR KIRCHNER: So you're saying you can't have a break at the flange essentially?

DR. LU: NuScale says. And then the Chapter 3 reviewer decided to approve that at this point.

CO-CHAIR KIRCHNER: Okay. So therefore that limits --

(Simultaneous speaking.)

MEMBER RICCARDELLA: -- you decided review this --

DR. LU: To approve that claim, to have the exclusion from --

CO-CHAIR KIRCHNER: But you cannot --

DR. LU: Yes.

CO-CHAIR KIRCHNER: Okay. So you can't have the valve body break loose, essentially is what you're saying.

DR. LU: Right.

CO-CHAIR KIRCHNER: So therefore the RRV is equivalent LOCA. I know the applicant doesn't want to call it a LOCA, but it is a LOCA. If the RRV opens, then you're just limiting to the actual valve diameter, right?

DR. LU: Right.

CO-CHAIR KIRCHNER: Okay.

DR. LU: And another reason is I think we are -- from our reactor system perspective we are supporting that; I personally also support that point, is because we did -- actually our staff handed that to Carl and then the also Office of Research did additional analysis. Just use the maximum possible location. Let's assume it's flange also fail.

CO-CHAIR KIRCHNER: Yes.

DR. LU: What's going to happen? And it doesn't really hurt. It does not really hurt the peak containment pressure nor the clad 4 level above the core, so figure of merit is covered. So therefore, with additional measure NuScale decided to maintain the flange and check that one during each cycle. And

I think with it or without it it's still okay because by design they have sufficient water there. And they also -- the nozzle size not that large, and so then that will not challenge the peak containment pressure. So I -- from reactor system perspective we supported that decision. So that issue was resolved from that perspective.

CO-CHAIR KIRCHNER: Did you actually do TRACE or other analysis for --

DR. LU: Yes, we did.

CO-CHAIR KIRCHNER: So you did it with a nozzle probably with a different --

DR. LU: Different size.

CO-CHAIR KIRCHNER: -- either Moody or whatever factors you used?

DR. LU: Right.

CO-CHAIR KIRCHNER: One for the actual valve and one for a nozzle?

DR. LU: Right.

CO-CHAIR KIRCHNER: Okay.

DR. LU: That's right. That's right. So the nozzle size -- of course the throat size is much smaller than a nozzle size. The nozzle size -- when we enlarge it to the nozzle size, then it's not that bad. And actually it's a -- there is an increase

there of the peak containment pressure, and there's no really impact for the -- on the clad 4 level. So therefore, we don't feel that it's not -- it's not as much of challenge there for us to prove that.

CO-CHAIR MARCH-LEUBA: Yes, and that conclusion is very intuitive because the peak pressure happens with the steam valves when the RVV is open, not when the RRV is open.

CO-CHAIR KIRCHNER: No, no. No, no.

PARTICIPANT: Peak pressure is --

CO-CHAIR KIRCHNER: No, actually it's the opposite.

(Simultaneous speaking.)

PARTICIPANT: -- is the RRV.

CO-CHAIR KIRCHNER: Yes.

CO-CHAIR MARCH-LEUBA: I don't --

CO-CHAIR KIRCHNER: It's the limiting case.

CO-CHAIR MARCH-LEUBA: No --

CO-CHAIR KIRCHNER: That's why they're addressing it.

CO-CHAIR MARCH-LEUBA: No, the --

CO-CHAIR KIRCHNER: This is a liquid discharge, two-phase. This is the worst break for them, of the different valve system.

(Simultaneous speaking.)

PARTICIPANT: Worse than a steam --

CO-CHAIR KIRCHNER: A burst in the steam break on the top.

CO-CHAIR MARCH-LEUBA: No, it is an RRV break that dumps liquid into containment. And when you hit the ECCS saturation level, then it opens the RVVs and that causes a big break.

CO-CHAIR KIRCHNER: No, when your primary system blows down from full pressure and temperature, this is the limiting case. Trust me. Not a steam line break.

DR. LU: Yes, I think we are going to --

(Simultaneous speaking.)

DR. LU: Back to March, the staff is going to talk about that one, so that was the details. But the bottom line, I think this particular issue is resolved. We identified. We launched the RAI. And then very few RAIs. We -- this one of them. And then the next one is also a sensitive one. And as the Chairman mentioned about this one, the NuScale did not want to declare as a LOCA, but it's a lower point of the RPV can discharge coolant to the containment.

CO-CHAIR KIRCHNER: Yes.

DR. LU: So we were -- as part of a PIRT

and think our consultant raise the question: How come the lowest possible, largest potential discharging point? We don't do this to say the breakpoint. Why there was no test was not done? And is the PIRT really covering this one?

So that took us a while and -- almost more than year to talk to the NuScale. And then they finally responded to this and with additional test. And they did the RRV test and then discharge this one. And the good news here turn out to be, yes, so although it's a different number, came up with different number, different peak containment pressure number, different clad 4 level, but it's still within the acceptable limit. And therefore there is no challenge to the existing claim. And then -- or from our perspective, from staff's perspective there's no integrity issue, nor the core cover issue.

So therefore, this particular issue was resolved. But that's part of a PIRT review identifying this issue. How come never thought about this one? The lowest level. There was no test.

CO-CHAIR KIRCHNER: Well, just one member's opinion, to avoid using the word LOCA, because their design is such that they don't have the extended piping system of the commercial fleet --

DR. LU: That's right.

CO-CHAIR KIRCHNER: -- so now we are calling this -- and here I think in one of our write-ups that you will receive on the chapters there's inconsistency because they've called it an AOO. By definition an AOO is a frequent event, something that happens one or more times in the lifetime of a plant. And that's in Chapter 6 of your SER.

DR. LU: Right.

CO-CHAIR KIRCHNER: So there's some inconsistent terminology that -- between Chapter 1.6 and 15.

DR. LU: I understand your comment. Actually I personally understand it, but I think the bottom line from a staff perspective if they are applying AOO criteria of CHF or SAFDL and then they definitely can meet 50.46.

CO-CHAIR KIRCHNER: Yes.

DR. LU: So what's the problem? There is no problem there, so we let it go. That's how we decide, made the decision. Say, okay, this issue is resolved. With additional testing saying that there's no additional new phenomena or change the PIRT. So therefore, this particular issue was closed. But I agree with you that's the -- there are a lot of places

we are having -- we have to learn how to jump out of the historic thinking of what we have been doing with operating fleet and then to accommodate new design features. I think that's a part. There are some of the regulatory inconsistency or consistency we have to develop to address design features.

CO-CHAIR KIRCHNER: So one question that occurs to me is once you admit to the RRV being an AOO, is the other equipment in containment protected against the blast that's going to come out of this valve?

Where I'm going with this is that, for the conventional valves up on the top of the reactor vessel, there are usually shrouds or other things to prevent the impingement zone affecting cables.

DR. LU: Right. Right.

CO-CHAIR KIRCHNER: We've got the lines that come in to actuate the ECCS valves and other things inside containment.

DR. LU: Right.

CO-CHAIR KIRCHNER: If this RRV lets go, is there the possibility that that two-phase mixture coming out could impact other equipment inside containment?

DR. LU: That's a very good question.

MR. THURSTON: Yes.

DR. LU: Actually I think we are going --

MR. THURSTON: So the valve design for the RRV is a downward projection.

DR. LU: Yes.

MR. THURSTON: So the nozzle is projected in the downward inside the containment. And in the RRVs the valves at the top, they have some type of device on the outlet of the RVVs.

DR. LU: Yes.

MR. THURSTON: So they have accounted for those phenomena.

DR. LU: Yes, and that --

DR. CORRADINI: But I think what Walt's question is even though it's pointed downward within that annulus, is there --

CO-CHAIR KIRCHNER: Well, there's the --

DR. CORRADINI: -- equipment or cabling --

CO-CHAIR KIRCHNER: The level detection sensors is the first things that occurs to me. The second thing is all the lines that go to the ECCS valves, et cetera. So there's --

DR. LU: Yes.

CO-CHAIR KIRCHNER: It's not as clean a containment as we're lead to believe. There is

equipment inside that annulus that could be impacted. The first one that I would worry about would be the level sensor.

DR. LU: Right. Actually I think the -- as Carl mentioned, is designed to be -- have a different orientation of discharge. There was -- also device was designed to protect the equipment against that impingement. And that device review was done as part of Chapter 3. We only heard of the -- but they -- I think the final disposition of that one was done by Chapter 3 guys. My understanding is that issue was raised and it was also resolved.

CO-CHAIR KIRCHNER: I know it was for the --

DR. LU: Right.

CO-CHAIR KIRCHNER: -- because you have to worry about it for the reactor safety valve.

DR. LU: Right.

CO-CHAIR KIRCHNER: And also the reactor vent valves. And I think there were shrouds or other measures taken to reduce impingement on equipment that's inside containment.

DR. LU: That's right.

CO-CHAIR KIRCHNER: But the RRV, the potential of that opening, yes, it's oriented down

because its real function was to bring water into the reactor vessel.

DR. LU: Right, right, right.

CO-CHAIR KIRCHNER: Hence, it's oriented down, a little bit different than you would expect.

DR. LU: Right.

CO-CHAIR KIRCHNER: But to my knowledge from reviewing Chapter 6, I didn't see any shrouds or anything to prevent a two-phase jet impingement on anything down in the containment.

DR. LU: Yes, that was --

CO-CHAIR KIRCHNER: So my concern would be the level detector.

DR. LU: Okay. Yes.

MEMBER BLEY: Well, the --

DR. LU: Go ahead.

MEMBER BLEY: Remember the level detector is expected to be a radar detector --

DR. LU: yes.

MEMBER BLEY: -- from the top down. Now I don't know -- it wasn't specified --

DR. LU: Right.

MEMBER BLEY: -- at the time they did that analysis, near as I heard.

DR. LU: That's right. That's right.

MEMBER BLEY: So it probably didn't get looked at at that time.

CO-CHAIR KIRCHNER: Yes. So I just raised that as an observation.

DR. LU: Yes.

CO-CHAIR KIRCHNER: That's fine.

MS. PATTON: So that's done in Chapter 3 as part of the hydrodynamic loads evaluation. So I'm trying to -- it's in their SER. It should be for Chapter 3 where it talks about jet impingement and the loadings and the review for that. But that's where you would find it is in that chapter.

CO-CHAIR KIRCHNER: I know it's there for the valves that are on the top of the reactor vessel, but I don't know that it was considered for the RRV. See where I'm going with this?

MS. PATTON: Yes, I mean, I'm trying --

CO-CHAIR KIRCHNER: I just make it as a note.

MS. PATTON: It should be in that section.

DR. LU: Okay. We'll take back your comment.

CO-CHAIR KIRCHNER: Yes, the upstairs valves, they're shrouded.

DR. LU: Okay.

CO-CHAIR KIRCHNER: I already looked at that.

MS. PATTON: Right, but either rate that's part of the DCA review.

DR. LU: Right. Yes, it's in our topical report. Yes, indeed.

Next slide. That's a PIRT. Sorry. And the models, the -- we are very glad that NuScale selected RELAP5-3D computer come from INL, the base code. And they developed the specific component to model the NuScale reactor. And the one along the top is a helical coil steam generator. So they have a specific NRELAP5 component. They have pipe component, they have a branch component, a whole bunch of different specific hydraulic components, but they have the specific steam generator component to model the heat transfer and the pressure drops.

And then we have been talk about what the -- our findings are and then also how -- what's after it in detail as part of a non-LOCA topical report presentation.

And they also develop their own and added, selected the right -- the CHF correlation, and specifically for the fuel selected there. And obviously since containment functions as one part of

ECCS system and provided the cooling, so therefore steam condensation is a unique feature of this device from a code and modeling perspective. So therefore, they spent a lot of effort to identify and to make sure that the correct correlation is used and also benchmarked against their test facility. So that's the part. And we can talk about details in -- during the proprietary session.

In the interfacial draft for large diameter pipes, because of the riser above the core, basically it's a large -- huge, large-diameter pipe, excuse me. So therefore, the interfacial draft becomes important there. So that's part of the model. They had a unique adjustment there.

Okay. And we also -- staff also review their NPM model, NuScale power module, the action model from the actual access model to the point to change it to the nodal RELAP5 model. So every step -- and then we audit it and we did the calculation. We identified issues and we resolved. So what we are proving here is they really -- the final -- the NPM model based on our review and the resolution of the comments we had.

CO-CHAIR KIRCHNER: Shanlai, I'd just like to make an observation.

DR. LU: Yes.

CO-CHAIR KIRCHNER: You have brackets after stem condensation and critical flow models.

DR. LU: Right.

CO-CHAIR KIRCHNER: We'll get to it in closed session, but --

DR. LU: Right.

CO-CHAIR KIRCHNER: -- most of these models are in the public domain.

DR. LU: That's right. The --

(Simultaneous speaking.)

CO-CHAIR KIRCHNER: -- the public domain.

DR. LU: Yes, you're right. And that --

CO-CHAIR KIRCHNER: So I just personally object to the idea that that somehow becomes proprietary, which is different than CHF where they actually developed the CHF correlations.

DR. LU: Right.

CO-CHAIR KIRCHNER: But in general, like the break discharge models, those are right out of the public domain.

DR. LU: That's right.

CO-CHAIR KIRCHNER: That's an observation.

DR. LU: In Appendix K you see they are --

(Simultaneous speaking.)

CO-CHAIR KIRCHNER: Or Appendix A. Yes.

DR. LU: There is a specific critical flow break discharge. I think that's just a proprietary check we would like to have, so I can -- I respect that decision. So we will talk about in details --

CO-CHAIR MARCH-LEUBA: In defense of the applicant, which I don't do often, they spent a lot of money verifying that those open domain correlations work for their system.

DR. LU: That's right.

CO-CHAIR MARCH-LEUBA: And the fact that this particular correlation work for my system, I spend a lot of money doing it. And they have the right to keep it.

DR. LU: That's right.

CO-CHAIR KIRCHNER: It's an expensive choice.

CO-CHAIR MARCH-LEUBA: Yes.

DR. LU: The choice is they decided to use public domain. It's still --

PARTICIPANT: That one I'm confident is all public domain, but --

DR. LU: That's right.

(Off-microphone comments.)

DR. LU: Then you can see that point, yes.

Indeed.

Okay. Next slide. So we have done extensive testing into -- with a scaled -- I think a partial height, not full height, in one-third height test facility. And then this one is -- as NuScale already presented at their NIST facility, covers reactor pressure vessel, steam generator and DHRS, containment and the containment pool. And so we have all the valves and vessels, whatever the NPM module has. So what is the -- really the extensive review by Dr. Peter Lien and also Upendra from Brookhaven National Lab, who was on the scanning analysis.

So this one-third height and devices was -- can that be scaled back to NPM and do we have any distortions or do they have identified sufficient detail to justify the distortions that are acceptable? Those are the issues we dived into.

And because we dived into this one and as part of scaling and the distortion analysis review, and I think the team identified one issue, which was also -- at that time it was a significant issue we had to report to the office director. And by the time we hit this issue we found, oops, all the previous LOCA test, scaled LOCA test was bound with only half of the decay heat power. It was supposed to do with a scaled

to this much of power. All the previous test was done half the power. And then that was a shock. And then so we issued RAI and then we pushed the -- it took a long time and then also to push that for another test. Most limiting LOCA test with full decay heat power, scale decay heat power.

So staff -- I think our team, we did our job and then we identify this issue. And then the good news is the conceptual design by itself has so much water. You add 100 percent additional power level into the core. And then for this NIST facility, they say okay. Now all other designs, all other PWR designs you have this kind of condition. You raise the power, decay heat power by a factor of two. Pretty much none of the power can now make it, but they made it. So that's good to see that. And that's issue resolved. And then resulting additional tests.

That's the reason why it's three years. And I think early the staff -- you know, ACRS was asking why took that long? Because there are additional tests being -- that were done.

DR. CORRADINI: Where is that documented in the SER? I can't find that.

DR. LU: I think that we --

DR. CORRADINI: What you just said I can't

find in the SER.

DR. LU: Okay. The SER -- okay. The -- what --

DR. CORRADINI: Because the scaling portion of the SER is -- the key scaling parameter is on keeping time in variant at the various scales.

DR. LU: Right.

DR. CORRADINI: But this is a power mistake. I don't remember reading that.

DR. LU: Yes. Well, this -- the staff was working on that. The final SER was based on all the revised version of the topical report, Revision 1. And there's a final version of the NRELAP5. So we are approving although the goal is still the same as the Rev. 0. But by the time we reach Rev. 1, those are the -- so when we wrote our SER --

DR. CORRADINI: There's no historical reference to this observation.

DR. LU: Yes. Yes. Yes.

DR. CORRADINI: Okay.

DR. LU: So we're --

DR. SCHULTZ: Isn't there --

DR. LU: Yes, go ahead.

DR. SCHULTZ: Isn't there an RA response which documents the results of the test as compared to

what had been done previously?

DR. LU: Yes, we actually have a -- specifically it's the middle of the -- I think that was part of HP-06b about this one. So this was -- that's additional test.

DR. SCHULTZ: Can you get the RAI response for us?

DR. LU: We'll get back to you. And then either RAI responses or audit report, because we had that -- this issue.

DR. SCHULTZ: Well, not just identifying the issue, but the results of the test --

DR. LU: Okay. Results of test, yes.

DR. SCHULTZ: -- in comparison to the original half-power test.

DR. LU: Yes, actually the -- yes, we'll get back to you about the ML number or whatever information. And actually part of --

MS. PATTON: Just to explain a little bit more how we're doing it. So the office undertook and effort to change a little bit how we're writing the safety evaluations. So now it's -- right now is like the latest revision. Like you said, when you're Revision 1, you write the SE against that revision.

So issues that were brought up and

resolved are documented through audit reports, through the RAI process and the issuance of those. But it's whatever that final rev -- and so if that final rev resolves the issue, that's what you're writing the SE against. But we're presenting some of the issues that have been encountered during the review so that ACRS could see some of those major issues, because we understood that that was part of what you wanted to see.

MEMBER BLEY: I admit to being a little confused because other -- and maybe it's different section by section and the way you treat the SERs. I think for the chapters -- the NuScale design cert, we've been told the SERs include the revision they're written on plus all of the resolved RAIs up to that point, which it really looks like the next rev of the design cert.

MS. PATTON: So the way that we write SEs evolves throughout the review. And so I can't speak to some of the other chapters as to how much they went towards the office director's goal of writing SEs this way. Some of those chapters I think they may have been further along in the writing at the time that decision was made and it may have been written more the old way. But the new way that the office is going

is you write it against whatever the current rev is.

So in other words, if you had written RAIs that said you guys need to specify X, Y and Z in this document. And then later they end up specifying X, Y and Z. When you write the SER, you're writing it against the revision that does have that specification. So you don't go through all the history of we had a problem with this. We made them fix it. They fixed it. Here's what they did.

MEMBER BLEY: Okay.

MS. PATTON: That's historical. But we do have enough for the legal basis of why everything is okay as it's currently written.

DR. CORRADINI: At least in this one I guess this is a big enough difference. At least I'd like to know what the RAI number is so we can --

MS. PATTON: Yes, we can get you that.

DR. SCHULTZ: No, and the reason is that there was a spectrum of tests that were performed with the half power.

DR. LU: Right.

DR. SCHULTZ: And a limiting case performed with the full power.

DR. LU: Right.

DR. SCHULTZ: So it would be interesting

to know. You said it was good to go, but the technical --

DR. LU: Yes. Want to know the technical basis.

DR. SCHULTZ: -- basis behind that would be good to have.

DR. LU: Yes.

DR. SCHULTZ: Thank you.

DR. LU: That's fair. And actually I think the slides we are going to present you today probably has already had some of those in comparison there. So that can may be --

(Simultaneous speaking.)

DR. SCHULTZ: That will be helpful also.

DR. LU: Right. Yes.

DR. SCHULTZ: Thank you.

DR. LU: Okay. All right. Let's get to the confirmatory analysis. As usual that for a new design, and especially for light water reactor design, we perform extensive confirmatory analysis to just to make sure everything is okay. And unlike AP1000 and these PWR, which we had develop our own test facility, this one we did not. So we put our effort on the confirmatory analysis with the great support from Office of Research.

And so we went through their separate effect tests. We set it to the NuScale power module, KAIST model relate to DHRS tube condensation. It's almost -- I think I recall it's full scale of the tubes there. And the SIET test model. And then they had their -- over the test facility Italy. And then we run our code and also develop our TRACE model. And we identify the issues in them. Currently is resolved at this point.

And also the NIST-1 model. By itself the NIST facility, we model that one, so that we can -- be make sure that how much uncertainty they are claiming is really agreeable with our own independent analysis, which is a part of our confirmatory analysis effort.

The integral effects test, and then as I mentioned, they did a LOCA/non-LOCA test. And so therefore we used a TRACE code and NRELAP5 code to understand their integral effect test and then make sure that our code also has the capability to have a same accuracy, or similar accuracy, or even better to assess their NRELAP5 model. And I think very good agreement were obtained at the end, so we were happy with that one. So that's how we end up saying, okay, this model are good.

And of course we used both TRACE and the

NRELAP5 code. And we had -- Carl ran many, many cases, 55 sets of calculations with sensitivity cases of different separated factors or integral factors and NPM models. There are a lot there.

And then because of that effort we identify a lot of audit issues and those are result by resolving those all the issues. Plus we had a QA audit. And then they decided to upgrade the code version from 1.3 to 1.4. And then good agreements were obtained between their code prediction against the test results. So therefore, we feel comfortable to prove this topical.

All right. Keep going.

DR. SCHULTZ: Shanlai, for the open session, can you categorize those items that you've identified in terms of what were the important items that you identified that --

DR. LU: Those are -- yes, we --

(Simultaneous speaking.)

DR. SCHULTZ: -- code modifications and so forth?

DR. LU: Yes, for example, we already mention about the RRV location. The RRV, the nozzle size versus the throat size. We also are trying to understand the impact on the peak containment

pressure. There are a lot of stuff can impact the peak containment pressure. And then if we -- I also mention about that we were diving into whether that rod ejection went with -- assuming it's a rod in controlled housing failure --

DR. SCHULTZ: Yes.

DR. LU: -- what would be the impact on peak power? Those are part of 54. We have our excellent expert on NRELAP5. He ran all those cases. And then, so we also have the Office of Research running the same things as we are, too, and with different subset, with a focus on -- we normally ran NRELAP5 with the initial runs and then we stay focused with TRACE focusing on specifically do we have a figure of merit and -- of the issue? And then Dr. Weidong Wang and Dr. Peter Lien, those are the key contributors of that.

DR. SCHULTZ: And the good agreements were obtained on Version 1.4?

DR. LU: Yes, 1.4 --

(Simultaneous speaking.)

DR. LU: Yes, the 1.4. 1.4 and then some of the 1.3 version for specific tests. Yes, normally --

MR. THURSTON: Yes, it's a very small

difference. There are minor changes and corrections that they made to Version 1.4. So it's -- I would characterize it --

DR. SCHULTZ: There were some changes there that you --

(Simultaneous speaking.)

MR. THURSTON: Yes, and we will do those in the closed session.

DR. SCHULTZ: Okay. safety

DR. LU: Yes.

DR. SCHULTZ: The Version 3 results cover --

MR. THURSTON: Are very similar.

DR. SCHULTZ: -- most of the issue, most of the good comparisons, but then some modifications were done that --

MR. THURSTON: That's correct.

DR. SCHULTZ: -- helped out others?

MR. THURSTON: That's correct.

DR. SCHULTZ: Okay. Thank you.

DR. LU: We actually had -- we are going to have the slides comparison.

DR. SCHULTZ: No, that's very helpful to know that this wasn't a major modification that changed everything to --

DR. LU: No, the -- yes, we have --

DR. SCHULTZ: Some things needed to be tweaked?

MR. THURSTON: Correct.

DR. LU: Right. Okay, let's go back to, so, CHF model, after they introduce the IORV, of course the SAFDL and also CHF becomes key. But even before that, because they also claim that the collapse level is so high and they never reach CHF, even doing LOCA.

So, that's also, you know, we dived into their CHF correlation, make sure they are really okay. And the good news here is, most of the limiting time, it's only a couple of seconds.

Right after that one, if both LOCA and IORV goes down, you have a scram, the power goes down and now CHF is not limiting. But because of that fissures of only a couple of seconds of CHF is limiting, if you just think about it for IORV or LOCA case, your initial couple of seconds of destructuration.

If you have a sudden from single phase to suddenly you have flashing of liquid in the core. Actually, that takes a lot of energy away from the cladding.

So therefore you, anybody apply the steady state, no matter whether public domain or proprietary correlation, which are thoroughly conservative. So therefore we don't, we agree with them about the design with such a passive feature.

And also, so much margin. And the CHF is not an issue for them, for IORV and LOCA.

All right, let's keep going. All right, since this can be a public domain, so I'm going to talk about Item 5. And so that I don't have to repeat this pro-limitation as a condition is part of the proprietary session presentation. So you have any questions on that one, let me know, okay?

So first, of course, they ask for the removal, or reduce of requirement, from Appendix K. Even before they submitted a topical report.

And so anybody, any applicant or licensee decided to view the plan with this one, they need to ask for the exemption from Appendix K requirement. Which is, can be done fairly easily.

So, since the figure of merit of this, the LOCA EM model is limited to the evaluation of LOCA and of CHF is not exceeded, it has to be, any revision of this in this application of this topical report limited to the point that the containment temperature

and pressure remain below the delay limit. And the core collapsible level is above the core.

If it goes into core uncovery, this approval can no longer be valid.

And the, no credit for DHR heat removal exits right now. Which is conservative I think to their credit. And I think if they do have, DHRS doing LOCA, it definitely will remove a lot of heat. There is no doubt about it.

And therefore, but they decided not to take the credit from that, which is very conservative. But if they decided to take on that one, they have to accredit that one, DHR's heat removal at certain point. And they need to submit a revision of these topical report with further review and approval.

CO-CHAIR MARCH-LEUBA: So once ECCS actuates and you reach agreement, this simulator is completely uncovered, right?

DR. LU: Yes.

CO-CHAIR MARCH-LEUBA: This DHRS doesn't really work.

DR. LU: That's right.

CO-CHAIR MARCH-LEUBA: That's the only reason --

DR. LU: Yes. No, that's right. But it

does, you still have a cooling there. If you have cooling there --

CO-CHAIR MARCH-LEUBA: Cooling steam condensing but --

(Simultaneous speaking.)

DR. LU: But, yes, you're right that the, actually, because the steam bays above the wall level in the collapsible level in the downcomer are very stagnant.

But if you have condensation going on, so you can, still can calculate steam coming down to the, through the steam generator and the condensate ejection water towards the downcomer collapsible level. That's can happen to it.

CO-CHAIR MARCH-LEUBA: If DHRS is available for LOCA, at least long-term, it's a very small addition to the heat removal.

DR. LU: Yes. So --

CO-CHAIR MARCH-LEUBA: It's not worth for them if they can live without to have to answer all your 45 RAIs on the subject.

DR. LU: Yes, I got it.

CO-CHAIR MARCH-LEUBA: So, everything is good.

DR. LU: Right. The next bullet is, use

of LOCA EM is limited to evaluations of analysis of the downcomer described in the TR, which is the condensation pressure CHF and the collapsible level for LOCA and IORV event.

And the approval is limited on version 1.4 and NPM Model Revision 2. And if they decide to make a change they need to submit a change process for our approval for any other revision change of the dyn (phonetic) changes there.

Other part of the features of this design, single failure criteria, electrical power assumption and the need for operator action are not approved within this methodology. This just methodology approved for LOCA and IORV analysis.

And we did have specific CHF limitation conditions there. And we are going to talk about that one in detail during the proprietary session.

So those are the approval limitations and conditions. Yes?

CO-CHAIR MARCH-LEUBA: So, I'd like to bring up the same topic I brought up this morning with Applicant.

DR. LU: Okay.

CO-CHAIR MARCH-LEUBA: After the ECCS valves were tested and the actuation pressure values

were identified, which I call set points, and I want to call it set points, they had to rerun all of their analysis because they have calculated with the wrong set point.

And one of the things they did was stagger the opening of the valves. You have to open the second IORV valve first and then you open the RPV.

DR. LU: Right.

CO-CHAIR MARCH-LEUBA: Where is that specified that that's part of the methodology?

It has to be either on this topical report or it has to be on Chapter 15 specifying what we mean by the IORV. And I don't --

DR. LU: Okay.

CO-CHAIR MARCH-LEUBA: -- see it in either of the two.

And I think the proper place to have it is a condition here that say, for future applications of this methodology to IORV, though shall use the worst staggering of valves.

DR. LU: Right.

CO-CHAIR MARCH-LEUBA: What do you think? I mean, that's what I would do. I would add a condition here.

DR. LU: Yes.

MR. THURSTON: So, the LOCA TR does acknowledge that there's a potential for staggered release of the ECCS valves. But the actual evaluation of those set points are targeted, are appointed to Chapter 15 analysis.

CO-CHAIR MARCH-LEUBA: Yes.

MR. THURSTON: Yes.

CO-CHAIR MARCH-LEUBA: Well, it is 500, 700 or 1,200 psi, that's a design issue. But the fact that this methodology, though shall use the worst opening time possible within your tolerances. It's part of the methodology.

MR. THURSTON: It should be. It should be yes. So that I agree with. Yes.

DR. LU: Okay.

CO-CHAIR MARCH-LEUBA: So, I mean, I would like to add the recommendation to overview. With somebody at overview.

MR. THURSTON: Well --

MS. PATTON: So I think, and Carl, maybe you can explain this, but I think you looked at some of those sensitivities when we audited, right? So how big was the difference?

MR. THURSTON: Yes. So the differences between the staggered and changing, so initially the

release set point was at 1,000. And through the testing, the verification testing that they did, they found that it could be as low as 900.

CO-CHAIR MARCH-LEUBA: That might be proprietary.

MR. THURSTON: Oh, that might be proprietary.

DR. LU: Yes.

MS. PATTON: I guess what I meant is, how much of a difference did it make, Carl?

MR. THURSTON: Thanks.

CO-CHAIR MARCH-LEUBA: The difference was a few psi --

DR. LU: Yes.

CO-CHAIR MARCH-LEUBA: -- but it was half of the margin.

MR. THURSTON: So, for the LOCA cases that we considered --

CO-CHAIR MARCH-LEUBA: I don't know, maybe we can postpone this to the closed session.

DR. LU: It's better to talk about it later on.

MR. THURSTON: I think the message we want to leave with the ACRS is that the difference is, between the previous analysis and the revised analysis

is small. For the limiting case of collapsed liquid level. Which is the limited case for LOCA.

CO-CHAIR MARCH-LEUBA: I see it refers to peak containment pressure. It was a small change, but that small change equals half the margin.

MR. THURSTON: No, that's not the case.

CO-CHAIR MARCH-LEUBA: No?

MR. THURSTON: Yes, that's not the case.

CO-CHAIR MARCH-LEUBA: That's what I remember.

MR. THURSTON: It's a small change.

CO-CHAIR MARCH-LEUBA: They have to increase the margin, the operating pressure of the containment just in case.

MR. THURSTON: Well --

CO-CHAIR KIRCHNER: No, that's not correct.

MR. THURSTON: -- that was NuScale's prerogative.

CO-CHAIR KIRCHNER: They, NuScale changed it once they admitted to the fact that you could have inadvertent operation, opening of the RRV. That's the dominant scenario for peak containment pressure.

MR. THURSTON: Yes.

CO-CHAIR KIRCHNER: So they had to change

the containment pressure from 1,000 psi --

CO-CHAIR MARCH-LEUBA: To 1,050.

CO-CHAIR KIRCHNER: -- in Rev 2, to 1,050 in Rev 3. I think that's a matter of public record.

The actual change in the sequences, sequencing the valves, the various combinations was less than ten psi difference.

CO-CHAIR MARCH-LEUBA: Which was half of the margin.

MR. THURSTON: No.

CO-CHAIR MARCH-LEUBA: To the existing --

CO-CHAIR KIRCHNER: No.

MR. THURSTON: No, it's not half of the margin.

CO-CHAIR KIRCHNER: Not half of the margin.

DR. LU: Maybe that's the part I think that more detail --

CO-CHAIR MARCH-LEUBA: Maybe we can --

CO-CHAIR KIRCHNER: Yes, we can do that --

MR. THURSTON: This margin is significantly larger.

DR. LU: Yes. Actually, I think Carl is right. And because of the margin of so much margin we have in terms, from the PCT perspective, peak

containment pressure --

CO-CHAIR MARCH-LEUBA: PCT, this --

DR. LU: -- in the peak pressure, even for the peak containment pressure side, we did a lot of sensitivity cases.

When we had IAV issued, they, you know, when the opening and the timing and the closed timing. So we had those calculations there. And --

CO-CHAIR MARCH-LEUBA: Let's talk in the closed session --

DR. LU: Yes. Yes.

CO-CHAIR MARCH-LEUBA: -- so we can talk.

DR. LU: We can talk about it, yes.

CO-CHAIR MARCH-LEUBA: And I have --

DR. LU: We can go to the closed session and see whether we can address your comments, yes.

CO-CHAIR MARCH-LEUBA: In my opinion, if the staggering of the valves makes a difference, does it make a difference? Does it make a significant difference?

DR. LU: Yes. If it does make significant, I think I would agree with you. At this point we have not seen that.

MR. THURSTON: That's right.

DR. LU: Because that's the point --

MR. THURSTON: Because think about it, when one of the valves open --

CO-CHAIR MARCH-LEUBA: That's not what --

MR. THURSTON: Well, it's not going to be a large difference.

(Off-microphone comments.)

CO-CHAIR MARCH-LEUBA: Yes, let's postpone it to closed.

DR. LU: Okay. I exceeded my time by a factor of three, sorry.

CO-CHAIR KIRCHNER: We'll take it out of your next colleague.

DR. LU: Okay. All right. Conclusion slide. And then we are proving that the version NRELAP5.14 and we agree that it's in compliance with 10 CFR 50.46, Appendix K.

And the 1.4 is associated with the NPM model is approved. And then we also proved that CHF modeling is up to limitation and condition for both LOCA and IORV. Those are the conclusions that we have.

So, pretty much what they asked, and then we approved.

CO-CHAIR KIRCHNER: Shanlai, could you just go back to, you don't have to show it, it was on

Slide 11 where you mentioned the new limiting LOCA test was conducted.

For the record, how late in the game, so to speak, was that discovered?

DR. LU: It's hard --

CO-CHAIR KIRCHNER: That they are running at half power.

DR. LU: Half power it was, when we launched the first initial face-to-face onsite audit, and then we got a lot of information, so we came back. We were digesting a lot of stuff. The team had to work, and then somehow we asked them, what is distortion reported, at that time the distortion report was not completed yet.

So we want to know, where is this, what exactly there. So our scaling team, and then also the staff, we dive into their, later they submit a technical report regarding this scaling and distortion analysis. The update scaling and the distortion analysis.

So we checked that one and said how come, a very specific statement saying, the power, decay heat power previously done before was half of the scale of the power.

We were surprised, and then so we, and we

asked what's going on there. It's confirmed that actually the previously LOCA tests were done at half of the decay power.

So we said, sorry --

CO-CHAIR KIRCHNER: But that was early in the process.

DR. LU: Early in the process. That's very early phase of, within half, half year. Nine months.

CO-CHAIR KIRCHNER: Okay.

DR. LU: Yes.

CO-CHAIR KIRCHNER: Because I'm reflecting on Jose's statement earlier this morning where had you discovered that very late in the process of approving we would perhaps, maybe this, they have enough margin it didn't have an impact, but that would be something of concern.

DR. LU: Yes. Sure.

CO-CHAIR KIRCHNER: And throw a question about the validity of RELAP validation for the Chapter 15 analysis.

DR. LU: Yes. I think that's exactly the reason I think the management asked the team to engage with NuScale as early as possible. And in support, asked for additional onsite audit and then face-to-

face meeting. Which enabled us to identify the, on the proper issues there.

CO-CHAIR KIRCHNER: So, at some point we will do a lessons learned, from our perspective, of our review, of this application. And we're taking notes and looking forward to future reviews of advance concepts.

And so there is something learned there about doing a good thorough, early engagement and audit with the Applicant.

DR. LU: Yes.

CO-CHAIR KIRCHNER: You wouldn't want this to be uncovered in December of last year --

DR. LU: At our 11th hour.

CO-CHAIR KIRCHNER: -- finishing your SER.

DR. LU: Thank you.

CO-CHAIR KIRCHNER: Okay.

DR. LU: And actually -- that's actually all our review and our branch review. We worked even late in March in order to do that.

CO-CHAIR KIRCHNER: Okay.

DR. LU: As early as possible.

CO-CHAIR MARCH-LEUBA: I'm getting a little tired of being corrected, so I do on the record. I'm looking at the slide that we were

presented earlier this month and it says nonproprietary everywhere, so I can talk about these numbers.

When the containment pressure was 1,000 psi the original analysis of peak pressure come out with 986 psi.

DR. LU: Right.

CO-CHAIR MARCH-LEUBA: When you stagger, this here. When you stager the opening of the valves it moved to 994. Very small change.

But this is more than half of the margin to 1,000. And after this test, the pressure was increased to 1,050.

MR. THURSTON: The margin for peak containment pressure was initially 1,000, you're right. And the first analysis that provided us with peak containment pressure I think was 951.

CO-CHAIR MARCH-LEUBA: 986.

DR. LU: 951.

MR. THURSTON: It was 951.

CO-CHAIR MARCH-LEUBA: The Revision 1, something.

DR. LU: Right.

MR. THURSTON: So the numbers that you are referring to are later iterations of the peak

containment calculation. At that time, the limiting design pressure had been increased to 1,050.

CO-CHAIR MARCH-LEUBA: Right. But the staggering of the IAV increased by eight psi, which is not much.

MR. THURSTON: Under ten, that's right.

CO-CHAIR MARCH-LEUBA: But it got to 994.

MR. THURSTON: That is correct. So that is the new --

CO-CHAIR MARCH-LEUBA: So it's not insignificant.

MR. THURSTON: Right. So, in comparison to the original design pressure, I see your point. But, all right.

CO-CHAIR MARCH-LEUBA: Yes.

MR. THURSTON: Okay.

DR. LU: With the 1,050 there might be enough to accommodate --

MR. THURSTON: Yes. With 1,050.

CO-CHAIR MARCH-LEUBA: You have 50 psi margin, which you didn't have it.

MR. THURSTON: That's right.

DR. LU: Right. Okay, that's --

CO-CHAIR KIRCHNER: Okay, Members, further questions of Shanlai before he escapes?

All right, so let's move on then to the non-LOCA, no, is this right?

CO-CHAIR MARCH-LEUBA: Non-LOCA. Yes. Non-LOCA open.

CO-CHAIR KIRCHNER: Yes, the non-LOCA open session. And their staff. Alex, do we turn to you?

MS. SIWY: Yes.

CO-CHAIR KIRCHNER: Please, proceed.

MS. SIWY: If everybody is all set. My name is Alex Siwy and I am the technical reviewer in NRR. And I will present to you the staff's review of NuScale's non-LOCA methodology topical report.

As you can see from this slide, we had a lot of contributors to the review. This included staff from NRR, the Office of Research, as well as consultants from Energy Research Incorporated.

And at the table with me here today are Dr. Mohsen Khativ-Rahbar of ERI, and Walter Tauche supporting ERI.

So, to provide a pretty broad overview of the staff's review process, the staff safety evaluation is based on Revision 2 of the topical report.

The staff conducted two audits in four phases as part of its review. And we examined about

140 different audit issues as part of the audits.

Ultimately the audits really helped to confirm the staff's understanding of docketed information, as well as to inform RAIs.

And that's also evidence in the number of RAIs that were ultimately issued. We issued 33 different RAI questions in five RAI letters.

And as of now, all of these issues are resolved. And the responses have been incorporated into Revision 2 of the topical report as appropriate.

The staff conducted its review in accordance with NRC requirements and guidance. Several general design criteria are applicable to non-LOCA transient analysis.

While the staff didn't make direct findings with regard to the GDC as part of the non-LOCA review, the staff did consider how the non-LOCA methodology would be applied in order to support those findings.

The staff also utilized several guidance documents as part of its review. These include SRP Section 1502, review of transient and accident analysis methods, Regulatory Guide 1.203, transient and accident analysis methods, DSRS Section 15.0, introduction, transient and accident analyses.

As well as the event specific SRP in DSRS sections that are applicable to the various non-LOCA events.

So NuScale already discussed some of the Chapter 15 acceptance criteria, but this slide just provides an overview of the acceptance criteria outlined in DSRS Section 15.0. Which is applicable to NuScale design basis events.

And the key takeaways here are that there are acceptance criteria for primary and secondary pressures, fuel failures and event escalation.

NuScale also covered the scope of the non-method, non-LOCA methodology in detail, but just to go through it quickly again.

The non-LOCA methodology is, it provides a conservative methodology for performing system transient analysis of the specific non-LOCA design basis events for the NPM.

The non-LOCA methodology directly evaluates the primary and secondary pressure figures and merit. And it's also used to evaluate whether a staple condition has been reached at the end of the transient.

The non-LOCA EM provides input to the downstream analyses that evaluate margin to specified

acceptable fuel design limits and radiological consequences. So the non-LOCA methodology itself doesn't evaluate against those acceptance criteria.

The non-LOCA EM covers the timeframe during which mixture level is maintained above the top of the riser and natural circulation is maintained. There are separate methodologies that cover the longer term periods after a non-LOCA.

And finally, the scope of the non-LOCA EM includes certain event specific assumptions, as well as conservative bias directions for initial conditions.

So this slide --

CO-CHAIR MARCH-LEUBA: We have seen this slide before.

MS. SIWY: Yes, we can move on. But the key here is that the methodology covers several different events from all event categories. With the exception of the decreased in RCS flow events because they're not applicable to the natural circulation design.

I just wanted to highlight some of the items that the staff is not reviewing in detail or that the staff is not making a finding on as part of the non-LOCA topical report review because these are

more appropriate for a design specific application of the methodology.

These include the representative calculation results in Section 8 of the topical report. Example, values of parameters or parameter biases that are discussed in Section 7 and 8 of the topical report.

The limiting loss of power and single failure scenarios. Any need for operator actions. As well as credit for secondary main steam isolation valves that are not safety related.

So the staff reviewed the entire non-LOCA transient analysis process to make sure that it would result in an acceptable calculation. The non-LOCA EM takes input from upstream methodologies, including nuclear and fuel design.

And the steps within the non-LOCA EM include adapting NRELAP5 base model for event specific analyses. This includes assumptions related to the initiating event, conservative biases and things of that nature.

Then steady state initialization calculations are performed. This includes running no transients. Or basically running a steady state calculation in transient mode to ensure that the

steady state convergence criteria have been met.

The next step is performing the transient calculations, which is pretty self-explanatory. Then the results of the transient calculations will be evaluated against the primary and secondary pressure acceptance criteria.

And finally, the cases for downstream analyses are identified. These include the subchannel analyses and radiological analyses.

CO-CHAIR MARCH-LEUBA: And I've been complaining about this step the whole morning. And part of the afternoon now.

Are those acceptance criteria, and criteria, to define downstream, need for downstream analysis more than identified in the methodology, I don't think so.

MS. SIWY: So, I would say that there are no numerical types of criteria. But the methodology does specify that the pre-screening process can be used as an aid. And if necessary, a spectrum of cases will be sent for downstream subchannel analysis.

CO-CHAIR MARCH-LEUBA: Yes. My problem is, when the CHF correlation that is used is so way off reality, how, unless you use a very conservative screening criteria, how can you rely on it?

MS. SIWY: So, we have pretty good confidence that the trend remains valid. I mean, as NuScale said, it's not going to correctly assess the margin but it does, from what we've reviewed, provide a reliable indicator as to which cases would be limiting.

And for the challenging events, there will be more cases that are subjected to the downstream subchannel analysis.

CO-CHAIR MARCH-LEUBA: What really saves us is that none of these transients' challenge limits anyway. We don't even have to run to know that everything is going to be okay.

But the, as long as you say, will run them all and pick the five worst and then analyze those instead of leaving them to the next, you know, they still going to happen to work out there to do this issue. I mean, there should not be judgement.

You should not have to rely on the good judgement, on the analyst, 20 years from now, to do the right thing. I mean, you should just say, pick the worst five in these categories, or whatever, and then those need to be run with VIPRE. You know what I mean?

I mean, we're leaving it hanging. We're

relying on the good judgment of, grab the student that just has been hired to run these calculations. To do the right thing.

DR. KHATIB-RAHBAR: I think it is desirable to have a perspective value.

CO-CHAIR MARCH-LEUBA: Pull the microphone.

DR. KHATIB-RAHBAR: It is desirable to have a --

CO-CHAIR KIRCHNER: Down below.

CO-CHAIR MARCH-LEUBA: There.

DR. KHATIB-RAHBAR: -- descriptive number. But what they've done I think there is sufficient margin here that a qualitative way of doing it is probably deceptive.

CO-CHAIR MARCH-LEUBA: What I'm saying is that we know what the result is.

DR. KHATIB-RAHBAR: Yes.

CO-CHAIR MARCH-LEUBA: We're just demonstrating it. But I would have liked to see something more prescriptive.

MR. TAUCHE: You know you have margin, but you have to have something for your licensing basis. So I need to determine which are the limiting cases that I want to show as my licensing basis.

And since you have potentially a large number of different cases for the various events, the question becomes then, how do you categorically decide I need to run five cases out of these ten or two cases out of these 20.

And I think what they have done is a reasonable approach to say that these cases have been clearly identified as based upon the dummy hot rod as those cases that are going to be the closest to the limit and therefore those are the ones that need to be subjected to the VIPRE analysis.

CO-CHAIR MARCH-LEUBA: What I mean is that, out of the cases that we run VIPRE already identify?

MR. TAUCHE: They're not already identified but -- and they are subject to interpretation, but what they have shown so far is that their ability to identify the cases that are going to be most limiting, can come from that selection from the hot rod criteria.

So the question is, do I run three or four cases out of 20, do I run five or ten cases out of 20 or do I need to only run those that I know based upon the hot rod dummy analysis are the ones that are going to give me the worst situation.

CO-CHAIR MARCH-LEUBA: And my point is that you are the experts on it, on the applicant, you've done it now. Make the tough decisions now instead of postponing to allow us to do it 20 years from now. That's what I would have done if I was applying this year.

DR. SCHULTZ: Yes. With potentially a higher power core.

CO-CHAIR MARCH-LEUBA: Oh yes. Yes. And we'll go into closed session and you can see how bad the CHF correlation is. I mean, it's embarrassing.

DR. CORRADINI: I'd be very careful on continuing saying bad. Different, maybe, bad, I would disagree.

CO-CHAIR MARCH-LEUBA: It's not conservative.

DR. CORRADINI: I just think that's not, I mean personally, I would call it different.

CO-CHAIR MARCH-LEUBA: What do you mean different?

DR. CORRADINI: Different doesn't mean bad or good, it means different.

MS. SIWY: Right. So I think if a limit were to be established on that correlation that would also look different from the limit on the NSP4

correlation.

CO-CHAIR MARCH-LEUBA: Okay, we'll wait for the closed session.

MS. SIWY: Okay. So anyway, to conclude this slide, the staff reviewed the entire analysis process, including the interfaces with the other methodologies and finds this general process to be acceptable.

The staff focused its review on several key design features of the NuScale design that include, but aren't limited to, the natural circulation design. Which introduces different transient behavior as well as different initialization characteristics relative to a traditional large light water reactor.

In addition, the staff really focused on the helical coil steam generators which are used to transfer heat from the RCS to the feedwater. As well as the passive DHRS condensers. Which transfer the decay heat to the reactor pool via the steam generators.

And the staff also considered the evacuated containment vessel, which, as NuScale pointed out, presents the possibility for a new type of overcooling event.

The staff reviewed the applicability of NRELAP5 to non-LOCA analysis. The staff reviewed how NuScale develops the non-LOCA EM, starting from the LOCA EM, using the graded approach in Regulatory Guide 1.203. And agrees that the regulatory guidance has been followed.

In addition, the staff audited the entire non-LOCA PIRT. And also reviewed the highly ranked phenomena that are described in the topical report. And the staff really wanted to ensure that the important phenomena were adequately addressed as part of the evaluation model.

So, as the Applicant indicated earlier, they identified the deltas and highly ranked phenomena relative to the LOCA EM and addressed those deltas using a variety of methods. Including the separate effects tests at NIST as well as the KAIST and SIET assessments, which the staff really evaluated as part of the non-LOCA review since those assessments are more applicable to non-LOCA transients.

In addition, there were three integral effects tests included as part of the non-LOCA topical report. A code-to-code benchmark against RETRAN-3D. As well as use of bounding input values and other analysis methodologies. In particular, a lot of the

highly arranged phenomena were addressed using the subchannel analysis.

And this brings up one of the first significant issues that the staff encountered during its review. And it's very high level for this slide. But I wanted to mention it for the benefit of the public.

The staff requested additional justification for how certain multidimensional flow effects in the RCS, as well as thermal stratification in the reactor pool are addressed as part of the non-LOCA EM.

The Applicant provided an RAI response, which the staff was able to confirm via audits and some audit discussions with the Applicant.

DR. CORRADINI: Which one was that?

MS. SIWY: This is RAI 9351, Question 15.0.2-31.

DR. CORRADINI: Right. Okay.

(Laughter.)

DR. CORRADINI: You were ready for this, I --

MS. SIWY: Some of this is just ingrained into my memory by now.

(Laughter.)

MS. SIWY: Anyway. The RAI response ultimately confirmed, through the staff's audit, resolved the issue. And I'll provide more details on this in the closed session.

The staff reviewed --

DR. CORRADINI: If I might back you up.

MS. SIWY: Sure.

DR. CORRADINI: So we are going to hear about this in the closed session?

MS. SIWY: Yes.

DR. CORRADINI: Okay.

MS. SIWY: There will be more details then.

DR. CORRADINI: Okay, fine. Thank you.

MS. SIWY: Yes. Okay, so the staff reviewed NRELAP5 assessments against the KAIST. And this one, HP-03 and HP-04 tasks, which predominately deal with DHRS phenomena. Particularly, the condensation inside and the heat transfer across the DHRS tubes. This cooling pool vessel fluid.

The staff confirmed that the ranges of the KAIST and missed one validations adequately cover the expected ranges during DHRS operation in the NPM. In addition, the staff finds that the facility nodalization is acceptable relative to the NPM

nodalization.

DR. CORRADINI: So, can I ask about this?

MS. SIWY: Sure.

DR. CORRADINI: Did the staff do an audit calculation?

MS. SIWY: By audit calculations do you mean --

DR. CORRADINI: Something else with somebody else?

MS. SIWY: So did we perform confirmatory calculations?

DR. CORRADINI: Yes. Yes.

MS. SIWY: The staff performed confirmatory calculations against KAIST in order to validate its TRACE models. So that was one thing that was performed.

DR. CORRADINI: Use TRACE --

MS. SIWY: Yes.

DR. CORRADINI: Okay. Because, I don't know how much I should say now or not. If you're going to come back to this in closed session I'll just wait.

MS. SIWY: We will touch a little bit more on the NIST tests in the closed session, but I wasn't planning to cover KAIST.

DR. CORRADINI: Well, there was a discussion about force convection versus natural, force circulation versus natural circulation.

MS. SIWY: That will come up in the closed session.

DR. CORRADINI: Okay, fine.

MS. SIWY: That relates to the NIST tests.

DR. CORRADINI: Okay, thank you.

MS. SIWY: Sure. And through the staff's audit of underlying calculations, as well as review of the topical report and RAI responses on these topics, the staff finds that the NRELAP5 prediction of the KAIST and HP-03 data is reasonable to excellent. And the agreement, the code-to-data agreement for HP-04 is reasonable.

DR. CORRADINI: What do those adjectives means? Reasonable versus excellent.

So, is there a number that goes with reasonable and a number that goes with excellent or --

MS. SIWY: They are --

DR. CORRADINI: I'm trying to understand because I'm reading it here and --

MS. SIWY: Right. So this terminology is taken out of Regulatory Guide 1.203.

DR. CORRADINI: Okay.

MS. SIWY: So, excellent agreement could be characterized as the trends follow the data in there within, well within the uncertainty bands.

DR. CORRADINI: Okay, fine.

MS. SIWY: And reasonable is they still follow the trends reasonably well but they might disagree in magnitude a little bit more.

DR. CORRADINI: All right, thank you very much.

MS. SIWY: Sure. So, this is a good segue into one of the significant issues that we encountered during our review. And just to preface this, there will be more detail during the closed session.

But one of the issues was removal of the DHRS heat transfer bias from the methodology that was described in Revision 0 and 1 of the topical report.

And the Applicants decided to remove this bias because they performed extensive sensitivity studies that showed that there was an insignificant effect of DHRS performance and nodalization on non-LOCA figures of merit.

And there was some key underlying reasons for this. The first is that the minimum margin to acceptance criteria typically occurs before the DHRS is fully active.

In addition, peak RCS pressure is typically limited by the lifting of the RSB that's credited in NRELAP5. And peak secondary pressure is really a function of the secondary inventory and primary site conditions at the time of DHRS actuation.

So it tends to be insignificantly affected by DHRS heat transfer variations. Especially given the margin in secondary pressure.

So the staff, in conclusion, finds that the sensitivity studies that the Applicant performed, as well as the physical dynamics, support removal of the DHRS heat transfer bias for NPM Model Revision 2. However, there is an associated limitation and condition that I'll discuss at the end of the presentation.

The staff also reviewed the assessment of NRELAP5 against SEIT TF-1 and TF-2 test data. The TF-1 test assessed flow within the steam generator tubes, and TF-2 assessed the primary to secondary heat transfer.

DR. CORRADINI: Can I just, I'm sorry that I'm slow to catch up to you, but can I make sure I understand.

MS. SIWY: Okay.

DR. CORRADINI: When you say bias, you

mean it's a timing bias?

I turn it on or off and it's essentially inapplicable based on SRV operation and timing of the event? That's how I read your three sub-bullets there.

MS. SIWY: Okay. I'm sorry that wasn't clear for you. I'll back up.

DR. CORRADINI: It's okay. A lot of things are unclear for me.

(Laughter.)

MS. SIWY: I take for granite that I've been working on this for quite some time.

DR. CORRADINI: No, no, no, I was just trying to understand. Okay, go ahead.

MS. SIWY: Okay. So the heat transfer bias is basically applied as a multiplier on the heat transfer through the DHRS tubes. So it doesn't have anything to do with timing.

What I was trying to get at with those three bullets is the underlying reasons why that bias doesn't have a significant effect for non-LOCA events.

DR. CORRADINI: But I read the three dashes to mean, the first one occurs because, the MCFR occurs before it kicks in, it's fully active.

MS. SIWY: Yes.

DR. CORRADINI: The second one is because the valve opens before it fully gets to high pressure. So, in some sense it's a matter of timing and operation of a component that limits you. So the bias is --

MS. SIWY: Okay. Yes.

DR. CORRADINI: That's how I read what you were saying.

MS. SIWY: Yes.

DR. CORRADINI: Am I close?

MS. SIWY: Yes.

DR. CORRADINI: Okay.

MS. SIWY: I would agree with that.

DR. CORRADINI: Okay, got it. Thank you.

MS. SIWY: Yes.

DR. CORRADINI: Got it.

MS. SIWY: Okay. So picking up where we left off here.

So, for the SIET RF-1 and TF-2 assessments, the staff audited the underlying calculations and also issued RAIs. Based on this review, the staff finds that there is reasonable to excellent agreement between the NRELAP5 predictions and the SIET test data.

However, the staff raised a related

concern about the range of conditions validated by the TF-2 test compared to expected ranges in the NPM and what impacts there might be on uncertainties and steam generator heat transfer as a result.

This leads into another significant issue that the staff looked at in its review. Which is the removal of the steam generator heat transfer bias.

DR. CORRADINI: Was the bias up or down?

MS. SIWY: It depended. But it could be plus or minus 30 percent. Depending on --

DR. CORRADINI: At a small effect?

MS. SIWY: Yes. That's ultimately what we're getting to. The Applicant performed a lot of sensitivity studies for various events and came to the conclusion that there really wasn't a significant impact on non-LOCA figures and merit.

The staff audited the sensitivity calculations and agrees with those conclusions. The staff also considered the impacts of steam generator heat transfer on both post-trip conditions and ultimately concludes that you're really limited by the DHRS heat transfer at that point because it's so much less than the heat transfer you can get through the steam generators.

In addition --

CO-CHAIR MARCH-LEUBA: Because at this point you have very low heat transfer from the inside of the DHRS tubes. I mean, I realize that there is a limit on the condensation rate on the outside, in the pool simulator, but on the inside of the vessel, when you are in DHRS there is no flow.

You're just basically a stagnant core with water with boiling, pool boiling in there. That's not very high heat transfer.

MS. SIWY: Yes, I think what we're trying to say is that the DHRS is designed to remove decay heat while when the steam generators are removing heat during normal operation. That's a lot --

CO-CHAIR MARCH-LEUBA: When you have flow through there.

MS. SIWY: Yes. That's fair.

CO-CHAIR MARCH-LEUBA: Now, you remove the flow, then you're not removing as much heat as you used to. Anyway, keep going.

(Off-microphone comments.)

CO-CHAIR KIRCHNER: You better turn on your green light.

MR. TAUCHE: Well, it builds up over time through the DHRS once you open the valves. So of course you have low flow at the very beginning, then

the steam condensation occurs, you build the flow through the system.

MS. SIWY: Right. Yes. Then it moves more constant.

CO-CHAIR MARCH-LEUBA: The moment you isolate the secondary, the steam line and you open DHRS, no more amounts of circulation. You only have whatever steam comes out.

MR. TAUCHE: You'll have a decay in the natural circulation while the DHRS builds up.

CO-CHAIR MARCH-LEUBA: Oh. No, the system won't by having a water level and having steam coming out and condensing and dropping down.

MR. TAUCHE: Yes.

CO-CHAIR MARCH-LEUBA: So your flow is sealed. Whatever steam comes up from the top is that water that goes in from the bottom.

MR. TAUCHE: Right. When you open the valve initially there is zero flow. From the DHRS, through the DHRS. So it relies upon the condensation. But that effect is, builds up rather rapidly.

CO-CHAIR MARCH-LEUBA: Right. Some people can call it natural circulation. It's not natural circulation, it's boil off condensation.

MR. TAUCHE: Yes.

MS. SIWY: Boiling condensing, yes.

CO-CHAIR MARCH-LEUBA: The boil off, the velocity of the liquid in the inside is essentially zero?

MR. TAUCHE: At times zero, that's correct.

CO-CHAIR MARCH-LEUBA: On a timing -- that too.

DR. CORRADINI: It's small.

CO-CHAIR KIRCHNER: It's very small.

DR. CORRADINI: I think he was saying small, you're saying super small.

CO-CHAIR MARCH-LEUBA: It's equal --

DR. CORRADINI: But I think you guys are in violent agreement.

CO-CHAIR MARCH-LEUBA: It's equal to whatever --

MR. TAUCHE: Condensate rate basically is what returns to the vessel. That's what it is.

CO-CHAIR MARCH-LEUBA: Which isn't reachable.

MR. TAUCHE: Yes. It's just three parts condensation.

CO-CHAIR MARCH-LEUBA: As far as this board is concerned it is zero.

(Laughter.)

CO-CHAIR MARCH-LEUBA: Anyway, keep going.

MS. SIWY: Okay. The staff also considered the effects of steam generator heat transfer on normal operation and found that technical specifications would account for the effect of variations in steam generator heat transfer.

So these considerations, in addition to the sensitivity studies that the Applicant provided, support removal of the steam generator heat transfer bias for NPM model Revision 2. Again, there will be more details on this during the closed session.

The staff also reviewed NRELAP5 assessments against the NIST-1 integral effects test. This included auditing the underlying calculation notes.

The NLT-02a test was a loss of feedwater transient to the point of a reactor trip. And ultimately the staff concluded that there was reasonable to excellent code-to-date agreement.

NLT-02b test was DHRS driven cooling following initial DHRS actuation. And the staff finds that there is reasonable or a better agreement for this test.

Due to staff questions on the version of

the assessment summarized in Revision 0 and 1 of the non-LOCA topical report, the Applicants also added the NLT-15P2 test, which is a loss of feedwater transient with subsequent DHRS cooling. And the staff finds that there was reasonable code-to-date agreement for this test.

So, looking at all three of these tests, the staff finds that they demonstrate the applicability of NRELAP5 to evaluate non-LOCA transients.

In addition, the staff evaluated the NRELAP5 assessment using the benchmark against RETRAN-3D. RETRAN-3D is a general purpose thermal hydraulics code that the NRC staff has approved for generic use as well as plant-specific applications.

So the staff found that the comparisons of NRELAP5 to RETRAN-3D are useful as part of the NRELAP5 assessment.

The staff reviewed some of the modeling assumptions that the Applicant implemented in the RETRAN-3D model. Which included a simplification since RETRAN-3D does not account for helical coil steam generators.

The staff reviewed, do you have a question? No, okay.

CO-CHAIR MARCH-LEUBA: I was waiting for you to finish.

MS. SIWY: Okay.

CO-CHAIR MARCH-LEUBA: But I do. I do have a question.

MS. SIWY: Okay. The staff reviewed the Applicant's comparison of four different reactivity transients. This includes both a slow and a fast, uncontrolled control rod withdrawal from full power, a power reduction from 100 to 50 percent in a dropped control rod assembly.

Based on these comparisons, the staff finds that there is reasonable or better agreement for each of these transients, which provides confidence in the ability of the NRELAP5 connects model.

CO-CHAIR MARCH-LEUBA: Now, I want to mess with your memory of this event. On this code-to-code comparison, did the Applicant calculate the reactivity worth of the control rod using CASMO/SIMULATE or did they use a RETRAN-3D edit of the reactivity?

Because if they used a written 3D edit of the reactivity, written 3D output, the number that you need to put into the point kinetics to get down sort of the return to RELAP. And I suspect that's what they did.

MS. SIWY: Yes, I don't recall.

DR. KHATIB-RAHBAR: Yes. The reactivity worth of control rods were input to the RELAP model. So RELAP is not used to calculate --

CO-CHAIR MARCH-LEUBA: That is correct.

DR. KHATIB-RAHBAR: -- is retrying this case.

CO-CHAIR MARCH-LEUBA: RETRAN-3D was also point kinetics?

DR. KHATIB-RAHBAR: My understanding was, RETRAN was used with the same control rod worth as it was used in NRELAP.

CO-CHAIR MARCH-LEUBA: Why do you even bother to do the benchmarking. Number one, it's worthless, number two, its guaranteed to succeed. So --

(Simultaneously speaking.)

DR. KHATIB-RAHBAR: -- getting from one point versus the other --

CO-CHAIR MARCH-LEUBA: If both of the models are point kinetics and both of the models have the same reactivity input, both of the models get the same output. I mean, duh. Please, let me repeat it, duh.

MS. SIWY: Okay.

CO-CHAIR KIRCHNER: Remember, this is just one member's opinion.

(Laughter.)

MS. SIWY: Point taken.

CO-CHAIR MARCH-LEUBA: One member in fact understands what you're doing.

MS. SIWY: Okay. Now that we've covered the assessments of NRELAP --

DR. KHATIB-RAHBAR: I guess all I was trying to show, if we use the same feedback effects, the same for hydraulic conditions, the same point kinetics approximation, you get the same answer. That's all it does.

CO-CHAIR MARCH-LEUBA: Was there any doubt?

DR. KHATIB-RAHBAR: Well, there may have been, yes.

MR. TAUCHE: But if RETRAN 3, if RETRAN is acceptable as an NRC method to calculate reactivity effects, then the use of NRELAP5 should be acceptable by the same means for future reactivity calculations.

So, sure, it's an extrapolated step. A equals B, therefore B equals C for future use.

CO-CHAIR MARCH-LEUBA: And it explains your margin for all these terms anyway. So, it's not

an issue but --

MS. SIWY: All right, moving on to some important general aspects of the non-LOCA methodology that the staff reviewed. The staff reviewed the Applicants use of suitable conservative input rather than performing an uncertainty evaluation.

This includes the use of bounding reactivity parameters, conservatively biased initial conditions, as well as conservative valve characteristics.

The methodology also specifies that plant control systems, operation, single failure and loss of power assumptions --

MEMBER BROWN: Can I calibrate myself the first bullet?

MS. SIWY: Yes.

MEMBER BROWN: The way that was read off to me, that sounds like they used a deterministic approach vice a uncertainty, uncertainty type approach. They made judgements as to what was conservative and then went with that as part of their, that's the way I read the fancy language.

I just wanted to make sure I can related it to my past experience. Thank you.

MS. SIWY: Yes, that's correct. Okay,

anyway --

CO-CHAIR KIRCHNER: Also, Charlie, then it's, so they're not using best estimate and uncertainty. I mean, that's really what she's saying.

MEMBER BROWN: That's what I meant.

MS. SIWY: Right.

CO-CHAIR KIRCHNER: Yes.

MEMBER BROWN: Yes, they kind of, so I'm going to weight it here, I'm going to stop it at that point. That's the worst thing I can think of. Roughly. And therefore it's okay. I'm very familiar with that technic.

MS. SIWY: Okay. So the methodology specifies that plant control systems operation, single failure and loss of power assumptions should maximize consequences with regard to acceptance criteria of interests, which is conservative and therefore acceptable.

In addition, the methodology specifies that if acceptance criteria are not sensitive to certain inputs or assumptions, nominal values may be used. Which is fine because there is really not a clear conservative bias direction for those.

In addition, if an acceptance criterion is not challenged by a particular event, it will not be

subjected to in-depth sensitivity studies. And the staff agrees that performing representative calculations for those non-challenged figures of merit is sufficient to determine that they are not challenged.

So, overall the staff finds the general approach for the non-LOCA methodology to be acceptable subject to the applicable limitations and conditions, which we'll get to towards the end of the presentation.

The staff also reviewed the event specific non-LOCA methodology. And these included methodology aspects specific to each initiating event.

The general assessment of which acceptance criteria are challenged by each events. As well as the initial conditions, biases and conservatism.

The staff always finds it acceptable for sensitivity studies to be performed on various parameters as part of a plant specific calculation. Because it's going to ensure that the most limiting cases are captured.

In addition, some conservative bias directions, or assumptions, are specified as part of the non-LOCA methodology. And for those items the staff confirmed that they are indeed appropriate

conservative with respect to their effect on the acceptance criteria.

Overall the staff finds that the event specific methodologies will ensure that conservative results are achieved when the methodology is implemented.

Finally, the staff looked at the representative calculations in Section 8 of the topical report. As far as they demonstrate the application of the non-LOCA methodology. Because the staff is not approving those representative calculations as part of the topical report review.

The staff concludes that the calculations provided show how the non-LOCA methodology can be used for conservative transient analysis simulations.

And the final calculations for the NuScale design are reviewed in the safety evaluation for DCA Chapter 15.

So this slide summaries the limitations and conditions in the staff SER. The first is that future changes to the LOCA topical report must be assessed for impacts to the non-LOCA evaluation model. And that's because the non-LOCA model was developed from the LOCA model.

The second condition is really concerned

with the scope of the non-LOCA evaluation model. Specifically that it's limited to the events described in the topical report prior to the time of riser uncovery for evaluation of primary and secondary pressures and the potential for loss of system functionality.

And the actual limitation and condition specifies some things that it is not approved for.

The third limitation and condition states that additional justification must be provided for the elimination of steam generator in DHRS heat transfer biases if the methodology is to be applied to a design other than NPM model Revision 2 or a model update that is made pursuant to a change process, specifically approved by the NRC for changes to the NPM model.

DR. CORRADINI: So, translate that.

MS. SIWY: Basically, a lot of the justification that the Applicant provided for eliminating those biases was based on sensitivity studies that we don't know how the results might change if there are evolutions of the design.

DR. CORRADINI: Okay. All right, so as designed and as analyzed it's fine, but if they start changing the operating space they may have to go back, someone may have to go back and do additional

sensitivities?

MS. SIWY: Yes, that's correct.

DR. CORRADINI: Thank you.

CO-CHAIR MARCH-LEUBA: Would that include a power operate?

MS. SIWY: That's one potential scenario.

CO-CHAIR MARCH-LEUBA: The other scenario.

MS. SIWY: Yes.

CO-CHAIR KIRCHNER: Now, when you say model, you're referring to actually the nodalization that's used in the code. So if they were to change the approach that was used in that model, how they nodalize the plant, there are a few instances that come to mind where they use pipes for certain components and so on.

Any change to that RELAP model, NRELAP5 model, would then have to be approved.

MS. SIWY: So, we plan, or NuScale plans to submit a change process because we recognize that there are certain changes that might be made that are not important and would not necessitate additional justification.

So we're not saying that any little change to the model would require this. We're still working on this. But we do recognize the burden that could be

placed if we required justification for any little modeling change.

CO-CHAIR KIRCHNER: Okay. See, where I was going with this is, even though my esteemed colleague impugned our ability to understand what's going on here, I do have some background in this area.

And I do know that, actually, the nodalization that you would use with RELAP to try and simulate the actual plant versus the integral tests, is going to be different.

And I would be interested in the closed session to hear from the staff on their own confirmatory calculations. Because it seems to me that without those confirmatory calculations you wouldn't have a lot of confidence going from the integral effects test that NuScale conducted to the actual NPM. Because of scaling and other issues.

And quirkiness, if I might say, of applying, in how the user applies the models and nodalizes the plant going from the integral effects test to the actual plant.

So this is where I see great value in the staff doing confirmatory analysis. Because in many cases, that's a big leap in modeling an integral facility and then modeling the actual prototypical

plant.

And so, in the closed session I'll be interested to hear from the staff, both in LOCA and non-LOCA, how their confirmatory analyses compared with those of the Applicant.

MS. SIWY: Okay. And I will add that one of the areas we reviewed was the, how comparable the nodalization of the test facility was to the actual NPM.

CO-CHAIR KIRCHNER: Yes, precisely. Thank you.

MS. SIWY: Yes.

MEMBER BROWN: Can I ask another? I want to understand the changes.

MS. SIWY: Okay.

MEMBER BROWN: I mean, they've based all this on a specific revision. And obviously, from my past experience in the Naval nuclear program, our labs always were trying to improve the models that for any specific design reactor plant and its application, we had specific versions that were used.

And they were not allowed to alter or change anything for application, for operations of the plants already in place, without our specific agreement to the changes. The headquarters agreement.

And it sounds like you're looking at a way to allow them to make changes without headquarters agreement. Even though you, how you define insignificant versus significant changes is kind of difficult.

Particularly when the licensee thinks he, I'm not impugning anybody's integrity when I say that thinks that these are insignificant whereas your conclusion may be different. It just seems to be a reach to me.

I'm just making a question out of that, it's not a statement. Again, one member's opinion of practice, it's kind of hard to imagine that you would allow the analysis basis that we've used for licensing to be modified, in any way, without you all knowing what the changes were first.

MS. SIWY: Right.

MEMBER BROWN: Now, that's just a comment.

MS. SIWY: Okay.

MEMBER BROWN: That's all it is.

MS. SIWY: That's kind of the idea behind the change process because we recognize that there will be minor error corrections that --

MEMBER BROWN: I hate to say this but we, error corrections still had to go through us, they

were not done without headquarters approval.

MS. SIWY: Okay.

MEMBER BROWN: Before we used them to draw conclusions to acceptability on operations. Maybe we were picky, but that's just the way it is.

DR. CORRADINI: Yes.

MEMBER BROWN: We were really picky. Just as I never allowed changes to my reactor plant protection safeguard systems or instrumentation systems, no changes were made unless they went through me. I was the director of the division, so they all had to go through me one way or the other. And my staff.

So, I'm really particularly picky when it comes to models and --

DR. CORRADINI: We wouldn't have noticed that.

MEMBER BROWN: Pardon?

(Laughter.)

MEMBER BROWN: If you hadn't figured that out yet.

DR. CORRADINI: That escaped my --

MEMBER BROWN: You what?

DR. CORRADINI: That escaped me.

MEMBER BROWN: It did?

DR. CORRADINI: Yes.

MEMBER BROWN: Okay. I'm just, that's a real caution. I think you're walking a very tenuous path when you start delegating change process to what you'd approve for your licensing basis. It's similar to the changes made to the plant issue.

MS. PATTON: So, just to clarify. You know, there are other vendors that have approved changed processes as well. So this is to basically say, if they came in with a change process, and if we approved that change process, then that could be used. That hasn't been done yet.

And obviously any changes to the plant, to the DCA, still have to go through the change processes that are specified in the regulations. So this one is specific to the methodology and this one aspect. And it allows that out, if we approve it.

Right. But it's not inconsistent with what's been done for other vendors.

CO-CHAIR KIRCHNER: I think, Charlie, that if you look at the last bullet, if they actually wanted to change the code and/or the model of the plant they're using, that limit and condition would essentially force them to come back.

So it's RELAP, NRELAP Version 1.5. They

have to come back and I would expect the staff would make them run all the separate effects and integral tests over again to --

MEMBER BROWN: That's not what she said.

CO-CHAIR KIRCHNER: But isn't that the process for change?

You are limiting it to this version of RELAP, Version 1.4.

MEMBER BROWN: You can always go to 1.5, if you get their approval subsequently. But they're saying they're going to allow them to make, a process may be put in place that allows them to make changes under the process, if they fit some category where NRC does not have to review them.

CO-CHAIR MARCH-LEUBA: 1.4 refers to a major release. In-between 1.4 and 1.5 there is probably a hundred changes.

CO-CHAIR KIRCHNER: No, I understand that. Yes.

CO-CHAIR MARCH-LEUBA: Okay. And each of those changes are tested and only major releases get the whole suit of, over education, in this review.

And, Charlie, when you were asking for that review in your office, it probably was taking two days. Almost a week if you're on vacation.

MEMBER BROWN: I hate to say it never just the two days. It was like sucking blood out of rocks. Okay.

CO-CHAIR MARCH-LEUBA: Hold on.

MEMBER BROWN: You had to go through reactor engineering, the safeguards people, the materials people, the physics director, me, who did the plant analysis. Plus the fluid system guys to make sure we didn't muck with their analysis.

CO-CHAIR MARCH-LEUBA: When you send the topical report for Alex to review it takes 18 months.

MEMBER BROWN: Yes. Some of our changes took a year, 18 months. Some of them. They were not done casually, that's all I'm saying.

CO-CHAIR MARCH-LEUBA: Yes.

MEMBER BROWN: I'm just, just an observation.

CO-CHAIR MARCH-LEUBA: All the cause that the industry used are modified, I wouldn't say daily --

MEMBER BROWN: I'm not arguing about that. It's using, when you started using them to do your, evaluate your plant when you've used an earlier version for setting everything in the basis of your operations, your AOO responses, et cetera, et cetera,

et cetera.

CO-CHAIR MARCH-LEUBA: Okay.

MEMBER BROWN: I just think that's a potential problem.

DR. CORRADINI: I think all we're hearing is that there's going to have to be a change process and that that change process has to be approved. That's what I sense is over there in that backroom.

MEMBER BROWN: Their words were used, changes was out, NRC having to see them in terms of --

DR. CORRADINI: Okay. That isn't what I heard by --

MEMBER BROWN: That's what --

DR. CORRADINI: Don't blame her, she's up front. I think that's what Becky said to us.

MEMBER BROWN: She's doing the talking so I presume she talks and then she is amplifying.

(Laughter.)

MS. PATTON: I'm just louder.

MEMBER BROWN: She's putting the brakes on she is providing the insight that we needed.

MS. PATTON: So we would have to approve the change process.

MEMBER BROWN: I understand that.

MS. PATTON: Right.

MEMBER BROWN: Details matter.

(Laughter.)

CO-CHAIR KIRCHNER: Keep going, Alex.

MEMBER BROWN: Excuse me.

MS. SIWY: All right.

MEMBER BROWN: Go ahead. Thank you.

MS. SIWY: So, condition in Limitation 4 is that any credit for the secondary MSIVs, which are not safety related, must be approved through the specific design review.

Limitation Condition 5 is that event-specific electrical power assumptions, single failures and operator actions must be approved through the specific design review.

And finally, Number 6 looks similar to one from the LOCA topical report that the non-LOCA evaluation model use is limited to NRELAP5 Version 1.4 and Version 2 of the NPM model, unless changes are made pursuant to the previously described change process.

So, in conclusion, all technical issues from the course of the non-LOCA review have been resolved. And the staff finds that the use of NRELAP5 with the non-LOCA methodology described in the TR is acceptable for non-LOCA safety analyses of the NuScale

NPM design, subject to the limitations and conditions as summarized on the last slide.

CO-CHAIR KIRCHNER: Thank you, Alex. Members, any further questions of Alex?

PARTICIPANT: That is all.

CO-CHAIR KIRCHNER: Okay. So at this point I would like to use this opportunity for public comment. This finishes are, essentially, open session.

So it's time to turn to anyone in the audience. Member of the public in the audience or others.

If you would wish to make a comment, please come forward to the microphone. State your name and your comment. And then we will open up our bridge line next.

CO-CHAIR MARCH-LEUBA: It's open.

CO-CHAIR KIRCHNER: Seeing that no one has stepped forward, is there anyone out there in the public on the bridge line who wishes to make a comment? If so, state your name and please make your comment.

MS. FIELDS: This is Sarah Fields. The only comment I have is on the NRC slides. They didn't have a list of acronyms at the end. Having that list

at the end to refer to is helpful.

I have a master list that I got from the staff, I don't know, a year or so ago and some of the -- a lot of the acronyms being used were not on that list. So, it's for future slides really helps. Thank you.

CO-CHAIR KIRCHNER: Thank you for your comment. Anyone else wish to make a comment?

Using the five second rule, not hearing further, okay, we can close the bridge line. And at this point I would like to recess for a break. And then we will re-engage at 3:30 on this clock here and go into closed session. Thank you.

(Whereupon, the above-entitled matter went off the record at 3:15 p.m.)

February 17, 2020 Docket No. 52-048

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

SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Subcommittee Presentation: NuScale Topical Report - Rod Ejection Accident Methodology," PM-1019-67365, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on reactor Safeguards (ACRS) NuScale Subcommittee Meeting on February 19, 2020. The materials support NuScale's presentation of the "Rod Ejection Accident Methodology" topical report.

The enclosure to this letter is the nonproprietary presentation entitled "ACRS Subcommittee Presentation: NuScale Topical Report - Rod Ejection Accident Methodology," PM-1019-67365, Revision 0.

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

If you have any questions, please contact Matthew Presson at 541-452-7531 or at mpresson@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

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Enclosure: "ACRS Subcommittee Presentation: NuScale Topical Report - Rod Ejection Accident Methodology," PM-1019-67365, Revision 0

Enclosure:

"ACRS Subcommittee Presentation: NuScale Topical Report - Rod Ejection Accident Methodology," PM-1019-67365, Revision 0

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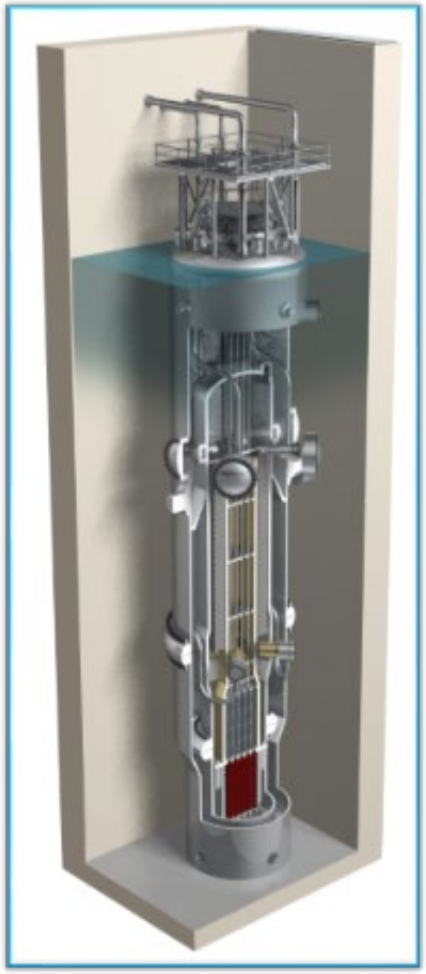
ACRS Subcommittee Presentation



NuScale Topical Report

Rod Ejection Accident Methodology

February 19, 2020

PM-1019-67365
Revision: 0

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Template #: 0000-21727-F01 R5

Presenters

Kenny Anderson
Nuclear Fuels Analyst

Matthew Presson
Licensing Project Manager

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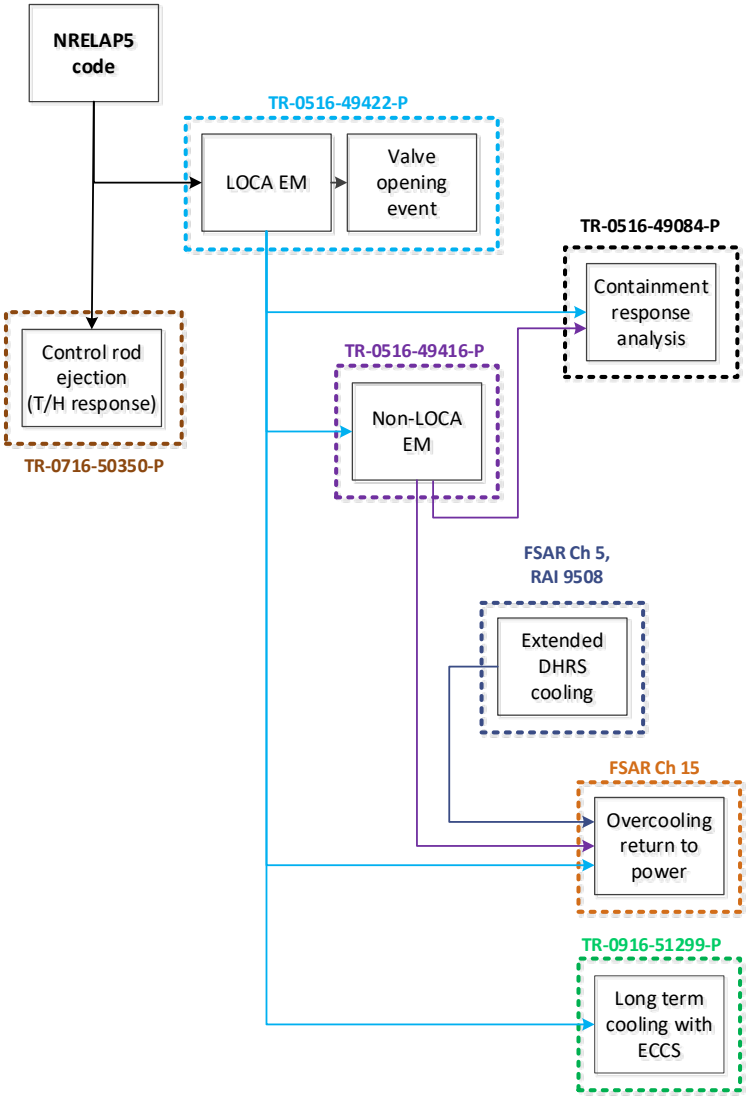
Opening Remarks – NuScale T/H Methods

System T/H Analysis Basis

- NRELAP5 code developed from RELAP5-3D
  - Modified to address NuScale-specific phenomena/systems
- LOCA Evaluation Model (EM) developed following RG 1.203 EMDAP
  - LOCA EM extended to derive EMs for other events as shown in this figure.
  - LOCA EM assessment basis leveraged for non-LOCA.
- Additional supporting EMs include
  - Nuclear Analysis Codes – TR-0716-50350-P-A
  - Critical Heat Flux – TR-0116-21012-P-A
  - Subchannel Analysis – TR-0915-17564-P-A



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Agenda

- Event Overview
- Acceptance Criteria
- PCMI Criteria – DG-1327
- Method Flowchart
- Steady State Initialization
- Event Evaluations
- Summary

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Overview

- NuScale seeks approval of methodology for modeling rod ejection accident (REA) events
- Bounding reactivity initiated accident (RIA) from General Design Criteria (GDC) 28
- REA is unique in comparison to other Ch. 15 events

Description	Rod Ejection	Other Events
Dominant Physics	Nuclear	Thermal-Hydraulics
Timing	milli-sec	sec to hr
Spatially	Local	Global
Peak power	~5x Full Power	~1.2x Full Power
Integrated Energy	Low	Low to High
Postulated Cause	Failure of ASME Class 1 Pressure Boundary	Single Equipment Failure
Acceptance Criteria	Specialized	Generic

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Unique Event Acceptance Criteria

Criteria Description	Topical Section	Unique?
Maximum reactor coolant system pressure	5.3	No
Hot zero power (HZP) fuel cladding failure	5.5.2	Yes
FGR effect on cladding differential pressure	N/A	Yes
Critical heat flux (CHF) fuel cladding failure	5.4.1	No
Cladding oxidation-based PCMI failure	5.5.3	Yes
Cladding excess hydrogen-based PCMI failure	N/A	Yes
Incipient fuel melting cladding failure	5.5.1	No
Peak radial average fuel enthalpy for core cooling	5.5.2	Yes
Fuel melting for core cooling	5.5.1	No
Fission product inventory (failed fuel census)	5.6	Yes

- Submitted NuScale design and method inherently precludes fuel failure, thus no accident radiological consequences are evaluated.
- PCMI: Pellet-Clad Mechanical Interaction

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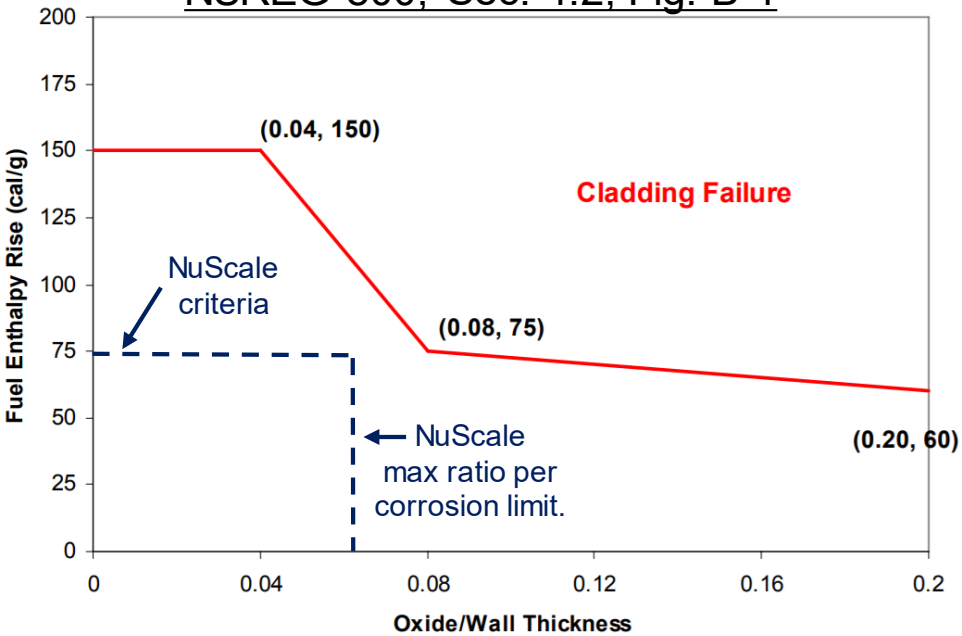
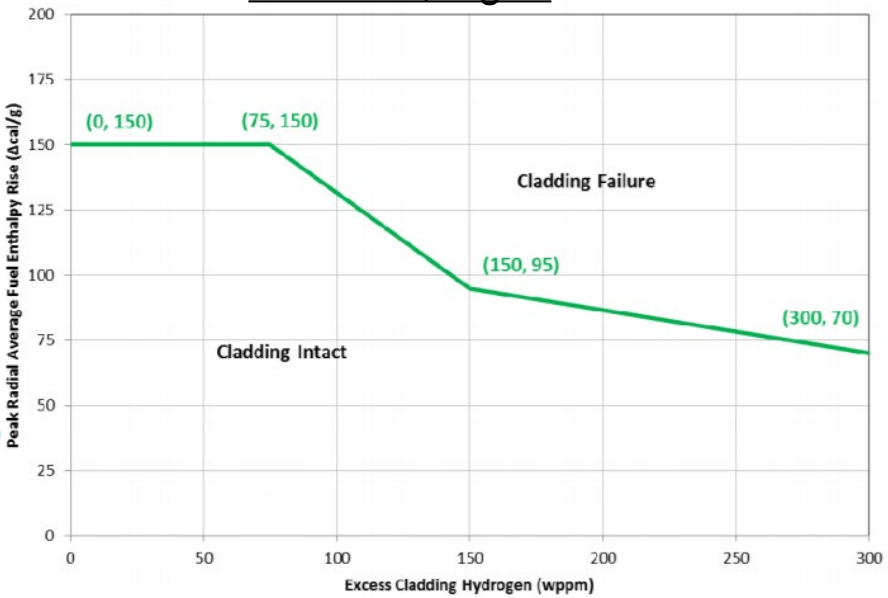
Revised PCMI Criteria

- In general, the NuScale REA methodology has adopted the limiting criteria of the 'Clifford Letter' (ML14188C423), now included in draft guide DG-1327 (ML16124A200). In spirit, NuScale is prepared for this regulatory change:
  - Closed session presents example results, showing large margins for enthalpy rise
  - A technical 'formality' inhibits complete adoption at this time. NuScale does not currently have a validated cladding H_2 model to convert local exposure to excess cladding hydrogen
  - Oxidation criteria from NUREG-0800 Section 4.2, Appendix B (ML07074000) is used
  - To simplify method, no exposure is credited (Limit: 75 Δcal/gm)
  - NuScale M5 cladding less susceptible than other zirc alloy-type clad used in the industry

NUREG-800, Sec. 4.2, Fig. B-1



DG-1327, Fig. 2



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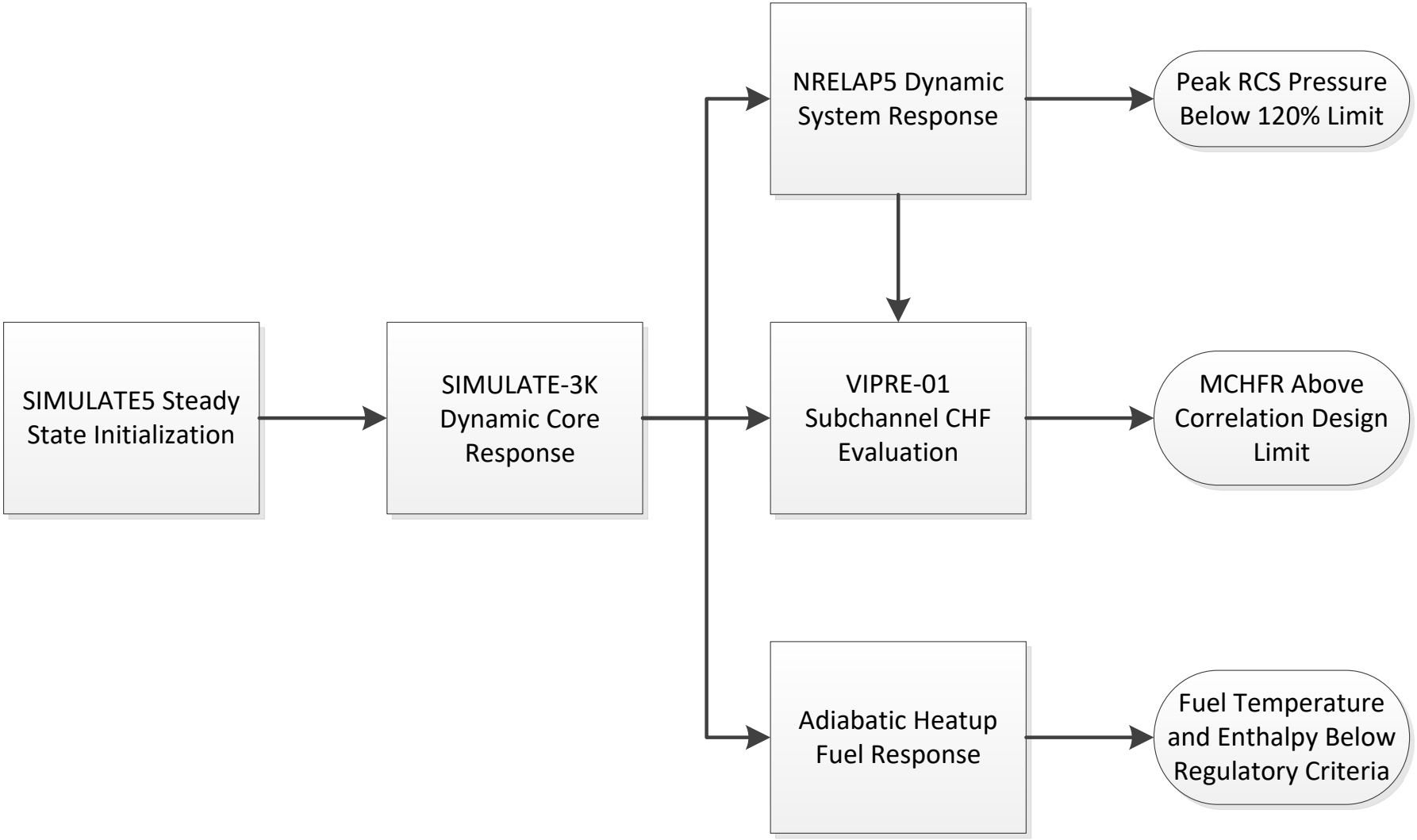
Unique Event Method (Flowchart)



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Steady-State Initialization

- SIMULATE5: Setup the core response analysis
- Code shown to be appropriate in TR-0616-48793-A (Nuclear Analysis Codes and Methods Qualification)
- Determination of the worst rod stuck out (WRSO)
  - Assumption bounds potential for ejected assembly to damage adjacent control rod assembly
  - Due to rapid nature of the event, location does not significantly affect the results in NuScale application

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Dynamic Core Response

- SIMULATE-3K: Model transient core response
- Benchmarked to SPERT-III experiment and NEACRP computational benchmark
  - Benchmarks demonstrate the combined transient neutronic, thermal-hydraulic, and fuel pin modeling capabilities
  - SIMULATE-3K results generally in excellent agreement with the results from the two benchmark problems
- Uncertainties applied for each simulation:
  - Delayed Neutron Fraction
  - Ejected Rod Worth
  - Doppler Temperature Coefficient
  - Moderator Temperature Coefficient

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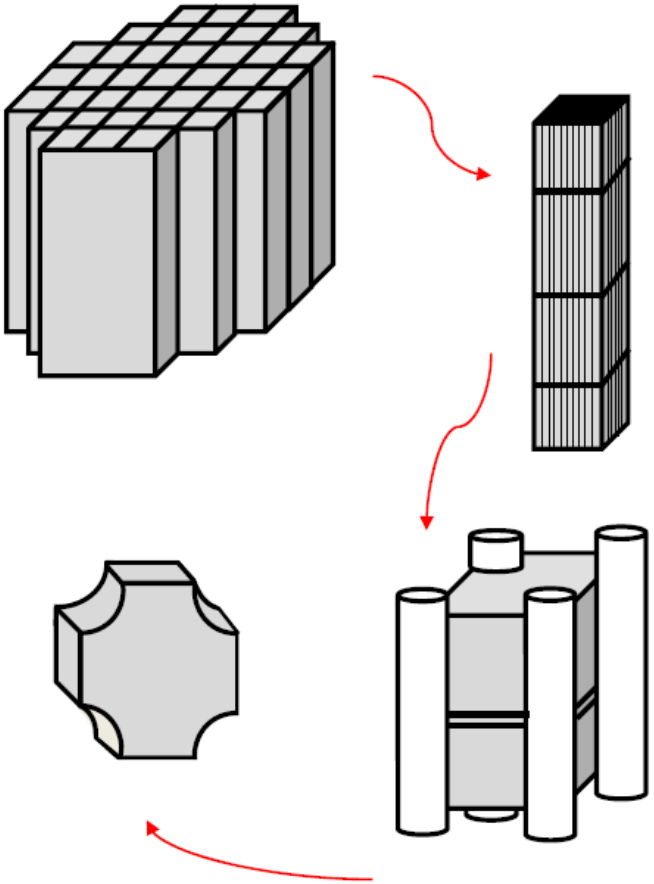
CHF Evaluation

- VIPRE-01: Model detailed thermal-hydraulics
- Evaluate critical heat flux (CHF) acceptance criteria
- Code shown to be appropriate in TR-0915-17564-A (Subchannel Analysis Methodology)
- Unique event differences in method:
  - Smaller axial nodalization (smaller time steps)
  - Radial power distribution (case-specific)
  - Axial power distribution (peak assembly)
  - Convergence parameters
- Additional parametric sensitivity cases performed with each application to holistically justify differences



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Adiabatic Fuel Heatup

- Hand-Calculation: Model fuel response
- Total energy (from SIMULATE-3K) during the transient is integrated
- Conservative as no energy is allowed to leave the fuel rod
- Energy is then converted into either a temperature or enthalpy increase
- Fuel rod geometry, heat capacity, and power peaking factors taken into account
- Calculated values compared to NRC developed acceptance criteria
  - Example values provided in closed session

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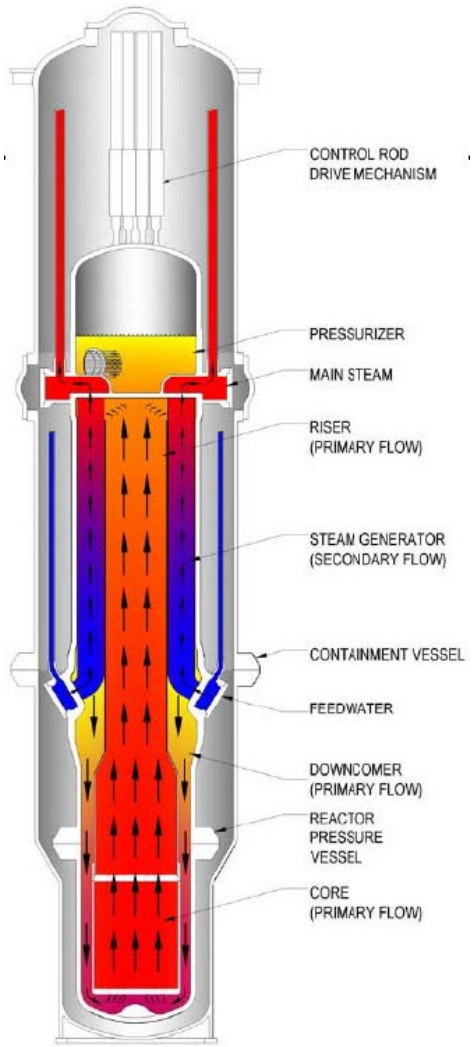
Dynamic System Response I

- NRELAP5: Evaluate system response for input to CHF Evaluation
- Code shown to be appropriate in TR-0516-49416 (Non-LOCA Methodologies)
- Transient power from SIMULATE-3K utilized as input
  - No reactivity calculation performed in NRELAP5
- Provides system thermal-hydraulic conditions to subchannel (CHF) evaluation
  - System flow, pressure, and inlet temperature
  - 'Screens' cases for potential to be limiting
  - Family of limiting cases evaluated with VIPRE-01



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Dynamic System Response II

- NRELAP5: Evaluate system response for pressurization
- Limiting scenario: Low ejected worth that raises the power quickly to just below both the high power and high power rate trip ‘setpoints’
- Point-kinetics model used based on bounding static worth
- Peak system pressure calculated compared to acceptance criteria
- Example results to be presented in closed session

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Summary

- A conservative analysis method for the unique rod ejection accident
- Topical Report provides details and justification for:
  - Software tools and acceptance criteria used
  - Applicability of the method and tools
  - Appropriate treatment of uncertainties
- Results from application of the method provide input to FSAR Chapter 15

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Acronyms

- CHF – Critical Heat Flux
- GDC – General Design Criteria
- HZP – Hot Aero Power
- MCHFR – Minimum Critical Heat Flux Ratio
- NEACRP – Nuclear Energy Agency Committee on Reactor Physics
- PCMI – Pellet Clad Mechanical Interaction
- REA – Rod Ejection Accident
- RIA – Reactivity Initiated Accident
- WRSO – Worst Rod Stuck Out

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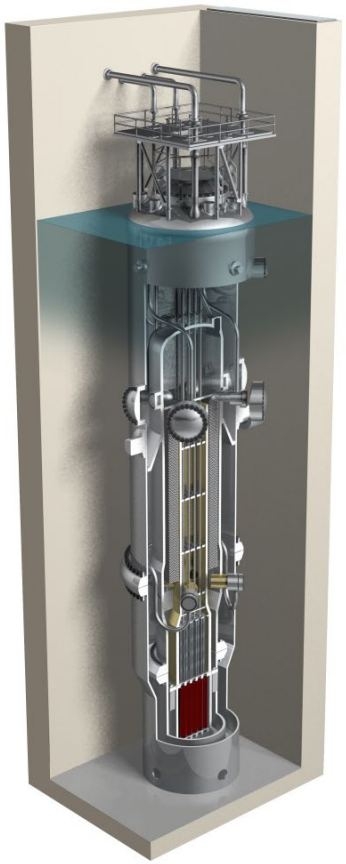

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February 17, 2020 Docket No. 52-048

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SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Subcommittee Presentation: NuScale Topical Report – Loss-of-Coolant Accident Evaluation Model," PM-0220-68917, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) NuScale Subcommittee Meeting on February 19, 2020. The materials support NuScale's presentation of the "Loss-of-Coolant Accident Evaluation Model" topical report.

The enclosure to this letter is the nonproprietary presentation entitled "ACRS Subcommittee Presentation: NuScale Topical Report – Loss-of-Coolant Accident Evaluation Model," PM-0220-68917, Revision 0.

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If you have any questions, please contact Matthew Presson at 541-452-7531 or at mpresson@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
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Enclosure:

"ACRS Subcommittee Presentation: NuScale Topical Report – Loss-of-Coolant Accident Evaluation Model," PM-0220-68917, Revision 0

ACRS Subcommittee Presentation

NuScale Topical Report

Loss-of-Coolant Accident Evaluation Model

February 19, 2020

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Presenters

Matthew Presson
Licensing Project Manager

Dr. Pravin Sawant
Supervisor Code Validation and Methods

Dr. Selim Kuran
Thermal Hydraulic Analyst

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Agenda

- Methodology Overview
  - Background
  - Regulatory Requirements
  - Methodology Roadmap
- NPM Safety Systems Overview
- Element 1: PIRT
- Element 2: Assessment Base
- Element 3: NRELAP5 Evaluation Model
- Element 4: Applicability Evaluation
- Extension of LOCA EM to IORV
- Conclusions

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Background

- Unique NPM Design Features
  - Integrated design eliminates piping and limits potential breaks
  - Coolant captured completely in containment, cooled and returned to RPV using a large pool as ultimate heat sink
- Simple LOCA Progression with Well-Known Phenomena
  - Choked/un-choked flow through break and ECCS valves
  - Core decay heat and RCS stored energy release
  - CNV heat transfer to pool (condensation, conduction, convection)
- EM Development Approach
  - Follows Regulatory Guide 1.203 EMDAP (Table 2-1)
  - Compliance with 10 CFR 50.46 and Appendix K requirements (Table 2-2)

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Regulatory Requirements

- 10 CFR 50.46 Acceptance Criteria
  - Max. clad temperature < 2200 ºF
  - Cladding oxidation > 0.17 times thickness
  - Hydrogen generation < 0.01 times total hydrogen from oxidation of all cladding
  - Core remains amenable to cooling
  - Long-term cooling maintained
- Maximum PCT at steady state, no clad heat up
- Conservative LOCA EM Acceptance Criteria (FOMs)
  - Core remains covered: collapsed level > TAF
  - MCHFR > CHFR Limit (1.29)
  - Containment pressure and temperature below design limit

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Methodology Roadmap

- 10 CFR 50.46 Appendix K Compliance (Section 2.2.3 of LTR)
- RG 1.203 EMDAP (Section 2.1 of LTR)

Element 1
Establish Requirements for EM Capabilities
LOCA PIRT (*LTR Ch. 4*)

Element 2
Develop Assessment Base
SET and IET Assessment (*LTR Ch. 7*)

Element 3
Develop Evaluation Model
NRELAP5 Code (*LTR Ch. 6*)
Evaluation model (*LTR Ch. 5*)
Break spectrum and sensitivity calcs. (*LTR Ch. 9*)

Element 4
Assess EM Adequacy
Top-down and Bottom-up evaluation (*LTR Ch. 8*)

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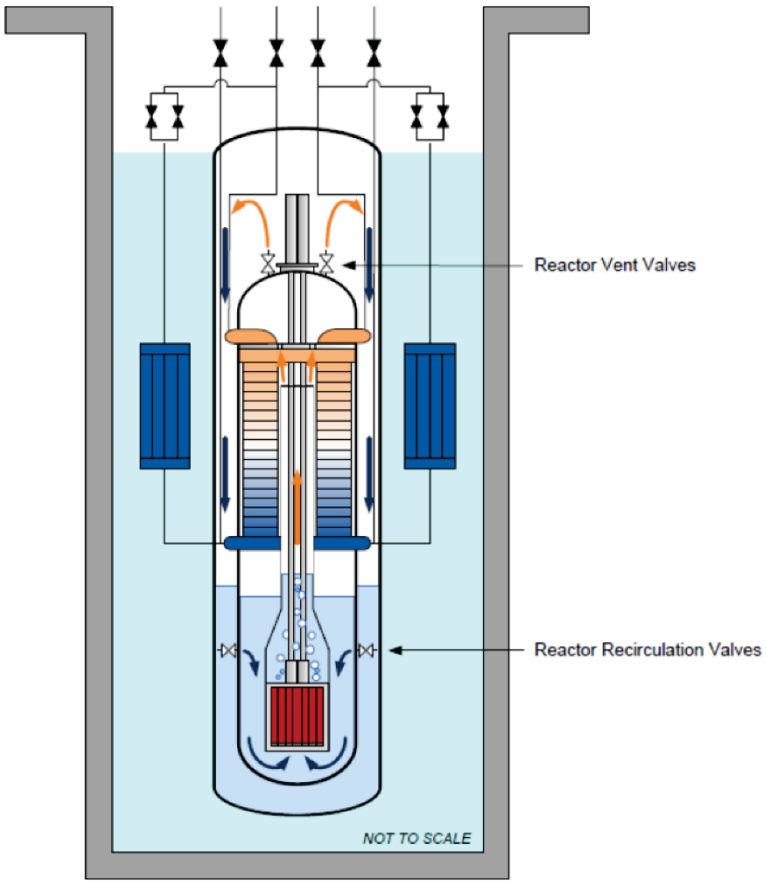
NPM Safety Systems

- ECCS
  - Opens a boiling/condensing circulation flow path to transfer decay and residual heat to reactor pool
  - Reactor Recirculation Valves (RRV): 2 valves
  - Reactor Vent Valves (RVV): 3 valves
  - Actuation Signals: High CNV level, 24-hour loss of AC power
  - Fail safe: ECCS trip valves open on loss of DC power
- Inadvertent Actuation Block (IAB)
  - Prevents inadvertent opening of ECCS valves at high RCS pressure
  - Actuation based on differential pressure between RPV and CNV
- Module Protection System (MPS)
  - Reactor scram
  - Steam Generator (SG) and Containment (CNV) Isolation
  - Passive safety system activation (ECCS and DHRS)
- Decay Heat Removal System (DHRS)
  - Passive, boiling-condensation system
  - Removes heat from RCS through SG via two trains
  - Each trains capable of removing 100% decay heat
  - Not credited in LOCA EM



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Element 1
PIRT

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PIRT Process

- Assessment of relative importance of phenomena
  - Unique phases
  - Key components
- PIRT panel included recognized experts and NuScale subject matter experts
    - State-of-knowledge, design description, LOCA description, NRELAP5 calculations
- Figures-of-Merit
  - CHF, Collapsed level above top of the active fuel, CNV P & T
- Rankings
  - Importance: High, Low, Medium, Inactive
  - Knowledge: Well known (small uncertainty), Known (moderate uncertainty, partially known (large uncertainty), very limited

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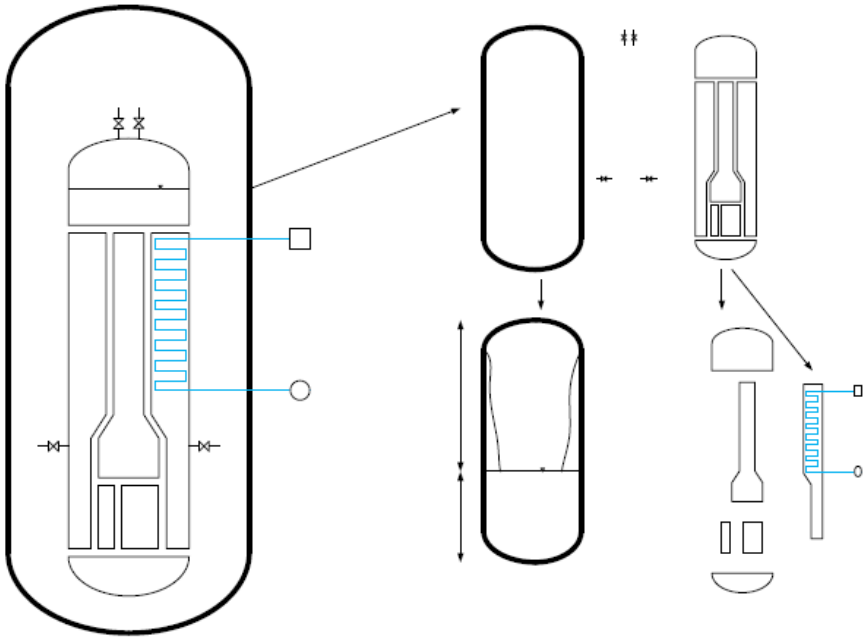
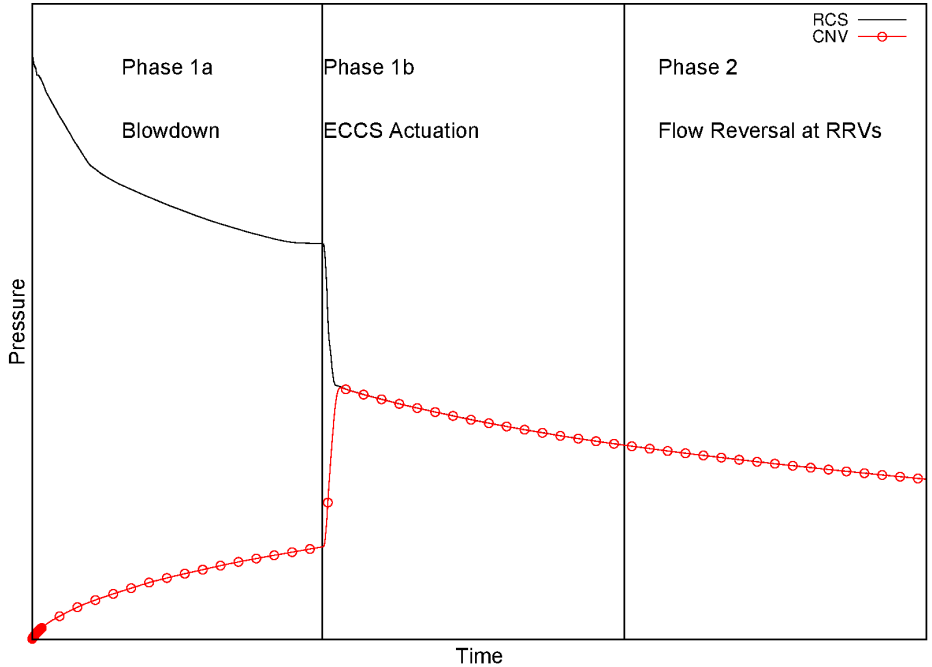
Spatial and Temporal Decomposition

- Phenomena identified for Systems, Structure, Components (SSCs) and LOCA phases
  - Phase 1a: Blowdown
  - Phase 1b: ECCS activation (opening)



System/Subsystem/Module decomposition



Distinct phases of a typical NPM LOCA

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Element 2
Assessment Base

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NRELAP5 Code

- RELAP5-3D© v4.1.3 used as a baseline code
  - Two-fluid model (thermal and mechanical non-equilibrium) for hydrodynamics with
    - Non-condensable gases with gas phase
    - Semi-implicit scheme for time integration
  - Heat conduction across 1D geometries (slab, cylinder, sphere)
  - Neutron Kinetics with thermal hydraulic feedback
  - Special Process Models
  - Comprehensive control/trip system modeling
- Code configuration control and development consistent with NuScale's NQA-1 2008 / 2009a QA program
- Modifications for NRELAP5:
  - NuScale specific components (e.g., helical coil SG)
  - Regulatory requirements (i.e., Appendix K)
  - Error correction

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IET and SET Data

- Extensive database with adequate coverage of all high-ranked phenomena
- Integral effects tests (IET)
  - Six (6) NIST-1 tests
- Separate effects tests (SET)
  - Two (2) NIST-1 SETs
  - Four (4) other NuScale SETs
  - Nine (9) Legacy SETs

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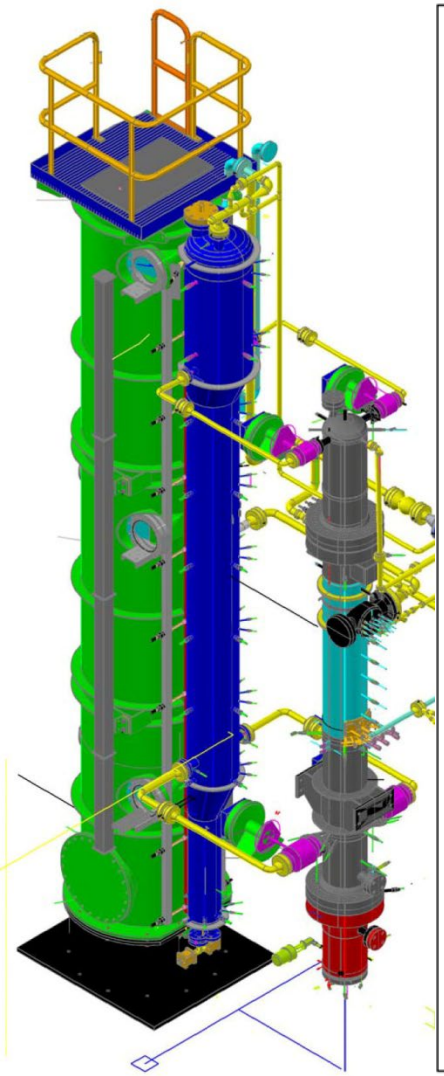
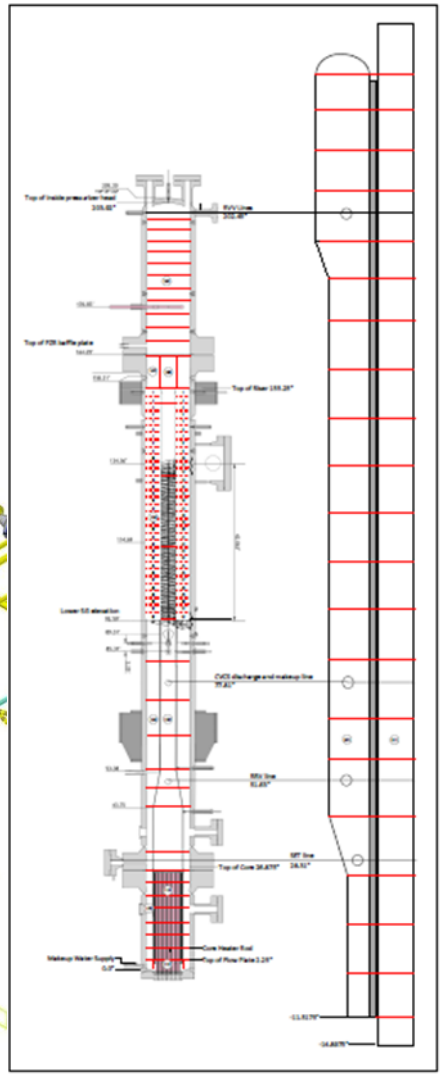
NIST-1 Facility





- Primary source of NuScale-Specific IET and SET data
- Design Features
  - Integral Reactor Vessel with electrically heated rod bundle core, helical coil steam generator, and pressurizer
  - Containment with HTP and Cooling Pool
  - DHRS, ECCS, CVCS lines represented
  - ~700 instruments
- Scaling Basis
  - Power/Volume Scaling
  - Reduced height and reduced volume scale
  - Full Pressure and Temperature
  - Same Time Scale (isochronicity)

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Element 3
NRELAP5 NPM LOCA

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NPM LOCA Model Overview

- Analysis and Justifications
  - NRELAP5 model nodalization and input options
  - Time-step control
  - Initial and boundary condition biases
  - Treatment of setpoints and trips
- LOCA break spectrum
  - Break location and sizes
  - Single failures
  - Power availability
- Methodology sensitivity calculations
  - Required by Appendix K
  - Phenomena-specific
  - To establish conservative biases

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Element 4
Applicability Evaluation

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Applicability Evaluation

- Evaluated models and correlations (bottom-up)
  - Identified dominant models/correlations for ‘H’ phenomena (Table 8-1 of LTR)
  - Identified key model/correlation parameters and phenomenological domain where models/correlations are used (Tables 8-2 and 8-4)
  - Reviewed models/correlations (Table 8-18 of LTR)
    - Pedigree, Applicability range, Fidelity to SET data, Scalability
- Evaluated integral performance of EM (top-down)
  - Reviewed code governing equations and numerics
  - Evaluated integral performance of code using IET data (Table 8-19 of LTR)
  - Evaluated IET data applicability and NRELAP5 scalability
    - Scaling and distortion analysis
    - Differences and distortions between NPM and NIST can be accounted using NRELAP5

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Conclusions

- Number of conservatisms built into the NuScale LOCA EM
  - 10 CFR 50 Appendix K
  - Other methodology conservatisms
- Cycle independent bounding LOCA analysis
- Supported by extensive experiment database, well qualified code, and several sensitivity calculations
- Applicability evaluation consistent with RG 1.203
- CHF not challenged
- Collapsed level in RPV remains above TAF
- No clad or fuel heat-up
- CNV P&T remain below design limits

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Appendix B to LOCA LTR

Extension to IORV Event

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IORV Background

- LOCA EM Extended to IORV
  - Liquid space (RRV) and steam space (RVV, RSV) discharge
  - Similar transient phenomena and progression
- EM Development Approach
  - Compliance with DSRS for NuScale SMR Design 15.6.6
  - Follows RG 1.203 EMDAP
  - Element 1 (PIRT), Element 2 (Assessment), and Element 4 (Applicability) remains same as LOCA EM
    - Initial LOCA PIRT addressed IORV
  - Element 3 (NRELAP5 Model) unique due to event classification

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Differences from LOCA EM

- Minor methodology differences given AOO classification
- Key Acceptance Criteria
  - MCHFR ≥ Limit (≥ 1.13 high flow range, ≥ 1.37 low flow range)
- Conservatisms same as LOCA with exceptions:
  - Fuel properties still biased to maximize stored energy, but additional 15% bias removed
  - Limiting axial power shapes and radial peaking based on subchannel analysis
  - Moody choked flow model for 2-phase flow choking applied to initiating valve
  - Initial conditions biased to minimize MCHFR

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Conclusions

- IORV is an extension of LOCA EM given similar transient phenomena and progression
  - PIRT, Assessment, and Applicability same as LOCA
- Minor methodology differences for AOO classification
  - Focused on conservative CHFR evaluation
- MCHFR occurs early in transient, then rapidly rises given increasing flow to power ratio
- Collapsed level in RPV remains above TAF

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Acronyms

1-D	one-dimensional
3D	three-dimensional
AC	alternating current
ANS	American Nuclear Society
CCFL	counter current flow limitation
CHF	critical heat flux
CNV	containment vessel
CVCS	chemical and volume control system
DC	direct current
DCA	Design Certification Application
DHRS	decay heat removal system
ECCS	emergency core cooling system
EM	evaluation model
EMDAP	evaluation model development and assessment process
FW	feedwater
FSAR	Final Safety Analysis Report
FOM	figure of merit
HL	hot leg
HP	high pressure
HS	heat sink
HTP	heat transfer plate
H2TS	hierarchical two-tiered scaling
IAB	inadvertent actuation block
IET	integrated effects test
INL	Idaho National Laboratory
KATHY	Karlstein thermal-hydraulic test facility
kW	kilowatt
LOCA	loss-of-coolant accident
LTR	Licensing Topical Report
Max	maximum
MCHFR	minimum critical heat flux ratio
Min	minimum
$Mlb/ft^2 \cdot hr$	pounds mass per square foot per hour
MPS	module protection system
MSIV	main steam isolation valve
NIST-1	NuScale Integral System Test Facility
NPM	NuScale Power Module

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Acronyms

P&T	pressure and temperature
PCT	peak cladding temperature
PIRT	phenomena identification and ranking table
psi	pounds per square inch
psia	pounds per square inch absolute
PZR	pressurizer
QA	Quality Assurance
RCS	reactor coolant system
RG	Regulatory Guide
RRV	reactor recirculation valve
RPV	reactor pressure vessel
RVV	reactor vent valve
SG	steam generator
SET	separate effects test
SIET	Società Informazioni Esperienze Termoidrauliche
StDev	standard deviation
TAF	top of active fuel

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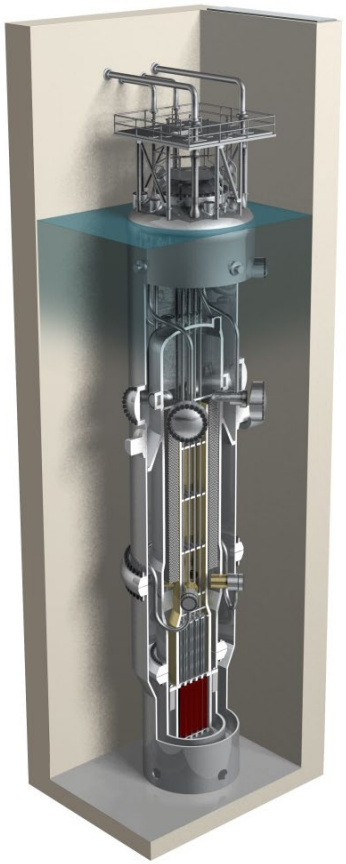

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Backup IORV Slides

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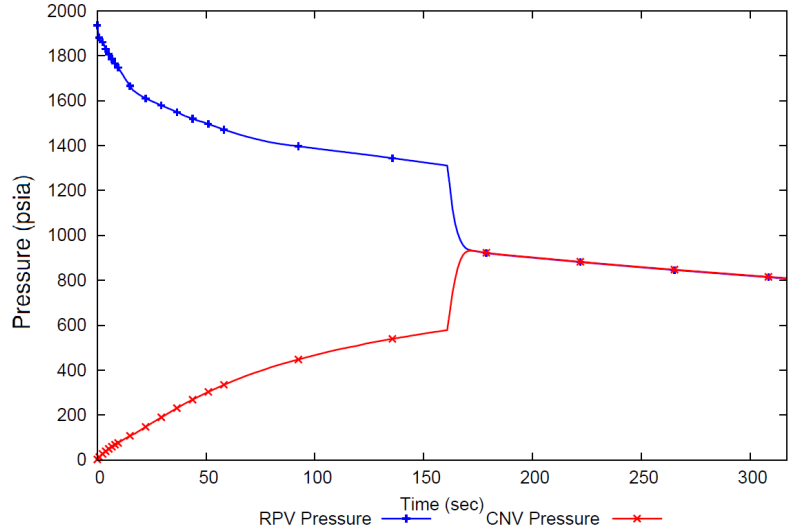
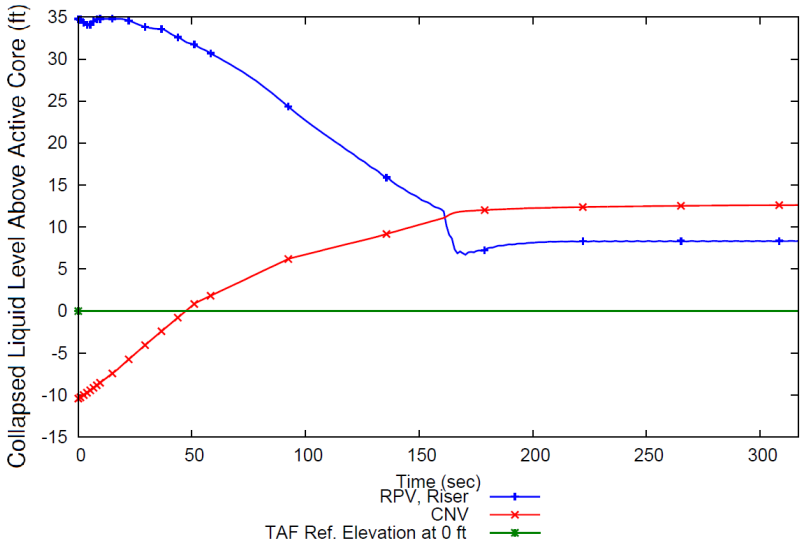
RRV Opening Results

Sequence of Events	Time (s)
RRV opens	0
Minimum CHFR occurs	0.5
Containment pressure reaches analytical limit	0.7
Control rods begin to fall	2.7
Peak steam generator pressure is reached	64
Remaining ECCS valves open	161
Minimum collapsed liquid level above the core	170
Peak containment pressure is reached	171
Natural circulation from containment to RPV is established	197





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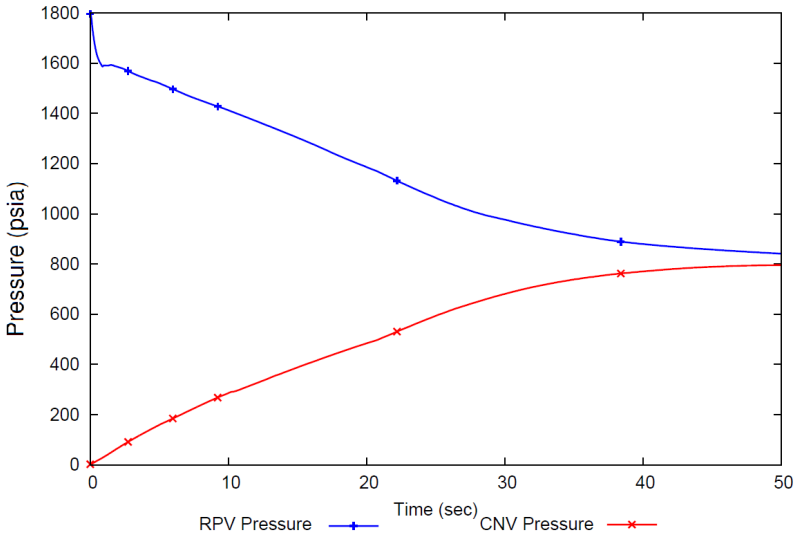
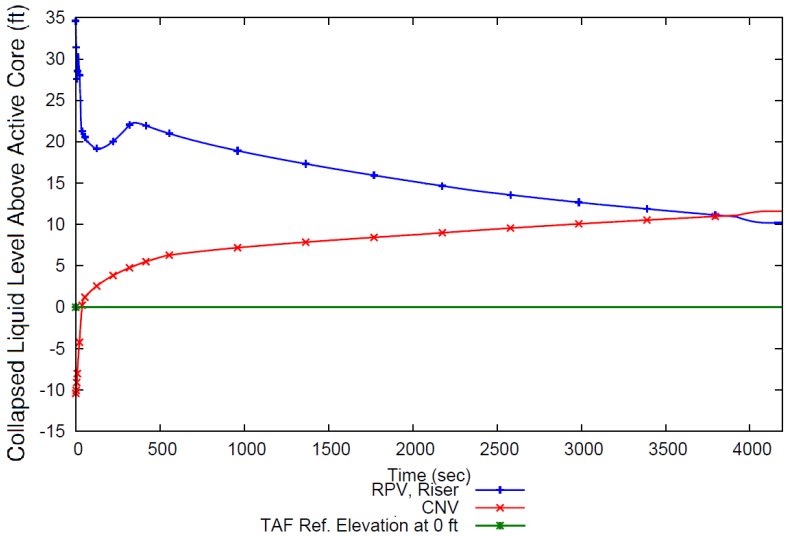
RVV Opening Results

Sequence of Events	Time (s)
RVV opens	0
Containment pressure reaches analytical limit	0.3
Minimum CHFR occurs	0.3
Control rods begin to fall	2.3
Peak steam generator pressure is reached	25
Peak containment pressure is reached	52
Remaining ECCS valves open	3925
Natural circulation from containment to RPV is established	4072
Minimum collapsed liquid level above the core	4192





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February 17, 2020 Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Subcommittee Presentation: NuScale Topical Report – Non-Loss-of-Coolant Accident," PM-0220-68852, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) NuScale Subcommittee Meeting on February 19, 2020. The materials support NuScale's presentation of the "Non-Loss-of-Coolant Accident Analysis Methodology" topical report.

The enclosure to this letter is the nonproprietary presentation entitled "ACRS Subcommittee Presentation: NuScale Topical Report – Non-Loss-of-Coolant Accident," PM-0220-68852, Revision 0.

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

If you have any questions, please contact Matthew Presson at 541-452-7531 or at mpresson@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

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Enclosure: "ACRS Subcommittee Presentation: NuScale Topical Report – Non-Loss-of-Coolant Accident," PM-0220-68852, Revision 0

Enclosure:

"ACRS Subcommittee Presentation: NuScaleTopical Report – Non-Loss-of-Coolant Accident," PM-0220-68852, Revision 0

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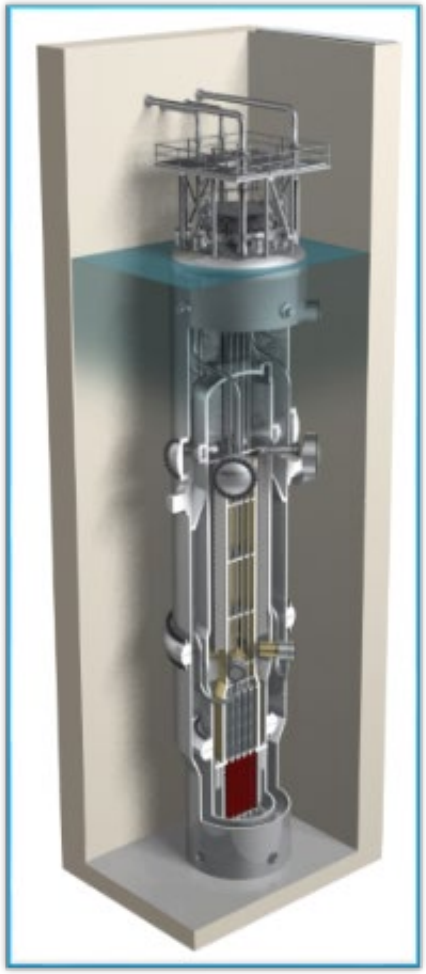
ACRS Subcommittee Presentation



NuScale Topical Report

Non-Loss-of-Coolant Accident

February 19, 2020

PM-0220-68852
Revision: 0

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Template #: 0000-21727-F01 R5

Presenters

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Outline

- Scope of non-LOCA LTR
- Non-LOCA events
  - Events and acceptance criteria
  - Interface to other methodologies
  - Factors controlling margin to acceptance criteria
- Development of non-LOCA EM
  - PIRT and gap analysis
  - Focus of NRELAP5 validation for non-LOCA
- General event analysis methodology
- Specific event analysis

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Scope of Non-LOCA Topical Report

In Scope

- NRELAP5 system transient analysis of non-LOCA events
- Interface to subchannel and accident radiological analysis
- Short-term transient progression with DHRS cooling

Out of Scope

- SAFDLs evaluated in downstream subchannel analysis
- Accident radiological dose analysis
- Control rod ejection
- LOCA and valve opening events
- Peak containment pressure/temperature analysis
- Long term transient progression with DHRS
  - Riser uncovery
  - Return to power

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Non-LOCA EM

EM applicable to NuScale Power Module plant design
Applicable initiating events:

- **Cooldown events**
  - Decrease in FW temperature
  - Increase in FW flow
  - Increase in steam flow
    Inadvertent opening of SG relief or safety valve
  - Steam piping failures (postulated accident)
  - *Loss of containment vacuum*
    Containment flooding

- **Heatup events**
  - Loss of external load
    Turbine trip
  - Loss of condenser vacuum
  - Closure of MSIV
  - Loss of non-emergency AC power
  - Loss of normal FW flow
  - Feedwater system pipe breaks (postulated accident)
  - *Inadvertent operation of DHRS*

- **Reactivity events**
  - Uncontrolled bank withdrawal from subcritical
  - Uncontrolled bank withdrawal at power
  - Control rod misoperation
    - Single rod withdrawal
    - Control rod drop
  - Inadvertent decrease in RCS boron concentration

- **Inventory increase event**
  - CVCS malfunction

- **Inventory decrease events**
  - Small line break outside containment (infrequent event)
  - Steam generator tube failure (postulated accident)

NuScale unique event

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Non-LOCA Event Acceptance Criteria

Description	AOO Acceptance Criteria	Infrequent Event Acceptance Criteria	Accident Acceptance Criteria	Analysis
Reactor Coolant System Pressure (P_{design}= 2100 psia)	**≤ 110% of Design**	**≤ 120% of Design**	**≤ 120% of Design**	**Non-LOCA NRELAP5**
Steam Generator Pressure (P_{design}= 2100 psia)	**≤ 110% of Design**	**≤ 120% of Design**	**≤ 120% of Design**	**Non-LOCA NRELAP5**
Minimum Critical Heat Flux Ratio	> Limit	If limit exceed, fuel assumed failed (1)	If limit exceed, fuel assumed failed (1)	Subchannel
Maximum Fuel Centerline Temperature	< Limit	If limit exceed, fuel assumed failed (1)	If limit exceed, fuel assumed failed (1)	Subchannel
Containment Integrity	< Limits (pressure, temperature)	< Limits (pressure, temperature)	< Limits (pressure, temperature)	Containment P/T analysis
Escalation of an AOO to an accident (AOO) or Consequential loss of system functionality (IE or accident)?	**No**	**No**	**No**	**If other acceptance criteria are met**
Radiological Dose	Normal Operations	< Limit	< Limit	Normal or Accident radiological

(1) NuScale safety analysis methodologies developed to demonstrate fuel cladding integrity maintained.

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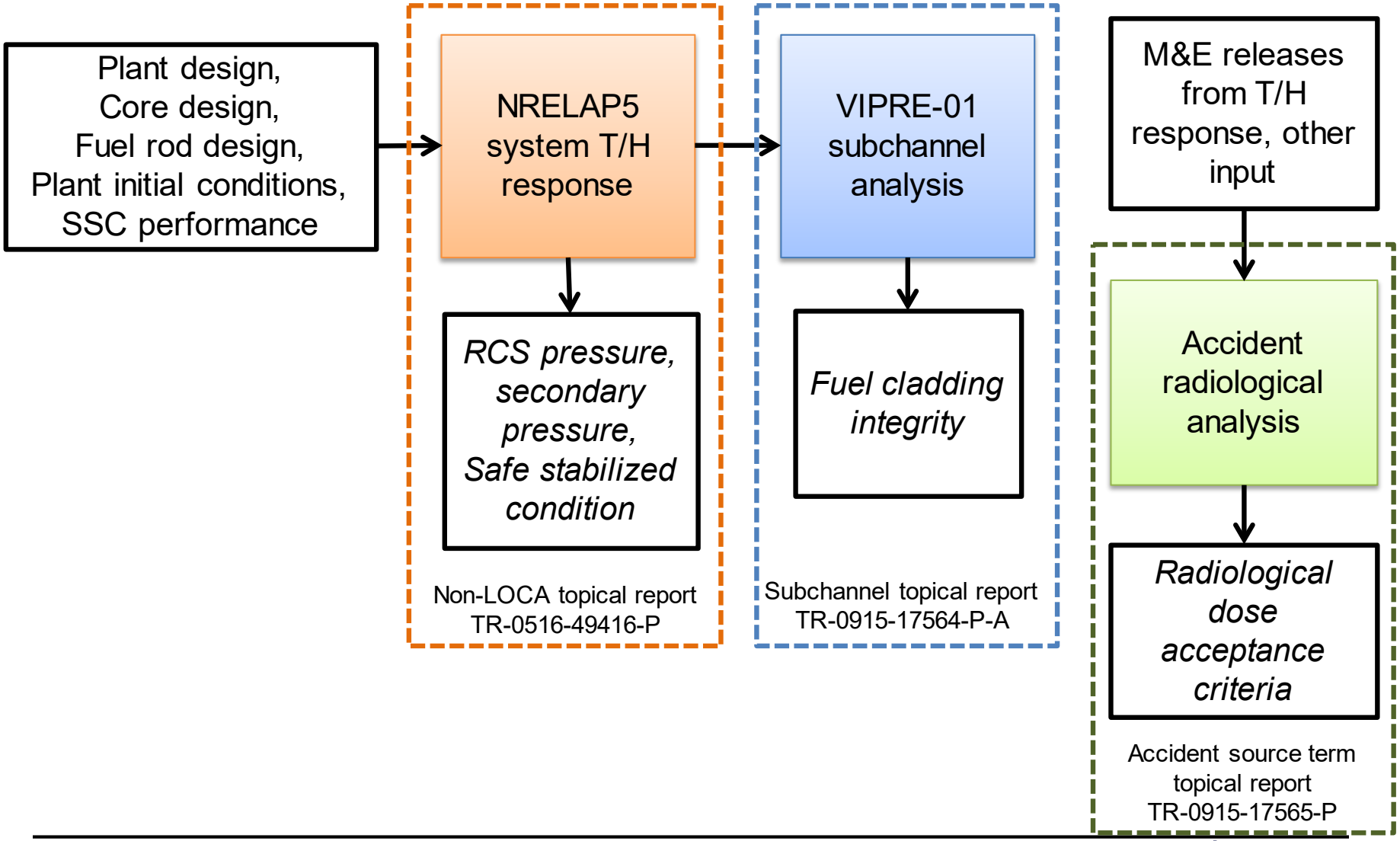
Evaluation Models – General Non-LOCA Approach



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Non-LOCA Events - Margin to Acceptance Criteria

Design characteristics governing non-LOCA event transient response and margin to acceptance criteria

- MCHFR: Limited by combination of high power, high pressure, high temperature conditions occurring around time of reactor trip, for reactivity insertion events
- Primary pressure: Protected by RSV lift
- Secondary side pressure: Limited by primary side temperature conditions
- Radiological release: MPS designed to rapidly detect and isolate based on measured conditions
- Establishing a safe, stable condition: MPS designed to trip, actuate DHRS to protect adequate inventory in at least 1 steam generator

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Non-LOCA EM Development

- Non-LOCA evaluation model developed to perform conservative analyses, following intent of the RG 1.203 EMDAP and applying a graded approach
- Element 1 – Establish applicable transients and acceptance criteria, develop non-LOCA PIRT
- Element 2, 3, 4
  - Leverage NRELAP5 development, NRELAP5 assessments performed during LOCA evaluation model development.
    - Gap analysis performed to evaluate how high ranked phenomena are addressed
    - Focused on differences in high ranked PIRT phenomena between LOCA and non-LOCA
    - Additional NRELAP5 code validation performed focused on DHRS and integral non-LOCA response
  - Suitably conservative initial and boundary conditions applied for non-LOCA analyses
  - Sensitivity calculations used to demonstrate factors controlling margin to acceptance criteria

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Non-LOCA PIRT Development

Event Types
Increased heat removal
Decreased heat removal
Reactivity anomaly
Increase in RCS inventory
Steam generator tube failure

SSCs Considered in PIRT	
Reactor coolant system	Main feedwater system
Containment vessel	Main steam system
Decay heat removal system	Chemical volume control system
Reactor pool	Containment evacuation system

Phase	Identification	RCS Response	DHRS Operation *	PIRT Figures of merit
1	pre-trip transient	higher flow levels at full power levels	inactive	CHFR RCS pressure
2	post-trip transition	transitional flow levels at transitioned power levels	startup	CHFR RCS, secondary, containment pressures
3	stable natural circulation	lower flow levels at decay power levels	fully effective	CHFR RCS mixture level Subcriticality

* If DHRS actuated by protection system

- Different non-LOCA events involve different plant systems and responses
- PIRT developed considering all non-LOCA event types and important SSCs
- Short-term response divided into 3 generic phases with associated FoM

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NRELAP5 Applicability for Non-LOCA

After non-LOCA PIRT developed, gap analysis performed to determine how to address high-ranked phenomena:

- Validation performed as part of NRELAP5 assessment for LOCA evaluation model
- Additional validation or benchmark for non-LOCA
- Conservative input
- Subchannel analysis

Key areas identified from gap analysis for short-term non-LOCA analysis:

- DHRS modeling and heat transfer
  - NRELAP5 validation against KAIST tests; NIST-1 SETs HP-03, HP-04
  - NPM sensitivity calculations
- Steam generator modeling and heat transfer
  - NRELAP5 validation against SIET-TF1, SIET-TF2 tests
  - NPM sensitivity calculations
- Reactivity event response
  - NRELAP5 benchmark against RETRAN-3D
- NPM non-LOCA integral response
  - NRELAP5 validation against NIST-1 IETs NLT-2a, NLT-2b, NLT-15p2

Overall conclusion: NRELAP5 code, with NPM system model, is applicable for calculation of the NPM non-LOCA system response

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Non-LOCA Analysis Process

Topical report Section 4

1. Develop plant base model NRELAP5 input (geometry, control and protection systems, etc)
2. Adapt NRELAP5 base model as necessary for specific event analysis and desired initial conditions
3. Perform steady state and transient analysis calculations with NRELAP5
4. Evaluate results of transient analysis calculations:
   - Confirm margin to maximum RCS pressure acceptance criterion
   - Confirm margin to maximum SG pressure acceptance criterion
   - Confirm appropriate transient run time execution to demonstrate safe, stabilized condition achieved
5. Identify cases for subchannel analysis and extract boundary conditions (if applicable)
   - Conservative bias directions:
     - Maximum reactor power
     - Maximum core exit pressure
     - Maximum core inlet temperature
     - Minimum RCS flow rate
   - NRELAP5 CHF calculations for dummy hot rod may be used as a screening tool to assist analysts in determining limiting cases to be evaluated in downstream subchannel analysis
6. Identify cases for radiological analysis (if applicable)
   - Maximum mass release case
   - Maixmum iodine spiking case

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Non-LOCA Methodology

General Methodology (Section 7.1):

- Steady-state conditions
- Treatment of plant controls
- Loss of power
- Single failure
- Bounding reactivity parameter input
- Biasing of other parameters: initial conditions, valve characteristics, analytical limits and response times
- Operator action

Event-specific Methodology (Section 7.2)

- Description of event initiation and progression
- Acceptance criteria 'of interest'
- Limiting single failure, loss of power scenarios, or need for sensitivity calculations
- Initial condition biases and conservatisms, or need for sensitivity calculations
- Tabulated representative results of sensitivity calculations

Example analysis results provided in Section 8

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Conclusions

- Non-LOCA system transient evaluation model developed following a graded approach in accordance with guidance provided in RG 1.203
- Applies to NPM-type plant design natural circulation water reactor with helical coil SG and integral pressurizer
- NRELAP5 used to simulate the system thermal-hydraulic response
  - Demonstrate primary and secondary pressure acceptance criteria are met
  - Demonstrate safe, stabilized condition achieved
- System transient results provide boundary conditions to downstream subchannel and radiological analyses

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Acronyms

- AOO – Anticipated Operational Occurrences
- CNV – Containment Vessel
- CVCS – Chemical and Volume Control System
- DHRS – Decay Heat Removal System
- ECCS – Emergency Core Cooling System
- EM – Evaluation Model
- EMDAP – Evaluation Model Development and Assessment Process
- FW – Feedwater
- FWIV – Feedwater Isolation Valve
- IET – Integral Effects Test
- KAIST – Korea Advanced Institute of Science and Technology
- LOCA – Loss of Coolant Accident
- MCHFR – Minimum Critical Heat Flux Ratio
- MPS – Module Protection System
- MSIV – Main Steam Isolation Valve
- MSS – Main Steam System
- NIST-1 – NuScale Integral System Test-1
- NPM – NuScale Power Module
- PIRT – Phenomena Identification and Ranking Table
- RCS – Reactor Coolant System
- RPV – Reactor Pressure Vessel
- RSV – Reactor Safety Valve
- RVV – Reactor Vent Valve
- SET – Separate Effects Test
- SG – Steam Generator
- SSC – Structures, Systems, and Components

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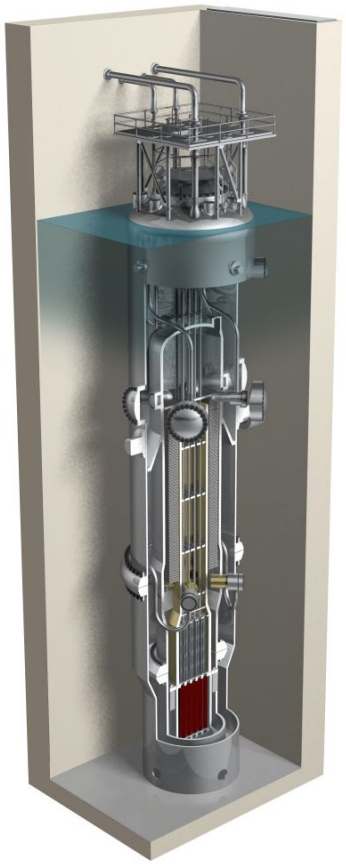

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Additional Material for Public Presentation

Previously presented background material

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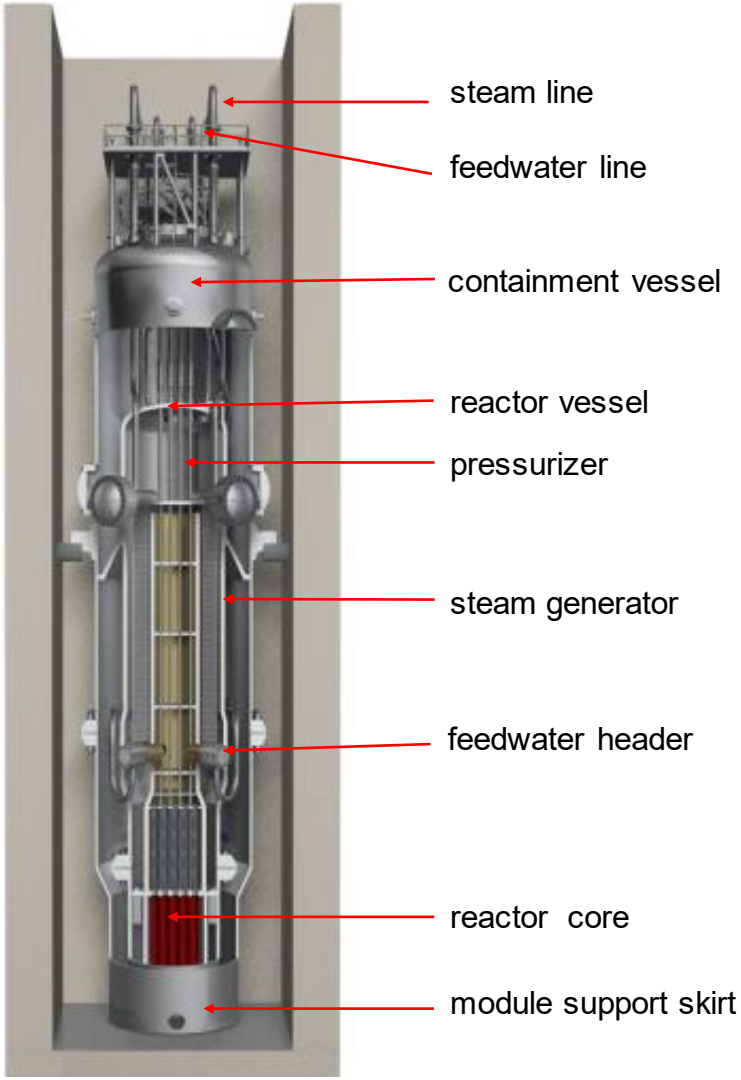
Power Module Overview

Integral Pressurized Water Reactor

- Core, steam generator and pressurizer in one vessel
- Integrated reactor design, no large-break loss-of-coolant accidents
- Reactor coolant system operated in single phase (liquid) density driven flow
- Safety decay heat removal systems are passive and fail safe
- Module protection system designed to automate event mitigation actuations (no operator actions)



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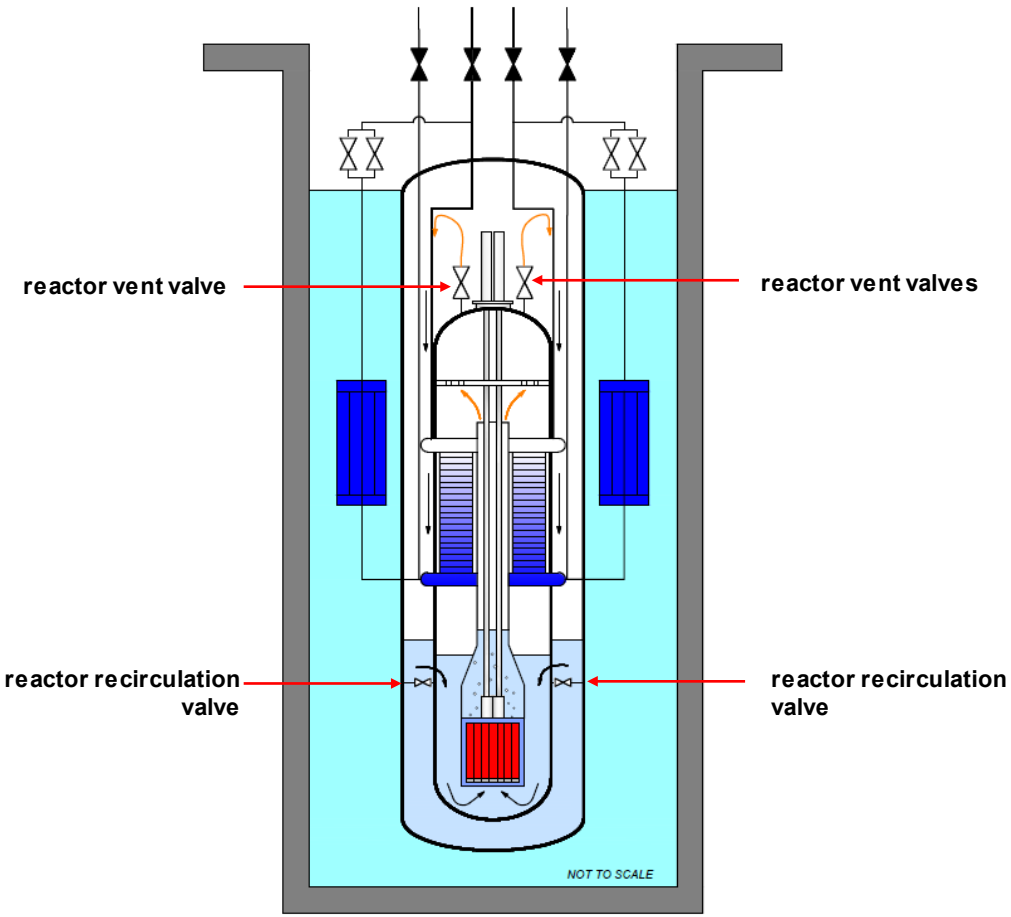
ECCS

Emergency Core Cooling System

- ECCS valves open to a boiling/condensing circulation flow path to transfer decay and residual heat to reactor pool
  - Liquid from containment vessel enters RCS through reactor recirculation valves
  - Vapor vented from RCS to containment vessel through reactor vent valves
  - Steam condenses on inside surface of containment vessel
  - Heat transfer through vessel walls to the reactor pool
- Actuation Signals: High CNV level, 24hr loss of AC power
- Fail safe: ECCS valves open on loss of DC power



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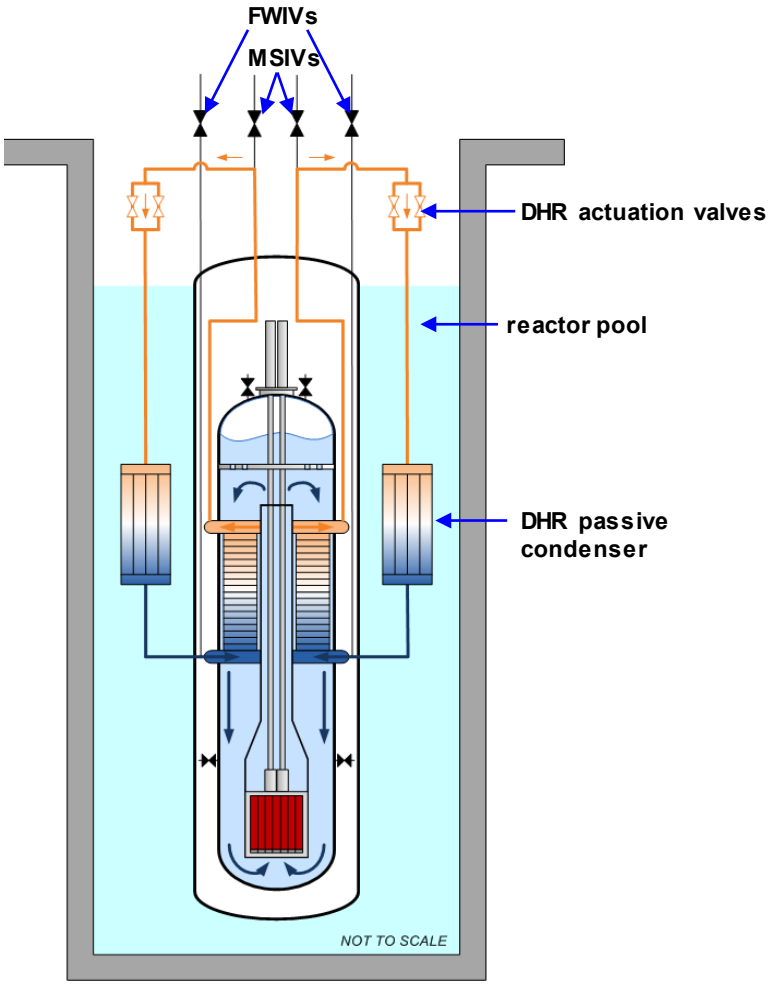
Decay Heat Removal System (DHRS)

- Removes heat after loss of normal cooling
- Boiling/condensing loop
- Two redundant trains
- Redundant actuation and isolation valves for each train
- Initiates on:
  - Loss of power
  - Loss of cooling indication (ESFAS Signal)



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Deterministic Event Mitigation

Module Protection Functions

Reactivity Control

- Reactor trip
- CVCS/Demineralized Water Isolation

RCS and Secondary Inventory Control

- Containment Isolation
- Secondary Isolation

Heat Removal

- DHRS Actuation
- ECCS Actuation

Subcooling

- Reactor trip

Event Mitigation

Increase in heat removal transients

- Reactor trip Secondary Isolation

Decrease in heat removal transients

- Reactor trip DHRS Actuation

Reactivity and power transients

- Reactor trip Demineralized Water Isolation

Increase in RCS inventory transients

- Reactor trip CVCS Isolation

Decrease in RCS inventory transients

- Reactor trip CNV Isolation
  ECCS actuation

Stability

- Reactor trip

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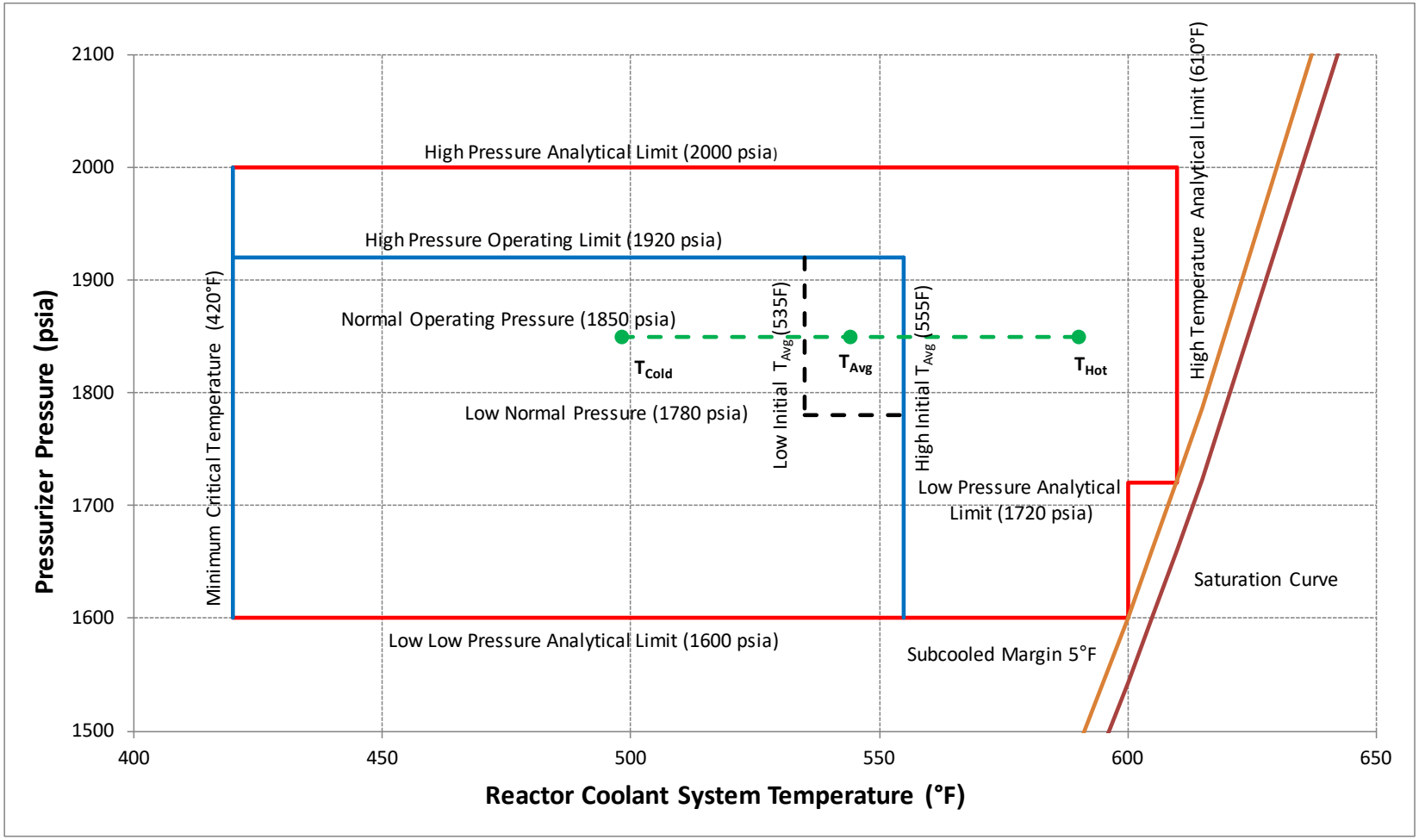
Pressure vs. Temperature Operation Map



- Module protection system (Ch. 7, red)
- Technical specification LCOs (Ch. 16, blue)

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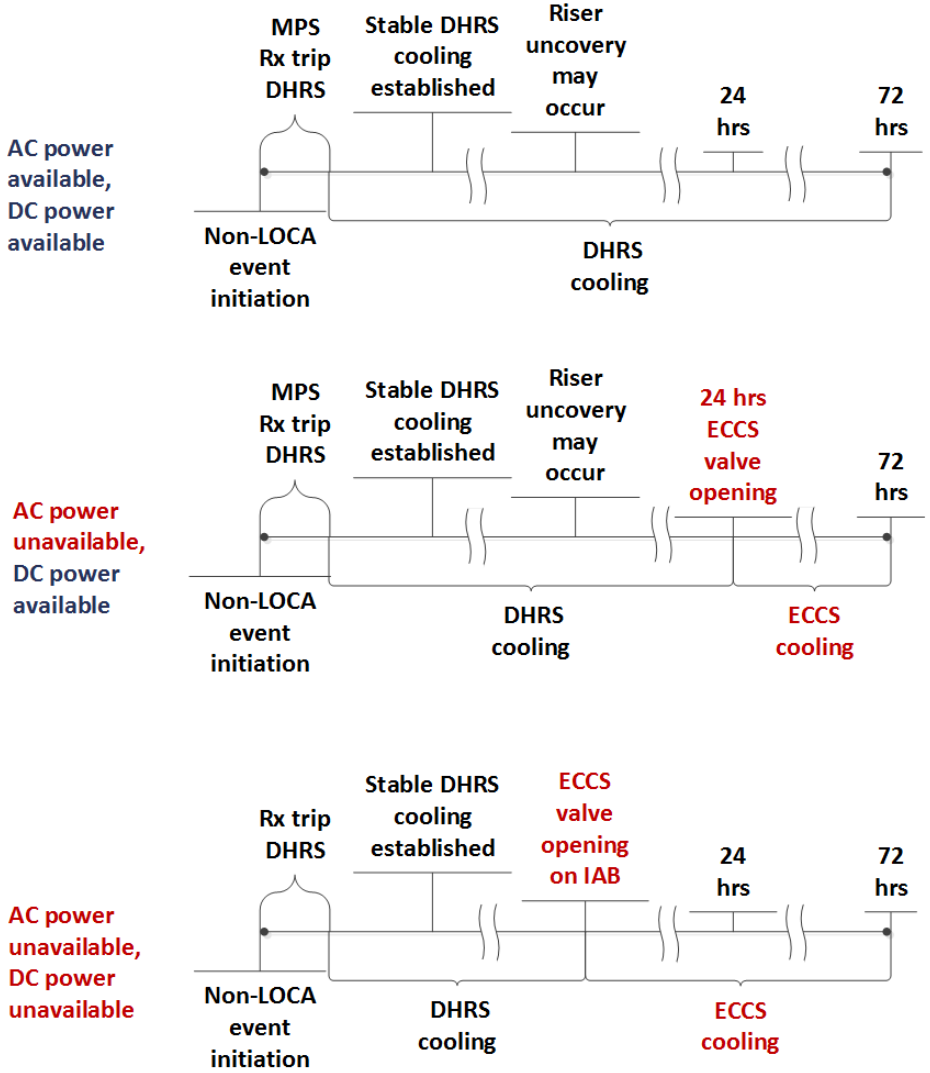
Loss of Power – Non-LOCA Event



- Availability of AC, DC power affects whether ECCS valves actuate, and what time they open

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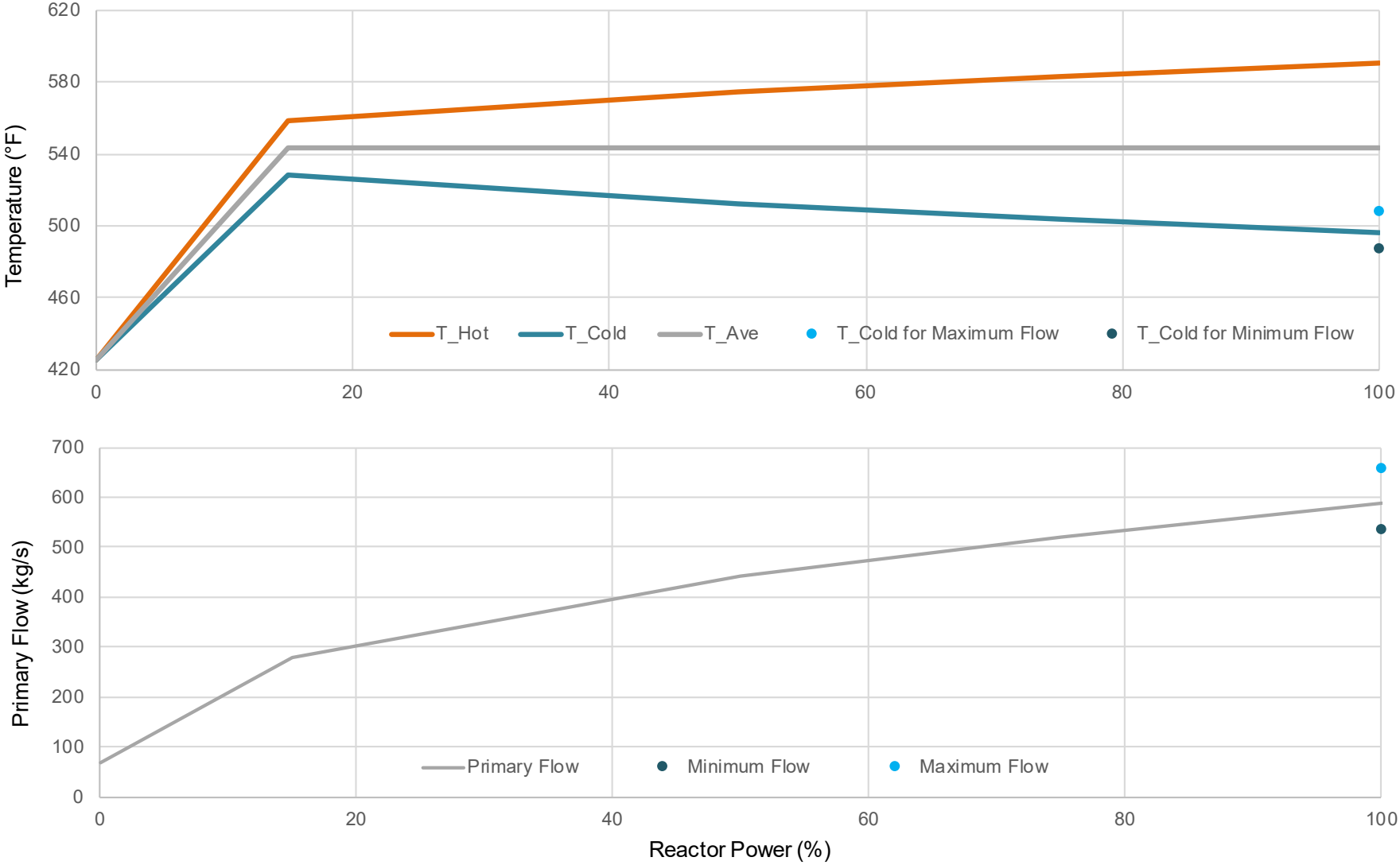
5.1 – Summary Description



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Presentation to the ACRS Subcommittee

Staff Review of NuScale Topical Report

TR-0716-50350, REVISION 1

"Rod Ejection Accident Methodology"

Presenters:

Chris Van Wert – Senior Reactor Systems Engineer, Office of New Reactors

Jeff Schmidt– Senior Reactor Systems Engineer, Office of New Reactors

February 19, 2020

(Open Session)

NRC Technical Review Areas/Contributors

- NUCLEAR METHODS, SYSTEMS & NEW REACTORS BRANCH / NRR:
  Rebecca Patton (BC)

- ADVANCED REACTOR TECHNICAL BRANCH / NRR:
  Jeff Schmidt
  Chris Van Wert

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Staff Review Timeline

TR-0716-50350, "ROD EJECTION ACCIDENT METHODOLOGY"

- NuScale submitted Topical Report (TR)-0716-50350, "Rod Ejection Accident Methodology," Revision 1, on November 15, 2019, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19319C684).
- Staff plans to brief advisory committee on reactor safeguards (ACRS) full committee on March 5, 2020.
- Staff plans to issue its final SER in March 2020.
- Staff plans to publish the "-A" (approved) version of the TR prior to finishing Phase 6 of the NuScale DCA.

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Staff Review

- The staff's review included:
  - Evaluation of the analysis criteria
  - Evaluation of the code suite used within the analysis methodology
  - Evaluation of the plant and cycle assumptions used in the analysis methodology
  - Evaluation of the rod ejection accident analysis methodology

- The staff's review does *not* include the licensing basis RIA analysis for the NuScale DCA.
  - Contained in Section 15.4.8 of the SER for the NuScale Design Certification

- During its review, staff audited calculations and other supporting information

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Analysis Criteria

- Reactor Coolant System Pressure
  - Maximum RCS pressure ≤ 120% of the design pressure
- Fuel Cladding Failure
  - High temperature at zero-power conditions:
    - Set to 100 cal/g for <u>all</u> cladding differential pressures
  - For intermediate and full-power conditions:
    - Failure assumed if local heat flux exceeds CHF limits
  - PCMI failure threshold follows Figure B-1 of SRP Section 4.2 Appendix B
  - Failure is assumed if fuel temperature anywhere in the pellet exceeds incipient fuel melting conditions

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Analysis Criteria (cont)

- Core Coolability
  - Peak radial average fuel enthalpy will remain below 230 cal/g
  - Fuel melt is not allowed
- Fission Product
  - NuScale does not rely on SRP guidance regarding fission product inventory. Instead, the methodology precludes this need by:
    - Requiring that no failure occurs due to fuel melt, enthalpy increase, or exceeding MCHFR limits
    - Incorporating the most limiting criteria which SRP 4.2 Appendix A associated with the largest cladding differential pressure, therefore any increase in pressure due to FGR will not change allowed peak radial average fuel enthalpy
- Staff reviewed criteria and found them acceptable relative to the guidance in SRP Section 4.2, Appendix B.
- Staff also noted that DG-1327, "Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents" is currently being developed.

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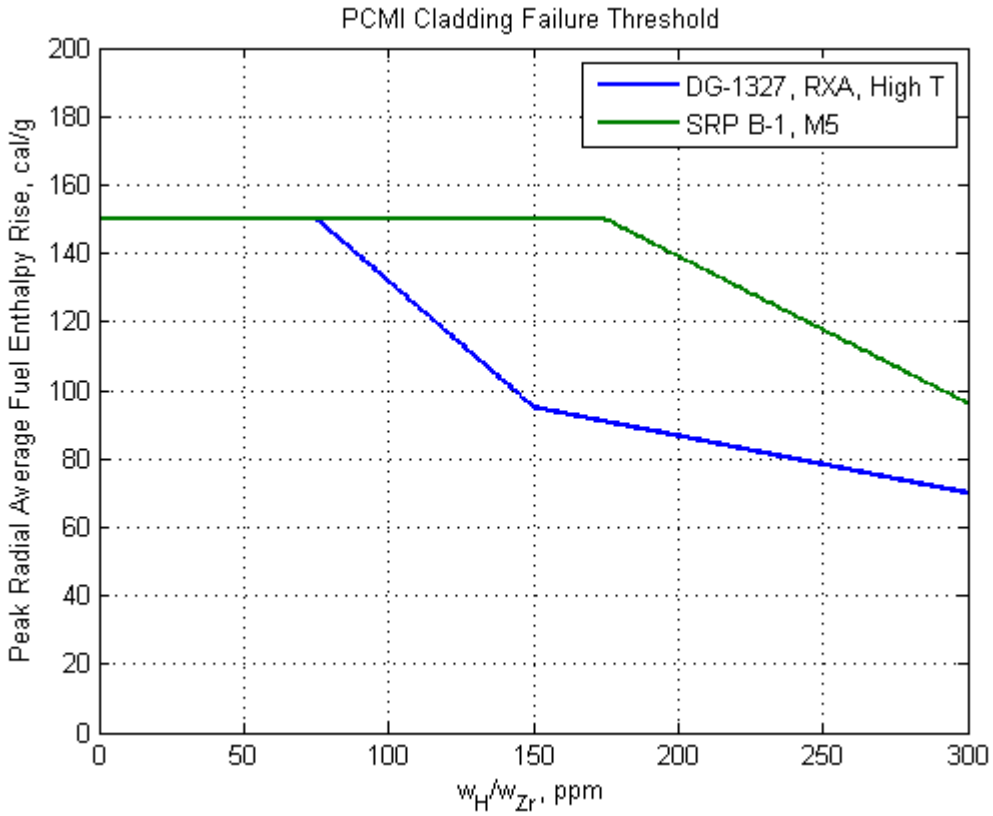
Comparison of Current versus Draft Regulatory Guidance



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Evaluation of Code Suite

- The NuScale REA analysis is based on the following codes and packages:
  - CASMO5/SIMULATE5: provides reactor core physics parameters
  - SIMULATE-3K: 3-dimensional nodal reactor kinetics code which supplies power input to downstream analyses
  - NRELAP5: transient system response
  - VIPRE-01: subchannel analysis

- Applicability of CASMO5, SIMULATE5, NRELAP5, and VIPRE-01 has been reviewed and approved for NuScale in TR-0616-48793-P-A, Revision 1, "Nuclear Analysis Codes and Methods Qualification".

- The validation of SIMULATE-3K is included as part of TR-0716-50350 and is therefore included in the staff's review.
  - Staff concluded that NuScale successfully validated S3K against experimental data and the NEACRP control rod ejection problem computational benchmark

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Plant and Cycle Assumptions

- The staff reviewed the plant and cycle assumptions used in the NuScale rod ejection analysis methodology
  - The staff determined that the methodology included ranges in power, time in cycle, and core power that covered a wide range of operating conditions and would capture the most limiting condition
  - The staff agreed that the assumptions associated with the automatic system response of non-safety systems were conservative
  - The staff determined that the methodology regarding the timing of loss of AC power conservatively biases the RCS pressure evaluation

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Rod Ejection Accident Analysis Methodology

- The staff reviewed the analysis methodology including steady-state initialization, dynamic core response, dynamic system response, subchannel critical heat flux evaluation, and the adiabatic heatup fuel response
- The staff's review included the methodology by which information is passed between codes, application of uncertainties, modelling assumptions used for inputs, and handling of reactor trips.
- The staff concluded that the methodology for calculating the system response, subchannel, and fuel response analyses was conservative and acceptable for demonstrating compliance with the acceptance criteria

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Staff SER Conclusions

- The staff concludes that the NuScale criteria used for evaluating REA either follows or is more conservative than staff guidance

- The staff concludes that the methodology accounts for the various potential operating conditions and time in life, and conservatively addresses uncertainties and plant conditions

- The staff finds the use of TR-0716-50350-P acceptable for evaluating reactivity initiated accidents for the NuScale plant design.

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Questions?

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Review of TR-0516-49416, "Non-Loss-of-Coolant Accident Analysis Methodology"

NuScale Design Certification Application

ACRS Subcommittee Meeting
February 19, 2020

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NRC Staff Review Team

- NRC Technical Reviewers:
  - Antonio Barrett, NRR
  - Jeff Schmidt, NRR
  - Alex Siwy, NRR
  - Ray Skarda, RES
  - Peter Lien, RES
  - Ron Harrington, RES
  - Jason Thompson, RES
- Consultants (Energy Research, Inc.):
  - Mohsen Khatib-Rahbar
  - Walter Tauche (subcontractor)
  - Morgan Libby
  - Michael Zavisca

Review Process Overview

- SER is based on TR-0516-49416, Revision 2
- Two audits conducted in four phases
  - About 140 audit issues
  - Helped to confirm staff's understanding and inform RAIs
- 33 RAI questions issued
  - All resolved and responses incorporated into TR-0516-49416, Revision 2, as appropriate

Regulatory Requirements and Guidance

- Regulatory requirements include several GDC (5, 10, 13, 15, 17, 20, 25, 26, 27, 28, 31, 34)
- Guidance Documents
  - SRP Section 15.0.2, “Review of Transient and Accident Analysis Methods”
  - RG 1.203, “Transient and Accident Analysis Methods”
  - DSRS Section 15.0,“Introduction – Transient and Accident Analyses”
  - Event-specific SRP/DSRS sections

DSRS Section 15.0 Acceptance Criteria

- Anticipated Operational Occurrences
  - Primary and secondary pressures < 110% of design values
  - Minimum DNBR > 95/95 limit
  - Should not generate a postulated accident without other faults occurring independently or result in a consequential loss of function of the RCS
- Postulated Accidents/Infrequent Events
  - Primary and secondary pressures < acceptable design limits, considering potential brittle and ductile failures
  - Fuel assumed to be failed if minimum DNBR ≤ 95/95 limit
  - Offsite doses ≤ guidelines of 10 CFR Parts 52.47(a)(2)(iv) and 100 (for infrequent events, 10 percent of 10 CFR 52.47(a) and Part 100)
  - Shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including RCS and containment

Non-LOCA Methodology Scope

- Provides a methodology for performing system transient analysis of specified non-LOCA design-basis events for the NPM
- Directly evaluates primary and secondary pressure figures of merit
- Provides input to downstream analyses that evaluate margin to SAFDLs and radiological consequences
- Covers time frame during which mixture level is above top of riser and natural circulation is maintained
- Includes certain event-specific assumptions and conservative bias directions for initial conditions

Non-LOCA Methodology Analyzed Events

Increase in heat removal by the secondary system

- Decrease in feedwater temperature
- Increase in feedwater flow
- Increase in steam flow
- Inadvertent opening of SG relief or safety valve
- Steam system piping failure inside or outside containment
- Containment flooding/loss of containment vacuum
- Inadvertent DHRS actuation

Decrease in heat removal by the secondary system

- Loss of external load
- Turbine trip
- Loss of condenser vacuum
- Main steam isolation valve closure
- Loss of nonemergency AC power to station auxiliaries
- Loss of normal feedwater flow
- Inadvertent DHRS actuation
- Feedwater system pipe break inside or outside of containment

Reactivity and power distribution anomalies

- Uncontrolled control rod assembly bank withdrawal from a subcritical or low power startup condition
- Uncontrolled control rod assembly bank withdrawal at power
- Control rod misoperation (single rod withdrawal, single rod or bank drop)
- Inadvertent decrease in boron concentration in the reactor coolant

Increase in reactor coolant inventory

- Chemical and volume control system malfunction that increases reactor coolant system inventory

Decrease in reactor coolant inventory

- Failure of small lines carrying primary coolant outside containment
- Steam generator tube failure

Non-Review Scope Items

Several items described in the Non-LOCA TR are outside the scope of the staff TR review and within the scope of a design-specific application of the methodology:

- Representative calculation results
- Example values of parameters or parameter biases
- Limiting loss of power and single failure scenarios
- Need for operator actions
- Credit for secondary MSIVs that are not safety related

Non-LOCA Transient Analysis Process

Upstream Methodologies
- Nuclear Design
- Fuel Design

↓

Non-LOCA EM (NRELAP5)

↓

Downstream Methodologies
- Subchannel (VIPRE-01)
- Accident Source Term

- Adapt NRELAP5 Basemodel for Specific Event Analysis
- Perform NRELAP5 Steady-State Initialization Calculations
- Perform NRELAP5 Transient Calculations
- Evaluate Results of Transient Calculations against Acceptance Criteria
- Identify Cases for Downstream Analyses

Key Design Features for Non-LOCA

- Staff focused its review on several key features of the NuScale Design:
  - Natural circulation design
  - Helical coil SGs
    - Transfer heat from RCS to feedwater
  - Passive DHRS condensers
    - Transfer decay heat to reactor pool using the SGs
  - Evacuated containment vessel

Applicability of NRELAP5 to Non-LOCA Analysis

- Non-LOCA EM developed from the LOCA EM using graded approach described in RG 1.203
- Applicant's non-LOCA PIRT identified highly ranked phenomena
- Applicant identified deltas in highly ranked phenomena relative to LOCA and addressed them using:
  - Separate effects tests: NIST HP-03, HP-04
    - Staff also evaluated KAIST and SIET assessments
  - Integral effects tests: NIST NLT-02a, NLT-02b, NLT-15p2
  - Code-to-code benchmark against RETRAN-3D for reactivity-initiated events
  - Use of bounding input values
  - Other analysis methodologies (e.g., subchannel)

Significant Issue – Multi-Dimensional Flow Effects

- Staff requested additional justification for how multi-dimensional flow effects in the RCS and thermal stratification in the reactor pool are addressed
- RAI response, as confirmed by audit and multiple audit discussions, resolved the issue
- More details in closed session

NRELAP5 Assessments: KAIST, NIST-1 HP-03, and HP-04

- Phenomena addressed: condensation inside and heat transfer across DHRS tubes to CPV fluid
- Staff confirmed that the ranges of the KAIST and NIST-1 validations adequately cover expected ranges during DHRS operation in the NPM
- Staff finds the facility nodalization acceptable relative to NPM nodalization
- Staff finds that the NRELAP5 prediction of KAIST and HP-03 data is reasonable to excellent, and the prediction of HP-04 data is reasonable

Significant Issue - Removal of DHRS Heat Transfer Bias

- The applicant performed extensive sensitivity studies that showed an insignificant effect of DHRS performance and nodalization on non-LOCA figures of merit
- Underlying reasons:
  - Minimum margin to acceptance criteria typically occurs before DHRS is fully active
  - Peak RCS pressure limited by RSV
  - Peak secondary pressure a function of secondary inventory and primary side conditions – insignificantly affected by DHRS heat transfer variations
- The staff finds that the sensitivity studies and physical dynamics support removal of a DHRS heat transfer bias for NPM model Revision 2

NRELAP5 Assessment: SIET TF-1 and TF-2

- TF-1 assessed flow inside SG tubes
- TF-2 assessed primary-to-secondary heat transfer
- Staff finds reasonable to excellent agreement between NRELAP5 predictions and SIET test data
- Staff raised concerns about the range of conditions validated by TF-2 compared to expected ranges in the NPM and associated uncertainties in SG heat transfer

Significant Issue - Removal of SG Heat Transfer Bias

- The applicant performed comprehensive sensitivity studies that showed an insignificant effect of SG heat transfer on non-LOCA figures of merit
- Staff audited the sensitivity calculations
- The staff finds that these sensitivity studies, in addition to post-trip and normal operation considerations, support removal of SG heat transfer bias for NPM model Revision 2
- More details in closed session

NRELAP5 Assessments: NIST-1 Integral Effects Tests

- Staff audited detailed assessments
- NLT-02a, loss of feedwater transient to the point of a reactor trip
  - Reasonable to excellent code-to-data agreement
- NLT-02b, DHRS-driven cooling following initial DHRS actuation
  - Reasonable or better agreement
- NLT-15p2, loss of feedwater transient and subsequent DHRS cooling
  - Test assessed by NuScale to address staff questions
  - Reasonable agreement
- The staff finds that these assessments together demonstrate applicability of NRELAP5 to evaluate non-LOCA transients

NRELAP5 Assessments: Benchmark against RETRAN-3D

- Primary purpose: assess performance of NRELAP5 point kinetics model
- Applicant made appropriate modeling simplifications since RETRAN-3D does not account for helical coil SG
- Staff reviewed applicant's comparison of four reactivity transients:
  - Slow uncontrolled control rod withdrawal
  - Fast uncontrolled control rod withdrawal
  - Power reduction from 100 to 50 percent
  - Dropped control rod assembly
- Staff finds that there is reasonable or better agreement for these transients, providing confidence in NRELAP5 point kinetics model

General Non-LOCA Methodology

- Use of “suitably conservative” input in lieu of uncertainty evaluation, e.g.:
  - Bounding reactivity parameters
  - Biased initial conditions
  - Valve characteristics
- Plant control systems operation, single failure, and loss of power assumptions should maximize consequences with regard to acceptance criteria of interest
- If acceptance criteria are insensitive to certain inputs/assumptions, nominal values may be used
- If an acceptance criterion is not challenged by a particular event, it will not be subjected to in-depth sensitivity studies
- The staff finds the general approach acceptable subject to applicable Limitations and Conditions

Event-Specific Non-LOCA Methodology

- The staff evaluated each event-specific methodology, including:
  - Methodology aspects specific to the initiating event
  - General assessment of challenge to the acceptance criteria
  - Initial conditions, biases, and conservatisms
    - Always acceptable for a parameter to be varied as part of a plant-specific calculation
    - Staff ensured that bias directions or conservatisms specified are appropriately conservative with respect to effect on acceptance criteria
- The staff finds that the event-specific methodologies will ensure conservative results when implemented

Representative Calculations

- Staff reviewed the representative calculations insofar as they demonstrate application of the non-LOCA methodology
- Staff concludes that the calculations show how the non-LOCA methodology can be used for conservative transient analysis simulations
- Final calculations for the NuScale design are reviewed in the SER for DCA Chapter 15

Staff SER Limitation and Condition Summary

1. Future changes to LOCA TR must be assessed for impacts to Non-LOCA EM
2. Non-LOCA EM scope limited to non-LOCA events defined in the TR prior to the time of riser uncovery for evaluation of primary and secondary pressures and potential for loss of system functionality
3. Additional justification must be provided for elimination of SG and DHRS heat transfer biases if applying methodology to a design other than NPM model Revision 2 or a model update made pursuant to a change process specifically approved by NRC for changes to the NPM model
4. Any credit for secondary MSIVs (not safety-related) must be approved through design review
5. Event-specific electrical power assumptions, single failures, and operator actions must be approved through design review
6. Non-LOCA EM use limited to NRELAP5 v1.4 and NPM model Revision 2, unless changes are made pursuant to a change process specifically approved by the NRC staff for changes to NRELAP5 and the NPM model

Conclusions

- All technical issues from the course of the review have been resolved
- Use of NRELAP5 with the non-LOCA methodology described in the TR is acceptable for the non-LOCA safety analyses of the NuScale NPM design subject to the specified limitations and conditions

Backup Slides

NuScale DCA GTS 3.4.1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Resistance Critical Heat Flux (CHF) Limits

LCO 3.4.1 Each RCS CHF parameter shall be within the limits specified in the COLR:

a. Pressurizer pressure,

b. RCS cold temperature, and

c. RCS flow resistance.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. RCS pressurizer pressure or RCS cold temperature CHF parameters not within limits.	A.1	Restore RCS CHF parameter(s) to within limit.	2 hours
B. RCS flow resistance not within limits.	B.1	Evaluate flow resistance effect on safety analysis and verify that the reactor coolant system flow rate is acceptable for continued operation.	7 days
C. Required Action and associated Completion Time not met.	C.1	Be in Mode 2.	6 hours

NuScale DCA GTS 3.4.1 (2)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is greater than or equal to the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.2	Verify RCS cold temperature is less than or equal to the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.3	----------NOTE---------- Not required to be performed until 96 hours after exceeding 50% RTP. ---------- Verify RCS flow resistance is within the limits specified in the COLR.	 Once prior to exceeding 75% RTP after each refueling

United States Nuclear Regulatory Commission

Protecting People and the Environment

Loss-of-Coolant Accident (LOCA) Evaluation Model Topical Report Review (TR-0516-49422-P)

NuScale Design Certification Application

ACRS Subcommittee Meeting
February 19, 2020

Review Team

- **NRC**

Mr. Carl Thurston
Dr. Peter Lien
Dr. Weidong Wang
Mr. Ron Harrington
Dr. Shanlai Lu
Mr. Antonio Barrett
Dr. Tim Drzewiecki
Dr. Syed Haider

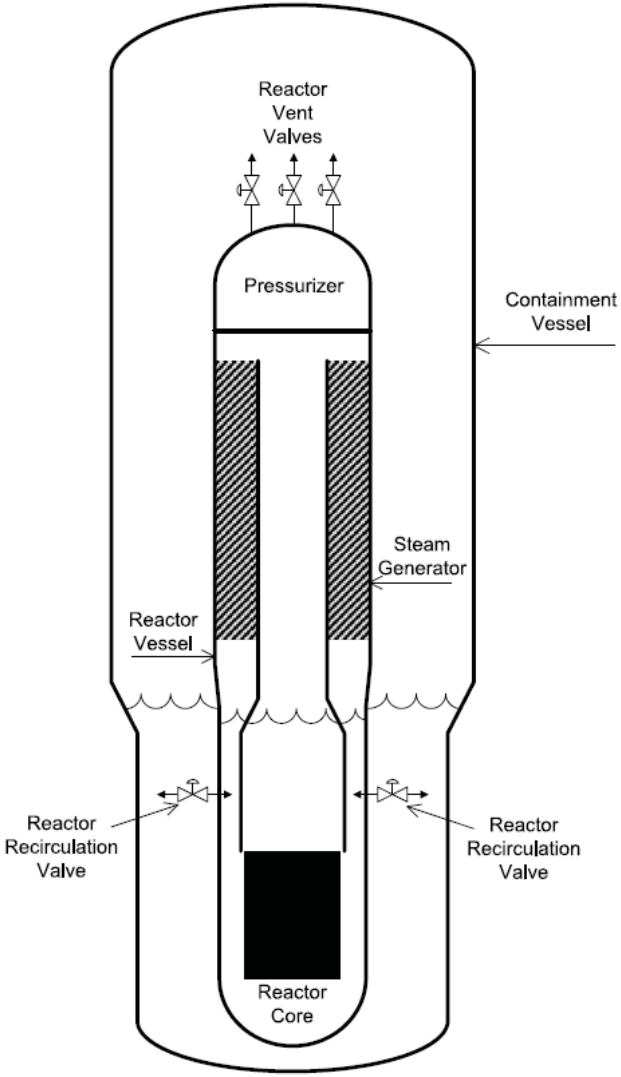
- **NuMark Associates**

Mr. Marvin Smith
Dr. Leonard Ward
Dr. Donald Rowe
Mr. Bert Dunn

- **Brook Haven National Lab**

Dr. Upendra Rohatgi

Design Features And Scope



3 Reactor Vent Valves

2 Reactor Recirculation Valves

Each ECCS valve has its own IAB, trip valve and trip reset valve

Containment functions as part of ECCS

Topical Report Application Scope

- Provide a methodology to analyze NuScale LOCA scenarios
- Provide a methodology to analyze the inadvertent opening of RPV Valves event
- Support other methodologies:

  Peak containment pressure analysis

  Non-LOCA analysis

  Long-Term Cooling analysis

Review Approaches

- Early Engagement And Extensive Audits Through Electronic Reading Room
  - Pre-application engagement
  - Initial on-site visits and audit meetings
  - Two phases of continuing audits throughout review period
- Issues Raised:
  - 45 RAI Questions
  - 210 Audit Issues
- Staff performed sensitivity analysis with NRELAP5 and confirmatory analysis with TRACE
- Major Issues Summarized on Later Slides

Regulatory Basis and Acceptance Criteria

- Regulatory Basis For LOCA Analysis: 10CFR50.46 Appendix K
  - Peak Cladding Temperature < 2,200 ºF
  - Local maximum fuel cladding oxidation < 17%
  - Whole core fuel cladding oxidation < 1%
- Phenomenon Excluded:
  - Post CHF heat transfer, Reflood & Refill, Pump Model
  - With this NPM design, LOCA Peak Clad Temperature (PCT) is the fuel temperature during normal plant operation and PCT is not a FOM
- The FOMs for the NuScale NPM during a LOCA are:
  - Maintaining containment integrity to provide water flow back to the RPV after ECCS valve operation and pressure equalization between the PRV and CNV
  - Maintaining the fuel heat flux below the critical heat flux ratio (CHFR) to prevent significant clad heat-up
  - Keeping the RPV collapsed liquid level (CLL) above the core is to demonstrate that there is an adequate supply of liquid water available to the core

Regulatory Basis and Acceptance Criteria

- Regulatory Basis For Inadvertent Opening of RPV Valve Event: GDC 10
  - NuScale categorized as Anticipated Operational Occurrence (AOO)
  - GDC 10 requires protection of Specified Acceptable Fuel Design Limits (SAFDLs)
- The FOM for the NuScale NPM during an IORV event is:
  - Maintain margin to critical heat flux: CHFR > CHFR-limit

Phenomena Identification and Ranking Table

- NuScale engaged expert panels to develop a Phenomena Identification and Ranking Table (PIRT) to identify important phenomena and processes in the NPM during a LOCA
- The PIRT panels identified twenty-one phenomena as important to capture in the NuScale LOCA Evaluation Model
- NuScale then performed NRELAP5 code and model validation to show that their NRELAP5 code and LOCA EM is able to evaluate these important phenomena and processes
- The NuScale validation suite included legacy separate effects tests (SETs) and integral effects tests (IETs), and SETs and IETs developed and run specifically for the NuScale NPM design.

Phenomena Identification and Ranking Table

- Staff identified the following significant issues:
  - RRV flange not considered a break location
    - Resolved via DCA Chapter 3 review – RVV and RRV flanges considered break exclusion zones
  - Not clear whether all the phenomena were identified for the lowest discharge point of the system (RVV or RRV opening).
    - NuScale performed additional NIST tests with RRV opening
    - Confirmed that the PIRT is appropriate for IORV discharge

NRELAP5 Code and Models

- NRELAP5 was derived from the Idaho National Laboratory (INL) RELAP5-3D computer code
- Models and correlations used by NRELAP5 (V 4.1.3) were reviewed and modified as required for use within the NuScale LOCA EM
  - ❑ Helical coil S. Generator heat transfer and pressure drop
  - ❑ Core CHF, MCHFR limit, and run termination (mCHFR<1)
  - ❑ Steam condensation heat transfer { }
  - ❑ Critical flow break discharge { }
  - ❑ Interfacial drag for large diameter pipes
- The development of NPM model and nodalization were reviewed resulting in model update

NIST Tests And Scaling Analysis

- NIST-1 scaled facility test used in validation of NRELAP5 code
- Main components included RPV, helical coil SG system with DHRS, CNV, and CPV representing the reactor pool (separated vessels)
- NIST-1 ECCS connects the RPV to the CNV and consists of two RVVs and two RRVs, each on separate lines
- The scaling analysis was developed and reviewed.
  - A RAI was issued to question the validity of all half power test results.
  - A new limiting LOCA test was conducted - Full decay heat power test (HP-06B)

NRC Sensitivity & Confirmatory Analyses

- Separate Effect Tests (SETs):
  - KAIST model: DHRS tube condensation experiment, non-LOCA
  - SIET model: helical coil steam generator tube/shell side heat transfer, non-LOCA
  - NIST-1 model: high pressure condensation test (HP-02)
- Integral Effect Tests (IETs):
  - NIST-1 models: loss of coolant accident (LOCA) and inadvertent emergency core cooling system (ECCS) operations

  NPM models: licensing calculation confirmation and sensitivity studies, LOCA, non-LOCA
- Both TRACE and NRELAP5 codes were used. More than fifty five sets of calculations were performed. RAIs were issued and NRELAP5 code was updated from V1.3 to V1.4. Good agreements were obtained with NuScale analysis results

CHF Correlations For LOCA and IORV

CHF Models:

- For LOCA and LOCA-like events (Loss of RCS Coolant/IORV) a combination of the { } and { } CHF correlations are used to calculate CHF.
  - { } is used for high-flow conditions
  - { } is used for low-flow conditions

- STERN and KATHY test facilities provide CHF databases which are used to assess the NRELAP5 models ability to predict CHF.
  - Stern CHF database is used to determine CHFR limit for low-flow conditions for LOCA and LOCA-like events
    - Preliminary Prototype Fuel
  - KATHY CHF database is used to determine CHFR limit for high-flow conditions for LOCA and LOCA-like events
    - Prototypical fuel

Approval Limitations & Conditions

With all the issues resolved, the staff approved this topical report with the following limitations and conditions:

- Exemption required for an applicant or licensee to exclude required features of Appendix K
- NuScale EM is limited to the evaluation of LOCAs where CHF is not exceeded, the collapsed water level above the core and containment temperature and pressure remain below the design limits
- No credit for DHRS heat removal unless it is further reviewed and approved
- Use of the LOCA EM is limited to evaluations of the analyses for the FOMs described in the TR: the short term LOCA or an IORV event
- Limited to NRELAP5 v1.4 and NPM model Revision 2, unless changes are made pursuant to a change process specifically approved by the NRC staff for changes to NRELAP5 and the NPM model
- Single Failures, Electrical Power Assumptions (ac/dc) and Need for Operator Actions Not Approved Within this Methodology – (done in design review)
- Several CHF C/Ls (covered in closed session slides)

Conclusions

- NuScale LOCA EM and NRELAP5 V1.4 are approved for determining critical heat flux and collapsed liquid level for NuScale reactor in compliance with 10CFR 50.46 Appendix K requirements
- NRELAP5 computer code V1.4 is also determined applicable to predict containment pressure and temperature subject to specific modeling requirements
- The CHF modeling is approved subject to limitations and conditions