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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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NuSCALE SUBCOMMITTEE

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OPEN SESSION

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MONDAY

MARCH 2, 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, Chair, presiding.

COMMITTEE MEMBERS:

WALTER L. KIRCHNER, Chair

RONALD G. BALLINGER, Member

DENNIS BLEY, Member

CHARLES H. BROWN, JR., Member

VESNA B. DIMITRIJEVIC, Member
JOSE MARCH-LEUBA, 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:
MIKE SNODDERLY

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

P R O C E E D I N G S

(8:29 a.m.)

CHAIR KIRCHNER: Good morning. The meeting will come to order. This is a meeting of the Advisory Committee on Reactor Safeguards, NuScale Subcommittee.

I am Walt Kirchner, the Chairman of the NuScale Subcommittee. Members in attendance today are Ron Ballinger, David Petti, Joy Rempe, Matt Sunseri, Jose March-Leuba, Charles Brown will be joining us later -- Dennis Bley, and Vesna Dimitrijevic.

That was good.

We are also joined by Pete Riccardella. And we have two consultants with us today, Michael Corradini, our former committee chair, and Stephen Schultz.

Mike Snodderly is the designated federal official for this meeting. The Subcommittee will conduct an area of focus review on the NuScale emergency core cooling system and valve performance, boron dilution and return to criticality, and probabilistic risk assessment, among other matters.

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 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 under Federal Register on June 13th, 2019.

The ACRS section of the US 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 US Code 552bc4.

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 the Subcommittee may do so and should contact the designated federal official five days prior to the meeting as practicable.

We have 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 US 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 identify themselves and speak with enough volume and clarity so that they can be readily heard.

Telephone bridge line has been established for the public to listen to the meeting. To minimize disturbance, the public line will be kept in listen-in only mode. To avoid further disturbances, I request that attendees put their electronic devices like cell phones in the off or noise-free mode.

We'll now proceed with the meeting. And let me say this is a three-day meeting. So when we finish this afternoon, we'll start with Chapter 15 today, we'll recess, and then you will not have to hear me read this again tomorrow morning or Wednesday. Thank you.

And with that, from the staff, Rebecca, any comments?

MS. PATTON: Just wanted to thank Committee for its time. There's obviously a lot of, you know, fairly complicated topics that we're going to be covering over the next couple of days, so we hope to have a very productive discussion on how we've resolved these matters and where we are today. Thank you.

CHAIR KIRCHNER: Thank you.

So with that, we'll turn to NuScale and Matthew Presson.

Go ahead, Matthew.

MR. PRESSON: Thank you and good morning. So the scope of this morning's meeting will be discussing the revisions to the NuScale DCA FSAR, Chapter 15, since revision 2. And we will also be presenting on some of the remaining focus areas centered on safety analysis.

We'll have a few proprietary details for our discussions on return to power and boron transport, but we'll still cover as much in open session as we can.

So the presenters for the day are myself, Matthew Presson, licensing project manager for Chapter 15. We have been Ben Bristol, supervisor of system thermal-hydraulics. We have Megan McCloskey, our thermal-hydraulic analyst, and we also have Paul Infanger, who is available as needed.

This is a quick summary of the items we have to get through today. And with that, I'll pass it over to Megan -- Ben.

MR. BRISTOL: So this is Ben Bristol with NuScale, I'd like to thank the Committee for their time and certainly for the staff for their view on these issues.

The agenda topics that we're going to cover today first is a little bit of kind of a design overview philosophy from the NuScale perspective to set the framework for the two primary issues I think we'll spend a fair amount of time discussing today, which are related to the, you know, kind of passive cooling and hold-down, shutdown capabilities of the NuScale design.

And then, with that, we'll get into the updates that we made to the FSAR return-to-power analysis and then the work that we've done supporting the boron transport conclusions and distribution assessment. We also have some slides related to some of the other FSAR changes with respect to some of -- minor module changes as well as the RELAP code version updates, and then some of the ECCS logic changes. And then, we have some overall analysis updates if the Committee's interested.

Next slide.

So I wanted to kind of kick today off with a little bit of an overview. I don't think this is new information for anyone in the room, but I think it to me -- it helps set the context of the two major topics we're talking about related to the long-term reactivity control and hold-down capabilities of NuScale design.

So, in general, the characteristics of the NuScale design are passive. So, we have simple, passive heat-removal systems. We have a very large ultimate heat sink, and a fairly small core, and a large RCS volume.

And so, that sets the objective for the NuScale design in the module itself as being completely passive. And the safety capabilities of the design we believe are actually much more reliable because of those fundamental characteristics.

In particular, the safety systems, the heat-removal systems, actuate -- can be actuated passively, they actuate on loss of power. And, in particular, the module in the core can reach cold shutdown using CRAs alone, which includes the capability of accommodating the reactivity addition through complete xenon burnout. And that's something that as we get a little bit more into the return-to-power analysis, I'll address a little bit more specifically.

Yes?

MEMBER MARCH-LEUBA: When you mean CRAs alone, do you mean boron-free or xeon-free? Do you still need boron to keep the reactor shutdown at the beginning of cycle? If you remove all of the boron from the core, and you burn out all the xeon, are you subcritical?

MR. BRISTOL: No.

MEMBER MARCH-LEUBA: Okay. Because that sentence could be misunderstood.

MR. BRISTOL: Sure. Of that, there's two, I think, important things that we'll spend a fair amount of time talking about our analysis of, and one of those is the boron distribution in the system as a function of the passive heat-removal operations, including extended conditions up to the 72-hour analysis period for the design basis analysis.

MEMBER MARCH-LEUBA: So what you meant by that sentence is that maintaining the normal operating condition of boron concentration, control rooms alone gives you a lot of margin for shutdown?

MR. BRISTOL: That's right.

MEMBER MARCH-LEUBA: And you can then go to complete cold conditions and still remain subcritical?

MR. BRISTOL: That's right. So, as a result, there are no active systems in the design, and because of that, we do not have a requirement for safety-related power or operator -- safety-related operator action to mitigate design-basis events.

Next slide.

So, I think that sets us up with a little bit of a challenge, I think, particularly with the GDCs. And the Committee's been briefed throughout the review process of where the NuScale design stands with respect to particularly GDCs 26 and 27 related to reactivity controls.

One of the things I think I wanted to circle back on was that the design basis and event analysis historically has assumed a worst rod stuck out as part of the analysis of the design-basis events. So for the NuScale design, that's one-sixteenth of our overall hold-down capability.

That's specifically and particularly penalizing or is a substantial margin assumption with respect to the analysis of the larger PWR designs. In particular --

CHAIR KIRCHNER: Ben, I'm going to interrupt you. I don't think it uniquely penalizes you. You decided to design the core the way you did. I think it's a good design.

But, you know, one of those things that comes with having a small core is that if you use a traditional control rod assembly, then it's got a lot more worth.

So I'm quibbling with the way you phrased that, and also one of the reasons is that you're using -- I think it's good what you've done with the stainless steel reflector in flattening the radial profile, but that's why, you know, the exterior assemblies have more worth. But I don't think it's uniquely penalizing you. It's your set of design choices. You could have used a different control rod drive system. You could put, you know, twice as many control assemblies in.

So you stuck with the traditional fuel design and hence, the control rod assemblies, but I just -- I'm just quibbling with your words, not with your design.

MR. BRISTOL: Sure, and I think the point

I was trying to make there is one of 16 on a percentage basis is larger than one of 53 or more. So, I understand the comment.

So, in the evaluation of the stuck rod, and I think this may be what the Committee was alluding to is that the GDCs indicate that margin for stuck rods is part of the design basis. And so, for us, that is what we had assumed.

In review of the GDCs, and I think this is where our understanding of the applicability to passive designs, particularly passive designs that have the capability of extended passive cooling conditions, the implications of having enough hold-down to accommodate the entire xenon burnout is one of the things that it doesn't seem that GDCs were initially written to accommodate or allow for.

And so, particularly with respect to GDC 27 and close to accident mitigation and hold-down capability, it's specifically written in respect to addressing core cooling, and not necessarily hold-down or shutdown.

MEMBER MARCH-LEUBA: Is that because the xenon will burn out in a 24, 36, 48-hour time constant, and you would allow for operator actions?

MR. BRISTOL: Yeah, that's right. So

because of the analysis conditions for the NuScale design include consideration of hold-down and shutdown for periods greater than 36 hours, that wouldn't come up necessarily in the traditional, you know, plant fleet and in the analysis of those events.

DR. CORRADINI: Can you just say that again, I want to make sure I understand your point.

MR. BRISTOL: So our immediate shutdown and hold-down capability are more than sufficient to keep a subcritical for periods greater than 36 hours. And it's because we're analyzing out to a 72-hour period that many of these issues --

DR. CORRADINI: Okay.

MR. BRISTOL: -- that we're going to talk about today are an issue.

Next slide.

So boron-addition considerations, one of the -- we'll spend a fair amount of time talking about this today as well, but the mechanisms of ECCS cooling results in a natural-born accumulation in the core. We have a fair amount of analysis that's supporting that conclusion. So this is a similar phenomenon to PWR post-LOCA boron accumulation, and the fact that we do not have a continuous boron source is important in our demonstration of margin to precipitation limits.

So one of the concerns of over-accumulation of boron is if you would hit saturation limits and create some sort of solidification effects and core coolability concerns from that. For most of the cycle, this phenomenon actually helps and supports and enhances the long-term shutdown capability except for late in the cycle, and that's something we'll spend some time discussing.

So the consequences of, you know, a latent cycle, loss of shutdown margin at low temperatures are really driven by the very slow xenon burnout and are not a safety concern, you know, based on our analysis of the conditions. And so, therefore, the conclusion of consideration of the enhancement of some sort of boron-addition system, we determined that it was not required in order for the NuScale design to be analyzed and determined to be safe.

MEMBER MARCH-LEUBA: Yeah, I agree more with what you said than with what you wrote, okay? You wrote it would not make the design safer, and everybody agrees that would make the design safer. But to meet our design goal risk characteristics, we'll omit it.

It's a different thing than it doesn't have any impact. It would have an impact. And we all know that for any of these instances we're going to be talking about today, the operators will take control of the reactor, and we'll add boron --

MR. BRISTOL: Sure.

MEMBER MARCH-LEUBA: -- one way or another, even if they have to go there with a bucket on the top of the core to put it in. They'll have 72 hours to do it.

MR. BRISTOL: Right.

MEMBER MARCH-LEUBA: So, yeah, I agree with what you said, not with what you wrote.

MR. BRISTOL: Next slide.

So as the Committee knows that -- the NuScale DCA included an exemption request from GDC 27. The NuScale design doesn't align with precedence-based compliance of the GDC due to the lack of secondary -- or the lack of boron addition through ECCS quite frankly in reading GDC 27 in -- we'll talk a fair amount about the analysis of the core coolability versus the hold-down.

So what NuScale wrote in our exemption request was a principal design criteria. It's written -- the first sentence is written verbatim to the actual GDC, and then there's an additional sentence related to the capability of the long-term hold-down.

And I will cover that in the next slide.

MEMBER REMPE: So, I'm going to go back a slide and argue with my esteemed colleague, Dr. March-Leuba. I thought a long time ago NuScale was arguing that if they had put the boron-addition system that it would require additional penetrations and other things, and therefore, you did do some sort of risk assessment and decided this was the safer way to go.

So, you're maybe right for one characteristic about it's good to add boron, but I thought if you look at the whole plant -- am I misremembering things? Didn't you guys use to argue that it was safer to not have this additional capability so you could put the additional boron in?

MR. BRISTOL: Yeah, I think it's hard to maybe do a quick analysis of all the potential design solutions that we did look pretty heavily at options and potential design solutions, and I think we presented some of that to the Committee.

Some of those do actually increase the risk, right? Others may or may not have. I, you know -- I think I'd be speculating to some degree related to that.

But in general, there's definitely some

complexities about how much boron, when you're adding it, how you're adding it, that makes the design more complex, certainly, if not, potentially create additional safety concerns or challenges.

MEMBER MARCH-LEUBA: But the boron system and all the penetrations are already there in the CVCS.

MR. BRISTOL: That's right.

MEMBER REMPE: But it's the --

MEMBER MARCH-LEUBA: But you don't need any new -- they're there.

MEMBER REMPE: But isn't the addition to make it active or extra power or something? There are certain attributes if they tried to do this that would make it less safe is what thought that they were arguing is why they didn't do this.

MEMBER MARCH-LEUBA: I will argue --

MR. BRISTOL: So, I think --

MEMBER MARCH-LEUBA: -- him and me are going to start this statement.

MR. BRISTOL: -- our position was that the design considerations for a completely passive system that could preclude this hold-down issue, latent cycle where some of them could potentially be safety challenges or concerns, but in general, they weren't necessary to meet the safety goals.

MEMBER REMPE: I agree with the final thing of what you said, that it's not necessary for the design. But to say that it could always make it safer, I'm not sure I agree with that statement.

MEMBER MARCH-LEUBA: The boron addition system is built. It's there.

CHAIR KIRCHNER: It's already there.

MEMBER MARCH-LEUBA: It's there. All you have to do is operate it.

MEMBER REMPE: Yeah, but I thought there were some attributes of trying to get it to work that would make it less --

MEMBER MARCH-LEUBA: It's different if you want to install --

MEMBER REMPE: Make a safety-related or whatever --

MEMBER MARCH-LEUBA: If you want to make a passive simulator injection inside the vessel, for example.

MEMBER REMPE: Right.

MEMBER MARCH-LEUBA: You will have to modify other stuff.

MEMBER REMPE: Right.

MEMBER MARCH-LEUBA: That's not what we're talking about.

MEMBER REMPE: Okay.

MR. BRISTOL: Yeah, and one note, the operation of the CVCS would require, in some events, the isolation of containment. And so, that may have been part of the consideration of the overall event sequences, and if containment's unisolated, is that safer or less safe?

Our analysis of the design concluded that bottling the module up and letting it slowly, you know, remove heat to the pool was always the safest approach.

MEMBER REMPE: Okay. Anyway, go ahead.

MR. BRISTOL: So, one of the review areas that we have with the staff was related to the language of the PDC. So through the review process, we'd agreed on revising the language specifically. I've got a mark out there.

The original submittal had following -- the second sentence, following a postulated accident, the control rod should be capable of holding a reactor core subcritical under cold conditions without margin for stuck rods provided SAFDLs for critical heat flux are not exceeded by the return to power.

One of the concerns in the interpretation of the SECY related to this particular issue was whether an event had experienced fuel damage would end up with a return to power, could that be an additional radiological consideration? Within our DCA, there are no events that result in fuel failure, including our LOCA events.

And so, we agreed to basically remove the last part of that sentence and update the FSAR, committing to analyzing all of the events to meet SAFDLs, such that the additional consideration of a return to power wouldn't be a question from a fuel-integrity perspective or additional source-term perspective.

So in conclusion, we eliminated the second sentence, following a postulated accident, the control rod should be capable of holding a reactor core subcritical under cold conditions with all rods fully inserted.

MEMBER MARCH-LEUBA: Is this a modification from draft two to draft four, or, I mean, the timing?

MR. BRISTOL: Yes.

MEMBER MARCH-LEUBA: So your initial proposal had the red marking --

MR. BRISTOL: The red.

MEMBER MARCH-LEUBA: -- on the final one

is the green marking?

MR. BRISTOL: That's right.

MEMBER MARCH-LEUBA: Okay.

CHAIR KIRCHNER: So the implication then is that for your design-basis events, you would not exceed your SAFDLs?

MR. BRISTOL: That's correct.

MEMBER MARCH-LEUBA: And that's not part of your PDC, it's part of your FSAR commitments?

MR. BRISTOL: That's right.

DR. CORRADINI: But nothing -- I guess that's where I was going to go. Nothing has changed relative to how you're analyzing and looking at your figures of merit. It's just where you're committing to it?

MR. BRISTOL: That's --

DR. CORRADINI: Am I understanding this correctly?

MR. BRISTOL: That's correct.

DR. CORRADINI: Okay.

MEMBER MARCH-LEUBA: And again with what you're writing and what you're saying, I think you also commit to the acceptance criteria of the SAFDLs for LOCA? Which is a postulated accident, right? Or it's a DBE?

MR. BRISTOL: Yeah, it's a design-basis event, and it's a postulated accident.

MEMBER MARCH-LEUBA: Okay. But you do commit for LOCA?

MR. BRISTOL: That's right.

MEMBER MARCH-LEUBA: Okay.

MR. BRISTOL: Okay. So, with that, we'll get into some of the specifics related to the updates of the return-to-power analysis. So, as we just kind of discussed a little bit, the compliance with PDC 27s, the immediate shutdown capability is sufficient to protect a reactor coolant pressure boundary as well as SAFDLs, including consideration of the worst rod stuck out. Cold shutdown is achieved with all rods fully inserted. And that's a design commitment for all future core designs.

And the basis of that is analysis of the long-term loss of shutdown margin consequences considering the highest worth rod fully withdrawn do not exceed SAFDLs, and don't challenge DHR ECCS heat removal capability. In addition, the overall probability of the event is sufficiently small.

So, the initial submittal of loss of shutdown margin was what I'll characterize as a fairly -- is a very conservative and fairly simplified event that looks a lot like our non-LOCA event analysis. A pretty simple reactivity balance was performed, and the basis of that event was maximizing the transient power overshoot as the return to power progressed and then analyzing that using our, you know, methods for CHFR.

And throughout the review, there were some questions that we had by the staff about the actual characterization of the event. So, as we looked into the event further, it was clear that the event really was dominated by the xenon decay, and xenon burnout provided the boron redistribution wouldn't generate a problem or a concern.

So of the mechanisms for loss of shutdown margin, moderator cooling was the one that we'd initially looked at and had dominated the transients, and the analysis that we have updated to is really more driven by the fission product decay and loss of shutdown margin related to that.

DR. CORRADINI: Can you repeat that again? I'm trying to -- because your original response RAI was 9444, and that was the conservative bounding calculation as I remember.

MR. BRISTOL: Yeah, which was consistent with what we had initially submitted.

DR. CORRADINI: Right, and so can you just repeat the change in the --

MR. BRISTOL: And I'll get into the details as we go --

DR. CORRADINI: Oh, okay, that's fine. I'll wait.

MR. BRISTOL: -- in the next few slides, but I think the point here is that we had originally looked at the event using a pretty simple reactivity balance based on maximizing ECCS and DHR cooling. So it was -- we had presented it as a moderator-cooling-driven event. In reality, it is not a moderator-cooling-driven event. It is a xenon-decay-driven event. And that's the point here.

DR. CORRADINI: But if I might -- if I remember the response though, in your original conservative calculation, you ignored xenon burnout.

MR. BRISTOL: That's right.

DR. CORRADINI: Okay.

MEMBER MARCH-LEUBA: In the open session, I don't know if you could -- I read too many slides over the weekend, I don't know if it's open or -- you'll be talking about later today sometime different branches, ECCS is working, it's not working. The DHRS recovers or not recovers, but can you give us an overview, simple thing of what conditions lead to return to power?

In my mind, we need to have a very cold core, which happens when you are running ECCS and DHRS together. You have to be end of cycle, so you monitor temperature coefficient, it's very negative. And because you have low boron.

MR. BRISTOL: Low boron.

MEMBER MARCH-LEUBA: And in addition, you've got to have one stuck control rod. And while we talked to you earlier, you also were taking credit of high-decay heat that you could not have decay heat and reach return to power.

Is that still the case? Or you're not taking credit for high heat -- I mean, when we talked with you two or three years ago, you said you needed to have been operating at a steady state, shut down for a month, restart, don't have time to build up your decay heat and have the scram then.

That's what you told us then. Is that still the case?

MR. BRISTOL: So, I think we've -- the reason that we had looked -- we had been evaluating those scenarios was really a characterization of the probability. In review with the staff, the frequency of the event doesn't really come into play with respect to deterministic assumptions, deterministic analysis, and acceptance criteria. So most of what we'll be talking about today is how we reach those final analysis conclusions.

I think with respect to the question about decay heat, what we found is that the event is really more driven by the pool temperature conditions and the overall ECCS-cooling capability. Decay heat would help mitigate its -- I don't know if I would say it would completely preclude it from occurring in a 72-hour period.

MEMBER MARCH-LEUBA: So, easier for my mind, you can't keep this matter out of one of the top two parameters?

MR. BRISTOL: That's right. Yeah, it's much more driven by what's the boron in the system, and then what's the temperature of the coordinates --

DR. SCHULTZ: On moderator boron --

MEMBER MARCH-LEUBA: What is seen on disappearing?

MR. BRISTOL: That's right.

DR. SCHULTZ: But again, Ben, what -- you're going past this pretty quickly. That is, you're trying to make sure that you're maintaining the passive-design concept and the 72 hours that go along with that, I'd say. And that's why when you first looked at it, you made some assumptions you focusing on early events, something similar to what had been evaluated and analyzed for typical PWR non-passive, and then the longer period of time and maintaining a passive environment was what led to additional -- a need to look at further additionally at xenon decay?

MR. BRISTOL: Yeah, and I think the questions really in -- and during the review were related to the tools we were using and their capability of analyzing the problem. And we'd made assertions as to the conservative nature of them, which is true, but I think we have a better characterization of the events, and the analysis in the FSAR is a better characterization now of the actual conditions that could be reached.

DR. SCHULTZ: And also your input assumptions?

MR. BRISTOL: Right.

DR. SCHULTZ: You were looking at it in this -- if you will, a broad sense --

MR. BRISTOL: Right.

DR. SCHULTZ: -- in the first place. It needed to go into some level of detail --

MR. BRISTOL: That's correct.

DR. SCHULTZ: -- later.

Thank you.

MR. BRISTOL: So the boron redistribution, we have quite a bit of slides later in the presentation related to our analysis of that, but in general, the boiling-condensing systems cause -- can cause boron redistribution, boric acid is not readily volatized in the vapor phase, and that's really what's driving the accumulation of the boron in the boiling region. For us, that's the core.

And so, in conclusion, boron redistribution through extended ECCS operation is non-limiting from respect -- with respect to analysis of the limiting, you know, return-to-power condition and power -- the associated power level. That's supporting the conclusion that, you know, the events conservatively analyzed at the end of cycle were boron concentrations.

MEMBER MARCH-LEUBA: But going back -- so we will be to talking later on. You're talking about the normal transition of the event, not the recovery phase?

MR. BRISTOL: That's correct.

MEMBER MARCH-LEUBA: When you have to gain control of the other core and eventually take it to the refueling station?

MR. BRISTOL: That's right.

MEMBER MARCH-LEUBA: During recovery of the core, we have some concerns about that return to power. Okay, we will talk about that, right?

MR. BRISTOL: That's right, yeah. We have some slides.

So just a quick overview of the updated analysis, FSAR Section 1506 contains the evaluation of the return-to-power condition along with the conclusions. The updated method is described there.

At a high level, what we have done now instead of using conservative reactivity balance with our point-kinetics methods, we're actually applying boundary conditions in our SIMULATE code in evaluating the three-dimensional reactivity characteristics as part of the power evaluation.

So the RELAP long-term cooling model is used to do basically a state-point calculation for a given power level, what's the associated steady-state core-inlet temperature conditions? Those conditions then are passed to SIMULATE code, where the power distribution is evaluated.

DR. CORRADINI: So to put it in the

simplest terms, you iterate between a reactor-physics calculation, a thermal-hydraulic calculation to find the equilibrium temperature, which then creates a power which matches?

MR. BRISTOL: That's right.

DR. CORRADINI: Okay.

MR. BRISTOL: And I have a figure hopefully that will help eliminate this in a couple more slides. But essentially, we've got a couple of lines that look at core-inlet temperature -- or core-average temperature as a function of either DHR-cooling capability or ECCS-cooling capability, which is really driven by pool temperature.

And then we have a SIMULATE series of calculations that look at core-average temperature with a stuck rod, and the associated steady-state power level associated with that. And where those overlap is essentially where the critical power level could be, or the maximum power level could be.

And then, in our analysis of CHF, there are some additional conservatisms that are applied to that power, the power distribution, to ensure that we're getting a conservative evaluation.

So some summary conclusions of the updated analysis, re-criticality is precluded during DHR cooling with rise and recovery. The temperatures associated with those conditions are too high because of the limitations of DHR.

Once the circulation path has been interrupted, DHRS cooling -- the DHRS loop characteristically has a little higher resistance for heat removal then the ECCS loop. And so what we see is ECCS cooling really is what drives us to the cold conditions where the return to power is maximized.

MEMBER MARCH-LEUBA: I think we see that better on your figure.

MR. BRISTOL: Yep.

MEMBER MARCH-LEUBA: I've been looking ahead. But what you're saying basically is that the ECCS cools better than the DHRS and therefore results in lower temperatures?

MR. BRISTOL: That's right.

MEMBER MARCH-LEUBA: Okay. Now, go back to the previous bullet, control the action with additional stuck rod.

No, no, same slide.

The rules of the game -- and that's just completely arbitrary for analysis in Chapter 15, you have an accident, and you assume one single worst failure? So, in that sense, when you eject the control rod, you should assume that second controller also fails to insert when you scram.

MR. BRISTOL: Yeah.

MEMBER MARCH-LEUBA: What you have done is, you have run the short-term transient with that stuck rod, but you have not considered the two stuck rods for long-term return to power? Is that correct?

MR. BRISTOL: That's correct.

MEMBER MARCH-LEUBA: And the staff has to agree with you that that is reasonable? Can you explain why it's reasonable? Yes, I'm just trying to put you on the spot. I'm not complaining about it, but you really need to go through the logic.

MR. BRISTOL: So we do have a couple of slides in the -- I'll cover that if we want to hold --

MEMBER MARCH-LEUBA: I'll wait.

MR. BRISTOL: -- that question. In about three slides, we'll get into that, and we can -- we've got a couple of slides that address that.

DR. SCHULTZ: Ben, the SIMULATE uncertainties, I saw a bullet coming up. Are you going to talk about that more later?

MR. BRISTOL: Yeah, I think we --

DR. SCHULTZ: Are you in agreement with the staff with respect to the nature of that uncertainty and how you're addressing it, how you're thinking about it --

MR. BRISTOL: Yes.

DR. SCHULTZ: -- and how staff has?

MR. BRISTOL: Yes.

DR. SCHULTZ: Okay. We'll see that later in more detail?

MR. BRISTOL: In closed session.

DR. SCHULTZ: I saw one bullet.

MS. McCLOSKEY: I think the details of those numbers is something we discussed in the closed session.

DR. SCHULTZ: Very good, thank you.

MR. BRISTOL: So again, just an overview, the average core temperature is determined using RELAP with a state-point method. There's a spectrum of calculations looking at different initial conditions and cooling modes. Their evaluated critical power level is determined using SIMULATE with uncertainties addressed, including the worst rod stuck completely withdrawn.

And those analyses are performed for a series of different boundary conditions. And then the results are overlaid, and that's how we get the resultant power level. And then, the CHF and power distributions are conservatively analyzed as part of the CHF evaluation.

MEMBER MARCH-LEUBA: And that CHF correlation will regard the LOCA CHF correlation benchmarks well with the Stern tests, and it's verified to be conservative with respect to the prototypical Cathie tests, right?

MR. BRISTOL: That's right.

MEMBER MARCH-LEUBA: So this is an accurate CHF correlation?

MR. BRISTOL: So, the summary of the loss of shutdown margin evaluations that the limiting initial conditions boron is minimized. So it's basically an end of cycle hop power starting condition, starting with the equilibrium xenon and then cooling is -- DHR ECCS cooling is maximized. And that's done by looking at maximum pool level, minimum pool temperature and high-biased, steam-generating DHR heat transfer capabilities, as well as looking at the maximum ECCS capacity.

MEMBER MARCH-LEUBA: Just out of curiosity, what's the capacity of the CCS?

MR. BRISTOL: So that's with respect to single failure of a valve to open or the flow coefficients. So all five valves are active in their

-- the float capacity is set to the maximum.

So reactivity bias is applied in SIMULATE in order to account for the code bias or code uncertainties, and conservative peaking's applied. In addition, a dynamic factor of two was added to the local heat flux in the CHF evaluation.

And what that is intended to account for is the events -- we're analyzing the event in a state-point mode. In reality, there will be some dynamic effects. We expect them to be very small because of the rate of reactivity insertion of xenon. And so, this is just an engineering uncertainty factor that's applied to ensure that we've got a conservative calculation.

DR. SCHULTZ: So you've looked at what you had before you applied these uncertainty factors as you're calling them?

MR. BRISTOL: Yes.

DR. SCHULTZ: And are you going to show the results without the factors, or you just move forward with some conservatisms applied?

I think we'd like to know what -- why these are applied and then move on.

MR. BRISTOL: I think we move forward with conservatisms applied.

DR. SCHULTZ: But, you mean -- you're not telling the whole story if you don't talk some about best estimate expected values, and so on.

MR. BRISTOL: Certainly. And just as a little bit of context, the original temperature-driven transient response had maybe a three or four X power overshoot. The reactivity insertion rates associated with that were much, much larger than what we get with the boron-driven -- or with the xenon-driven transient response.

So our analysis has shown that we will reach the -- with the consideration of xenon burnout, we reach practically equilibrium or steady-state temperature conditions if we're looking at DHR cooling or ECCS cooling. So there's really very little temperature transient response in the return to power.

The rate at which xenon's increasing is very, very slow. So you will expect to have almost zero dynamic effect. As reactivity is slowly increasing and you approach K effective of one, then there's some power response that's associated.

DR. CORRADINI: Did that help you? Because it didn't help me.

DR. SCHULTZ: A little, you know, there's two things anytime you say, well, I'm going to provide

-- I'm going to apply a conservative factor, I'm going to increase the power factor in order to demonstrate that I'm good to go. It's nice to know what impact that each of those has on the result. So you understand what conservatisms, in fact, you're applying.

So, we'll see.

(Simultaneous speaking.)

MR. BRISTOL: Sure, and I think we have a little bit more of the details of the analysis in the closed session, and maybe we can circle back to this specific uncertainty.

MEMBER MARCH-LEUBA: On the top of your head, do you remember what the CHFR value is? Is it 1.7, 1.9, 3.2 during the total criticality?

Top of the head -- it's not 1.2.

MR. BRISTOL: No, it's not. The way that we've performed the calculation, it's greater than four.

MEMBER MARCH-LEUBA: Right. So there's 400 hundred percent margin to critical heat flux?

MR. BRISTOL: Yes.

MEMBER MARCH-LEUBA: So uncertainties -- I don't know why you even bother to put them in.

DR. SCHULTZ: Yeah, that's part of what

I'm trying to understand because it sounds great, but --

MEMBER MARCH-LEUBA: No, I mean, the power they're going to reach is so low that the temperature there -- the PC -- the cladding won't even see a temperature increase.

DR. CORRADINI: I'm still confused about the evolution, but I'll wait.

MR. BRISTOL: And the picture, finally a picture. Okay, so what I've been describing here, I think we presented this figure last June when we were out discussing the analysis updates for Phase 2 --

DR. CORRADINI: Is it the same picture? It looks different.

MR. BRISTOL: It is a similar picture.

DR. CORRADINI: Okay.

MR. BRISTOL: This is the final picture. What we've had previously was usually based on preliminary results, yeah.

So the various lines here, we have analysis of DHR cooling with the riser uncovered, DHR cooling with a covered riser essentially, inventories maintained in -- or hot zero power conditions are assumed, but regardless that the inventory is sufficient to keep the natural circulation loop.

And then ECCS -- you'll see characteristically that that ECCS results in equilibrium conditions that result in much lower temperatures.

MEMBER MARCH-LEUBA: So, Ben, each of these dots, the red, blue, and purple dots on the screen, each of those represent an independent and steady-state real applied calculation?

MR. BRISTOL: That's correct.

MEMBER MARCH-LEUBA: So you input the power, decay heat, say, one percent?

MR. BRISTOL: Yep.

MEMBER MARCH-LEUBA: And you calculate what the average temperature would be --

MR. BRISTOL: That's right.

MEMBER MARCH-LEUBA: on each of those?

And then you have a straight line which you will tell us what it is, the solid line?

MR. BRISTOL: That's right. And so, the critical power is essentially translation of those boundary conditions in to SIMULATE -- one of the additional considerations and conservatisms that we have is that there's a flow factor applied in the SIMULATE conditions. SIMULATE is effectively at zero void feedback associated with the way that the power is calculated.

So under ECCS conditions, we have low pressures. We have very little flow, and we actually expect some boiling effects. None of those are considered as part of the steady-state power calculations.

DR. CORRADINI: So just, so void is ignored, void feedback is ignored in the black line?

MR. BRISTOL: That's correct.

MEMBER BALLINGER: There are five, four lines but five -- oh, oh, I -- ah.

MR. BRISTOL: Oh, yes, so the last line, 72-hour decay heat. Essentially, we were running the analysis, the safe point calculations for RELAP up to the point where we would have 72-hour decay heat conditions. So where the black line crosses over the vertical line, that's essentially beyond the 72-hour conditions.

MEMBER BALLINGER: Red on the screen but black on my computer.

MR. BRISTOL: So essentially because the covered, or the uncovered DHR line would cross the critical power line past 72 hours, that led us to the conclusion that we don't, we can't get low enough decay heat to get to a critical configuration if we're just in DHR cooling for a 72-hour period in the riser.

DR. CORRADINI: So I want to make sure I understand this. So the black line would actually be lower if you considered the local void, that's 0.1. The black line would be lower if I consider early after xenon burnout, because I'm at 36 hours, I start worrying about this. So the black line would be lower there for an equilibrium.

You don't have the uncertainties identified. In other words, it's not a black line, and it's a black fuzzy line. So there's uncertainties that it would be lower still than that.

I'm trying to understand, though, the little dots, and how would they, how they move if I start feeding in the uncertainty and I did a best estimate calculation versus your bounding calculation. I can kind of guestimate where the black line goes, I'm having a harder time with the other ones.

MR. BRISTOL: So the things that would change, the elevation, I'll call it, of the dots. So the vertical axis is temperature, right. And what goes into feeding the RELAP temperature is mostly what the pool temperature conditions are assumed to be.

DR. CORRADINI: So everything would fall.

MR. BRISTOL: Everything would increase.

Those dotted lines are as low as they can be.

DR. CORRADINI: Okay, because you've maximized cooling, excuse me, excuse me.

MR. BRISTOL: That's right.

MEMBER REMPE: So you've said that a couple of times about you assumed a minimum pool temperature. What temperature did you --

MR. BRISTOL: Sixty-five degrees.

MEMBER REMPE: Sixty-five, and how high did it get in the analysis? Or do you remember?

MEMBER BROWN: Are you talking about the return to power power, or the decay heat?

MEMBER REMPE: I'm talking about the pool temperature.

MEMBER BROWN: The pool temperature is assumed to be constant.

MEMBER REMPE: It stays constant, you've left it there the whole time. Okay, you didn't, okay, because -- anyway.

MR. BRISTOL: If we were dumping all of the module decay heat to the pool, it would increase over a 72-hour period, that's right. We did reperform this calculation at different pool temperatures, which led to, just as a benchmark, the 140-degree pool temperature brings that ECCS line about the black line.

So essentially if the pool is heating up at all the return to power is limited because of the temperature conditions. Okay.

CHAIR KIRCHNER: What is the, roughly what is the power level, that line -- when it returns to power, what's the, what is the power?

MR. BRISTOL: So --

CHAIR KIRCHNER: X megawatts?

MR. BRISTOL: One percent is 1.6 megawatts.

DR. CORRADINI: That's what I was going to say.

CHAIR KIRCHNER: Oh, you're using, I see what you're doing. You're using it for double duty, decay heat and power.

MEMBER MARCH-LEUBA: With a decay heat of one percent, you have sufficient power to shut down the reactor. If you go for ECCS core, the purple line. If you have decay heat that is lower -- if your total power in the core is less than 1 percent, then you can't reach critical heat. And you have to alternate between nuclear and decay heat, so whatever you don't make with decay heat, you have to make it with nuclear.

MEMBER BROWN: Why does decay heat return you to power?

MEMBER MARCH-LEUBA: It doesn't.

MEMBER BROWN: Normally it's critical that returns you to power. I've lost the ball.

MEMBER MARCH-LEUBA: If you have sufficient decay heat, the core is hot enough.

MEMBER BROWN: Oh, you're trying to stay about 175 or 180 degrees or something like that.

MEMBER MARCH-LEUBA: Yeah.

MEMBER BROWN: If your, once your decay heat goes -- now I get the connection, because once your decay heat goes away, then --

MEMBER MARCH-LEUBA: Your ---

(Simultaneous speaking.)

MEMBER BROWN: Then you need and you can have as much as anywhere from one percent to three percent, or 2.9, according to their FSAR. Is that, my memory is correct?

MR. BRISTOL: That's right.

MEMBER MARCH-LEUBA: And this is only with Doppler, not void. Doppler and moderator.

MR. BRISTOL: That's right, yeah.

So the results summarized, ECCS is most limiting results in an equilibrium power of around one to two percent. That's with the additional consideration of some of the uncertainty factors. I believe the graph presented there is the nominal simulated calculations, and so those are also performed at, with uncertainties applied in addition to the CHF power distribution penalties that are applied.

Core temperature must be low 200 degrees. As we discussed, if pool temperature effects are considered, then up to 140 degrees the return to power would be precluded. The onset of the return to power is --

MEMBER BROWN: That means below 140, that's when you're not precluded, is that correct?

MR. BRISTOL: That's right.

MEMBER BROWN: Okay, just trying to make sure my -- so we want lots of decay heat just to stay around all the time, is that right? That's not going to happen, but I mean it's, sooner or later we're going to return to power under the scenario if your temperature goes, decay heat goes away and the plant cools down.

MR. BRISTOL: Yes.

MEMBER BROWN: And it could sit there for weeks. Same discussion we had months ago.

MR. BRISTOL: Sure.

MS. McCLOSKEY: If you assume the worst rod is stuck out.

MEMBER BROWN: I understand that, that's part of the -- but that's one of the required evaluation things, that's all.

MS. McCLOSKEY: Yes.

MEMBER BROWN: I still have my same objection as I voiced before.

MR. BRISTOL: So the timeframe, again, based on the analysis of the xenon is around the 40-hour time frame after scram the combination of the temperatures you could reach, plus the xenon poison that's left. That's when the critical, criticality could begin to start. And then obviously we analyze it at the state point of zero xenon left.

So the CHF results show this is non-limiting based on the actual critical power levels that we can get to. Again, there's a fair amount of conservatisms in those.

The other AOO acceptance criteria are met, and there was some analysis looking at some of the other SAFDLs, particularly rod pressure, that concluded this was also non-limiting for this.

DR. CORRADINI: So the analysis you're

showing us here, or the conservative end of the analysis, is an RAI response that we should have had?

MR. BRISTOL: Yeah, so most of this information's actually in the, in 1506, in the FSAR.

DR. CORRADINI: Oh, okay, fine.

MR. BRISTOL: The RAI response that included some of the details is 9485.

DR. SCHULTZ: And Ben, just remind us what, where you are in cycle here. These are end-of-cycle conditions, concerns?

MR. BRISTOL: Yeah, zero boron. Or close to zero boron, yeah.

DR. SCHULTZ: Where in cycle does that become, does this all become an issue? Talking about the last 60 days?

MR. BRISTOL: We haven't --

DR. SCHULTZ: He's again looking for March.

MR. BRISTOL: Yeah, so in terms of the characterization, I think we've got a better characterization of the event presented. We have not evaluated the entire cycle burn-down condition to try to target in where this -- at what concentration levels we could expect this or not expect this. Part of the review process was, again, looking to maximize

-- or minimize the margin, and so we focused on the end-of-cycle considerations.

DR. SCHULTZ: Sure, okay. More details to follow.

MEMBER MARCH-LEUBA: Ben, that's a very simple calculation. You are trying to do it the hard way, but the easy way is you know how much your posterior activity is.

MR. BRISTOL: Sure.

MEMBER MARCH-LEUBA: And see how much worth -- how much boron you need to compensate for that, and then you know how many ppms, and that tells you how many days.

PARTICIPANT: Yeah, just trying to understand the sensitivities.

MEMBER MARCH-LEUBA: It's a very simple hand calculation, you can do it right now. I mean, you were trying to do a very difficult, do the whole analysis at every time slice.

MR. BRISTOL: Sure.

MEMBER MARCH-LEUBA: But you can do it much simpler.

PARTICIPANT: Thanks.

MEMBER PETTI: And the decay heat is the nominal decay heat, that you put the multiplier on the

ANS?

MR. BRISTOL: So the, in terms of maximize -- or minimizing the temperature, we're looking at minimum decay heat conditions.

DR. CORRADINI: That's what I thought, that's what I thought.

MEMBER RICCARDELLA: Excuse me if this is a dumb question, because this isn't my area, but if it does return to power, it doesn't stay there, right? I mean, it starts heating up and then it, then the void coefficient brings it back down. What's the rough time period of those oscillations? You know, is it like a minute or an hour?

MR. BRISTOL: So because the way the event is driven by the slow burnout of xenon, it will actually be just an equilibrium condition that's reached. It will be offset more by the actual thermohydraulic, local thermohydraulic conditions. Because of the concern of addressing the uncertainties with respect to those, we've provided some pretty bounding analysis of those conditions.

MEMBER RICCARDELLA: So the xenon burning out then cancels out the moderator effect?

MR. BRISTOL: No, so the xenon burning out provides the reactivity to drive the increase in neutron power level.

MEMBER RICCARDELLA: Yeah.

MR. BRISTOL: And that will reach effectively state point equilibriums for given hour conditions after 40 based on the local temperature feedback effects and local void feedback effects.

MEMBER RICCARDELLA: So a quick stay at like one percent power for -- under these extreme assumptions, it could stay at one percent power.

MR. BRISTOL: Yeah, so the one percent power level is the limit.

MEMBER MARCH-LEUBA: Yeah, if it reaches criticality as Charlie says, there is nothing that will bring it down. It will stay heated forever until the operator takes control.

MEMBER BROWN: And it depends on the temperature.

PARTICIPANT: Or the pool heats up.

MEMBER BROWN: Sixty-five is a higher power than it is at 140 degrees.

MEMBER MARCH-LEUBA: Yeah, one thing you could do is warm up the pool. I would, no, the --

DR. CORRADINI: I think Charlie, you're -- Charlie, I think you're misinterpreting on how conservative the calculation is. That's my interpretation. They've ignored void effects, they've ignored pool heatup, which are two biggies.

So the calculational number -- but the calculational number that they come up with is -- I think that's where I was, I think Steve was pushing them to know what it is versus what the best estimate is, that's what I thought he was after.

MR. BRISTOL: And maybe to illuminate on that a little further, in the evaluation of the likelihood of the event, the two things that seemed to dominate were the probability of the rod being stuck to start with.

So that's why further, you know, I guess pencil-sharpening of precisely characterizing deterministic assumptions as related to realistic operating conditions wasn't an objective we were looking for for demonstration of the safety.

MEMBER PETTI: I just want to note for the Committee's benefit, this is an issue in lots of small advanced reactors. And there have been benchmarks done to try to predict this. This is not easy to predict, like to predict the time at which you will go re-critical. You see all the different things.

They've done a, sort of an interesting state approach. But to take the best codes you have and to try to calculate it, they are all over the place in terms of the time. They all get, yes, it gets critical and it'll oscillate to the steady state. But they're not all sitting there together. Because of all the different parameters, it's not an easy calculation.

MEMBER MARCH-LEUBA: Yeah but the key argument here is is it possible or is it not possible to go recritical. And with a bounding calculation it's possible. What are the consequences of it? And the bounding calculation says none.

MEMBER PETTI: Right.

MEMBER BROWN: Except your reactor's critical forever.

MEMBER MARCH-LEUBA: Except that your reactor's critical forever.

DR. CORRADINI: If that worries you, get ready for a bunch more of them.

MEMBER BROWN: Come on, Mike.

MEMBER BLEY: In power reactors you allow operator actions in any of the design basis events. And in your old reactors, you allowed operator actions, and we're ignoring that completely too.

MEMBER MARCH-LEUBA: Yeah, but we're ignoring it because they want us to ignore it. They don't want to take, they don't want to take credit for it. The whole issue is a paper exercise, because when we move to Chapter 19, we allow for operator actions with a certain probability of failure. And then this doesn't happen.

MEMBER BROWN: Yeah, well, our operator, just as a point, the Navy reactors don't have boron, so if your rod's stuck, it's stuck. You don't, you can't do anything else unless you go drill a hole in it and start pumping something in.

PARTICIPANT: And if it gets cold water?

MEMBER BROWN: We were down pretty -- like I can't tell you the number, it's pretty damn cold, okay. You're just a little above freezing. Forty degrees.

CHAIR KIRCHNER: But since it was brought up that other advanced reactors have this set of issues, it's not a given going in that the consequences are so benign. And that's going to challenge us going forward significantly. Because some of the factors that were discussed here are much better qualified for LWR systems than they will be for advanced designs with different coolants and such.

And so we can be fairly confident here in the analysis about not exceeding, for example, CHF, but in other designs that's going to be much harder to demonstrate.

MEMBER BROWN: Fundamentally if CDCS was classified as a safety system, then you could count on it to be able to keep yourself sub-critical. And the issue here is they don't get, they don't want to call it safety system.

MEMBER MARCH-LEUBA: Yeah, my concern I think is what Walt, in the back of your mind, you're thinking, is that we don't want this PDC-27 to create a precedent for every other reactor.

MEMBER BROWN: Exactly.

CHAIR KIRCHNER: Well, and also, and for the advanced reactor design criteria, I think the staff has clarified this particular issue in a way that will give us a little more certainty about what the requirements are going forward.

MR. BRISTOL: Okay, so circling back to the topic that was brought up a few minutes ago, evaluation of GDC-28 with respect to PDC-27, NRC had an RAI related to this.

MEMBER MARCH-LEUBA: For those of us that don't talk numbers, what's GDC-28?

MR. BRISTOL: GDC-28 is the requirement to analyze the maximum postulated reactivity insertion and demonstrate that essentially the fuel design is capable of withstanding that.

And so just to build on that, the purpose of the return-to-power analysis, again, demonstrate compliance with PDC-27. And is bounding with respect to the analysis of the control rods that are evaluated with, for the rod ejection event.

So what I mean by that is the worst rod is one of the -- for the return-to-power analysis is one of the outer control rods that's used for shutdown purposes and not used for control.

In the NuScale design, there's two separate banks that are used for reactivity control purposes. And those a limited by our power-dependent insertion limits we talked about a couple weeks ago.

And the rods that are evaluated as part of the rod ejection event are those rods that are actually inserted during the cycle. The rods that are looked at for the long-term hold-down perspective are the ones that are fully withdrawn throughout the cycle operation.

MEMBER BROWN: To jog my memory, PDC is proposed design criteria -- principal. That's, in other words, that's your-all's new design criteria, as opposed to the present rules general design criteria.

MR. BRISTOL: Correct.

MEMBER BROWN: Based on the exemption you're requesting.

MR. BRISTOL: Correct.

MEMBER BROWN: All right, just make sure I understood that. Thank you for principal, same thing.

MR. BRISTOL: So again, the control rod ejection analysis is performed to demonstrate compliance with GDC-28.

As we discussed a couple weeks ago, what's primarily driving that event is the Doppler feedback effects due to the reactivity insertion and the enthalpy disposition, or the enthalpy deposited is evaluated to ensure that the pellet design is sufficient and the reactivities that we're inserting are insufficient to challenge the actual mechanisms of the mechanical fuel assembly.

So in response to the RAI, we evaluated the postulation of the ejected control rod. And we discussed this a little bit a couple weeks ago. The summary is that the rod ejection break, or the break of the weld or the nozzle, is non-mechanistic in the NuScale design. However, it is still postulated for the purposes of evaluating GDC-28.

And we concluded that the extension of PDC-27 to the rod ejection event was not required because of the conclusion, the mechanical design basis of the control rod assembly and the way it's attached, welded to the vessel head, such that we wouldn't need -- we wouldn't analyze the dynamic effects of a control rod potentially taking out a control rod associated with it or surrounding it.

And that the basis of an additional control rod being assumed to not insert as part of the rod ejection event was based on analysis of or disposition of the dynamic effects of the actual break itself.

MEMBER RICCARDELLA: Maybe my colleagues could help me. With an existing, with the existing fleet, is it common to assume a control rod ejection in conjunction with a LOCA? I mean to get to the GDC-27 concern, basically you postulated a LOCA, right? And then on top of that, you have to assume a control rod ejection? That doesn't seem.

MEMBER MARCH-LEUBA: GDC-27 postulates a loss of power for an extended period of time. The return to power assumes you have a scan. A rod fails to get in for whatever reason, and then you lose power for 72 hours.

MEMBER RICCARDELLA: But if ECCS is running -- you're running ECCS because you've had a LOCA, right?

MEMBER MARCH-LEUBA: No, no, it because you've lost power.

MEMBER RICCARDELLA: Oh, okay.

CHAIR KIRCHNER: Yeah, and then the rod ejection scenario, it's just assumed that the rod is in a step change of reactivity based on the rod worth.

MEMBER RICCARDELLA: I understand that --

CHAIR KIRCHNER: There's no, there's no LOCA associated with --

MEMBER RICCARDELLA: I'm just questioning the rod ejection in conjunction with some other, you know, postulated event or postulated accident. That's what that seems to be, that's what --

MR. BRISTOL: So I think we'll maybe get into that in the next slide.

MEMBER MARCH-LEUBA: No, wait, wait, don't go. Because the most common in my mind, I mean the failure of one rod not to go in could be binding. And you have bowing of the channel. So you could have bowing on rod 22 -- well, you have only 16, rod 13, and then you eject rod 2. And then you try to scram, and the channel doesn't let you go in, right. That would be the scenario. But why don't you analyze that?

MR. BRISTOL: So in, I've got a little bit more detail, our review of the precedent associated with the analysis of GDC-28 and the control rod event, there was -- we did not find evidence of a link to a consideration of hold-down in GDC-27. So --

MEMBER MARCH-LEUBA: What's hold-down?

MR. BRISTOL: Reactivity control, post-accident reactivity control. So the question that was asked by, you know, in the RAI was rod ejection is an accident. GDC-27 cites accidents, the reactivity control for accidents. So why wouldn't you consider the event which takes out one of your rods in addition to a stuck rod.

And our review of the analysis and methodologies related to GDC-28 did not find a link to, back to GDC-27 as a consideration of an ejected rod with a stuck rod and its effect on reactivity control.

DR. CORRADINI: So can I say it in a little bit simpler? You're saying that the likelihood of the combination is low.

MR. BRISTOL: The likelihood of the combination's extremely low, so based on that we could build, you know, we have an argument to build that as outside of the design basis.

DR. CORRADINI: So I want to get back, though, to Pete's question, because we've never -- and I guess I didn't know how to answer it. But I don't sense that current LWRs are required to consider simultaneous stuck rod in a rod ejection. But I'm not an expert in this, that's why I didn't want to answer your question.

MEMBER MARCH-LEUBA: I'm not sure about that.

MEMBER BALLINGER: I think there's a periodic requirement to measure the force it takes to put a rod --

MEMBER MARCH-LEUBA: Because otherwise they're stuck.

MEMBER BALLINGER: Right, and so that precludes a stuck rod.

MEMBER MARCH-LEUBA: No, it doesn't preclude. You do it because the stuck is likelihood. There is a likelihood of sticking. You spend all that money and time and effort to test all the BWRs, and even in the control room where they were doing it. It's a lot of time during a whole weekend. Because you cannot preclude the fact that it will stick otherwise. By testing it, you reduce the probability of it sticking, but the sticking happens.

MEMBER BLEY: Forty years ago we had a number of stuck rods.

MEMBER BALLINGER: I remember that.

MEMBER BLEY: And we started doing this testing.

MEMBER BALLINGER: Right.

MEMBER BLEY: And we still do it. I don't think we've had any other --

MEMBER PETTI: But do the rods get stuck anymore?

MEMBER BLEY: It's been a long time.

MEMBER BALLINGER: Yeah, they measure the force, and it's beyond a certain amount, they fix it. That's my understanding.

MEMBER MARCH-LEUBA: They reduce the likelihood, certainly, but testing you have a lower likelihood.

MEMBER RICCARDELLA: It just seems to me that the combination of those two events is a very, very low likelihood. But I guess that's maybe ---

(Simultaneous speaking.)

CHAIR KIRCHNER: We'll hear from the staff why they asked the RAI. But for the existing fleet, again, they'd have a lot more control rod assemblies and individual relative worth is lower. So a stuck rod is not such a problem.

Because for the hypothetical ejection, rod ejection accident, I mean, it's the Doppler that saves, you know, the local assemblies from damage. And then you scram. But you've got a lot more rods to scram in a large existing plant. But I think we can query the staff when they.

MEMBER BALLINGER: This core is about one-third height, right?

CHAIR KIRCHNER: One-half height.

MEMBER BALLINGER: One-half height. So that mitigates against sticking, because big PWRs have a big, tall core.

CHAIR KIRCHNER: Well, here they have much more mass in the control rod extension because the assembly, the drive mechanism is way up in the top of the containment vessel, or the reactor vessel. And so it's a much heavier control rod assembly, versus the existing fleet. And you'll remember they presented on things like drop and other issues that become mechanical design problems for this design that don't exist for the fleet.

DR. SCHULTZ: Appreciate the staff coming forward with their evaluation, but my recollection was that one reason why this came up at all was that you were going for an exemption to GDC-27, and the staff wanted to be absolutely sure that you had considered any reactivity impacts associated with the control rods in creating the PDC-27 concept.

MR. BRISTOL: So again, the, in our review of the application of GDC-27, it is not cited as part of SRP 1548, that's the guidance on rod ejection analysis, or in Reg Guide 1.77. And the application of GDC-27 was not required for other approved rod ejection methodologies. Therefore, the extension of PDC-27 into the rod ejection event by similarity was not warranted due to some additional unique design considerations.

MEMBER MARCH-LEUBA: And again, I'm having a problem with your language. GDC-27 is required to be applied for every accident. I mean, this are general design criteria you must satisfy always. It is required. It's just not used as the methodology. Because you satisfy the first 20 seconds, you reach acceptable low power, probably shut down. And what happens 72 hours later is no concern to the Chapter 15 analysis.

But GDC-27 is applicable to everything in the operating reactor, unless they get an exception, like you are. I mean, it's just --

CHAIR KIRCHNER: It's the ensemble.

MEMBER MARCH-LEUBA: Yeah.

CHAIR KIRCHNER: It's the ensemble of GDCs, too, it's not just you pick and choose one.

MEMBER MARCH-LEUBA: And you satisfy all GDCs all the time.

CHAIR KIRCHNER: It's 26, 27, and 28 together that define the reactivity limits and controls for, as basic design principles.

MEMBER MARCH-LEUBA: So when the staff asked why don't you analyze the return to criticality with two rods stuck, one that was ejected and one that gets stuck, they have logic behind it. Now, what you're saying is that this hasn't been done historically for all of the reactors.

MEMBER PETTI: Yeah, but --

MEMBER MARCH-LEUBA: That's correct. And that's because all the reactors have safety-grade boron, safety-grade power, and therefore the stuck rod for 72 hours makes no difference.

MEMBER BROWN: Doesn't matter.

MEMBER MARCH-LEUBA: And in this reactor, we have good quality boron, good quality power, and in reality, they will be injecting boron anyway. The problem is trying to make the argument we omitted.

MEMBER BROWN: Well, their argument that they, if you look at part of their FSAR says the overcooling and return-to-power analysis can be safely cooled by DHRS or the ECCS. So they can operate forever at one percent or two percent or three percent.

And they'll remove heat, you know, while people go home and have a few beers and come back three months. It'll be just fine, because that's all, you don't need anybody to do it. So if there's nobody they can abandon the plant, and it'll take care of itself for months.

I'm being a little bit exuberant with my comment, but that's fundamentally what you're claiming with the design of your DHRS system and your ECCS system, which are just non-active NC systems fundamentally.

MEMBER RICCARDELLA: Except if while they're home having that beer they have a rod ejection.

MEMBER BROWN: Well, come on, you can't -- I, even I won't throw that one into the hopper.

MEMBER RICCARDELLA: I mean, that's what

we're talking about here.

MEMBER BROWN: No, it's the stuck rod issue.

MEMBER MARCH-LEUBA: If you -- you eject the rod then one rod sticks.

MEMBER BROWN: Yeah, I, and that's, I have a hard time even signing up with that one.

MR. BRISTOL: Yeah, I think none of us expect whoever builds and owns one of these to ever walk away.

MEMBER BROWN: I'd say --

MR. BRISTOL: However --

MEMBER BROWN: But that's the point, Ben, is that you're fundamentally saying that we can cool this thing forever and don't worry about it. So we got you can't go in the compartment, you can't go in the place because you're generating power. You have to do everything from an external source if you're going to do anything now.

MEMBER MARCH-LEUBA: Since we're having so much fun on this, can I interest the staff in letting us know what is your proposal for not considering two rods out of the core for long-term. What's the regulatory basis for accepting only one rod is stuck, not two?

MR. SCHMIDT: So we'll go over that in a staff presentation, but it's basically what NuScale has up there.

MEMBER MARCH-LEUBA: Which is?

MR. SCHMIDT: Which is basically it's a stylized event to really check control rod and loading patterns so you don't have a mismatch that could if you were to postulate an ejected rod such that you disassemble the rod. We don't look at the combination of what would be like an N-2 configuration.

MEMBER MARCH-LEUBA: But we'll grill you on that later.

MEMBER BROWN: How far behind are we now?

MR. BRISTOL: Yeah, I think we're moving along okay. Sorry.

MEMBER PETTI: A question of the staff. You know, we went into this what do we really mean by these GDCs. Is there a place where the rationale is actually documented so that -- I mean I am thinking forward where we're going to have a ton of reactors with different coolants and different configurations potentially.

MEMBER BLEY: When they came out with rod -- I'll jump in and the staff can. The ARDC is the advanced reactor design criteria. They have a table in there, and on the righthand column, they give a pretty thorough explanation of their rationale for these. And they might have something better to say, but that's one place we've seen it in the recent past.

MR. BRISTOL: So just, this is my last slide. But just to follow up that the basis had two primary parts, one was precedent-based in our interpretation of the purpose of analyzing a maximum reactivity insertion was not coupled with demonstration of the reactivity control capabilities, right.

So that's where we're parsing the two GDCs from being independent in terms of their, what their intent is. Which, again, is ensuring that the fuel design is capable of withstanding -- or the fuel design in combination with your control limitations fit together to ensure that you can withstand this postulated event.

But within that, deterministically it's not actually a postulated mechanical failure, and therefore wouldn't be considered like a LOCA scenario that then progressed with a stuck rod.

And so I think that's where we're parsing these two issues is if the rod ejection mechanical failure of the CRDM housing isn't a postulated break location and there's no other mechanism for it to be ejected, then the application of one worst rod stuck out for the purposes of demonstrating long-term hold-down and the safety of that is sufficient.

An in conclusion, the estimated likelihood of the failure of the rod to insert and the failure of the boron addition system would result in a return to power that's extremely unlikely.

CHAIR KIRCHNER: But it would, Charlie, this is addressing the missile, the hypothetical missile. And Pete. Any particular questions you want to ask here?

MEMBER BROWN: I didn't particularly understand that they were now missile doing it. My concern on the missile thing was the breaking of the containment.

CHAIR KIRCHNER: No, yes, but this indirectly addresses is it a likely event to have a control rod ejected or the shaft become a missile.

MEMBER MARCH-LEUBA: Today is better day to argue what we were arguing a couple weeks ago than during the methodology topical report. This is Chapter 15, this is where we apply it. So if you want to follow up your argument, now is your opportunity. Peter?

MEMBER RICCARDELLA: Well, control rod eject --

MEMBER BROWN: Go ahead.

MEMBER RICCARDELLA: Well, somebody wants to come to the mic. Are you addressing this issue?

MR. INFANGER: Paul Infanger, Licensing. We've had some discussions with our engineering, structural engineering, about the control rod missile. And the evaluations that were done were based on previous reactors, the pressure boundary was a quarter-inch plate of metal inside a concrete containment. So the missile that was of concern was that a missile could crack that quarter-inch plate of steel.

Our situation -- and so what most of them did was they, in order to avoid doing that analysis, they put in missile shield. So what we have is a containment head that's two to three times as thick as a missile shield and a higher tensile strength steel.

So just from a kind of engineering judgment, the top of our containment vessel is stronger than the existing missile shields, plus we're not protecting a thin liner. So the situation is considerably different.

MEMBER RICCARDELLA: I understand that,

and I don't have a problem with that. You know, I'm an ASME code person and done a lot of analyses of pipe breaks and leak before break and all that stuff, and I just have sort of a philosophical problem with that sub-bullet there, okay.

To say, well because it's a class one vessel, then gross failure is not considered credible. That's, to me that's not consistent with the intent of what we've been doing in that area all along, which is the reactor vessel, you know, we postulate breaks of pipes. We don't postulate breaks of the vessel.

But to take a little pipe that sticks out of the vessel and say, well, failure of that is incredible because it's part of the vessel, that's really not consistent with that vessels don't, you know, pipes can break but vessels don't.

Now, in terms of the missile, I've seen the probability arguments, the P1 plus P2 plus P3, and I agree, you'll probably meet that ten to the minus seven probability very likely, and you know, there's a relatively low probability of failure of that little nozzle, coupled with the probability that if it does fail, it's going to go through the containment, the three inch thick containment. I don't have any problems with that.

I just have a philosophical problem with say, oh, it's a vessel, and therefore it's incredible that it can break when it's a little, tiny pipe sticking out of the vessel, okay?

MR. PRESSON: And that is part of where the other bullets came in for us as this is part of the vessel, and to kind of strengthen that discussion, you have the CRDM nozzles, which are integral parts of the vessel. And those nozzles are, you know, full penetration butt welds too, so it's.

MEMBER RICCARDELLA: Just like any other small nozzle on a BWR or PWR, you got a four-inch nozzle sticking out of a vessel, it's full penetration weld to vessel, and it's got a safe end that welds it to the pipe. But we assume a break of that safe end, okay. It's no different.

MEMBER BROWN: Fundamentally a little bit different from him. You're saying that the top of the containment vessel is one inch thick as opposed to a quarter inch --

PARTICIPANT: Three, two and half --

MEMBER BROWN: I don't know, you said, made some comment about what's in the existing plants, and this was three times thicker. You used, I thought you said a quarter of an inch, so.

MR. INFANGER: A quarter of an inch is the existing older plants, the liner. Our containment head is about five inches thick.

MEMBER BROWN: The containment head is five inches thick.

MR. INFANGER: Correct.

MEMBER BROWN: My only point was if the thing was ejected, is the characteristic that comes out, can it hit a point force like that, and does it hit with sufficient force to crack it. And at that, all these arguments are strictly engineering judgments that it wouldn't.

I'm not a mechanical engineer from the standpoint of stress analysis, so I can't, you know, I can't argue one way or the other. It's just that what does it look like.

Now, if the rod or whatever configuration in your mechanism had a stop in it somewhere that as it comes up to the nozzle and it's got a flange, and it comes and hits that stop, then I really fully accept the idea that, you know, you've got those ASME requirements for the nozzles and having the full penetration welds, etc. That you've got a blocking mechanism as it comes out, it's never really going to do it. That's a good judgment.

But if it's just a straight, you know, missile, like a spear, then you've got a different circumstance. Five inches is pretty darn thick, I agree.

MEMBER BALLINGER: To your point, Pete, there's, so CRDM nozzles are integral parts of the reactor vessel closure head forging. So they're not welded to the vessel, they're part of the forging, right. Did that make difference?

MEMBER RICCARDELLA: But there's a weld to the other end of that forging.

MEMBER BALLINGER: Yeah, but that's just like a normal PWR then.

MEMBER RICCARDELLA: Yeah, and that is considered a potential pipe break.

DR. CORRADINI: But you'd consider it in the LOCA analysis, which they do. What they consider that sort of break as one of the break spectrums in the LOCA analysis.

MEMBER RICCARDELLA: Yeah, but it's the mechanistic effects of it. There's two different things. It's considering the LOCA analysis, whether you can shut down the reactor. But then there's also a criteria for whether you got to consider the mechanistic effects of a pipe break. And that's why we do leak before break or pipe whip analyses, to address the mechanistic. I think that's GDC-4 maybe, it's a different GDC, GDC-4.

And, but I'm not complaining, I don't have a concern with the design. I just have a concern with saying philosophically, well, that's part of the vessel so we don't have to assume it'll break, okay.

MEMBER BALLINGER: We need to coin a new unit. The AEU, the age-of-the-earth unit, of which ten to the minus tenth is probably a hundred age-of-the-earth units.

MEMBER BROWN: All those numbers, they're all guesses in the first place. There's no data that substantiates a single one of those ten to the minus anythings.

MEMBER RICCARDELLA: Yeah, I mean, for a small pipe break, we've historically used ten to the minus third as the probability of a pipe rupture, okay, that's historically been what we use.

MEMBER MARCH-LEUBA: And that's why I asked them to evaluate the --

MEMBER RICCARDELLA: And I certainly agree that it's probably significantly lower than that in the case of this nozzle. And then to meet the ten to the minus seventh criteria, you already got to say is well then the probability of that control rod getting through the top head is less than ten to the minus fourth. And my judgment is you'll meet that, you know, it's not a problem, okay.

MEMBER BROWN: To me, this is less of an issue than the return to power, so.

MEMBER MARCH-LEUBA: Yeah, but I, sorry, I'll have the pleasure of writing the first draft of the Chapter 15 letter, so it is, this is my understanding, operating reactors analyze a missile from the rod ejection, is that correct?

MR. INFANGER: They typically put up a missile shield instead of doing the analysis.

MEMBER MARCH-LEUBA: You are free to tell your name to the.

MR. INFANGER: Paul Infanger, Licensing.

MEMBER MARCH-LEUBA: But the way they resolve it, operating reactors, by putting a shield that they have calculated that is sufficient to prevent it from damaging a very weak component of the containment, correct?

MR. INFANGER: Correct.

MEMBER MARCH-LEUBA: Our containment is much larger, much thicker and much stronger than that shield is. So you've done a scoping calculation in your head which is not documented, but the scoping calculation says that even if we went to do the whole analysis, it would not even be close to limits. So it's not necessary to perform it.

MR. INFANGER: That's correct.

MEMBER MARCH-LEUBA: So I'll write that in a paragraph, and then we'll have the -- I'll have the pleasure of getting your input during the letter review part.

MR. BRISTOL: So that concludes the, this first subsection of the topics today. And --

CHAIR KIRCHNER: So let me interject here. I think this is a good place to stop and take a break, because we're going into an extensive set of view graphs for the next section. So with that, let's take a break here and come back at 20 after ten. So we are, whatever we are, recessed.

(Whereupon, the above-entitled matter went off the record at 10:02 a.m. and resumed at 10:20 a.m.)

CHAIR KIRCHNER: Okay, we are back in session. And Matthew, I'll turn it back to you. I think we're at boron transport on the agenda.

MR. PRESSON: Yeah, that'd be correct. With that, I'll pass it over to Megan.

MS. McCLOSKEY: All right. In this session, we'll give a high level discussion of boron transport aspects in the NuScale power design, first focusing on ECCS cooling --

PARTICIPANT: Is your mic on?

MS. McCLOSKEY: Yes, is it -- okay. And hopefully the paper shuffling won't be too bad.

So first we'll talk about ECCS cooling and the boron transport analysis that we've done, followed by differences when you think about extended DHRS operation and the RCS conditions during DHRS, riser uncovery conditions and how that affects boron transport. And then we'll speak a little bit about beyond design basis conditions in ATWS.

And this is a high level summary, and we have a fair bit more material and details to discuss as part of the closed session.

So during long-term ECCS operation, the phenomena of interest are driven by boiling in the core and condensation in the containment vessel, which will tend to result in boron accumulating in the core and riser region and the boron concentration of fluid in the containment and the downcomer decreasing.

MEMBER MARCH-LEUBA: So in your cartoon over there, you don't, I don't see any steam transfer from one, region one to two and three. Is that meant to be?

MS. McCLOSKEY: The cartoon on the right was focused on realistically with a hot core in the center and cold, cool conditions on the outside, we would expect some degree of buoyancy-driven recirculation flows to develop in both the core riser and the downcomer in the containment regions that would tend to mix concentrations.

MEMBER MARCH-LEUBA: If DHRS happened to be working in the middle region, the downcomer, then that would be even more accentuated.

MS. McCLOSKEY: Then you could potentially get condensation on the steam generator tubes that would result in recirculation further in the downcomer.

MEMBER MARCH-LEUBA: So the cartoon doesn't include two additional lines from core to downcomer and core to containment, but you consider them.

MS. McCLOSKEY: Yes. So --

MEMBER BROWN: For the uninitiated, could you explain these three levels of circulation in terms of how you get diluted with -- you've got natural circulation going on. Why does the -- I didn't understand that from reading the stuff in the papers.

MS. McCLOSKEY: So --

MEMBER MARCH-LEUBA: Can you use the mouse?

MS. McCLOSKEY: Sure.

MEMBER BROWN: I mean, there's three regions. There's the reactor core region, there's the RPV, decay heat. I presume the purple line or whatever it is is the decay heat.

MS. McCLOSKEY: That's the downcomer region within the reactor vessel, and the steam generators are above here.

MEMBER BROWN: Oh, okay.

MS. McCLOSKEY: Cut off of the figure.

MEMBER BROWN: All right.

MS. McCLOSKEY: And this is the core in the riser region going up. And then this is the containment vessel and fluid in containment that's --

MEMBER BROWN: But the decay heat is -- decay heat system is where? It's inside the RPV, right?

MS. McCLOSKEY: The steam generators are up above and then the decay hear --

MEMBER MARCH-LEUBA: He's asking for the core.

MS. McCLOSKEY: The heat exchangers are outside of containment in the reactor pool.

MEMBER BROWN: They're, okay, all right. I've got the other pictures from your previous.

DR. CORRADINI: This is a cartoon that's not driven to scale and it's missing the top of the containment. But other than that it's right on --

MEMBER BROWN: Yeah, well that's where I've lost the bubble, okay.

MS. McCLOSKEY: Yeah. And so when we started discussing with the staff concerns about boron dilution analysis during ECCS cooling, initially some of the concerns were driven by non-LOCA events, non-LOCA type events, feedwater line breaks, reactor component cooling water line breaks that initially put an amount of dilute water into containment that would be there when the ECCS valves open.

And during the -- but considering the blowdown during ECCS value opening, we had evaluated at that point in time that that would tend to put borated water into containment prior to establishing recirculation.

And that initial blowdown would also tend to increase the concentration in the core and riser region due to flashing, vaporization, such that when you establish long, the very low flow rates that we have during long-term circulation, you would maintain a highly borated core and riser region.

MEMBER BROWN: That's good, isn't it?

MS. McCLOSKEY: Yes.

MEMBER BROWN: Okay. Bottom line, I'm sorry.

MS. McCLOSKEY: So we discussed that qualitatively, and then later quantitatively evaluated that boron transport analysis, and that's the boron dilution analysis that we'll talk about a fair bit today. Particularly to evaluate the potential for a lower boron concentration to accumulate and to -- to accumulate in fluid below the core region and then get into the core region. And would that cause a concern for a recriticality, from a recriticality perspective.

So ultimately we're doing this to confirm that we've done an appropriate scope in the return-to-power analysis that as Ben was talking about this morning, was focused on the late-in-operating-cycle conditions with essentially zero boron.

MEMBER MARCH-LEUBA: Say again, because boron dilution is a beginning-of-cycle issue, when you have 1000 ppm.

MS. McCLOSKEY: Yes.

MEMBER MARCH-LEUBA: Return to power is an end-of-cycle issue when you don't have any boron.

MS. McCLOSKEY: So we --

MEMBER MARCH-LEUBA: When I exaggerate, say do not have any boron I mean 100 ppm. Go ahead.

MS. McCLOSKEY: So we were confirming that the, if you had a beginning-of-cycle condition, you would tend to accumulate boron in the core to higher than the initial concentration.

MEMBER MARCH-LEUBA: During the boil-off period.

MS. McCLOSKEY: Due to the boil-off in flashing, and --

MEMBER MARCH-LEUBA: During the boiling stage, you obviously were concentrating the core and riser.

MS. McCLOSKEY: Right.

MEMBER MARCH-LEUBA: And depending on how much that circulation happens, it would be uniform or not.

MS. McCLOSKEY: Mm hm. So that we had an appropriate scope in the return-to-power analysis as it existed.

MEMBER MARCH-LEUBA: Yeah, this does not affect the return to power, because return to power only happens when you don't have any boron.

MS. McCLOSKEY: Right. So the concern, what I'm trying to say is the concern was if you had a beginning-of-cycle condition and you somehow accumulated a deep, a lower boron concentration at the core inlet or in the core region, that could create different return-to-power concerns that hadn't been analyzed. That was the concern from the boron dilution analysis.

MEMBER MARCH-LEUBA: My concern is that you accumulate low borated water in the downcomer, and then you reestablish flow.

MS. McCLOSKEY: Which, yeah. And we'll touch on that in this session and have further discussions in the closed session.

DR. CORRADINI: But just to help Charlie, though, I think the way Jose just spoke about it is the reason the concern arises.

MEMBER MARCH-LEUBA: No, there are many -- I said before, when you talk boron dilution, there are three topics, okay. And number one is you add, you have an inadvertent actuation of the CVCS system, so you're putting deborated water, and just slowly deborate the whole vessel by adding clean water. And that eventually we'll get to return to power if they don't notice, but it might take several hours and it will be of no consequence.

For the staff, during most of the review, their concern was boron plating. So that you, because you concentrate so much boron in the core, when it boils off it can -- like when you're boiling a pot of water in your stove, you see that salt accumulates. And that way when you recover our level, you would not have enough boron to reach critical.

My concern is there are mechanisms to accumulate deborated water in the purple region, in the downcomer, and then when you reestablish flow, that can go into the core. So those are the three I see, I'm sure there are five more.

MEMBER BROWN: Your purple, is that maroon-like, that's the center one?

MEMBER MARCH-LEUBA: I think it's fuchsia.

MEMBER BROWN: That's it.

MS. McCLOSKEY: Yeah.

MEMBER BROWN: That's it, that's your purple one, right? That's the purple.

MS. McCLOSKEY: This is the down, yeah, in the downcomer region.

PARTICIPANT: Fuchsia?

MEMBER MARCH-LEUBA: That's what computers call it.

MEMBER BROWN: I'm used to the old-time Crayolas, they only came in five colors in 1940.

MS. McCLOSKEY: So the first, of those three scenarios, the first is considered in the Chapter 15.4 typical boron dilution analysis that we discussed previously. The second scenario is what we're -- the first part of what we're focused on here. And then the third and done is according to Chapter 15 design basis considerations.

And the third part in terms of recovery from a specific event and recovery operations, what we'll talk about today is focused on the design capabilities to be able to safely recover the reactor, and how that will be a, that takes a number of steps if you're in ECCS cooling operation or extended DHRS operation for the operators in order to un-isolate containment and then take steps to eventually reestablish level in the RCS and move back into tech spec conditions.

So we can talk about the design capabilities today, but the specific procedures for that are developed at a later date.

MEMBER MARCH-LEUBA: In previous meetings I used the analogy of the toothpaste getting out of the tube. Once you separate it, it's very difficult to put it back together. I go through the slides and you figure out a way to put the toothpaste back in the tube. So that's very good that we're attempting to do it, but I'll wait for you to describe it.

MS. McCLOSKEY: Okay. So if we go on to the next slide. We're talking -- this slide describes the methodology for the boron dilution analysis that we've performed. We started with the long-term PIRT and the high ranked phenomena in that PIRT that affected boron transport, and we evaluated those phenomena and how they could be addressed to conservatively evaluate dilution in the core.

We performed a control volume approach to analyze transport between the regions. One of the inputs to that control volume approach were fluid masses and flow rates from NRELAP5 to calculate transport of liquid and vapor between different regions in the reactor vessel and containment. But the actual boron transport is all calculated separately from that outside of the NRELAP5 code.

We also calculated volatility effects and entrainment separately because that's not included in the code package.

MEMBER MARCH-LEUBA: So you have the word volatility during a meeting a couple weeks we were talking about how much boron dilutes in the steam phase. Do you have a reviewed, I mean, we have one Russian paper -- two, actually two, that says that, actually the number, it gets stuck to my face, atmospheric pressure is 0.14 percent of the boron transfers to the steam, and it's proportional to pressures.

When you're in these conditions at these temperatures, you're roughly at 100 psi, so you would transfer like one percent of the boron to the same. Is that your numbers?

MS. McCLOSKEY: Roughly speaking. It is a function of temperature and the correlation that we applied, the bulky correlation, is a function of a couple of other factors. And I've got, we've got more detail about that to discuss in the closed session, including comparison of the volatility that we've calculated as a function of temperature for the NPM conditions compared to a number of other data sets that are available to demonstrate that we have a reasonable calculation of volatility.

And one of the key conservatisms that we have is we assume that all of that volatilized boron is lost from the problem.

MEMBER MARCH-LEUBA: You just dump it out.

MS. McCLOSKEY: It's just removed. Whereas, more realistically you would expect that this is when the steam condenses in containment, that boron would return to solution and continue to be available to recirculate.

MEMBER PETTI: Is, I mean, I don't want to get into the closed session, but as I remember reading Chapter 15, you have experimental data that's been, this has been looked at experimentally, correct?

MS. McCLOSKEY: Yes.

MEMBER PETTI: That I was at least unaware of. So I'm excited to hear about it, because there's actual real data that I think will help us think this all through.

MS. McCLOSKEY: Next, one of those bullets got a little bit jumbled, it looks like. We conservatively model transport of boron between the regions. And the premise, and so one of the concerns is how much mixing do you get in these different regions of the reactor pressure vessel and the containment as boron is transported there. Within the boron dilution methodology --

MEMBER BLEY: I'm sorry, I was just thinking. I can think of various ways that could play out, and it's not transparent to me what are the conservative assumptions on transport, because it can get, if you get a lot in the core, it can play out as you were talking, and get a lot somewhere else, may come back. So what makes it conservative?

MS. McCLOSKEY: So from the boron dilution perspective, the main conservatisms that we applied in the transport are to, are focused on minimizing the amount of boron that's available to transport into the lower plenum core riser region into the hot region of the vessel, where it can accumulate. And also maximizing the transport of boron out of that.

So maximizing it is on the volatility side, minimizing it is a combination of assumptions that throws away a lot of the boron in the system that's available to transport to minimize how much can come in and the rate at which it comes back in. And conservatively treating the concentration that's transporting into the core. And that'll be -- we have a -- we'll go into that in more detail in the closed session.

MR. BRISTOL: And I think I'd add we look at both sides of that problem.

MS. McCLOSKEY: Yeah.

MR. BRISTOL: So we look at the conditions where boron is initially maximized in the core and its effect on things like volatility played out considerations. And we also look at long-term very slow transient progressions where we're minimizing the boron returned to the core.

MEMBER BLEY: When both of you speak of volatility, does that include if there is boiling, your transport and the moisture that gets carried out as well?

MS. McCLOSKEY: We --

MEMBER BLEY: So volatility is not just becoming vapor, it's being transported as droplets?

MS. McCLOSKEY: We considered entrainment of droplets. That turns out to be a fairly small factor in our analysis because we've got such a tall riser, so there's very little droplet entrainment that could ---

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: Basically we read, I mean volatility is not probably the right name. It's solubility of boric acid in steam, it's not creating -- because at room temperature you can boil off boric acid at 300 degree C and it becomes a gas. But it remains dissolved in the steam phase. It's not boiling it off.

MS. McCLOSKEY: So I think that's touched on some of the key areas in the review that we'll talk about in more detail in the closed session.

I'll also say that as part of preparing for this meeting and following up on some of the discussions last week, we have opened a condition report to evaluate the -- and scrutinize the scope of the scenarios that were considered as part of the boron dilution analysis and to consider the design capabilities and the expected progression of events.

MEMBER MARCH-LEUBA: Would you describe what the condition report, the CR you have in there, what does it entail, and what documentation will we get to see out of it?

MS. McCLOSKEY: That's part of our internal corrective action program. It's the initial report that identifies there could be a condition here that needs further analysis and scrutiny and it's part of our QA program.

MEMBER MARCH-LEUBA: If you run through the CR and determine that there was no error, there won't be any output out of Corvallis. We'll never see anything. We only see a we evaluated it, nothing happened.

MS. McCLOSKEY: The, yeah, I think it

depends on what the Committee wishes to see and any follow-up there. The information is available to the NRC staff as part of audits and inspections.

MEMBER MARCH-LEUBA: The staff can always invite themselves to do an audit. The Committee does not have that prerogative.

MS. McCLOSKEY: Right.

MEMBER REMPE: What is the schedule? I mean, how long does this take? How does it affect your DCA? I don't fully understand what a Condition Report is.

MS. McCLOSKEY: We are working expeditiously on this to evaluate if there's a concern and if there are changes to the DCA that are needed. So, schedule-wise, it's dependent. It varies with the severity of the condition that's identified and the schedule of other -- other scheduling strengths.

MEMBER MARCH-LEUBA: It's not out of the scope of possibility that it may affect the July time that we're working on?

MEMBER BLEY: Or does it turn into a COL condition?

MR. MELTON: Right. So, if I may, Mike Melton, Manager of Licensing.

We've identified this in the CR. We're

working through the process expeditiously, as stated. We'll work closely with the staff to minimize any impact on the DCA schedule. We are finalizing any DCA changes in the errata letter we're preparing. But the whole schedule impact we expect to be very minimal, if anything, but we need to work closely with the staff. There may be some discussion on this at the next full Committee meeting in April, depending on how quick things go. So, that's the process we're working through.

MEMBER MARCH-LEUBA: But anything in writing that we can look at?

MR. MELTON: At this moment, no. As you pointed out, the staff can see our material. That's probably the preferred path of the briefing at the moment, but we need more days this week.

MEMBER MARCH-LEUBA: You need to finish your evaluation first?

MR. MELTON: We need to finish, exactly. Exactly. Then, I think there will be a briefing in order.

MEMBER MARCH-LEUBA: You understand that at least some of the members at this table are very interested in this topic?

MR. MELTON: We get that and we appreciate that.

MEMBER MARCH-LEUBA: Okay.

DR. CORRADINI: So, in terms of documentation, the RAI 8930, the three different versions of it as they've progressed, is the documentation that I have been looking at. Is that the wrong place to look? That's the only place I'm aware to look at this.

MS. McCLOSKEY: That's the correct place.

DR. CORRADINI: Okay. And what we'll see later in closed session is potentially updates to that, as essentially a bounding set of analysis. That's my interpretation of what in closed session we're going to talk about.

MR. BRISTOL: Yes. So, the details of the assumptions, the methodology, conservatisms that go into that analysis, where we stand today is that analysis is valid.

DR. CORRADINI: Okay.

MR. BRISTOL: If we were, through this evaluation process, if we were to find that's incorrect, obviously, we would work through the process to update it, and then, that would go through the staff for review, and then, depending on the process, come back to the Committee.

MEMBER MARCH-LEUBA: So, if you find anything significant, the time will slip, almost guaranteed, right? So, the only way we're sticking with this timing is if you don't find anything significant?

MEMBER REMPE: Well, we could do a final letter and discuss this was discussed and that they have a Condition Report, but that's not a very good final letter for this.

MEMBER MARCH-LEUBA: I'm not worried about the letter. I'm worried about the SER.

MEMBER REMPE: Yes, well, the staff SER, but we're still -- what I'm thinking about is our letter and what we're reviewing and what's going to be in it. So, we could work fast. I mean, it could affect what we write.

MR. PRESSON: Yes, and given the combination of the ACRS schedule, NRC's schedule, and our schedule, we are definitely working on it.

DR. SCHULTZ: Ben, in terms of focus, you implied that March is the timeframe for this CR. In other words, you've got folks that are working on it and you hope to have an answer by the 1st of April.

MEMBER RICCARDELLA: But it --

DR. SCHULTZ: I know, no, the month of

March, that's what I heard.

MEMBER RICCARDELLA: But, at least at this stage, we can evaluate the analyses they plan to do, right, and the validity of those analyses?

MR. BRISTOL: That's correct.

MR. PRESSON: And that will be the expected timeframe, yes.

MS. McCLOSKEY: So, for the analysis that we have completed, we've evaluated boron transport during ECCS cooling, and the results that are summarized in that third version of the 8930 response show that the core boron concentration remains above the initial concentration throughout the long-term ECCS operation. So, that indicates there's no net boron dilution expected in the core region, even with the bias transport assumptions that we have.

We also did a more realistic analysis of boron transport with slightly less conservative assumptions. And that indicates that the accumulation in the core region is two to three times the initial concentration after three days and remains above the initial concentration for at least seven days, which ultimately supports our conclusions that systems such as CVCS or the containment flood and drain system are not needed to increase the boron concentration in the core region for at least seven days following an event.

So, that kind of bounds the scope of the Chapter 15 analysis. If we, then, move to thinking about recovering the actual plant operation and recovering from these types of conditions, you had an event. You had an engineered safety system actuation. And by that nature, you're outside of your tech spec conditions. You're not in normal operating mode.

Actions to recover the riser and reestablish and demonstrate that you're in tech spec conditions and reestablish Mode 3, safe shutdown conditions, take multiple deliberate operator actions following appropriate procedures.

DR. CORRADINI: So, I want to make sure because this is something we've worried about, and now you're stating it. Remind me what Mode 3 is?

MS. McCLOSKEY: It's cold shutdown conditions and -- safe shutdown. Safe shutdown, k-effective less than .9, .99.

DR. CORRADINI: And temperature?

MS. McCLOSKEY: Less than 420 degrees.

DR. CORRADINI: If you're allowed to say it?

MS. McCLOSKEY: It's in the tech spec,

modes.

DR. CORRADINI: Okay, fine. I'm sorry. The reason I was asking is that this, to me, is kind of the most important bullet here, which is your basically recovery, and the recovery has to take account of concerns that Jose has raised in the past meetings, so that you don't get there inappropriately.

MS. McCLOSKEY: Uh-hum.

MEMBER MARCH-LEUBA: Correct, and I believe it's closed session when we're going to see all those little "n's" and how you plan to do it.

MS. McCLOSKEY: Uh-hum.

MEMBER MARCH-LEUBA: There are two comments on this. First, is this going to reflect on the Chapter 19 risk analysis? Because we always say that there is no operator error of omission or commission that can possibly cause any problems. And we will talk about this on Wednesday and I will drill the person that will be sitting there instead of you, so you don't have to answer now. But, clearly, if we have some procedures that must be followed, otherwise we may have problems, the probability can be very low, but there will be errors of omission or commission that the operator can take.

MS. McCLOSKEY: And there is a COL item

that makes the commitment that the PRA will be updated to account for the as-built, as-operated plant and the procedures associated with that.

MEMBER MARCH-LEUBA: Yes, but, right now, we're taking clarity for risk-informed decisions that are based on a PRA -- my ongoing favorite topic -- which is incomplete. It doesn't account for errors of the operator that doesn't establish Mode 3 correctly. The probability of an operator failing is always 10 to the minus 4.

MEMBER BLEY: Or greater, I should say.

MEMBER BROWN: Or better.

(Laughter.)

MR. BRISTOL: So, we will get into to the ATWS conditions a little bit later in the presentation. So, the reason that we had opted to have most of this discussion in closed session is it's speculatory in nature in terms of what the procedures would be, right? We can't sit here and say exactly what specific procedures are going to be because they're not developed yet. However, we can describe the system capabilities and the system response, and I think we can help illuminate why we wouldn't consider a significant increase in risk based on recovery actions and their possibilities.

MEMBER BROWN: Can I make sure I understand? I thought all ECCS was no requirement -- I'm just echoing what Jose said -- no operator actions. Yet, deliberate operator actions are required following appropriate procedures. Is Mode 3 post-72-hours or something or is that within the -- I thought the no operator actions calibrate need was within the first 72 hours. Is that correct?

MR. BRISTOL: Yes, that's correct.

MEMBER BROWN: Based on other discussions, we could walk away for a week and nobody would care.

MR. BRISTOL: That's correct. After an event, though, we will immediately enter into, if we have a safety actuation function or an active safety signal, we're in LCO space. In order to exit LCO space back into normal operating mode, specific recovery actions are going to be --

MEMBER BROWN: Mode 3 is a normal operating mode?

MR. BRISTOL: That's right.

MEMBER BROWN: Oh, okay.

MR. BRISTOL: That's right. That's right.

MEMBER BROWN: I lost the --

MEMBER MARCH-LEUBA: Shutdown.

MEMBER BROWN: Safe shutdown? Okay.

MEMBER BLEY: Let's not get confused between the real world and the Chapter 15 world. In the real world, all those non-safety systems are working. All the operators are following their procedures, well, they're trying to follow their procedures.

MEMBER BROWN: Yes, I got that.

MEMBER BLEY: They're doing something.

MEMBER BROWN: Yes, I understand that. It's just that the concept, from my understanding, was this plant was a walkaway plant under the accident-type conditions.

MEMBER BLEY: For a Chapter 15.

MEMBER BROWN: For a Chapter 15 accident condition.

MEMBER BLEY: I think that's right, yes.

MEMBER BROWN: And now, all of a sudden, I'm starting to get the feel that it's not okay as a walkaway under Chapter 15 conditions. There are specific requirements for operator actions in order to maintain a safe plant after a reactor, a Chapter 15 --

DR. CORRADINI: I don't take it that way. In the current plants, and some of the past --

MEMBER BROWN: I'm looking at this plant.

DR. CORRADINI: I understand that, but

this is no different than what's post-72-hours in other design certifications we've done. This is no different.

MEMBER RICCARDELLA: You can walk away and it's safe, but you don't want to write off the plant necessarily.

MEMBER BROWN: Well, I understand that. I understand it. I'm just talking about what it's stated to be. I mean, I've got the return-to-power issue, and then, I've got the ECCS issues, and all this stuff. But now, I have to have operator actions. It's just sounds like we've turned to the power isn't even you can walk away. You've got to have operator actions if you're under the Chapter 15 circumstances.

MEMBER MARCH-LEUBA: I think now is the time for me to use a joke. I was reserving it for Jeff's presentation.

(Laughter.)

MEMBER BROWN: Is that this afternoon or tomorrow?

MEMBER MARCH-LEUBA: It will be this afternoon at the rate we're going.

MEMBER BROWN: Is that a closed session or --

MEMBER MARCH-LEUBA: No, it's an open

session. But let me give the analogy --

MEMBER BROWN: All right, I'll quit.

MEMBER MARCH-LEUBA: -- the analogy I'm having. And I'm ecstatic. I'm jumping up and down with joy that you guys are having this Condition Report and you're evaluating this condition.

Because the way we were December 12th, 2019, reminds me of the movie "Thelma and Louise".

(Laughter.)

Okay? Where, at the end of the movie, they are peacefully driving on a country road with a simple song and laughing, and that's where the movie ends. And we rely on the COL applicant to be the sequel to that movie and realize that there is a canyon behind them and they're going to jump into it.

(Laughter.)

Because the way we were on December 12th, 2019, we were on the country road driving peacefully and saying everything is safe and sound and everything looks good. And nobody in the SER, and anywhere I can find documented, was telling the COL applicant, "Watch out, there's a canyon coming." Now we have it. Now we're looking at it. So, we go leaps and bounds across the --

CHAIR KIRCHNER: I took the movie

differently than you did, I think.

(Laughter.)

I think that was an intentional ending.

MEMBER REMPE: Just with your comment about the procedures, remember the letter that Dennis or the memorandum that Dennis wrote last month about Chapters 13 and 18, and commenting about the procedures for the operators and some sentences he recommended for the final letter. And I don't recall exactly the words he had, but it seemed like he was emphasizing the crane and some other things. But maybe that discussion should be enhanced with what comes out of your letter or something before we're done with this. Because I think we've identified that there's a lot that's been cut off to the COL applicant with respect to the procedures, and this might be the place to --

MEMBER MARCH-LEUBA: We are in positive territory now.

MEMBER REMPE: Yes.

MEMBER MARCH-LEUBA: I mean, the way we were December 12th, 2019, we didn't know there was a canyon ahead. The COL applicant, where he was going to write the procedures, had not realized that you could have this boron distribution if you restart it.

If you have and you avert the CVCS actuation when you're in that condition, or if you start --

CHAIR KIRCHNER: Or CVCS without boron.

MEMBER MARCH-LEUBA: Yes, or things like that, I mean, maybe you had it in your mind, but it was not written anywhere. And now, we are, now we are there.

MS. McCLOSKEY: I think that I would add to that, though, that in the 1506 discussion of the FSAR, it does have an explicit discussion that boron distribution is expected, possible, expected during ECCS operation.

MEMBER MARCH-LEUBA: Under long-term cooling --

MS. McCLOSKEY: Under long-term cooling conditions.

MEMBER MARCH-LEUBA: -- the technical report says explicitly we have not identified any possible way to put that cold water into the core. It says that in bold, underlined letters. Now we have.

And it may be a difficult way to get there, and as I was telling Matt earlier, if you were trying to tell me that you were going to put the cold water into the core in an experiment, I would laugh at you and say, "You cannot possibly do it." But, the say way that I'm laughing at you, that you cannot possibly tell me it can happen without an analysis, and that's what you're looking at. And we're going in the positive direction. We've improved --

MR. PRESSON: Yes, and while I definitely appreciate the joke, just to put this slide into context --

MEMBER MARCH-LEUBA: It's not a joke. It's an allegory.

MR. PRESSON: Yes, the allegory, the metaphor. This slide does kind of bridge a bit between where we are in DCA scope and where we are looking to head towards in that COL applicant scope. You know, there are a number of COL items. There is plenty of work that is going to happen between now and us support a COL applicant. So, none of us are under the impression, us or the NRC, that we are done, but we do want to during the DCA establish that the design that we can come up with at least has a reasonable basis for being safe.

And this slide, kind of to your point, still maintains the Chapter 15 basis of you don't need operator actions within that 72 hours, but it is starting to look into that COL applicant space, or even an operating COL, and say what should be looked at as you develop procedures and what things do you need to consider once you do start taking operator actions.

You know, the Chapter 15 safety analysis, we're still good for that space. When you exit that space, when you're looking to get control of the reactor and the event, bring it all back into tech specs, what are some of the considerations you have to bear in mind? So, it's definitely a blended slide, but it's looking to, like you say, bridge that gap.

MEMBER SUNSERI: So, let me ask or comment here, one or the other. I think it seems to me to be a fairly normal situation to be at where a design certification, based on its design, has some endpoints that an operator has to consider when they write their procedures, but, clearly, okay to defer that to the people that will actually write the procedure later.

To tie it back into the PRA, that's why it's a two-step PRA. We've got the part now that addresses the design features and any feedback into the design that should be incorporated to certify a design, but, then, later, at fuel load, that's the more comprehensive one that incorporates all the operator actions, where we would check for the operator commission and omission perhaps impact on risk. Am I understanding it right? I mean, that seems to be the normal process that I've experienced in past DCAs.

MR. PRESSON: Yes, that's correct.

MEMBER SUNSERI: And we're not departing from that kind of, I'll call it, philosophy or practice with what you're doing here, right?

MR. PRESSON: No, we are not. We are simply presenting on how one might be able to bridge that gap.

MEMBER SUNSERI: Okay. I just wanted to make sure I was following the flow here. Thank you.

MR. PRESSON: Yes.

DR. SCHULTZ: That's bridging the gap in terms of process. You don't really expect design-specific features to affect these procedures. You'll work with the licensees and they're all expected to have the same procedures to react to this event.

MR. PRESSON: They are all expected to have fairly similar sets of procedures. I would have to talk to Ops to see if we are looking to enforce any mandatory procedures. But since you have owners and --

DR. SCHULTZ: Consistency is good.

MR. PRESSON: Yes.

DR. SCHULTZ: Just remember that as you go forward.

MR. PRESSON: We are looking to provide consistency across those procedures, yes.

DR. SCHULTZ: That's also the responsibility of the designer, sure. Thank you.

MR. RAD: This is Zachary Rad, NuScale Director of Regulatory Affairs. If I could just add one comment?

So, I want to make sure that it's clear that, when we talk about not requiring operator actions, that is within the context of the regulatory framework established for the Chapter 15 Design Basis Analysis. We expect to have emergency operating procedures and we expect to have operators, but that regulatory framework establishes requirements for things like control room instrumentation and controls and the classification of those items. And so, that's one of the important factors to consider when you determine what operator actions are required.

What you want to have in your design is a design that doesn't require operators to respond. That doesn't mean that you are going to have a walkaway design, as it has been stated. We intend to have operators and we intend to have operating procedures that will address emergency in the same context as the existing legacy fleet. They just won't have the same requirements established for them in order to address those accidents, if that makes sense. They won't have the same need.

So, we'll have to be careful about what we ask our operators to do because acts of commission in this case are really the only pathway to creating errors in the control room.

Thank you.

MEMBER MARCH-LEUBA: Since you guys are working on the CR, can we go back to slide 20? Let me give you a couple more things to consider.

No. 1, when you're in this situation, your CR needs to consider the built-in actuation of CVCS will establish the circulation and will slide the fuchsia water into the core. So, when you do the evaluation, you need to figure out what you're going to do about that. That's one.

No. 2, if you remember the small break LOCA, the water levels for red and fuchsia -- and that's the core and the downcomer -- are very close to the top of active fuel because the ECCS valves haven't opened yet. And all that water that is not there is in the containment. So, the containment level is really high. Eventually, you reach the right pressure and the IAV opens, and you open the floodgate into the downcomer, and that pushes the fuchsia into the red.

So, your Condition Report needs to evaluate those and 100 others. But, certainly, there's more break with a delayed ECCS valve opening. It will push the downcomer water into the core.

CHAIR KIRCHNER: And replenish it with fresh water, de-borated water from the containment.

MEMBER MARCH-LEUBA: It has been --

CHAIR KIRCHNER: If it's a small break LOCA.

MEMBER MARCH-LEUBA: A small break LOCA, yes. So, it is possibility when you have that high -- and we're talking 15 feet of water difference. I mean, it's not a couple of inches.

MS. McCLOSKEY: It's not 15 feet, but it's --

MEMBER MARCH-LEUBA: When the staff audits it and we eventually will look at it -- I mean, we can guarantee we'll look at it -- I'll be looking for that. And the fact that you have already another 25 --

MS. McCLOSKEY: Thank you. We understand the comment.

Slide 23. So, one of the other areas of interest has been extended DHRS operation, riser uncovery conditions. So, what this slide does is gives a comparison between ECCS and DHRS conditions. Whereas, an ECCS cooling is established by the boiling and condensing mode, during DHRS operation you only get to and sustain riser uncovery due to cooling if you have significant convective heat transfer through the riser wall. And during that, your RCS levels are significantly higher than they are during ECCS operation because you've maintained all of the fluid inside the reactor pressure vessel. And so, in terms of a condensation potential, you've only got the top of the steam generator region and that's really going to be limited by the vapor heat transfer on the secondary side inside the steam generator tubes in terms of how much condensation can you get.

MEMBER MARCH-LEUBA: Have you performed this analysis with a code?

MR. MELTON: With what?

MEMBER MARCH-LEUBA: With a code, with a validated analysis tool, instead of my gut feeling. But what happens, if you don't condense sufficient steam on the uncovered steam generator, what you do is you raise the pressure of that steam. I mean, you could have coolant in the steam and it will raise pressure, and you will start condensing more mass with the same surface.

So, I mean, these things are not easy to talk out of unless you do the analysis.

MS. McCLOSKEY: And in the closed session, we'll present results from a range of sensitivity calculations that we've done and looking at the combination of heat transfer through the riser and flow over the top of the riser that's needed to sustain DHRS cooling in the plant design, which ultimately results in really minimal drivers to redistribute boron in the RCS compared to your ECCS operating conditions.

But, regardless of that, if you have a DHRS operation and, then, extended operation and you reach conditions of riser uncovery, the plant is in a state where the levels below the top of the pressurizer, which actuates containment isolation, recovering from that condition and getting back into your tech spec conditions will take, again, multiple deliberate operator actions, following the appropriate procedures at that time to assure that you can -- to recover the module, confirming that you have met your tech spec shutdown margin requirements.

MEMBER BLEY: I know you've said it several times already, but would you summarize what Mode 3 means exactly?

MS. McCLOSKEY: Mode 3 means safe shutdown conditions, k-effective less than .99, and temperatures below 420 degrees Fahrenheit.

MEMBER BLEY: 420?

MR. BRISTOL: So, one of the things that in developing the material that we'll be going through today, and then, later this afternoon in the closed session, we wanted to provide a little bit more context of the -- oh, thank you. Hello?

So, in the development of the material for this presentation, we wanted to provide additional context to both the design basis consideration and beyond design basis considerations, particularly related to the phenomena at play for DHR cooling versus ECCS cooling, and then, how that relates back to the analyses that we've performed, both about boron distribution as well as the return-to-power and shutdown safety analysis condition.

So, I've got a couple of slides here that kind of touch on the ATWS considerations. Just from a frequency perspective -- and I'm a safety analysis, not a PRA analyst, so I'll do my best to cover this --

but the postulation of an ATWS is typically related to common cause failure modes in I&C and related systems associated with reactor trip breakers and getting the rods, the CRDMs, de-energized such that they would fall.

Because of that, it's typically considered from the short-term transient response. So, the context there being, in our PRA-supporting thermal-hydraulic analysis we apply a similar coping period between the design basis event, Chapter 15, and the analyses that they support. In general, the long-term cooling effects of a post-ATWS, because of the overall likelihood and the relative simplicity of inserting a fair amount of reactivity just be de-energizing the CRDMs, provides a ton or -- we don't believe that there's necessarily a tremendous amount of risk insight that is derived from the thermal-hydraulic results as a result of the common cause failure of 16 CRAs to insert. And that's primarily based on the likelihood of that condition existing.

MEMBER MARCH-LEUBA: Okay. But let's go back to the Thelma and Louise analogy. Operating reactors have become very safe, in my opinion, because of the emergency operating procedures, that they are symptom-based. And the way that the guy that was writing them described to me is, I'm a pilot. The initial thing is you keep the shiny side up on the plane and you're doing good.

(Laughter.)

And I don't care how you got there. So, eventually, you're going to develop emergency operating procedures, and it has to be there that, if you uncover the riser, via ATWS, via LOCA, via DHRS actuation, you go into the Mode 3 condition, and that's how you recover. And that takes care of it whether ATWS happens or doesn't happen.

So, there is a good regulatory argument to get those guys' signatures, but, in reality, the safety argument to this ACRS member is it's not going to happen, but if it happens, I have an EOP that will take care of it.

MR. BRISTOL: That's right. That's the eventual summary of this condition.

MEMBER MARCH-LEUBA: Because I don't want to be arguing with you about what the MTC of a unit cycle is or your ATWS calculation that the PRA guys keep quoting. Where does it exist? I haven't seen it.

If you tell me that, at least from the point of view of boron dilution, if you have an ATWS and you uncover the core, the riser, that would be the worst possible scenario because you have the highest power and you're boiling out of steam. So, you could tell me it has a core. Whenever I finally control my ATWS, I will sample my downcomer and I will make sure that I don't just start without the circulation.

MR. BRISTOL: That's right, absolutely. And the commitment to ensuring that k-effective is below .99 is the regulatory hook to ensure that, in order to exit from the LCO into Mode 3, shutdown margin is the first consideration.

MEMBER MARCH-LEUBA: But that only happens if you know the canyon is in front of you, and we didn't. Or maybe we did, but we didn't recommend it.

MR. BRISTOL: Sure. So, back to the context, within the ATWS as it's described in the PRA, it is basically considering none of the rods to insert. The numbers that go into the likelihood of the ATWS consider something more conservative than that. So, the frequency is based on the combination of three stuck rods. However, the thermal-hydraulic analysis performed with 16, all of the rods out.

And the next slide gets into a little bit of the summary conclusions. We have a few more details in the closed session. But what I was trying to do is bridge that gap. In Chapter 15, we consider one stuck rod. Reality is there is some frequency of two or three, up to 16, and there's a domain of thermal-hydraulic responses that range from what we have in Chapter 15 in the analysis there, and then, what the PRA T-H runs look like, which is all of the rods being stuck for an extended period.

MEMBER BLEY: I agree with what's analyzed. Some of your characterization of the ATWS condition in the PRA, you know, you talk about common cause failures in that whole system. The only cases where we've actually, in reactors of this type, where we've actually had an ATWS occur happened because two reactor trip breakers failed to open, and it looked like they had because they had the signal to open. So, that's a confusing factor for the operators and took a little time to get out of it.

I'm only suggesting maybe it's not as unlikely as your words seem to suggest; that's all. The three stuck rods, that depends on a lot of factors that we don't know anything about for your reactor right now.

MR. BRISTOL: Certainly. So, some of the considerations there, the module protection system design and the redundancy within that. Obviously, the breaker design is important in terms of and as a contributor to that likelihood.

MEMBER BLEY: And at least for us, we haven't seen that yet.

MR. BRISTOL: Really, the point that I'm getting at in the previous slide there is that the long-term -- so, boron redistribution and its concern is really a long-term effect. So, the combination of applying multiple stuck rods that are most likely due to the fact that the CRDMs didn't energize, for whatever reason, looking at that for the long-term effects is even less insightful from a risk perspective because of its overall probability.

The short-term ATWS response is analyzed, and that's considered from, are there enough feedback mechanisms to protect the reactor coolant pressure boundary in the secondary side, right? That's the short-term response, at which point you would expect operators to go de-energize the rods. They do have indication of where the rod position is, so they do have insights as to if the rods have fallen or not, if they didn't expect that. When we get into the longer-term effects in boron redistribution, I think it becomes even less significant as an overall contributor to risk.

MEMBER BLEY: Agree.

MR. BRISTOL: Okay. So, the thermal-hydraulic response, the short-term response for LOCA and IORV-type events where we go immediately to ECCS generally we see a fair amount of depressurization and avoid feedback effects that result in an immediate subcriticality, and as part of that, the riser uncovers relatively quickly. We enter into ECCS cooling mode.

If we're at beginning-of-cycle conditions, the boron accumulation effect ends up mitigating the return. So, for an ATWS condition, we would have a short-term response, some feedback effects that cause the power to drop or go subcritical, and then, that would be sustained under the beginning-of-cycle conditions. Under end-of-cycle conditions, the combination of the xenon effect and the balance of the moderator void feedback and temperature distributions would result in some equilibrium power level based on ECCS's cooling capabilities.

For the non-LOCA event response, the short-term response at beginning of cycle is more severe because the moderator feedback is less dominant. It is still negative and it's sufficient to eventually increase the RCS temperature until the power level is reduced and it matches the DHR heat removal capabilities.

MEMBER MARCH-LEUBA: At some point, I would like somebody, even if it's offline, to explain to me why the figures in Chapter 4 of MTC versus boron concentration are so wrong. Because they're not consistent with what you're saying there.

MR. BRISTOL: Sure, and --

MEMBER MARCH-LEUBA: And once you told me that they were very conservative, but I don't see what conservative is. And you don't have to do it now, but --

MR. BRISTOL: Okay.

MEMBER MARCH-LEUBA: Our tech specs say that the MTC can be zero. I mean, it's going to be positive sometimes at low power. But, then, for ATWS we're saying, oh, it's very negative. I'd like to see those calculations. I mean, because I suspect the figure in Chapter 4 comes from a simulated calculation, which is the wrong standard.

MR. BRISTOL: So, in the establishment of the core design limits, we use specific state points as opposed to the overall characterization of reactivity as a function of operating, you know, the entire operating domain. So, they're looking to ensure that the core limits are met and a simple a summary of what those can be.

And then, in design basis space we often use those limits and just unphysically postulate that they could occur through the entire temperature range, through the entire power range. And so, that's the inconsistency between what we would characterize as the core operating limit and the practical operating domain of the actual plants. When we're at hot conditions, it is impossible to add boron, to the point, and be critical at full power, to have a positive MTC.

MEMBER MARCH-LEUBA: Yes. No, no. Full positive, no, but very close to zero, yes. According to your SIMUL8 calculations, at about 1200 ppm of boron, if you look at figure 4.x -- I don't have it in front of me -- it tells you that MTC is very, very low. "Low" I believe to mean very close to zero.

MR. BRISTOL: And maybe we can get a little bit more into those details in the closed session. This topic will come up again, and we can discuss the specifics.

I think one consideration is, in simplifying the amount of analysis that's used in the core design, we often take the hot zero power MCT condition and call that the most positive limit or the least negative limit. And then, we take the end-of-cycle --

MEMBER MARCH-LEUBA: MTC --

MR. BRISTOL: -- MTC, and that's the most negative limits. And those are, again, what's used in the Chapter 15 analysis domain to ensure that we've covered the entire allowable spectrum of the core design. But it's essentially taking all of the power and xenon defect that is really in the design and replacing it with boron.

MEMBER MARCH-LEUBA: I'm concerned about the BOC and MTC that you create in this analysis being too negative or for all possible conditions in your in the domain. But even if it was less negative, you will eventually uncover the riser, generate a lot of voids, and equilibrated the power generation of the DHRS capabilities.

So, I'm concerned, only concerned that we are telling the operators not to worry about events that may happen afterwards. The operators need to keep a skeptical -- we need to teach the operators that things go wrong. We keep telling them, "Don't worry about it. Whenever something happens, put it on DHRS and controlled cooling, remove the keys, and go home. Take the keys out of the ignition and go home because that's the safest thing you can do." That's what I've been told by the PRAs I've seen at that table, in that chair.

And what we need to be telling the operators is, yes, that's one possibility, but that's not what you need to do.

MR. BRISTOL: Certainly.

MEMBER MARCH-LEUBA: And the same with ATWS. Because especially with two or more units, when you have an event, and we will have an event that will move the core modules, they're going to go to the most troublesome one, and the other ones, forget about them. Because "I was told by my instructor that that's okay." Keep looking at it.

MR. BRISTOL: Just to build a little bit -- and I'll cover this in the longer-term response -- even in the shorter-term response, there is some 10-minute timeframe where the MTC reaches some combination of balance with the reactivity rate. As power drops, then the xenon build-in starts to become a factor. And so, that helps reduce the power level until it starts to burn back out again, which is a much longer-term effect that we talked about earlier today.

So, end of cycle, obviously, the MTC is much more negative and the equilibrium condition is reached much more quickly than at the beginning of cycle for events that are mitigated by DHR cooling.

MEMBER MARCH-LEUBA: So, you were there when we were doing the simulated ATWS transients, and I was looking at you and saying, "How can this possibly be?" Because the difference between BOC and EOC on that, the MTC, the difference was 25 percent between the two. I mean, it should be 700 percent. And when we're looking at the core for the Module 6, which was end of cycle, and module 9, which was BOC, it was nothing. I mean, it was a few degrees difference. So, there's something fishy with that simulation.

You want to say something, Matt?

MEMBER SUNSERI: Well, I was behind you a couple of sentences here.

(Laughter.)

But, earlier when we were talking about the operators, I think I'm hearing actually - let me say it differently. You have multiple audiences for various messages, right? So, one of your audiences you want to tell them, hey, you know, even in the worst-case situation our operators don't have to take much action, if any. I do not hear you telling the operators, though, you don't have to take any action.

You're going to have procedures. You're going to have abnormal operating procedures. You're going to have emergency operating procedures. There's going to be things when there's an indication of an ATWS or a coolant leak, or this or that. And the operators are going to be highly engaged, I think. Is that correct?

MR. BRISTOL: Yes.

So, the longer-term response, our calculations do not have as extreme MTC conditions simulated. They're more of a best estimate reactivity calculation, at which we do see that the BOC conditions reach an equilibrium while the level is still in the pressurizer. As xenon starts to decay, power will slowly begin to increase, and under those conditions, pressure starts increasing as well. Again, this is in the 36 hours and beyond kind of space. That will result in the pressurizer relief valve cycling several times.

That actually has -- it's a fairly small effect, but there's about 20 percent of the RCS volume in the pressurizer. So, there's quite a bit of boron that's up in that space. As the RCS depressurizes and vapor is generated, the boron will slowly begin to work its way back into the RCS loop, which helps suppress the overall event reaction.

End of cycle is similar, although the temperature effects, the boron accumulation wouldn't be considered. And so, it's just a combination of the feedback mechanisms between temperature and xenon.

So, the final conclusion, the overall results of the thermal-hydraulic runs supporting the PRA support the conclusion that operator action isn't required to mitigate the events, although certainly operators would be expected to do something relatively quickly well before the xenon build-in would be expected to be an important phenomena.

MEMBER MARCH-LEUBA: Xenon burnout?

MR. BRISTOL: Burnout.

MEMBER MARCH-LEUBA: Which would be in 24 to 36 hours?

MR. BRISTOL: That's right.

CHAIR KIRCHNER: So, in the first class of events, the LOCA events, is there a danger when that riser is uncovered, or even before it gets uncovered, of getting saturation conditions in the riser and going into kind of a menometer U-tube oscillation?

MR. BRISTOL: Under ECCS cooling

conditions.

CHAIR KIRCHNER: Yes.

MEMBER MARCH-LEUBA: For me, the low pressure is even worse. So, yes, I'm not --

CHAIR KIRCHNER: So, unlike the other Chapter 15 analyses, in this case with an ATWS, and if you go to an oscillation, if you hit saturation conditions in the riser, it strikes me there's the potential to go into an oscillatory mode in the core, where you could, then, if you get a cold slug of water in, you'll collapse the voids, get a power pulse, and then, that would reverse itself. Did that show up in any of your calculations with RELAP?

MR. BRISTOL: So, I can follow up with the guys that looked at those conditions.

CHAIR KIRCHNER: Because for normal operations --

MR. BRISTOL: Yes.

CHAIR KIRCHNER: -- you know you've got to cut -- you've got a limit. You've got a 5-degree -- you're looking for saturation conditions in the riser and you would trip the system, obviously, and scram the reactor if you exceeded that. But, with the ATWS conditions, it's very likely that that would occur.

MEMBER SUNSERI: But don't you also get an increase in pressure, though, with the ATWS which would mitigate?

MR. BRISTOL: Yes. So, the equilibrium conditions, I'm not as familiar with those calculations. Again, we can follow up in the closed session.

CHAIR KIRCHNER: Yes. All right. So, just save that for the closed session. Thank you.

MR. BRISTOL: So, next slide.

Okay. So, just as an overall summary of the topics we've discussed so far this morning, the inherent design characteristics provide ample safety for the module in the safety analysis, including the considerations of the shutdown capability as well as the boron redistribution. Again, that's the heat removal capabilities of the pool and the passive systems coupled with the low overall core power.

Compliance with the intent of the GDCs is demonstrated for the reactivity control systems. We discussed the Principal Design Criteria 27 and its analogy to GDC 27 as well as the analysis of the return to power and the safety analysis of that.

And then, finally, the boron redistribution is evaluated and demonstrated to not be a safety concern. Natural accumulation of boron in the core during ECCS operation tends to help with the shutdown conditions, both for design basis events and severe accidents.

MS. McCLOSKEY: So, shifting gears now, the remainder of the open session discussion, we wanted to cover a range of changes that were made in Chapter 15 between the FSAR Rev 2 and FSAR Rev 4, and what effect that those changes had on the Chapter 15 analysis results compared to what was discussed with the Committee last year.

MEMBER MARCH-LEUBA: Before you delve into that, one of the primary reasons -- well, all these discussions we've been having, obviously, we would have had anyway, but the reason we haven't had a review of Chapter 15 is because, back in July when we saw Rev 2, there were 34 open items in this SER, 11 of which were UOIs, where "U" means unresolved, unopen, unusual, depending on who you ask.

So, while you're going through this, can you tell us -- I think the staff is going to give us a list of them -- but can you tell us how they were resolved? Most of the open items were they're doing a LOCA calculation and the LOCA Topical Report hasn't been approved yet. Those are not the UOIs, but there were 11 UOIs. So, if you can focus on those, that will help us resolve a lot.

MS. McCLOSKEY: Yes, and I think the NRC staff is going to present the specific UOIs, and what we tried to cover here was the technical content that supported those UOIs.

So, this slide, again, speaks to what we were just talking about. And when we spoke with the Subcommittee and the full Committee last summer, we were at FSAR Rev 2. Changes that were incorporated into FSAR Rev 3 in the fall of 2019 included updating the analyses in NRELAP5 Version 1.3 to 1.4; incorporating updated base model input into those Chapter 15 analyses. In some cases, we incorporated more conservative core design input and more bounding ranges into the analysis, as well as the DHRS actuation signal changes and addition of secondary site isolation. So, that's the material that looks at the impact of that on the transient progressions as well ECCS actuation signal changes.

And then, the changes made in Rev 4 focused on the ECCS IAB threshold and pressure release changes that were discussed with the Committee at the beginning of February.

So, in terms of the code changes made to create NRELAP5, Version 1.4, the modifications were made due to routine code maintenance and error corrections. There were 26 specific code fixes, and the most notable are summarized here. And the primary one of interest, I think, is some error corrections in the condensation correlation that ended up having less than 2 psi impact the containment pressure calculations. So, overall, fairly minor error corrections being made.

We also have a number of new features. These don't impact the DCA calculations at all.

DR. SCHULTZ: Megan, as you've been required to make modifications, are any of these modifications part of those that you ought to report back to the code developers? That is, the RELAP5 code developers.

MS. McCLOSKEY: The developments.

DR. SCHULTZ: And do you do that on a consistent basis?

MS. McCLOSKEY: I would defer that question to the code development team and what their processes are for feeding back to the RELAP5 code development. Some of it is specific to -- like the condensation correlation corrections were specific to modifications we made to the code.

DR. SCHULTZ: Right. I understand that.

MS. McCLOSKEY: And so, that wouldn't necessarily impact RELAP5-3D.

DR. CORRADINI: My memory was from the summertime, last summertime, that your code developer lead that was discussing with us did report them back, but now I'm fuzzy.

MS. McCLOSKEY: That's also my recollection, but I would want to confirm that with the --

DR. CORRADINI: Would you do that maybe offline?

MS. McCLOSKEY: Yes.

DR. SCHULTZ: Just confirm, and if it's not true, then let us know.

DR. WOLF: This is Brian Wolf on the phone. Am I able to speak?

DR. SCHULTZ: Yes.

MS. McCLOSKEY: Yes.

DR. SCHULTZ: We hear you, Brian. Thank you.

DR. WOLF: Yes, so this is Brian Wolf, Code Development Supervisor.

We do report back errors that impact the IRUG releases of RELAP5-3D. We typically will provide those errors as we find them, and then, every year we are involved in the IRUG and domestic user group, which is like an international RELAP user group that INL supports. And so, we go to those meetings and discuss those errors with them, as well as receive errors from the international community. And we evaluate those for impact on our NRELAP5 code also.

DR. SCHULTZ: Perfect, Brian. Thank you.

MS. McCLOSKEY: Thank you.

Okay. Going on to the next slide and the base model changes, just to give some history here, Revision 0 of the base model was released in 2015, and that was what was used to support the DCA submittal at the end of 2016.

Revision 1 was released in August of 2017 with a number of updates for design consistency to incorporate minor geometry changes that were based on drawing updates and get that aligned, some minor flow loss updates that result in changes to the best estimate values, as well as updates for analysis consistency and downstream use for the Chapter 15 analyses. So, we incorporated nodalization changes to match the LOCA model as well as adding passive heat structures in the riser region that are insignificant with respect to non-LOCA event transients, but are included in the LOCA model to maximize stored energy.

Other changes made included the change from an elevation-based to the volume-based calculation of collapsed liquid level, which was discussed as part of the LOCA Topical Report meeting, and error correction in specifying the material properties of the lower containment material. That error correction had already previously been corrected in specific impacted analyses, like the containment pressure analysis, prior to the DCA submittal, and we were now folding it into the base model update.

So, it is a range of, it's a collection of relatively minor changes that affect a lot of different parts of the model and look like a lot of differences, but the overall impact is fairly small.

In Revision 2 of the base model, done at the beginning of January of last year, which ultimately supported the Rev 3, is where we incorporated removing the ECCS actuation on the riser level, as well as, again, flow loss updates and minor geometry corrections.

DR. CORRADINI: Can we stop on that one?

MS. McCLOSKEY: Uh-hum.

DR. CORRADINI: So, maybe I'm the only one that gets nervous about that, but this seems like an important redundancy that was lost. And can you repeat why it was taken out? This is the design?

MS. McCLOSKEY: Yes.

DR. CORRADINI: The calculation I'm --

MS. McCLOSKEY: Which we will get to in a couple more slides as well.

DR. CORRADINI: Okay, fine. If we're going to get to it, then I'll wait.

MS. McCLOSKEY: Yes.

DR. CORRADINI: Okay.

MS. McCLOSKEY: It was a redundant actuation with respect to ECCS actuation, but I think it is important to note that the sensors are still in the design. They are still required for post-accident monitoring and indication to the operators as to what the riser level is.

DR. CORRADINI: I remember that. What's bothering me is the fact that you don't have redundancy in the actuation of ECCS.

MS. McCLOSKEY: Okay.

DR. CORRADINI: So, let me ask a different question you can defer. So, you get certified. Life is good. You get a COL. What if the COL owner wants to put it back? Does that require a change in the rule? Because this is a redundancy that bothers me.

MS. McCLOSKEY: I would defer.

DR. CORRADINI: You can think about that, and the staff can think about that. But you're taking it out. If I want to put it back because I just happen to be a super-conservative owner, that's an allowable since the instruments are there; it's a matter of the instrument, the I&C rods to basically get it back into the protection system. Can you just kind of think about that?

MR. PRESSON: Yes, and we'll defer that. But a quick thought is, if they wanted to incorporate that, that wouldn't be any different than any other departure made to an applicant.

DR. CORRADINI: That would be a departure?

MR. PRESSON: Yes.

DR. CORRADINI: Okay. Fine. That's what I thought.

MEMBER REMPE: Could I be picky about your words when you said that the sensors are still there because they're required for post-accident monitoring? Don't you want to say that they provide additional assurance for post-accident monitoring? That's what you meant, right? Okay. Just to make sure, because there's been some issues where in the DCA that people kind of slide into the other verbiage and it causes me heartburn. Thank you.

MS. McCLOSKEY: Thank you. I understand.

Okay. In terms of the neutronics range changes that were incorporated into the Chapter 15 analysis, there was no change to the actual core design, but we did run a couple of additional depletions of the core design for high and low flow rates and generated some more bounding ranges of some of the parameters, such as a more negative Doppler temperature coefficient, some changes in delayed neutron fraction, augmentation factors for asymmetric reactivity events.

And what we are trying to do here is really get ourselves better situated to support --

MEMBER MARCH-LEUBA: Corvallis, can you please put your phones in mute? We're hearing an interference.

It might be the public line? Okay. Please mute all your phones and we'll live with the "crush, crush, crush."

Keep going.

MS. McCLOSKEY: Yes. Why did we do this? We were trying to situate ourselves a little bit better as we get ready to do reload analyses and to decouple some of our safety analyses from some of the variations in best estimate flow rate that we had seen over the time.

They still need to mute the phone line.

MEMBER MARCH-LEUBA: Just speak a little louder.

MS. McCLOSKEY: Okay. Going to the DHRS actuation changes, what we did was we added a module protection system signal for secondary side isolation for a range of conditions indicating an upset in the secondary side conditions. That closes the main feedwater and main steam isolation valves. It does not actuate DHRS. And the DHRS actuations were limited to a smaller subset of signals indicating insufficient secondary side cooling. This was done to support expected plant startup progressions.

In terms of the impact on the transient analyses, there's minimal or no impact in heat-up events where you get DHRS actuations on high pressurizer pressure or high RCS temperature. In cool-down events, you would typically actuate secondary side isolation first, and then, later DHRS on high steam pressure conditions, minimal impact on other events that aren't related to secondary side upsets.

CHAIR KIRCHNER: I was going to observe -- I'm jumping ahead to your next slide where you summarize the changes -- but, certainly, in terms of operational reliability, less is more here. Get down to the essential trip signals that you need and not have -- like your list is less than half, which I view as a probable improvement in the operational reliability of the system.

So, I assume you went through it. You said it was also driven in part by the actual startup progressions. So, I sense that you went through a logic exercise where you realized that that very long list on the left versus the new list on the right would get you into a better operating configuration.

MR. BRISTOL: Yes, that's correct. In addition, even after an event, the minute the DHR cooling takes over, the signal will clear to allow the operators to pick a module up potentially and work on recovery procedures.

CHAIR KIRCHNER: Okay.

MR. BRISTOL: The old design, those actuations stayed in until we reached some cold conditions. And so, they're basically locked into DHR cooling, and that was the key consideration that went into making the change.

MEMBER MARCH-LEUBA: But by removing some sensors from -- I mean, is there any sensor that became non-safety grade?

MS. McCLOSKEY: No. No. I think what you can see here is that, basically, everything that actuated DHRS before, now actuates secondary side isolation, and the DHRS actuations are limited.

MEMBER MARCH-LEUBA: Because, yes, my concern was, if you used to have four sensors and now you have only one because they're not safety grade, that would not be good.

MS. McCLOSKEY: No, there was no change there.

MEMBER MARCH-LEUBA: I don't see the riser level anywhere. That's a non-safety grade?

MS. McCLOSKEY: That was never a DHRS actuation.

MEMBER MARCH-LEUBA: Right, but is it safety grade or not?

MS. McCLOSKEY: It was in Rev 2 and now it's not a safety grade module protection system signal.

DR. CORRADINI: But I think his question is that it may not actuate what -- does it still remain as a safety grade measurement system? That's what I thought you were asking.

MEMBER MARCH-LEUBA: Uh-hum. Yes, my

question is, do you have four or do you have only one?

MS. McCLOSKEY: We still have four in the design.

MEMBER MARCH-LEUBA: Okay. Keep it that way.

MS. McCLOSKEY: All right. Let's go on to the --

CHAIR KIRCHNER: Typical caution: members' individual opinions on how you design your plant are not the Committee's.

MS. McCLOSKEY: Understood.

The next couple of slides give examples of the limiting event progressions that were impacted by the DHRS actuation change. So, if we start on the decrease in feedwater temperature event, what's happening there is the feedwater temperature is decreasing. You're getting a reactivity response from the colder water. You're also getting a response from the control rod, control system, trying to maintain RCS temperature, which increases power and RCS hot temperature.

And the limiting cases in both Rev 2 and Rev 4 occurred where the high power and the high RCS temperature limits were reached around the same time, when you account for different signal actuation delays. There's a longer delay that we account for in the temperature signals. And so, both cases tripped near high power, high temperature limits, and the DHRS actuation goes on high temperature -- or sorry -- yes, high RCS temperature.

MEMBER MARCH-LEUBA: When we're talking about DHRS, I remember that, for startup conditions, to avoid overfilling the DHRS with cold water, you had to follow some procedures. Does this mean it's strictly controlled? I mean, you have to pre-heat feedwater before you could -- I mean, if you overfill the DHRS with water, then it doesn't work. And for the startup procedures, we were doing something to prevent that. Is that still there administratively?

MR. BRISTOL: So, it's actually built into -- the overfill protection signal is related to the high, or the low superheat trip, right? So, in order to establish that the DHR is operable, the steam generator must be maintained in a super-heating-type mode. That's very challenging if you're trying to heat the RCS up.

So, in order to avoid that, the change that was made for low temperature conditions was to isolate one of the feedwater reg valves once level had been established through whatever boil-off procedure that they would use. Then, the low super-heat trip signal would be bypassed if one of the two feedwater isolation valves was closed. So, effectively, an overfill event, then, would be protected from during startup conditions because you have one of the steam generators bottled up, and therefore, confirming you have DHR functionality. And that's a limitation in order to, as part of the startup progression, the containment is slowly evacuated and the level is drawn down. In order to get to the condition where you're allowed to unflood the containment, you have to demonstrate operability of one DHR train.

MEMBER MARCH-LEUBA: And that's based on steam super-heat?

MR. BRISTOL: That's right. Well, the level -- under startup conditions, we actually have some dP sensors that are designed to confirm where the level is with respect to the steam generator.

MEMBER MARCH-LEUBA: The level inside the steam --

MR. BRISTOL: The level inside the steam generator, yes. So, the differential pressure between the steam line and the feedwater line is, then, used as an indication of where the collapsed level is in the steam generator while under really low flow conditions, where that dP is meaningful.

MEMBER MARCH-LEUBA: That's only, then, at zero power, right?

MR. BRISTOL: That's done before the containment is unflooded. So, as part of Mode 3, we go from a flooding containment passive heat removal condition --

MEMBER MARCH-LEUBA: I'm thinking of later on this week, and what would happen if those tubes were oscillating wildly, and the flow, what would happen to the signal? But that will be during power operation, not --

CHAIR KIRCHNER: But they would do this on low, very low flow, right?

MR. BRISTOL: Yes. Yes, really low flow as part of the startup progression.

MEMBER MARCH-LEUBA: And then, just ignore during power operation?

MR. BRISTOL: So, as part of the startup progression, the low super-heat, once the level has been established, then the containment level is drawn down and the containment is evacuated. That allows for continued heat-up of the RCS. You've got the one steam generator bottled up, so that you know if the DRH actuation signal comes in, it will work. And then, operationally, the other steam generator is used to balance the heat-up rate.

MEMBER MARCH-LEUBA: Okay. We'll ask that question later in the week.

MR. BRISTOL: Okay.

MS. McCLOSKEY: Just to give another example on a second cool-down event, the increase in steam flow rate, the sequence of events here looks a little bit different, but in both cases it's similar to the decrease in feedwater temperature cases. The limiting cases are where the high power and high RCS hot temperature occur around the same time.

In the Rev 2 analysis, that reactor trip was first on the high RCS hot temperature. In the Rev 3 and 4 analysis, it was first on high power. And so, with the high power trip, that doesn't actuate secondary side isolation or DHRS. The secondary side isolation comes later when low pressurizer pressure, and then, further on after the secondary side is bottled up, and you're still transferring decay heat. You generate a high steam pressure condition in the secondary side and actuate DHRS.

MEMBER PETTI: And which change, or is it a combination that really -- it is the condensation heat transfer? I can't remember.

MS. McCLOSKEY: I think in terms of why the high temperature versus high pressure or high power conditions, I'd say it's the combination of changes in the base model and some of the ranges conditions, but, unfortunately, what the table doesn't convey is that in both cases you're right up near the high temperature limit and the high power limit in both cases. And so, you're roughly the same conditions.

In terms of a heat-up event, if you look at the feedwater line break limiting DHRS case, this is a large feedwater line break inside of containment that disables one train of DHRS. In both cases, you reach a high pressure limit very quickly that actuates reactor trip. In the SR Rev 2 analysis, that also actuated DHRS very early in the transient. In the Rev 3 and 4 analysis, it's the DHRS actuation is delayed a few seconds until you get to the high pressurizer pressure condition, but, overall, there's really no impact.

And then, if we look at a reactivity event for the uncontrolled bank withdrawal at power, similar to the cool-down events, this is the limiting cases occur where high power and high temperature occur around the same time. And that's what is seen in both the Rev 2 and the Rev 4 analysis. And so, you follow the reactor trip with DHRS actuation due to the high pressurizer pressure signals.

All right. So, in terms of the ECCS valve operation and the changes to remove the RCS level signal for ECCS actuation and modify the inadvertent actuation block threshold and release pressures, as the Committee is familiar with this from the beginning of February, the emergency core cooling system valves now receive an actuation demand on high containment level or on loss of DC power to the trip valves. If that demand comes while the RCS pressure is significantly higher than the containment pressure, which would be typically only under -- for a loss of coolant accident condition, you would only get that if you assumed loss of DC power to the trip valves. So, if the RCS pressure is still high, then the IAV functions; if the actuation is above the threshold pressure, the IAV will block to hold the valves closed until the release pressure is reached. If the ECCS valves are actuated while the RCS is already depressurized through the break to containment, and the pressure is below the threshold pressure, the IAV does not engage and the ECCS valves open directly.

I think we can go to the next slide.

This summarizes the changes made in the actuation signal. The containment actuation signal levels were increased and the riser level was removed.

CHAIR KIRCHNER: May I ask, I think, more a question of clarification? Why such a high level in the containment before you actuate? To fill the containment that full, quite a bit of water from the primary system has been lost. Is that roughly the height of the RRV? Is that what you're limiting? I'm just trying to clarify why you wait so long.

MR. BRISTOL: So, the nominal recirculation or equilibrium level is kind of around the 260-inch range.

CHAIR KIRCHNER: That's what I would guess.

MR. BRISTOL: So, there's sort of two items at play here. The range is fairly broad, and that's to allow for sensor uncertainty and processed air considerations, right? So, that's one consideration.

So, waiting until the containment level has reached the point at which recirculation flow would occur was part of the thought process of ECCS. So, the reason to actuate ECCS is when you've lost enough inventory that you're ready to recirculate, and that's in and around that range.

CHAIR KIRCHNER: So, in the mass balance between the primary and the secondary, where is the level at that point in the primary? Below the riser?

MR. BRISTOL: Yes, well below the riser.

CHAIR KIRCHNER: Well below the riser?

MR. BRISTOL: Yes.

CHAIR KIRCHNER: That's what I would think. Okay.

MEMBER MARCH-LEUBA: Yes, the numbers you told us were 10 feet above the top of active fuel all three levels are equivalent. Are those inches of above top of active fuel or what?

MS. McCLOSKEY: No, that's relative to the inside bottom of containment.

MEMBER MARCH-LEUBA: Oh, in units of feet above top of active fuel, what are those numbers? Do you know?

MR. BRISTOL: So, I think that's where the 260 is close to the 10-foot equilibrium.

MEMBER MARCH-LEUBA: Okay.

MR. BRISTOL: Depending on initial conditions, it's going to move a little bit, right? How much mass you assume as your starting point --

MEMBER MARCH-LEUBA: That's a very good

reference point. So, we just moved it a little bit above the equilibrium where it --

MR. BRISTOL: Yes, it went on the other side. Yes.

MEMBER MARCH-LEUBA: And operating below the equilibrium is not good because, then, you're losing water from inside.

MR. BRISTOL: Yes. So, what we observed is, for most transients, the ECCS actuation includes actually an injection of water and vapor to the containment. So, just conceptually, from an ECCS performance perspective, we would wait for that event until recirculation was ensured.

MEMBER BLEY: I may have lost where we are here. All of these changes we've been talking about are changes in the analysis. Are they also changes that are to be reflected back into Chapter 7 for the set points for all of these systems?

MR. BRISTOL: They're already incorporated in Chapter 7.

MEMBER BLEY: Okay. As shown here?

MR. BRISTOL: Yes. Our Rev 4 was -- oh, these were incorporated in Rev 3 or Rev 4?

MS. McCLOSKEY: Yes, these were incorporated in Rev 3, except for the IAV changes which were incorporated in Rev 4.

MEMBER BLEY: Okay.

MR. BRISTOL: Which was last year.

MEMBER BLEY: Okay. Thanks.

MS. McCLOSKEY: And then, the IAV changes in terms of the threshold pressure, if the threshold pressure of ECCS is actuated above that, then you get an IAV block and that changed from a range of somewhere in the 1,000 to 1200 psid to greater than 1300 psid. And if you get a block --

MEMBER BLEY: But this was a result of the testing on the valves?

MS. McCLOSKEY: Yes, yes. This was a result of the testing. And if you get a block, then the release pressure changed from the range 1,000 to 1200 to 950 plus or minus 50 psi.

And so, the changes in the IAV affected the FSAR Chapter 6.2, the FSAR 15.6.5, and the FSAR 15.6.6 analysis for heat containment pressure, loss of coolant accident analysis, and inadvertent valve opening. And those revised analysis results were submitted in September of last year and reviewed by the staff in audit in October, and Revision 4 was formally submitted in January of this year.

I think this slide, this table summarizes the results that the Committee is familiar with from the beginning of February. The heat containment pressure results increased by about 8 psi due to evaluation of staggered IAV release points for where the second recirc valve opens at 1,000 psi and the vent valves open later at 900 psi.

The LOCA minimum water level above the top of active fuel decreased by about .2 feet due to the lower minimum release pressure of 900 psi. And for the inadvertent valve opening events, the minimum critical heat flux ratio decreased, and that was due to the model revisions discussed, and not specifically due to the threshold change.

MEMBER MARCH-LEUBA: See, that 1.32 is getting -- I mean, we used to say that this core has so much margin that we can't do anything about it, but the limit is 1.28. I think very close to limits.

MS. McCLOSKEY: The limit for the recirc valve opening event is 1.13.

MEMBER MARCH-LEUBA: Oh, okay. Well, then, I'll wait for your next slide to complain about it.

(Laughter.)

MS. McCLOSKEY: I think the next slide summarizes basically what we just talked through, and specifically, the valve opening events confirm that the 13 psi threshold pressure was still low enough to hold the ECCS valves closed while the control rod is inserted. And overall, all the events continue to demonstrate margin to the acceptance criteria.

And finally, one slide that summarizes the overall impact of changes on the SR Chapter 15 analysis for results. In terms of maximum RCS pressure in the events, there's effectively insignificant changes, where the limiting results remain around 2170 psia compared to acceptance criteria of more than 2300 psi.

Maximum steam generator pressure for AOOs is the advertent operation of the DHRS event, which continues to show significant margin to the acceptance criteria. The limiting accident is the steam generator tube failure event where the maximum pressure is less than 900, so it's 600 pounds of margin to the acceptance criteria.

The limiting MCHFR cases for single rod withdrawal is the limiting AOO, and that has less margin, considering more conservative bounds of the core design, particularly in the augmentation factors and some of the reactivity parameters.

MEMBER MARCH-LEUBA: So, the change

between 1.6 and 1.37 is uncertainties were widened out? You widened the uncertainties?

MS. McCLOSKEY: Yes.

MEMBER MARCH-LEUBA: My concern is that 1.37 is getting very close to 1.28, and when we go to an extended power, we might start hitting limits or getting really close to limits.

DR. CORRADINI: They may stop doing as conservative -- well, what you're saying is correct.

MEMBER MARCH-LEUBA: You're shooting yourself in the foot by doing it now instead of doing it later. But, of course, this is on the books, so you're going to change it. But it is always harder to change these uncertainties later on than to do it from the beginning.

MS. McCLOSKEY: Understood.

MEMBER MARCH-LEUBA: We've been saying that this reactor has so much margin because it's low power, very small. Hey, it's only 1.1. Most operating reactors operate with more margin than that to CHFR. And this is only because you are too much concerned about this in your calculation.

MEMBER BLEY: I have a question about the IAV. We know it's there to prevent inadvertent valve operation. I don't remember if this came up in a meeting or in a discussion afterwards. If the valve should open when the pressure differential is higher than specified, would there be any particular challenges to the containment if you start from a higher pressure? If the IAV doesn't work and the valve pops open for some reason?

MS. McCLOSKEY: We analyzed the valve popping open as an initiating event. And that is the limiting containment pressure case.

MEMBER BLEY: And where did you start from?

MS. McCLOSKEY: From normal operating conditions.

MEMBER BLEY: Normal operating conditions? Okay.

MS. McCLOSKEY: Yes, 102 percent power.

MEMBER BLEY: And it's limiting, but it's not real -- how close is it? I don't remember.

MS. McCLOSKEY: That's the 986 psi pressure case for the recirc valve opening.

MEMBER BLEY: That's when you raised the limit?

MS. McCLOSKEY: Yes.

MEMBER MARCH-LEUBA: So, your IORV, in the event of opening early of a valve, it assumes a failure of the IAV and a failure of the valve and a failure of everything. You're going to assume an additional failure of another IAV afterwards, is that correct? So, you assumed your first IAV failed. The second works.

MS. McCLOSKEY: Correct. We assume a mechanical failure of the valve that results in it opening in some way. And we do not take a second IAV as a --

MEMBER MARCH-LEUBA: The first one failed. It's a failure.

CHAIR KIRCHNER: Yes, it's a LOCA.

MS. McCLOSKEY: Overall, our limiting events remain the same between Rev 2 and Rev 4, and we continue to demonstrate margin.

MEMBER BLEY: Now that analysis led to raising the containment design pressure, but you don't actually have a containment now. My guess -- and, Pete, this is in your -- my guess is when you actually design the final containment, it's probably good well beyond the assumed design pressure that we're talking about here. I mean, it's a pretty sturdy vessel.

MEMBER RICCARDELLA: Typically, you design to a safety factor of three.

MEMBER BALLINGER: Yes, it's a code-designed vessel.

MEMBER RICCARDELLA: Code-designed.

MEMBER BALLINGER: So, it's a factor of three.

MEMBER RICCARDELLA: Safety factor of three.

MEMBER BLEY: A safety factor of three. Okay.

MS. McCLOSKEY: Yes. Right. And this is compared to 100 percent of the design pressure for the acceptance criteria.

All right. I think we're going to go to the next slide.

So, in conclusion, the return-to-power analysis results demonstrate that ECCS cooling conditions are limited and the equilibrium power level is 1 to 2 percent, in the 1 to 2 percent range, depending on the number of uncertainties included.

The boron transport analysis results for ECCS demonstrate that the boron concentration remains higher than the initial concentration.

We've discussed several of the changes incorporated into FSAR Revision 3 and Revision 4 that resulted in relatively minor changes in the overall results.

And our Chapter 15 analysis results continue to demonstrate margin to the acceptance criteria.

CHAIR KIRCHNER: Thank you, Megan.

Members, any quick questions here at this point? We will have a closed session on this material later this afternoon. But if there are any immediate questions? None? Okay.

Let's, then, recess until 1:00, and we will, then, hear from the staff on this topical area.

(Whereupon, the above-entitled matter went off the record at 12:08 p.m. and resumed at 12:59 p.m.)

CHAIR KIRCHNER: We are back in session. Now we're going to hear from the staff on Chapter 15, the NuScale and DCA, and I'll turn to Omid, please.

MR. TABATABAI: Thank you, Dr. Kirchner. Thank you so much, members. We really appreciate your time. This is staff's opportunity to talk about Chapter 15, and specifically related to boron redistribution, return to power in ECCS, and some other focused others.

Today with me are Ryan Nolan, Jeff Schmidt, and Carl Thurston. They will be leading the technical discussions and answering your questions.

We have this afternoon and tomorrow morning, half the day for this discussion, so there is plenty of time, but we have a full agenda as you can see on this slide.

I just wanted to, before we get started, give credit to all of the technical reviewers who contributed to the SER and preparation for this ACRS briefing.

With that, I'd like to ask Ryan Nolan to get us started on the closure of open items.

MR. NOLAN: Yes, all right, thank you, Omid. So the last time we presented you Chapter 15 was last year in July, and we presented, in that presentation, we had a list of 11 UOIs, and so most of those UOIs are also being presented today. You'll see them identified in some of the headers on the slides.

A couple of them that we won't be discussing since we presented them last month is the NRELAP Version 1.4, as well as the steam generator heat transfer uncertainties since those were part of the methodologies. Sprinkled in here are also some phase 2 open items we'll be presenting as well.

So I'll be presenting the first two topics, credit for the non-safety valves, as well as treatment of the IAB and single failure. These were both presented as part of the phase 3 presentation in July, so I'll go through them fairly quickly at a high level.

So Chapter 15 for NuScale safety analysis credits non-safety related valves as a backup to their safety-related counterpart when applying to single failure criteria. They are specifically identified in Table 15 0-9. It's primarily the feedwater reg valves, a feedwater check valve, and a secondary MSIV.

The credit for these components in secondary line breaks is consistent with past practice, and this is described in NUREG 0138. The technical document primarily focuses on the main steam line break scenario where credit in a lot of the safety analysis, it credits the turbine stop and control valves as providing some means of isolation when you assume a single failure of the safety-related MSIV.

NuScale is following a similar methodology here, and at a high level, it looks at, you know, the augmented quality associated with those valves, the surveillance requirements, as well as operating experience.

So our conclusion is the crediting of the feedwater reg valves, control valves, and secondary

MSIVs is consistent with this guidance.

MEMBER MARCH-LEUBA: This non-safety equipment that is credited, does it become part of the tech specs for repairability requirements?

MR. NOLAN: They are --

MEMBER MARCH-LEUBA: The thing is whenever you have a safety grade equipment that goes out of service, you shut down the plant if you cannot put it back in service. Are these --

MR. NOLAN: Many of them are in tech specs. I don't have them in front of me right now.

MEMBER MARCH-LEUBA: The concern is just, okay, I'm going to relax to the Class 1-E and you don't have to have -- I understand, but are you going to allow it to be out of service if it goes bad and for how long?

MR. NOLAN: Yeah, I believe many of these are in --

MR. SCARBROUGH: Ryan, this is Tom Scarbrough. Yes, those main steam isolation valves and the bypass valves, they are in the tech specs, so there are requirements for them to --

MEMBER MARCH-LEUBA: Well, these are repairability requirements.

MR. SCARBROUGH: Right, right.

MEMBER MARCH-LEUBA: Good.

MR. NOLAN: Omid, you can go to the next slide. One extension of this sort of methodology is with respect to steam generator tube failures. The NuScale safety analysis credits the non-safety related secondary MSIVs for providing isolation during that event.

And the focus of the NUREG was on secondary line breaks and is comparing the consequences of a primary line break versus a secondary line break, and in this case, NuScale is crediting it for a primary line break.

So we requested NuScale perform its sensitivity analysis to show us the consequences if that secondary MSIV also failed to isolate and the results are shown here. It's approximately 50 percent more mass in release and proportional increase in doses. However, large margins still remain to the acceptance criteria.

So now I'm going to talk a little bit about the treatment of the IAB. As you are all aware, the ECCS system, it's a valve system, includes an inadvertent actuation block feature with the primary goal of reducing the inadvertent opening of these ECCS valves. These inadvertent opening can occur on a loss of power or a spurious signal.

The Chapter 15 analysis does not assume a single failure of the IAB closing function, so any assumption of two or more valves failing at the same time would be considered a beyond design basis event.

The IAB is an active component, safety significant, first of a kind, and we issued a SECY paper to request the Commission direction, and that is documented in SECY-19-0036. You can go to the --

And so the staff has implemented the Commission direction from that SRM, specifically that the assumption of single failure does not apply to the closing valves, the closing function of the IAB for the purposes of the Chapter 15 review, and that's how we've implemented that direction.

That concludes the single failure to the IAB discussion. We'll certainly be presenting some results. Carl, as part of his LOCA presentation, you'll see an example of the ECCS system functioning, the IAB when it blocks and releases to give you guys an example of how that would work.

MR. SCHMIDT: This is Jeff Schmidt. So this is a slide we've seen before. It's again the exemption for GDC 27. First, we state what GDC 27, and then the staff took a position in the preapplication letter that reliably controlling reactivity in GDC 27 meant shut down as the final state. When we look at kind of the totality of our regulations, we came to that conclusion.

But as we know, as we discussed earlier today, following initial shutdown, the NuScale reactor can return and maintain criticality on a cool down event using the safety-related passive heat removal systems, which is the key heat removal system and the ECCS under certain bounding conditions.

NuScale submitted an exemption to GDC 27 and requested approval for our principal design criteria in GDC 27, or PDC 27, sorry, and you saw that also in NuScale's presentation and I think you'll see it here again.

So we also sent a SECY up. It was an INFO-SECY. SECY-18-0099 specifies the following three criteria for the exemption. The design of the reactor must provide sufficient thermal margins such that a return to power does not result in a failure of the fuel cladding fission product barrier as demonstrated by not exceeding the SAFDLs for analyzed events.

A combination of circumstances and conditions leading to the actual post-reactor trip return to criticality is not expected to occur in the lifetime of the module, and incremental risks to the public health and safety from a hypothetical return to criticality at the NuScale facility with multiple reactor modules does not adversely erode the margin between the Commission's goal for new reactor designs related to the estimated frequencies of core damage and large releases and those calculated for the NuScale design.

So as part of the Chapter 15 review, I mean, we're really focused on the first bullet, and that one is protection of the SAFDLs during a design basis event.

MEMBER MARCH-LEUBA: I agree with that. The rest are icing on the cake.

MR. SCHMIDT: Yes.

MEMBER MARCH-LEUBA: Let's assume it happens, no consequences.

MR. SCHMIDT: Right, and, you know, we're taking this from, you know, a cool down can occur from an AOO or a postulated accident in the NuScale design, so that's why we picked SAFDLs.

MEMBER MARCH-LEUBA: My example is always the step up testing and demonstration of DHRS cooling as a feature of our reactor. I mean, that has probably 100 percent within the first three months of operation.

MR. SCHMIDT: I don't know if I'd say that, but, yes, I think it's likely to occur potentially in the lifetime of the plant.

MEMBER MARCH-LEUBA: Now, we have seen that if you maintain DHRS cooling and another circulation, you can reach criticality, but if you reach the low temperatures required for criticality, you will not have enough circulation unless you have an addition, a CVCS addition because, I mean, to reach --

MR. SCHMIDT: Yeah.

MEMBER MARCH-LEUBA: -- the cold temperature required, you would need to raise the level.

MR. SCHMIDT: Yes, so in the various scenarios we'll talk about coming up is like, for example, in a riser uncovery, you're just going to sit kind of in that equilibrium condition. Where the riser was assumed to be preserved, you have to add, typically add water mass to keep that natural circulation going, right, and ECCS is relatively stagnant.

MEMBER MARCH-LEUBA: Because the example I like to concentrate on is we build one in south

Florida next to my house and there's a hurricane coming, and you shut down all of your 12 modules and go into passive cooling in preparation that you're going to lose offsite power, which you probably will if it's a Category 5 hurricane.

So you have now all 12 modules in passive cooling with different levels of boron concentration in each of them and everything that can possibly happen in that scenario, and that's not such a farfetched scenario.

MR. SCHMIDT: I mean, so if it was a controlled shutdown, I think they would maintain RCS levels such that you have natural circulation that wouldn't necessarily uncover the riser if you have time to --

MEMBER MARCH-LEUBA: You are anticipating a loss of offsite power. If you shut down every single module and you see a Cat 5 going towards you, you know you're going to lose reliance from the coal power plants, or gas power plants in that case, and you are going to completely passive. So, I mean, it's not a farfetched scenario.

MR. SCHMIDT: I guess I agree that you could lose power, but I guess it's a function of what actions you might take before you lose power, I think, that is different, and that's a scenario that I can't really speak to, but I think there are actions that could be taken prior to that loss of offsite power and there are other -- you know, they do have other means of AC power potential that could be available.

So getting back to SECY-18-0099, ACRS endorsed the proposed staff criteria with the addition of adding the overall facility risk, which is reflected in the third criteria above, and satisfying the three criteria of SECY-18-0099 ensures that there is no undue risk to the public health and safety.

MEMBER MARCH-LEUBA: But going back to the analogy we used this morning of the Thelma and Louise thing, if you shut down your 12 modules, and you know the hurricane is coming and you are going to lose all of your lines around for the next week, you will have an emergency operating procedure that says you're going to flood them with boron before you lose power.

MR. SCHMIDT: That's kind of what I was getting at.

MEMBER MARCH-LEUBA: Right, but I don't see that. I mean, by delegating all of the operating procedure decisions and guidance to the sequel for the movie, which is the COL, are we doing ourselves a favor or --

I mean, because right now is the time to think about those things. That's why we have generic technical guidelines. We don't have procedures, but we have guidelines, so I don't know.

MR. SCHMIDT: Yeah, I think --

MEMBER MARCH-LEUBA: If it was my reactor and I know I'm going to lose power, I would like to have 2,000 PPM boron in there.

MR. SCHMIDT: Yes, I would fill it up and do the same, but I think those procedures come later as part of the COL.

MEMBER MARCH-LEUBA: Okay.

MR. SCHMIDT: I mean, they're not -- those procedures aren't available for the staff to review and they come as --

MEMBER MARCH-LEUBA: The Thelma and Louise thing is if you don't know there is a cliff coming, how do you know that the procedures will be -- a reactor -- when we get to the COL, they're going to be in a hurry.

They're going to be pouring concrete and they want to get all of the side tacks done. They won't be thinking calmly and intelligently about things. They're going to want to get done as soon as possible.

MR. SCHMIDT: Hopefully they'll go back and review our discussion today.

MEMBER MARCH-LEUBA: Okay.

(Simultaneous speaking.)

PARTICIPANT: We will not assume the future COL's integrity.

MR. SCHMIDT: Again, NuScale showed us this morning their PDC 27. Again, the real change is probably the last sentence, "Following a postulated accident, the control rods shall be capable of holding the reactor core subcritical under cold conditions with all rods fully inserted."

Again, it's very similar, and like they discussed this morning, what we've done is we've taken the SAFDL -- earlier versions of this had the SAFDLs in as part of the 27.

What we've done is removed it from that and put it really into Chapter 15, 15.0 as the criteria. There's a couple of tables in there that says, you know, basically the criteria is don't fail fuel under AOOs or postulated accidents, so it was removed from here and put in Chapter 15. The table is right there, tables three and four.

So as we talked a little bit this morning, three scenarios can potentially lead to a return to power, and that's the decay heat removal system, water remaining above the riser, water dropping below the riser, release was evaluated, and the decay heat removal cool down without DC power, which is basically ECCS actuation at the IAB set point.

I guess that shouldn't be used, but there is a threshold is the proper term, and then ECCS cool down. As we just discussed, it can occur as a result of most Chapter 15 AOOs and postulated accidents.

Key assumptions in the return to power are no operator action, safety-related equipment used to mitigate, and worst, a stuck rod is out consistent with the GDC.

A return to power is possible at EOC conditions, but not when significant RCS boron exists, BOC and MOC, and we'll have a slide on that in the closed portion with some numbers for you.

EOC return to power, as NuScale mentioned, the new analysis method was in Rev 3 and Rev 4 which took the super conservative analysis that was originally done in Rev 2 and kind of broke it out into the individual scenarios with finer fidelity, I believe.

As they discussed, NRELAP is used to determine the average moderator temperature for a series of constant core powers for each one of the cool down scenarios. Those were the individual squares or dots on their return to power graphs.

It basically eliminates the NRELAP reactivity feedbacks. That's all effectively picked up in the SIMULATE model now. Various reactor building cool temperatures were analyzed. Again, they went down to the lowest tech spec value.

Multiple riser heat transfer assumptions were analyzed. SIMULATE5 was used for the reactivity bounds using a power search to determine the critical power for the various input conditions of flow and inlet temperatures.

And then NRELAP core power versus RCS flow is used to determine basically you take these range of conditions into a single SIMULATE5 criticality line versus average moderator temperature, and they showed that plot earlier.

The intersection of the RELAP constant power cases and the average moderator temperature curve and SIMULATE line determines the equilibrium return to power value. MCHF is calculated using a pool boiling correlation.

Again, continuing on for EOC return to power, some of the conservative assumptions, reactor pool level is maximized, temperature is minimized. They have the 30 percent increase on the decay heat removal system.

ECCS valve capacity is maximized and the conservative decay heat is assumed, and that's one of the OUIs from the Phase 2, staff's Phase 2 SE. That was specifically looked at.

MEMBER MARCH-LEUBA: In this, conservative means low, right?

MR. SCHMIDT: Yes, conservative means low, yes.

MEMBER MARCH-LEUBA: Okay.

MR. SCHMIDT: SIMULATE5, so this addressed OUI 1506-3. It was one of the reactivity issues with kind of the earlier methods. Basically, the overall redo of the method kind of addressed this OUI and the use of SIMULATE addressed the OUI.

Reactivity is biased to address SIMULATE5 uncertainties. Conservative coolant density, as they mentioned, there's no credit for voiding here. Conservative peaking factors for MCHF determination, you know, correspond to a stuck rod. Zero xenon concentration is assumed consistent with the 72 hours following a reactor trip.

And then as they mentioned, MCHF is

calculated with the peaking factors, plus this factor of two to account for potential transient overshoot conditions. And in the closed session, we'll show some graphs that really don't indicate that transient, a significant transient condition occurs.

MEMBER MARCH-LEUBA: The factor of two is on the peaking factor?

MR. SCHMIDT: The factor is, yeah, the localized peaking factor. The decay heat removal system cool down, assuming the riser remains cover, and ECCS cool down return to power, so these are the two conditions as NuScale showed that can have a potential return to power, so it's the riser remains covered and on ECCS cool down.

A return to power is less than two percent. A significant CHF margin exists. We'll give specific numbers in the closed slides. I assume that NuScale will also, and the staff concluded that general design criteria can if SAFDLs is met.

Decay heat removal system cool down with water level dropping below the riser and riser uncovered remain subcritical due to sufficient decay heat. They kind of showed that in their plot earlier today, and this was again at OUI 15.0.5-1.

Staff independent confirmatory analysis

yielded very similar results, and we'll see that also. Staff recommended approving the exemption to GDC 27.

Here is the staff's confirmatory results. I would just, you know, that's about one percent power. That's probably something to remember as we get into the other sessions. Again, this is EOC --

DR. CORRADINI: That's what, I'm sorry?

MR. SCHMIDT: One percent power.

DR. CORRADINI: Oh, thank you.

MR. SCHMIDT: Yeah, look at kind of the yellow line if you have a color version of it.

DR. CORRADINI: Yeah.

MR. SCHMIDT: Yeah, so this is EOC kind of biased or conservative reactivity coefficients, zero PPM and a very large pool to try to maintain pool temperature at 65 degrees.

MEMBER MARCH-LEUBA: Why does the power keep going down towards 60, 72 hours?

MR. SCHMIDT: Yeah, so --

MEMBER MARCH-LEUBA: You have less xenon?

MR. SCHMIDT: Yeah, so if you remember from the earlier phase 2, this plot actually showed it going down much more. It's because we allowed the bay temperature to increase, the local bay. Here, we're trying to preserve as best we can the pool staying at a constant temperature, and that's kind of why it's going down a little bit.

MEMBER MARCH-LEUBA: So it's changes in your pool temperature?

MR. SCHMIDT: It's a little bit changes in the pool temperature, yeah.

MEMBER MARCH-LEUBA: But it's not Samarium or xenon?

MR. SCHMIDT: If you look back at the earlier ones -- well, it's also, there is also xenon decay that's happening at the same time --

MEMBER MARCH-LEUBA: But that would go the other way.

(Simultaneous speaking.)

MR. SCHMIDT: -- but it would go the other way. Up to about 72 hours, it would be -- well, at 72 hours, it's about zero reactivity change at that point.

MEMBER MARCH-LEUBA: Yeah, but how long does Samarium take to build?

MR. SCHMIDT: Samarium takes to build, I do not remember. Samarium --

MEMBER MARCH-LEUBA: I don't know either.

MR. SCHMIDT: Yeah, I think it's a long time period, yeah.

MEMBER MARCH-LEUBA: So that could be Samarium?

MR. SCHMIDT: I'd have to actually ask.

MEMBER MARCH-LEUBA: It's not important.

MR. SCHMIDT: Yeah, obviously I don't know if they modeled Samarium necessarily in this. I would tend to doubt it, but I could ask for help at that point.

MEMBER MARCH-LEUBA: What code were you using?

MR. SCHMIDT: This is TRACE.

MEMBER MARCH-LEUBA: TRACE?

MR. SCHMIDT: So, yes, so this is PARCS.

MEMBER MARCH-LEUBA: PARCS, so PARCS will have --

MR. SCHMIDT: Well, PARCS TRACE.

MEMBER MARCH-LEUBA: Yeah, they will have a similar scenario.

MR. BIELEN: This is Andy Bielen with the staff. I ran the kinetics calculations. Samarium is not a, doesn't -- it's not a radioactive isotope. It only --

Samarium is produced in fission, and that poison effect is a function of what power level you're operating at, but unlike xenon, when you hit a reactor trip and you don't have appreciable power production, the Samarium just sits there, so our reactivity curves were input as a function of xenon versus time, or xenon reactivity versus time.

MEMBER MARCH-LEUBA: Oh, it wasn't looking at the calculation?

MR. BIELEN: Well, so the kinetics calculations inform the point kinetics inputs that we put into TRACE.

MEMBER MARCH-LEUBA: It's okay. It's not relevant.

DR. CORRADINI: So does that mean we were still not sure why it's kind of slowly going down?

MR. BIELEN: I think, so the, you know, the feedback we got from the subcommittee after the June meetings was, well, you, you know, your analysis looks pretty good, but you only assumed a single bay, so you should really, you know, try to keep the bay temperature, you know, as low as possible.

And the way that we executed that was basically to increase the bay size that was surrounding the module by a factor of 12 to account for, you know, having 12 modules where only one was operating.

And so even though -- you know, so in the initial calculations, that power really started coming down once you get out because the bay heated up. We're still seeing a teeny bit of heat up here, but, you know, the effect is much more --

DR. CORRADINI: So the ultimate heat sink is the reason?

MR. BIELEN: Yes.

MR. SCHMIDT: Right.

DR. CORRADINI: Right, that's what I was -- okay, thank you.

MR. BIELEN: Yeah.

MR. SCHMIDT: So this is the boron redistribution, non-EOC potential for return to power. Excess reactivity is controlled by soluble boron, which is just a normal PWR situation. Loss of soluble boron in the core doing cool down can cause a recriticality similar to EOC, ECCS cool down.

That was the postulated potential part of it is, you know, where does the boron redistribute since you have a lot of reactivity controlled by the soluble boron, as was pointed out this morning.

So we looked at a variety of loss mechanisms and they're listed there, flashing, liquid discharge, entrainment, boron volatility, core and riser boron gradient, and diluted CNV water entering the core, so those were like kind of the loss mechanisms that were evaluated.

So the staff review is documented in SCR 15.0.6 as was mentioned already. Staff conducted a detailed audit and numerous public meetings on the topic as I'm sure everybody is aware at this point.

As NuScale pointed out, a control volume method using RELAP to calculate fluid transport. Boron transport is informed by the RELAP fluid transport. Methodologies, conservative assumptions, and we'll get into that in the closed session, to minimize core boron concentration.

Mass is removed by conservative treatment of physical phenomena as well as boron mass being effectively artificially removed to ensure an overall methodology conservatism, and I think we'll be able to spell that out clearer in the closed portion of what is removed like by physical phenomena and what is just removed by other methodology means.

DR. CORRADINI: But you're driving it towards an end of the cycle calculation?

MR. SCHMIDT: What we're trying to show is that the core concentration never drops below the initial value that -- so criticality is maintained. So EOC, we demonstrated that with conservative assumptions, we will go recritical again.

DR. CORRADINI: Right.

MR. SCHMIDT: This, the goal of this is to show that there is no criticality concern earlier on in cycle.

DR. CORRADINI: Okay.

MR. SCHMIDT: In other words, boron is not displaced in other parts of the system such that you would have a diluted concentration in the core and have to consider basically a recriticality with a higher boron concentration in the core, so we want to preclude criticality earlier on.

DR. CORRADINI: You're demonstrating, but from the standpoint of if it can occur, if it were to occur under conservative assumptions, it would first occur at end of cycle? That's what I'm trying to --

MR. SCHMIDT: No, not necessarily. So if the -- let's just take the worst-case example, and Jose kind of mentioned this this morning, is if all of the boron were to just stay in containment.

Let's just say for argument's sake, just play it out on containment, that would be a large reactivity insertion, right, because it's around 1,200 PPM, 12,000 PCM. The control rods would not be able to control that reactivity insertion.

So you would have a situation where you had a lot of excess reactivity in the core that your control rods could not control, and potentially you would return to a different higher power level.

But as NuScale alluded to this morning is the core riser region basically increases in boron concentration so you don't have to worry about this potential return to power scenario.

MEMBER MARCH-LEUBA: But all you're saying does not include the recovery phase of --

MR. SCHMIDT: It does not. It does not.

MEMBER MARCH-LEUBA: My concern is if you can figure out a mechanism to get -- you have a lot of boron in the core. We'll call it 4,000 PPM as some of the calculations show, and you have a little bit of cool unborated water and you can push it in there.

What gives me nightmares is that you put unborated water at the bottom of the core and you have 4,000 PPM at the top of the core, so you reach reactivity plus one cent, and now your efficient is positive.

MR. SCHMIDT: Right.

MEMBER MARCH-LEUBA: Highly positive in the top of the core.

MR. SCHMIDT: Right, right.

MEMBER MARCH-LEUBA: So there is no way that you can smoothly reach an equilibrium. The moment you get plus one cent, it's going to go boom, boron critical.

MR. SCHMIDT: Yeah, and I think we'll allude to that in some slides going forward, that you would want to prevent that --

MEMBER MARCH-LEUBA: Yeah.

MR. SCHMIDT: -- situation from occurring.

MEMBER MARCH-LEUBA: And I would like to see from the risk informed or the risk evaluators what is the chance of an operator error of omission or commission that -- I mean, if that possibility is there. I said this morning if I were to design an experiment to do that, I would laugh at myself. I would say you cannot possibly do it, resulting that from doing, but to disqualify it in principle, that's pretty difficult to do without analysis.

MR. SCHMIDT: Right, right, I understand what you're saying.

MEMBER MARCH-LEUBA: So we're really, this member would like to see, whenever the condition report analysis comes through, I don't know what that documentation will be. We may have to ask for an information meeting somehow or --

MR. SCHMIDT: Okay.

MEMBER MARCH-LEUBA: -- a subcommittee to see what it was.

MR. SCHMIDT: Okay, and then finally, riser and core boron gradient was evaluated based on this data and VEERA data, and we'll go into that fairly extensively in the closed session.

Non-EOC or redistribution, the staff agrees that boron will concentrate in the core riser due to boiling. Staff concluded that boron loss terms informed by NRELAP are conservative.

Staff concluded that assuming the elimination of the down cover, which is one of the very great conservatives, and lower plenum boron mass is conservative with regard to the core boron concentration.

Boron volatility correlation was reasonable based on the NuScale operating conditions and conservative by not including any rewetting and return to the core.

VEERA test data demonstrates that the core boron concentration is uniform once saturated boiling conditions are reached. Evaluation of a fully diluted water mass as zero PPM below the saturated boiling core elevation demonstrated the core remains subcritical.

MEMBER MARCH-LEUBA: So this bullet is incompatible with the -- this sub-bullet is incompatible with the previous bullet, with the main one. The main bullet says you cannot have non-uniform --

MR. SCHMIDT: Well --

MEMBER MARCH-LEUBA: -- but just in case --

MR. SCHMIDT: Well, no, what I'm trying to communicate actually is, you know, the saturated boiling occurs at a certain height into the core.

MEMBER MARCH-LEUBA: Yes.

MR. SCHMIDT: So the question is could you have a diluted slug below that? So the VEERA tests clearly show, and we'll see that, that once you get to that condition, you have a lot of mixing due to boiling.

The staff was concerned that, okay, what happens below that, and can you introduce a diluted slug that would still have enough uranium mass down there to go critical even below that saturated boiling point? And that was a specific analysis that the staff asked for and NuScale performed.

MEMBER MARCH-LEUBA: The other thing in

defense of NuScale, if you have essentially zero flow, the saturated boiling boundary would be at elevation zero, because whatever water you had in the bottom, it will warm up until it boils.

MR. SCHMIDT: Yeah, it's --

MEMBER MARCH-LEUBA: You won't have sub-cool water down there.

MR. SCHMIDT: So it's pretty low and you'll see the numbers when we get to that.

MEMBER MARCH-LEUBA: Where does the coolness come from?

MR. SCHMIDT: Well, there might be some in the downcomer, I guess, was the postulated.

MEMBER MARCH-LEUBA: Okay, so we'll see those numbers?

MR. SCHMIDT: Yeah, they're presented, but that was -- the staff was worried that -- you know, you could also argue that there is still convection down there and mixing, but it just becomes a harder argument to make, so we chose the brute force conservative method.

MEMBER MARCH-LEUBA: Okay.

MR. SCHMIDT: NIST long-term cooling core exit void test data demonstrated there was enough two phase flow mixing to promote riser core mixing.

Staff concluded that the final boron concentration at 72 hours is greater than the initial core boron concentration, maintaining subcriticality, again preventing a return to power.

Staff is aware of a condition report written by NuScale last week that is still being discussed, so we're still in the information gathering stage there.

We also looked at it beyond 72 hours from like a Chapter 15 perspective out to seven days as kind of a RTNSS test for the CVCS system.

Staff considered NuScale capability to cope with potential boron redistribution without the need of, again, non-safety systems, thinking primarily here the CVCS system for a period of seven days consistent with SECY-96-128, which is RTNSS B.

Staff reviewed the NuScale calculation, initial conditions, assumptions, and results. Staff agrees that there is sufficient decay heat removal and the core would remain subcritical throughout the seven-day period.

Boration from the CVCS is not required in the first seven days, and we'll go through some specific details in the closed portion of the presentation on this.

MEMBER MARCH-LEUBA: The boration is not required to satisfy SAFDLs. Is that what you mean?

MR. SCHMIDT: To keep it subcritical.

MEMBER MARCH-LEUBA: But we don't -- we return to criticality.

MR. SCHMIDT: So this is --

MEMBER MARCH-LEUBA: Or it's non-EOC?

MR. SCHMIDT: Yeah, this is the non-EOC. This is the BOC when you have sufficient boron concentration.

MEMBER MARCH-LEUBA: Yeah, and when you review the condition report because you guys have an audit or you have a meeting or something --

MR. SCHMIDT: Yes.

MEMBER MARCH-LEUBA: -- make sure you understand this LOCA, this low break LOCA, even I thought this morning, when the inside level is at the top of the fuel and the containment is 15 feet higher.

MR. SCHMIDT: Yes.

MEMBER MARCH-LEUBA: And you open the flow.

MR. SCHMIDT: Yes.

MEMBER MARCH-LEUBA: What happens there?

MR. SCHMIDT: We understand that, yes.

MEMBER MARCH-LEUBA: And think of all of

the audit. I mean, I'm not the only one thinking about passing.

MR. SCHMIDT: Right.

MEMBER MARCH-LEUBA: You're supposed to --

MR. SCHMIDT: Right.

MEMBER MARCH-LEUBA: -- to do it yourselves.

MR. SCHMIDT: Yes, you are correct. So this is now switching gears a little bit. This is the long term decay heat removal system operation. As we talked about, it's a safety-related heat removal system used to mitigate non-LOCA transients, normal shutdown.

So this is, I'm trying to separate kind of the function of the decay heat removal system from, say, a normal plant evolution. Normal shutdown uses the secondary side to cool down. CVCS is available to maintain RPV level above the riser. This is where you would borate high.

Say if you're going down for a refueling outage, for example, obviously you have control of the plant and you can borate sufficiently high. In that one, you'd flood it with the containment flood and drain system that you would maintain a subcritical configuration.

RPV level may drop below the riser elevation following a reactor trip. We've discussed that and subsequent cool down. Without makeup water, the water level will drop below the riser within the range depending on your initial conditions at three to six hours, and the function of the core decay heat. RPV temperature is below 700, or 400, I'm sorry, 420 degrees when the riser becomes uncovered.

MEMBER MARCH-LEUBA: And the figures in the DCA, I think it's Chapter 9 when they're talking about the DHRS performance show like two to three hours from uncovering, and that's probably with a conservative DHRS which is, it may happen in two hours.

MR. SCHMIDT: Yeah, I'm going to turn to Carl here because he --

MR. THURSTON: Yeah, I want to say our worst-case analysis was it was somewhere around five to six hours.

MR. SCHMIDT: Yeah.

MEMBER MARCH-LEUBA: The ones on the DCA show the speed of hot and cold temperature at two-and-a-half hours, and I would assume that that was for a conservatively low heat distribution coefficient which is not conservative in our -- just so you -- let's not make any -- I mean, if it happens earlier than --

MR. THURSTON: Maybe it could.

MEMBER MARCH-LEUBA: It may happen earlier than we think.

MR. THURSTON: And our sensitivity studies.

MR. SCHMIDT: Again, long term decay heat removal operation here. When the water level drops below the riser, two scenarios are possible depending on the heat transfer. I think NuScale kind of covered this.

With sufficient heat transfer across the riser, two separate internal natural circulation loops are established, one in the core riser and one in the downcomer. In this scenario, the riser remains uncovered.

So your magenta lines, what they showed earlier today, if heat transfer across the riser is insufficient to remove decay heat, either a continuous or intermittent single phase natural circulation is established.

Collapsed liquid level remains well above the top of the active fuel, and a significant fraction of the steam generator tube surface area remains covered, so this is like the lowest collapsed level on the decay heat removal system, and I think we have numbers in the closed session.

The applicant concluded and the staff agreed that adequate cooling exists if the riser becomes uncovered after 72 hours.

MEMBER MARCH-LEUBA: But as I said earlier, in this condition for the second bullet, if you have not uncovered sufficient steam generator area to condense the steam than whatever steam you could use, then the pressure would rise.

MR. SCHMIDT: Yes.

MEMBER MARCH-LEUBA: And you will condense more mass because first, the delta H is smaller and the density is higher, so either -- I mean, the pressure will continue to rise until you condense all of the steam that it generates, that the core generates. So as long as part of the steam generator is uncovered --

MR. SCHMIDT: Right.

MEMBER MARCH-LEUBA: -- you will condense.

MR. SCHMIDT: Right, yeah. Yeah, so on the riser uncovered scenario, I guess we just talked about this, some water vapor will condense will condense on the exposed steam generator tubes.

This has the potential to dilute downcomer over a long period of time as water vapor is assumed to have negligible boron concentration. Dilution of the downcomer is limited by the fraction of the steam generator tube surface area covered.

MEMBER MARCH-LEUBA: I do not agree with that.

MR. SCHMIDT: I think it's a factor in it.

MEMBER MARCH-LEUBA: No, either you condense all of the steam you produce in the core or you open the safety valves. Those are the only two options.

MR. SCHMIDT: Okay.

MEMBER MARCH-LEUBA: And I don't think you open the safety valves.

MR. SCHMIDT: No, probably not.

MEMBER MARCH-LEUBA: Not at one percent power.

MR. SCHMIDT: Probably not, no.

MEMBER MARCH-LEUBA: So you would be condensing all of the steam that you generate?

MR. SCHMIDT: Over time, yeah, over time.

(Simultaneous speaking.)

MR. SCHMIDT: I mean, you're still going to be generating steam from the core, right?

MEMBER MARCH-LEUBA: Yeah.

MR. SCHMIDT: There is still some decay heat, so it's --

MEMBER MARCH-LEUBA: The thing that I think is going to happen is that there is not going to be much steam coming out to the top surface of the riser because you have all of this heat transfer to the --

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: -- walls that will collapse it inside of the riser.

MR. SCHMIDT: Right.

MEMBER MARCH-LEUBA: But you have to realize it.

MR. SCHMIDT: Right, I think we have analyzed it to the point of, I think, reaching a safe, stable condition. You're talking about the recovery from that condition, I believe.

MEMBER MARCH-LEUBA: Yeah, I'm talking about the potential of distilling steam and condensing water on the downcomer.

MR. SCHMIDT: Right, but I think as long as you stay in that configuration, you're okay. It's when you go to recover, I believe, that there's a concern.

PARTICIPANT: But you got to be, recover

how you recover.

MEMBER MARCH-LEUBA: Yeah, but you --

MR. SCHMIDT: Yeah, you've got to be careful how you recover.

MEMBER MARCH-LEUBA: As the Thelma and Louise analogy goes by, you're driving on a condensed road and that's not a safe, stable condition, so this is equivalent to --

MR. SCHMIDT: If you didn't touch it, it would -- all right, so potential exists for reestablishing a single phase natural circulation which again is what we were talking about here, causing a recriticality similar to the event analyzed in the ECCS return to power.

All I'm saying there is basically what Jose is saying, is that you could have a return to power. If you sufficiently diluted that downcomer, that situation could exist.

MEMBER MARCH-LEUBA: See, the thing that I notice, I know it doesn't take an hour. I don't know if it takes five hours, 10 hours --

MR. SCHMIDT: Right.

MEMBER MARCH-LEUBA: -- 100 hours.

MR. SCHMIDT: Agreed, agreed.

MEMBER MARCH-LEUBA: And that would be

something that we need to know.

MR. SCHMIDT: Right, reestablishing RPV level above the riser after extended again requires the operator to initiate action to recover the module through the addition of water. Again, this is dealing with the concern of reestablishing single phase and moving a diluted slug from the downcomer into the core.

Post-accident module recovery is not required to be evaluated in Chapter 15 design basis, so it's not really a part of the DCA, but it is part of the recovery procedures.

CHAIR KIRCHNER: Okay, though it's not required, I would imagine that you and the applicant had discussions about how they would effect this recovery?

MR. SCHMIDT: We have not had detailed discussions.

CHAIR KIRCHNER: Well, I'm just trying to think through with the design as evaluated, the DCA, the normal recovery would be borated water from the CVCS system, right?

MR. SCHMIDT: Yeah, I think I lay out a possible scenario, but we have not really spent a lot of time analyzing that recovery scenario.

CHAIR KIRCHNER: Okay --

MR. SCHMIDT: So I think I have an example of a potential one coming up.

MEMBER MARCH-LEUBA: You don't need to have a detailed procedure of how you will do it, but it would not be prudent or even responsible to certify a design for which you don't know --

CHAIR KIRCHNER: Whether they can do it.

MEMBER MARCH-LEUBA: -- whether it can be done.

CHAIR KIRCHNER: That's where I'm going.

MEMBER MARCH-LEUBA: At least one way.

CHAIR KIRCHNER: Yeah.

MR. NOLAN: I think at this point, our position is the design is capable of addressing it. It's not a design issue. There's enough flexibility in the design as we know it, as part of the certification, which would address this scenario.

DR. CORRADINI: But there must be a generic way to recover. That's what I think --

MEMBER MARCH-LEUBA: As long as there is one way.

CHAIR KIRCHNER: Yeah, that's where I'm going with this is as long as, as Jose is saying, there is in the DCA as you evaluated it, a means for a safe recovery, I think that the procedure isn't needed, but certainly the functionality is, right?

You wouldn't approve a DCA if there was no feasible way of recovering in a safe manner, right? So you've thought it through at least that much, Jeff.

MR. SCHMIDT: Well, we have, we have. I don't know if we've thought of all of the possible scenarios to recover from that.

MEMBER MARCH-LEUBA: If there are a dozen scenarios for a loss of recovery, even better, but before we certify, we need to have one.

MR. SCHMIDT: Well, I think I have one example.

MEMBER MARCH-LEUBA: And I've seen it and it looks good.

MR. SCHMIDT: Yeah, so I think there is one that I propose, but I put it as an example because I can't necessarily speak for the applicant here.

MEMBER MARCH-LEUBA: You need to be able to, with a straight face, raise your right hand and say there is at least one way of getting out of this, because the way I put it on my other is you got the toothpaste out of the tube --

MR. SCHMIDT: Right.

MEMBER MARCH-LEUBA: -- but separating the boron.

MR. SCHMIDT: Right.

MEMBER MARCH-LEUBA: How do you put it back together?

MR. SCHMIDT: Right.

MEMBER MARCH-LEUBA: You figure out a way to put it back together.

MR. SCHMIDT: Right.

MEMBER MARCH-LEUBA: That's good, but before this presentation, I don't think we have thought what we would do.

CHAIR KIRCHNER: How does that fit with Thelma and Louise?

(Laughter.)

CHAIR KIRCHNER: Go ahead.

MR. SCHMIDT: All right, thank you. NuScale indicated a recovery of the NPM following extended heat decay heat removal would be procedurally controlled. We talked about that this morning again. Plant procedures are not part of the DCA review. Procedures would be developed by the COL applicant or applicant holder.

Chapter 13 has COL items that addresses the development of operating procedures. Staff believes procedures could be developed to adequately address recovery from this condition, which is what we were just speaking about.

Plant design allows for the following operational strategies that could address recovery from this condition, excuse me, mixing core and downcomer boron concentration by simultaneous injection of letdown preserving the RCS level. This is really just the no additional mass such that you would reestablish single phase natural circulation.

Again, downcomer boron concentration would be sampled to ensure that, you know, you have adequate mixing before you did establish single phase natural circulation, and then as part of that, you would be confirming that shutdown margin exists before restoring the level. So all of those, I think, are means of getting the plant back safely.

So now we're going to switch gears a little bit and talk about the ATWS scenario. This is kind of derived from, I think, scenarios you guys have seen for ATWS that research has done supporting Chapter 19.

Limiting RPV pressure ATWS was initiated by a loss of AC power. The loss of AC power causes the feedwater pump and turbine to trip. Control rods are assumed to fail to insert.

RPV pressure increases due to loss of heat sink. You get a high RPV pressure that trips the decay heat removal system to actuate. RPV inventory is lost by lifting the safeties and discharging into containment.

ATWS is not considered a design basis event due to the design of the reactor trip system within the MPS lowering the probability of occurrence below one times 10 to the minus per reactor year. Again, that's kind of denoted in Section 15.8. Again, all this is saying, this is a beyond design basis event.

MEMBER REMPE: I'm sorry, I'm slow. Could we back up a bit on the slide about -- I'm always hung up about whether you need water level in the RCS or not. You're going to have procedures and you're going to be restoring water level. Doesn't that imply you need that sensor? It's more than just additional assurance for post-accident conditions?

MR. SCHMIDT: You mean as far as sampling the lift?

MEMBER REMPE: No, I'm talking about under the --

MR. SCHMIDT: I'm sorry.

MEMBER REMPE: Yeah.

MR. SCHMIDT: Is this ATWS or is this --

MEMBER REMPE: The long term DHRS operation.

MR. SCHMIDT: Okay.

MEMBER REMPE: And that slide as well as the prior slide infers that you kind of need reestablishing RPV water level.

MR. SCHMIDT: At some point, yes, for recovery.

MEMBER REMPE: Don't they need that sensor then?

MR. SCHMIDT: The sensor exists. It's just not connected to the --

MEMBER REMPE: But, I mean, oh, they don't need the sensors, or they do need the sensors, and I'm still kind of -- do they need it or not?

MR. SCHMIDT: I guess, could you be more specific of what sensor you're --

MEMBER REMPE: Water level in the RPV.

MR. SCHMIDT: Oh, RPV water level.

MEMBER REMPE: Yeah.

MR. SCHMIDT: Right, right --

MEMBER REMPE: Like on the slide.

MR. SCHMIDT: -- for the recovery. So it's not used for mitigation. It's used for recovery.

MEMBER REMPE: So post-accident monitoring, we've been told numerous times the only thing I need is something in the spent fuel pool. Everything else is just to provide additional assurance, and we kind of, we keep going back and forth.

This morning, the NuScale person said, oh, yeah, it's required by the operator. Is it required or not? What should the procedures have in it? Should they rely on it or not?

MR. NOLAN: So, yeah, it's not required to mitigate the design basis event. So what we're talking --

MEMBER REMPE: But it's required for recovery.

MR. NOLAN: Oh, yes.

MEMBER REMPE: That's the way it's required, I know, but it's a real fuzzy thing.

MR. NOLAN: It's after the beyond design basis event.

MEMBER REMPE: Yeah, it's just kind of fuzzy to me. Am I the only one who is kind of saying this is fuzzy?

DR. CORRADINI: I don't think I understand your worry. I thought the applicant said that all of these instruments would remain. They just aren't connected to essentially actuation of --

MEMBER REMPE: Right.

DR. CORRADINI: -- ECCS, but these are all post-accident safety grade instrumentation that will remain.

MEMBER REMPE: Yes, but I'm kind of thinking about Chapter 20 more and post-accident monitoring. All they need is something in the spent fuel pool is what we've heard, and yet post-accident monitoring, isn't that sort of recovery?

MR. SCHMIDT: I guess I would say not necessarily. I mean, you can monitor the as-stable state condition and just monitor it. The recovery is when you want to, in my mind, make power again, you know, or move the module to the refueling if you need to refuel or something.

So in other words, you can monitor the plant after the event as long as it's monitoring a safe, stable state. I don't know if that's the same as recovery.

MEMBER REMPE: I guess it isn't. It's just a fine line, I guess, but I just kind of --

MEMBER MARCH-LEUBA: My concern with that safe, stable state of, let me say it again, driving another road, but it is not in a -- I mean, you create a safe, stable state, but it's only safe and stable as long as the operator doesn't do anything.

MR. SCHMIDT: Yes, I'm assuming no operator action, right.

MEMBER MARCH-LEUBA: Or you have --

MR. SCHMIDT: I'm assuming no operator action that's adverse.

MEMBER MARCH-LEUBA: You're assuming that you are not going to have inadvertent actuation of CVCS or shutting down the DHRS so the system warms up and raises the --

MR. SCHMIDT: Yes.

MEMBER MARCH-LEUBA: Yeah, those things have to be included on the PRA, all of those actions that could cause something bad.

MR. SCHMIDT: I'm going to have to defer to a PRA expert on that.

MEMBER MARCH-LEUBA: Yeah, I bet you they're not, and it would be -- certainly it may not be required for mitigation of the first 15 seconds of the transient, but it would be nice to have it.

MEMBER REMPE: History has shown that it became even more than nice with things you didn't expect over the years.

MEMBER MARCH-LEUBA: No, but I was happy that they told me that the RPV water level in the riser is safety grade, and there are four of them, and it's powered by the batteries.

MEMBER REMPE: Well, they said they were there. Did they explicitly say they were safety grade?

MEMBER MARCH-LEUBA: They did.

MEMBER REMPE: Yeah, okay, okay, so we'll look at it later, but I just kind of --

MEMBER MARCH-LEUBA: For recovery, we have to wait for the recovery for the closed session, but you're going to need more than that to recover. You're going to need active power, active pumps --

MR. SCHMIDT: Oh, yes.

MEMBER MARCH-LEUBA: -- sampling systems.

MR. SCHMIDT: Yeah, absolutely.

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: And all the time, you are just heating next to the cliff.

MR. TABATABAI: Dr. Rempe, you mentioned Chapter 20. Actually, the agenda for Chapter 20 is on the 4th, so we can revisit and we can probably discuss in more detail.

MEMBER REMPE: I just, I keep hearing

things and it's going back and forth, and I'm -- well, again, it's confusing me.

MR. SCHMIDT: So, again, so this is an ATWS scenario where you're relieving mass through the safety relief valve into containment. So, two ATWS scenarios are possible, control rods, insert rods early into the event or operators delay or take no action to mitigate the ATWS.

In both cases, again, the safety relief valves relieve pressure and discharge into containment. If the operators insert the control rods early in the transient as expected, the ATWS event effectively looks like a long term decay heat removal cool down scenario with the riser potentially becoming uncovered because you've lost mass to containment, probably more likely.

If operators delay or take no action to insert control rods and enough RPV inventory is lost, the riser, or the level drops below the riser breaking natural circulation and establishing a new equilibrium power, which I think you've presented that scenario.

Pete has presented that scenario in the ATWS scenario where you basically get to a new equilibrium power. The safe state is reached with the collapsed liquid level remaining above the top of the active fuel.

So this is ATWS mitigation and recovery now. If the operator acts to insert rods before the CNV level reaches the lowest CNV level ECCS set point, that recovery would be the same as a design basis decay heat removal cool down scenario.

Staff's conservative analysis demonstrates that the lowest CNV level is reached in approximately one hour. The likelihood of operators failing to insert control rods within one hour is highly unlikely.

If the operator could not insert control rods after reaching the lowest CNV level ECCS set point, additional analyses may be needed to determine the appropriate actions.

ATWS mitigating procedures are dependent on the specific ATWS event and the available equipment. Operator actions to recover the plant following beyond design basis events are not within the scope of the DCA review and are developed by the COL applicant or holder. There's a Chapter 13 COL item that addresses the development of operating procedures like we talked about earlier.

So, that slide just denotes that I think if you get the rods in early and don't get to the ECCS set points, you're basically on the decay heat removal system and it's very similar. Above that point, I think you have to think about how you would recover or mitigate that situation with more analysis.

A return to power with ejected rod, DCA does not address the potential return to power following a postulated ejected rod. Ejected rod is evaluated for the short term reactivity response only consistent with the requirements of GDC 28 and the guidance of SRP-15.4.8, the appropriate limit on the -- to appropriately limit the rate of reactivity increases associated with certain postulated activity accidents, including rod ejection, are primarily a check of loading pattern and control rod designs such that a coolable geometry is maintained.

Staff determined that the provisions in GDC 27 for evaluating design basis accidents in the long term are met for the NuScale design because inter-rod ejection accident need not be considered in the long term due to the robust design of the control rod housing, and staff evaluated the control rod housing design in the safety evaluation Section 3.9.4.

CHAIR KIRCHNER: Could you elaborate on that last bullet, Jeff, for the record?

MR. SCHMIDT: On the staff evaluated the

control rod housing design?

CHAIR KIRCHNER: Yeah.

MR. SCHMIDT: I can't go into great detail, but I think basically, as the discussion was this morning, was that it's a robust design. If it weren't a robust design, we would have other considerations, say an ejected rod plus LOCA. I don't know if there is somebody from the mechanical group here to report.

MR. NOLAN: I think all we're trying to say is it's determined to not be a credible missile source, the housing failure.

CHAIR KIRCHNER: Let me make a note here and just say that we would want, when we get to the full committee meeting, I think that topic should be addressed more thoroughly.

MR. NOLAN: Okay.

CHAIR KIRCHNER: Okay?

MR. NOLAN: Sure, and just to add a little bit of perspective here, from a precedence standpoint, most new reactors, these missile CRDM housing failures are not considered credible.

I went back to look at how we addressed this for the other new reactor designs, even the ones that we didn't certify but we did write safety evaluations on. Only one out of the six designs actually consider this a credible missile source, and NuScale followed a very similar methodology as those other new reactor designs.

CHAIR KIRCHNER: Can you elaborate on why one was credible?

MR. NOLAN: It was the APR1400. I think it was maybe just a legacy from the System 80, System 80+ design. That design happened to have missile shields, and so it was easier to just say it's credible. We did a barrier analysis and the hazard has been addressed.

MEMBER BROWN: I think this plant is different than the other ones. They were all large light water reactors. This is a tiny light water reactor, very compact, the other ones, expansive.

DR. CORRADINI: That makes this better.

MEMBER BROWN: I'm not sure. That's your opinion.

DR. CORRADINI: True.

MEMBER BROWN: That's your opinion.

PARTICIPANT: Since we're --

MR. NOLAN: I would just say from a credibility, a missile credibility standpoint, the methodology has not changed. So even though the design has changed, the method for deeming an internal missile credible or not has not changed.

MEMBER BROWN: How did they deem it credible or non-credible?

MR. NOLAN: Just as --

MEMBER BROWN: Did they say it could never happen? Did they analyze the head to determine you could never break the seals?

MR. NOLAN: No, it's --

MEMBER BROWN: Did they identify you didn't have this robust structure?

MR. NOLAN: It's a qualitative engineering judgment approach --

MEMBER BROWN: You looked at it and thought it was okay.

MR. NOLAN: -- based on the design requirements --

MEMBER BROWN: Fundamentally.

MR. NOLAN: -- placed on the component.

MEMBER BROWN: You looked at the design, and without any analysis, determined that it didn't appear to be credible, and therefore, it was marked credible and it was accepted.

There was no analysis necessarily to develop stresses and everything else as to what happens to differential pressures? What acceleration could you get, if you're going to break a -- none of that was done. You just looked at it. I'm not questioning --

MR. NOLAN: Right.

MEMBER BROWN: We do that frequently.

MR. NOLAN: Right, it was --

MEMBER BROWN: But saying it was credible or not credible solely based on that, to me, it's a judgment call. You could say -- to stake your points on previous credible or non-credible determinations without some specific analysis that goes along with it, that doesn't hold a whole heck of a lot of water except we've done it before.

MR. NOLAN: Right, it's based on primarily design requirements place on the component.

MR. SCARBROUGH: This is Tom Scarbrough. We'll take that back and we'll find the specific reviewer that looked at that, and we'll report back to the committee.

MEMBER BROWN: At some point in the past, somebody made a decision that there was a basis for it not being credible. All we've heard is people have determined it was not credible.

MEMBER BLEY: And some point before then, it was -- at least they devised methods to protect against it.

MEMBER BROWN: Exactly.

MEMBER BLEY: And, well, Matt's not here. Matt's seen -- I've been in a lot of plants, but I never saw missile shield, but Matt tells me in his plants he's been in, and it's on the railroad tracks, I guess, but they've actually got these shields in almost all of the existing plants.

MR. NOLAN: That's true, yeah.

MEMBER BLEY: I've never seen one, so --

MEMBER BROWN: So what was the basis?

MEMBER BLEY: What changed?

MEMBER BROWN: Yeah, what's the basis?

MEMBER BLEY: That's kind of it. If it was something you ought to protect against before, why isn't it any longer?

MEMBER RICCARDELLA: Well, I think --

MEMBER BROWN: I'm not sure it ever was, but --

MEMBER RICCARDELLA: Let me read a couple of sentences from GDC 4. Okay, GDC 4 says, if I can get into it, the structures, systems, and components --

PARTICIPANT: Use your mic a little more, Pete.

MEMBER RICCARDELLA: These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging flows that may result from equipment failures and from events and conditions outside the nuclear power unit.

However, dynamic effects associated with the postulated pipe ruptures in nuclear power units may be excluded from design basis when analyses reviewed and approved by the commission demonstrate that the probability of fluid system piping ruptures is extremely low under conditions consistent with the design basis for the piping.

So I think there's, you know, that's really what we're invoking, is that we believe the probability of this rupture happening and creating a missile is very low, but at some point in time, and maybe it doesn't need to be done as part of DCA, there should be some analyses.

I would probably -- my judgment is the same, that the probability is sufficiently low, you know, and the more recent work has said that the sufficiently low conservative criteria is 10 to the minus six probability of a rupture. That's what the xLPR program is using at least for an initial guide, but --

CHAIR KIRCHNER: Ten to the minus six or seven?

MEMBER RICCARDELLA: Ten to the minus six. That's separate from this --

PARTICIPANT: It's not in use now.

MEMBER RICCARDELLA: Ten to the minus seven is something different. That says 10 to the minus seven is the combination of the probability to rupture --

CHAIR KIRCHNER: Right.

MEMBER RICCARDELLA: -- times the probability of it hitting something, times the probability of it actually doing damage to it. That's the 10 to the minus seven, but I don't know that it really needs to be as low as 10 to the minus six, but that's what they're proposing as criteria in the xLPR program.

But I think it's certainly believable that the probability of one of these things coming out of the head is very, very low and meets this GDC, but at some point, perhaps an analysis should be done to demonstrate that, and I think that's what Charlie was saying, right?

MEMBER BROWN: Just saying it ain't so just doesn't meet the eyeball test.

MEMBER PETTI: So for me, it would be helpful when the staff goes back, is to understand historically, because you've said this has been done recently.

If the designs were all really similar, and I don't know if there's lots of different designs of a rod housing with the vessel, it would be good to know how much is in the design bucket versus how much is in what's called the probability bucket in terms of the argument. That would, I think, help provide some context.

MEMBER RICCARDELLA: Certainly this is much, much better than the existing -- I'm sorry, then the existing control rod drive mechanisms like the Davis-Besse type.

It's for sure. But it's not --

MEMBER PETTI: But it's the five --

MEMBER RICCARDELLA: But for me it's not all that much different then say, the safety injection nozzle in a PWR. Which is, you know, typically a small bore nozzle attached.

We don't want to assume the nozzle comes out of the vessel. But, we do assume that the safe --

that I breaks at the safe end. Right?

MEMBER REMPE: Pete, earlier today you were uptight about a sentence that was in a slide that basically said that they don't have to do this.

MEMBER RICCARDELLA: Because it's quote, part of a vessel. I don't -- I think that's a misuse of it.

MEMBER REMPE: And you said that you -- because we talked later. And what's on a slide isn't that important.

But, what's important is if it's in the staff SER and the DCA.

MEMBER RICCARDELLA: It's in the DCA. Chapter three.

MEMBER REMPE: And it might be good if you've gone through and looked at other DCAs, if that sentence has -- the offending sentence is still there and you might, I don't know if there's time to fix the SER.

But, it would be good to at some point, perhaps revise it so it's more appropriate.

MR. NOLAN: For clarification, was that statement made in an SER or on a slide?

MEMBER RICCARDELLA: It was on a slide. And I'm pretty sure it's in the DCA in Chapter three.

MR. NOLAN: Okay.

PARTICIPANT: I'm looking at Chapter three. And not --

MEMBER RICCARDELLA: And well, it's all -- you're a fast reader if you're --

MR. NOLAN: Yeah. It's a big chapter.

(Laughter.)

MEMBER REMPE: It would be good to see if what's in the slide is in the DCA. And it maybe, especially your SER.

PARTICIPANT: I know what description various words are.

MEMBER RICCARDELLA: Yeah. Yeah. I'm pretty sure it's there. I was the reviewer of Chapter three. And I'm pretty sure it's there.

MR. SCHMIDT: Okay. So, this slide, I think there was a question on the boron concentration requirements in mode 4.

So, I kind of tried to walk through what I thought the issue was.

MEMBER MARCH-LEUBA: And mode 4 is the refueling? Or the transition one?

MR. SCHMIDT: The transition. I think it's transition.

MEMBER MARCH-LEUBA: And section mode.

Yeah, mode 5 is refueling.

MR. SCHMIDT: Yes. That's correct. Again, just a reminder, technical specifications apply to normal operations.

And allowed design basis event initial conditions. Technical specifications are not applicable during a design basis event.

Generic technical specification table 1.1-1, Modes, defines mode 4 as transition as requiring having a K effect of less than .95.

Conditions to enter mode 4 from other modes need to be met before entry into mode 4.

LCO 3.5, .3, ultimate heat sink, states the bulk average boron concentration shall be maintained within the limits specified in the COLR.

A combination of inserted control R worth and RPV boron concentration ensured mode 4 K effective is less than .95. Ultimate heat sink boron concentrate is set to ensure entry into mode 4 and refueling operations also.

MEMBER MARCH-LEUBA: I believe we wrote a recommendation in the value of one of our letters. And since the transition, this transition to the fueling, --

MR. SCHMIDT: Right.

MEMBER MARCH-LEUBA: You might as well get the boron concentration for refueling before you move it.

Since moving the core with a crane and all of it is what ends up to be the limiting event when you help it, maybe you don't need to. But, it would be safe to require to be at mode 5 EPM of boron at mode 4.

Instead of yes, make a calculation for .95. That was our recommendation. It was just, there is no scientific basis for it.

Just yes. Why don't you put it at 2000 ppm and forget about calculations and numbers?

MR. SCHMIDT: All right. I guess I can't speak for NuScale. But, you know, when they go to refuel one of these it's kind of in the same body of water.

So, you're going to be putting in a fresh core at that point. And I would assume you're going to establish the fuel concentration to perform that evolution.

And that's probably permitting.

MEMBER MARCH-LEUBA: If I remember correctly, this .95 k effective, there's a table in Chapter four. Towards end of cycle the required ppm is only like 150 or 200 to reach that.

And sure it does give you .95. You have a calculation that does it. And why don't you put it at 2000 and forget about calculation. Make it subcritical.

That was all I thought.

MR. SCHMIDT: I guess that's up to that. I'm just saying that, so the plant, one of my used to plants, I mean this was the k effective for refueling, .95.

And we would calculate it every time for the new load that went in.

MEMBER MARCH-LEUBA: Well, you're going to open the core in the UHS, in the pool?

MR. SCHMIDT: Um-hum.

MEMBER MARCH-LEUBA: If you have it at 100 ppm and open it, you're going to flood it with 2000 ppm that comes from the pool, and there's going to be mixing.

And that is -- I don't know. That was our recommendation. That mode 4 should not be different than mode 5 on boron.

And I guess there cannot be a regulatory requirement that uses that. But certainly knowledge.

MR. SCHMIDT: I think, you know, having a mode 4 of .95 is appropriate. I mean, that seems -- I mean, you'll have your control rod still in there too, right?

MEMBER MARCH-LEUBA: Yeah.

MR. SCHMIDT: They're decoupled. So they're a negative reactivity that can be accounted for.

So, I guess I don't -- I mean, again, it would seem like when you open this up as you're saying, you're going to want a pool concentration that's high. Especially when you put in new fuel into that core that will likely set this pool concentration.

MEMBER MARCH-LEUBA: All right.

CHAIR KIRCHNER: I think that's the answer to that. I think the answer, you just said it, is that that reload calculation will set the concentration in the pool.

MR. SCHMIDT: But the pool concentration effectively.

CHAIR KIRCHNER: It has to be. And I think it will be pretty high.

MEMBER MARCH-LEUBA: That's for mode 5. Mode 5 they're going to get 2000. And so it is --

MR. SCHMIDT: Well, it's the same body of water.

CHAIR KIRCHNER: The same body of water. And once they open up, they have to have it at that level.

MEMBER MARCH-LEUBA: Well, I wouldn't say and that came from Member Skillman at the time. That was our recommendation since.

CHAIR KIRCHNER: Um-hum.

MEMBER MARCH-LEUBA: I'm worrying about moving a critical core out there with a crane. Why would you want to be close to criticality?

Just fly this as cold as you can before you move it.

MR. SCHMIDT: Again, I guess, again, I'm used to refueling with a k effective of .95. I don't see how transition is different then refueling at our current PWR.

CHAIR KIRCHNER: You know, that's pretty low k effective point. You know, we're .99 rhetorically.

If it were .99, then I would really be with you. And worried. But at .95 is pretty --

MEMBER MARCH-LEUBA: It was Dick's recommendation. I'm just defending an ex-member here.

MEMBER RICCARDELLA: Are we getting into

procedure space again on that one?

MR. SCHMIDT: Yeah. We're getting into operation of the plant effectively. All right. Let's --

MEMBER MARCH-LEUBA: It's called defense in-depth.

MR. SCHMIDT: Okay. Let's go to the next slide.

MEMBER MARCH-LEUBA: Which is a dirty word.

MR. SCHMIDT: And I put these together just to try to provide some context and examples. Mode 3 to mode 4, this is a transition to -- you know, this is a normal operation. RCS is cooled down by the secondary side.

CVCS would add borated water to the RPV. At some point you would probably trip your control rods in.

Ultimate heat sink water would be added to containment and mixed with -- through the RPV to the ECCS valves. And a combination of inter rod worth and RPV boron concentration again ensures k effective is less than the .95 prior to transition of the module basically.

MEMBER MARCH-LEUBA: And let me beat a

dead horse. Say that you are doing the refueling at, going into mode 4 at end of cycle.

MR. SCHMIDT: Uh-huh.

MEMBER MARCH-LEUBA: And looking at the table in Chapter four, I wish I had it in front of me. I will in a moment. You only require 300 ppm of boron.

MR. SCHMIDT: Um-hum.

MEMBER MARCH-LEUBA: So you put the 300 ppm boron, you're in mode 4. You disconnect your CVCS lines. You disconnect everything and start moving it.

MR. SCHMIDT: Um-hum.

MEMBER MARCH-LEUBA: Gets to the refueling station. Now how did you increase the boron to mode 5 condition once it's out there?

MR. SCHMIDT: I would say that they would do it when they flood up on CVCS system already. I go back to my original statement.

MEMBER MARCH-LEUBA: Yeah. Because if you move your module at the -- and the boron concentration is less than mode 5, you cannot go into mode 5 when you get to the station.

MR. SCHMIDT: Correct. You can't enter that mode unless you've established those conditions.

MEMBER MARCH-LEUBA: I guess that's what

Dick Skillman was thinking about when he recommended that.

MR. SCHMIDT: I don't think -- again, I don't know this table in Chapter four, but just because that table says something in Chapter four, does not mean that you would operate the plant like that.

MEMBER MARCH-LEUBA: I just don't see what mode 4 buys you. You've got me at mode 4.

MR. SCHMIDT: Well -- yeah.

MEMBER MARCH-LEUBA: If it is safe, the regulatory basis for your condition of --

MR. SCHMIDT: I guess you could set it to mode 5 and not have a mode 4. Sure. But, I don't see anything wrong with it.

Example for beyond design basis recovery. Again, just to be clear, you know, the plant stabilizes following a design basis event.

Again, the plant may or may not be in a mode to find per the generic tech specs table 1.1-1. Return to power is a good example.

You know, as we talked about before, you could be effectively at return to power at a low temperature. You would be not -- you are not in that mode table.

Operators will establish an operating mode defined by table 1.1-1 using whatever available systems they have. For example, mode 3, they could take it effective to mode 3 with k effective less then .99.

Again, transition to mode 4 is then the same as any other normal plant operation.

And now I'm going to turn it over to Carl.

MEMBER MARCH-LEUBA: I found the table. It's table 4.3-2. And it allows you to go into mode 4 at end of cycle with 700 ppm of water, 736. That's the number, proprietary number.

MR. SCHMIDT: Right. But the overriding is your trans -- is your tech spec k equals .95. Whatever you have to do to do that is what guides you.

CHAIR KIRCHNER: Yeah. It's probably more boron then that.

MR. SCHMIDT: Yeah. I'd have to look at the tables.

MEMBER BLEY: And mode 4 requires that you've -- your CRAs are incapable of withdrawal.

MR. SCHMIDT: That's right. You --

MEMBER BLEY: You've disconnected the CRVM.

MR. SCHMIDT: You can count those, as I

mentioned, you can count it. So the k effective it goes .95 includes your reactivity associated with your control rods and whatever boron concentration.

CHAIR KIRCHNER: Yeah.

MR. THURSTON: Okay. Carl Thurston, Reactor Systems. So, I'm going to speak at a high level for the impact of Chapter --

CHAIR KIRCHNER: Carl, could you pull your microphone closer? Or speak at a higher level.

(Simultaneous speaking.)

CHAIR KIRCHNER: Use the other microphone. That one doesn't work.

MR. THURSTON: I remember this from the last time. So, Carl Thurston, Reactor Systems. I'm going to speak at a high level. The changes to Rev 3.

So, the staff reviewed the impact to the design changes. Primarily those changes were changes to RELAP5 code from NRELAP 1.3 to 1.4. And then updating the base model from Rev 0 to Rev 2.

I think NuScale did a fairly good job of describing the changes made. So, we won't go into extended details now since they did a good job of explaining most of the changes.

The other changes included ECCS logic. Changes to the IAB. The block pressure and the release pressure is changed. Had minimal impact on the results that we'll see. Some changes with DHRS logic. That's what they discussed.

So, staff reviewed the underlying calculations. And we'll review those on the next slide.

MEMBER MARCH-LEUBA: Do you, Carl, remember why there was a new OI? Because those look to me more like an open item.

They decided to change their logic, and I have not evaluated it.

MR. THURSTON: Yes. That's correct. So essentially the changes have been made. But, we have not audited all those calculations at that point. So, they became open items.

MEMBER MARCH-LEUBA: But, back when we were talking in July about all those things, those use, in my mind, remained unresolved.

We -- I mean, and that's what the SCR says. Is we don't have a path forward. And these were not really used. It was really old.

MR. THURSTON: They were really open. I don't know if Becky wants to chime in.

MS. PATTON: so, it was based on a lot. It was, like I said, it was not necessarily the significance of the issue. It was the state we were in, what was left to be done. And whether or not there was like a defined path forward.

So, we hadn't seen at that time, like the calculations hadn't yet been performed. So, that was determined then to constitute unclear based on the -- it was more of a schedule risk determination.

That's what -- that's what differentiated regular OIs from UOIs. So, that's --

MEMBER MARCH-LEUBA: But more an impact on the schedule.

MS. PATTON: Yes.

MEMBER MARCH-LEUBA: Then any of the thing it was going toward.

MS. PATTON: Yes. Or which was anything we either didn't have agreement on. Or things that were like a large amount of work moving forward.

MEMBER MARCH-LEUBA: That's not the impression you left with me. And I assume with the public that would be, can be -- and just the members that can be following this.

Because those UOIs, they are special OIs. And they can all -- how did they all get resolved so fast?

MS. PATTON: So, it -- I mean, like I

said, so the determination was, you know, you have to have a certain, it was based on phase discipline.

Okay. And that, so that was, you know, that's how the office is operated at phase two. You're supposed to have like a clear path to resolution.

And what that means is that, you know, it's supposed to be straightforward to resolve those between the end of phase two and phase four.

So, when it's analysis that you have not yet seen, so you don't know how much they're going to change, whether you're going to potentially have new questions.

So, it's not supposed to be stuff that are expected to potentially generate new RAIs after phase two.

That's why I said, it's more of like a schedule risk type of a determination. Not necessarily, you know, anything.

That's -- it's based on phase discipline and maintaining that.

MEMBER MARCH-LEUBA: As I mentioned earlier, I have the pleasure of writing the draft letter for this. And I'm going to have to describe how we resolved these UOIs.

And the more I look at them, the less view I give them. I want to try to describe it in a positive way.

Okay. It was more of a scheduled risk rather than a clinical risk.

MS. PATTON: It was maintaining phase, phase discipline. So, you know, what happened between the end of phase two and the end of phase four, is we conducted, you know, detailed audits of those calculations.

The results of those were updated in the DCD. And you know, we reviewed those updated summaries as well.

So, that was all that basically what happened in that last phase. And like I said, it's not normally intended that that much work and potential for new RAIs could happen in that phase.

MEMBER MARCH-LEUBA: Take for example the ECCS logic change. It's just coming near like that. If we -- it's an open item, because in July when we wrote phase two SER, we didn't know it. We hadn't resolved.

But, if we review it, do some preliminary analysis and it doesn't work, they can always go back to the previous one. So, it was not a safety risk.

MS. PATTON: No, no, no. So not safety risk, right. So, you know, it's based on, like I said, the definition in terms of project management between phase two and phase four.

And so your -- when you have, when you know and you've been notified that you're going to be seeing a new analysis rerun with potentially different logic or different set points, then you have to ask, you know, is there an expected potential I might have an RAI?

And when you haven't seen the analysis yet, you have to believe there's that potential. And the need to write RAIs in that phase is considered, you know, not to be within the phase discipline.

So, that's how those were established.

MEMBER MARCH-LEUBA: Well, for instance like an abundance of caution.

CHAIR KIRCHNER: That's right.

MEMBER MARCH-LEUBA: The label it was new was an abundance of caution. You didn't for 100 percent knew how we're going to solve that.

But, you had a strong suspicion that it was going to work. Right?

MR. THURSTON: There were things we didn't know.

MEMBER REMPE: That suspicion if they hadn't seen the analysis.

MR. THURSTON: We hadn't seen the analysis.

MEMBER REMPE: It sounds to me that she was just doing what the rule said.

MEMBER MARCH-LEUBA: Yeah. Well, to everybody those UOIs sound very bad. Right? It does. We need to address it.

And I think that would be one way to address it.

MR. THURSTON: Right. So, in continuing, NuScale essentially recalculated all of the Chapter 15, 15.5, 15.6 events with the change in the RELAP5 code and the latest Rev 2 of the NRELAP5 based model.

So, staff presented that information a couple, a week or so ago in a subcommittee meeting for the topical report.

Again, for the ECCS logic changes, NuScale removed the actuation on riser low level. So, it's retained as a non-safety related sensor, but it is not an input to the ECCS logic.

That activation was not used in any of the Chapter 15 analysis. So now the ECCS actuates on loss of DC power, high containment level, or low AC voltage after 24 hours.

And there was a mistake on this slide, because the trip set point and the containment increased from 220, the minimum from 220 to 264. Which is an increase of 44 inches, not 24 inches.

The other changes included the block and release set points. So the release set points decrease from 1000 to 950, plus or minus 50, which is the low point now, is 900.

And the block essentially is not -- it's not used unless you assume a loss of DC power, which is very unlikely.

But, if you do, then that -- the valve will trip. And the IAV block will activate to block the opening of the valves.

MEMBER MARCH-LEUBA: But even on this small break LOCA, the IAV is not great? Well, the small break LOCA, when the level goes past the, not the set point, but the -- the set point. You know what I'm talking about?

It goes higher than that and doesn't trip. Isn't the IAV holding it?

MR. THURSTON: Yes the IAV is holding it. That's right. Because the set point hasn't been reached. So we have a slide to show that.

MEMBER MARCH-LEUBA: Actuation first? That's the new --

MR. THURSTON: Yes. It can hold it. That's what I was -- that's what I said. Okay. Next slide.

Okay. So this is an example. So this is the limiting small break LOCA, which is injection break 5 percent.

We can see that the blue line is the reactor coolant pressure. And the red line is containment pressure.

So, you can go past several of the ECCS set points. Why is it not moving? All right.

So, here is RCS pressure. And the minimum ECCS setpoint is 264. We can see that that's bypassed.

So, if the ECCS tripped at that point, the IAB would hold the valve closed. And the next setpoint, the nominal setpoint for the revised ECCS setting is 282.

This is not really helping.

(Laughter.)

MR. THURSTON: And the EC -- the valves will continue to be blocked. And even at their highest setpoint, at 300 inches, the valve will remain engaged.

And so, we can see here that the valves don't open until it reaches 318. And then the DP is satisfied so the valves can open.

MEMBER MARCH-LEUBA: And this is a liquid phase LOCA?

MR. THURSTON: It's a liquid LOCA. It's an R -- CVCS injection.

MEMBER MARCH-LEUBA: Because if it was a steam phased LOCA it would depressurize much faster?

MR. THURSTON: It would depressurize a lot faster. And the level would come up much slower.

MEMBER MARCH-LEUBA: So this takes roughly four hours?

MR. THURSTON: Yeah. Right. Okay, next we'll review the DHRS changes.

So, as NuScale indicated, the DHRS signal is going to be split into two. The DHRS actuation and a secondary side isolation signal.

So, the number of inputs to the DHRS is reduced from 13 to four. Those four now are RCS high pressure, RCS high temperature, high steam pressure, and low RCS voltage to the batteries.

This will function to open up the DHRS valves and close the primary and secondary MSIV. And bypass main feed water and main feed water reg valves.

And stagnant release pressures also were considered in the REV 3 analysis, or some of the Rev 3 analysis. Question?

MEMBER MARCH-LEUBA: Can you go back to the slide with the figure?

MR. THURSTON: Yes.

MEMBER MARCH-LEUBA: DHRS is not active here, right? DHRS is assumed failed?

MR. THURSTON: DHRS is not assumed to be active for any other LOCA analysis. Not credited.

If DHRS was on, the depressurization would be much faster. And so you would actually lose less inventory, right, because it's going to activate earlier.

MEMBER MARCH-LEUBA: This is where I was going to the condition report. You have roughly four -- I mean, I don't know how early you would uncover the rising in this transient.

Because you're losing inventory. So your wet level is dropping.

MR. THURSTON: Yes. It's coming down.

MEMBER MARCH-LEUBA: So, at most you'll have at most four hours of operation with the rising uncovered. And the pressure is going down, which makes me believe that you have sufficient contact with the pressure vessel surface.

And the rising water level and containment to condense all of the steam on the vessel wall. Even though DHRS is not working.

The pressure is going down. You're condensing the steam inside the vessel.

MR. THURSTON: You're condensing the steam, sure.

MEMBER MARCH-LEUBA: All that steam is going into the downcomer. I mean, if you condense it, so where is condensing on the wall?

MR. THURSTON: It's going into the containment. Right, because you're still blowing down.

MEMBER MARCH-LEUBA: No. No. The liquid --

MR. THURSTON: It's not going to the downcomer.

MEMBER MARCH-LEUBA: The liquid there.

MR. THURSTON: There was some, some steam that will go onto the tubes, right, and will condense into the downcomer. But most of the -- well, so this is --

CHAIR KIRCHNER: So you've already lost

this to containment.

MR. THURSTON: Yeah.

MEMBER MARCH-LEUBA: But you are still generating steam.

MR. THURSTON: Yes.

MEMBER MARCH-LEUBA: For all this time. So, I suspect the steam pressure is the one that counts. Not the volume of liquid that you're dumping.

MR. THURSTON: All right. This is -- this is steam. This is cover riser pressure.

MEMBER MARCH-LEUBA: Okay. Potentially you've got up to four hours of steam condensation on the pressure vessel wall falling down into containment, into the downcomer potentially. Likely won't. But potentially you can.

At time 1300 seconds, you dump all that water that you had in containment. You open the flood gates to the RV and whoosh, race everything into the core.

In this case, the water you have in containment is borated. But, potentially the bottom part of the downcomer is unborated.

So, that's something for the official report and that needs to be evaluated.

MR. THURSTON: Sure.

MEMBER MARCH-LEUBA: And probably the condensation on the vessel wall is not sufficient. And if you do have DHRS working, then it will happen much faster.

But, it has to be -- I cannot do it in my head.

MR. THURSTON: Yeah. We are looking at that.

MEMBER MARCH-LEUBA: It has to be calculated. So whenever you guys do the audit for the condition report, please look into this.

MR. THURSTON: Okay. So, we're talking about the secondary side, actuation signals. So, those 12 signals now are moved as NuScale showed.

And they only isolate the feed water and steam valve closure. So you bottle up the steam generators.

And this is -- and most of the time the actuation of DHRS will then be pushed to high steam pressure.

So, it allows for better operator control at start up. And it reduces the frequency of DHRS actuation.

Delays RCS actuation until much later in the transient. Minimum change to Chapter 15 figure merit margins. Next slide.

So, here is a review of the effect of the LOCA transient in Chapter 15.6.5. This shows the minimum CHF criteria changing from 1.8 to 1.72.

A relatively small difference. And this difference is due too again, the changing of the release setpoint of the IABs. Primarily because you're losing more liquid into the containment.

And then NuScale also discussed the change in the method for calculating collapsed liquid level in the core. So that changed from an actual based method to a volume based method.

It increased the level from very small margin, 0.14 feet to 1.7 feet of collapsed liquid level.

So overall there are minimum changes too minimum CHF. The margin is slightly reduced. The minimum collapsed level increased, but due to a change in methods.

CHAIR KIRCHNER: What Carl -- Carl, what uncertainty would you put on that? I mean, that was skirting their criteria in the first Rev 2, .14 feet, a couple of inches.

MR. THURSTON: That is correct.

CHAIR KIRCHNER: I mean that, given all

the uncertainties, that's --

MR. THURSTON: That's right. It's very close.

CHAIR KIRCHNER: Including the uncertainty in the code's ability to --

MR. THURSTON: That's right.

CHAIR KIRCHNER: Predict that collapsed level.

MR. THURSTON: So, staff looked particularly at the CHF correlation that NuScale used. And we thought that the correlation was -- had adequate conservatisms that even if NuScale violated their top of fuel criteria, that the CHF margin was conservative enough to protect SAFDLs. And protect the CHF margin.

MEMBER MARCH-LEUBA: But the change from .14 to 1.7 was mostly IAB setpoint? Do you think?

MR. THURSTON: Which one?

MEMBER MARCH-LEUBA: The collapsed liquid level in revision 2 has -- was ceiling point .14 feet and it raised 1.7.

MR. THURSTON: So they changed how they calculated the collapsed liquid.

CHAIR KIRCHNER: Yeah. They changed it. The method changed.

MEMBER MARCH-LEUBA: Oh, the collapsed level, already, right.

MR. THURSTON: Yeah. So we went through that in pretty much detail last week. Or the week before last.

Do you want -- are you doing this slide? I mean, --

MR. SCHMIDT: Sure. You can get that one. So, long term cooling analysis. So, that's in 15.0.5.

There's two long term cooling analyses or scenarios evaluated. We've talked a lot about them already, the decay heat removal system, and the ECCS. Again, it's 15.0.5 and 15.6.5 for ECCS.

Long term cooling methodology is documented in a technical report. It's part of -- it's referenced as part of DCD Chapter One.

Long term cooling addresses ECCS cooling after recirculation is established. A long term cooling methodology assumes subcriticality. Return to power as we talked about is really addressed by the decay heat, or by the DCD 15.0.6.

Phase two SER included an open item on the long term cooling technical report that stated that cooling was demonstrated to 30 days. NuScale revised the statement. And the staff SE documents the review to 72 hours for the long term cooling analysis.

So, that's effectively parsed in 15.0.5 and 15.6.5. And the scope of the analysis goes to 72 hours.

MEMBER MARCH-LEUBA: So, they changed 30 to 72, because --

MR. SCHMIDT: Yes.

MEMBER MARCH-LEUBA: They couldn't --

MR. SCHMIDT: No. So think the situation was, we were also reviewing this in parallel with the boron redistribution issue.

MEMBER MARCH-LEUBA: Um-hum.

MR. SCHMIDT: And we weren't sure how that was going to turn out. So we to simplify the review and process we asked them to denote that only out to 72 hours, because that's where we were likely to come up with the boron redistribution finding. Be consistent with that.

MEMBER MARCH-LEUBA: Is there any regulatory basis for that? I mean, obviously common sense tells you, after 72 hours the operator will be able to do something.

MR. SCHMIDT: Yeah, well right. But, we were just told to get consistent again with like the Chapter 15 analysis.

So, you know, we were not looking at what other actions could happen. We're not even saying that maybe no actions could be required.

Like we talked about a little bit is, you know, they've done a boron redistribution analysis. And it looks like it's good to seven days.

MEMBER MARCH-LEUBA: Um-hum.

MR. SCHMIDT: But, we didn't know that at the time. And this was just trying to make the review consistent with Chapter 15 and where that boron issue was going to turn out.

MEMBER MARCH-LEUBA: Okay.

MR. SCHMIDT: Carl, I think this is you.

MR. THURSTON: Yeah. So, the next slide looks at long term cooling for extended state when you have a LOCA.

So, the long term cooling starts after a quasi-steady state. Recirculation is obtained during the ECCS steaming mode.

So NuScale, they basically ran the NRELAP code with the abbreviated ICA long term cooling model. And they ran that for 12 hours.

It has a collapsed liquid co -- collapsed core and other simplifications for long term cooling. And then they use state point method to infer the conditions at 72 hours.

And that's basically based on the decay heat with whatever multiplier, they assume whether it's a 1.2 for high decay heat, or .8 or lower for low decay heat.

There were three recriticality scenarios that were evaluated in 15.0.6.5. So, that's the open item.

NuScale ran two tests that support the long term cooling assessments. And that was HP 19A and 19B.

Those two, one was with non-convincibles and the other one without. So, we found good comparison between their NRELAP code and the data for HP 19A and B.

The final results, so there's several cases they ran to look at maximum temperature, minimum temperature, boron precipitation, and minimum collapsed liquid level. And these values show the limiting results that NuScale had.

For boron precipitation, the margin was about 17 degrees f, for the precipitation limit and the minimum level, minimum temperature in the core. The top off fuel was not really challenged. It's 2.8 foot.

MR. NOLAN: Presented in this slide are the DCA revision four results for the inadvertent opening of a reactor valve, 15.6.6.

Specifically, these results are for the limiting MCHF case. Which is an inadvertent opening of an RVV.

The changes that resulted here were not necessarily associated with changes to the IAB block and release thresholds, because the limiting MCHF case occurs so early in the transient. It's more associated with the changes NuScale had presented earlier to the base model.

The staff did perform audits of the underlying calculations. And we found that the applicant's analysis was acceptable. And we concluded that the resulting figures of merit are within their design limits.

And if there's no questions on this event, this concludes sort of group one of the staff's presentation. We need to go through a tech staff shuffle at this point.

CHAIR KIRCHNER: Let's take a break here while we do the shuffle. And then we'll continue at -- I'm pretty confident about our time.

So, let's reconvene at three o'clock on

that clock.

(Whereupon, the above-entitled matter went off the record at 2:41 p.m. and resumed at 3:00 p.m.)

CHAIR KIRCHNER: Okay. We are back in session. We are going to turn to -- who is first? Syed? Go ahead, please.

MR. HAIDER: Okay. So, my name is Syed Haider. I was given the task of summarizing the entire containment safety analysis review on a single slide.

(Laughter.)

MR. HAIDER: And I've tried to do my best.

CHAIR KIRCHNER: We appreciate the brevity.

MR. HAIDER: So, yes. So, --

(Off-microphone comments.)

MR. HAIDER: Yeah. I mean, if I'm given an opportunity to. So, that will be the only side that I will have on containment analysis.

So, I'm the lead reviewer of NuScale FSCR Section 6.2.1.1 on containment structure, which is mainly related to the peak containment pressure and peak containment wall temperature that results from the respective limiting design basis events involving mass and energy that leads into the containment.

The applicant provided the supporting information in the containment response analysis methodology or CRAM technical report that is incorporated by reference.

The staff reviewed various considerate decisions used in the applicant's NRELAP model, and its initial and boundary conditions.

NuScale application had provided NRELAP modeling reserves for seven containment design basis events that included three LOCA breaks, two AOO events involving inadvertent opening of RVV and RRV valves, one main steam line break, and one feed water line break.

The staff performed confirmatory analysis using MELCOR and TRACE for the same spectrum of seven containment design basis events.

Both the applicant and staff ran numerous sensitivity cases of the peak containment pressure and water temperature to investigate several key factors such as the condensation heat transfer modeling on the containment inside surface effect of non-convincible gases, nodalization of containment volume, and heat structures, and liquid permanent stratification affects inside the containment as were as the cooling pool.

In general, the peak containment pressure and water temperature were found to be not much sensitive to these factors to significantly reduce the safety margins.

After phase two, the applicant modified and the staff audited, the limiting peak containment pressure case of one inadvertent RRV opening at full power.

The change was made to account for the IAB release pressure range of 950 psia plus minus 50 psi, and potential for staggered ECCS valves opening at different pressure over the range.

It turned out that the new limiting break containment pressure event with the staggered opening of the remaining RRV and the three RVVs, increased the peak containment pressure from 986 psia to 994 psia, which still demonstrated a 5.3 percent margin.

In the containment design pressure with respect to the 1050 psia containment design pressure, only at 986, which was about 6 percent margin, but when they considered the staggered opening, all the ECCS valves, the pressure increased by 8 psia. But the staff believes that it was not a significant change in the margin.

So, the peak containment pressure of 526 degree Fahrenheit remains unchanged for the CVCS injection line break event. The staff concludes that there is sufficient conservatism built in the containment design conditions, as has been demonstrated by the confirmatory analysis and the sensitivity studies.

And the staff accepts that the NuScale containment design meets all regulatory requirements for sufficient safety margin for pressure, temperature, and heat removal, because there is reasonable assurance of adequate protections.

MEMBER MARCH-LEUBA: We just did this last week, so. Everything sounded familiar. Thank you.

MR. HAIDER: Thank you.

MS. SIWY: Okay. I'm Alex Siwy, Reactor Systems. The staff looked at changes in Chapter 15 since we issued phase, our phase two safety evaluation. And we looked in particular at how the margin to figures of metric changes.

The staff noticed that for the non-LOCA type events there were two static reactivity events that showed pretty large changes in the calculated MCHFR between revisions 2 and 3 of the DCA.

These were the control rod misalignments that's analyzed as part of the control rod misoperation events. And the inadvertent loading and operation of a fuel assembly in an improper position event.

And as you can see, there was a 25 percent to 43 percent decrease in MCHFR. It's also worth noting that the revision 3 results are -- or revision 4 results are the same as the revision 3 results.

MEMBER MARCH-LEUBA: But you have such a large change back on the two events, but not on the other events?

MS. SIWY: We see --

MEMBER MARCH-LEUBA: I mean, the peaking factor is sufficient to give you this.

MS. SIWY: Right.

MEMBER MARCH-LEUBA: For the events you get the same peaking factor, don't you?

MS. SIWY: This is -- so, for these two events, the large difference is primarily due to the -- the radial peaking augmentation factor that's applied to these two events.

MEMBER MARCH-LEUBA: Is the factor of two that we had before?

MS. SIWY: It's not a factor --

MEMBER MARCH-LEUBA: So, sorry, that was steam power.

MS. SIWY: Okay.

MEMBER MARCH-LEUBA: Yeah. Okay. This is radial peaking bundle to bundle.

MS. SIWY: Yes.

MEMBER MARCH-LEUBA: Okay.

MS. SIWY: So yeah, the staff audited the underlying calc notes to confirm that. And to answer your question, you know, maybe you're thinking a single rod withdrawal event like NuScale brought up earlier.

MEMBER MARCH-LEUBA: Um-hum.

MS. SIWY: And it's because each of these events uses a different -- different peaking assumptions. So, in this case, it just -- it had more of a pronounced effect here.

MEMBER MARCH-LEUBA: Okay.

MS. SIWY: And another factor that led to some of the differences was the use of a more conservative core inlet temperature. It increased by a little over 10 degrees. So, that would also led to a decrease in margin.

So, overall the staff finds that the inputs and assumptions applied to these analysis are acceptable. And the resulting MCHFR still remains above the 95/95 limit.

MR. BARRETT: Yeah. This is Antonio Barrett of the staff. And the staff also reviewed the 15.1 and 15.2 non-LOCA events, which are the increasing cooling events and decreasing cooling events between DCA Rev 2 versus Rev 3, and noted that there weren't significant changes to the figures of merit.

The table that's presented here on this slide is the results for the 15.1.5 steamline break analysis results from DCA Rev 2 versus Rev 3. The figures of merit that we looked at were minimum critical heat flux ratio, the max RCS pressure, and the max steam generated pressure.

And you can see that there weren't significant differences. The staff audited the underlying calculations for the different events in 15.1 and 15.2 for Rev 2 versus Rev 3. And again, didn't notice any significant impacts.

There were some small impacts observed due to the DHRS logic change for RCS pressure and steam generator pressure. But, in general, they were pretty small.

Whenever the DHRS would come in a little bit later then it normally would have before, when both the secondary side isolation would happen at the same time as the DHRS. And in some case -- in a lot of cases, they still happen about the same time.

So, the staff finds that all the changes were acceptable. And that the resulting figures of merit remain within their design limits.

DR. LU: Okay. Shanlai Lu from staff division of ECCS system and NRR. I'm going to cover about the water hammer issue.

And it's one of the open items resolved through testing. And at this point, I think we reached the closure with NuScale about that part.

And just to give you a little background and when we asked to launch this RAI into that, that's for typical ECCS system operating fleet of the, you know, we worry about it downstream of the pump. Are there any potential after you, you know, turn on the ECCS, the remaining potential water hammer.

So, for this one, they don't have, you know, downstream and you know, piping system. But, we have the trip valve hydraulic line doing the actual, actuation of ECCS system.

And it may --

MEMBER MARCH-LEUBA: So Shanlai, this is the line that goes from the actuation valve outside containment --

DR. LU: That's right.

MEMBER MARCH-LEUBA: To the IAB?

DR. LU: From IAB to -- onto the trip reset valve entry, you know, set valves there. So it's, there's a long line.

MEMBER MARCH-LEUBA: Um-hum.

DR. LU: So, and then each valve has different length into that region.

MEMBER MARCH-LEUBA: And why does it flash? Because the containment is getting hot when it has a LOCA?

DR. LU: Once the ECCS -- well, the reason it starts flashing is because of during the normal operation, it's the experience of the high temperature and high pressure.

MEMBER MARCH-LEUBA: And even with a vacuum of a critical containment?

DR. LU: Right. Because of radiation. That's where we --

CHAIR KIRCHNER: It's attached onto the outer part.

DR. LU: It's attracting in the gap between the containment and the outer part.

MEMBER MARCH-LEUBA: So it's continuously boiling?

DR. LU: What?

MEMBER MARCH-LEUBA: It's continuously boiling?

DR. LU: No. No. No, there's no -- unless you -- you know, it's just like a -- it's pressurized too.

MEMBER MARCH-LEUBA: Yeah. But that wasn't really my question. Where did the steam go?

DR. LU: Well --

MEMBER MARCH-LEUBA: If you have a solid line, you cannot flush it.

DR. LU: Right. But then once you open that one, you have a trip reset valve. And then that for the hydraulic you have fluid that first started to move in a flash.

MEMBER MARCH-LEUBA: Yeah.

DR. LU: So, that's the exact reason I think we, from my perspective, because we are wanting what's the water hammer?

MEMBER MARCH-LEUBA: Um-hum.

DR. LU: Do we have water hammer through this hydraulic line? And are there any potential water hammer issues?

And then before the testing, when Tom asked for the specific demonstration testing, and we did not have any information on the table to make a decision and then whether water hammer can be resolved, or is that the issue or not.

And then so after Tom asked for the requested tests, which is a wonderful idea. And then so we added this on top of what he asked, because we also want to know what's hydraulic in line with water hammer issue. And whether it has any impact on the valve performance.

MEMBER MARCH-LEUBA: Anything with two phase flow with compressibility --

DR. LU: Yes.

MEMBER MARCH-LEUBA: It's hard to see that water hammer will be a problem.

DR. LU: That's right. So that's the exact reason we were looking into this one with an area on that.

MEMBER MARCH-LEUBA: But, what I am concerned is when you open the pilot valve, and the recent valve, --

DR. LU: Right.

MEMBER MARCH-LEUBA: And you start flashing, then you're not going to get the same flow out of the IAB that you did in testing and with cold water conditions.

How do we know that the IAB and valve

work?

DR. LU: Okay. Our --

MEMBER MARCH-LEUBA: Right, but --

MR. SCARBROUGH: So this is Tom Scarbrough, --

MEMBER MARCH-LEUBA: But now are you proprietary?

MR. SCARBROUGH: Well actually, when they actually conducted the design demonstration testing, they did apply high temperature pressure water. Right, so it was very similar to an actual installation, right.

Because they had a very long run of the small tubing, right. And with many turns, right. And then it opened up to atmospheric conditions.

But, it was such high temperature, you know, it was flashing. So you had flashing all the way from the main valve, through the IAB, out those -- out that tube line, that hydraulic line.

So, yes, it was able -- the design demonstration testing was able to show whether or not you would have water hammer.

MEMBER MARCH-LEUBA: And that was -- well, I'm not worried about water hammer anymore. I'm worried about whether the IAB and the valve work.

MR. SCARBROUGH: Well, that's -- that was the whole goal of the design demonstration testing. And it showed it did work.

I mean, they had some issues. And we talked about those issues.

MEMBER MARCH-LEUBA: Um-hum.

MR. SCARBROUGH: And they had to make some adjustments in design and they had, you know, different things they had to change in the design. But, at the end of the day --

MEMBER MARCH-LEUBA: But would it work at different temperatures?

MR. SCARBROUGH: No, they were -- they were working at basically reactor, you know, similar to reactor conditions.

MEMBER MARCH-LEUBA: But what if it is colder?

MR. SCARBROUGH: I'm sorry?

MEMBER MARCH-LEUBA: What if it is colder? I don't know why, but --

MR. SCARBROUGH: Oh, okay.

MEMBER MARCH-LEUBA: But, I mean, how much sensitivity to a -- well, how much do the conditions need to change before the valve won't open?

DR. LU: Okay. Let me answer that.

MR. SCARBROUGH: Go ahead.

DR. LU: So, basically that's why we asked for the simulation of the actual pipelines through the gap. What's the temperature? What's might be the -- pressure is really at the primary site for the pressure.

MEMBER MARCH-LEUBA: Yeah.

DR. LU: And then the temperature, they did a hand calculation. And then figured out what might be the bounding temperature for that one.

And then based on the radiation heat transfer they determined there was a bounding temperature. It's lower than the primary side temperature.

But, to still be, you know, as it goes through, it's supposed to have a gradient. But at the higher the temperature and then the, and you know, more challenging for the water hammer or flashing issue in terms of water performance on that.

MEMBER MARCH-LEUBA: We need to have a whole session on ECCS valves maybe Wednesday, I believe. And --

DR. LU: Wednesday?

MEMBER MARCH-LEUBA: And it will be proprietary, maybe I'll ask my questions then. Not used, on the PRA.

(Off-microphone comments.)

MEMBER MARCH-LEUBA: Tomorrow afternoon?

DR. LU: Tomorrow afternoon? Okay.

MEMBER MARCH-LEUBA: I'll probably ask the questions there. But, I see some variability on the flow that's going to go through the IAB depending on what your conditions are.

I never thought of all this flashing.

MR. SCARBROUGH: Well, they had a whole range of conditions they tested at. We had -- on February 4 we went through all of those various test runs they made, right.

They made like 63 test runs at various conditions, right. And they found some plot times when it didn't work very well, right.

So, they had to go back and make a number of changes in the design, right, to have it work, right. So, but we can go back through some of that if you'd like.

MEMBER MARCH-LEUBA: My question is going to be, when you consider the uncertainties, how much confidence do you have that the valves works?

MR. SCARBROUGH: Well, that --

MEMBER MARCH-LEUBA: Uncertainty meaning

it has uncertain operating conditions. You made some hand calculations.

MR. SCARBROUGH: Right.

MEMBER MARCH-LEUBA: I made some hand calculations before that happened to be wrong. Right? I'm sure you've done it too.

MR. SCARBROUGH: What they, you know, this is just design demonstration testing to show that the physics works, right.

MEMBER MARCH-LEUBA: Um-hum.

MR. SCARBROUGH: They still have to go through QME-1 qualification testing of the various ranges of temperatures that it could see and such, right.

So they're -- there's still a lot more qualification work that has to be done for this valve system, right. It's not done yet.

This was just to show that the physics works. And they can demonstrate it through the qualifications.

MEMBER MARCH-LEUBA: Not done yet?

MR. SCARBROUGH: No. They will have to do qualification testing.

DR. LU: That's the next slide.

MR. SCARBROUGH: This is just design

demonstration testing.

MEMBER BLEY: So go on after design cert?

MR. SCARBROUGH: Oh, absolutely. Yes. There's a lot more review that the staff will do of the qualification.

Actually, that's -- are you done with your slide?

DR. LU: Yes. I'm done with water hammer. Let me --

MR. SCARBROUGH: Okay. We'll jump to this one then.

DR. LU: Okay. Yeah, go ahead.

MR. SCARBROUGH: Because during the, we had some follow up questions from our February 4 discussion. And one of them had to do with the small, the lines, the tubing, and the ports and such.

This was a significant discussion item for our ECCS valve, design demonstration audit. And also the FMEA audit we had.

And in the latest report, there were like four pages where we went through, what were the various ideas? Because they've already, they had developed draft qualification plans.

But, you know, so we went through these questions. And a lot of these questions had to do with like the plugging, the fouling, and things -- of course boron precipitation. All of these issues.

Now, in QME-1, to meet the QME-1 standards, they're going to have to look at internal, you know, clearances and tolerances.

They're going to have to look at environmental conditions, the fluid conditions, all that has to be demonstrated as part of the qualification testing.

And so that's the -- now, that's the next step. So, we did ask them about their precipitation temperature. You know, would the temperature always be above precipitation for the boron?

And that's going to have to be addressed during the qualification testing. They're going to have to show that over a significant period of time that they're not going to have, you know, boron precipitation.

It's going to have to be part of the qualification plan. And we've identified these in the audit report.

And then the last item there had to do with could there be an accumulation of boron or different particles during plant operation now.

They are going to flush out these lines, because to be able to reset the valve system, they're going to have to fill the lines back up with CVCS water. And force the valves to close.

And so -- and so what they're going to do, is it's going to flush all that through. So, they are going to be flushing the lines before every restart.

So, they -- but they're going to have to show that over the time frame of the plant operation that you don't have precipitation or any particulates or something built up in the lines.

So, there's still a lot of qualification work that has to be done on these valve systems. So, it's -- this is -- this is focusing on what was, you know, design demonstration was for design certification.

And for qualification, that's the COO applicant. And then ever after that, those ITAAC are going to have to be addressed.

But, for the COO applicant, we'll be reviewing their updated qualification plans to make sure that they've incorporated all of these types of lessons learned in the plan as part of the COO applicant review.

So, there's still a lot of work to do. But, we did go back and we did ask these questions, because they're going to have to demonstrate it as part of the qualification.

CHAIR KIRCHNER: So, the qualification plan is captured how? You mentioned also an ITAAC. So, the qualification plan is -- is a condition on the DCA? Or on the --

MR. SCARBROUGH: Well, as part of the normal staff review, in Part 52, when you get to the COO application review, there's a provision in there which says that the NRC staff may need to audit, you know, procurement specifications and things of that nature.

CHAIR KIRCHNER: Okay.

MR. SCARBROUGH: And how we, as part of the audit process, we look at the qualification plans to make sure that the FSAR information, the design specs, and the qualification plans are all consistent with the operation of the valves.

So, we'll be looking at that as part of that review for the COO applicant. To make sure that they have qualification plans for the valves that are consistent with the design function of the valves.

So, we'll be doing that.

MEMBER BLEY: Tom, do you expect to be able to observe any of these testings?

MR. SCARBROUGH: Oh, absolutely. We're going to -- we're going to ask about that. Because we -- with AP1000, we observed the squib valve testing.

MEMBER BLEY: Right.

MR. SCARBROUGH: We made sure that the designers know, Westinghouse knew we were interested in observing the testing. And we went down to Wiley and watched it there.

So, we plan to do the same thing. Because they're both first of a kind. And so the staff is going to want to observe that testing.

MEMBER BLEY: Excellent.

CHAIR KIRCHNER: And the ITAAC that you referenced associated with this, what does that entail?

MR. SCARBROUGH: Yeah, the ITAAC is a requirement that for qualification that the valves are demonstrated to be capable. And from the discussions with NuScale, and they have a specification in the FSAR that they will meet QME-1.

And so to satisfy that ITAAC, they will have to be able to demonstrate that they have satisfied the QME-1 qualification testing.

CHAIR KIRCHNER: Um-hum.

MR. SCARBROUGH: And that's what we're

doing right now with Vogtle 3 and 4. We're conducting inspections.

We were just at Cranberry back in November, looking at their qualification to make sure they were consistent with their requirements to meet QME-1 as part of an ITAAC closer process.

So, that's what we'll be doing.

CHAIR KIRCHNER: Okay.

MR. SCARBROUGH: So, there's still a lot more review to do.

MEMBER MARCH-LEUBA: One concern we always have is you have a highly rated water, because you are refueling duration. It's stagnant for two years in a highly ionized environment.

How are you going to handle that?

MR. SCARBROUGH: They will have to demonstrate it during qualifications that they will can handle that over that length of time. Because they have to be able to show that the valve can operate over that -- over that two year period --

MEMBER MARCH-LEUBA: But the test is for two years?

MR. SCARBROUGH: For the end of cycle.

MEMBER MARCH-LEUBA: They have to test it for two years?

MR. SCARBROUGH: They're going to have to figure out some way to show that there's no degradation over time. Now, I'm not sure how they're going to do it. But --

(Simultaneous speaking.)

MEMBER BLEY: Have you done this before? Extended life kind of testing?

MR. SCARBROUGH: Yeah. They're going to have to -- what typically what they would do is --

MEMBER BLEY: I really liked what you had to say about this last time.

MR. SCARBROUGH: Yeah.

MEMBER BLEY: They've never said this stuff to us, so.

MR. SCARBROUGH: Yeah. What they're going to have to be able to show, and they had to do the same thing when they were doing squib valves. They have to show that over a long period of time that they don't have like, for the squib valves, that they didn't have leakage.

You know, so they hold it in a compressed chamber for a long period of time to show they -- and actually they had, you know, they had a leakage. But, they dealt with that.

So, they will have to show that over a

long period of time. Now, they might not have the full system, but they're going to have to show that over a long period of time at the conditions that this thing will be sitting at, they will not start to have precipitation.

So, that's going to be part of their challenge in terms of developing a qualification plan. But, that's something we'll be reviewing when they get to that point.

DR. LU: So, in conclusion basically, water hammer issue as an open item was resolved based on actually quite a lot of testing. They did 63 for?

MR. SCARBROUGH: Sixty-three tests.

DR. LU: Sixty-three tests that they put pressure tap there. For temperature distribution they put, you know, on the thermal couple and measured the temperature. And also electrical all heating system to maintain the temperature.

So, that's a bounding, you know, information and it might be a hand calculation, but it's bounded to make sure that there is no water hammer issue at high enough so that it's going to cause the flashing.

So, we based it on that one, I think the staff closed that open item. And then we felt that there is no issue, particularly of interest that relate to water hammer.

Any questions?

CHAIR KIRCHNER: Just as a practical matter, it seems like to me that this is in effect going to require them to take off the ECCS valves each reviewing.

MR. SCARBOROUGH: For the first --

CHAIR KIRCHNER: For only the first time?

MR. SCARBOROUGH: Yes, what they're going to have to do, and because of the ASME OM codes requirements for these valves, they submitted alternative request that for the first outage, they would take all five off and run full diagnostics on all five.

And then, the opinion on timing of when the other modules come into play, it's possible that they might have to take another five off in the next outage, but if they have other modules come into play the next step would be that they could take like one RPV and one RRV off, and they would sample that and show that there's no degradation in the operation of that valve by the threshold of relief pressures and such in diagnostics.

After that, over a time period, once they get data they can demonstrate a longer interval because they can sample it and the code allows a sampling process.

But for the first time, the first time they run the first unit, they're going to have to take all five off and test them all. Because we're not, you know, we want to verify that you have all that qualification was going to be proper and that there's no degradation over time. Definitely for that first one, and then they'll sample after that.

CHAIR KIRCHNER: Okay, thanks. Keep going?

MR. TANEJA: This is Dinesh Taneja from the NRC branch. I just want to give you, I guess, what the design contains on the containment vessel and the reactor pressure vessel level instrumentation.

For both of these measurements, the design has these guided-wave radar technology for measuring the level. Containment level, basically the range is 683.5 inches from the top of the containment, which is approximately 220", I guess, from the bottom of the containment vessel to the top.

The functions that they perform is the ECC saturation is one, and they also have an interlock. I think that is their -- one of the other functions that it is used for, in addition to that is PAMs variables by B, C, and D.

So Type B, C, and D of PAMs variable essentially means that it's there to assure the operability of the safety function. That's really all they do there.

Type A is the variable that says that this indication is required to perform a credited manual operator action, so these are not used for that.

So they are classified as safety related, type A1. The interesting thing about the next pressurizer level and the RPV riser level, it's the same exact instrument. We call them two separate things but it's all transmitters, they are four separator transmitters, they go from the top of the reactor vessel to the, it's basically 554.9" is the measuring length of it. The top portion of it, that's in the pressurizer, is scaled to measure the full measurement of the pressurizer level, and the rest of the measure, the RPV riser level.

What they did is, when they changed the logic, they declassified the function to be non-safety related, but it's a safety-related instrument. The function is strictly, the RPV riser level is not used for any initiations, it's strictly for PAM-variable measurements, so they declassified the function to be

B2 but it's a safety-related instrument.

MEMBER MARCH-LEUBA: Apparently they are also used for recovery.

MR. TANEJA: Well, it's there. Recovery, what I understand recovery is now, it's post, you've already basically gone beyond your accident and you are in recovery mode. This really is, there's no action required to really, so there is no Type A variable. There is no operator action for mitigating any of the accidation events.

MEMBER MARCH-LEUBA: But if recovery is messed up, the potential for a very serious accident exists.

MR. TANEJA: Recovery should really start when you've actually reached a condition which allows for recovery. You will not start recovery mode unless our plant is considered to be in a safe state.

MEMBER MARCH-LEUBA: I will 100 percent agree with you if it wasn't for the time when they call a safe, unstable condition, it ain't. It takes physical recovery to put it in a safe and a stable condition. This is nomenclature, right?

CHAIR KIRCHNER: Jose, I agree with the problems that you've identified, but I don't see how the level indicator would change, would provide you with the information about where the boron --

MEMBER MARCH-LEUBA: It provides you with the indication that you need to worry about it.

CHAIR KIRCHNER: No. No.

MEMBER MARCH-LEUBA: You uncovered the riser, therefore you have to worry about it. you shouldn't uncover the riser because you get --

CHAIR KIRCHNER: Well, you'll know that from other --

MEMBER MARCH-LEUBA: From what?

CHAIR KIRCHNER: You'll have the containment level. There's a mass balance. Steam isn't going to have much of the mass, so you're going to have a containment level.

MEMBER MARCH-LEUBA: You will have these things without having an opening, and have a, yes, the answer is cooling. There is no containment involved,

CHAIR KIRCHNER: That one room ---

(Simultaneous speaking.)

CHAIR KIRCHNER: --- set for separation.

MEMBER MARCH-LEUBA: Yeah, it is.

CHAIR KIRCHNER: That the LOCA does.

MEMBER MARCH-LEUBA: The test cooling, within two hours you are separating.

MR. TANEJA: Buy you know these are four

independent circulatory sensors, so they are designed and qualified to be available under all postulator conditions.

MEMBER MARCH-LEUBA: Am I reading this correctly , that the sensor is the same, the function is different? You're saying here that the recuse uses the same transmitter for the safety level pressurizer level than for the riser. It's the same guy.

MR. TANEJA: Same radar guided wave from the top.

MEMBER MARCH-LEUBA: And it's the same two, the same --

MR. TANEJA: It says the same exact. There are only four level transmitters in the design. They are basically inserted from the top, so it's coming down, so this portion of the top -- Go to the next slide, I think I put a pictorial.

So you know, the top portion, the pressurizer, it just scaled to measure pressurizer and that's where all your actuation or safety functions are. Then the rest of it still measures it, it's just they degraded it basically because it doesn't have a safety function so they're calling it a non-safety related function.

The instrument itself is a Class 1A

safety-related qualified to operate in that given environment under all postulator conditions.

MEMBER REMPE: Could you to slide 33 real quick? Some of my confusion is that I'm looking at the first bullet under EECF logic changes, and this distinction of function versus sensor gets blurred a lot, right? And this is an example of where it got blurred.

MR. TANEJA: Right.

MEMBER REMPE: Okay. So, again, I'm kind of thinking that it probably got blurred other places that I can't pull up one in the SCR, but I'm kind of saying it's very easy to get confused.

MR. TANEJA: I know. I know. We were also confused initially, when we got that change and designed while we're taking the signal out, and we are going to move it to containment level and not use the RPV riser level, and we are going to degrade it to non-circulated. I have the same question. I thought there's only four sensors. Are you adding four different sensors for RPV riser?

And then they clarified, no, it's the same sensor, we are just calling that function non-safety related but it's a safety-related instrument.

MEMBER REMPE: Looking to see if I can find that discussion somewhere.

(Simultaneous speaking.)

MEMBER REMPE: Because that's kind of interesting that I share Member March-Leuba's concerns about monitoring versus recovering the reactor, because it's like suddenly now it's important. It wasn't important for a while, but you know how it is now again, because of its function.

MR. TANEJA: Right. So they are circulated instruments and they are really designed to be Class 1A, for the environment of the reactor vessels.

MEMBER MARCH-LEUBA: Why did you make them Class B2? Why is the same instrument Class A1 for the first 130 inches and is Class B2 for the rest?

MR. TANEJA: We have a technical report on the sensor technical report. The sensor technical report called it a circulatory instrument. That's where I think it's basically called out with the specification of the instruments are. Then the B, C, D, the function, they basically call the function being non-circulated for the RPV riser level function.

But the instrument that's used for this, this is the, and I think you guys probably have a copy of it, it's a rev 2 of the sensor technical report. The title is, Nuclear Steam Supplies Us Some Advances technical report?

MEMBER DEMITRIJEVIC: That's a NuScale report.

MR. TANEJA: Right.

MEMBER DEMITRIJEVIC: We were just trying to remember. I'm not sure if we reviewed that when we reviewed Chapter 7, did we?

MEMBER REMPE: It was after Chapter 7.

MEMBER DEMITRIJEVIC: It was after Chapter 7.

MEMBER REMPE: We started asking about the containment, radar-based water sensor. And that's when it showed up.

MEMBER DEMITRIJEVIC: So we didn't actually review it.

MEMBER REMPE: No.

MR. TANEJA: It was part of the DCA.

CHAIR KIRCHNER: Well, I understand that, which means we probably should have reviewed it when we reviewed the chapter. Mike? I'm not sure we have that accessible for all of us.

MEMBER REMPE: The reason why it wasn't reviewed with Chapter 7 was the disclosure being a radar space sensor did not come in until after Chapter 7, as you may recall. Because I know Charles said, wait a second, that wasn't mentioned, and you're right.

MEMBER DEMITRIJEVIC: Is this just about the leveler sensor?

MR. TANEJA: No, it's about all the instruments. Basically this technical report that NuScale put together was for doing the sensor development for this. They are basically, I think there are four phases to it, and you know, the other phase, discipline they call that, it's essentially identifying what technology and then the proof of concept testing and then doing qualification and then taking it through those phases.

So they talk about temperature sensors, pressure sensors, level sensors, and pole sensors. In a hot event they are selecting the technology and now with the proof of concept testing that they are going to put it through. I think Section 7 talks about the level instrumental testing.

MEMBER BROWN: It's a technical report, not a topical report?

MR. TANEJA: It's a technical report.

MEMBER BROWN: Attached to Chapter 7?

MR. TANEJA: It's TR-0316-22048. It's at revision 2 right now.

MEMBER BROWN: Say that again?

MR. TANEJA: TR-0316-22048, revision 2.

MEMBER BROWN: 0316-22048. And what came after the 8?

MR. TANEJA: Revision 2.

MEMBER BROWN: Revision 2, that's rev 2. And it's proprietary?

MR. TANEJA: Parts of it is proprietary in there, yes. This is a proprietary class document. The information that I am sharing with you is not proprietary. I think essentially the performance specifications on accuracy requirements. That's proprietary information. There's some other proprietary information in there.

MEMBER REMPE: I don't have it easily accessible, but be sure you get rev 2, because I can see an earlier version of it that I have accessible. It may be on my --

MR. TANEJA: We'll get it.

MEMBER REMPE: -- anchor that's back in Idaho.

MEMBER BROWN: I looked for it, I don't have it either.

MR. TANEJA: So this really, I'm listening to the previous ACRS meeting, I think that was a confusion that I was getting on the sensors. I wanted to clarify that in today's meeting, as to exactly how these instruments are designed and implemented.

MEMBER BROWN: Rev 0? Is that what you have, Joy?

MEMBER REMPE: Yes, I have rev 0, but I could have rev 2 maybe on my --

MR. TANEJA: This is what they did. They initially submitted rev 0, which was essentially -- And we went through this process of selecting what technologies would work. Rev 0, these are the technologies we've identified and then next rev was like, you know, they've started to do proof of concept testing and we've done that.

Rev 2 was, I think the flow was one of the very critical ones for the RCS flow and that was a lot of the time and money they spent on trying to prove that that would work. It was ultrasonic type of flow measurement technologies. So this is where the technical report was all about.

MEMBER REMPE: What is the date on rev 2, if you have it in front of you?

MR. TANEJA: Let's see.

MEMBER BROWN: May 2019.

MEMBER REMPE: I can't remember when we

became cognizant of the fact that --

MR. TANEJA: That was one of the very early ones that we did, right?

MEMBER REMPE: I think I learned about it later, like in Chapter 9 or something.

MEMBER BROWN: Well, 5/2019 is the date I've got for installing this in my file, of the Rev 0, dated December '16. I'm not sure about that file. Is that what it says on the rev 2 papers, 5/'19? That's the download date for the rev 0 that I have.

CHAIR KIRCHNER: Any other questions?

MEMBER BROWN: It was vague. I remember looking at this. This was not a lot of detail. It was --

MR. TANEJA: Yeah, it's basically a summary of the technologies and what they've selected. But it does give you the safety classification of the instruments and what the ranges are, and all that information.

MEMBER BROWN: My memory also is that when we discussed this at one time, Joy's continual questioning. The accuracy they wanted was something like a half a percent or one percent, something very, highly accurate instrument. Is that in the new --

MEMBER REMPE: Today he said that they have a much broader range. Does rev 2 reflect that range?

MR. TANEJA: You know, for example, if you look at the ECC saturation set point, the range is given 264 to 300 inches, it's like plus or minus 16 inches.

MS. ROGERS: 264 to what?

MR. TANEJA: Plus or minus 18 inches. It's 264 inches to 300 inches is the high level, is ECC saturation set point. So basically, it's the set point when you look at the set point --

MEMBER BROWN: About six percent, then.

MR. TANEJA: It's 282 inches plus or minus 18 inches.

MEMBER REMPE: But the report, when we looked at rev 0 or whatever rev, had a very high uncertainty or limited, so now what I'm asking is does rev 2 say plus or minus 16 inches?

MR. TANEJA: Yes.

MEMBER REMPE: Really. Okay. I've not seen that.

MR. TANEJA: So they really did look at what the performance requirements were from the Chapter 15 perspective and they really broadened the specifications for the performance requirements.

MEMBER BROWN: But it's still only six

percent, and if you go back, I don't know, I remember digging up a whole bunch of stuff on radar in a frothy environment, and they were, some of them were as bad as 20, 25 percent. Plus you add frothiness compensation, whatever frothiness adds. Density compensation effectively, I think that's what that is. It will be interesting.

MEMBER REMPE: And in a radiation environment --

MEMBER BROWN: Well, water doesn't get irradiated that quick, does it?

CHAIR KIRCHNER: Any further questions? Is this, Alex, is this it? You have backup slides, but --

MS. SIWY: That looks like the end.

CHAIR KIRCHNER: What I would like to do, then, is turn now and provide opportunity for any public comment. So I'll turn around in the room and see if anyone wishes to make a comment. Please come up to a microphone, state your name and make your comment. After that we'll go to the public bridge line.

Seeing no one stepping forward, we'll just wait for indication that the public line is unmated. Sounds like that was the signal. Is there anyone on the public line who wishes to make a comment? If so, please state your name and make your comment.

MS. FIELDS: Yes. This is Sarah Fields. I have a few comments. First of all, it would help if presenters identified the slide number they are discussing. Sometimes it's easy to get lost, particularly in the NuScale presentation this morning. So that would help.

And I have a concern about the DCA rulemaking schedule, which the NRC has now posted on the DCA website. The NuScale DCA website. According to the schedule, the rule will be published for public comment in the Federal Register on June 1, 2020. The signature authority will be on March 19, which is just in a couple of weeks. The final rule would then be signed on November 23rd and published on January 27th.

So the public comment period will start before the ACRS completes its final review and before the NRC completes the final SEA. If the NRC provides a 90-day comment period starting June 1, the comment period would end before the September date of the final SEA.

I really don't, and this is basically addressed to NRC staff, I don't understand this scheduling. I don't think it's right to commence the rulemaking until all the i's have been dotted and the t's have been crossed when the ACRS and the NRC complete their final reviews. I would definitely like to have that rulemaking schedule be adjusted and commence after the ACRS and NRC complete the final documents.

Also, on slide 44, bullet 3, said that the NuScale plans to flush valves in lines during the refueling outage, that's a very wrong assumption. NuScale isn't going to be a COL applicant. NuScale is a designer.

This also goes back to the rulemaking issue. There's no hurry to finalize this rule, commence or finalize the rulemaking because it's unlikely that there will be a COL applicant that references the DCA. The only COL applicant or prospective applicant that has stepped forward is the Utah Associated Municipal Power System, or UAMPS.

Both UAMPS and NuScale have indicated that the power rate, or the 12-module reactor they intend to site at Idaho National Lab will be a 25 percent increase on the power that the DCA is being reviewed for.

So, the COL applicant, that is UAMPS, would not rely on the DCA. What they would rely on would be a standard approval and that process has not commenced. According to NuScale, that application is not going to be submitted to the NRC until the latter part of 2021. And of course, the SDA would rely on the design certification but there doesn't seem to be any rush in this process. Both UAMPS and NuScale want to have a small modular reactor that has 25 percent increase in power from that outlined in the DCA. Thank you.

CHAIR KIRCHNER: Thank you, Ms. Fields. Is there anyone else on the public line who wishes to make a comment? Okay, with that I think we can -- Yes, we need to make an announcement before we go to closed session, and that is tomorrow morning we will not have any open sessions. We will have a closed session on Tuesday, March 3, in the morning. Then we will break at lunch and we will, at 1:00, have a closed session with the Applicant on PRA.

After an appropriate break at 3:00, we will then go to an open session on PRA. So, for members of the public, we will be in open session again tomorrow afternoon at 3:00 to take up PRA. With that, we're going to close this open session and take a short break here, recess, and come back at 4:00.

(Whereupon, the above-entitled matter went off the record at 3:49 p.m.)

February 28, 2020 Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Subcommittee Presentation: NuScale FSAR Topic – Resolution of Chapter 15 Phase 2 Open Items," PM-0220-69062, 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 March 2 - 4, 2020. The materials support NuScale's presentation of the NuScale resolution of Chapter 15 Phase 2 open items.

The enclosure to this letter is the nonproprietary presentation entitled "ACRS Subcommittee Presentation: NuScale FSAR Topic – Resolution of Chapter 15 Phase 2 Open Items," PM-0220-69062, 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
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Enclosure: "ACRS Subcommittee Presentation: NuScale FSAR Topic – Resolution of Chapter 15 Phase 2 Open Items," PM-0220-69062, Revision 0

Enclosure:

"ACRS Subcommittee Presentation: NuScale FSAR Topic – Resolution of Chapter 15 Phase 2 Open Items," PM-0220-69062, Revision 0

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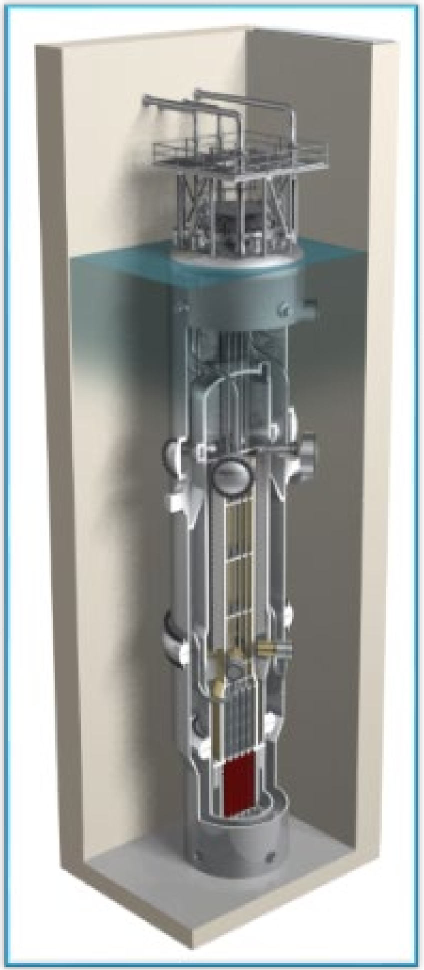
ACRS Subcommittee Presentation



NuScale FSAR Topic

Resolution of Chapter 15 Phase 2 Open Items

March 2-4, 2020

Presenters

Ben Bristol
Supervisor, System Thermal-Hydraulics

Meghan McCloskey
Thermal-Hydraulic Analyst

Matthew Presson
Licensing Project Manager

Paul Infanger
Licensing Specialist

Agenda

- NuScale Design objectives and long term shutdown implications
- FSAR 15.0.6 Return to power analysis
- Boron transport
  - Design basis ECCS cooling
  - Design basis DHRS cooling
  - Beyond design basis conditions
- Changes from FSAR Rev. 2 to FSAR Rev. 4
  - Incorporates NRELAP5 v1.4
  - Minor module model update
  - DHRS actuation logic changes
  - ECCS changes
- Overall changes in Chapter 15 analysis results FSAR Rev. 2 to FSAR Rev. 4

Completely Passive Design Basis

- Fundamental design characteristics enable passive design objectives
  - Low core power and large RCS volume
  - Simple decay heat removal systems
- Actual plant capabilities for heat removal and reactivity control are much more reliable than existing fleet
  - Fail safe valve positions activate passive heat removal
  - NPM can reach cold shutdown using CRAs alone
    - Accommodates reactivity insertion from complete Xenon burnout
    - NuScale PDC-27 commitment for all future core designs
- No active safety systems – no requirement for safety related power and or safety operator mitigation actions

Traditional Analysis Limitations

- Design basis events are analyzed considering highest worth CRA fails to insert.
  - 1 of 16 as opposed to 1 of 53 (AP1000)
  - Small core (larger leakage) leads to proportionally more excess reactivity and larger CRA worth for exterior assemblies.
  - Application of WRSO is uniquely penalizing for the NuScale design
- Origins of GDCs indicate no intention of application of stuck rod margin for the purposes of long term hold down.
  - Redundant system intended to be used to compensate for Xenon burnout (GDC 26 and 27).
  - Not required for the NuScale design due to CRA worth and natural Boron redistribution phenomena during extended ECCS operation

Boron Addition Considerations

- Mechanisms of ECCS cooling result in natural Boron accumulation in the core region
  - Same phenomena as typical PWR post LOCA Boron accumulation
  - Lack of continuous boron source supports sufficient margin to precipitation limits
  - Boron accumulation phenomena enhances long term shutdown margin, except late in cycle
- Consequences of late in cycle loss of SDM at low temperatures and power levels due to very slow Xenon burnout are not a safety concern.
  - GDC-27 exemption request
- NuScale conclusion: An active or passive safety Boron addition system does not make the design safer.

Principle Design Criteria 27

- DCA includes an exemption request from GDC-27
  - NPM design does align with precedent based compliance with GDC-27 due to lack of second safety reactivity control system
- Principle Design Criteria 27
  - Passive reactor GDC-27 equivalent
  - Ensures the safety related reactivity control system is designed to achieve and maintain subcritical core
  - Ensures fuel integrity for an extended overcooling in combination with a partial failure of reactivity system (stuck rod)

PDC-27 Clarification

- RAI-9498 Q# 15-9S1
  - Revised PDC language consistent with Staff interpretation of acceptable consequences for the return to power condition.
  - FSAR was updated committing to SAFDLs acceptance criteria for all DBEs.
  - Ensures an accident with fuel failure does not precede a return to power where additional source term would need to be analyzed.

The reactivity control systems shall be designed to have a combined capability of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained. Following a postulated accident, the control rods shall be capable of holding the reactor core subcritical under cold conditions with all rods fully inserted.

~~Following a postulated accident, the control rods shall be capable of holding the reactor core subcritical under cold conditions, without margin for stuck rods, provided the specified acceptable fuel design limits for critical heat flux would not be exceeded by the return to power.~~

Agenda

- Overview of boron transport analyses and evaluation of N-x reactivity balance conditions
- FSAR 15.0.6 Return to power analysis
- Boron transport
  - Design basis ECCS cooling
  - Design basis DHRS cooling
  - Beyond design basis conditions
- Changes from FSAR Rev. 2 to FSAR Rev. 4
  - Incorporates NRELAP5 v1.4
  - Minor module model update
  - DHRS actuation logic changes
  - ECCS changes
- Overall changes in Chapter 15 analysis results FSAR Rev. 2 to FSAR Rev. 4

Compliance with PDC-27

- Immediate shutdown is sufficient to protect RCPB and SAFDLs with margin for the worst rod stuck out of the core
- Cold shutdown is achieved with all control rods fully inserted
- Loss of Shutdown Margin Consequences Benign
  - Evaluated with single highest worth control rod fully withdrawn
  - Critical power level does not challenge DHRS or ECCS heat removal or SAFDLs
- Probability of the combination of conditions that results in a loss of shutdown return to power with a single rod stuck out of the core is small

Return to Power Mechanisms

- Moderator overcooling
  - ECCS and DHRS designed to removed decay and residual heat
  - Under cold conditions DHRS or ECCS can cause a fairly rapid temperature decrease and increased moderation
- Fission product decay
  - Xenon decay causes a slow post shutdown reactivity insertion
- Boron redistribution
  - Boiling/condensing systems cause boron redistribution
    - Boric acid is not readily volatilized to the vapor phase and would be expected to recondense
    - Results in increasing concentration in boiling region and decreasing concentration in condensing region
  - Conclusion: Boron redistribution during extended ECCS operation increases SDM in the core (neglected in OCRP analysis)

Loss of Shutdown Margin

- FSAR 15.0.6 evaluation of return to power conditions
- Updated method described in FSAR 15.0.6
  - Statepoint analysis with SIMULATE5
  - NRELAP5 quasi-steady analysis
  - Critical power level at overlap
  - SIMULATE5 uncertainties accounted for
  - Control rod ejection with additional stuck rod only analyzed for short-term response
- Updated results presented in FSAR 15.0.6
  - Recriticality precluded for DHRS cooling with riser uncovery
  - DHRS cooling with covered riser non-limiting
  - ECCS cooling limiting

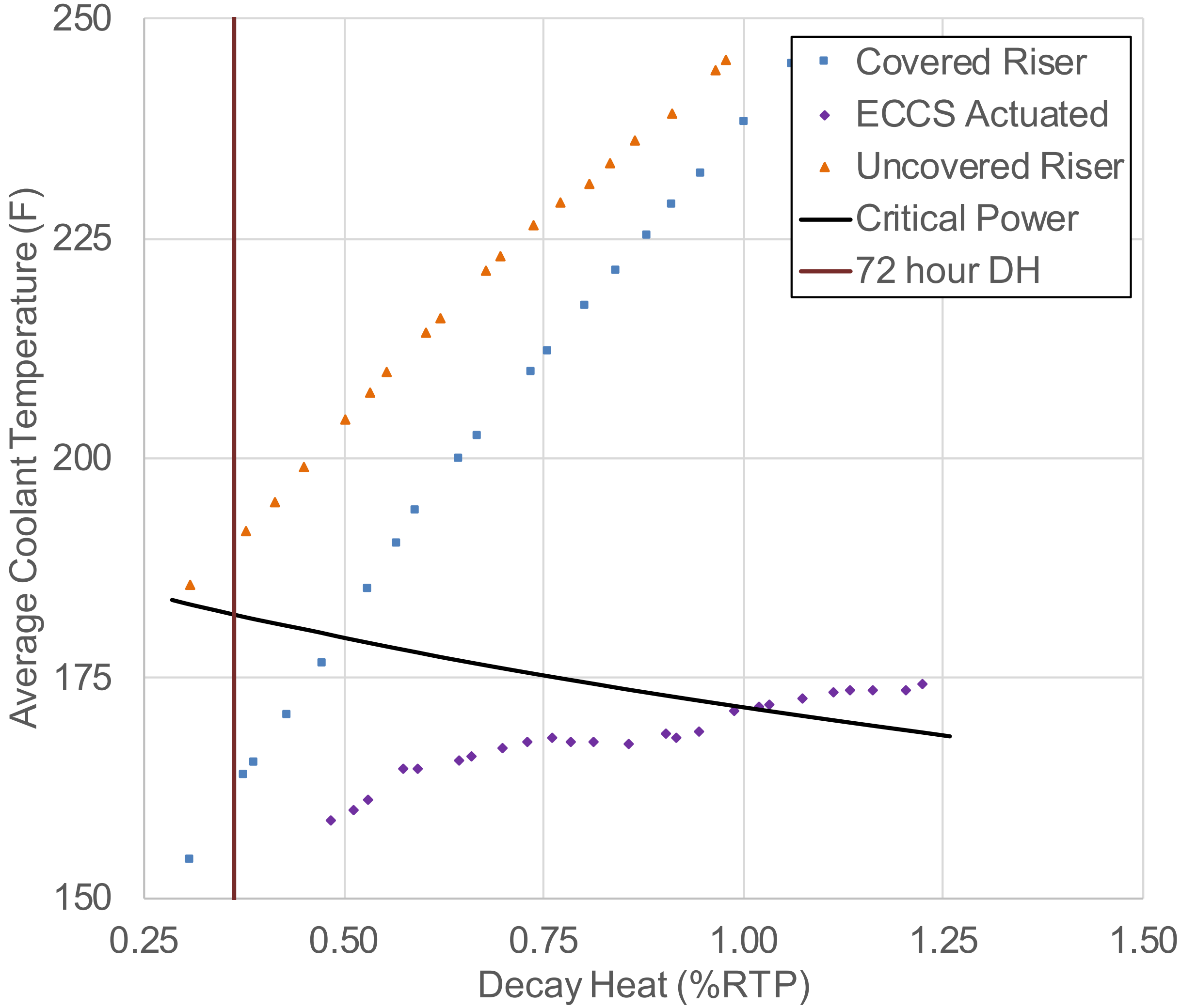
Loss of SDM Evaluation

- Average core temperature determined with the NRELAP5 state-point method described in LTC LTR
  - Performed for spectrum of initial conditions and cooling modes
- Critical power level determined using the SIMULATE5 core model with WRSO
  - Performed for a spectrum of boundary conditions (pressure, temperature, flow)
- CHF is evaluated using the zero flow CHF correlation described in the LOCA LTR
  - Margin is reported to the appropriate analytical limit also described in the LOCA LTR

Loss of SDM Evaluation

- Limiting Initial Conditions
  - Minimized Boron (Hot Full Power, Eq. Xe, EOC Core)
  - Maximized Cooling (Max pool level, min pool temp, biased high SG & DHRS heat transfer coefficient, max ECCS capacity)
- Additional Penalties for MCHFR Evaluation
  - Reactivity bias applied to SIMULATE5 to account for methodology uncertainty (Increases critical power level)
  - Conservative local peaking factor applied to core heat flux
  - Dynamic return to power factor of 2.0 applied

Equilibrium Power Results



Results – Return to Power Analysis

- ECCS cooling most limiting with equilibrium power limited to 1-2% RTP.
- Core temperature must be <200°F for recriticality
- Increased pool temperature decreases the magnitude of the return to power, with 140°F precluding a recriticality
- Earliest recriticality determined to occur approximately 40 hours post-scram
- MCHFR for most limiting results non-limiting relative to other events
- Other AOO acceptance criteria met
- Other SAFDLs demonstrated with OCRP conditions bounded by existing analyses developed for the DCA

Control Rod Ejection, GDC 28, and PDC-27

- RAI 9647/q15-29
- Return to power analysis
  - Performed to demonstrate compliance with PDC-27
  - Is bounding with respect to long-term holddown with a single control rod ejected
- Control rod ejection analysis
  - Performed to demonstrate compliance with GDC 28
  - GDC 28 imposes core design limits distinct from reactivity control system capabilities addressed by GDC 27
  - REA is non-mechanistically postulated for the purpose of evaluating the consequences of a limiting reactivity insertion event as required by GDC 28
  - Extension of PDC 27 to REA not warranted by unique design considerations

Control Rod Ejection, GDC 28, and PDC-27

- GDC 27 has historically not been applied to a rod ejection accident
  - GDC 27 is not cited in SRP 15.4.8 or RG 1.77
  - Application of GDC 27 not required in other approved rod ejection methodologies
  - Extension of PDC 27 to NuScale REA is not warranted by unique design considerations
- Control rod ejection is non-mechanistically assumed in NuScale design
  - FSAR 3.9.3.1.2: CRDM pressure housing is a Class 1 appurtenance per ASME BPVC, Section III, NCA-1271
    - As with other Class 1 vessels and appurtenances, gross failure is not considered credible
  - CRDM nozzles are integral parts of reactor pressure vessel closure head forging
  - CRDM nozzle to Alloy 690 safe-end welds are full penetration butt welds
  - Estimated likelihood of rod ejection, failure of a control rod to insert, failure of boron addition system that could result in return to power is ~ 1E-10 per year

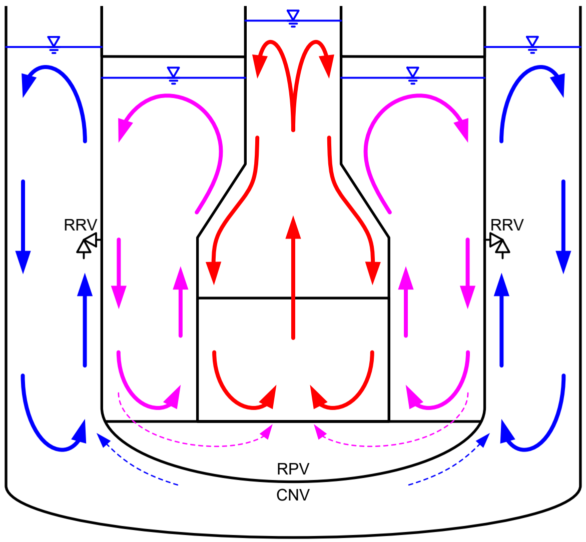
Agenda

- Overview of boron transport analyses and evaluation of N-x reactivity balance conditions
- FSAR 15.0.6 Return to power analysis
- Boron transport
  - Design basis ECCS cooling
  - Design basis DHRS cooling
  - Beyond design basis conditions
- Changes from FSAR Rev. 2 to FSAR Rev. 4
  - Incorporates NRELAP5 v1.4
  - Minor module model update
  - DHRS actuation logic changes
  - ECCS changes
- Overall changes in Chapter 15 analysis results FSAR Rev. 2 to FSAR Rev. 4

ECCS Boron Transport – Context

Context for ECCS boron transport analysis:

- As boron accumulates in the core/riser region, boron concentration in the CNV and DC decreases
  - Boron precipitation analysis performed as part of ECCS long term cooling analysis
- Boron dilution analysis performed to:
  - Evaluate potential for lower boron concentration fluid in core or near core inlet
  - Confirm appropriate scope of return to power analysis by demonstrating that core region concentration remains above initial concentration
  - Response to RAI 8930



Boron transport governed by:

- boiling in the core
- condensation in the containment vessel

ECCS Boron Transport – Method

- Method summary for dilution analysis:
  - LTC PIRT high ranked phenomena affecting boron transport evaluated
  - Control volume approach to analyze transport between regions
  - NRELAP5 used to provide volume fluid masses, flow rates as input for boron transport calculation
    - Volatility, entrainment calculated separately
  - Boron transport calculation performed separate from NRELAP5
  - Cotransport out of RCS hot region
  - Demonstrate that RCS hot region concentration remains above initial concentration
- Key areas of NRC review:
  - Treatment of boron volatility
  - Mixing
- CR opened to evaluate scope of scenarios considered
- Additional discussion in closed session
  - conservatively model transport between regions:
    Boron distribution factors applied to minimize boron transport in, maximize boron

ECCS Boron Transport – Results

- Boron transport evaluated during ECCS cooling
  - Results summarized in RAI 8930 show core boron concentration remains above initial concentration
    - No net core boron dilution is expected even with biased transport assumptions
    - More realistic analysis of boron transport indicates boron concentration in RCS core region is 2-3 times the initial concentration at 72 hours. Core boron concentration remains above initial concentration for at least 7 days.
- Realistically, long term, high boron concentration expected in RCS hot region, with low concentration in RCS cold region, containment
- Recovering the riser and establishing Mode 3 conditions will take multiple deliberate operator actions following appropriate procedures
- Procedures are developed on a site-specific basis (COL commitments 13.5-2 and 13.5-7.)

Comparison of ECCS and DHRS Conditions

ECCS

- Cooling established by boiling/condensing mode
- Will tend to redistribute boron into the RCS hot region and out of RCS cold, CNV regions
- RCS level well below top of the riser
- Recovering the riser and establishing Mode 3 conditions will take multiple deliberate operator actions following appropriate procedures

DHRS

- Riser uncovery sustained by significant convective heat transfer through the riser wall
  - RCS level significantly higher than during ECCS operation – top of steam generator relatively limited condensing potential compared to CNV
  - When riser remains covered, primary side natural circulation maintained and boron distribution should remain close to initial well-mixed condition
- Minimal drivers to redistribute boron in RCS compared to ECCS cooling
- Recovering the riser and establishing Mode 3 conditions will take multiple deliberate operator actions following appropriate procedures

Additional discussion of extended DHRS cooling in closed session.

PRA Considerations for ATWS

- ATWS is typically postulated due to common cause failure of I&C systems to generate a reactor trip signal
  - Events would easily be resolved by removal of power from CRDMs
- Focus of ATWS analysis is generally limited to short term RPV/Secondary pressurization analysis.
  - Short term effects are not challenging due to small core power, large RCS volume, and large RSV capacity in NuScale module design.
- Analysis of long term effects of an ATWS are less meaningful for risk insight due to low combined likelihood of mechanical CCF of 16 CRAs
- In the NuScale PRA, ATWS frequency is conservatively based on mechanical CCF of 3 CRAs.
  - N-3 transients will immediately shutdown similar to design basis events where WRSO is considered (Ch. 15)
- PRA supporting T/H calculations evaluate failure of all CRAs to insert for 72hr coping period.

PRA Insights for ATWS

- LOCA/IORV Events
  - Short term response
    - BOC/EOC – Break flow cause sufficient depressurization and void feedback to make core subcritical.
  - Long term response (riser uncovery)
    - BOC – Boron accumulation in core leads to complete shutdown (no power return)
    - EOC – Equilibrium power level achieved due to balance ECCS cooling with reactivity feedback mechanisms from void, temperature, and Xenon.
- nonLOCA Events
  - Short term response
    - BOC – MTC is less negative but still sufficient to reduce reactor power to match DHRS heat removal.
    - EOC – Large negative MTC cause a quick stabilization of core power and DHRS heat removal.
  - Long term response (no riser uncovery)
    - BOC – Equilibrium power level achieved due to balance DHRS cooling with reactivity feedback mechanisms from temperature, Xenon, and Boron accumulation. RSV venting and subsequent boron concentrating identified as important factor in overall reactivity balance.
    - EOC – Equilibrium power level achieved due to balance DHRS cooling with reactivity feedback mechanisms from temperature, and Xenon.
- T/H results support the conclusion that no operator action is required to mitigate event and prevent core damage for short or long term ATWS mitigation.

Conclusions

- Inherent design characteristics provide ample safety
  - Low core power, large RCS inventory, small high pressure containment, and large ultimate heat sink
- Compliance with intent of GDCs is demonstrated for reactivity control systems
  - Conservative analysis of the low probability return to power condition demonstrates safety margin
- Boron redistribution is evaluated and demonstrated to not be a safety concern
  - Naturally accumulating boron in the core adds to shutdown margin for design basis event and severe accidents.

Agenda

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  - ECCS changes
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Ch 15 Changes FSAR Rev. 2 to Rev. 4

- Results from FSAR Rev. 2 presented to ACRS in June, July 2019 in subcommittee and full committee meetings for Chapter 15
- Changes in FSAR Rev. 3 include
  - Update from NRELAP5 v1.3 to v1.4
  - Updated NRELAP5 base model input
  - More conservative core design input in some cases
  - DHRS actuation signal changes, addition of secondary side isolation signal
  - ECCS actuation signal changes
- Changes in FSAR Rev. 4 include
  - ECCS IAB threshold/release pressure changes

NRELAP5 v1.4

- Modifications made from v1.3 to v1.4 were due to routine code maintenance
- 26 specific code Fixes (documented in error reports) with most notable being:
  - Condensation correlation error corrections (< 2 psi increase in CNV pressure calculations)
  - Correction to choking model quality factor (little to no impact)
  - Updated Windows executable to 64-bit version (not used for production calculations)
- 5 new Features – None of which impact DCA calculations
  - Added proprietary classifications marking to source files
  - Expanded number of elements allowed in water property file (no water property file update)
  - Interpolation update for CHF correlation not used in DCA calculations
  - Added warning message to users if mass error stop (1%) is disabled
  - Removal of Developmental Options from user access

NRELAP5 Base Model

- Revision 0 released 12/2015 (DCA submittal 12/2016)
- Revision 1 released 8/2017
  - Updates for design consistency
    - Minor geometry changes based on drawing updates
    - Minor RCS flow loss updates (changes in best estimate values)
  - Updates for analysis consistency and ease of downstream use
    - Minor nodalization changes to match LOCA model
    - Added passive heat structures defined in LOCA model
  - Other changes
    - Change from elevation based to volume based calculation of collapsed liquid level
    - Error correction when specifying lower CNV material (had been previously corrected in impacted analysis calculations)
- Revision 2 released 01/2019 (FSAR Rev. 3 submittal 8/2019)
  - Removed ECCS actuation on RCS riser level signal
  - Minor RCS flow loss updates
  - Minor geometry error corrections

Neutronics Range Changes

- For FSAR Rev 3, analyzed more bounding ranges of core design input, including 2 additional depletions for high and low flow rates.
- Different parameter ranges included:
  - Most negative DTC (from -2.25 pcm/°F to -2.5 pcm/°F)
  - Delayed neutron fraction (β_{eff})
  - Augmentation factors for asymmetric reactivity events
- No changes to MTC range

DHRS Actuation Changes

- Summary of change:
  - Add secondary side isolation actuation for range of signals that indicate upset in normal secondary side cooling conditions
  - DHRS actuation limited to subset of signals indicating insufficient secondary side cooling
  - DHRS actuated following secondary side isolation
- Purpose of change: Support expected plant startup progressions
- Effect of change on transient analyses:
  - Heatup events – No change to expected DHRS actuations on high pressurizer pressure or high RCS hot temperature
  - Cooldown events – Secondary side isolation may be actuated first; DHRS actuated afterwards on high steam pressure
  - Reactivity events, inventory increase, inventory decrease events not significantly impacted

DHRS Actuation Changes

FSAR Rev. 2	FSAR Rev. 3, Rev. 4
DHRS actuation on: - High pressurizer pressure - High RCS hot temperature - High CNV pressure - Low pressurizer pressure - Low-low pressurizer level - Low main steam pressure - Low-low main steam pressure - High main steam pressure - High main steam superheat - Low main steam superheat - High under bioshield temperature - Low AC voltage	SSI actuation on: - High pressurizer pressure - High RCS hot temperature - High CNV pressure - Low-low pressurizer pressure - Low-low pressurizer level - Low main steam pressure - Low-low main steam pressure - High main steam pressure - High main steam superheat - Low main steam superheat - High under bioshield temperature - Low AC voltage DHRS actuation on: - High pressurizer pressure - High RCS hot temperature - High main steam pressure - Low AC voltage

DHRS Actuation Changes

- Example impact on cooldown event:
  Decrease in FW Temperature MCHFR Case

Event	Time (sec) FSAR Rev. 2	Time (sec) FSAR Rev. 3, 4
Feedwater temperature begins to decrease	0	0
Feedwater temperature reaches 100°F	160	86
High RCS hot temperature limit reached	125	184
High reactor power limit reached	131	187
Reactor trip (high reactor power)	133	189
DHRS actuation (high RCS hot temp)	133	192
SSI actuation (high RCS hot temp)	n/a	192

Limiting case occurs where high power, high RCS hot temperature occurs ~ same time

DHRS Actuation Changes

- Example impact on cooldown event:
  Increase in Steam Flow Rate MCHFR Case

Event	Time (sec) FSAR Rev. 2	Time (sec) FSAR Rev. 3, 4
Steam flow begins to increase	0	0
High RCS hot leg temperature reached	60	n/a
High reactor power limit reached	n/a	63
Reactor trip	68	65
Low pressurizer pressure limit reached	n/a	123
SSI actuation (low pressurizer pressure)	n/a	125
High steam pressure	n/a	1692
DHRS actuation	68	1697

Maximum power in both cases ~ 200 MW
Limiting case occurs where high power, high RCS hot temperature occurs ~ same time

NUSCALE Power for all humankind

DHRS Actuation Changes

- Example impact to heatup event:
  FWLB Limiting DHRS Case

Event	**Time (sec) FSAR Rev. 2**	**Time (sec) FSAR Rev. 3, 4**
Large FW line break inside CNV	0	0
High CNV pressure limit reached	1	1
RTS actuated (high CNV pressure)	3	3
Secondary system isolation actuated (high CNV pressure)	n/a	3
High pressurizer pressure limit reached	*{ does not cause additional actuations }*	7
DHRS actuation	3 (high CNV pressure)	9 (high PZR pressure)
RSV lift point reached	25	25
DHRS actuation valves open	33	39

DHRS Actuation Changes

- Example impact on reactivity event:
  Uncontrolled bank withdrawal at power – MCHFR case

Event	Time (sec) FSAR Rev. 2	Time (sec) FSAR Rev. 3, 4
CRA bank begins to withdraw	0	0
High RCS hot temperature limit reached	178	144
High pressurizer pressure limit reached	184	150
Reactor trip actuated	186	152
SSI actuated	n/a	152
DHRS actuated	186	152

Limiting case occurs where high power, high RCS hot temperature occurs ~ same time when different signal delays are accounted for

ECCS Valve Operation

- Emergency core cooling system (ECCS) valves receive actuation demand on:
  - ECCS actuation signal on high CNV level, or
  - Loss of DC power to ECCS trip valves
- Inadvertent actuation block (IAB) feature prevents ECCS valve opening if the differential pressure between the reactor coolant system (RCS) and containment (CNV) is above the IAB threshold pressure
  - This feature prevents opening due to spurious signals or equipment failures at normal operating pressures but permits opening in loss-of-coolant accident (LOCA) conditions
- If IAB is actuated by ECCS demand at high differential pressure, IAB releases at lower pressure and then ECCS valves open

ECCS Actuation Changes

FSAR Rev 2

- ECCS actuated on
  - High CNV level (220-260 in)
  - Low RCS riser level (350-390 in)
  - Loss of DC power to valve actuators
- RCS riser level is post-accident monitoring Type B and Type C variable
  - 4 total divisions of RPV riser level
- IAB threshold/release
  - Threshold:
    Block if ECCS actuated above threshold pressure that is in the range of 1000-1200 psid
  - Release: If IAB blocks, release between 1000-1200 psid

FSAR Rev 4

- ECCS actuated on
  - High CNV level (264-300 in)
  - Loss of DC power to valve actuators
- RCS riser level is post-accident monitoring Type B and Type C variable
  - 4 total divisions of RPV riser level
- IAB threshold/release
  - Threshold:
    Block if ECCS actuated above 1300 psid;
    does not block below 900 psid
  - Release: If IAB blocks, release at 950 psid +/- 50 psi

ECCS Changes - Revised FSAR Analyses

- Impacted FSAR Sections
  - FSAR 6.2 Peak CNV Pressure
  - FSAR 15.6.5 Loss of Coolant Accidents
  - FSAR 15.6.6 Inadvertent Operation of ECCS
- Revised assumptions
  - Assumes all ECCS valves remain closed due to IAB block function above 1300 psid
  - Evaluated ECCS valves opening on IAB release pressure between 900 and 1000 psid
- Revised analysis results submitted in September 2019 and reviewed in NRC October audit in Corvallis
- DCA Revision 4, including revised FSAR analysis results, formally submitted January 2020

ECCS Changes - Updated Analysis Results

Event / Acceptance Criteria	DCA Rev 3 Results	Updated DCA Rev 4 Results	Comments
Peak CNV Pressure (RRV Opening) CNV Design Pressure - 1050 psia	986 psia	994 psia	Change in peak pressure due to staggered IAB release (2nd RRV at 1000 psid, RVVs at 900 psid)
LOCA - Minimum Water Level Above Top of Active Fuel	1.7 ft	1.5 ft	Change due to lower IAB minimum release pressure 900 psid
Inadvertent ECCS valve opening – MCHFR limit 1.13	1.41	1.32	Change due to model revisions not IAB threshold change

Conclusions – ECCS Valve Changes

- CNV peak pressure results slightly more limiting (8 psi) due to explicit evaluation of ECCS valves opening at different IAB release pressures
- LOCA minimum water level above fuel results slightly more limiting (~0.2 feet difference) due to lower minimum IAB release pressure of 900 psid
- Inadvertent ECCS valve opening MCHFR slightly more limiting due to evaluation of error corrections and more bounding model input, not from IAB change
- All updated event results demonstrated margin to acceptance criteria

Agenda

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FSAR 15 Limiting Transient Results

Parameter	Event	Acceptance Criterion	Limiting Result FSAR Rev. 2	Limiting Result FSAR Rev. 4
Maximum RCS Pressure	Several	< 2315 psia (110% P_{design}) or < 2520 psia (120% P_{design})	~ 2170 psia	~ 2170 psia
Maximum SG Pressure	Inadvertent Operation DHRS	< 2315 psia (110% P_{design})	1582 psia	1592 psia
	SG tube failure	< 2520 psia (120% P_{design})	1806 psia	1871 psia
MCHFR	Single rod withdrawal	> 1.284	1.614	1.375
	Inadvertent Opening RRV	> 1.13	1.41	1.32
	LOCA	> 1.29	1.796	1.74
Level above top of core	LOCA	> 0 ft	1.5 ft	1.5 ft

Conclusions

- Revised return to power analysis shows ECCS cooling conditions result in equilibrium power at 1-2% RTP
- ECCS boron transport analysis demonstrates that core boron concentration remains higher than initial concentration
- Changes incorporated into FSAR Revision 3:
  - Several minor changes in NRELAP5 code, NPM plant base model
  - DHRS, ECCS actuation changes
- ECCS IAB changes incorporated into FSAR Revision 4
- FSAR Ch 15 limiting transient results consistent between FSAR Rev. 2 and Rev. 4
- FSAR Ch 15 analysis results demonstrate margin to acceptance criteria

Acronyms

AOO – Anticipated Operational Occurrences

CHF – Critical Heat Flux

CNV – Containment Vessel

COL – Combined License

COLR – Core Operating Limits Report

CRDM – Control Rod Drive Mechanism

CVCS – Chemical and Volume Control System

DHRS – Decay Heat Removal System

DTC – Doppler Temperature Coefficient

ECCS – Emergency Core Cooling System

EOC – End of Cycle

GDC – General Design Criteria

IAB – Inadvertent Actuation Block

LCO – Limiting Condition for Operation

LOCA – Lossof Coolant Accident

MCHFR – Minimum Critical Heat Flux Ratio

MTC – Moderator Temperature Coefficient

NPM – NuScale Power Module

OCRP – Overcooling Return to Power

PDC – Plant Design Criteria

PIRT – Phenomena Identification and Ranking Table

RCPB – Reactor Coolant Pressure Boundary

RCS – Reactor Coolant System

REA – Rod Ejection Accident

SAFDL – Specified Acceptable Fuel Design Limits

SDM – Shutdown Margin

WRSO – Worst Rod Stuck Out

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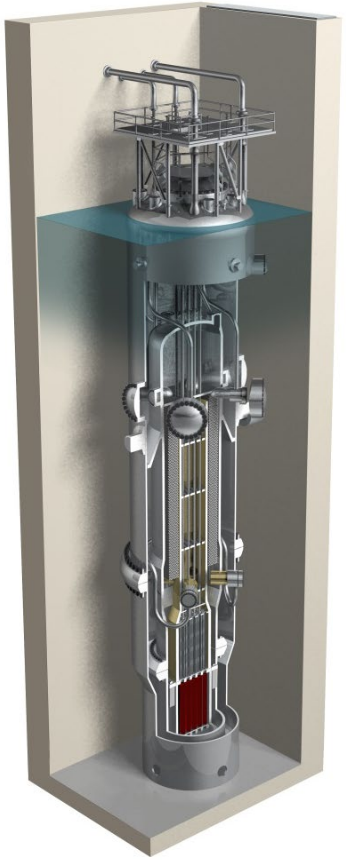

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Additional Information

MODE Definitions

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	INDICATED REACTOR COOLANT TEMPERATURES (°F)
1	Operations	≥ 0.99	All ≥ 420
2	Hot Shutdown	< 0.99	Any ≥ 420
3	Safe Shutdown [(a)]	< 0.99	All < 420
4	Transition [(b)(c)]	< 0.95	N/A
5	Refueling [(d)]	N/A	N/A

(a) Any CRA capable of withdrawal, any CVCS or CFDS connection to the module not isolated.

(b) All CRAs incapable of withdrawal, CVCS and CFDS connections to the module isolated, and all reactor vent valves electrically isolated.

(c) All reactor vessel flange bolts fully tensioned.

(d) One or more reactor vessel flange bolts less than fully tensioned.

SG Modeling

- NRELAP5 validation and NPM sensitivity calculations:
  - SIET-TF1 secondary side heat transfer and pressure drop
  - SIET-TF2 primary and secondary side heat transfer and pressure drop
  - NPM sensitivity calculations for steam generator modeling
    - Axial nodalization, heat transfer variation
- Key conclusions for NPM non-LOCA analysis:
  - Steam generator heat transfer variation affects steady-state conditions:
    - Steam generator initial secondary side inventory, steam temperature
    - RCS initial flow and temperature conditions due to influence of secondary side conditions on natural circulation driving head
  - Steam generator heat transfer impact on initial conditions affects which process condition is first reached that actuates reactor trip and/or other engineered safety systems
    - For events analyzing a spectrum of change, changes in steam generator secondary initial conditions will tend to shift the magnitude of the limiting change but not otherwise change the type of event progression.
      Example: the limiting temperature decrease for the decrease in FW temperature event analysis
  - Steam generator heat transfer does not directly affect margin to MCHFR
    - RCS steady-state flow rate is biased low for MCHFR cases
    - After reactor trip, power decrease much faster than flow rate decrease, even considering variation in steam generator heat transfer

Analysis Results – FSAR Rev. 2; Rev 4

15.1 Increase in heat removal by secondary system

Sec.	Event [1] (Acceptance criteria)	Peak RCS Pressure (< 110% P_{design}: 2310 psia) (< 120% P_{design}: 2520 psia)		Peak SG Pressure (< 110% P_{design}: 2310 psia) (< 120% P_{design}: 2520 psia)		MCHFR (> limit: 1.284)	
15.1.1	Decrease in feedwater temperature	1959	2005	1432	1541	1.921	1.847
15.1.2	Increase in feedwater flow	1936	2002	1424	1491	1.944	1.854
15.1.3	Increase in steam flow	2018	1981	1208	804	1.957	1.881
15.1.4	Inadvertent opening of steam generator relief or safety valve	NA	NA	NA	NA	NA	NA
15.1.5	Steam piping failures	2156	2081	1346	1495	1.861	1.866
15.1.6	Loss of containment vacuum/containment flooding [1]	1992	1937	1342	1426	2.761	2.66

(1) NuScale unique event

Significant margin to acceptance criteria for all events

Analysis Results – FSAR Rev. 2; Rev 4

15.2 Decrease in heat removal by secondary system

Sec.	Event [(1)] (Acceptance criteria)	Peak RCS Pressure (< 110% P_{design}: 2310 psia) (< 120% P_{design}: 2520 psia)		Peak SG Pressure (< 110% P_{design}: 2310 psia) (< 120% P_{design}: 2520 psia)		MCHFR (> limit: 1.284)	
15.2.1	Loss of external load	2158	2161	1474	1545	2.579	2.441
15.2.2	Turbine trip	2158	2161	1474	1545	2.579	2.441
15.2.3	Loss of condenser vacuum	2158	2161	1474	1545	2.579	2.441
15.2.4	Closure of main steam isolation valve	2160	2161	1481	1512	2.567	2.670
15.2.6	Loss of non-emergency AC to station auxiliaries	2162	2160	1361	1415	2.569	2.539
15.2.7	Loss of normal feedwater flow	**2165**	**2171**	1434	1528	2.569	2.426
15.2.8	Feedwater system pipe breaks	**2164**	**2164**	1328	1389	2.607	2.496
15.2.9	Inadvertent operation of the decay heat removal system [(1)]	2163	2161	**1582**	**1592**	2.489	2.67

(1) NuScale unique event

Significant margin to acceptance criteria for all events

Analysis Results – FSAR Rev. 2; Rev 4

15.4 Reactivity and Power Distribution Anomalies – focus on SAFDLs

Sec.	Event [1] (Acceptance criteria)	MCHFR (> limit: 1.284)		Fuel centerline (< Tmelt)		LHR (< 21.22 kW/ft)	
15.4.1	Uncontrolled control rod assembly withdrawal from subcritical or low power	>10	>10	890.8 F	1051.8F	NA	NA
15.4.2	Uncontrollled control rod assembly withdrawal at power	1.624	1.499	NA	NA	8.97 kW/ft	9.16 kW/ft
15.4.3	Control rod misalignment	2.509	1.437	NA	NA	7.10 kW/ft	8.39
15.4.3	Control rod withdrawal	**1.624**	**1.375**	NA	NA	7.84 kW/ft	8.29
15.4.3	Control rod drop	1.641	1.432	NA	NA	8.42 kW/ft	6.71
15.4.6	Inadvertent decrease in boron concentration in RCS	NA	NA	NA	NA	NA	NA
15.4.7	Inadvertent loading and operation of a fuel assembly in improper position	1.916	1.437	NA	NA	7.87 kW/ft	8.39
15.4.8	Spectrum of rod ejection accidents	2.477	1.838	2162 F	2345 F	NA	NA

Control rod withdrawal has limiting MCHFR for reactivity events

Analysis Results – FSAR Rev. 2; Rev 4

15.5 Increase in reactor coolant inventory

Sec.	Event [1] (Acceptance criteria)	Peak RCS Pressure (≤ 110% P_{design}: 2310 psia)		Peak SG Pressure (≤ 110% P_{design}: 2310 psia)		MCHFR (≥ limit: 1.284)	
15.5.1	Chemical and volume control system malfunction	2130	2160	1418	1430	2.379	2.702

Significant margin to acceptance criteria

Analysis Results – FSAR Rev. 2; Rev 4

15.6 Decrease in reactor coolant inventory

Sec.	Event (1) (Acceptance criteria)	Peak RCS Pressure (< 110% P_{design}: 2310 psia) (< 120% P_{design}: 2520 psia)		Peak SG Pressure (< 110% P_{design}: 2310 psia) (< 120% P_{design}: 2520 psia)		MCHFR		Additional	
15.6.1	Inadvertent opening of reactor safety valve	NA		NA		NA		NA	
15.6.2	Failure of small lines carrying primary coolant outside containment	2047	2067	1368	1473	NA		Note 2	
15.6.3	Steam generator tube failure	2073	2158	**1806**	**1871**	NA		Note 2	
15.6.5	Loss of coolant accidents resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary	NA		NA		1.796	1.74	1.5 ft	1.5 ft
						Acceptance criteria: > 1.29		Minimum level above top of core	
15.6.6	Inadvertent operation of emergency core cooling system (1)	NA		NA		**Result: 1.41**	**Result: 1.32**	NA	
						Acceptance criteria: > 1.13			

(1) NuScale unique event

(2) Mass release and iodine spiking time provided as input to radiological analyses

SG tube failure maximum secondary pressure remains below design pressure

Valve opening and LOCA events demonstrate margin to acceptance criteria

Chapter 15, “Transient and Accident Analyses"

Focus Areas on: Boron Redistribution/Return to Power and ECCS

NuScale Design Certification Application

ACRS Subcommittee Meeting
March 2-3, 2020

Non-Proprietary

Agenda

- NRC Staff Review Team
- Closure of Unclear Open Items
- Credit for Non-Safety Valves
- IAB Single Failure
- Return to Power
- Boron Redistribution
- Recovery from certain DHRS and ATWS scenarios
- Boron concentration requirements for Mode 3 to Mode 4
- Changes to Design in Phase 4
  - NRELAP5 v1.4
  - ECCS Actuation Logic
  - IAB block/release pressure
  - DHRS Logic
- LOCA
- Long-Term Cooling
- Changes to Selected Analyses in Phase 4
  - IORV Analysis
  - Peak Containment Pressure
  - Control Rod Misalignment
  - Inadvertent Loading of an Assembly
  - Steam System Pipe Failure Inside/Outside Containment
- ECCS Design
- CNV and RPV Level Instruments

NRC Staff Review Team

- Chapter 15 Phase 4 Technical Reviewers:
  - Antonio Barrett, NRR/DANU
  - Andrew Bielen, RES/DSA
  - Tim Drzewiecki, NRR/DANU
  - Michelle Hart, NRR/DANU
  - Andrew Ireland, RES/DSA
  - Shanlai Lu, NRR/DSS
  - Ryan Nolan, NRR/DSS
  - Jeff Schmidt, NRR/DANU
  - Alex Siwy, NRR/DSS
  - Ray Skarda, RES/DSA
  - Jason Thompson, RES/DSA
  - Boyce Travis, NRR/DANU
  - Carl Thurston, NRR/DSS
  - Chris Van Wert, NRR/DANU
- Additional Technical Reviewers for Boron Redistribution/Recriticality and ECCS Focus Areas:
  - Syed Haider, NRR/DSS (containment peak pressure)
  - Peter Yarsky, RES/DSA (ATWS)
  - Tom Scarbrough, NRR/DEX (ECCS valves)
  - Dinesh Taneja, NRR/DEX (I&C)

Closure of Unclear Open Items

- July 10, 2019, Phase 3 Chapter 15 ACRS meeting discussed status of Chapter 15 review
- Listing of 11 Unclear Open Items provided
- The following presentation notes these OI numbers as each is discussed
  - Selected additional Phase 2 OIs are also included
- OI 15.0.2-2: unclear portion of OI related to staff review of NRELAP5 v1.4
  - Discussed in February 19, 2020, ACRS SC on LOCA topical report
- OI 15.0.2-4, unclear portion of OI related to staff review of the steam generator heat transfer uncertainty
  - Discussed in February 19, 2020, ACRS SC on Non-LOCA topical report

Credit for Nonsafety-Related Valves

- Relied on in Chapter 15 as a backup to safety-related valves when applying single failure criteria.
- DCA Tier 2, Table 15.0-9 identifies the events and assumed single failure for the credited nonsafety-related valves.
- Credit for secondary line break events is consistent with NUREG-0138.
  - consequences due to failure
  - surveillance and operability requirements
  - augmented quality and testing requirements
  - operating experience

Credit for Nonsafety-Related Valves

- Relying on the nonsafety-related secondary MSIVs for steam generator tube failure events is an extension of NUREG-0138.
- Staff requested a sensitivity on the limiting SGTF event considering the failure of the secondary MSIV.
  – Results in ~50% more mass release and proportional increases in dose
  – Large margins remain to both offsite and control room dose criteria

Treatment of IAB Single Failure

- Each ECCS valve includes an IAB design feature to reduce the frequency of inadvertent opening of the valve during power operation.
- Chapter 15 analysis does not assume the potential for single failure of the IAB valve closing function and considers the inadvertent opening of more than one ECCS valve a beyond-design-basis event.
- The IAB valve is an active safety-significant, first-of-a-kind design feature, and more complex than components traditionally excluded from the SFC.
- The staff requested Commission direction in SECY-19-0036.

Treatment of IAB Single Failure

SRM-SECY-19-0036:

- “The staff should review Chapter 15 of the NuScale Design Certification Application without assuming a single active failure of the inadvertent actuation block valve to close.”

Return to Power: GDC 27 Exemption

- General Design Criterion 27 states,
  - The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.
- Staff took the position in the pre-application Gap 27 letter (ML16116A083) that “reliably controlling reactivity” in GDC 27 means shutdown as the final state when considering the totality of NRC regulations regarding reactivity control
- Following an initial shutdown, the NuScale reactor can return and maintain criticality during a cool down on the safety-related, passive heat removal systems (DHRS and ECCS) under certain conditions
- NuScale submitted an exemption to GDC 27 and requested approval of a principle design criteria, PDC 27

SECY-18-0099

- SECY-18-0099 (ML18065A540) specifies the following three criteria to evaluate the exemption
  - The design of the reactor must provide sufficient thermal margin such that a return to power does not result in the failure of the fuel cladding fission product barrier, as demonstrated by not exceeding SAFDLs for the analyzed events.
  - The combination of circumstances and conditions leading to an actual post reactor trip return to criticality is not expected to occur during the lifetime of a module.
  - The incremental risk to public health and safety from the hypothesized return to criticality at a NuScale facility with multiple reactor modules does not adversely erode the margin between the Commission’s goals for new reactor designs related to estimated frequencies of core damage or large releases and those calculated for the NuScale design.

- ACRS endorsed the proposed staff criteria with the addition of evaluating the overall facility risk, which is reflected in the third criteria above (ML18052A532)
- Satisfying the three criteria in SECY-18-0099 would ensure no undue risk to public health and safety

Proposed PDC 27 (OI 15.0.6-1)

- NuScale revised PDC 27 in DCD Section 3.1.3.8 to state:
  - “The reactivity control systems shall be designed to have a combined capability of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained. Following a postulated accident, the control rods shall be capable of holding the reactor core subcritical under cold conditions with all rods fully inserted”
  - NuScale revised DCA Chapter 15, Tables 15.0-2, 15.0-3 and 15.0-4 acceptance criterion to ensure that “capability to cool the core is maintained” refers to meeting the specific acceptable fuel design limits (SAFDLs), including margin for a stuck rod, for all design basis events (DBEs)

Return to Power Scenarios

- Three scenarios can potentially lead to a return to power
  - DHRS cooldown with dc power (EDSS)
    - RPV water level remains above the riser
    - RPV water level drops below the riser
  - DHRS cooldown without dc power (EDSS)
    - ECCS actuation at IAB setpoint
  - ECCS cooldown
- Can occur as a result of most Chapter 15 AOOs or PAs
- Key assumptions in the return to power scenarios
  - No operator action
  - Only safety-related equipment is used to mitigate the event
  - The worst stuck rod is assumed stuck out consistent with GDCs
- A return to power is possible at EOC conditions, but not when significant RCS boron exists (e.g., BOC and MOC conditions)

EOC Return to Power Analysis Methodology

- New analysis method applied in DCA, Section 15.0.6, Revision 3, using NRELAP5 and SIMULATE5
  - NRELAP5 used to determine the average moderator temperature for a series of constant core powers for each cooldown scenario (e.g., DHRS riser uncovery)
    - NRELAP5 reactivity feedbacks not used
    - Various reactor building pool temperatures analyzed
    - Multiple riser heat transfer assumptions analyzed
  - SIMULATE5 used for reactivity balance SIMULATE5 power search used to determine critical power for variety of input conditions
    - NRELAP5 core power verses RCS flow rate curve used to determine a single SIMULATE5 criticality line verse average moderator temperature
  - The intersection of the NRELAP5 constant power verse average moderator temperature curve and the SIMULATE5 line of criticality determines the equilibrium return to power magnitude
  - MCHFR calculated using a pool boiling correlation

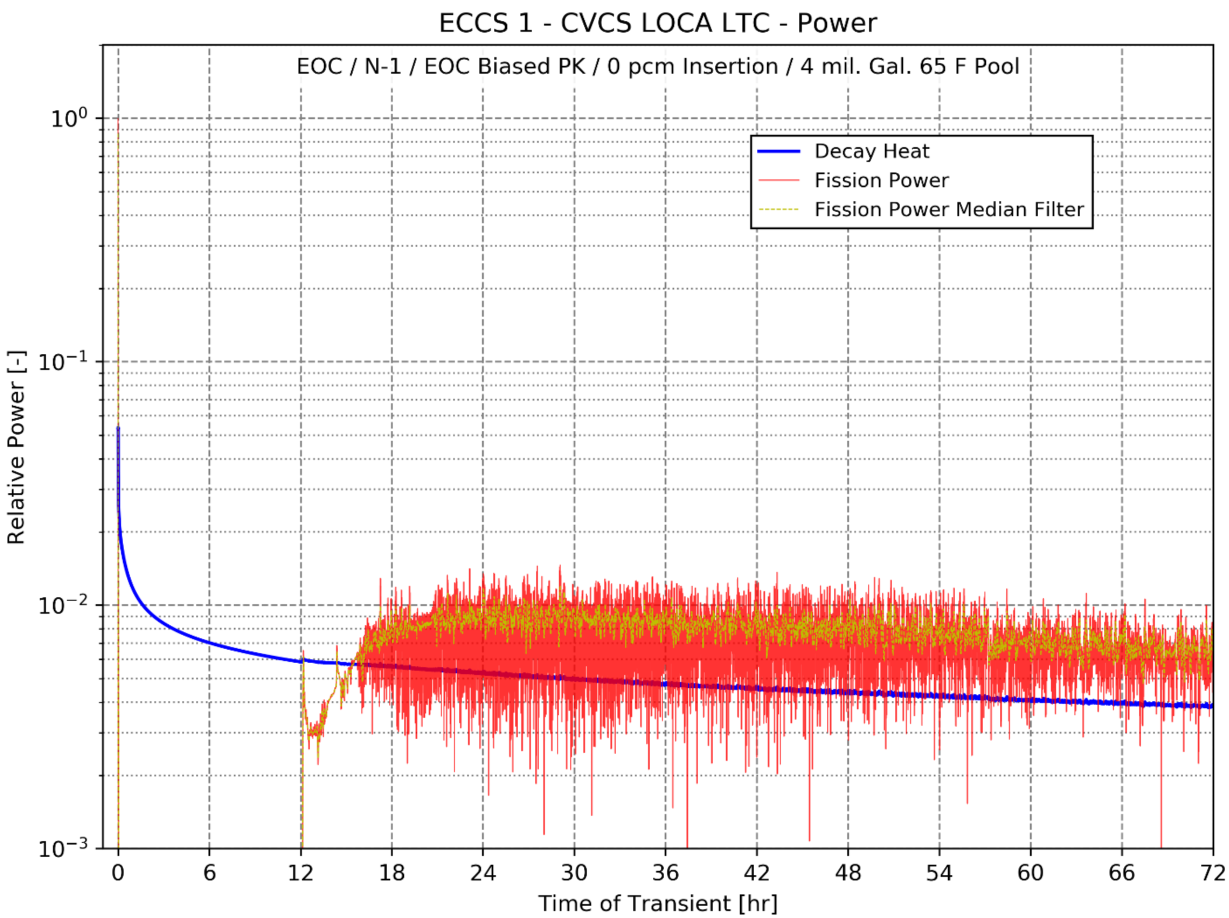
EOC Return to Power Analysis Methodology Inputs

- NRELAP5
  - Reactor pool level is maximized and temperature is minimized
  - DHRS heat transfer increased 30 percent
  - ECCS valve capacity maximized
  - Conservative decay heat assumed (UOI: OI 15.0.6-4)
- SIMULATE5 (UOI: OI 15.0.6-3)
  - Reactivity biased to address SIMULATE5 uncertainties
  - Conservative coolant density assumed
  - Conservative peaking factors for MCHFR determination
  - Zero Xenon concentration assumed consistent with 72 hours following a reactor trip
- MCHFR
  - Calculated using stuck rod peaking factors increased by a factor of 2 to account for potential transient overshot effects

EOC Return to Power Analysis Results

- DHRS cooldown, assuming riser remains covered, and ECCS cooldown return to power
  - Return to power is less than 2% rated thermal power
  - Significate MCFHR margin exists
  - General Design Criterion 10 met
- DHRS cooldown with water level dropping below the riser (riser uncovered) remains subcritical due to sufficient decay heat at 72 hours (UOI: OI 15.0.5-1)
- Staff's independent confirmatory analysis yielded similar results
- Staff recommended approving the Exemption to GDC 27

Core Power, EOC ECCS Confirmatory Analysis



Potential Non-EOC Return to Power (UOI: OI 15.0.6-5)

- Excess reactivity controlled by soluble boron
- Loss of soluble boron in the core during a cooldown could cause a recriticality similar to the EOC ECCS cooldown scenario
- Core boron can be reduced by:
  - Flashing/Liquid Discharge
  - Entrainment
  - Boron volatility
  - Core and riser boron gradient
  - Diluted CNV water entering the core

Return to Power at Non-EOC Analysis Methodology

- Staff review documented in SER Section 15.0.6
  - Staff conducted detailed audit and numerous public meetings on topic
- Control volume method using NRELAP5 to calculate fluid transport
  - Boron transport informed by NRELAP5 fluid transport
- Methodology uses conservative assumptions to minimize core boron concentration
  - Boron mass is removed by conservative treatment of physical phenomenon
  - Boron mass is artificially removed to ensure overall methodology conservatism
- Determination of boron loss using NRELAP5 information include:
  - Flashing/Liquid discharge
  - Entrainment
  - Boron volatized and redeposited outside the core
  - CNV level
- Riser and core boron gradient evaluated based on NIST-1 and VEERA test data

Staff Findings Non-EOC Analysis Methodology

- Staff agrees that boron will concentrate in the core/riser region due to boiling
- Staff concluded that boron loss terms informed by NRELAP5 are conservative
- Staff concluded that assuming the elimination of the downcomer and lower plenum boron mass is conservative with regard to core boron concentration
- Boron volatility correlation was reasonable based on the NuScale operating conditions and conservative by not including boron rewetting and return to core
- VEERA test data demonstrates that core boron is uniform once saturated boiling conditions are reached
  - Evaluation of a fully diluted water mass (0 ppm) below the saturated boiling core elevation demonstrated the core remained subcritical
- NIST-1 long-term cooling core exit void test data demonstrated that enough two-phase mixing would occur to promote riser and core mixing
- Staff concluded that final core boron concentration at 72 hours is greater than the initial core RCS boron concentration, maintaining subcriticality
- Staff is aware of a Condition Report written by NuScale late last week that is being discussed

Return to Power at Non-EOC 72 hours ➡ 7 days

- Staff considered NuScale capability to cope with potential boron redistribution without the need for additional nonsafety-related equipment for a period of 7 days consistent with SECY-96-128 (RTNSS ‘B’).
- Staff reviewed NuScale calculation initial conditions, assumptions, and results.
- Staff agrees there is sufficient decay heat removal and the core would remain subcritical throughout the 7-day period.
- Boration from the CVCS is not required in the first 7 days.

Long Term DHRS Operation

- The DHRS is a safety-related heat removal system used to mitigate non-LOCA transient events
  - Normal shutdown uses the secondary side to cooldown and depressurize
  - CVCS is available to maintain RPV level above the riser
- RPV water level may drop below riser elevation following a reactor trip and subsequent cooldown (riser uncovered)
  - Without makeup, water level will drop below the riser within 3-6 hours depending on initial conditions and core decay heat
- RPV temperature is below 420 °F when the riser becomes uncovered

Long Term DHRS Operation (cont)

- After water level drops below the riser, two scenarios are possible depending on heat transfer across the riser:
  - With sufficient heat transfer across the riser, two separate internal natural circulation loops will be established; one in the core/riser and one in the downcomer. In this scenario the riser remains uncovered
  - If heat transfer across the riser is insufficient to remove decay heat, either continuous or intermittent single-phase natural circulation is established
- Collapsed liquid level remains well above the top of the active fuel and a significant fraction of the steam generator tube surface area remains covered
- Applicant concluded, and staff agreed, that adequate cooling exists if the riser becomes uncovered out to 72 hours

Long Term DHRS Operation (cont)

- In the riser uncovered scenario, some water vapor will condense on the exposed steam generator tubes
- This has the potential to dilute the downcomer over a long period of time as water vapor is assumed to have a negligible boron concentration
  - Dilution of the downcomer is limited by the fraction of steam generator surface area uncovered
  - Boron volatility, entrainment and rewetting may help limit downcomer dilution but are not quantified
- Potential exists that reestablishing single-phase natural circulation could transport the diluted downcomer to the core causing a re-criticality similar to the event analyzed during ECCS return to power
- Reestablishing RPV water level above the riser after extended DHRS operation requires the operator to initiate action to recover the module through the addition of water
- Post-accident module recovery is not required to be evaluated in Chapter 15 design basis review

Long Term DHRS Operation (cont)

- NuScale has indicated the recovery of an NPM following extended DHRS operation will be procedurally controlled
  - Plant procedures are not part of the DCA review
  - Procedures would be developed by the COL applicant or holder
  - Chapter 13 COL item addresses the development of operating procedures
- Staff believes procedures could be developed to adequately address recovery from this condition
- Plant design allows for the following operational strategies that could address recovery from this condition:
  - Mixing core and downcomer boron concentration by simultaneous injection and letdown preserving RCS level
  - Downcomer boron concentration sampled to ensure adequate mixing before single-phase natural circulation is reestablished
  - Confirming adequate shutdown margin before restoring level

ATWS Scenario

- The limiting RPV pressure ATWS event is initiated by a loss of A/C power.
- Loss of A/C causes the feedwater pump and turbine to trip.
- Control rods are assumed to fail to insert.
- RPV pressure increases due to loss of heat sink.
- High RPV pressure trips the DHRS to activate
- RPV inventory is lost by lifting the RSVs and discharging into containment
- ATWS is not considered a design basis event (DBE) due to the design of the reactor trip system within the MPS lowering the probably of occurrence below 1.0E-5 per reactor year (see SER Section 15.8)

ATWS Mitigation

- Two ATWS scenarios are possible:
  - Operators insert control rods early in the event
  - Operators delay or take no action to mitigate the ATWS
- In both cases, the RSVs relieve pressure and discharge into containment
- If operators insert the control rods early in the transient as expected, the ATWS event progression resembles the long term DHRS cooldown scenario with the riser potentially becoming uncovered
- If operators delay or take no actions to insert the control rods, enough RPV inventory is lost, the level drops below the riser - breaking natural circulation and establishing a new equilibrium power.
  - A safe state is reached and collapsed liquid level remains above the top of the active fuel

ATWS Mitigation and Recovery

- If operator acts to insert rods before CNV inventory reaches the lowest CNV level ECCS setpoint, the event recovery would be the same as a DBE DHRS cooldown
  – Staff's conservative analysis demonstrates the lowest CNV level is reached in approximately 1 hour
  – The likelihood of operators failing to insert the control rods within 1 hour is highly unlikely
- If the operator could not insert control rods after reaching the lowest CNV level ECCS setpoint additional analysis maybe needed to determine the appropriate operator actions
- ATWS mitigating procedures are dependent on the specific ATWS event and available equipment
- Operator actions to recover the plant following a beyond design event are not within the scope of the DCA review and are developed by the COL applicant or holder
- Chapter 13 COL item addresses the development of operating procedures

Return to Power with Ejected Rod (UOI: OI 15.0.6-6)

- DCA does not address the potential return to power following a postulated rod ejection
- Rod Ejection is evaluated for the short term reactivity response only
  - Consistent with the requirement in GDC 28 and the guidance in SRP 15.4.8 to appropriately limit the rate of reactivity increases associated with certain postulated reactivity accidents, including rod ejection
  - Primarily a check of loading pattern and control rod design such that a coolable geometry is maintained
- The staff determined that the provisions in GDC 27 for evaluating DBAs in the long term are met for the NuScale design because:
  - Control rod ejection accident need not be considered in the long term due to the robust design of the control rod drive housings
  - The staff evaluated the control rod housing design in SER Section 3.9.4

Boron concentration requirements for Mode 4

- Technical Specifications apply to normal operations and allowed DBE initial conditions
  - Technical specifications are not applicable during a DBE
- Generic Technical Specification Table 1.1-1, “MODES” defines Mode 4, “Transition” as requiring a keff less than 0.95
- Conditions to enter Mode 4 from other Modes need to met before entry into Mode 4
- LCO 3.5.3, “Ultimate Heat Sink” states that bulk average boron concentration shall be maintained within the limit specified in the COLR
- Combination of inserted control rod worth and RPV boron concentration ensures Mode 4 keff is less than 0.95
- Ultimate heat sink boron concentration will be set to ensure entry into Mode 4 and refueling operations are possible

Example Mode 3 to Mode 4 Transition, Normal Operation

- RCS is cooled down by the secondary side and CVCS adds borated water to the RPV
- Control rods inserted
- Ultimate heat sink water added to containment and mixes with RPV water volume
- Combination of control rod worth and RPV boron concentration ensure Keff is less than 0.95 prior to Mode 4 entry

Example DBE Recovery

- Plant stabilizes following a DBE
  - Plant may or may not be in a defined Mode per Generic Technical Specifications Table 1.1-1
  - Return to power is an example of stable state not defined by Table 1.1-1
- Operators will establish an operating Mode defined by Table 1.1-1 using available systems
  - For example, Mode 3 with Keff less than 0.99
- Transition to Mode 4 is then the same as a normal plant operation

Changes to Design in Phase 4

- Staff reviewed impact of design/method changes on Chapters 6 & 15 during Phase 4
  - NRELAP5 v1.4 & NPM Model Rev. 2 (UOI: OI 15.0.2-1)
  - ECCS Actuation Logic (UOI: OIs 15.0.0.4-1, 15.6.5-1)
  - IAB block/release pressure
  - DHRS Logic (UOI: OIs 15.0.0.4-1, 15.6.5-1)
- Updated analysis results provided for impacted events in DCD Rev. 3
  - Staff audited revised calculations

Changes to Design in Phase 4

NRELAP5 and NPM Base Model changes (NRELAP5 v1.4, Base Model Rev. 2)

- Reviewed by staff in LOCA topical report and described during Feb. 19, 2020 ACRS subcommittee meeting

ECCS Logic changes

- Removed actuation on riser low level (level indication retained as nonsafety-related sensor)
  - Level activation was not used in Chapter 15 results
  - Actuation only on either loss of DC power, high CNV water level, or low AC voltage after 24 hrs
- CNV water level ECCS trip increased by 24”

IAB Block/Release Pressure changes

- IAB release 950 psid (±50 psi), IAB blocks ≥ 1300 psid
  - 900 psid is limiting for LOCA CLL
  - Block ability important only if loss DC is assumed

Review of ECCS Logic Changes on LOCA (5% injection break)

Break & Loss of AC power (T=0s)

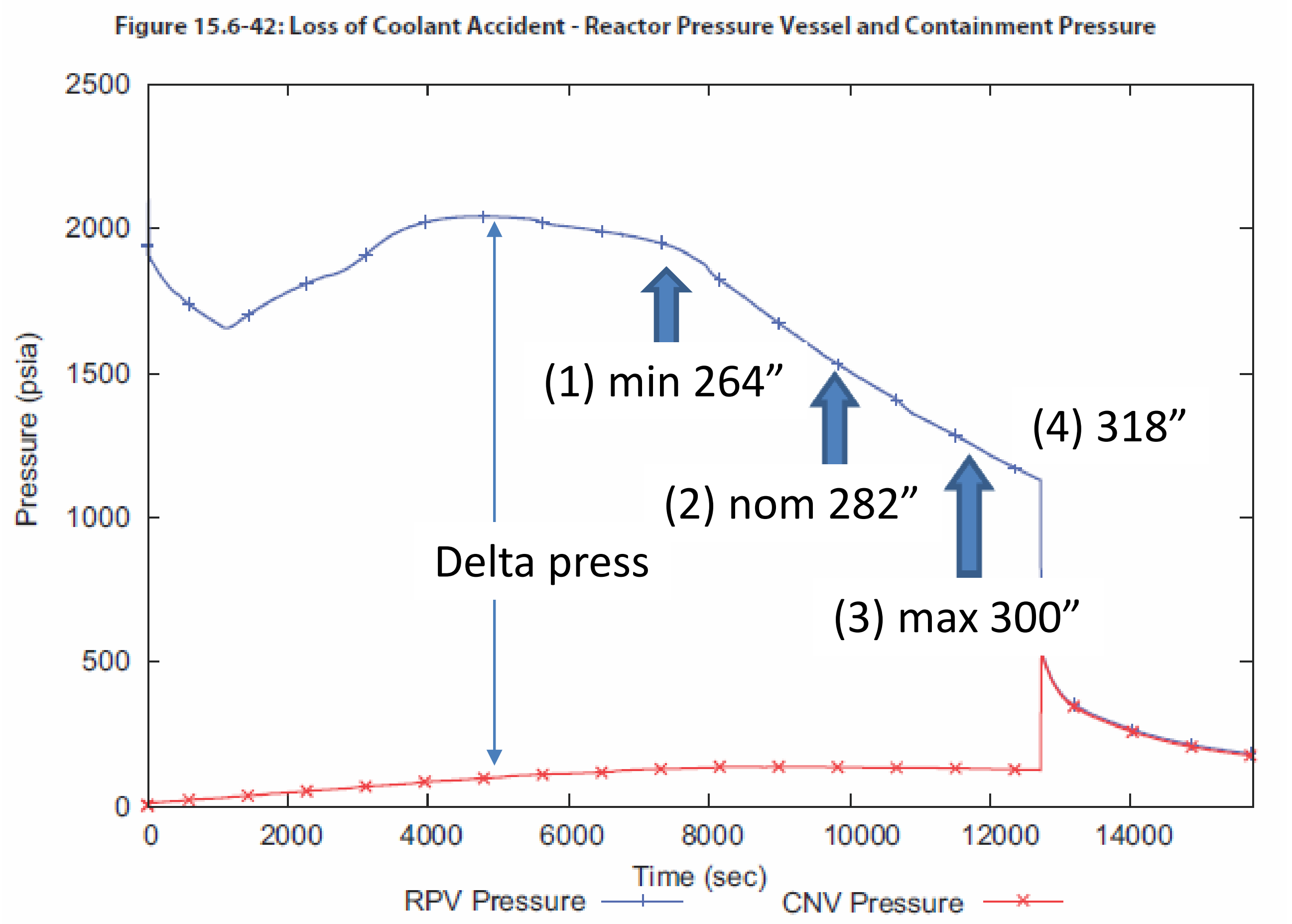
Rx Trip – high PZR press (T=8s), Rods in (T=14s), High CNV press (T=16s)

(1) IAB blocks

(2) "

(3) "

(4) 900 release pressure reached



Changes to Design in Phase 4

DHRS Logic changes

- DHRS signal split into two signals (DHRS actuation and Secondary isolation (SSI))
- Direct DHRS actuation inputs reduced from 13 to 4 signals, high (1) RCS press, (2) temp, (3) steam press, and (4) low AC voltage to batteries, function opens DHRS valves and closes primary and secondary MSIV and bypass, MFIVs and MFRVs Staggered release pressures considered
- New SSI actuation with 12 signal inputs, function only closes primary and secondary MSIV and bypass, MFIVs and MFRVs
- Allows better operator control at startup, reduce frequency of actuation
- Delays DHRS actuation until much later in transient; min change to Chapter 15 FOM margins

Loss of Coolant Accident Analysis

- Staff reviewed changes to LOCA results in Phase 4
- Considered NRELAP5 v1.4 code and min modeling changes, ECCS logic, IAB block/release pressure

Figure of Merit	MCHFR (DCA Rev. 2)	MCHFR (DCA Rev. 3)	Percent Difference
MCHFR	1.80	1.72	-4.65%
Min CLL (ft)	0.14	1.7	1214%

- Minimal change in results: MCHFR 1.29, margin slightly reduced, min CLL margin increased, due to revised method to compute CLL (which staff have accepted)

Long-Term Cooling Analysis

- Two LTC situations evaluated by NuScale
  - DHRS cooling
  - ECCS cooling
- Staff review documented in SER Section 15.0.5 and 15.6.5
- LTC methodology documented in technical report incorporated by reference into DCD Chapter 1
  - LTC methodology addresses the ECCS cooling after recirculation is established
  - LTC methodology assumes subcriticality; return to power addressed in DCD Section 15.0.6
- Phase 2 SER included OI (UOI:15.0.5-2) as LTC technical report had stated cooling was demonstrated to 30 days
  - NuScale revised statement and staff SER documents review to 72 hours

Long-Term Cooling Analysis (ECCS Cooling)

- LTC starts after quasi-steady state recirculation in ECCS steaming mode
- NRELAP5 run to 12 hrs (collapsed core), state point method used to infer 72 hr condition, based on decay heat multiplier assumed (OI 15.6.5-2)
- Worst re-criticality scenarios evaluated in 15.0.6 (OI 15.0.6-5)
- NIST-1 Tests HP-19a & b provided validation

Figure of Merit	Acceptance Criteria Coolable Geometry	DCA Rev. 4 Results
Max RCS Temp	-	280 F
Min RCS Temp	-	94 F
Boron Precipitation	> 0 F	17 F
Minimum CLL	>TAF	2.8 ft

- Min & max RCS temp to 72 hr, considering boron precipitation, and min CLL, all showed adequate margin to FOMs

Changes to Selected Analyses in Phase 4: IORV (15.6.6)

Results for the limiting MCHFR case (inadvertent opening of an RVV):

Figure of Merit	Acceptance Criteria Limit	DCA Rev. 4 Results
RCS Pressure	≤2310 psia	1796 psia
SG Pressure	≤2310 psia	1037 psia
MCHFR	1.13	1.32
Minimum CLL	>TAF	10.2 ft

- Staff audited the underlying calculations.
- The change to the IAB block and release thresholds did not have an impact on the limiting case since MCHFR occurs early in the transient.
- The staff finds the applicant's analysis acceptable, and the resulting figures of merit are within their design limits.

Changes to Selected Analyses in Phase 4: 6.2.1.1, Containment Structure

FSAR + Containment Response Analysis Methodology (CRAM) Tech. Report

- Conservatisms in the CNV model/initial & BCs
- Staff confirmatory analyses of 7 CNV DBE scenarios
- Applicant & staff sensitivity studies:
  - CNV inner surface condensation heat transfer modeling
  - Effect of non-condensable gases
  - Nodalization of containment volume and heat structures
  - Thermal stratification effects inside CNV & cooling pool
- CNV Design Margins: **994 psia** vs. 1050 psia; **526 ^{o}F** vs. 550 ^{o}F
- NuScale CNV design has sufficient conservatism.
- NuScale CNV design meets all regulatory requirements.

Changes to Selected Analyses in Phase 4: 15.4.3, 15.4.7 & 15.1.5

- Two static reactivity events showed significant changes in calculated MCHFR between DCA Revisions 2 and 3:

Event	MCHFR (DCA Rev. 2)	MCHFR (DCA Rev. 3)	Percent Difference
15.4.3, Control Rod Misoperation (Misalignment)	2.509	1.437	-42.73%
15.4.7, Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	1.916	1.437	-25.00%

- The staff audited the underlying calculation notes and confirmed that the differences are primarily due to use of:
  - A more conservative, bounding radial peaking augmentation factor
  - A more conservative core inlet temperature (from 497.4 °F to 510 °F)
- The staff finds that the inputs and assumptions used are acceptable, and the resulting MCHFR remains above the 95/95 limit

Changes to Selected Analyses in Phase 4: 15.4.3, 15.4.7 & 15.1.5

- Increase in RCS cooling events (15.1) and decrease in RCS cooling events (15.2) did not show significant changes in figures of merit between DCA Rev. 2 and Rev. 3
- 15.1.5 Steamline break analysis results DCA Rev. 2 vs Rev. 3:

Figure of Merit	MCHFR (DCA Rev. 2)	MCHFR (DCA Rev. 3)	Percent Difference
MCHFR	1.861	1.866	0.27%
Maximum RCS Pressure (psia)	2156	2081	-3.48%
Maximum SG Pressure (psia)	1346	1495	11.07%

- The staff audited the underlying calculation notes and confirmed that the changes from DCA Rev. 2 to Rev. 3 do not have significant impacts
- A small impact observed on RCS and SG maximum pressure for some events due to timing delay from the DHRS logic change (Secondary System Isolation and DHRS actuation signals)
- The staff finds that the changes are acceptable, and the resulting figures of merit are within their design limits

ECCS Design: Water Hammer Issue Resolution

- ECCS Valves Trip Valve Hydraulic Line
  1. Different trip valve hydraulic line length for each valve
  2. Fluid inside the lines experiences two-phase flashing
  3. Staff requested full scale, high temperature and high pressure tests to confirm no water hammer effects
- As part of ECCS/IAB testing, NuScale performed full scale testing using the longest hydraulic line and multiple 90 degree bends
  1. Test results of all test cases did not show any pressure spike
  2. Water hammer phenomenon were not observed

ECCS Valve System Follow-Up Items

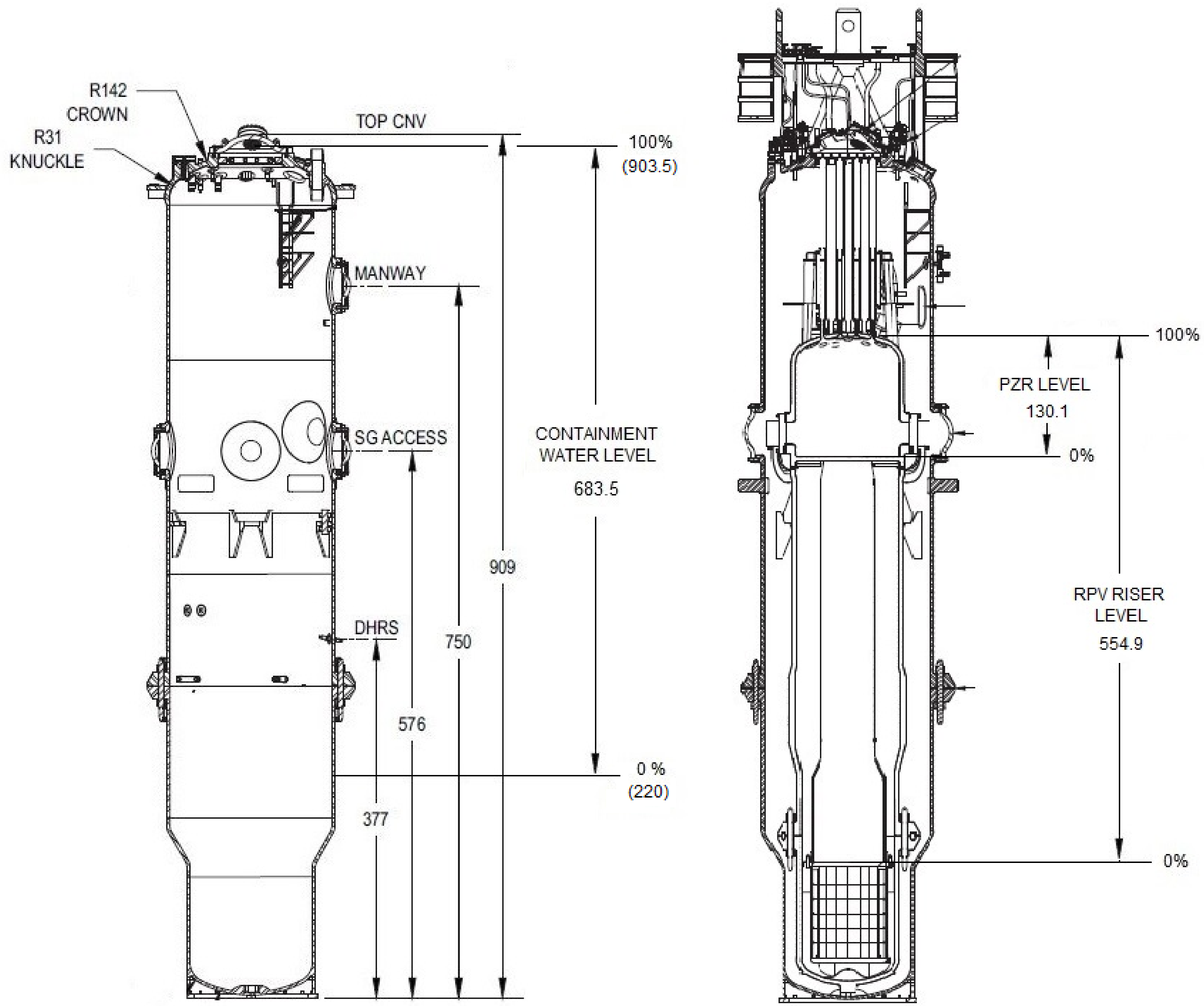
- ASME Standard QME-1 qualification of the ECCS valves and their hydraulic lines will need to include demonstration that small ports and tubing are not subject to plugging or fouling due to boron precipitation and other degradation modes over time.
- NuScale has stated that the temperature of the ECCS valves and their hydraulic lines will remain above the precipitation temperature of boron during plant operation.
- NuScale plans to flush the ECCS valves and their hydraulic lines during each refueling outage to remove any particulates that might unexpectedly accumulate during plant operations.

CNV and RPV Level Instruments

Level Transmitter	Indicated Range (Span) [Process Range]	Nominal (100% RTP)	Function	Safety & Risk Classification
Containment Water Level	0 to 100% (683.5 Inches) [approx. 220 to 903.5 Inches[1]]	0%	ECCS Actuation 264" to 300"[2] High Level L-1 Interlock >540" & RT-1 active (Reactor Trip Breakers Open) PAM Variable Type B, C, D	A1
Pressurizer Level[3]	0 to 100% (130.1 Inches) [Full height of PZR]	50%	Reactor Trip 80% High Level 35% Low Level Secondary Sys Isolation 20% Low-Low Level Containment Sys Isolation 20% Low-Low Level Demin Water Sys Isolation 80% High Level 35% Low Level CVCS Isolation 80% High Level 20% Low-Low Level Pressurizer Heater Trip 35% Low Level L-2 Interlock >20% & T-3 active (RCS T_{hot} <350°F)	A1
RPV Riser Level	0 to 100% (554.9 Inches) [Top of upper core plate to top of PZR]	100%	PAM Variable Type B, C, D	B2[4]

[1] Levels are reported in terms of module elevation with the global zero elevation at the bottom of the reactor pool.
[2] The ranges allow ±18" from the nominal ECCS level setpoint of 282"
[3] Common Level Transmitter is used for Pressurizer Level and RPV Riser Level
[4] Common Level Transmitter is used for Pressurizer Level and RPV Riser Level. However, function of RPV Riser Level is classified as B2

CNV and RPV Level



Acronyms

•	AC	alternating current
•	ACRS	advisory committee on reactor safeguards
•	AOO	anticipated operational occurrence
•	ASME	American society of mechanical engineers
•	ATWS	anticipated transient without scram
•	BOC	beginning of cycle
•	CFR	code of federal regulations
•	CHF	critical heat flux
•	CHFR	critical heat flux ratio
•	CLL	collapsed liquid level
•	COL	combined license
•	CNV	containment vessel
•	CVCS	chemical and volume control system
•	DBA	design basis accident
•	DBE	design basis event
•	DCA	design certification application
•	DHR	decay heat removal
•	DHRS	decay heat removal system
•	ECCS	emergency core cooling system
•	EDSS	highly reliable dc power system
•	EOC	end of cycle
•	FOM	figure of merit
•	FSAR	final safety analysis report
•	GDC	general design criteria
•	HTC	heat transfer coefficient
•	IAB	inadvertent actuation block
•	IORV	inadvertent opening of a RPV valve
•	LOCA	loss of coolant accident
•	LTC	long term cooling
•	MCHFR	minimum critical heat flux ratio
•	MOC	middle of cycle
•	MPS	module protection system
•	MFIV	main feedwater isolation valve
•	MFRV	main feedwater regulating valve
•	MSIV	main steamline isolation valve
•	NPM	NuScale Power Module
•	OI	open item
•	PA	Postulated Accident
•	PDC	principle design criteria
•	PZR	pressurizer
•	QME	qualification of active mechanical equipment
•	RAI	request for additional information
•	RCS	reactor coolant system
•	RG	regulatory guide
•	RPV	reactor pressure vessel
•	RSV	Reactor Safety Valve
•	RTP	rated thermal power
•	RTNSS	regulatory treatment of non-safety systems
•	Rx	reactor
•	SAFDL	specified acceptable fuel design limits
•	SER	safety evaluation report
•	SFC	single failure criteria
•	SG	steam generator
•	SGTF	steam generator tube
•	SH	super heat
•	SLB	steamline break
•	SRP	standard review plan
•	TAF	top of active fuel
•	TR	topical report
•	UOI	unclear open item

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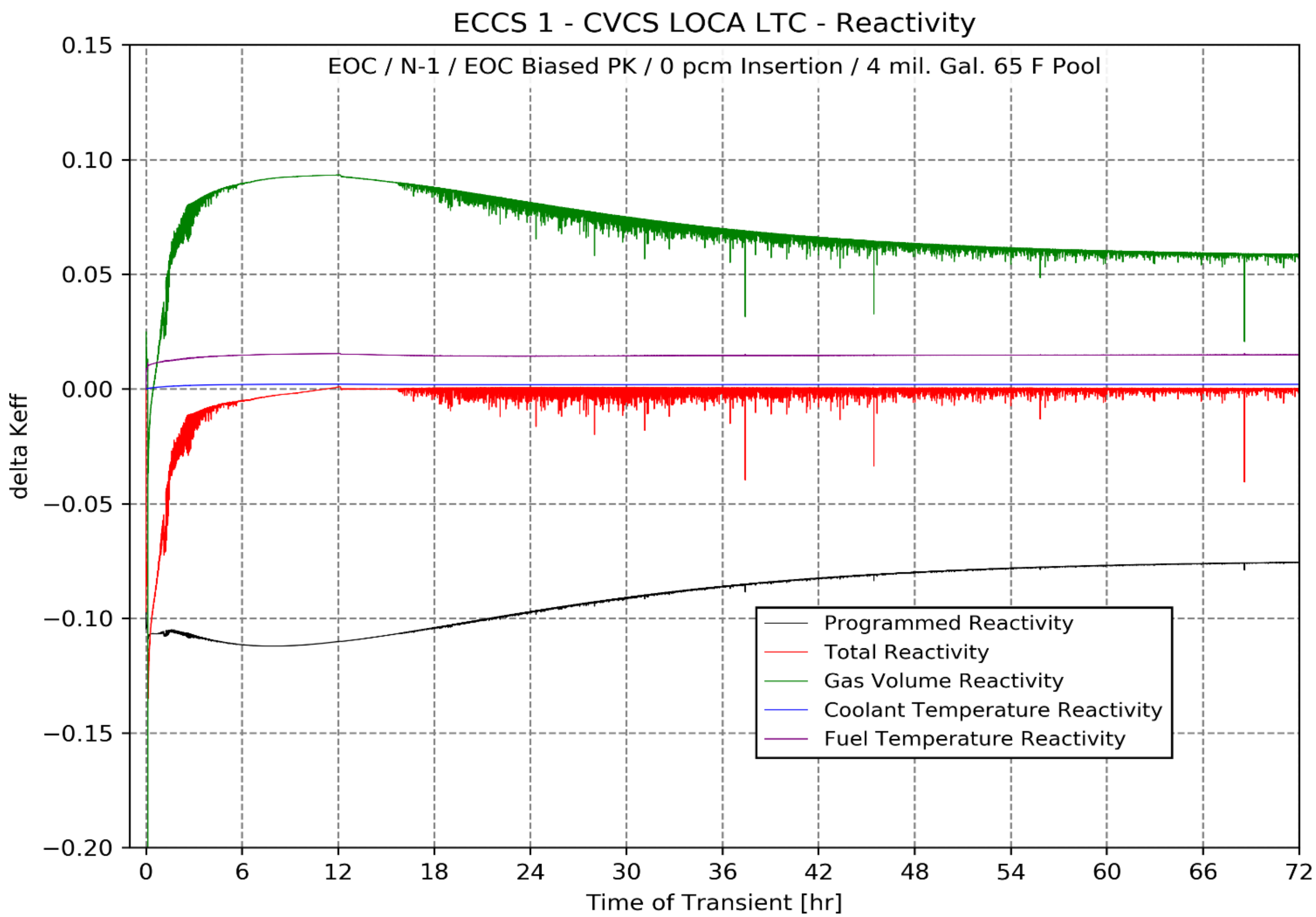
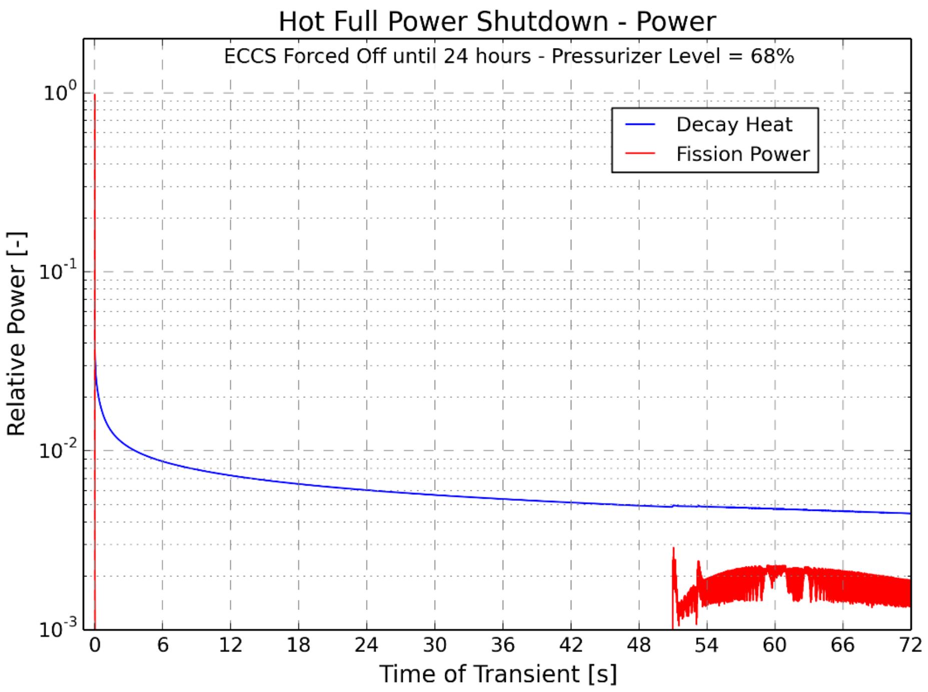
Backup Slides

Reactivity Worths, EOC ECCS Confirmatory Analysis

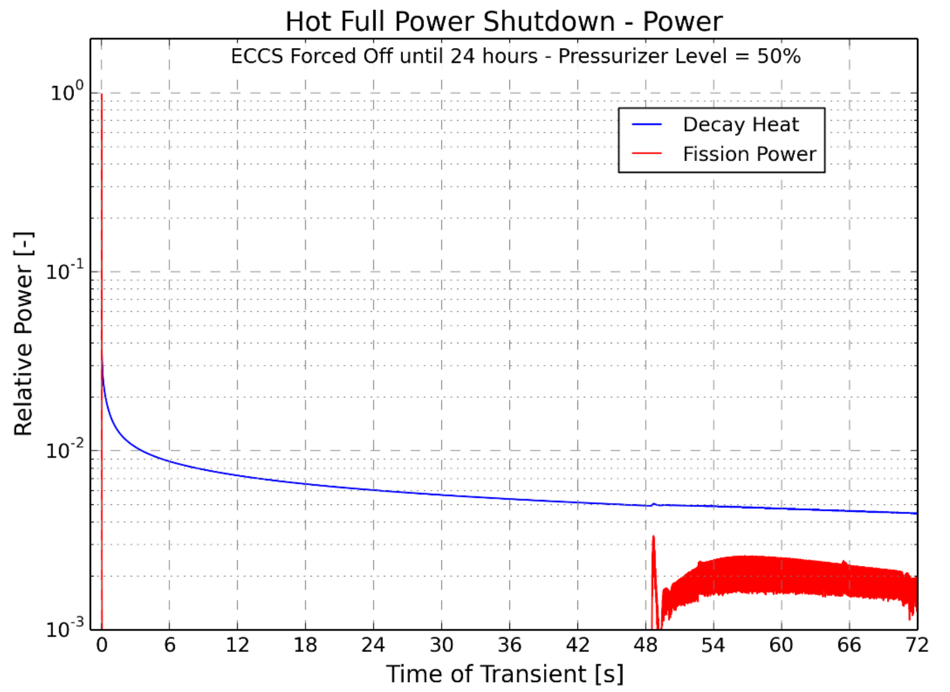


Core Power, EOC, HFP DHRS Cooldown

High PZR Level

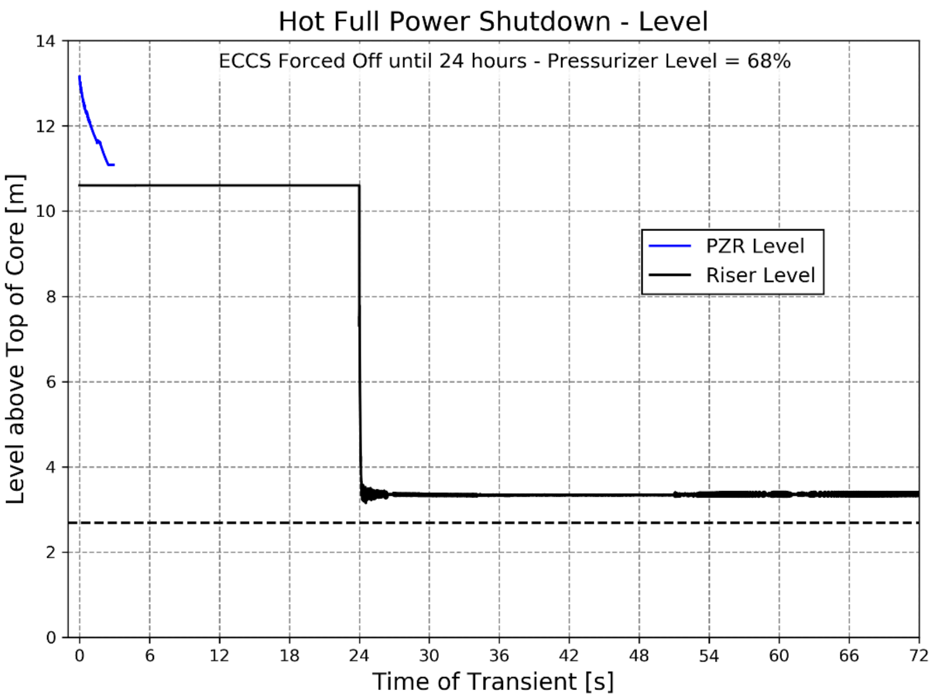


Low PZR Level

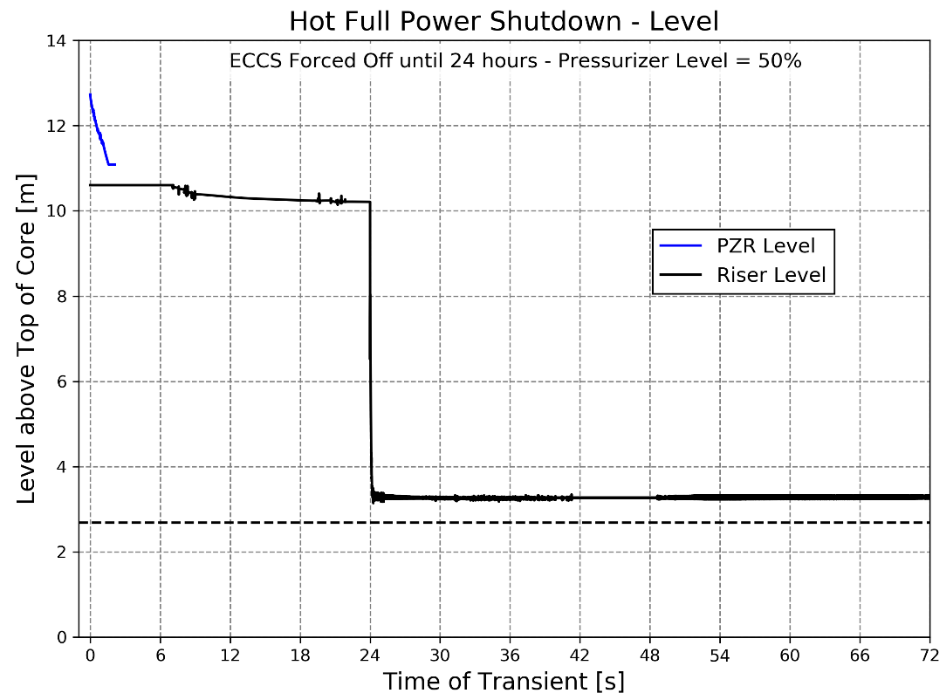


HFP DHRS Cooldown – Level in RCS Components

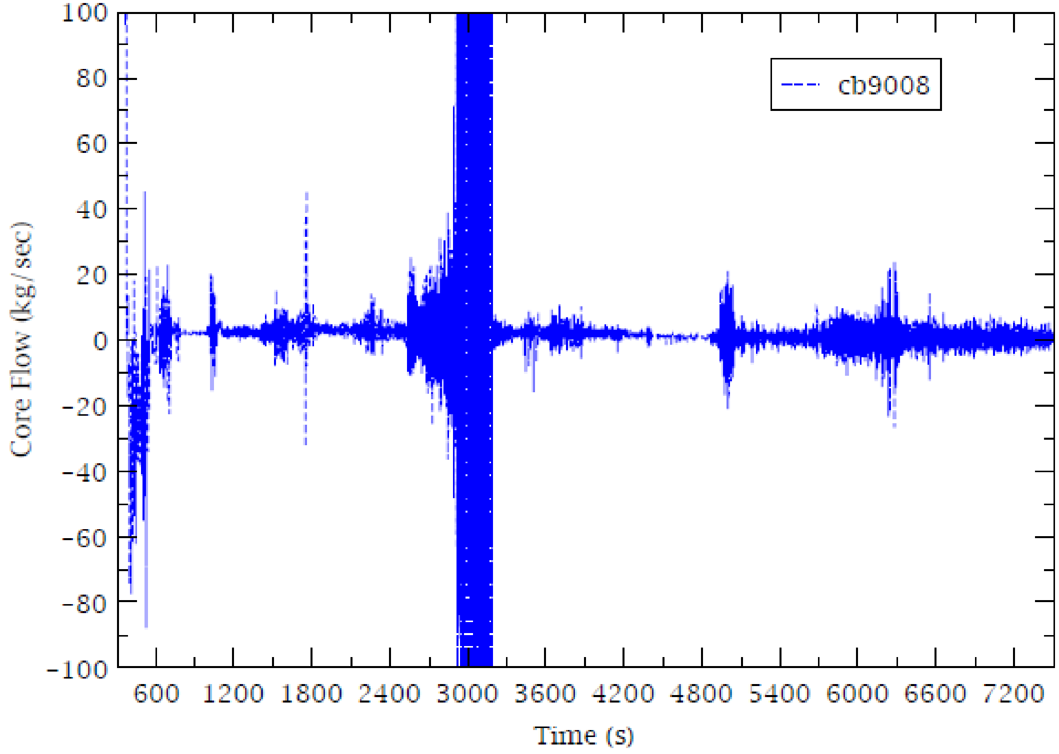
High PZR Level



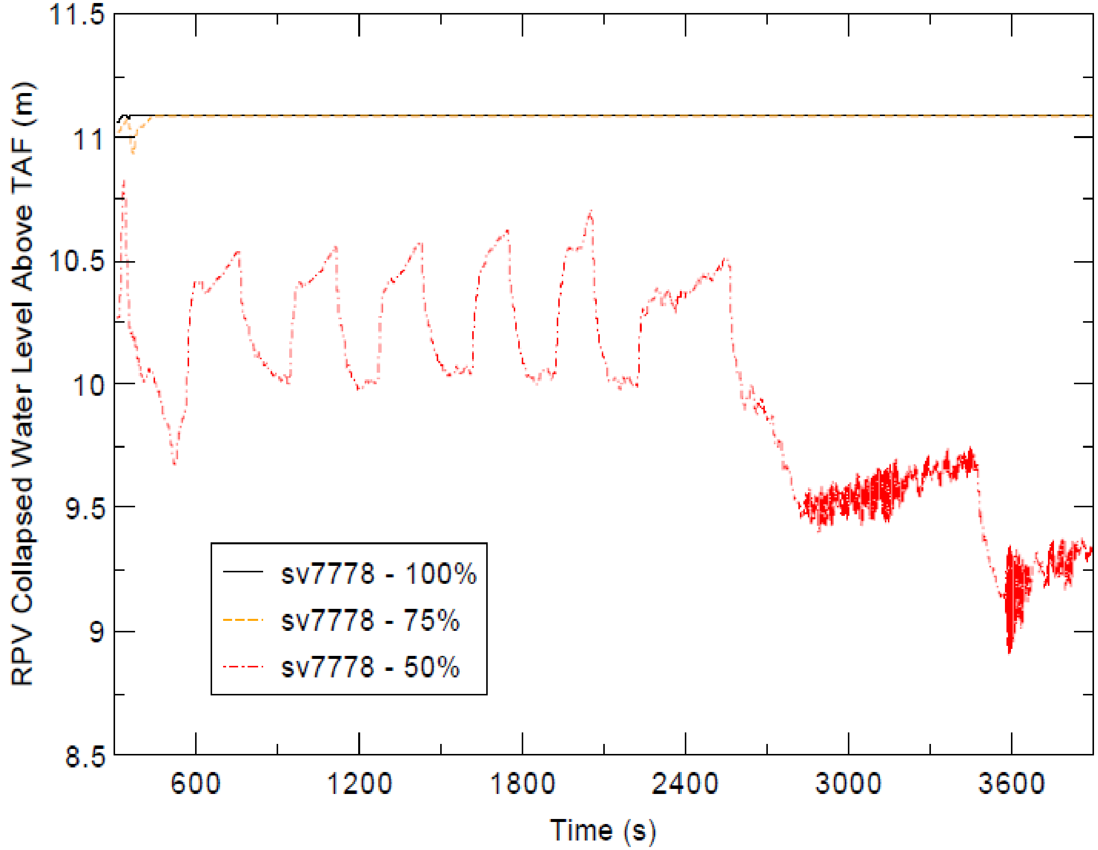
Low PZR Level



Core Flow during RRV ATWS w/o ECCS

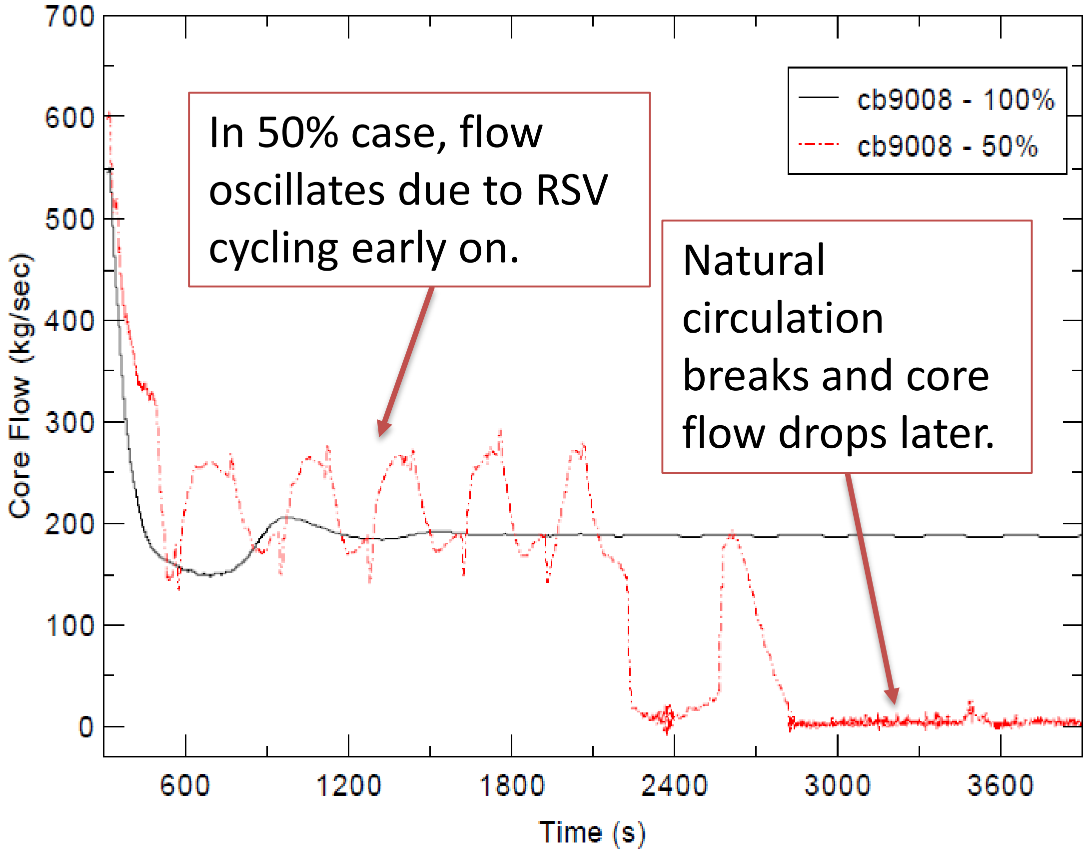


RPV Collapsed Level (1 hour)



- In the 50% case, voiding occurs in the riser due to high RCS temperature.
- After pressurizer drains, the riser level begins to drop further.
- Eventually level drops below the top of the riser and the natural circulation flow loop is broken.

Core Flow (1 hour)



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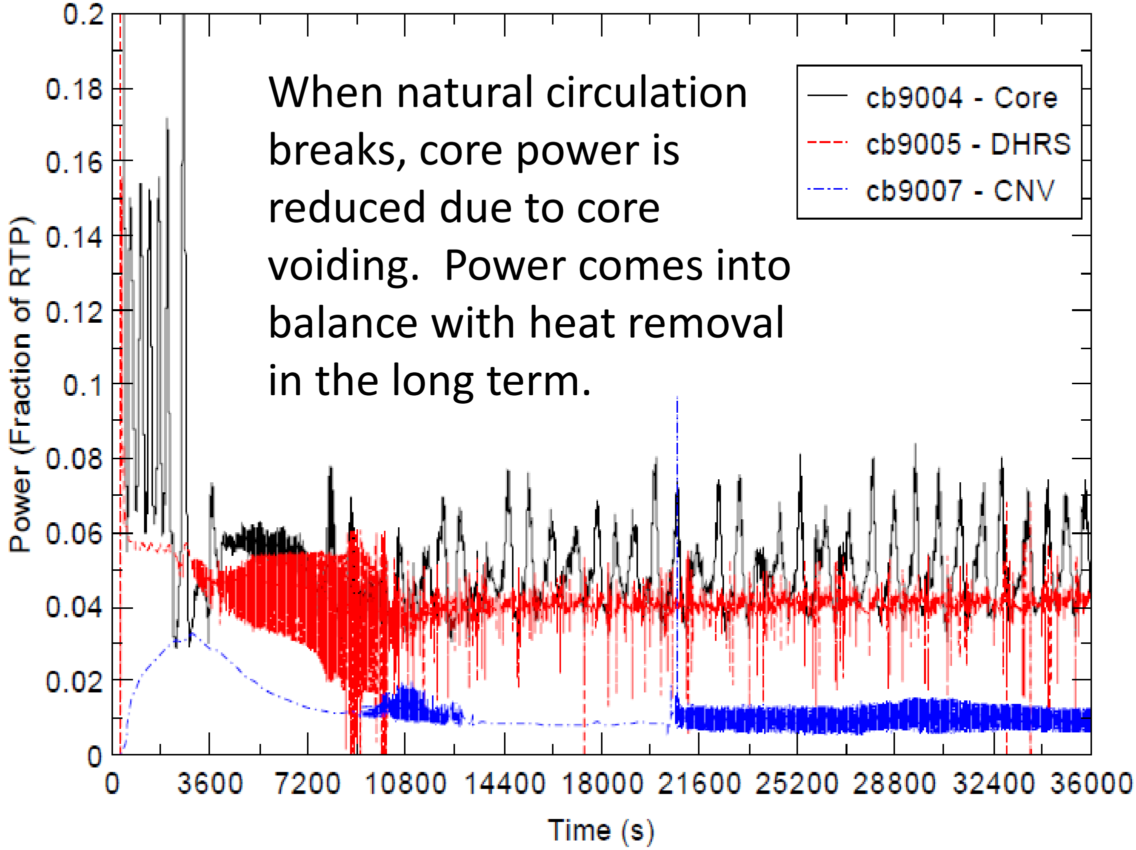
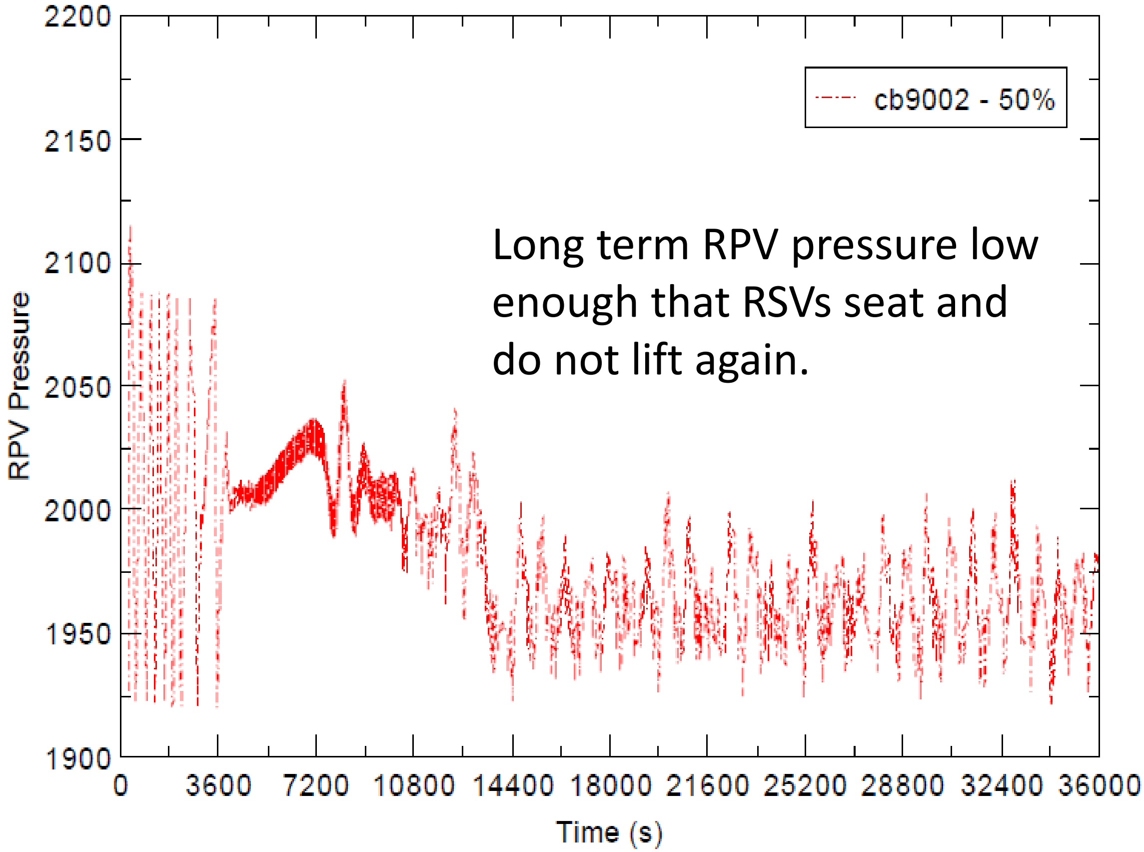
50% Case - Heat Balance (10 hours)



50% Case - RPV Pressure (10 hours)



50% Case – RPV Level (10 hours)