Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title:	Advisory Committee on Reactor Safeguards NuScale Subcommittee: Open Session
Docket Number:	(n/a)
Location:	Rockville, Maryland
Date:	Thursday, June 20, 2019

Work Order No.: NRC-0389 Pages 1-255

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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NuSCALE SUBCOMMITTEE

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THURSDAY

JUNE 20, 2019

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ROCKVILLE, MARYLAND

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The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2D10, 11545 Rockville Pike, at 8:30 a.m., Jose March-Leuba, Chair, presiding.

COMMITTEE MEMBERS:

JOSE MARCH-LEUBA, Chair

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CHARLES H. BROWN, JR. Member

MICHAEL L. CORRADINI, Member

VESNA B. DIMITRIJEVIC, Member

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MATTHEW W. SUNSERI, Member

ACRS CONSULTANT:

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

MIKE SNODDERLY

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P R O C E E D I N G S

8:30 a.m.

MEMBER CORRADINI: Okay. Why don't we get started. Just to remind everybody, we're in the third day of our subcommittee meeting on NuScale's DCA.

I'll remind you of all the normal things. Make sure all your appliances are muted, turned off or subdued so that we don't have interruptions. We're going to be speaking to the staff today about their SER review at least with open items. And I'll turn it to Dr. March-Leuba.

CHAIR MARCH-LEUBA: Not much to say except before management gives an introduction, yesterday we presented some information reading from the slides about the initial critical heat flux ratio, CHFR, that there are different numbers on different plots. And I did a compilation on that, and there seemed to be two clusters of initial CHFRs, one around 2.8 and one around 1.6, which will be consistent with having two different evaluation methodologies.

But even around those clusters, there are 2.8, 2.6, 1.6, 1.8. There are differences which might be accountable for different initiated conditions or different versions of the code that were run over time. So I would like -- you don't have to do this now. But before we finalize the SER, maybe for the full committee, have some confidence that the staff really understands the differences.

Now from the point of view of strategy, it's supposed to be flying at 40,000 feet. What is the CHFR, 1.6 or 2.8? I mean, there is a big difference. And in particular for events, 15.1.1 and 15.1.2, the delta CHFR is almost 1. It's 0.85. So if the CHFR at time zero was really 1.6, it would have been bad situation. Clearly it's not. But this needs to be addressed.

And frankly, we were talking earlier before and among ourselves that this is the 21st century. We can't have a factor of 2 in CHFR evaluations in a steady state. And so when the LOCA and non-LOCA topical reports come from the review, we'll again bring this to the floor. I mean, we can have a little uncertainty on the transient. But on the steady state, we should know what it is.

MEMBER CORRADINI: My guess it's in initial conditions. But I agree with that the staff has got to at least be clear as to what's going on relative to the applicant's numbers.

CHAIR MARCH-LEUBA: Is it biased on the initial conditions because of Appendix K for the staff? Or is it because you're using two different correlations? I would like to know that when we review the topicals. And now I think Rebecca wants to give us some wise words.

MS. KARAS: This is Becky Karas from Reactor Systems. I just want to thank the committee for their time in this. In reviewing this, as you're aware, we have a number of issues in this draft SER that are not yet with a clear path to resolution where we could -- they would be legitimately characterized as open items.

So we'll be talking through many of those as well as some of the issues where we do have a clear path to resolution and/or they may even have been closed since the time that the SE was drafted. We understand and appreciate your comments on CHF. We'll be discussing some of that today. And we can certainly also address any remaining items when we come with the topical reports.

MEMBER CORRADINI: So can I follow up? Because you had a -- you kind of had a preamble, a special note in the SE. And the sentence that hit me was, however, in this preliminary SE, the uncommon OIs have been presumptively characterized as OIs, recognizing their advancement in status.

So as the staff goes through this, if you can identify the ones that are uncommon because there's enough of them. I had a bit of a problem deciding what were common and uncommon, right? And that links back to 6.2.1 and 6.2.2 which you're going to present today anyway, which at least as far as I could tell, there were three there also.

And somehow, I guess, I want to be -- personally want to be educated so we're clear that you guys have captured what you're worried about and what bin they fall in, at least from your perspective.

MS. KARAS: If I can just interject into that real quick. The decision of when it moves from an open item that doesn't yet have a clear path to resolution, whatever we call them, uncommon or I'm not sure what we're calling them today. But when it moves from that to moving to a legitimate open item, that decision is made at my level.

So addressing that with an individual staff member, they may not yet know when that decision has been made, not just my level but my management. So on any individual item --

(Simultaneous speaking.)

MS. KARAS: -- that you're asking, I can tell you the status today of it. They'll talk about where they are and the status of the review. But as far as the ones where we don't yet have, like, a common understanding with NuScale or a clear path to resolution, you could -- if you have questions on a specific one, you can address that and I'll respond.

MEMBER CORRADINI: So we'll let you chime in when the time is right.

MS. KARAS: Sure.

MEMBER CORRADINI: We trust you will.

CHAIR MARCH-LEUBA: Yeah, and I'm sure you will here on the July full committee meeting when we write the letter. Because this is an unusual circumstance and it's going to make for a funny letter and we can make mistakes, what is uncommon, what is not. And we will talk about it. Maybe we will have to list every one of them when we get to the problem at least. But because you've written the SER, I see they were resolved. It could feel like we are approving what you wrote.

MS. KARAS: Right. Yeah, just to be clear, it's presumptively written as an open item. We didn't differentiate because the status of any one is in flux as you move through time and the SE moves through concurrence. So to the extent that the committee feels the need to differentiate, then we could provide that clarity at that time. But yeah, in July, we understand.

CHAIR MARCH-LEUBA: We need to --

MS. KARAS: We'll support with whatever you need.

CHAIR MARCH-LEUBA: We'll need your -- you're here for technical ideas. Shanlai or Alex? Who's in charge?

MS. SIWY: Thank you. Good morning, everyone. My name is Alex Siwy, and I'm a technical reviewer in the Reactor Systems branch. This presentation will cover the staff's review of Chapter 15 of the NuScale DCA Part 2 transient and accident analyses, and it will also touch on some aspects of Chapter 6, specifically Section 6.2.1, containment functional design, and 6.2.2, containment heat removal.

So here's a high-level agenda of what we plan to cover today. First, I'll introduce the staff review team and give a very high-level summary and status of where we are with the review. Then we'll go into some kind of generic issues related to Chapter 15 analyses including methodology, input assumptions, event categorization, and generic evaluation. Then we'll dive into some of the key Chapter 15 events, and then we'll move into the containment performance aspects.

So this slide lists the NRC review team. Obviously, it's pretty large so I won't read you the whole list. Several of these people will be presenting today, and so we'll kind of have to play musical chairs and switch out several times. So just bear with us when we do that. In addition, Rani Franovich is the project manager for Chapter 15 as well as three of the four related topical reports.

So as a high-level summary and status of our review, the staff safety evaluation report is based on DCA Revision 2. The SER contains 32 open items and 9 confirmatory items, and that's part of the reason for the special note at the beginning of the SER which also acknowledges that not all of the open items may have a clear path to resolution at this time. However, we just wanted to note that significant progress has been made on several of the items since the SER was transmitted to you. And we will highlight some of the major open items throughout this presentation.

In addition, the staff evaluates two exemption requests in Chapter 15. The first one is exemptions to portions of Appendix K which is in

Section 15.0.2 of the staff's SER and an exemption to GDC 27 which is discussed in Section 15.0.6 of the staff's SER. And finally, the staff has initiated four audits for Chapter 15.

MEMBER CORRADINI: Since we're on this one, I think I understand the GDC 27 since we kind of went over that a bit, a few times. I don't understand the 15.0.2. What was the reason -- I think we asked this of NuScale yesterday, and I guess I want to ask the staff today. What was the reason you need an exemption if there are certain physical phenomena that just simply do not occur? Then tool and the portions of the evaluation model tool just simply don't apply.

What was the need for the exemption which leads it to more of a -- it seems like it raises the issue and I don't see the need. I'm confused.

MR. LU: This is related to LOCA, so -- oh, okay. This is related to LOCA, a topical report. So actually, I have some slide about that.

MEMBER CORRADINI: Okay. All right.

MR. LU: And --

MEMBER CORRADINI: So if it's going to come up later, we can wait.

MR. LU: Yeah, yeah. I can explain actually. I have other very high-level summaries.

It's just the licensing process. We consider its complete process from their perspective. I think that's a good decision. It was also --

MEMBER CORRADINI: It was what?

MR. LU: It was part of the licensing process, and then they want to follow the practice. And then remember that even when we have a few -- we have it operating fully. We have a different new view, then they still have the exemption request to a certain degree. So it's not that much a big deal as --

MEMBER CORRADINI: Okay.

MR. LU: -- I'm going to cover. But I think they're following the right process --

MEMBER CORRADINI: Okay.

MR. LU: -- to do it.

MEMBER CORRADINI: Okay. All right.

CHAIR MARCH-LEUBA: But we said yesterday it has bad visibility from the point of view of the public. If I'm an interested member of the public and I see that NRC is issuing exceptions for this reactor, it feels like they cannot get the license unless they have friends in high floors or something like this which is not the case. The case is that these things are methodical and don't make a difference.

MR. LU: You are correct --

CHAIR MARCH-LEUBA: The visibility is very bad.

MR. LU: -- that's really bad. Yeah, for Appendix K part of the exemption, yes. I don't think that's the case we have, a specific -- allow that one to happen only for NuScale. And then so it's not something --

(Simultaneous speaking.)

CHAIR MARCH-LEUBA: Typically, what we've done in the past is the applicant has in the topical report, my methodology says that this point or this assumption doesn't apply to NuScale.

MR. LU: Yeah.

CHAIR MARCH-LEUBA: And the staff says, we concur.

MR. LU: Yeah.

CHAIR MARCH-LEUBA: That's it.

MR. LU: Yeah, but from another perspective, they do have a good design. That's what comes out of the features here.

CHAIR MARCH-LEUBA: After they're going to accept you legally, the applicant has a much stronger case to apply it.

MR. LU: That's right.

CHAIR MARCH-LEUBA: But the visibility is bad. So it certainly needs to be explained in that sense that it's not something related to safety whatsoever.

MR. LU: That's right. So that will be covered as a part of my summary for LOCA topical.

CHAIR MARCH-LEUBA: Okay.

MS. SIWY: Okay. So here's an overview of all the more generic issues related to Chapter 15 that may touch several different Chapter 15 events. So first, we'll go over the topical reports at a high level and we'll talk about the re-analysis of Chapter 15 events using NRELAP5 Version 1.4 in the revised base model. We'll also touch on design changes to DHRS and ECCS logic in actuation signals. We will then talk about credit for nonsafety-related valves, treatments of IAB single failure. We'll touch on the GDC 27 exception and then finally return to power and long-term cooling analyses. And I'll turn it back over to Shanlai for the --

MR. LU: All right.

MS. SIWY: -- LOCA topical.

MR. LU: All right. I'll cover the LOCA topical report. As part of 15.0.2, and that refers a lot spectrum methodology. So as part of that GDC review, I'm asked to give a summary of a LOCA topical report or review status.

So it's not full blown. I'll give you guys a presentation starting from PIRT, scaling all the way to the testing, everything. It's just a summary, so we are hoping that we can resolve all the issues in the fall of 2019 and then give you a full blown, one day or two day, whatever you ask for, the detail presentation started from PIRT.

CHAIR MARCH-LEUBA: I believe it's October 17.

MR. LU: October 17? Okay.

CHAIR MARCH-LEUBA: Mark your calendar.

MR. LU: Okay. Well, I think so at a very high level, this topical report is a significant effort from both, mostly applicant and NRC staff. And then that was viewed on top of almost ten years of testing and the conceptual design of NuScale. So really it was heavily invested.

And from the staff perspective, myself, Carl Thurston, Jeff, and then we have also Peter Lien from research and we have a consultant from Brookhaven National Lab and then four consultants from Numark, and we all were involved. And then from another side of that, we do have also NuScale side. We have their analysis team and the testing team too. So this was really how we invest in the topical report.

As it is today and actually we are, from grand scheme-wise, we still have issues. And then we are trying to resolve CHF issues related to inadvertent opening of ECCS valves. We also have been working on 1.4. But in terms of number of issues, we're past the peak. We are grinding down towards the resolution. I think we have good feelings that we may be able to according to the schedule and then come back to you guys and then talk to about it. Depends on the NuScale side the response time, right?

So with that, I'm going to cover what's the high level of the topical report, the application scope. Initially, it was just LOCA. So after, we dive into the containment analysis and we found, okay, yeah, indeed they have a technical report summarizing the containment pressure analysis, a technical report as a methodology. But the methodology defines most of the input for the model.

So the methodology itself, the RELAP5 and then the assessment refers to this topical report. So that's part of this approval. We plan to at least mention about or at least in our conclusion and the proof by saying, okay, this methodology and the testing data PIRT is applicable for NuScale to use to calculate the peak containment pressure.

And after we issue some RAIs and I'm going to get into a little more details and then the NuScale added is the inadvertent opening of ECCS valve case as part of AOO but just one case in the appendix too. So we have staff covering the CHF part to give you a status. We are not saying that we are -- we reached a point we are very close to everything. And so that's the -- if you look at the scope or the high-level scope, those are the three items there. Okay, next slide. Okay. All right.

This part, I'm going to answer Mike's question about exemption stuff. So the first part of the 50.46, the peak cladding temperature criteria, I don't even want to go through that one because I've gone through that so many times, right, 2,200 degree Fahrenheit, 1,700, 17 percent maximum local oxidation, one percent of the hydrogen generation and long-term cooling.

That's actually you put it, but it's addressed in a separate technical report and in support of DCA 15.0.5. So the LOCA topical will stop a short time, a short term. And the long term is covered by that technical report related -- anything related to decay heat removal or return to power. That's covered over there.

All right. Let me just go through why and they ask for the exemptions. And as I mentioned, as part of 6.3, if just from decay heat removal perspective, this design -- the ECCS design is very simple, how it works, and the gravity treatment and passive. And with that feature, we call it a really elegant design from my perspective.

And with that elegant design and all those phenomena bothering the new operating fleet of applicants for a long time. And therefore, so every single LOCA topical contained in the past have to address swelling and the rupture of the cladding and a few rod thermal parameters, reflood and refuel and pump model, of course. And then there's specific BWR. It's basically going back to 1979. Those items were wrote into Appendix K and part of 50.46.

So those issues are specifically targeted to, at that time, the operating fleet and some of the plans. And although they are still we consider safe, but they are having changes for a while. They had changes, but they demonstrated they are okay at this point to satisfy all those models, radiation, heat transfer, and reflood, the refuel.

A lot of testing was done there too. But since with NuScale's passive ECCS design and these issues of -- it's far really from the real phenomena that NuScale would encounter. So therefore, we have no problem.

CHAIR MARCH-LEUBA: I often call my approach to that of a sales -- car salesman, repeat -- a used car salesman, repeat it over and over again.

MR. LU: Okay.

CHAIR MARCH-LEUBA: So for example, you have the three with the exceptions that have been requested and been pushed to all the way to the commissioners --

MR. LU: Right.

CHAIR MARCH-LEUBA: -- that they don't have a pump model. They don't have a pump.

MR. LU: That's right.

CHAIR MARCH-LEUBA: The simplest solution, the path of least resistance and best for the visibility with the public would say that the SER from the staff says NuScale methodology is not approved for slowing the rupture of the cladding. If any transient results in rupture of the cladding --

MR. LU: Right.

CHAIR MARCH-LEUBA: -- you don't have an

approved methodology.

MR. LU: Right.

CHAIR MARCH-LEUBA: And the SER says that. And I'm going to stop there because they're never going to slow down rupture of the cladding. And the same with reflood and refuel and certainly with the pump is obvious. You don't need a pump model because we don't have a pump. It's all easier to satisfy with an SER limitation than moving it up to the tenth floor. And that's certainly the one member's opinion --

MR. LU: Right.

CHAIR MARCH-LEUBA: -- which may or may not get reflected in the letter.

MR. LU: Yeah. Well, I think from a staff perspective, I agree with that statement. And on top of that is simply because that's the next two items there. And we can with the current design and we can for LOCA. In the basis of LOCA, they can maintain collapse if liquid level remains above the top of active fuel.

CHAIR MARCH-LEUBA: So those three models --

MR. LU: You don't need it.

CHAIR MARCH-LEUBA: -- you don't need it?

MR. LU: The phenomena would not show up --

CHAIR MARCH-LEUBA: So this --

MR. LU: -- in the design basis LOCA.

CHAIR MARCH-LEUBA: If I was writing the SER, I would say, you are not approved. These models are not approved for those three options. Now it's up to NuScale to run the risk of not having it approved. But an exception won't give them any more flexibility.

MR. LU: Yeah, definitely. That's the staff's SER as a limitation. Since we approve this one and if they ask for this one. So anything beyond that point is not covered. And then if they decided to go forward with the power operating or different design features or alternation. And then they can still use a large chunk of this topical report. But they have to reexamine those items there. You're right.

CHAIR MARCH-LEUBA: Just for the applicant, there will be a time to make comments from members of the public, if you want to expose more on this, the direction that you want, that you prefer to go. If there's something you're thinking about of why you're going this way, that would be a perfect time for that.

MR. LU: Okay. All right. So I just want to touch one point there. And from a PCT perspective or a CHF perspective, they are very conservative because we have collapsed water level. But from a regulatory perspective, even they have a tufis (phonetic) level on top of the core or should be sufficient cooling to prevent from boiling, especially at the decay heat level.

So therefore, they are shooting for a very conservative target and based on their actual design, the passive design features. So that part is really they earned that. It's not something they grant that. Okay.

But we're still in the process. I don't want to give the conclusion. We agree with every single bullet they ask for. But right now, that's part of the proposal and acceptance criteria.

MEMBER CORRADINI: So we don't have to take it up now, but I want to at least make sure we know when it's going to be taken up. So staff has done a series of audit calculations relative to 6.2.1, 6.2.2, and a lot of 15.

MR. LU: Right.

MEMBER CORRADINI: Are we going to take those up in the closed session?

MR. LU: Yes, the answer is yes. And now I'll give you a high-level summary.

MEMBER CORRADINI: That's fine. Whenever it fits into your thing.

MR. LU: It's a lot of --

MEMBER CORRADINI: But I just want to make sure --

MR. LU: -- proprietary and ECI data. So we cannot --

MEMBER CORRADINI: Okay, fine.

MR. LU: -- present it in a public fashion.

MEMBER CORRADINI: Okay. So we'll wait till closed session. That's fine. Thank you.

MR. LU: Okay. All right, next slide. Okay. So what's our area of focus? We have been reviewing this one for two and a half years. So we start from PIRT, Phenomena Identification and Ranking Table, and then based on understand their facility, understand their testing and the measurement and so that we start from there.

And then one of the important aspect of their justification of this design is a NIST facility. So it's a facility that has all the bells and whistles with scaling done pretty typical to try to capture as much as we can as part of a scaling approach to perform integral test facility which is very important basis, based on that to justify why their NRELAP5 code is qualified to predict the LOCA. Okay.

Of course every single test facility, once you scale down, you have scaling issues to deal with and then the distortions are expected. The question here is how we're going to deal with the distortion. So that's part of the evaluation model development we are focusing on.

We also focused on the code assessment against integral test data and separate effects data, for example, steam generator because they have their unique features of steam generators. So we have to go through that part. And then we have a very -- they have done quite a lot. Even before they submit everything, a lot testing has been done. So that's part of our review focus.

And another aspect of review focus is applicability of NRELAP5 LOCA analysis. Of course it's starting from initial conditions. For example, Jose mentioned about CHF stuff, and then that's part of his now LOCA stuff. However, how you're starting from, at what condition and then what's the operating -- from operating pressure, temperature you start of that LOCA analysis and even to progression there too.

And NRELAP5, it has its own numerics and condensation model. So we have a focus on that too because specifically when it's applied to containment, when it's used to capture containment pressurization, the numerics needs to be examined.

We understand how the -- at the break flow condition. In the past, most of the codes combined with other containment codes. And they you have numerics, figure out how much mass energy discharge and another code would take on. They combine this one.

It's a very good approach, but it does introduce the questions we have related to the numerics. We know that there may be -- there is a potential with issues. And then so we were examining that point, especially condensation. As I mentioned, as part 6.3, the condensation, the containment is one of unique features of ECCS. So we spent quite a lot of effort on that.

And of course, the evaluation input model and how they comply with Appendix K and then the -- so that is they can perform conservative analysis and whether they have Appendix K requirements and all the model is turned on and also the bounding input values there. So that's the focus of staff's review of this topical report.

Now I'm going to go through major issues in the past two and a half years. One of the issues we identified are not all of that. Actually if you go through the RAIs we issued, this time, we only issued a total 13 RAIs for a topical report, 13 total RAIs. But each areas has subquestions. So we total lumped it directly under LOCA topical. We only have about 40, 42 questions.

CHAIR MARCH-LEUBA: Any of these open items will qualify as the uncommon ones?

MR. LU: Specifically under LOCA topical, at this point, we do not have. We do have open items there. Okay? And yeah, I think it was the margins they have the core are up to that high.

MS. KARAS: This is Becky Karas. We are still characterizing some of them like the switch to NRELAP 1.4 which we just received the executable for last week, I think.

CHAIR MARCH-LEUBA: That's not an uncommon -- that's an open item.

MR. LU: It's an open item.

MS. KARAS: So --

MR. LU: But it's not uncommon.

MS. KARAS: -- I mean, we have the threshold sort of decided at the management --

CHAIR MARCH-LEUBA: My opinion --

MS. KARAS: -- for that.

CHAIR MARCH-LEUBA: -- that's an open item that has a clear path of resolution. We approve Version 4 or we don't.

MS. KARAS: So there's been a lot of discussion within NRO management as to what --

MR. LU: Okay, yeah. Well, I just wanted to touch on that since you mentioned about 1.4. I think 1.4, as Brian Wolf from NuScale mentioned about that one, it was made available internally to NuScale back to early last year. But we just received that recently, the executable for us, not the source code, the executable. So it was very tedious to get there and then it consumed a lot of time and energy and delay on the schedule for us to get there. And then if that showed up earlier, then we can have better answers right now.

But the good news is we obtained the executable. We can run additional cases to confirm their models and whatever, put it into the Version 1.4 when it's still okay, so --

CHAIR MARCH-LEUBA: So while we were

talking about that 1.4, would you expect the whole spectrum of Chapter 15 analysis to be rerun with 1.4? Or the ones that have plenty of margin, is it okay to keep the on the record calculation with 1.3?

MR. LU: At this point, it's a good question. And at this point, I think 1.4 is still in our advantage. We are going to make that determination once we --

CHAIR MARCH-LEUBA: You will determine --

MR. LU: Yeah.

CHAIR MARCH-LEUBA: -- it's okay. If you determine it's bad, that's a bad path.

MR. LU: Yeah.

CHAIR MARCH-LEUBA: And it's a possible one because of the review. But let's assume you find it's okay and you approve it in December.

MR. LU: Yeah.

CHAIR MARCH-LEUBA: Will you expect for them to rerun every single calculation?

MR. LU: Okay. That decision will be made at the management level based on our technical reviewer's review because we just obtained their executable. But we are going to tackle that one and provide the staff's opinion to give it to the -- yeah?

MS. KARAS: This is Becky Karas. We

actually have a slide on this coming up as we go through the variety of topics. But we already have a schedule from NuScale as to when they're going to rerun the various calculations, and so --

CHAIR MARCH-LEUBA: I just was talking about the process. I mean, if for some calculations that have so much margin that nothing will --

MS. KARAS: Yeah, so remember there are multiple changes going on, right? There's the change to 1.4. There's also changes to the model itself, so --

CHAIR MARCH-LEUBA: You know what you're doing.

MS. KARAS: Right.

MR. LU: Yeah, but I think actually NuScale provided the access to -- a long time before we received the executable. So we were able to dive into their code and revision documentation and just check the QA record. So we understand what they are doing, and we just wanted to make sure quantifiable results are obtained. And then we provided the recommendation to management to make a decision on that one. All right.

So let's cover the significant findings as part of PIRT. Okay. The review team identified one issue which was early, very early on. And I think within about two months we found this one. And in terms of the analysis and testing, they did not cover the lowest possible elevation.

So as a part of a spectrum of LOCA or the inadvertent opening of ECCS valves cases, we said, well, how come you have a passive design? But you're bound fundamentally. That's something you want to cover in a CVCS line break. And then their testing was only done to address CVCS line break. At least their test results provided to us was only that's the case. So we issued RAI. And this RAI triggered an action from their side. I really appreciate what they did. Okay.

What they did was they did a specific test to address this RAI as a part of HP-49 test and to capture the lowest elevation of the -- from a testing perspective. We got our management's full support. So we asked of this issue and justified why we need that. Because from a PIRT perspective, if you have a lower elevation of a break has not been covered, how come we know you captured the important phenomena or ranked it right or well?

Okay. So that's one of the issues. And then on top of that and we did not see their analysis.

So they decided to include as part of Appendix B of this topical report to include an analysis of the inadvertent opening cases of both RVV and RRV. But RRV covered this one. And they performed additional assessment against HP-49.

And then so we are -- staff is still in the process to review this one. But it looks good at this point. We are not approving yet and we are not writing on that perspective. But we review that one. That's a significant finding from that perspective. And --

CHAIR MARCH-LEUBA: Chapter 15 has an AOO --

MR. LU: Yes.

CHAIR MARCH-LEUBA: -- opening of an RRV.

MR. LU: Yes, so from --

CHAIR MARCH-LEUBA: So it's not just part of this report. It's part of LOCA also, Chapter 15.

MR. LU: You are right. So they did actually characterize that as the AOO. So that just give them the credit. But are they testing? And since they are applying this NRELAP5, a lot of referencing on documents came from that topical report and the supporting documents there. So really the key there was the additional testing.

MEMBER CORRADINI: So let me ask a technical question at this point. So under the bullet for NuScale response, have they submitted their NRELAP analysis of HP-49?

MR. LU: Yes.

MEMBER CORRADINI: And you guys are looking at it? Are you doing audit calculations to compare to it?

MR. LU: We did.

MEMBER CORRADINI: And will we hear about that? Or is that ongoing that we're going to hear about it later?

MR. LU: I would say that in October you are going to hear all details. But at this point, we don't see a problem.

MEMBER CORRADINI: Okay. The reason I ask that is if I understand the -- I'm not sure which of the open items this all comes into. But that's not -- the numbering doesn't matter. From a technical standpoint, the concern by the staff was stratification. But in both calculations I'm assuming by staff and NuScale, they're assuming a one-dimensional TRACE NRELAP are essentially one dimensional unless you purposely try to nodalize it to look at it to be two dimensional. In other words, you have to have a nodalization that you allow for essentially an actual circulation pathway because I've got a cold containment with a cold cool which causes a natural circulation path that neither of them would capture with a one-dimensional nodalization.

I didn't see that in NuScale nor did I sense that it was done by the staff. So when eventually this comes back, I'd like to see a nodalization that can capture an a natural circulation pathway because I've got a cold wall which is going to cause a flow down and essentially a refreshing of the surface. So the concern about having stratification that I have it too warm or too cold may not be there because I might have a well-mixed situation by essentially a cold wall circulation pattern.

Because I've got a hot surface, I've got a cold surface, which means I should have a Rayleigh number that causes natural circulation. I didn't see that in either -- I didn't see in a NuScale calculation and I didn't sense it was there in the audit calculation. So that, to me, is a technical issue that might be a conservatism that's been unidentified.

MR. LU: Okay. I'll give you the high level. We did our own nodalization study with TRACE.

And then the TRACE, one thing I -- well, actually, I have one slide if you want.

MEMBER CORRADINI: Okay. Well, then we can wait.

MR. LU: I'll cover that one --

MEMBER CORRADINI: Okay, fine.

MR. LU: -- for you.

DR. SCHULTZ: Shanlai, before we go there, the last bullet, the benchmark case nodalization study, whose study is that? Is that a NuScale study or is that on both --

MR. LU: The --

DR. SCHULTZ: -- sides, your work as well as NuScale?

MR. LU: Okay. Their study and was -- at the time we are preparing the slide and they were still doing this and the study. And now they submitted the results already. And we did our own calculation a long time ago. That's the reason we ask that question. And actually, Mike, to you, you have a very nice description. It was a concern, yes. But that's pretty much where we came from too, so --

MEMBER CORRADINI: But the reason I bring it up as I did is at least the way the RAI was written, it implied that there was potentially a conservatism.

MR. LU: Yeah.

MEMBER CORRADINI: I guess my view of it is it might be the exact opposite. Since I have a cold wall, hot wall situation --

MR. LU: Right.

MEMBER CORRADINI: -- there's a traditional natural circulation problem that I'd want to compute what the characteristic dimensional group is to see whether I actually have a circulation pattern. And I don't think either calculation can do that with a one-dimensional formulation.

MR. LU: I'm going to cover that one.

MEMBER CORRADINI: Okay.

MR. LU: So I don't want to jump to the conclusion and say yes or now.

MEMBER CORRADINI: That's fine.

MR. LU: And so I'll share you with the staff's perspective on that point. I have one slide specifically on that point.

MEMBER CORRADINI: Okay.

MR. LU: And we can use that slide for that one purpose. Okay. So that's the first one. You probably heard from NuScale already, 1.4. They are changing the condensation correlation and to modify the code. And obviously, since because of the potential -- the challenge on a couple years ago about the peak containment pressure. And also they rely on the condensation heat transfer on the wall and the containment as part of ECCS because therefore we focused on that a lot.

And the actual implementation of the correlation uses different hydraulic diameter from our correlation. And then their actual -- the current -- even the current correlation, if you compare the actual conditions of the pressure, temperature void fraction, it's outside the bound of the previous supporting testing database for that wall condensation.

CHAIR MARCH-LEUBA: Outside the bound, assuming meaning conservative?

MR. LU: That's correct. That's --

CHAIR MARCH-LEUBA: What we were told is there was an error in the FORTRAN --

MR. LU: Well --

CHAIR MARCH-LEUBA: -- of the implementation. There was a multiplier in the wrong place.

MR. LU: Yeah.

CHAIR MARCH-LEUBA: So it was incorrect.

But it was fortunate that was conservative?

MR. LU: Yeah.

CHAIR MARCH-LEUBA: Okay, good.

MR. LU: That's a whole line from where we come from. And then it's to cover that wide pressure range from right beginning almost vacuum to the point that you reach the peak containment pressure. It's 986 pounds and then the predicted 82. And so that range -- actually, there's no condensation correlation which has the base to cover that range. And they did that wrong.

And as part of HP-49 and the previous NIST test to show based on those test data -- integral test data to show the correlation we are using. Although it's a simple one and it's a conservative laminar flow condensation correlation and where it allowed them to pick a conservative peak containment pressure. And from low with respect with the core, a class 4 level and then the PCT prospective rates, it's not sensitive to that much based on the analysis and testing.

So our current position on that one and we used to have one open item with this one and we consider it's closed and they have conservative correlation and benchmark it against our test data. So this particular issue has been going on and then resolved. So that's next slide.

DR. SCHULTZ: Shanlai, has the degree of conservatism been identified? I didn't see that in NuScale's presentation yesterday --

MR. LU: It's --

DR. SCHULTZ: -- as something that they --

MR. LU: Okay. Yeah, the --

DR. SCHULTZ: -- have identified as -- it showed some deltas --

MR. LU: Yeah, actually that was part --

DR. SCHULTZ: -- but not associated with this.

MR. LU: -- that will be presented as part of 6.2.2 for the peak containment pressure.

DR. SCHULTZ: Okay, great.

MR. LU: And the short answer is yes, they did that one. And it may not be in your submittal for you so you can see. But we definitely cover that one as part of an audit and then went through all the documents they had.

DR. SCHULTZ: Look forward to that. Thank you.

MR. LU: Next slide, CFH correlation. So as Jose, you mentioned, right at the beginning that there is a -- okay, Antonio, that's your slide.

MR. BARRETT: My name is Antonio Barrett with the NRC staff. So the NRC staff reviewed the CHF correlation information provided in the LOCA topical report for the significant findings. The staff noted that for an inadvertent opening of an RRV event, there's a momentary core stagnation and reversal that's expected.

So the NuScale applies a combination of a high flow and a low flow CHF correlation to cover the expected range of the flows encountered during the LOCA or a LOCA-like event. LOCA-like event being like an inadvertent opening of an RPV valve.

So the staff had questions regarding the CHFR limits with respect to the applicability range as well as the method. And this is not just for LOCA-like events or the inadvertent opening of an RRV valve but also for all of LOCA events and inadvertent opening of RPV valves.

So the staff issued an RAI in which NuScale responded to which ended up being the Appendix to the LOCA methodology that I think you guys are aware of. And in addition, the staff issued an additional RAI to ask for more information about the database of different test data that they had in order to confirm that there's no conservative or nonconservative portions in their CHF correlations.

In addition, the staff had some questions about the application. On the low flow CHF, there's a modifier that was being applied. So staff had some additional questions of whether or not that modifier introduced some nonconservatisms. And so this entire review is still ongoing. So we're still reviewing that information.

And the staff also had questions about whether or not the interpolation scheme between the high flow multiplier correlation versus the low flow correlation, if that was implemented correctly and if it introduced any sort of nonconservative aspects.

MEMBER CORRADINI: But to get -- maybe you're getting to it and I didn't -- but I think Jose's question about the difference in terms of starting point doesn't answer -- is not answered by this.

MR. BARRETT: That's correct.

MEMBER CORRADINI: You're going to get to that eventually?

MR. BARRETT: I wasn't planning on addressing that.

MEMBER CORRADINI: But when you're done, can you at least give us a hint?

MR. BARRETT: Yeah, sorry.

MR. SCHMIDT: Yeah, this is Jeff Schmidt from the staff. So Jose's observation is correct. There's, like, two groupings, on VIPRE-based and one --

MEMBER CORRADINI: Yeah, that's what I thought.

MR. SCHMIDT: -- NRELAP-based. I wouldn't expect those necessarily to be different. I mean, they're different fidelity models, different assumptions. Like, say, VIPRE has a cross assembly flows. It could be bypass flows. It could be peaking factors, assumptions. So there are two classes. Like, if you say within the VIPRE regime where they use VIPRE, some of that biasing is due to the, like, initial conditions that you want to try to force a conservative answer out of.

Now we haven't -- as of last night, haven't spent to figure out what exactly the details are, like, within each group. But I think from a -- if you were to look at it from, like, a hot full power standpoint and wanted to know, like, what you think the CHFR is, I think you should use the VIPRE-like number.

CHAIR MARCH-LEUBA: I mean, I would expect when Antonio the review of the CHF correlation that the data points kind of follow the line of the correlation. You may have chosen to put one that bounds all of them or you can choose one that goes through all of them.

MR. SCHMIDT: So that is true.

CHAIR MARCH-LEUBA: That is? Okay. Now when you do the VIPRE analysis, you're having immensely super conservative bunch of assumptions. To the point of normal operation when you do the analysis, VIPRE assumes, like, a 30 percent void fraction when really there is no void fraction in there. So NRELAP is not calculating a void fraction. I cannot see and I would like to be doing the October presentation.

How come a NRELAP correlation -- NRELAP analysis using a correlation that kind of follows the data points is more conservative than a VIPRE super conservative model? I would expect them to be reverse. VIPRE would be 1.6 and NRELAP to be 2.8, not the other way around.

MR. SCHMIDT: Okay. All right. I think we'll --

CHAIR MARCH-LEUBA: That doesn't make sense to me.

MR. SCHMIDT: I mean, in detail, we'll have to address the differences in the summary report.

CHAIR MARCH-LEUBA: So my suspicion is that the NRELAP, they have a multiplier or something that makes it super conservative.

MEMBER CORRADINI: Well, but -- I guess they owe us answer. But the original Hench-Levy was a limit line approach which was it was designed to be everything under the data. And as I understood the applicant's response, they modified that. But I still have the funny feeling that since the Hench-Levy approach is a limit line approach, you're going to tend to underpredict or get a lower CHFR. Then I would have essentially a tuned correlation that VIPRE is using. But I think -- I'm still not exactly sure why.

CHAIR MARCH-LEUBA: Yeah, so I am going to shut up in this topic. You can follow as much as you want.

(Laughter.)

CHAIR MARCH-LEUBA: I will expect to follow it up in October because something does not make sense. Do you understand what I said?

MR. BARRETT: I understand what you're saying. So it's still under review. We've gotten some other materials from NuScale. We're looking at those. We didn't -- we hadn't yet looked at it with respect to VIPRE. But we can follow up later with those details.

So as I mentioned before, NuScale did provide audit material for us to look at. The review is still ongoing. NuScale has agreed to develop some additional LOCA TR markups. That should hopefully explain a lot more about what's going on with the correlation and some of the conservatisms that are there in the correlations. And so the staff is still waiting for those markups as well as going through the review for the audit material.

CHAIR MARCH-LEUBA: So a moment ago, I lied. I'm going to make one more remark. I didn't mean to say there's something wrong with it. I mean there's something that merits looking into it.

MR. BARRETT: Understood.

MR. LU: Okay. All right. So that's the reason that CHF is on the side, and then since we still have ongoing review activity plus 1.4, so we hope we can make it in October to get back to you. It depends on the applicant side, the response time, and then figure out how we can get there. All right. That CHF is definitely one of the items there. All right. We have a significant effort from staff side to perform a confirmatory analysis using both NRELAP5 and TRACE.

MEMBER CORRADINI: So I'll ask my question again. The inner box is the reactor vessel?

MR. LU: Yeah.

MEMBER CORRADINI: The fleshy stuff is the core? The okra sort of ugly stuff is the chimney?

MR. LU: Right.

MEMBER CORRADINI: The second box is containment. The third box I assume is part of the pool?

MR. LU: Yeah.

MEMBER CORRADINI: The problem that I see in all these nodalizations is it's not one dimensional. You have to with all of these tools, they're all pipe and junction models which means you've got to have at least two control volumes in any axial location to allow for a natural circulation path. Otherwise, you kill natural circulation and therefore you haven't modeled what apparently is going to be the physics involved. And well, I'll stop.

MR. LU: Okay. That's the exact reason we ask in the RAI.

MEMBER CORRADINI: Okay.

MR. LU: When we saw the first time, we said, oops, where's the natural circulation going? That's the exact reason we ask for it.

MEMBER CORRADINI: Okay, fine, fine.

MR. LU: Okay. All right. So I have to answer your question first. So I have still have comments related to it. But as -- it's correct that we dive into the pool. And we have a bay for each containment module. The NPM module has a bay. Closely surrounding at least three-quarter of the parameter of that module. So we dive into that one. And really from our perspective, we have a conservative approach. Conservative approach means that from in terms of peak containment pressure or collapse lower levels.

So we did actually a sensitivity study. We put a bounding scenario there and find out whether they are okay. And they actually did a similar thing too, demonstrate to us. So based on both confirmatory analysis and their analysis, the model itself, you're correct. It does not have the capability to model the natural circulation because simply it's a 1D type of a model, especially for the core, but --

MEMBER CORRADINI: So now I'm going to tease with you a bit. The one tool that I would've used would've been MELCOR because MELCOR has a natural circulation default for essentially Rayleigh calculation where I have a hot heat structure and a cold heat structure. It defaults to essentially a Nusselt number as a function of Grashof-Prandtl which means you don't have to have this sort of nodalization. You can stay with a major control volume. So that would be another way to check it.

MR. LU: Thank you very much. That's what the staff is going to present to you.

MEMBER CORRADINI: Okay.

MR. LU: In addition to TRACE analysis and NRELAP5 analysis and the containment branch contracted because the peak containment pressure issue is very sensitive and that one of the figure of merit. So we have the MELCOR.

MEMBER CORRADINI: Okay.

MR. LU: They did a MELCOR analysis too. But from NRELAP5 and TRACE perspective, we found the 1D model is not mechanistic but conservative.

MEMBER CORRADINI: Okay.

MR. LU: That's the answer your question. All right. So let's go back to the high-level summary of this confirmatory analysis. And we did have a lot of our research folks in support of this one. We have

Peter Lien develop the base model. And then we have Andrew perform a lot of the long-term cooling analysis and LOCA analysis. We are lucky to have also ACRS staff performing confirmatory analysis as part of a research team, and wondering whether you can just send him to make sure that the staff is doing confirmatory --

CHAIR MARCH-LEUBA: There might be a conflict of interest there.

MR. LU: Yeah. Okay. All right. Anyway, I just want to cover that one. And we -- Carl Thurston did a lot of NRELAP5 runs. And the reason we could reduce our total number of RAIs -- official RAIs on the docket, under the LOCA topical, there's only 13. And then we come to all these subquestions, about 45. And we did a lot of analysis that's following the office and the projects, the guideline to reduce the burden to applicant as much as possible. That's what we did.

So then we zoom in our issues, we even have done our homework ahead of time, performed a lot of analysis, figured out, is this issue really is going to affect the figure of merit? That's the reason that this topical report, we reduce the number of RAIs to almost, like, 20 percent of the previous, the LOCA topical report review.

And so that's the confirmatory analysis. We are going to get into details about the class as part of confirmatory and also the proprietary session and also in the public session, whatever can be shown. It's not proprietary data, but it'll show some curves there too.

CHAIR MARCH-LEUBA: Let me ask a process or high-level question. Did you use a risk informed approach to reduce this number of RAIs? Because let me tell you what I'm thinking. I'm thinking for this LOCA, as long as the RRV opens, all I care is whatever is left of containment is capable of condensing this thing. I don't care what the temperature of the water is, as long as thing just condense and the pressure doesn't go over 1,050. The thing is going to work. I don't need TRACE to calculate this. I need an NRO.

So this is a perfect application. I hurts me to say that. I joke that I'm a member of the PRA branch on this table. But this is a perfect application of risk informed where the consequences of being wrong are very low. You can do it backward in the bounding calculation. Do I have enough containment to condense the steam? I'm fine.

MR. LU: Okay. Let me address that one

from two aspects. One aspect, there is right now the methodology and the rule that the regulation we have specific to LOCA topical, it's deterministic. So that part is -- so we have to go through all the spectrum cases and understand the figure of merit.

CHAIR MARCH-LEUBA: No, sorry. My --

MR. LU: However --

CHAIR MARCH-LEUBA: -- point is --

MR. LU: I understand.

CHAIR MARCH-LEUBA: -- there is no regulation that you need to issue 150 RAIs. You can issue only 20.

MR. LU: Right.

CHAIR MARCH-LEUBA: And the reason you issue only 20 is because you don't need to know the message.

MR. LU: You're correct. You're correct. And actually, the second aspect, since the beginning of NuScale review and I think someone else would be the best person to answer that question. I think both my branch chief, the lead reviewer and then the project and office, all the way through. So they not only just reduce the burden of the applicant but also performing internal confirmatory analysis and then determining that they're really significant from the research and to reduce that one.

From that, the answer is yes, we have been following that practice even in the deterministic world of analysis and review. We have done that.

CHAIR MARCH-LEUBA: You may have done it unconsciously. It's hard to take credit for it.

MR. LU: We are given the marching order to do that actually.

MS. KARAS: Let me just clarify real quick. So the review approach that we're following is really one that's evolved over time. So we use a variety of tools to make the review as efficient -- as effective as possible, right? So we use audits. We use stops, internal confirmatory or sensitivity analyses. There's not any sort of -- I don't want there to be any confusion about number of RAIs or targets for that.

We typically use audits so that we can focus the review in the most important areas, right? We still have the regulations. We still have a deterministic look at things. But we do use our engineering judgment and our engineering tools to determine what are the most important topics that we need to look at and we need to consider.

CHAIR MARCH-LEUBA: My point is you can do a complete PSD thesis on the recirculation patterns on the lower part of the vessel. And I would love to do it. It would be real fun. But if you can make a calculation that says, let's assume the temperature is a set of initial conditions. Does it still work? Okay. It cannot be worse than this. I don't need to know how cold it is. And that's risk informed and that's a valid application of risk information.

See, I have a bounding calculation. I don't need to know more detail. And Dr. Ballinger was educating me on that this morning on how safe is safe enough. And this is valid information. I think we should take credit for it.

MR. LU: I think staff agree. And actually factor that one into the practice of this review and also across the board all the reactor reviews there. So if there is sufficient bounding analysis, we do not have to dive into the details. We are fine. Okay.

That's the end of my summary for LOCA topical. And anything else that's -- if you have -- feel free to give us questions and then we'll be prepared to come back to give you a detailed briefing.

DR. SCHULTZ: Just to comment, Shanlai. Just to follow up on the discussion we just had in terms of what you presented and the rationale that was used, technically, regulatory, all of the above. What that comes down to is to make sure that what you've described as your review process is very well documented in the safety evaluation. And so that the fact that you haven't -- how you reached your conclusion is very well documented in terms of your analysis and decision making, how you followed it, the audit process. Those are all good practices and they need to be documented well so that --

MR. LU: Right, yeah. Actually --

DR. SCHULTZ: -- your process is well appreciated by the public.

MR. LU: Yeah, that's a part of the process as we're preparing SER and so we got project support to finish all the -- wrap up all the audit reports and then put it on the docket too. And so that's the -- thank you for the comments. If there are no other questions and answers, so the next topic.

MS. SIWY: Okay. This is Alex Siwy again. I'm going to be covering the staff's overview of the non-LOCA topical report review.

And so just as a bit of background, the scope of the non-LOCA methodology using NRELAP5 includes non-LOCA design basis events except for static events such control rod misalignment or an improper -- a fuel assembly in an improper position, control rod ejection and inadvertent operation ECCS because they're all covered by separate methodologies.

The time frame that's covered for non-LOCA events is while the mixture level is still above the top of the riser, so that natural circulation is maintained. Anything beyond that would be covered by the long-term cooling analysis.

The non-LOCA methodology scope also includes how it interfaces with other methodologies such as receiving input from fuel performance and nuclear design codes and providing boundary conditions for the downstream subchannel or radiological analyses. And the non-LOCA methodology scope also includes certain event-specific assumptions as well as conservative bias directions for the initial conditions.

As far as the status of the non-LOCA review, as we've already discussed the review is still ongoing. Most of the technical issues are resolved at this point. But we are still continuing audits and public teleconferences for a few of the remaining technical issues. And as you know, the ACRS subcommittee meeting on the topical report is scheduled for this fall. Just to give you --

MEMBER CORRADINI: So just to -- can you go back?

MS. SIWY: Sure.

MEMBER CORRADINI: I want to make sure I understand. So it's not that you're using the different tool. It's the initial conditions and the protocol in terms of what is assumed is different between the two because you're using NRELAP again and all the same -- the same release version, et cetera. Am I correct in understanding that?

MS. SIWY: That's correct. I mean, there are certain areas that are a little bit different for the non-LOCA models --

MEMBER CORRADINI: Right. No, I understand that.

MS. SIWY: -- compared to LOCA.

MEMBER CORRADINI: The biggest one is, if I understand the applicant's explanation yesterday, is that because we're within a regime of normal operation -- more normal operation, they essentially couple NRELAP to VIPRE and we would have VIPRE calculations for a core response.

MS. SIWY: That's correct.

MEMBER CORRADINI: Am I correct?

MS. SIWY: Yes.

MEMBER CORRADINI: Okay, fine.

MS. SIWY: Okay. I'll kind of give you an idea of the areas that we focused on in this review. So we looked at conformance to existing guidance such as several SRP sections in Reg Guide 1.203 as well as NuScale design specific review standards. One of the biggest areas that we looked at is applicability of NRELAP5 to analyzing non-LOCA events, especially in terms of how the non-LOCA highly ranked phenomena are addressed.

So as NuScale mentioned yesterday, there's a lot of overlap with the LOCA EM. Non-LOCA has basically evolved from that a little bit. And so they used a graded approach to the evaluation model development and assessment process. And to address some of the differences between the LOCA and non-LOCA highly ranked events, they performed separate effects tests.

So we actually looked a lot at the SIET and KEIST tests that were mentioned as part of the LOCA topical report because of their importance to helical coil steam generator modeling and the high pressure condensation phenomena in the DHRS. And the NIST HP-03 and HP-04 tests are looking to validate the full length DHRS.

In addition, there were several integral effects tests that we looked at. NIST NLT-02a and 02b were originally in the submittal of the topical report and they look at the integral system response to a loss of feedwater events and the subsequent DHRS actuation. And then in response to a staff RAI, NuScale provided an additional test assessment against the NIST NLT-15p2 test which covers the loss of feedwater right into DHRS actuation.

NuScale also performed a code to code benchmark against RETRAN-3D for reactivity initiated events to kind of validate the neutronics models within NRELAP5. And other things that NuScale did to address highly ranked phenomena is using bounding input values for certain parameters. And simply some of the highly ranks phenomena are addressed by different methodologies including a subchannel.

CHAIR MARCH-LEUBA: I know we're going to have some discussion about the moderator temperature coefficient. I've seen -- I've looked at the slides ahead of time. But one concern I have with the non-LOCA -- actually, in the LOCA too, but non-LOCA always come first -- methodology is the use of a constant MTC independent on the conditions, temperature, voids, boron, other reactor. That really is a bad approach.

I mean, and it only works if you have a scram within the first couple of seconds and it doesn't make any difference. But for any other events when you maintain power, you seen a constant MTC. When you know it can change by a factor of five during the transient, it's not a very good approximation. Did you guys do some analysis to see if it was bounding or acceptable?

MS. SIWY: So the staff did run some confirmatory analyses to look at MTC at different power levels. But I didn't perform those, so I wouldn't be the best person to speak to about them.

CHAIR MARCH-LEUBA: See, for example, I mean, my background is in boiling water reactors where we use a density coefficient which is the same thing. But the difference between the reactivity feedback at zero percent voids and 80 percent voids is 500 percent, a factor of five. So you're using -- individually, you were never there. You use a constant reactivity coefficient.

MR. SCHMIDT: So this is Jeff Schmidt from the staff. So we did some PARCS calculations where we looked at MTC as a function of temperature and at different pressures, like, a family of curves. And we really didn't see much difference in the MTC versus temperature for a range of pressures.

CHAIR MARCH-LEUBA: Not pressure but temperature you see it --

(Simultaneous speaking.)

MR. SCHMIDT: Well, it was MTC versus temperature for a series of pressures.

CHAIR MARCH-LEUBA: Oh, the pressure has nothing to do with it other than a small change in density.

MEMBER CORRADINI: But I want to make sure what Jeff said. I think what he said is they did. What they were worried about then that the variation is small.

MR. SCHMIDT: That's right.

CHAIR MARCH-LEUBA: No, I'm not -- I'm worried about changes in temperature during the transient, not changes in pressure.

MR. SCHMIDT: Well, it's just a function of temperature as well. So there is a change of moderator --

CHAIR MARCH-LEUBA: Density?

MR. SCHMIDT: Well, there is moderator temperature change too also. There's a slope to that curve --

CHAIR MARCH-LEUBA: Oh, you're talking about --

MR. SCHMIDT: -- as a function of temperature.

CHAIR MARCH-LEUBA: Yeah, we said it before that a moderator has no Doppler.

MR. SCHMIDT: A moderator has no Doppler?

CHAIR MARCH-LEUBA: If the hydrogen is a single proton, the reason you have Doppler is because you have broadening of references.

MR. SCHMIDT: Are you talking about the Doppler on the fuel or the --

CHAIR MARCH-LEUBA: Doppler on the --

MR. SCHMIDT: So you mean the --

CHAIR MARCH-LEUBA: Doppler on the moderator.

MR. SCHMIDT: Moderator? Oh, the temperature change of the moderator?

CHAIR MARCH-LEUBA: The temperature change of the moderator should not give you any broadening of references and that's the only effect you will have. So that has to be minimal.

MEMBER CORRADINI: But I just want to make sure I understand his response. I think what he said is there's a table of temperatures and then various pressures and temperatures as long as I'm subcooled. And there's MTC values and all of those had nominal changes. That's what I thought you said.

MR. SCHMIDT: So they have nominal changes relative to pressure, but they do change versus of temperature.

MEMBER CORRADINI: But that is considered?

CHAIR MARCH-LEUBA: No, it's not.

MR. SCHMIDT: So in their analysis, they fix, like, the moderator temperature coefficient at a value, right? But there should be lookup tables in RETRAN that would then you can either hold it constant or let it flow according to the tables, right?

CHAIR MARCH-LEUBA: Do the methodology allow you to float versus temperature? The answer is no.

MR. SCHMIDT: If they used a constant, the answer is no.

CHAIR MARCH-LEUBA: The answer is no, I know.

MEMBER CORRADINI: But you're concerned that's a nonconservative result?

CHAIR MARCH-LEUBA: It could be either way. And as our friend, Dennis, says, conservative for what? Conservative in one direction or conservative in another. And the MTC, you will see the figure. I think it's in the closed session. It changes from positive to negative. The MTC goes positive at lower temperatures and negative -- I mean, it's not a little bit of change. It's positive to negative.

MS. SIWY: So what I will say is that we looked at whether the nature of the coefficients used would be bounding in terms of the type of event that it is. So for an overcooling, to maximize an overcooling event using the most negative MTC would be acceptable to us. And for an overheating event, we want to reduce the negative feedback.

CHAIR MARCH-LEUBA: Most likely, I know. In Chapter 15, they use -- and this comes from the public version of the report so you can see the numbers. They use 46 pcm per Fahrenheit or zero, depending on -- so they try to bind it -- bound it. So it's probably okay. But for events where the scram happens late, and I know my favorite design was, I would have rather seen a table of MTC as a function of the conditions that I have now, unchangeable during the transient. But it's in your evaluation that you see zero or 46 acceptable.

MS. SIWY: Yeah, I mean, that's -- at

least in what I've reviewed, I'm accepting that as --

CHAIR MARCH-LEUBA: Yeah, I explained --

MS. SIWY: -- valid.

CHAIR MARCH-LEUBA: -- claim on the record that we're in the 21st century. I mean, using zero or 46 was good in 1960. We should have had a table of cross section that you look up given your boron, your concentration, your voids, all the parameters of the calculation and calculate a density as a function of time instead of just going with this bounding analysis that don't give you an insight of what reality is.

MR. SCHMIDT: This is Jeff Schmidt from the staff again. So it can be in there. The user can choose to take that option or not, right? There is a table of reactivity feedback terms that are in there. So it's available.

CHAIR MARCH-LEUBA: It may be available. But on all the transients I have reviewed in Chapter 15, it's constant.

MR. BRISTOL: This is Ben Bristol from NuScale. Just a quick clarification. The methodology does commit to using a conservative moderator feedback. It doesn't necessarily specify a constant to be used. There is one event where we -- or a couple events where we don't use a constant MTC.

Generally, the non-LOCA events, though, in the FSAR do use a bounding temperature feedback based on whether it's a heat up event or a cool down event.

CHAIR MARCH-LEUBA: And what are the transients where you change, you have time dependent MTC?

MR. BRISTOL: So the IORV event, for instance, uses a curve. In addition, the single rod withdrawal event actually uses a variant because it's got some temperature dependence.

CHAIR MARCH-LEUBA: Okay. I didn't catch that on the -- that's good. I mean, that's the way to do it. I would have done it for all of them, but --

MR. BRISTOL: Sure.

CHAIR MARCH-LEUBA: Okay.

MS. SIWY: Okay. Moving on, the staff also looked the NRELAP5 facility models including nodalization and similarity of the test facility to the actual NPM. And staff also looked at the inputs and modeling assumptions used for the non-LOCA analyses, and this includes the use of what the SRP suitably conservative input rather than performing a full blow uncertainty evaluation, the adequacy of sensitivity studies to determine what the conservative bias directions will be for the various initial conditions, the consideration of single failures and loss of power scenarios, and treatment of plant control systems.

So this is just a really high-level summary of what some of the more challenging review areas were. So the first three I will discuss a little bit more in the closed session because it borders on proprietary information. But just to go over them real quickly, adequacy of steam generator heat transfer uncertainty based on some of the things that we are seeing from the SIET assessments. Highly ranked phenomena and transient multidimensional effects within both the primary loop and within the reactor pool. Nodalization and scaling aspects for DHRS.

And the last one is assessments against NIST test data which is -- I mean, I characterize it as challenging because there were a lot of different assessments to look at, lots of information there. And we have a lot of back and forth with the applicants on why we were seeing some of the behavior that NRELAP5 was predicting. And in addition, the applicants updated most of the assessments to use the updated version of NRELAP5 as well as to make some corrections in the models.

MEMBER CORRADINI: Did that change the result?

MS. SIWY: Not significantly, no.

(Simultaneous speaking.)

MEMBER CORRADINI: But based on what they -- oh, I'm sorry. Based on what they said about the five changes they made except for the condensation coefficient correction, I wouldn't have expected it. So would the staff agree that those were not supposed to be prototypical? Those were essentially benchmarks that were not expected to be prototypical. Because when I asked the applicant yesterday, that's the response I got is they're different but they're not prototypical. We're just trying to test out the model. So have you resolved that, or are you still reviewing the NIST data -- the NIST and NRELAP analysis of the NIST data?

MS. SIWY: We're still reviewing the markups that the applicant provided because there is a lot of information. But we haven't seen anything that we're very concerned about.

MEMBER CORRADINI: Have you guys done audit -- or your own calculations of this? Or you're simply auditing their calculations?

MS. SIWY: We're primarily auditing their calculations.

MEMBER CORRADINI: Okay, fine. Okay. Thank you.

CHAIR MARCH-LEUBA: Since you mentioned uncertainty on the steam generator heat transfer coefficient, were you here yesterday morning during the study of the topical?

MS. SIWY: I wasn't here for that.

CHAIR MARCH-LEUBA: Okay. I'll stay nonproprietary. With the design that is currently on the table that we are approving, it's likely that we will have some noise on the heat transfer coefficient because of some flow oscillations on the second day. Will that uncertainty that you already applied cover those, or do we need to add an additional just in case conservatism? And since you were not here yesterday, you're not the right person to ask. I'll just leave it open and you can think about it for the final SER.

MS. SIWY: Okay. And when I discuss it a little bit more in the closed session, it may give a little bit more clarity --

CHAIR MARCH-LEUBA: Okay.

MS. SIWY: -- as to where we need to go.

MEMBER CORRADINI: Let me ask a question about number three which we're going to go to closed session.

MS. SIWY: Sure.

MEMBER CORRADINI: So I'm kind of curious about what applicant said was, if I fill the -- well, I want to understand the limits of the operation of DHRS. Has staff looked at that?

MS. SIWY: Yes. Staff has looked at that. That particular staff member is not here today, but there was --

MEMBER CORRADINI: Well, we can talk about it in closed session?

MS. SIWY: Yeah.

MEMBER CORRADINI: Okay, fine. I will wait then. Thank you.

CHAIR MARCH-LEUBA: On that, let's just provide some questions ahead of time. One concern I have with the DHRS is overfilling of the ECCS liquid. And because NuScale is passive, it does not use any operator actions in their analysis. But passive means you should assume the worst possible operator actions that he can possibly do within reason, right?

So when you're doing your review, are you looking at what if I overcool and DHRS doesn't come on and the operator turns it on ten minutes later? He says, oh, something happened. Let's turn this. And by that time, you're overcooling and you are overfilled. The answer we got from the applicant is you cannot overfill the secondary side of the steam generator. Because the primary is so hot, there is no way you can prevent voids. But have you looked at that? When you were looking at those transients, have you looked at --

(Simultaneous speaking.)

MS. SIWY: Yes, so there's a slide later on about the increasing feedwater flow event with the single failure assumption that contributes to the potential steam generator overfill.

CHAIR MARCH-LEUBA: Okay. Thank you. Because particularly that one, the steam generator fills to 85 to 90 percent level. It really fills. And assuming everything works and the scram happens on time.

MS. SIWY: Understood. Okay. Now I'll turn it over to Chris for rod ejection.

MR. VAN WERT: Okay. Thank you very much. So similar to other areas, the rod ejection analysis presented in Chapter 15 relies on a methodology that is in the topical report. It's currently under staff review. We will be coming to you, I believe it's in November, for the actual presentation of that topical report. But we wanted to use a few slides here to just give you a high-level overview of what the staff's review has entailed or what it entails.

So first off, just the acceptance criteria chosen by NuScale closely mirrors the acceptance criteria that you find in 4.2 -- SRP 4.2 Appendix B. A few modifications were included that are more conservative. I guess we can go to the next slide.

But more importantly, the overall framework of the analysis methodology is presented here and it's copied from the application or the topical report. They rely heavily on the SIMULATE-5 and the S3K code package. We have the layout here. I will point out that I just noticed a typo. The upper right box should be peak RCS pressure instead of temperature.

But generally, they use SIMULATE-5 perform the steady state initialization and then S3K for the dynamic core response. That feeds into three other sections here, the dynamic system response performed by RELAP which then goes on to give you your RCS pressure analysis but also provides input to VIPRE to give you your MCHFR analysis.

The one part here that's maybe a little bit new is the bottom path there where we have the adiabatic heatup fuel response calculation. This is presented in a little bit more detail obviously within the topical report. And since that was new, the staff focused on that a little bit more heavily. Next one.

So just to give you a high-level overview of what you'll see in the presentation we give later in the fall. We focused a little bit more on validation of the code packages. So the only validation that's contained in this topical report was for SIMULATE-3K. The other codes were verified and validated elsewhere.

We also looked heavily at the input assumptions and uncertainties that were used in this analysis and the transmittal of inputs between the codes. And then as I mentioned previously, the adiabatic fuel response hand calculation itself received some attention.

As far as the status, we're currently completing the SER. It should be done fairly soon going through a peer and branch chief review. And then I put fall 2019, but I think I heard November 20th maybe is the date that we have currently on schedule for this.

CHAIR MARCH-LEUBA: Alex, you're in charge. We're hitting the time where we're scheduled to break. Do you think we should go and do the source term? We've overrun and we probably should finish it. But your call.

MS. SIWY: Yeah, I don't think it'll take too long.

MR. VAN WERT: Okay. Thank you.

MS. HART: So good morning. I'm Michelle Hart from the Radiation Protection and Accident Consequences Branch. And I'll talk to you about the accident source term topical report. It is currently under review. We just got Revision 3 last month of the topical report and some related changes to the FSAR. And so the topical report that's described does analyses that we would evaluate in Chapter 15. There's a listing of them there. Next slide, please.

Some of the changes that we've already completed the review of is they have described earlier yesterday that they have used the ARCON96 code, their own version of that in ARCON to do the atmospheric dispersion analyses because that's a very short distance. And so we have completed that review.

Also the review of the pH control is also done. However, we are still evaluating the core damage event which they have now categorized as a special event. And they've added a new iodine spike design basis source term to be a bounding surrogate for any kind of release to the containment from the reactor vessel. That does not include core damage.

They first proposed these changes to the methodology framework in January 2018, but we did have a lot of discussions. So the submittal that they just gave us is the result of those discussions. Next slide, please.

So we are still reviewing the changes to the core damage event including aerosol deposition and iodine spike design basis source term including the source that you develop for environmental qualification of equipment inside containment because now they propose to not use a core damage source term for environmental qualification inside containment, instead use the release from the coolant as the source term for that.

MEMBER CORRADINI: But that would include an iodine spike?

MS. HART: That would include an iodine spike, and we do have some further questions about other constituents in the coolant whether there is a spike as well.

MEMBER CORRADINI: I see. Okay.

CHAIR MARCH-LEUBA: So what environmental qualification -- what instruments other than the level?

MS. HART: So the particular pieces of equipment that we were concerned about are the electrical penetration assemblies at the top of the containment, the isolation valves, and also the radiations outside containment underneath the bioshield.

CHAIR MARCH-LEUBA: And by that, I'm very interested in the water level measurement sensors which are likely to be microprocessor based with a fine something or other that meets the radar.

MS. HART: Right. So environmental qualification of any equipment inside containment is what we're discussing.

MEMBER CORRADINI: I translate to be they may not be as worried as you are, but you're trying to increase their level of worry.

CHAIR MARCH-LEUBA: I told you I'm a used car salesman. I keep repeating the topic over and over until eventually somebody listens and buys the car.

DR. SCHULTZ: Michelle, did you say that they have a new version or a NuScale version of ARCON96?

MS. HART: That's correct. And so what PAVAN does, it looks 360 degrees. And ARCON only does point to point. And so to address that difference, they had to make some changes in the methodology, how they use it.

DR. SCHULTZ: And the reason they're using ARCON because it's a closeup dose?

MS. HART: Correct, it's 400 feet, I think, is the distance --

DR. SCHULTZ: Okay, fine.

MS. HART: -- of their calculation --

DR. SCHULTZ: So you're --

MS. HART: -- distance for the EAB and the LPZ.

DR. SCHULTZ: And you're reviewing what they have done in the modification of the code? You're not --

MS. HART: Correct, and in the use of the code, yes.

DR. SCHULTZ: Right. You're not creating your own version?

MS. HART: We are not creating our own version. That is correct.

DR. SCHULTZ: Going over their changes?

MS. HART: Correct.

CHAIR MARCH-LEUBA: And let me continue to sell cars on the record. The reason I'm concerned about that is NuScale is the most safer design I have ever seen for a light-water reactor. I mean, it's beautiful. It's really well designed, well conceived. Everything works well as long as the ECCS RRV valves open. And they open if that sensor senses water level of a particular elevation in the containment.

So it is -- the Achilles heel of the whole design is those four level sensors must work. And they are a little flaky in my opinion. I would like to see some diversity on that measure.

MS. HART: I understand your comment. Thank you. So we did start an audit of especially the iodine spike design basis source term because that is new pieces of information. We started that audit on June 3rd. And in fact, most of my colleagues are over in the NuScale offices right now talking about that. And we do have -- for the topical report, we do have an ACRS subcommittee tentatively scheduled for December of this year.

So if you have any questions. I didn't really go in any details, but the rest of the events are all -- basically, they're following Reg Guide 1.183 for the most part.

DR. SCHULTZ: Where does fuel handling accident fit in there? Are they following 1.183 for a --

MS. HART: They are following --

DR. SCHULTZ: -- fuel handling accident?

MS. HART: -- 1.183.

DR. SCHULTZ: Right down the line?

MS. HART: Yes, they had originally proposed to use different pool decontamination factors because they do have a larger depth of water.

DR. SCHULTZ: Right.

MS. HART: But after some questions, they decided to go conservatively and stick with the decontamination factor of 200.

DR. SCHULTZ: So the results we've seen yesterday would have reflected --

MS. HART: That's correct.

DR. SCHULTZ: -- the 1.183 calculation?

MS. HART: That is correct.

DR. SCHULTZ: Good, thank you.

CHAIR MARCH-LEUBA: Alex is doing a fantastic job in keeping us on time in time for a break.

MEMBER CORRADINI: We want to break for 15 minutes?

CHAIR MARCH-LEUBA: Yeah.

MEMBER CORRADINI: Okay, 10:15 we're back.

(Whereupon, the above-entitled matter went off the record at 10:01 a.m. and resumed at 10:16 a.m.)

MEMBER CORRADINI: Okay. Why don't we pull us back together. And Alex, do you want to tell us where --

MS. SIWY: Sure.

MEMBER CORRADINI: -- we're going?

MS. SIWY: We'll start off talking about the re-analysis of Chapter 15 events with NRELAP5 Version 1.4 in the revised based model. Is that --

MR. SCHMIDT: Yeah.

MS. SIWY: -- Jeff?

MR. SCHMIDT: That's fine. Yeah. So this is Jeff Schmidt. So these slides here in the public version are kind of just a schedule and overview. The actual details of the changes are obviously prop. So you'll see a more detailed slide in the prop. So again -- oh, sorry. Again, so this was, as NuScale mentioned yesterday, revised in 2018 for some minor non-conservatives in the condensation heat transfer. There was a variety of other improvements made. And again, we'll go into details in the proprietary.

MEMBER CORRADINI: So if I might just --

MR. SCHMIDT: Sure.

MEMBER CORRADINI: -- ask. These encompass the open items 15.0.2-1 through 4 as far as I can tell because Version -- I mean, you guys have given enough of these. I actually have to remember them or else they --

MR. SCHMIDT: Yeah, unfortunately, you probably remember them better than I do at this point.

MEMBER CORRADINI: I have cheat sheets.

MS. SIWY: I don't remember the numbers exactly. One of the open times relates to the --

MEMBER CORRADINI: Version 1.4?

MS. SIWY: -- NRELAP5 -- yeah. And then two of those will relate to the LOCA EM still being under review --

MEMBER CORRADINI: And the non?

MS. SIWY: -- and non-LOCA. And I think that's -1 is LOCA, -4 is non-LOCA, something to that effect.

MEMBER CORRADINI: And then three was how LOCA EM handles CHF during long-term calculations. But in any cases, as I understand --

MR. SCHMIDT: LOCA, the topical report you mean, the appendix?

MEMBER CORRADINI: It's -- I think it's a technical report. It's -- I'm sorry.

MR. SCHMIDT: No.

MEMBER CORRADINI: It's not a technical report. I back away. But let's just keep on going.

MR. SCHMIDT: I think that's LOCA topical report.

MEMBER CORRADINI: Okay, okay, fine. Sorry. Excuse me. So we'll hear more in the closed session?

MR. SCHMIDT: Yeah, we can't go into the specific --

MEMBER CORRADINI: That's fine.

MR. SCHMIDT: -- changes. Again, RELAP base model, so the base model was 2015, used up to rev 2. Staff noted that RELAP models were not consistent with ANSYS solid models.

MEMBER CORRADINI: You mean in terms of areas, the differences?

MR. SCHMIDT: Yes, yes.

MEMBER CORRADINI: Okay.

MR. SCHMIDT: RELAP base model subsequently updated in 2017, will be used, as we talked about many times at this point, DC rev 3 in conjunction with NRELAP code Version 1.4.

MEMBER CORRADINI: So can I ask Jose's question to you guys again and see if I understood it? You're leaving it to NuScale to decide what to re-analyze with 1.4. You're not telling them what they must re-analyze. You're leaving it to them to decide.

MR. SCHMIDT: Yeah, it's up to NuScale to decide, yeah.

MEMBER CORRADINI: Okay, fine.

CHAIR MARCH-LEUBA: Yeah, but what criteria will we use, just bring me one?

MR. SCHMIDT: Again, it's up to the applicant to decide.

CHAIR MARCH-LEUBA: I'm asking you --

MR. SCHMIDT: We will review what they bring.

CHAIR MARCH-LEUBA: Yeah, that's the staff -- that's the official staff position. I shouldn't have asked. My apologies. No, and it is true. I mean, it's a proper position to take. I'm not complaining.

MR. SCHMIDT: Yeah, I mean, they can obviously justify something else and we'd have to review it. NuScale -- well, so this is what they've, I think, committed to us at this point, and NuScale can jump in if it's changed. NuScale reran Chapter 15 events using updated base model and then RELAP Version 1.4.

MEMBER CORRADINI: To show differences.

MR. SCHMIDT: What's that?

MEMBER CORRADINI: No, I'm fine. I understand what it means.

MR. SCHMIDT: Okay. Revised analysis are scheduled for July 31st of this year.

MEMBER CORRADINI: But as you have your initial bullets, there's two things. One, there was a set of changes they made and then also changes in the plant model too. So it's the combination of both the plant model and the analysis tool that you're able to get.

MR. SCHMIDT: So I'm not going to cover this slide too much in depth because we just got the changes and they aren't reflected in Chapter 15. And the guy who's actually doing the review is not here.

So NuScale submitted a plan to revise the MPS logic for operational considerations. I think they talked about that yesterday. It seemed to be startup related. Existing decay heat removal, actuation signal will be split into two signals which they talked about. A secondary side isolation for steam and feed and then decay heat removal actuation signal.

NuScale further plans to review ECCS actuation signal -- or remove, I should say, ECCS actuation signal on low reactor pressure vessel level which the applicant will not credit.

CHAIR MARCH-LEUBA: Okay. I'm in my -- again, my car selling mode. Having a backup for a possible failure on the containment level, it's not a bad idea. I mean, and unfortunately, these other level sensors are exactly the same method. So a common cause failure will affect all of them. But I'm just trying to sell cars. Don't need to answer.

MR. SCHMIDT: Noted. Staff --

MEMBER CORRADINI: So the staff's position is it's under review, correct?

MR. SCHMIDT: Yes, this is a fairly recent -- the staff got this fairly recently --

MEMBER CORRADINI: So no --

MR. SCHMIDT: -- maybe a month ago.

MEMBER CORRADINI: -- RAIs have been issued?

MR. SCHMIDT: No, no.

MEMBER CORRADINI: Okay, fine.

MR. SCHMIDT: I guess I say that a staff review is ongoing, will not be completed after the planned re-analysis of Chapter 15 to incorporate the logic changes.

MEMBER CORRADINI: But it's more subtle. Staff has never taken credit for this in the Chapter 15 analyses. Is that correct?

MR. SCHMIDT: As far as the level on the RCS?

MEMBER CORRADINI: Yeah.

MR. SCHMIDT: We have in rev 2.

MEMBER CORRADINI: You have --

MR. SCHMIDT: It's in there in rev 2.

MEMBER CORRADINI: And therefore, you've taken credit that it would be a system that --

MR. SCHMIDT: It's a valid actuation system.

MEMBER CORRADINI: It's a valid actuation system?

MR. SCHMIDT: Yes.

CHAIR MARCH-LEUBA: But is it safe to say that -- I mean, if you have those two trips, one of the two will trip before the other. And always the containment level trips earlier and therefore you can say that the other one is credited.

MR. THURSTON: In general, that's true. This is Carl Thurston, Reactor Systems.

CHAIR MARCH-LEUBA: If your name is in there, you don't need to tell it.

MR. THURSTON: Okay.

CHAIR MARCH-LEUBA: He knows who you are.

MR. THURSTON: Okay.

MEMBER CORRADINI: But just to make sure we're on the same page with you. This is a redundant trip. But normally, as Jose said, the level in containment will happen before getting to the set point in terms of pressure level and RPV?

MR. THURSTON: Yes, the containment level will trip first, yeah.

CHAIR MARCH-LEUBA: One possible concern would be, I mean, if you have a LOCA outside containment that is unmitigated, you could lower the level in the vessel and there is not level in the containment. So by opening the ECCS, you actually drain the vessel. That would be a concern.

MR. SCHMIDT: So I think there is still going to be a sensor for isolation for a break outside containment.

CHAIR MARCH-LEUBA: We're talking then ten to the minus eleven type of Chapter 19 events. So you have a LOCA outside containment that goes unmitigated. Your level in vessel will drop which would -- and their logic will open your RRV and you will make it worse.

MR. SCHMIDT: Right. So you're not talking like small breaks outside?

CHAIR MARCH-LEUBA: This is --

MR. SCHMIDT: You're talking a different --

CHAIR MARCH-LEUBA: Something that is sufficient to lower the level inside the vessel and put the water into the pool.

MR. SCHMIDT: If you can postulate, yeah --

CHAIR MARCH-LEUBA: That's one of those --

MR. SCHMIDT: -- something like that, then --

CHAIR MARCH-LEUBA: -- I agree that this is ten to the minus eleven.

MR. SCHMIDT: But not be picked up by this, yes. You probably won't pick it up in pool level.

CHAIR MARCH-LEUBA: Yeah. No, this would be a bad actuation of the MPS because you would put the water at RRV level in the vessel. Because it would drain the vessel into containment.

MR. SCHMIDT: Right.

MEMBER CORRADINI: But it doesn't --

CHAIR MARCH-LEUBA: It still will be acceptable.

MEMBER CORRADINI: That's what I was going to say. Fine, okay.

CHAIR MARCH-LEUBA: So that's a consideration. I mean, you always have to consider that's such a low probability then that it certainly begins investment.

MEMBER CORRADINI: Can I make -- maybe because I remember the first bullet under the changes. Can we ask the applicant why the second change about the removal of the low pressure, pressure rise level? Because I don't remember that. Can the applicant explain the logic for removing that?

MR. BRISTOL: Can you clarify the question?

MEMBER CORRADINI: Yeah, why are you doing the second bullet under NuScale further plans to remove ECCS actuation?

MR. BRISTOL: On low reactor level? So as Carl kind of mentions, in reviewing the analyses, the containment level actuation was the one that came in -- always came in before low RCS level. The initial thought behind the low RCS level in the design was considerations of some severe accidents that ended up not working necessarily as intended or being necessary.

And in working through optimizing with the I&C folks, there was some advantages of -- from a sensor perspective, of removing that function as a requirement. And we determined it was necessary in safety analysis.

In addition, as part of the changes in the containment analysis, we were interested in delaying ECCS actuation. So the containment actuation level actually moved up as part of this change. And since we are making that change, the low RCS level was removed from the design.

MEMBER CORRADINI: So okay, I think I got all that. So my summary of that is it's an unnecessary redundancy.

MR. BRISTOL: From --

MEMBER CORRADINI: That's what I heard you say to me.

MR. BRISTOL: From a safety analysis perspective, it wasn't required.

CHAIR MARCH-LEUBA: But did you consult the Chapter 19 people? Did you do a PRA evaluation? Probably the PRA is not deep enough to consider this as a backup of the other sensors anyway, so --

MR. BRISTOL: Yeah, in the PRA, the events that you're alluding to, actuating ECCS can be detrimental depending on the conditions and the time. If the inventory has been drained down to that condition and then you actuate ECCS and it continues to drain, that can accelerate in some cases the time to which the code covers. But overall, it's in the noise of their events.

CHAIR MARCH-LEUBA: Yeah.

MR. NOLAN: Okay. I'm Ryan Nolan. I'll be presenting the next two topics for 15.0. First topic is related to crediting of nonsafety systems as backups as well as single failure treatment of the IAB.

So the NuScale Chapter 15 does credit nonsafety related valves as backups to the safety related valve when applying single failure criteria. And Table 15.0-9 identifies the different events in which these valves are credited. And I think NuScale did an overview of this yesterday and talked about the individual valves and identified them.

This approach goes back to a technical paper that was written, this NUREG-0138 about the mid-70s. And first let me talk a little bit about that and how typical PWRs have addressed this. This is a pretty common approach for secondary line breaks. SRP 10.3 for the main steam system even addresses criteria and goes into a little bit more detail.

But the general concept is if you have a main steam line break and you apply a single failure to the good intact steam generator main steam isolation valve, you could potentially go down depending on what design you have to steam generators. And what the paper does is it looks at any downstream valves that can provide isolation.

And they look at GDC-1 and they look at the consequences due to secondary line breaks and compares to the consequences typically applied to safety related components credited for mitigating LOCAs. And what that paper describes is turbine stop valves and control valves are typically credited for doing this. And the basis is the consequences for a main steam line break are not as severe as primary line breaks.

There augmented quality to be applied to those components. There's surveillances. Their in tech specs are tested fairly often, and then there's a lot of operating experience. So generally speaking to reliability of these nonsafety related components to provide an isolation as a backup function.

And so NuScale is applying this approach to a lot of their secondary line break scenarios. It's a little bit different in that they're not crediting turbine stop control valves. They're crediting -- they actually have a second MSIV and then some check valves and feed reg valves on the feedwater side.

And so fairly consistent. A lot of those valves are in tech specs. They are tested. There's augmented quality applied to them. NuScale briefly overviewed. They're seismically qualified. They're scoped into ISI, IST programs. And so we found it acceptable for the secondary line break scenarios.

CHAIR MARCH-LEUBA: So this is not even an open item. This has been decided.

MR. NOLAN: Yeah, no open item. Next slide. So one area where NuScale sort of goes beyond what was discussed in NUREG-0138 is crediting the secondary MSIV for providing isolation for a steam generator tube failure event. In this case, this is sort of a primary line break scenario. And so it goes a little bit beyond the existing guidance.

And so in this case, we requested NuScale perform a sensitive study assuming that that fails to function for a couple reasons. One, to make sure it doesn't trip the safety related definition, the criteria, primarily the third criteria for offsite doses as well as to judge generally what are the consequences if that valve does fail.

And the results are about a 50 percent more increase in mass and a proportional increase in dose. And so the results show large margins to both offsite and control room doses. And so we accepted it.

DR. SCHULTZ: So this is a sensitivity study. The results that NuScale is presenting and can rely upon are their original analysis taking that credit?

MR. NOLAN: Yeah, yeah.

DR. SCHULTZ: And this is just a demonstration that things don't go off in an unexpected direction?

MR. NOLAN: Yeah, correct. We just wanted to make sure that the consequences of the event are still significantly low.

CHAIR MARCH-LEUBA: So would you say that what this analysis shows is that you don't really need the MSIVs? You can leave them open and nothing happens?

MR. NOLAN: Yeah, and I think they had that discussion internally. And --

CHAIR MARCH-LEUBA: I'm glad --

MR. NOLAN: -- ultimately, it's up to them on how they want to treat it.

CHAIR MARCH-LEUBA: I'm glad they decided to put them in, but --

MR. NOLAN: Yeah, I mean, in reality, I think it's one step better than what the current PWRs are doing instead of crediting a stop valve that's hundreds of feet away plus a lot of smaller pipes and valves off of those lines. This valve is just a couple feet after the primary isolation valve.

CHAIR MARCH-LEUBA: I would like to go to the concept I used before that what you did was a risk informed decision only that you rely heavily on the consequences. Typically, when we talk risk informed, we're going to the frequency and we forget all the consequences. But doing a consequence analysis is part of a risk informed decision and we should take credit for it.

MR. NOLAN: Well, I think we were already comfortable with the reliability of the components since it's already -- it has a lot of augmented quality already tied to it. And so we just wanted a better feel for the consequences.

CHAIR MARCH-LEUBA: My claim is from the top down, we could get -- you should do more risk informed regulation. This was risk informed regulation. We already do it. Take credit for it.

(Laughter.)

MR. NOLAN: All right. We will. All right. Next slide. So the next topic I'll cover is treatment of the IAB and single failure. The first couple slides is more of just an overview of the policy in general and then the following slides are a little bit more specific to the --

MEMBER CORRADINI: We have --

MR. NOLAN: -- NuScale design.

MEMBER CORRADINI: -- your SECY. We have the staff SECY.

MR. NOLAN: Yes.

MEMBER CORRADINI: Is this -- I can't remember if it's 34 or 36. I get confused.

MR. NOLAN: 36.

MEMBER CORRADINI: 36? I knew there was a three in there somewhere.

MR. NOLAN: Yeah. So I mean, at a high level, I think most people are familiar with this. The single failure criteria is an important element of the NRC's defense-in-depth philosophy. It's primarily there to promote component reliability and competence that the systems will perform their safety function.

It's applied at the system design level and the plant safety analysis level. And what I mean by that is there's certain GDCs that drive the system itself. For example, ECCS as well as Appendix K talks about when performing the transient analysis to also look at the most damaging single failure. Lastly, it's required by regulations.

Here's an overview of the regulations guidance and policy. I mentioned the GDC's Appendix A. There's a couple of them. It's also defined in Appendix A. Appendix K also mentions single failure. It's identified in the SRP 15.0.

And then lastly, there's two primary policy papers that describe the approach to the single failure criteria. The first one is SECY-77-439, and that goes through the definitions of what's a single -- what's an active failure, what's passive failure, how to treat mechanical components, how to treat electrical components.

And the one thing I want to highlight here is there's an exception for simple check valves that they're typically reliable enough and close enough to a passive failure that we wouldn't apply single failure to them. Later in the '94 SECY, there was sort of a clarification and the Commission was re-engaged on this in that there were certain passive designs that were using check valves in low dP systems.

And so it was really the Commission reaffirmed that single failure does apply to check valves. They are a mechanical component. However, if you can show through testing and operating experience that its failure on the order of a passive failure, then you don't need to apply the single failure criteria.

MEMBER CORRADINI: To show what? I'm sorry.

MR. NOLAN: If the probability or the reliability of the check valve is sufficiently high, normally on the order of what a passive failure would be. So a pipe break --

MEMBER CORRADINI: Do we know what that is?

MR. NOLAN: -- or something similar.

MEMBER CORRADINI: Do we know what that answer is?

MR. NOLAN: So this SECY calls out I think

E to the minus four is what it uses in here.

MEMBER CORRADINI: Okay.

CHAIR MARCH-LEUBA: Is that per month, E to the minus four per month?

MR. NOLAN: So it's debatable, and I don't really want to go into the details.

CHAIR MARCH-LEUBA: I don't know what to go into --

MR. NOLAN: It's not really that important for the discussion. I just wanted to highlight --

CHAIR MARCH-LEUBA: In this case --

MR. NOLAN: -- that there are policy papers out there.

CHAIR MARCH-LEUBA: In this case, E to the minus four per month in NuScale would probably end up being once in the life of the universe.

MR. NOLAN: Yes. And through the interactions with NuScale, I think it's been determined that the '94 SECY NIST isn't really applicable to their design because it's so specific to check valves in low dP systems. It's not applicable to the IAB. And so most of the discussion were all going back to the 77 policy paper.

CHAIR MARCH-LEUBA: But that would be my comment again. The pressure on the little piston that moves in the IAB is for a containment vacuum to 1,700, 1,600 psi. When it has it up to 1,800 and above, right? So it's big delta P.

MR. NOLAN: Yeah, across a small area.

CHAIR MARCH-LEUBA: Which they would argue that is in their favor because the big delta P is pushing the piston out.

MR. NOLAN: Yeah.

MEMBER BROWN: If a check valve never gets exercised because it's never used in the life of the plant, who's to say it's ever going to work after it sits there and gets sticky after a while. Things that are passive like that, if you don't exercise them, there's crud in the water and everything else, eventually can stick and therefore it's not longer -- it is a single failure if they're not exercised. So it looks like the application, in the times in the application when they may be operated for other reasons would be a key element of whether you can consider it eligible for single failure or not.

MR. NOLAN: And I think that's a consideration, especially in Chapter 3. Most -- these are safety related components, and so there's a lot of programmatic controls that are tied to those components. Alex, you can go to the next slide.

So the NuScale Chapter 15 analysis does not assume a single failure of the IAB closing function. And let me talk --

MEMBER CORRADINI: Under the assumption that I might have an initiating event that would have a failure of the combined valve system.

MR. NOLAN: Yeah. So what I mean by the closing function is -- well, I guess, the opening -- so if the IAB closes successfully, the opening function, we're not as concerned with applying single failure because that's already inherently taken care of by just assuming a single failure of the main valve, right?

And so the focus here is the closing function and the fact that all -- I guess four others. If your initiating event is one main valve opening, it's all four other IABs have to quickly close. And if one of them doesn't, then you start to lose enough RCS pressure where all the others will go open. But it's tied to 15.6.6 analysis. And currently, it only assumes one valve. And so if you assume a single failure of another IAB, then you end up opening two valves and it challenges CHF primarily.

MEMBER CORRADINI: The challenge -- let me a question. The challenge of CHF is a function of pressure or stored energy?

MR. NOLAN: Yeah, it's the rate of depressurization.

MEMBER CORRADINI: Is it the rate --

MR. NOLAN: The rapid --

MEMBER CORRADINI: -- or the absolute -- that's what I'm trying to understand. Is it the -- it can't be the rate. It's got to be the absolute pressure, yes?

MR. THURSTON: It's primarily the pressure drop, the pressure drop.

MEMBER CORRADINI: Okay. So let me ask my question a little bit differently to make sure I'm clear. If I were to start the whole thing at 1,200, it's not an issue of rate. If I were to start the whole thing at 1,800, it's not a matter of rate. It's the matter of the absolute magnitude. And then when I take it down, I would have then a difference in the CHF ratio. Am I understanding it correctly?

MR. THURSTON: Correct.

MEMBER CORRADINI: Okay, fine.

CHAIR MARCH-LEUBA: So I don't understand. I mean, the pressurization, is it 15.0.6? No, not 0.6. It's 15 point something and 6. We the inadvertent opening of the ECCS. Maybe 15.6.

MR. THURSTON: 15.6.

CHAIR MARCH-LEUBA: Then we see the pressurization happening. And then when it hits the right pressure, then the other valves open and then the pressurization is faster. But what causes the CHF? The scram happened on second number one. I mean, it happened immediately. So you indicate heat already.

MR. NOLAN: Yeah, so it's a very fast -- it happens in the first half a second. And it's just -- it's the --

CHAIR MARCH-LEUBA: The CHR?

MR. NOLAN: Yeah.

MR. THURSTON: CHF happens.

MEMBER CORRADINI: It moves before the rods move.

CHAIR MARCH-LEUBA: Before the rods move?

MR. NOLAN: Yeah, it's the --

CHAIR MARCH-LEUBA: How fast does the pressure drops? I mean, did the pressure drop that fast?

MR. THURSTON: The pressure can drop that fast.

MR. NOLAN: Plus a momentary flow stagnation at the same time.

CHAIR MARCH-LEUBA: That too. That too.

MR. NOLAN: It's a very quick event from a --

CHAIR MARCH-LEUBA: I thought the pressurization was so large that it would give some time to -- but no, you calculated it, right, Carl?

MR. THURSTON: With the big valves at the top, right? So you're venting out a lot of steam quickly.

CHAIR MARCH-LEUBA: No, but this is something --

(Simultaneous speaking.)

MEMBER CORRADINI: But it's the RRV is the limiting --

CHAIR MARCH-LEUBA: The RRV.

MEMBER CORRADINI: -- I thought.

MR. NOLAN: They're both very close.

MR. SCHMIDT: Yeah, they're both very close. There's not a lot of separation between those two.

CHAIR MARCH-LEUBA: Yeah, I've never seen it blow up on the first few seconds of that time. And we have stamp size figures on 2,000 seconds. So you're just getting a line. It would be nice to see the first --

MR. SCHMIDT: It should be in the sequence of events.

CHAIR MARCH-LEUBA: It is. That's the conclusion. I like to see plots.

MR. SCHMIDT: Understood.

MEMBER CORRADINI: And are we going to come back to this in closed session? Or are the -- to put it in terms of his question, are we going to see the plots in closed session?

MR. NOLAN: We weren't --

MR. SCHMIDT: No.

MR. NOLAN: -- planning on --

MR. SCHMIDT: No.

MEMBER CORRADINI: Because I'm curious of what's the correlations being used to determine this. Is it Hench-Levy?

MR. THURSTON: It's Hench-Levy.

CHAIR MARCH-LEUBA: For us that don't speak the language, is it the conservative or the nonconservative one? I mean, the more conservative or the least conservative?

MR. SCHMIDT: We should leave that for closed session.

MR. NOLAN: And recognize that this is tied to Appendix B of the LOCA topical that we're still reviewing it.

CHAIR MARCH-LEUBA: Can you make a note to bring those blowup plots in October? Or would it be the LOCA topical? That would be very nice. Just the first ten seconds or maybe the first second, whatever is relevant. Because the plots that we get in the FSAR is a stamp size line that you can see it.

MEMBER CORRADINI: So I have one last one, then I want to make sure I understand about timing. So if I just say it back to you. The assumption is whether I have the RVV or the RRV as the initiator. And I essentially have an instantaneous opening of those. Within less than a second, I'm going to pass through essentially CHF.

MR. NOLAN: Yes, that's right.

MEMBER CORRADINI: And if I have a failure of the IAB upon closing which would delay the opening of the others, I'd pass through sooner? I can't believe that.

MR. THURSTON: Can you repeat the question?

MEMBER CORRADINI: I'm trying to understand. The way you answered Jose is that if I have the initiating event for the LOCA which is the assumed opening of the RRV or the RVV, I will pass through CHF quite soon.

MR. NOLAN: You reach the minimum value --

MEMBER CORRADINI: Quite soon?

MR. NOLAN: -- quite soon. And then so if you apply a single failure to another valve so you have two valves instantaneously opening at the same time --

MR. SCHMIDT: You've got two valves -- effectively two valves --

MEMBER CORRADINI: Then I pass through it?

MR. SCHMIDT: Yes.

MEMBER CORRADINI: Based on a criteria? Okay, fine. I misunderstood. I misunderstood that.

MR. THURSTON: So with two valves, you would likely violated your CHF criteria.

CHAIR MARCH-LEUBA: To the risk of the technical, I'm looking at Chapter 15, the public version, 15.6. And it drops -- the MCHFR drops to 158 to 142 instantly. I mean, the next plot point goes down and then it just stays at 142. It's not that -- well, to start with, it's one of those that starts with in the low CHFR which 158, which we believe to be the very conservative correlation. And the delta is only, like, 0.16.

MR. NOLAN: Yeah, and --

CHAIR MARCH-LEUBA: And that's not bad.

MR. NOLAN: Yeah, and I'll cover the 15.6.6 transient at a high level later on.

MEMBER CORRADINI: Okay, fine. That's fine.

MR. NOLAN: And yeah, so -- and what you'll see there is there is considerable margin to --

MEMBER CORRADINI: Well, that's fine. You clarified my confusion. That's all I wanted. I'll remember that, at least for now.

MR. NOLAN: All right. And so the second bullet is really getting to what we were just discussing it. We've identified it as -- or we categorize it as a safety significant component in part because it does challenge CHF. And here the regulatory requirement GDC 10 challenges the staff. So that's why we categorize it as a safety significant component.

MEMBER CORRADINI: So an argument, not the argument, but an argument might be made that the reliability of the IAB is of such a level that it looks like a passive component?

MR. NOLAN: Yes, and I was going to speak a little bit to it.

MEMBER CORRADINI: Okay, fine.

MR. NOLAN: And just to clarify, it's an active component. It's a mechanical component that requires movement. It's whether it's not subject to single failure.

MEMBER CORRADINI: I understand.

MR. NOLAN: And so yesterday you'll hear the term, treated as passive. That's synonymous with active not subject to single failure. We sort of used the terms interchangeably and I think it's --

MEMBER BROWN: How can you call something --

MR. NOLAN: -- causing confusion.

MEMBER BROWN: -- reliable when it's a first of a kind and never been used? There's no experience with it. You just -- it's a qualitative judgment.

MEMBER CORRADINI: We did it for squid valves on two plants that I can think of.

MEMBER BROWN: If you remember my comments in that letter --

MEMBER CORRADINI: I'm simply poking the bear.

MEMBER BROWN: -- an exploding valve that you can't -- you have to replace it. So you can never check it in service. It's done. But we've passed that one. Okay?

MR. NOLAN: Yeah, and that's sort of the --

MEMBER BROWN: You brought it up, not me.

MR. NOLAN: And that's sort of the staff's position on this. We've engaged the applicant heavily on this topic, and they wrote a letter the end of December putting forth their arguments and partly qualitative, a little bit of risk in there, correlating it to known components. And so they tried to correlate to other components that we do have lots of data on and whether it's similar or not.

MEMBER BROWN: Yeah, but it's a piece part. It's more believable than the exploding valves. I'll agree with that.

CHAIR MARCH-LEUBA: At least you can --

MEMBER CORRADINI: Are you on the record for that?

MEMBER BROWN: Yes, it's better than exploding valves which you can't test and service at all. Because once you test them, they're not working anymore. So --

MR. NOLAN: And so this --

MEMBER BROWN: -- you have to replace them.

MR. NOLAN: And this is why we wrote this, the SECY paper, is we recognize that there's some gaps in the policy. There's some ambiguity there. And so we needed the Commission engagement to interpret and help push some options forward or determine where we should go from here.

MEMBER CORRADINI: Is the issue functionality or functionality at the given set point? Because in my view, just one member's view, it'll work. Will it work at exactly 1,200?

MR. NOLAN: Yeah, and so in this case, the component has to respond in such a short period of time on the first attempt. Otherwise, the other valves are going to open. The main valves are going to start to open. So it's --

MEMBER CORRADINI: So --

MR. NOLAN: -- not really a --

MEMBER CORRADINI: So let me ask you a technical question that maybe we can reserve for later. But where I'm going with this is, again, we had a valve education earlier this week. Okay? And I think I got it. But when we had our valve education, my interpretation of that is it probably will open. Whether it'll open exactly at the given set point is where I think the uncertainty lies.

MR. NOLAN: So we're -- yeah, we're concerned with --

MEMBER CORRADINI: Are we going into --

MR. NOLAN: No, no. We're going to close with a closed function.

MR. SCHMIDT: Yeah, just clarify, it's the -- I'm sorry, Ryan. It's the closed -- it's closing --

MEMBER CORRADINI: So I'm misinterpreting?

MR. SCHMIDT: Yeah, so it's really key to pick up the fact that this is blocking a blowdown at high pressure. This is not the opening function to address a LOCA.

MEMBER SKILLMAN: This valve plugs --

MR. SCHMIDT: Doesn't block.

MEMBER SKILLMAN: -- to stop --

MR. SCHMIDT: Yeah, yeah.

MEMBER SKILLMAN: -- opening.

MR. SCHMIDT: Right. We're focused on the block.

MEMBER SKILLMAN: And there's a time challenge here. This has to close so quickly to block the other valves from opening that the internals of the IAB are critical. The clearances in there are probably down to ten-thousandths of an inch. But this valve has to close quickly to prevent the RRVs from opening.

MEMBER CORRADINI: Okay. I've got to go back to my cartoon picture. Thank you.

MEMBER BALLINGER: Yeah, experience with these pilot valves are such that if they close then have to open. And if they're closed for a long time, they tend to stick. And that doesn't mean they don't open. It just means that they don't open at the set point.

MEMBER CORRADINI: Okay. But I'm --

MEMBER BALLINGER: But if they have to close, then they're much more reliable. In other words, they have a stop.

MEMBER CORRADINI: You're making the argument for me. Thank you.

MEMBER BALLINGER: Well, it's just experience.

MEMBER CORRADINI: Okay, okay. But I misunderstood what you guys were concerned about. It's not the open function. It's the close function. Excuse me. Sorry.

CHAIR MARCH-LEUBA: Ron, can you explain to me? I mean, if the valve stem has to move, that's bad.

MEMBER CORRADINI: Well, they have to move in either direction. It's a matter whether they move --

MEMBER SKILLMAN: It has to move --

MEMBER CORRADINI: -- to close or move to open.

CHAIR MARCH-LEUBA: Move to function.

MEMBER SKILLMAN: I has to move 135 mils. It has to move just over an eighth of an inch to block.

(Simultaneous speaking.)

MEMBER BALLINGER: There are two problems with these valves that I'm aware of. One is sticking seat to disc which we can explain. The other is the shaft getting warped or something like that. That's much less common. But it's the seat sticking thing that's the most common. But that's the standard BWR set point drift problem for steam relief valves, pilot operated --

(Simultaneous speaking.)

CHAIR MARCH-LEUBA: Without going into proprietary --

MEMBER BALLINGER: Yeah, we should wait.

CHAIR MARCH-LEUBA: -- it's an open -- the closed -- think of it as a disc that has to move and close the flow. The disc is open and a spring is pushing against it. So you have to have a change in pressure to push it up. You have to move the disc against the spring and it has to close in a short time.

MEMBER CORRADINI: So if we're going to debate this, I think we better debate it in closed session. But I think I get it now. I misinterpreted what you guys were saying. That's my fault.

MEMBER BROWN: Well, there's one other aspect to the -- when you have tight tolerances as Dick alluded to, it doesn't take but minor contamination in the fluid to make it stick. And I can count on -- I don't have enough hands to count on the times in our service in the fleet -- Navy fleet that you dealt with contamination. We were always flushing the darn things to try to keep them -- to try to make sure they could make their operation. So I mean, there was a lot of down sides.

MEMBER BALLINGER: It depends -- it also now depends on --

MEMBER BROWN: The best argument would be -- the best argument for this would be if it doesn't have to operate, do you still -- all you're doing is preventing other valves from opening.

MEMBER BALLINGER: The crapping up part.

MEMBER BROWN: Does that mess anything up? I mean, are you still protected?

MEMBER BALLINGER: It depends on orientation of the valve, whether stuff gets into that region.

MEMBER SKILLMAN: I think we need to be careful there. I think we've got to give NuScale a salute for making a reasonable effort to prevent inadvertent blowdown.

MEMBER BROWN: I don't disagree with that. I don't disagree with that.

MEMBER SKILLMAN: I think that's a good thing. And I think they're between a rock and a hard place.

MEMBER BROWN: It's difficult.

MEMBER SKILLMAN: This is really a very tricky hydraulic problem to resolve. And they've got Target Rock working on this, and Target Rock's got a lot of experience in this exact realm.

MEMBER BROWN: I've got a lot of experience --

MEMBER SKILLMAN: They're doing the --

MEMBER BROWN: -- in Target Rock valves sticking also.

MEMBER SKILLMAN: I got that.

MEMBER CORRADINI: Why don't we move on.

MR. NOLAN: This is identified as an open item.

(Laughter.)

MR. NOLAN: This next slide is meant just to be a high-level summary. The intent of the presentation is not to go into detail on the paper but it's just to present what the options were that were presented to the Commission.

The first option is an extension of the existing policy. And we recognize that NuScale's design is unique here in that the stylized approach to Chapter 15 requires that you not credit nonsafety related power. And so a lot of the Chapter 15 transience assume the DC power system goes away. And that's essentially what would require the IAB to function because the DC power, you lose power to the trip solenoid valve. The IAB would then be required because you're operating at normal RCS pressure to then change position and keep the main valves closed.

And so because the staff couldn't come to the conclusion at the IAB by itself is reliable or as reliable as a passive component. Option 1 considers the combined reliability. What's the reliability of actually losing the DC power system that would require the challenge on the IAB in combination with the IAB itself?

And so we recognize there's an inherent reliability tied to the IAB. It's a safety related component. It is first of a kind. And so that's what Option 1 is, is looking at the combined reliability of the integral system, not just an individual component. And that would be -- Option 1 would be an affirmation that single failure does apply to the IAB.

Option 2 is also an affirmation single failure applies in the traditional sense. And this would require -- there's several different things that NuScale could do here. But it would require additional analysis, maybe using different acceptance criteria, potential design changes to the IAB or the ECCS valves or an exemption request.

MEMBER CORRADINI: But number three is -- it behaves from a reliability standpoint like a passive component?

MR. NOLAN: So number three would a Commission determination that the arguments or justifications NuScale put together in their letter are sufficient to determine that single failure does not apply to the IAB per the 77 SECY paper. Because we didn't have really quantitative data or the 77 policy wasn't necessary based on quantitative data. A lot of it was a qualitative approach.

CHAIR MARCH-LEUBA: But let me see if I understand where we're going. The initiating event is you lose DC power to all five solenoids. There are five IABs. You lose power to all five solenoids. All of them want to --

MR. NOLAN: The initiating event is one valve fails for whatever reason.

CHAIR MARCH-LEUBA: Let me do a different --

MR. NOLAN: Okay.

CHAIR MARCH-LEUBA: -- problem. My initiating event is you lose power to all five solenoids. You open the three valves, saw that it is low pressure. Let's not discuss how it works. One IAB fails and therefore one ECCS opens which is what we have analyzed, and that has no consequence.

MR. NOLAN: Right, yeah. The 15.6 --

CHAIR MARCH-LEUBA: 15.6.

MR. NOLAN: -- would stand as the analysis.

CHAIR MARCH-LEUBA: A second IAB fails and opens and you challenge a little bit of CHF. Wouldn't that be a special event where CHF is not my criteria? That definitely is not an AOO. You have to lose your DC. You have to fail one IAB and then fail a second IAB. And then you're supposed to write a common cause.

MR. NOLAN: Right. And so that would be embedded in Option 2, the first two sub-bullets there is just redo an analysis or look at different acceptance criteria.

CHAIR MARCH-LEUBA: But where I'm trying to go is I don't think we need an exception. If the event is recategorized or appropriately categorized as beyond design basis because it will require you have to lose all your batteries because you have the trip all five. You can have a failure of --

MR. NOLAN: I don't know if that would be considered beyond design basis. Maybe at best, a PA.

CHAIR MARCH-LEUBA: A special event. Because a little CHF doesn't do anything.

MR. THURSTON: But they are not allowed to violate the CHF criteria.

MEMBER CORRADINI: Yeah, we're with --

MR. THURSTON: They don't have a methodology for that.

MEMBER CORRADINI: -- you there. We're

with you there.

MEMBER DIMITRIJEVIC: Let me just ask you. What is your initiating event? That would mean they have a problem. Is your initial event loss of DC power? It's lose -- or your initial event is an inadvertent opening of that?

MR. NOLAN: Inadvertent opening.

MEMBER DIMITRIJEVIC: Okay. So how did this inadvertent opening of ECCS valves happen? What challenged the valves so it will open inadvertently?

MEMBER CORRADINI: They don't have to --

MR. NOLAN: They're not really postulating what the exact failure is.

(Simultaneous speaking.)

MR. NOLAN: It could be a mechanical failure, yeah.

CHAIR MARCH-LEUBA: Yeah, but that would not be an IAB issue.

MEMBER DIMITRIJEVIC: Yeah, that would be an IAB issue. That's what I want to say. It has a double prevention of that. So two events have to happen.

MR. NOLAN: But remember that single failure, it's an initiating event plus a single active failure.

CHAIR MARCH-LEUBA: Yes, but the reason an ECCS valve is going to open is not because the IAB is going to fail. It's because the disc that is closing the valve breaks. The main valve breaks. The IAB was never challenges. That's your initiating event. Now what's the second failure? You would have to lose power to the solenoid and fail the IAB. That's two failures.

MEMBER CORRADINI: Well, but I think they're thinking in a very straightforward manner which is something happens and then they apply the single failure criteria to all the components that have to respond to it.

CHAIR MARCH-LEUBA: I'm asking --

MEMBER CORRADINI: That's what I --

MR. NOLAN: The DC power system which is keeping the trip solenoid valves closed is nonsafety related. And so we wouldn't credit that in Chapter 15 as being available.

CHAIR MARCH-LEUBA: So you assume this always fail?

MR. NOLAN: Correct, unless it makes it worse to keep it there.

MEMBER DIMITRIJEVIC: So you're assuming a challenge to those ECCS valves all the time?

MR. SCHMIDT: So Option 1 is kind of -- I think what you're getting at is if you look at the combination of the reliability of the DC power which holds the main valve and the IAB, it seems like you're drifting towards -- I mean, it seems like you're describing to me maybe Option 1.

CHAIR MARCH-LEUBA: Absolutely. I'm saying the probability -- I mean, let's look at how you can open two ECCS valves by mistake, by failure. And I mean, the probability is ridiculous.

MEMBER DIMITRIJEVIC: Well, he's talking probability. I would think the probability would come and never mind. I think even --

MR. SCHMIDT: Is your green light on?

MEMBER DIMITRIJEVIC: -- if you simply count, like you count in Chapter 15. I think if you just simply count, you cannot have one initiating event and one single failure leading to this. Because even if your IAB fails, it has to fail in the moment the ECCS are challenged, right? It has to fail to close and the challenge to prevent opening. So there has to be something challenging them.

MR. SCHMIDT: I think it's just taken as a mechanical failure of the valve. That's the initiating event and then the other --

MEMBER DIMITRIJEVIC: And then it challenges the second one.

(Simultaneous speaking.)

MR. SCHMIDT: So the second valve has to block now.

MEMBER DIMITRIJEVIC: -- which requires opening the other valves and then IAB fails.

MR. NOLAN: Right. That's how the -- that's the stylized approach to Chapter 15 and meeting the regulatory requirements for that event.

MEMBER CORRADINI: It's not in the realm of probability. It must --

MEMBER DIMITRIJEVIC: I know. I was trying to even get counting down without probability.

MS. KARAS: Let me try and clarify and make sure everybody understands the event we're talking about with 15.6.6. Okay? The initiating event that has been stated is a mechanical failure of one of the valves, like the main valve, right? And that's what NuScale has postulated at an AOO frequency. Okay?

So you have the mechanical failure of that one valve. At the same time with Chapter 15 rules, it's not considered an additional failure. Just it can't be credited, right? Nonsafety related power is going to go away at the same exact time by Chapter 15 rules.

So what you're going to have happening is as that one valve opens and also at time zero, all of the other IABs when that power is lost immediately have to move to block the other four valves from opening. So that's how it's been analyzed currently in Chapter 15 without an additional single failure.

So if you take a single failure of one of those four IABs to rapidly reposition preventing their main valves to open, then you have at the very start of the event two valves -- to main valves opening instead of one. That has not been analyzed. So that's the specific situation that we're talking about. So it's an initiating event and a single failure of an IAB has not been considered. Is that --

CHAIR MARCH-LEUBA: But it has not been analyzed as with an AOO criterion which is CHF. But if you are with your Option No. 1, it would be analyzed a two-valve opening with more relaxed criteria.

MR. NOLAN: Option 1 would be an exception to the single failure criteria. If the combined reliability is sufficiently high, you don't need to apply the criteria. And so 15.6.6 would stand as is.

CHAIR MARCH-LEUBA: Right. But Option 1 basically what it's saying is that you will have a postulated accident or a special event that can get two valves open. Is that what you're saying?

MR. NOLAN: No, that's an AOO.

MR. SCHMIDT: It's one valve. The mechanical failure of the valve would be the failure. And their combined reliability would keep the other valves closed.

CHAIR MARCH-LEUBA: And you would never analyze two valves open?

MR. SCHMIDT: No, we would not.

CHAIR MARCH-LEUBA: And is -- but then maybe what I'm proposing is number four. We analyze two valves open with relaxed criterion which is no fuel damage like you do otherwise, like you do LOCA.

MEMBER CORRADINI: I think if --

MR. SCHMIDT: Yes, that is under Option 2.

CHAIR MARCH-LEUBA: That's Option 2?

MR. NOLAN: Yeah, alternate acceptance criteria.

CHAIR MARCH-LEUBA: But that one requires an exception request?

MR. NOLAN: No, exemption would be a totally separate option.

CHAIR MARCH-LEUBA: Then I'm reading it wrong.

MR. SCHMIDT: It would require a methodology change.

MS. KARAS: Option 2 has multiple sub-options that are paths that the applicant could follow, one of them being requesting an exception to not consider it subject to single failure.

CHAIR MARCH-LEUBA: So maybe what I'm looking for is 2B, two with an alternate acceptance criteria.

MEMBER DIMITRIJEVIC: But they're analyzing an impossible event because, okay, a valve can open eventually. But what probability in the next second?

CHAIR MARCH-LEUBA: The disc is going --

(Simultaneous speaking.)

CHAIR MARCH-LEUBA: The disc is going to be --

MEMBER DIMITRIJEVIC: That's not going to happen.

MEMBER CORRADINI: Okay. I think we need to move on. And I think we understand what you guys put up to the Commission, and I think there's a slide that's going to say it's in their court.

CHAIR MARCH-LEUBA: Yeah, but Mike, we need to ask ourselves, ACRS, for our discussion, are we going to communicate to the Commission what we think we should do?

MEMBER CORRADINI: That we have that option in the letter in July.

CHAIR MARCH-LEUBA: And therefore, we need to understand the issue in detail. And then we need to talk among ourselves to see what we would like.

MEMBER CORRADINI: But if you're going to get back to the technical options, I think what you've been debating is internal to Option 2 because they have multiple subsets. If you look at 36, they have multiple subsets of Option 2. And what I think Jose is suggesting could fit into that.

CHAIR MARCH-LEUBA: I'm saying 2B.

MR. SCHMIDT: So just to add full context to this because we're going to be getting into this subject next is the return to power SECY. If you go to a different criteria than CHF, let's say you go to fuel damage criteria, that SECY and the return to power analysis assume that you would not have CHF through the whole event. So if you ended up damaging fuel, I think that SECY would be in question.

MS. FLORES: For that event, yes, because that's the path you chose. But if that event has low probability, you can move it from -- you should not analyze it with AOO criteria.

MS. KARAS: So it's an event progression, right? So that has the acceptance criteria of the initiating event. So it's considered a progression from -- and we'll get into that, and I think Jeff will get into it later, the various scenarios that can lead you there. But it's considered an event progression.

MEMBER DIMITRIJEVIC: I have one more comment on this because, okay, so you assume you don't have DC power because it's not safe. But that's additional initiating event. Because loss of DC is an initiating event.

MR. SCHMIDT: No.

MEMBER DIMITRIJEVIC: So you're analyzing two initiating events.

MR. SCHMIDT: No, if it's not class power, we don't credit it. Just it's a --

MEMBER DIMITRIJEVIC: But then you assume the valves are a challenge to open.

MEMBER CORRADINI: No, it doesn't matter how it -- I think what they're trying to say is it doesn't matter how it opens. It opens.

MR. SCHMIDT: Yes.

MEMBER CORRADINI: By how it opens, that's not --

MEMBER DIMITRIJEVIC: No, it's important --

MEMBER CORRADINI: -- discussed.

MEMBER DIMITRIJEVIC: -- for timing because they have to need to -- those valves to open in a very short time. So that's important for you to have a problem with CHF, right? It has to happen in short time.

MR. SCHMIDT: Yes, we need to block.

MR. NOLAN: But the loss of power is not considered as a failure or a single failure initiating event. It's not safety related, so it's not credited in Chapter 15.

MEMBER DIMITRIJEVIC: No, I understand why you don't credit that. I just want to tell you the loss of DC power, they will have an initiating event even if the valves didn't open you had mentioned.

MR. NOLAN: That event would be bound already by 15.6.6 if you were to take a single failure of an IAB.

MR. SCHMIDT: That's right. All the valves would move -- the power main valves would move to the open position and then we would take a single failure of the active component being one IAB on one of those valves. That is effectively the 15.6.6.

MEMBER DIMITRIJEVIC: Okay. Why would those valves move to that position?

MR. SCHMIDT: Because you've lost DC power to the --

MEMBER DIMITRIJEVIC: That's an initiating event. You understand if you're operating without DC power, you will not be able to open it. That's an initiating event. It's an initiating event in the PRA. It's just an initiating event.

MEMBER CORRADINI: But I think what --

MR. NOLAN: It's a separate event.

MEMBER CORRADINI: But I think what staff is saying, at least what they're saying back to you is that's actually bounded by 15.6.6. What you're asking for essentially is less severe than what 15.6.6 is.

MEMBER DIMITRIJEVIC: No, I am just saying they're looking at two initiating events.

CHAIR MARCH-LEUBA: Yeah, but --

MEMBER DIMITRIJEVIC: They're not starting at power because if they don't have DC power, they're not at power.

CHAIR MARCH-LEUBA: No, but that's not --

MEMBER DIMITRIJEVIC: They're assuming

they start at power and open valve and then they say, I don't have DC. But if they don't DC, they have never been at power.

CHAIR MARCH-LEUBA: No, but the problem is this is not considered as initiating because those are the rules of the analysis. If you don't credit something, you have to analyze it with and without and see what happens. So it's not initiating, and you're doing two analyses.

MEMBER DIMITRIJEVIC: No, no. But they want everything to happen in a couple of seconds.

CHAIR MARCH-LEUBA: No, but these are the rules. If you don't credit it --

MEMBER CORRADINI: It's the deterministic rules they must follow under the current regulations.

CHAIR MARCH-LEUBA: You analyze it with power and without power and pick the worst. That means not creating it.

MEMBER CORRADINI: That's right.

CHAIR MARCH-LEUBA: And you know that without power is the worst, so that's the only one you analyze.

MEMBER CORRADINI: Right.

CHAIR MARCH-LEUBA: But I'm with her. That's really close.

MR. NOLAN: So the path forward on this, currently, the Commission is deliberating on a staff paper and the staff will implement the direction once the SRM is received.

MR. SCHMIDT: Okay. So we're going to go back over the GDC exemption regarding the return to power analysis. So just a highlight of the GDC 27, there's really kind of two embedded exemptions in here. One is that, as pointed out yesterday, no poison addition to the emergency core cooling system. And then the more difficult aspect of it was reliably controlling reactivity.

Staff put the position in the pre-application GDC 27 letter that reliably controlling reactivity in GDC 27 means shutdown is the final state when considering the totality of the NRC regulations regarding reactivity control.

Following the initial shutdown, the NuScale reactor can return to power and maintain criticality during a cool down on safety related passive heat removal systems, decay heat removal and ECCS which we'll get into detail this afternoon under certain conditions. NuScale submitted an exemption to GDC 27 and requested approval of a principle design criteria, PDC 27. Exemption evaluation includes the lack of ECCS which was --

MEMBER CORRADINI: But as we had discussed yesterday, this is based on what they view -- they postulate as a conservative bounding set of circumstances. Is that a fair characterization or unfair?

MR. SCHMIDT: I think it's fair to say that the DCA as is, is bounding for end of cycle events. And we'll go into detail --

MEMBER CORRADINI: Okay, fine.

MR. SCHMIDT: -- a lot of detail in the afternoon.

MEMBER CORRADINI: Okay.

MR. SCHMIDT: But the BOC one where you have a potential redistribution of boron is still ongoing.

MEMBER CORRADINI: Okay.

MR. SCHMIDT: So the SECY for the GDC 27, 18-0099, used the following three criteria. And we were kind of -- when we were talking about the IAB, we were saying you can't necessarily switch to a postulated accident criteria easily because this SECY here says that if you do your return to power analysis, you shall not violate an AOO type criteria. So it's the complication of using a postulated accident and this SECY that makes it more difficult.

CHAIR MARCH-LEUBA: But this would not be a postulated accident. This would be an AOO because this test we just done in a scram.

MR. SCHMIDT: But the criteria I thought you were referring to was saying, allowing CHF to be violated for two valve openings which would be typically a loss of fuel integrity relative to GDC 10.

CHAIR MARCH-LEUBA: I will move it to take about less than 2,200 just to the CHF.

MR. SCHMIDT: But then you're talking about fuel failures.

CHAIR MARCH-LEUBA: Correct.

MR. SCHMIDT: Right.

CHAIR MARCH-LEUBA: For if the frequency of the initiating event is low enough and I would add on the single failure, I mean, or both. I would have to think about it. Then I can relax. It's not an AOO anymore whereas the return to power is an AOO. It's a loss of offsite power.

MR. SCHMIDT: But it's important to realize that the return to power can happen either from an AOO or a postulated accident. So if you were to take --

CHAIR MARCH-LEUBA: In a scram.

MR. SCHMIDT: -- 15.6.6 -- so if you took 15.6.6, for example, and made that a postulated accident and had a fuel clad failure, then you would not be meeting the first bullet of this SECY --

CHAIR MARCH-LEUBA: Oh, what you're saying is --

MR. SCHMIDT: -- as a complication.

CHAIR MARCH-LEUBA: What you're saying is by the AOO rules, any of the AOOs, any and all of them convert into this one because you cannot create the AC power, you cannot create the DC power, and you cannot create any operation. So you don't need to lose AC power. You cannot create it. So any scram, even the end of cycle scram when the operator goes in and turns the key off and goes home becomes this by AOO rules.

MR. SCHMIDT: This criteria, if you're returning to power in that scenario, then this criteria applies.

CHAIR MARCH-LEUBA: Or all the scrams, if you do it at end of cycle and the right MTCs and all this --

MR. SCHMIDT: So --

CHAIR MARCH-LEUBA: -- it becomes this.

MR. SCHMIDT: Okay. I think you have to separate, though, a little bit. This is for an upset condition, right? This is a transient condition. Normally, when you, like, trip the reactor at end of cycle, there are all your other systems available to control the cool down, like, the secondary side.

CHAIR MARCH-LEUBA: Not under AOO rules. We just went through that. You just told me a moment ago that AOO rules you should assume that AC power is not there because it's not created. You should assume DC power is not there.

MR. SCHMIDT: On --

CHAIR MARCH-LEUBA: It's not created.

MR. SCHMIDT: -- upset conditions or transience, not as part of planned normal power operation.

CHAIR MARCH-LEUBA: I have a scram for -- I have a spurious scram, electrical. I have a ground problem and it just trips it. Now my condition --

MR. SCHMIDT: So that --

CHAIR MARCH-LEUBA: -- AC power doesn't exist. DC power doesn't exist. Operator is gone.

MR. SCHMIDT: But that is an -- so that's a nonplanned upset condition, right?

CHAIR MARCH-LEUBA: That's a scram because of any AOO scram cause me to scram. Any AOO in Chapter 15 cause me to scram. Now my condition is no

AC power created, no DC power. The operator is somewhere on vacation. And therefore, towards the end of cycle, this will happen to every one of the AOOs. And I think that's abusive.

MS. KARAS: I think just to be clear, it can. I has the potential to, and I think that's how we're treating it as an event progression, right? So you have an initiating event. Pick your AOO or accident from Chapter 15, right? Those are evaluated under Chapter 15 rules over a 72-hour period, right? So you have the initial short-term portion of the event, right? And those are all evaluated to CHF including the PAs under the NuScale framework, right?

Then if you also had the stuck rod which I need to assume just like I assume that I don't have nonsafety power and all that, right?

CHAIR MARCH-LEUBA: That's a given.

MS. KARAS: Then over some period of time as that progresses, depending on certain other factors like decay heat and some other things that Jeff will get into. Over some period of time, you would expect there's a potential return to power because that stuck rod exists, right?

So it's treated as an event progression, essentially as one event. But the way that NuScale has dealt with it is they've just pulled out the return to power portion and tried to conservatively bound that, right? But in terms of licensing space, right, it's an event progression from any one of those.

CHAIR MARCH-LEUBA: So what you're saying is we treat return to power as the possibility after any scram because we have to assume that a scram fails one rod. And we cannot take care for AC or DC power or operator action. So if certain initial conditions are applied which is toward end of cycle with very little boron and hopefully low decay heat, this applies to an AOO. So to this particular event, we need to apply the AOO rules. Is that what you're saying? That the AOO criteria, namely CHF, applies to return to power.

MS. KARAS: That's correct. But I mean, there's a nuance. The reason the -- it is because it could apply to any event. But it's also when you think about it in terms of the event progression, if you use -- like, say, you had an accident, right? If you were to use the accident criteria, right? And then I have a subsequent return to power. Now I have a return to power with previously failed fuel existing. And so that's another rationale why the

SECY paper, why that was in there that way.

CHAIR MARCH-LEUBA: I'm not complaining about this one at all. This makes sense. I mean, sure, if you have the LOCA and then you had a return to power, you may be able to relax. But you will have to analysis it for the normal scram. So you have to analyze it for the most restrictive one with the most restrictive criteria.

MR. SCHMIDT: That's right. So if a postulated accident leads to failure like Becky said, we would have preexisting fuel failure on the return to power and that would have to be analyzed. That additional -- it gets complicated because your failed fuel, you don't know what state that failed fuel is in exactly. So then you have to somehow try to evaluate that on the return to power.

CHAIR MARCH-LEUBA: Well, let's talk LOCA which actually fail fuel. But let's assume you have a LOCA where you fail some fuel. A return to power can happen with the LOCA.

MR. SCHMIDT: Right. That's what I'm saying. I think -- yes, and that's the problem. That's the problem by changing the --

CHAIR MARCH-LEUBA: But this exception would not be satisfied. So what do we do?

MR. SCHMIDT: That's why we limit local to CHF. That's why we don't -- that's what I'm trying to say. We don't allow fuel failure because we would run into this issue.

CHAIR MARCH-LEUBA: Oh, so this is a limitation for everything?

MR. SCHMIDT: Yes.

CHAIR MARCH-LEUBA: Okay. That's --

MR. SCHMIDT: That's --

CHAIR MARCH-LEUBA: -- news to me.

MR. SCHMIDT: So that's what I'm trying to communicate is that if you go down the option where you fail fuel, let's say any Chapter 15 event or any scram, you would run into this complication of this first bullet.

CHAIR MARCH-LEUBA: This is big news to me, and I assume most of my colleagues. If we offer NuScale an exception for GDC 27, every single postulated accident must meet CHF.

MR. SCHMIDT: Right. And that's what it does now.

CHAIR MARCH-LEUBA: That must be SAFDLs.

(Simultaneous speaking.)

MR. SCHMIDT: All events of Chapter 15 have to meet --

CHAIR MARCH-LEUBA: SAFDLs.

MR. SCHMIDT: Yeah, thank you.

MS. KARAS: But that's what their existing -- their existing submittal does use SAFDLs as a criteria for all of them.

MR. SCHMIDT: Yes.

DR. SCHULTZ: And just to be clear, Jeff, we're talking about fuel failure due to the event. If there is failed fuel due to a normal operation, that's okay.

MR. SCHMIDT: That's okay, yes. That's --

DR. SCHULTZ: This is --

MR. SCHMIDT: -- part of the normal aspects, yes.

DR. SCHULTZ: But the demonstration that NuScale has made is that there is no calculated fuel failure in the events in Chapter 15?

MR. SCHMIDT: That's correct.

CHAIR MARCH-LEUBA: That's big news to me and I know Ryan said anything to you too.

MEMBER CORRADINI: Well, I want to make sure I got it right. That means the judgment of whether I meet the SAFDLs or the judgment that I fail fuel is going through CHF.

MR. SCHMIDT: That's right. That's right.

That's why when we showed the acceptance criteria slides, it was collapsed liquid level and CHR for LOCA. We talked about the 50.46, but they're actually more restrictive or conservative or however you want to say it relative to those criteria.

CHAIR MARCH-LEUBA: And the applicant understands this?

MEMBER CORRADINI: Well, I think they're --

MR. SCHMIDT: They submitted it.

MS. KARAS: Just to be clear, they submitted it with the CHF criteria before, I think, the whole return to power discussions, right? This was their initial choice, right, for a variety of reasons. When the return to power issue came, that basically then created another reason why you would need to have CHF as the criteria under this review.

CHAIR MARCH-LEUBA: And then a corollary to my question. Do the commissioners understand what they're doing when they approved GDC 27?

MEMBER CORRADINI: You'll have to ask the commissioners.

MR. SCHMIDT: They -- so it was --

CHAIR MARCH-LEUBA: Have you communicated properly --

MR. SCHMIDT: -- an INPO paper, and this is the SECY number.

CHAIR MARCH-LEUBA: Right. But sometimes you put three pages of information when a bullet will suffice.

MR. SCHMIDT: I guess I can't really speak --

CHAIR MARCH-LEUBA: I've seen --

MR. SCHMIDT: -- to that.

(Simultaneous speaking.)

MEMBER CORRADINI: Ask the commissioners.

CHAIR MARCH-LEUBA: We have written a letter on GDC 27, and I didn't know it.

MEMBER DIMITRIJEVIC: I didn't know it. I'm not the exception to that. This is the first time I understand the GDC letter.

CHAIR MARCH-LEUBA: Communication -- I often tell my wife -- works both ways. Just because you told me doesn't mean I heard it.

(Laughter.)

MEMBER CORRADINI: And I think in the plant's control room, they call it three-way communication.

CHAIR MARCH-LEUBA: Yes, that's what the Navy always confirms, right?

MEMBER CORRADINI: Moving on.

MR. SCHMIDT: Well, that was not what I thought.

MEMBER CORRADINI: That was pleasant.

MR. SCHMIDT: The other two bullets -- the third one, you guys added. The second on is a combination of circumstances, conditions. Basically, a return to power isn't expected in the lifetime of the plant or the module -- I'm sorry, the module. And the incremental risk of public health and safety from a hypothetical or hypothesized return to critical. A NuScale facility with multiple modules does not adversely erode the margin between the Commission's goals for new reactor designs related to the estimated frequencies of core damage or large early releases and those calculated for the NuScale design.

CHAIR MARCH-LEUBA: Okay. But let me go back to this because it's a new concept. We're still -- maybe we should think about it first. But let's say I have a transient like opening two ECCS valves that violates CHF on the first few seconds of the transient. But it does not damage fuel.

MEMBER CORRADINI: So no way to know that. There's no way to know that.

CHAIR MARCH-LEUBA: So if you didn't hit

--

MEMBER CORRADINI: I mean, but based on -- I'm back to what they defined as failed. Fail is past CHF.

MR. SCHMIDT: Right. That's right.

MEMBER CORRADINI: So I don't think that's the case because I don't fail CHF and hold there as if it's a LOCA. I fail CHF and I pop back up. So you could make the argument -- nobody has made it. But you could make the argument that it's very similar to the KATHY test that I essentially have a rewet.

CHAIR MARCH-LEUBA: Yeah, that's the argument I'm making.

MR. SCHMIDT: And that is effectively under Option 2 of the IAB SECY.

CHAIR MARCH-LEUBA: So I would have a temporary CHF and rewet. They'll have to figure out how to demonstrate that my fuel didn't pop open. And then I have criticality and the fuel was okay from the beginning and it didn't go bad afterwards.

MR. SCHMIDT: That's correct.

CHAIR MARCH-LEUBA: That would be okay?

MR. SCHMIDT: Yes.

CHAIR MARCH-LEUBA: But they will have to -- so we have to figure out how -- CHF is the cheap and easy way to demonstrate no fuel failure.

MR. SCHMIDT: That is correct.

CHAIR MARCH-LEUBA: There is a more advanced method. I assume PCT less than 2,200 is a little obvious here.

MR. SCHMIDT: Yes.

CHAIR MARCH-LEUBA: And that's too much.

MEMBER CORRADINI: But I'm not sure what we're allowed to say in open session. But that isn't what was identified as the KATHY test as their criterion. It wasn't 2,200.

MR. SCHMIDT: No, it's CHF is their success criteria. Is that what you're saying?

MEMBER CORRADINI: But in terms of the testing, they saw CHF in rewet and continual heat.

MR. SCHMIDT: Right, but that is not submitted for --

MEMBER CORRADINI: I understand.

MR. SCHMIDT: -- the elimination of our typical CHF as a failure mode. And that is if they want to bring that data.

MEMBER CORRADINI: I'm not arguing with that. I'm just trying to go further as to that's a path. And you're saying that's a potential path within your Option 2?

MR. SCHMIDT: Yes.

MEMBER CORRADINI: Okay, fine.

MR. LINGENFELTER: Andy Lingenfelter, Fuel Engineering from NuScale. Just to sort of reemphasize something Becky said earlier and that is when NuScale submitted this application, right, we had to deal with LOCA and non-LOCA analysis. The LOCA submittal, the topical, right, is based on Appendix K which is very conservative. Part of that conservatism is associated with how you treat -- actually how you measure or calculate CHF, right?

And then as you've already heard, it's based on an extended Hench-Levy basis which as we've tracked it through the LOCA analysis through the figures of merit, as you said, it's a surrogate for we don't have fuel failures. Okay? And for the non-LOCA piece of it, since it's an AOO, the KATHY material that we've submitted is based on full scale testing of the assemblies.

And so just to keep things separated in your own mind, it's hard because we're talking about CHF in sort of two different contexts, accidents and AOO. But it's both CHF. Okay? And we're using Appendix K to help keep that figure of merit focused on an accident basis. So I don't know if that helps, but that's -- I think that's what Becky said. I just want to clarify it a little bit. And we don't fail fuel in any case.

MS. KARAS: Right. And I think as Jeff mentioned, I mean, there are other options. Under Option 2 where there could've been, say, revisions to methodologies and things to look at that for the two-valve opening event, that was part of potential options.

CHAIR MARCH-LEUBA: But the problem is I see this almost as a paperwork problem.

MEMBER CORRADINI: No.

CHAIR MARCH-LEUBA: That's the way I see it because --

MEMBER CORRADINI: Well, okay, fine.

CHAIR MARCH-LEUBA: And it's forcing us that the only available solution is to dictate. Speaking to that, that IAB -- a second IAB failure cannot possibly happen whereas we could do an analysis that shows that the second IAB failure doesn't really do anything. And it would be much more satisfying.

MR. SCHMIDT: So we're not dictating the path forward. We're setting the acceptance criteria.

CHAIR MARCH-LEUBA: Right.

MR. SCHMIDT: Those are two different

things.

CHAIR MARCH-LEUBA: But it almost drives to that solution.

MR. SCHMIDT: Actually I think if I recall right, in SRP 4.2, it says we use CHF as a figure of merit for fuel failure. If you want to come in and justify something else, that's acceptable.

CHAIR MARCH-LEUBA: We can move on, but --

MEMBER CORRADINI: But I think what Jose is thinking of which I would not disagree with is that if the committee in a potential letter wants to suggest an Option 2 pathway, that is a possibility.

CHAIR MARCH-LEUBA: Yeah, definitely the committee will discuss that in July because I will bring it up and we'll see where it goes. But I just don't see where the problem is. Honestly, I don't see where the problem is because a temporarily different CHF with no overheating of the clad.

MR. SCHMIDT: So I mean, that's, I guess, engineering judgment evaluation on your part.

MR. NOLAN: And for that transient, a regulatory finding would be that they meet GDC 10 and the criteria that's currently set for the SAFDLs would not be met absent additional analysis.

MS. KARAS: I mean, that was certainly an option where they could submit something. But I think what Jeff and my staff are trying to say, we don't dictate to the applicant which you have to follow. We don't consult with them on which path to follow, right? That's --

MEMBER CORRADINI: That's their problem.

MS. KARAS: -- their choice. And we'll review what comes.

CHAIR MARCH-LEUBA: From the point of view of an interested member of the public, I would like to know what the consequences are of opening two valves. And are we really sure that opening two valves doesn't really cause any problems? It violates an artificial surrogate. There is a surrogate for fuel failure of CHF. But we know we are marching past that and they explain the experimental theories of that.

MR. THURSTON: So you are correct. And we have through sensitivity studies looked at these case, of course. And the level will come back, right, because you have the water still in containment. So it will come back. But they have not submitted --

(Simultaneous speaking.)

CHAIR MARCH-LEUBA: You have run --

MR. THURSTON: -- to testify that.

CHAIR MARCH-LEUBA: You have run this

calculation, correct?

MR. THURSTON: Yes.

CHAIR MARCH-LEUBA: And how much does the clad overheat during that short --

MR. THURSTON: No, no. You don't get -- well, so they have a hard stop on CHF. There is a mechanism to turn off the hard stop and see what happens. So we have done that too.

CHAIR MARCH-LEUBA: You have done it?

MR. THURSTON: Right.

MEMBER CORRADINI: We can talk about that in closed session.

MR. RAD: This is Zach Rad from NuScale. I'm the director of reg affairs over there. A couple of topics. Your interest in the opening of multiple ECCS valves is analyzed in Chapter 19 of our application. So that is public.

Secondarily, going back to the options paper, there's more to -- there are further implications of treating that particular component as active relative to the single failure criteria. And it's not just this single event, although it is the one of interest right now. It would also mean assuming that addition active failure, at least considering it in every sequence that's analyzed. And so there's --

CHAIR MARCH-LEUBA: Every single AOO will become in an opening of ECCS plus a little there -- a little more.

MR. RAD: And each -- you're right. Each accident sequence as well. So there's significant implications to that relative to the amount of work that would need to be done to address it.

And then lastly on the overcooling return to power, we are working diligently with the staff to come to alignment on a final solution there. But our original submittal considered the SAFDLs during the overcooling return to power event only. And so it actually stated following a return to power, the SAFDLs will not be violated. And so that's the point of contention right now that we're working through.

CHAIR MARCH-LEUBA: And so the staff disagrees with you on your submittal?

MEMBER CORRADINI: They're still discussing it.

MR. RAD: So we're still working to come to a final alignment, especially in the wording of the PDC itself.

CHAIR MARCH-LEUBA: Is this one of those unusual open items?

MEMBER CORRADINI: I think we're about to get there.

MR. SCHMIDT: I don't know if --

MEMBER CORRADINI: Is it the next slide?

MR. SCHMIDT: -- I'd classify it as that. Yeah, next slide. I don't want to go through this PDC actually because we're -- as Zach just mentioned, we're in the process of redoing this. This is what exists currently. It's likely not to exist in the future.

So staff noted that the PDC, our problem was it didn't really apply the SAFDLs to the case where you have a stuck rod. So that was kind of the reason why we wrote an RAI that we had trouble with the original PDC. But we're working with them to resolve that.

So now we're going to get to return to power scenarios. So there are three potential return to power scenarios or three systems. So decay heat removal cool down with the DC power, decay heat removal cool down without DC power which would then actuate the ECCS at the IAB set point, can occur for most Chapter 15 AOOs and postulated accident. They all lead there as we kind of just discussed.

Key assumptions in the return to power

analysis is there's no operator actions. Only safety related equipment is used to mitigate the event, and it stems the worst stuck rod with the current GDCs, consistent with the current GDCs.

So this is the assumptions, and I'm going to go through these pretty quickly, of the current rev 2 return to power analysis which is basically a cool down on the decay heat removal system and then opening of the ECCS valves at the maximum return to power. So it's a stylized event as NuScale said yesterday.

It has two trains in service, lowest reactor building pool temperature, minimum shutdown margin, conservative peaking factor, conservative EOC, MTC analysis methods. So the methodology for return to power is kind of in 15.0.6. But it borrows from a lot of the topical reports, so it uses the inadvertent operation of the ECCS methodology in 15.6.6 which is that inadvertent opening we've been talking about.

Core modeling and nodalization are consistent with the LOCA topical report, uses, in this case, RELAP Version 1.4. MCHF determined using CHF correlations consistent with the LOCA analysis which we were talking about already, the Hench-Levy.

Results from that analysis, maximum return to power is about ten percent on a core-wide basis.

The equilibrium value, as it says, it's approximately 2.5 percent, and CHF was met through the whole transient.

MEMBER CORRADINI: So go a little bit slower. I'm sorry. I wasn't paying close enough attention. I apologize. And this is the analysis submitted that we have?

MR. SCHMIDT: This is DCD rev 2.

MEMBER CORRADINI: Okay, fine. And in terms of the three conditions, DHRS cool down with DC power, DHRS cool down without DC power, and then a 24-hour initiation and ECCS cool down. Which one are you referring to in the results? They're all about the same?

MR. SCHMIDT: So this is somewhat confusing. This is a stylized event to try to bound all of those scenarios, all of those three methods to get to potential return to power.

MEMBER CORRADINI: So is this the 9.4.4.4 response? That's what I'm --

MR. SCHMIDT: No.

MEMBER CORRADINI: -- potentially trying to understand what I look at to make sure I understand it.

MR. SCHMIDT: So the 9.4 -- I think I can speak to it in generalities. We're going to over that in --

MEMBER CORRADINI: You can just tell us to wait too. Okay.

MR. SCHMIDT: Well, I think it's safe to say that transient that's reflected in that, it starts with this analysis. It's better -- so I have it diagramed out --

MEMBER CORRADINI: Fine.

MR. SCHMIDT: -- in closed. It's probably easier.

MEMBER CORRADINI: Okay, fine. So this is intended to kind of envelope all the various ways --

(Simultaneous speaking.)

MR. SCHMIDT: -- to envelope all the various ways to get -- yeah. So when things get complicated, we like to simplify. And this is --

MEMBER CORRADINI: Understood.

MR. SCHMIDT: -- to simplify.

MEMBER CORRADINI: So now let me try one on you. I'm not sure if you were in the room late last night when we were asking the question. Is there a way to do a best estimate to compare to, what is this, a conservative enveloping?

MR. SCHMIDT: So you will see a lot of

that this afternoon.

MEMBER CORRADINI: Okay, fine.

MR. SCHMIDT: DCA does not adequate evaluate a return to power when water level drops below the riser. That was an RAI that staff had that the current rev 2 analysis we were just referring to on the previous slide was run to preserve the riser -- water level above the riser and maximize the cool down.

It became aware to the staff that in some situations coming from a higher power level and different RCS inventory that you could lose the risers. So we asked what happens on the loss of riser? That's really that. And we'll be talking about that this afternoon in detail again.

DCA does not adequately address a potential return to power following a postulated rod ejection. That is, I guess, what we're calling one of these open items that are difficult with the --

MEMBER CORRADINI: Uncommon.

MR. SCHMIDT: Thank you. So that's a public meeting scheduled to address the path forward. So --

MEMBER CORRADINI: I'm sorry. What are you --

MR. SCHMIDT: So this is -- this potential transient is a rod ejection where you now have lost one control rod during the ejection. And then because it's still a -- because rod ejection could be considered a postulated event or a postulated accident -- I'm sorry, postulated accident, then GDC 27 still applies where I also have to take a stuck rod. So I lose two instead of one. So two rods instead of one. So your margin that your shut down changes and your return to power value changes.

The analysis you saw that we just calling bounding in the DCA and what we're going to talk about at length in the afternoon only effectively has one stuck rod. So this is --

CHAIR MARCH-LEUBA: You have not analyzed two stuck rods?

MR. SCHMIDT: We have not.

CHAIR MARCH-LEUBA: But fortunately, the worth of the second rod is not as bad as the first one. So --

MR. SCHMIDT: It's not. So that is a true statement.

CHAIR MARCH-LEUBA: That's an extreme.

MR. SCHMIDT: But it's not that far away.

CHAIR MARCH-LEUBA: A single rod is 5,000 pcm.

MR. SCHMIDT: And then the second one you look at the worth of that one too and it's --

CHAIR MARCH-LEUBA: The worth that was left is another 5,000 pcm?

MR. SCHMIDT: I'm not saying it's 5,000 pcm. I'm saying it's less than 5,000 pcm.

CHAIR MARCH-LEUBA: Okay.

(Simultaneous speaking.)

MR. SCHMIDT: But it may not be as well as you're thinking. We haven't addressed this. We haven't --

MEMBER CORRADINI: This is the RAI 9647?

MR. SCHMIDT: 9647?

MEMBER CORRADINI: I'm tracking all of these things because I can't keep track.

MR. SCHMIDT: Thank you.

MS. KARAS: Yes, it's 9647.

MR. SCHMIDT: You're not the only one.

CHAIR MARCH-LEUBA: What I don't understand is you were sticking for the rules a moment ago. And now you don't want to follow the rules because the rules are clear. You should scram with one failed rod. I mean, that's what --

MR. SCHMIDT: So that's what this -- I'm

not sure exactly what you're saying. So are you saying that --

CHAIR MARCH-LEUBA: That you should run the two stuck rods.

MR. SCHMIDT: Okay. And that's --

CHAIR MARCH-LEUBA: If you follow the rules.

MR. SCHMIDT: And that's why -- so that's why we're having the public meeting.

CHAIR MARCH-LEUBA: What forum does it have? I mean, a public meeting is a conference call with the applicant when people can call in?

MR. SCHMIDT: Yes. I think NuScale is preparing information for us for the public meeting. DCA does not adequately address ECCS return to power. So there's a statement in the current rev 2 that decay heat greater than 100 kilowatts, that you have to have decay heat less than 100 kilowatts for a return to power. And I think we'll talk about that in the afternoon. But the staff is not convinced that that is accurate.

So the non-EOC. So we've been referring really to just EOC values just to be clear. The events that I referred to so far are cool down on the decay heat removal system, cool down on ECCS are all end of cycle at this point where you can have a negative reactivity to drive the return to power.

So during the review, the staff identified a potential way to lose or redistribute boron in the combined RPV and CNV. And so I'm referring to this as the non-EOC ECCS return to power due to boron plate-out or redistribution outside the core.

MEMBER CORRADINI: And that connects back to RAI 8930 in terms of --

MR. SCHMIDT: Yes, from what I remember.

MEMBER CORRADINI: -- where the boron is?

MR. SCHMIDT: So did you want to talk to this one, Shanlai?

MR. LU: Okay, yeah. I follow that. Okay. All right. So I covered that boron redistribution as Jeff mentioned that NuScale did submit the cases for the EOC case return to power. We considered that EOC case was supposed to be bounding. And so staff issued RAI 8930 and then a question that so what's going to happen to the boron. And then if there is a redistribution and there is a potential, let's assume that a potential core recriticality due to the boron redistribution on top of overcooling. So what's going to happen?

So that's the question. So the question

we raise this issue I think actually three months before the DCA submittal. So it took us two rounds of RAIs until we got the attention from NuScale. But their safety analysis group, actually once they were on this topic. So within about six to nine months actually they develop a very comprehensive model. They actually presented it yesterday and from their perspective starting from almost zero and using NRELAP5 code.

So what they did was very quick and then very -- they achieved a lot of stuff there indeed. And then I was talking to one other consultant who did the initial boron precipitation, boron transport analysis for Westinghouse, the combustion engineering. It took them three or four years. And then NuScale safety analysis group did a really good job and then they moved fast.

However, that's the fact. I'm not just saying something. That's a fact. But their current analysis as it was presented to us at this point and there was still significant gap between their perspective and our perspective. And we launched significant confirmatory analysis effort based on the supporting office research, Peter Lien and Andrew and Andy.

So we were trying to quantify and figure out, do we have a concern of boron redistribution which may cause the recriticality? If it does, that's a power level. What the figure of merit is the reactivity.

So going back to where we started at that time and back to three months before the DCA, when we ask the question, where is the boron, there was no answer. Now at least the NuScale based on their analysis, say, okay, as long as there is boron and the core will never get to recritical. That's the part we have a difference there.

So if you go back to the previous slide, just I want to hit on one item there. So then for a non-EOC case, ECCS return to power due to boron plate-out. So when they did analysis at this point only focused on EOC. The first order, zero ppm boron.

So staff's concern is that if you move from EOC two hours, see, you only move two hours, two ppm, ten ppm, whatever the ppm of that movement, or one month, maybe 100 ppm, what's going to happen? I think that's the part we had -- actually had a lot of interaction with them. We actually converted it to the point that collectively we agree that whenever you have boron, your initial SIET -- I'm going to use the next slide to show you. Okay.

Here's the plot of the reactor vessel lower plenum and the core and the riser after ECCS actuation. So if you have this kind of configuration. So what's happening there, the phenomena is you have a condensate and the RVV is releasing steam boil off and the condensate has very little boron concentration.

And then so we agree once you start ECCS, once you reach their re-circulation pattern, then the water comes in, fresh water where first diluted the water right above the RRV. Then it start to come in. Well, before it reach the core, actually the core region and the riser section with -- underneath the level is going to see an increase of boron concentration. So the margin actually is improved.

CHAIR MARCH-LEUBA: Can you grab the mouse and point to where you're talking about?

MR. LU: So the condensate, I did not draw the containment here. So that's just a -- so the condensate is accumulating in the upper part of containment. So its condensate is from the condensation from the steam. Of course, the condensate has very little boric acid. So there is a water level right above the level about the -- in the downcomer.

So there is a chunk of water initially after you blowdown. Initially, you have the chunk of water here. It's still boron. But when you keep -- I have this system keep going, then this diluted water will dilute this one first. The condensate will replace that water. That water keeps going into this downcomer in the lower plume and in the riser section.

And so the initial -- I think right now collectively in both NuScale and the staff converted on the point that your initial phase of the boron transport is going to increase the core region of the boron concentration because of the boil off. And so when this chunk of water and the water above the RRV is transported to the core, of course your boron concentrations go high.

So that's the part actually we agree with NuScale, and they did their three nodes model. They capture that phenomena. The question is after that. So if I go back to the previous slide. Let's go back to the previous slide. Okay.

So when we have -- when you have this initial peak. So that really comparing with you will see a freshwater case. Let's say you have one day or one hour before. So you have a couple ppm. What's going to happen there? Then you're going to learn the occurring case as ECCS of which is one they already predicted. But just let's say you only have two ppm. What's going to happen?

So they agree that because this initial spike it may delay that one. But staff's question, here's where the staff's question comes in. You have a delay, but you still have peak overcooling. That additional feedback due to the moderated density feedback that come in were still -- if you only accumulated, let's say, you have two ppm initially. You have ten ppm, let's say, five times increase in the core. But that can be readily overcome by the overcooling additional reactivity comes in.

So how do I know the power level right before you say will be lower than you will say? We do not know. They may still become recritical. But based on yesterday's presentation, they claim as boron --

MEMBER CORRADINI: So I'm trying to listen to understand what's -- you're saying that as I perceive away from EOC and I become middle of cycle and beginning of cycle, I actually could have a return to power of a higher power level?

MR. LU: Okay. Here's -- on the first

point, I want to question here -- actually, our question here with boron is going to recritical or not.

MEMBER CORRADINI: With boron?

MR. LU: Yes. If it's only two ppm, is it going to recritical or not? And from a staff perspective very close to you will say your feedback coefficients due to MTC are introduced as sufficient additional cooling or reactivity to overcome your initial spike of the boron concentration in the core. Then if you still get to the point that you're going to have the core become recritical. That's the first point I want to communicate with the committee. Mike, you have a --

MEMBER CORRADINI: No, I'm just listening.

MR. LU: Okay. All right. So if you move from the cycle from end of cycle to the point that you're close to a certain point, then there is a point that your moderator feedback coefficients that are past the reactivity due to the overcooling may not compensate additional, that initial spike of the boron concentration.

MEMBER CORRADINI: So I guess I think of it is as a monotonically changing function. But you're saying it could not be and you want them to show it.

MR. LU: Yeah, yeah. Well, it's monotonically -- actually, I agree with the term monotonically. It's not just -- whenever you have one ppm, the boron concentration, there's no criticality. That's what they claim. All right.

So our question here is between EOC and BOC, what's going to happen? They have no analyzed. Or they analyze that basing on their conclusion is that no criticality. But if you only have two ppm, you're going to get to criticality no matter what. And that's the first point I want to say that one.

Second, I want to go back to the next slide. Okay. And let me talk about the potential phenomena which we observed or we studied the literature based on the team. We have a review team --

MEMBER CORRADINI: Are these fluid calculations that make the little arrows or is this just cartoons?

MR. LU: That's cartoon. But this part is actually the -- we actually use this one and then actually NuScale did a full power and one percent of power CFD analysis trying to figure a pattern for the full power case. So its level is now it's out of the pressure. So they gave us some insight that's not as sufficient. We just pull out a cartoon for discussion purposes. Okay. All right.

So what's going to happen here is from our perspective, even they have a little boron, they are still going to have a possibility to reach critical. But it's a just a matter of time. It may be delayed, but it's still going to happen.

Then the question here, can we identify as a boron plate-out due to volatility? What it really means here, volatility here, is that when you have the vapor generated through the core. Vapor carries a certain percentage of the boron concentration from the fluid. That's the given. So I think we agree that phenomena is given.

So what's going to happen? It takes a long time to do that. But what's going to happen? If you have the power -- return to power already, then let's assume that they set the power level at three percent, two percent, whatever, then you keep cooking this fluid.

The water, you keep having fresh condensate coming in. Then you keep plating out. So in the long run, it takes a long time. We did the calculation too. And in a couple hours, a couple days, then this chunk of the concentration inside this region. Let's assume that's fully mixed for now. Okay. We'll start to reduce. That's a phenomena that's given. It's a natural phenomenon. That is nothing we need to debate. We can debate on what kind of correlation we can use.

So we also converted on one point is if this happens and then this phenomena were to happen and this will happen will bring the reactor to critical simply because of the plate-out. On top of overcooling, you haven't cooled the ECCS because you have a pool.

CHAIR MARCH-LEUBA: But let's stipulate that if you do plate-out in a pressurizer, you're depleting boron in the core and therefore you go critical. I mean, we'll stipulate that that would happen now. What's the mechanism for plating out in the pressurizer?

MR. LU: Okay. That's a good question.

CHAIR MARCH-LEUBA: That's a sticky point.

MR. LU: Okay. But the -- and honestly, we -- do we have sufficient data to say that 100 percent? We do not. But we have -- Westinghouse has data to show they did the test of atmospheric pressure. They cook the core just like this one, but they stop around here actually. One of the plot that actually NuScale showed yesterday was that you have --

(Simultaneous speaking.)

CHAIR MARCH-LEUBA: Yeah, the one with 40 percent weight.

MR. LU: So to answer your question here is when the boron start to plate-out, the structures right above this one and from that test is coated with white stuff.

CHAIR MARCH-LEUBA: Yeah, but that test was 30, 40, 50 percent weight.

MEMBER CORRADINI: Is that open information?

MR. LU: Yeah, I think according to them, it's maybe one of the --

(Simultaneous speaking.)

MEMBER CORRADINI: Well, I just want to make sure -- I want to make sure we're -- if we're going to start talking about this, I want to make sure we go into closed session and talk about it. If it's open information, that's fine. If not, I would rather we wait and talk about that.

MR. LU: Okay.

CHAIR MARCH-LEUBA: Well, let's not talk about that. The boiling happens in the core unless you're talking about flashing a bulb. So the steam that they saw a little bit of gaseous boric acid has to bubble through all of the chimney. If it was going to plate-out, why didn't it plate-out against the liquid that it's in contact with instead of waiting until it's on the surface?

MR. LU: That's true. You have a mechanism of that one. But just think about it. At that point, the liquid on top of that. If we assume it's mixed, the liquid on top of it already also has a high concentration. The steam generated --

CHAIR MARCH-LEUBA: It does not have high concentration. They're much lower than -- and these are open numbers. I mean, if you shut down. But to start with, we're talking about very low concentration. But if you have that condition, you have 2,000 ppm. And the limit of solubility is 40,000. So we're talking five percent of solubility is very little boron in the water.

MR. LU: Right. All I'm saying is when you still have -- as you stated, you have boiling inside this region. And then when you generate the vapor in the core region and then one of the observation from the test -- actually I think it was still open. And so it will show this part is fully mixed.

MEMBER CORRADINI: But I think what Jose is -- I'm kind of with him on this. Why would it wait to go there if I've got all this water? It would seem to me it would just stay right inside of that.

CHAIR MARCH-LEUBA: It's trying to -- the water pool is filtered much like when we do the source term analysis. When the iodine goes to the pool, it just stays in the pool.

MR. LU: Your question would be valid if you have zero concentration in the riser section. You'll lose the boric acid significantly.

CHAIR MARCH-LEUBA: A hundred ppm or two hundred ppm of boron is the concentration because the solubility limit is 40,000.

MR. LU: I agree with that particular statement. But if -- let's assume that this portion is the same, as we all -- that the NuScale assume, the components of the core were still going out to the two feet level. So that volatility will remain and that part of boron will go out of the core region.

CHAIR MARCH-LEUBA: I think the only way to prove this is to run some tests. But my gut feeling is if those bubbles of steam with a little bit of boron made it to the top of the riser, they're going to make it to the containment.

MR. LU: Oh, all the steam boil off will make it to the containment. The question here, on its way and goes through and we have a very long shaft of the control of the drive system. On top of that, you have a baffle plate of the pressurizer. So when you go through that part, how much --

(Simultaneous speaking.)

MR. LU: -- will be taken out of that portion? Do we know that number? We don't. Do we anticipate that there's a deposition? Yes, based on the previous test.

So there is a mechanism to remove this boron from this part of the system. Qualitatively, we have the question. So what's going to happen to the core?

CHAIR MARCH-LEUBA: The question is how many grams are you going to remove?

(Simultaneous speaking.)

MR. LU: Another dimension here was very important. As I mentioned that when you start, I have core boiling here. And there is -- their analysis submitted to us, we have a question there was what's going to happen to the core boiling? We have the core boiling, mixing, significant mixing that we have. And we have tests to show that Westinghouse -- well, in a previous test that was done up to this level. That's what they have. So it's very well mixed. And also a very high concentration for the boron presentation test.

But now when you have the core boiling, it keeps going. Then you have another dimension, the initial pool temperature. So if you run only 140 degree which was initially conservative decay heat removal calculation for long term calculation, then the core, based on the NRELAP5 calculation, you still have significant void fraction in the core region. We can talk about a specific number during the closed session.

MEMBER CORRADINI: So I just want to make sure where we are. I want to make sure Jose has all his questions answered. But we're kind of at potentially a break point after we get past the boron redistribution.

MR. LU: Okay. So maybe then I'll summarize a few bullets there in that case so we can go have lunch. So the question here is once you have boron, the boiling stops in the core region and move to the upper part. Then you're concentrating here. And then the freshwater could get in here.

So if I perform a conservative calculation to bound this, I could get to the point that core has a significant boron dilution. And it depends on if I divide the core region and the riser section into two parts. So that's another mechanism we have the question.

CHAIR MARCH-LEUBA: I'm more concerned -- are we going to have a discussion in the closed session later on?

MEMBER CORRADINI: Yes.

CHAIR MARCH-LEUBA: On this?

MR. LU: We have one slide, a proprietary slide prepared.

MR. SCHMIDT: Yeah, and you'll see a boron redistribution run in the closed session. It tries to work the problem backwards. How much -- if I lose a certain amount of ppm, is it safety significant? What happens? And you will see that in the --

CHAIR MARCH-LEUBA: How many ppms will I have to lose before I have to worry? Is that what you're saying?

MR. SCHMIDT: It's in that vein, yes.

CHAIR MARCH-LEUBA: That's a good question. That's a good answer to have. If I can lose 50 percent of my boron and still nothing happens, then I don't worry about it.

MR. LU: Yeah, we did --

(Simultaneous speaking.)

MR. LU: We actually did that calculation --

(Simultaneous speaking.)

CHAIR MARCH-LEUBA: But if it's at 0.5 percent and then I have to worry about it, then ooh.

MR. SCHMIDT: We will cover this in the closed session.

CHAIR MARCH-LEUBA: Good. Then I move that we will have lunch.

MEMBER CORRADINI: Well, wait. I think you've got another slide, don't you?

CHAIR MARCH-LEUBA: No.

MR. SCHMIDT: Two slides. So we're going to talk about this in the closed session a lot, and this is the scope which we've done, which we kind of talked about. EOC decay heat removal cool down which is kind of like the FSAR or the DCA is today, right, rev 2. And then we have ECCS EOC cool down. Again, both the first two are with no boron to try to maximize return to power with a defect. And then we have a BOC ECCS with and without biasing and a positive reactivity inserted that would mimic boron plate-out and loss.

CHAIR MARCH-LEUBA: And BOC, you have over 1,000 ppm.

MR. SCHMIDT: BOC, you'll see it in the closed.

CHAIR MARCH-LEUBA: At beginning of cycle, you should be operating over -- we'll see the results?

MR. SCHMIDT: Yeah.

CHAIR MARCH-LEUBA: During a big positive --

MR. SCHMIDT: In the closed session.

CHAIR MARCH-LEUBA: Okay. Nothing like a good calculation.

MR. SCHMIDT: And then following a long-term cooling analysis, I'm going to go quickly through that too. Focused on inventory reduction terms, so you've lost inventory for the ECCS system. I think the important point here is the long-term analysis assumes that you are shut down. So if you're not shut down, it goes to 15.0.6. So there's, like, a break point you've got to keep in mind when you're looking at these analyses.

Evaluates out to 72 hours. Figures of merit were discussed yesterday, decreasing clad temperature, minimum RCS to prevent boron precipitation, collapse liquid level above the top of the active fuel. Staff focused on validation of NIST test data, RELAP model, and assumed analysis conditions.

CHAIR MARCH-LEUBA: Have we made a bounding calculation by hand how long does is the ultimate that this thing can survive with the return to power and the DHRS working to remove it?

MR. SCHMIDT: No.

MEMBER CORRADINI: NuScale has not done that?

(Simultaneous speaking.)

MR. SCHMIDT: Well, they have done 12 modules in decay heat.

MEMBER CORRADINI: Okay.

CHAIR MARCH-LEUBA: But one module at two and a half percent power?

MR. SCHMIDT: Right. So we'll go into the -- we haven't done that calculation. But I think you'll get a better perspective when we go through the results. And you'll see whether you think that's important or not.

MEMBER CORRADINI: Okay.

CHAIR MARCH-LEUBA: Okay.

MEMBER CORRADINI: With that, anything

else before we go into the actual events? This looks like a logical break point from my standpoint.

MR. SCHMIDT: Yes, it is.

MEMBER CORRADINI: All right. So why don't we take a break for lunch and back at 1:00 o'clock.

(Whereupon, the above-entitled matter went off the record at 12:09 p.m. and resumed at 1:02 p.m.)

MEMBER CORRADINI: Okay, let's begin. Staff was going to now start going through the Chapter 15 key events.

MS. SIWY: Okay, so the key events that we're going to cover include two decrease in coolant inventory events, LOCA and inadvertent operation of ECCS.

We'll also cover inadvertent decrease in boron concentration, feedwater line break, increase in feedwater flow, and loss of AC power.

MR. THURSTON: So, I am the reviewer for 15.6.5, LOCA.

The LOCA break spectrum compared to traditional PWRs is essentially a break so the LOCA break spectrum is limited to a two-inch CVCS line injection or discharge, high point vent for pressurizer spray.

So, there are basically two distinct features of the LOCA blowdown, Phase 1A and Phase 1B. They differ slightly, whether it's a liquid space break or a steam space break.

Phase 1A is blowdown to ECCS opening and then Phase B is ECCS opening to establish long-term ECCS recirculation.

And after that you can essentially consider that the long-term cooling will begin after the ECCS recirculation phase is established. The key figures of merit are minimum CHF and collapsed liquid level above the core.

Okay, the limiting break is a small break, 10 percent CVCS injection line break. One of the things that the Staff noticed in the Applicant's analysis was that the T ave, the incorrect T ave temperature was used so the stored energy was underestimated.

So Staff requested additional information via RAI 9841 to correct the T ave, and that's being addressed. So that essentially is going to be a confirmatory item.

MEMBER SKILLMAN: Carl, what does the ten percent refer to, please?

MR. THURSTON: It's ten percent break

area.

MEMBER SKILLMAN: Thank you.

MR. THURSTON: So, a full break spectrum is considered from 0 percent all the way up to 100 percent of a two-inch line, which is still relatively small. So, the key assumptions are loss of AC power so DC power is still available.

So, the reactor trips on high RCS pressure and there's no single failure applied. They did a single failure sensitivity study and they confirmed that no single failure is a limiting case. And DHRS is not credited for any of the LOCA events.

So, in this particular transient, it's a very small break so the blowdown continues for a pretty extended period of time.

And when the containment fills, the IAB is still not clear so the inventory continues to come out of the RCS into the containment so it maximizes the inventory loss.

And the RCS is saturated at the time and so you get kind of like a flashing behavior and that minimizes the liquid level in the core momentarily.

So, we're going to look at that when we look at the benchmarks, the confirmatory runs that the research did and factored.

So, for this particular case, the minimum liquid level was less than five inches above the top of the fuel, but like I say, it's a very momentary drop and then it recovers back up to the ten-foot essentially above the reactor core.

The limiting CHF case is a 15.6.6 analysis but they have to do CHF analysis for this event as well and it's 1.8. And the design limit for postulated accidents is 1.29.

So, what's not reflected in this analysis, we are based on DCD Rev 2 so for Rev 3 we are expecting changes to the ECCS logic. I think that has been discussed already.

So, the riser level and the ECCS actuation trip is going to be eliminated and the CVCS line actuation is going to be changed from 220 inches to 264 minimum.

So, you're looking at about an additional 40 or so inches that they're going to add to that trip.

MEMBER CORRADINI: And again, the reason for that is what?

MR. THURSTON: Because the containment level trips essentially bounds all of the cases for Chapter 15, essentially. So, there really aren't any cases where we have used the level trip.

MEMBER CORRADINI: I guess I don't appreciate --

MR. THURSTON: By increasing the level. He kind of reviewed it a little bit earlier.

MR. BRISTOL: Ben Bristol with NuScale. Yes, so for the containment analysis, we are seeing some sensitivity to the time in which ECCS actuated. We kind of covered that a little bit yesterday.

As the level increases and then ECCS actuates, that sort of sets the limiting conditions. By delaying ECCS actuation, we recovered some margin in some of the cases for the containment analysis.

So the driver for changing, increasing, the containment level actuation point wasn't for LOCA specifically. Rather, it was for the containment analysis and I think as Carl mentioned, the limiting LOCA case is actually on IAB, not on the containment level.

So, we don't expect this change to influence the limiting cases, I guess is my point.

MR. THURSTON: That was my next point.

CHAIR MARCH-LEUBA: Don't go away. Go ahead.

MEMBER SKILLMAN: Isn't that increase

conservative? That's increasing the driving head from the containment out of the core when you want ECCS.

MR. BRISTOL: Yes, so there's sort of an equilibrium point at which recirculation would establish and so this sets the containment actuation of ECCS a little bit closer to when recirculation would establish at the level at which recirculation occurs from the containment.

So, yes, it's a little bit more in line with actuating your depressurization once the level -- there's enough inventory in containment to actually establish recirculation.

So, through this change we expect to lose less liquid through the process of ECCS actuation.

MEMBER SKILLMAN: Thank you.

CHAIR MARCH-LEUBA: In this unit, tell me where the RRV is located? Is it at 210?

MR. THURSTON: The RRV is on the side of the downcomer.

MEMBER CORRADINI: No, he wants to know in inches.

MR. BRISTOL: Yes, the elevation, I would have to do a quick check on that.

CHAIR MARCH-LEUBA: Okay, because 44 inches is an awful lot of movement. It'd be nice to know -- obviously, 220 was above the RRV.

MR. THURSTON: Yes.

CHAIR MARCH-LEUBA: It would be nice to know.

MR. BRISTOL: Okay, I'll follow up with that.

CHAIR MARCH-LEUBA: Is this the ten feet above the top of the fuel?

MEMBER CORRADINI: I guess to write something down like this, I'd like to know where the top of the fuel was.

CHAIR MARCH-LEUBA: And where the RRV is. RRV is not as relevant.

MR. THURSTON: These numbers are based on the bottom of the containment, the numbers that we are showing here.

CHAIR MARCH-LEUBA: The number I remember is after we opened the ECCS everything is going level inside and out at ten feet above the top of the fuel.

What is that with respect to these inches?

MR. BRISTOL: Okay, I'll follow up with that.

MR. THURSTON: The next thing we wanted to show is some confirmatory analysis that was conducted by research.

So, the TRACE model that was presented earlier was developed from the base model of the RELAP5 deck and we've used it to generate several transients.

So, the limiting case has been executed by our research and so this shows a comparison of that table, 15.6-12 in the DCD.

So, you can see that the basic trends trend very well so this is just a table of results, and then we have some plots that we're going to show so we just move to the next.

So, the first slide shows a break flow and you can see initially there is -- so, the local model uses the very conservative Appendix K. So they have Henry-Fauske that transfers to Moody for saturated blowdown.

MEMBER CORRADINI: Is the transfer point where it goes from 15 to 5?

MR. THURSTON: Right, so this is the saturated -- right.

CHAIR MARCH-LEUBA: You need to use the mouse on top of the microphone or he won't hear you.

MR. THURSTON: So, this is subcool blowdown and then it saturates here.

MEMBER CORRADINI: What happens when it

goes down a factor of three?

MR. THURSTON: So, the ECCS is opening out over here and this is initially saturating here. So, this is kind of blowing down, and we can look at the next plot too.

CHAIR MARCH-LEUBA: Hold on a minute. At 4000 seconds, a second ECCS valve opens?

MR. THURSTON: No, the second ECCS valves open out here at about 6000 seconds.

CHAIR MARCH-LEUBA: So, what happened at 4000 seconds?

MR. THURSTON: That I'll have to look...But you can see here that and Peter Lien may want to weigh in.

But there's a trend here that it rides the saturation curve and I think that corresponds to this drop.

MEMBER CORRADINI: Okay.

MR. THURSTON: Pete, do you want to weigh in?

MR. LIEN: This is Peter Lien from the Office of Research. If you look at the plot from 0 to 500 seconds, the first tick, half of the first tick, I'm speaking from TRACE results.

The red curve, the first part, 0 to 500

seconds, is a single phase. From 500 all the way to 5000 is about two phases. The whole fraction upstream of the break is about 0.2 to 0.3.

From 5000 seconds it drops because the upstream becomes a single phase steam so that void goes down. So, when two curves merge together going to that IAB opening, at that point it's only a single phase of steam.

MR. THURSTON: So, there's modest differences in the conservatism of the models and again, the Applicant uses Appendix K and the TRACE model is more best estimate calculations.

So, you're going to get slightly different -- so, as those differences augment through the transient, we still get a fairly comparison.

The ECCS opens about very close, and then we transition to long-term cooling here.

The next plot shows the primary pressure and the containment pressure and how those trends evolve. So, as the ECCS opens, of course the pressure equilibrates between the containment and the RCS. And then you just ride it out.

The next door plot shows the minimum collapsed liquid level in the core, which is this rise section, the blue line and the black line. And then the containment level rising at the green line and the red line.

CHAIR MARCH-LEUBA: Is this with the new or the old, the 220 inch or the 264 inch?

MR. THURSTON: This is with the 220 so this is Rev 2 of the DCD.

CHAIR MARCH-LEUBA: Because there is a significant delay.

MR. THURSTON: But it's riding the IAB so even though the containment level might rise to the trip set point, it'll trip but the IAB is going to close so the ECCS is still not going to open up.

CHAIR MARCH-LEUBA: Right, but the concern I have is starting at about 4000 seconds, the water level in the vessel is lower than the water vessel in the containment.

And disappointing, which I guess is due to the fuel in the blue calculation.

MR. THURSTON: Right, so it gets very close.

CHAIR MARCH-LEUBA: If you delay that actuation until you are at 260, if you delay by 44 more inches, that actuation it might hit the top of the fuel and it might uncover.

MR. THURSTON: Yes, we did studies, we

looked at varying temperatures.

One of the things we thought about was if we decrease the pool temperature and how that would affect -- and that has a small affect but it's still above the --

CHAIR MARCH-LEUBA: Now, the new logic delays the actuation of the return from containment vessel, delays the actuation of the ECCS flow by an additional 44 inches, which might be 2000 seconds.

And the trend is going towards uncovering core.

MR. THURSTON: Right, so we will --

CHAIR MARCH-LEUBA: There isn't that much margin in there to delay the actuation of the ECCS.

MR. THURSTON: Right, so that'll be in the Rev 3.

CHAIR MARCH-LEUBA: Okay, we are going to get some information.

MS. McCLOSKEY: Megan McCloskey, NuScale. If I can just hopefully clarify something. Carl talked about how when the ECCS valves open in this case, that's on the IAB release pressure.

The ECCS actuation comes much earlier in the transient response because this is a small break and per the methodology, any heat transfer through the

DHRS is not credited.

And so that's why in this case there's delay between the ECCS actuation time and when the valves actually open.

CHAIR MARCH-LEUBA: I was under the impression that the ECCS does not open unless the level is 264 inches. There is only one single going to ECCS actuation which is high level in containment.

MS. McCLOSKEY: Yes.

CHAIR MARCH-LEUBA: So, until you have high level in containment at 264 inches it won't open, whether the pressure is pressurized or not. By the time it opens, the pressure is nothing. IAB is not holding it.

MEMBER CORRADINI: Can we go back to what the pressure is? The pressure's not nothing.

MR. THURSTON: So, he's looking at the delta pressure, right? The valve is looking at a pressure between the containment and the reactor vessel.

MEMBER CORRADINI: But I'm waiting for somebody to give me -- can you go back now? The answer is you're achieving a steady state. By changing the set point they essentially are guesstimating where the steady state is.

So, when they open the valve I'm not going to see a big dip. That's what I would expect to be the case if I delay it by a level.

Because I'm coming to essentially zero slope before I open the valve..

CHAIR MARCH-LEUBA: So, the big question is in this scale what is the actuation time for the ECCS? When do you de-energize the solenoid even though IAB --

MS. McCLOSKEY: Carl, I think you have that in the sequence of events.

MR. THURSTON: Yes, we do. So, low pressurizer, low level. It has the containment level here and the RELAP case is 2200 seconds.

CHAIR MARCH-LEUBA: So, you have dumped sufficient water in containment to almost uncover the core and your IAB is not preventing you from opening?

MS. McCLOSKEY: No, you have not almost uncovered the core.

(Simultaneous speaking.)

MR. THURSTON: So, at 2200 seconds we're up here.

CHAIR MARCH-LEUBA: Yes, but when you trip it, it's at five. When you finally trip it, it's less than five and then it hits zero.

MR. THURSTON: That's why this is a limiting case.

CHAIR MARCH-LEUBA: So, the IAB is not doing you any favors and when you have any uncertainty on that, you are not going to be able to satisfy that Appendix K requirement that you don't uncover the core.

You're asking an exception for it.

MEMBER CORRADINI: It's not uncovering the core, it's getting the minimum CHFR. It's not clear you've uncovered -- that's the collapsed liquid level.

CHAIR MARCH-LEUBA: Blue?

(Simultaneous speaking.)

MEMBER CORRADINI: That's not the two-phase level.

MR. THURSTON: The CHF is way up by this time because you're way out in time.

CHAIR MARCH-LEUBA: How much margin do we have?

MR. THURSTON: Tons.

CHAIR MARCH-LEUBA: Would you agree that the IAB is not doing us any favors in this transient?

MS. McCLOSKEY: Not for this small scenario --

(Simultaneous speaking.)

MS. McCLOSKEY: -- with the conservative methodology that we have.

MR. NOLAN: And like was previously stated, keep in mind DHRS is not credited here.

CHAIR MARCH-LEUBA: We have always failures equipment. What happens if DHRS is credited? You still lose inventory.

MR. THURSTON: It'll depressurize much faster so you won't lose as much inventory.

CHAIR MARCH-LEUBA: Oh, because you're cooling off the steam?

MR. THURSTON: Mm-hmm.

MS. McCLOSKEY: And I think the Members are aware of this, the DHRS is a single-failure-proof safety-related system. So, that's very conservative not to credit it.

MR. THURSTON: So, again, this is the limiting case. Is that our last slide? Yes.

CHAIR MARCH-LEUBA: And you have run 8 percent and 12 percent, right? Ten percent looks very arbitrary.

MR. THURSTON: Right, so want 10 percent, 20 percent, and 5 percent.

CHAIR MARCH-LEUBA: Yes, because on a small-break LOCA we always do a spectrum of breaks to find the sweet spot.

And often you can even miss the bottom if you don't do it fine enough. You convince yourself that you did it fine enough?

MR. THURSTON: Yes, we did sensitivities and know this is the worst case.

MR. LIEN: This is Peter Lien again.

With the spectrum order calculation from 2 percent all the way to 100 percent so we did confirm that, the minimum there will occur at -- in this calculation we found out that the minimum level occurs at a five percent break.

But we had found this ten percent so we do the ten percent calculation for this comparison purpose. In our spectrum, it's about five percent is the minimum level and the level we found out is 4.3 feet above active core.

This case is 4.6 so I think even a 5 percent calculation, you still have a margin there.

CHAIR MARCH-LEUBA: So, the bottom is very flat?

MR. LIEN: No, the bottom is a two percent case. We will see the minimum there will go back up so five percent is our results. But I think it's very comparable with this.

MS. SIWY: All right, moving on.

CHAIR MARCH-LEUBA: So, have you run a sensitivity on IAB set point? Because one thing is the IAB won't fail, that I can understand it will.

But the IAB opens because there is a spring on tension, I guess a piston, and that has been there for several years and you haven't calibrated it. I don't know when the last time you calibrated that, certainly now within the last --

MR. THURSTON: Right, so the lowest set point is conservative for the smaller breaks. And then for the larger breaks, 1200.

CHAIR MARCH-LEUBA: So, NuScale, please, the set point for IAB, is that tech spec value?

MR. BRISTOL: It's in the ASME design specification for the valves.

CHAIR MARCH-LEUBA: The question is, is there a surveillance requirement?

MR. BRISTOL: Yes, I believe that the IAB is tested on --

CHAIR MARCH-LEUBA: Every refueling?

MR. BRISTOL: I don't know what the refrequency is, we could follow up with that.

CHAIR MARCH-LEUBA: Because the set point, being a spring against a plate with a little binding and things like that, it's going to drift.

MR. SCARBROUGH: This is Tom Scarbrough. Yes, we have been reviewing their OM ASME code testing program. These are going to be tested every outage. They'll actually remove them.

They're bolted on now so they can take them off and test the IAB on a bench to looking for the right pressure.

Because there's no set point on these so during the factory testing in the factory they're going to set with a shim to have them move at the right pressure.

And it's a difference in pressure they're going to be comparing.

CHAIR MARCH-LEUBA: But it's not a screw.

MR. SCARBROUGH: They're factory-designed and manufactured, but they're going to test them every outage on the bench to ensure that they're operating properly.

CHAIR MARCH-LEUBA: But looking at these results, there is some sensitivity to that point of actuation --

(Simultaneous speaking.)

CHAIR MARCH-LEUBA: -- set point.

MR. SCARBROUGH: Right, and we talked

during the closed session the other day about the testing they're doing right now at Target Rock to look at where what they call engagement pressure, where it closes and the release pressure, where it opens during this part of the testing they're doing right now so we can get a feel for where that is.

CHAIR MARCH-LEUBA: And there will be some acceptance criteria when they test it when they throw it away and get a new one?

MR. SCARBROUGH: Whenever they start -- actually, first, they qualify, right, in their QME1 qualification during the COL stage and then they'll manufacture them.

And they'll show that they operate within the band that they're required to, and then every outage, when you pull them off and test them on the bench, they'll have another criteria. Make sure they stay within that band.

CHAIR MARCH-LEUBA: And they have a band and the band is consistent with the analysis they perform?

MR. SCARBROUGH: It should be. That's kind of far down the road but when they set up those criteria for in-service testing, they need to make sure they're within those bands.

CHAIR MARCH-LEUBA: Historically, we forward those things to the cooperating report or through the tech specs.

MR. SCARBROUGH: Right, and this is the IST program and the tech specs, a lot of times they reference the IST program.

So, that's what they'll be doing so every outage they'll be checking to make sure they're within the band.

CHAIR MARCH-LEUBA: My concern is I didn't expect it to be this sensitive. We're close to limits.

MR. SCARBROUGH: Well, that's good to know because we feed that back to us and we can make sure that's taken care of during our review of the IST program for the COL Applicant when they come in.

CHAIR MARCH-LEUBA: Consider yourself fed back.

MR. SCARBROUGH: Thank you.

MR. BRISTOL: Ben Bristol, just following up on the elevation of the recirc valves. If we can go back one slide? Okay, so on this slide the bottom of containment is about -10 feet there.

The recirc valves are about 8 feet above the top of the fuel, so about 18 feet in elevation in containment.

CHAIR MARCH-LEUBA: And the set point is?

MEMBER CORRADINI: 220.

CHAIR MARCH-LEUBA: 264 divided by 12 minus 10.

MR. THURSTON: Well, the new one is 264. In Rev 2 it's 220.

CHAIR MARCH-LEUBA: Right, it's going to be...

MR. THURSTON: It's going to be 264.

CHAIR MARCH-LEUBA: Which is? I need a calculator.

MR. BRISTOL: 22 feet.

CHAIR MARCH-LEUBA: 22 feet minus 12?

MR. BRISTOL: So, on this figure, it would be actuated about the 12-foot mark.

CHAIR MARCH-LEUBA: So, it's four feet above the RRV?

MR. BRISTOL: That's right, yes.

CHAIR MARCH-LEUBA: So, 220 used to be right at the RRV?

MR. BRISTOL: Yes.

CHAIR MARCH-LEUBA: That's too close to the RRV. You want it four feet above.

MR. BRISTOL: Yes.

CHAIR MARCH-LEUBA: It's unreasonable but...Please continue.

MR. NOLAN: All right, are we ready for 15.6.6, round two?

MEMBER CORRADINI: Can't wait.

MR. NOLAN: Okay, so we're presenting this transient in part because it is a limiting event for MCHFR.

And just to refresh, I guess, some of the assumptions and the first few seconds of this transient, the initiating event is one RRV inadvertently opens for whatever reason.

And at the same time, we're only taking credit for safety-related SSCs to mitigate, and so that means we would assume the DC power system is not there.

So, at time zero power is lost to the trip solenoid valves, which would then require the IABs in the other four ECCS valves to close to keep the main valves from opening.

And then the minimum CHFR occurs at half a second into the event. So, it's a very, very quick transient.

CHAIR MARCH-LEUBA: But if I remember correctly from proprietary, this is not proprietary, right? This half a second? The delta CHFR calculated by the Applicant was only like 0.1 for this actually, was because unless you couldn't see it on the –- I mean, you couldn't see the deep. Whereas the delta CHFR for the overcooling event is almost one, ten times larger. And this is the limiting MCHFR because if you miss it you get a penalty.

But the delta is much more limiting on the overcooling than here. If you put them both on the same scale starting from the same point, overcooling is ten times worse than this.

(Simultaneous speaking.)

MEMBER CORRADINI: Are we comparing the same thing? Because in one case they're using VIPRE and the NSP4 correlation, and here they're using Hench-Levy.

So, eventually, when we get around to trying to look at that, my anticipation is that Hench-Levy starts off at a lower MCHFR.

CHAIR MARCH-LEUBA: Yes, but that's an artifice of the calculation.

What I think is if one calculation gives you a degradation of your CHFR of 0.1 and another one gives me a degradation of CHFR of almost 1, you cannot say this is the limiting one because --

MEMBER CORRADINI: I see your point.

CHAIR MARCH-LEUBA: That one is worse. Overcooling is ten times worse.

MEMBER CORRADINI: But they have to look at all of them.

CHAIR MARCH-LEUBA: It would be nice if we used consistent numbers. This is by no means even close to limiting, a factor of ten off. This is not on our mind even.

We need to talk about CHF and how to report these things because it really leads to the wrong conclusions.

MR. SCHMIDT: So, if you're saying you should look at the -- it's really the most limiting value that's calculated relative to its appropriate design limit, right?

That's the one that kind of challenges its design limit. Because we have two design limits here effectively.

CHAIR MARCH-LEUBA: Which are?

MR. SCHMIDT: So, one is the Hench-Levy and its corresponding design limit and then the NSP4. So, when you say most limiting, it's a little tricky to evaluate.

I would look at the delta between what you calculate and your design limit as being the limiting one. We might be kind of wrapped around the axle of what's limiting here.

CHAIR MARCH-LEUBA: The reactors I'm used to, which are BWRs, we report these type of events as a delta CPR.

Actually, delta CPR divided by CPR because that's how it scales properly and that's why I'm using that historical background to analyze these events, analyze the CPR over CPR.

MR. SCHMIDT: And I think we could do that and give you back something like that to show.

CHAIR MARCH-LEUBA: I just cannot understand how the same reactor using one analysis method will give you an initial CHFR of 2.8.

We use a different analysis method to give you an initial CHFR of 1.5 and it's the same reactor.

MR. THURSTON: Yes, it's different methodologies, different correlations. You know that.

MR. SCHMIDT: Different modeling methodologies and correlations.

CHAIR MARCH-LEUBA: I understand that but --

MR. SCHMIDT: I think you've already given us that action to piece that up.

MR. THURSTON: He has filled out their methodology.

CHAIR MARCH-LEUBA: In my mind, with a BWR you are biased. I would do a delta CPR over CPR. The overcooling is a lot worse than this.

MEMBER CORRADINI: But they can't get to one no matter what.

(Simultaneous speaking.)

MEMBER CORRADINI: That's their limit.

DR. SCHULTZ: For comparative purposes it would be nice to have that table with the events in order to support a statement like this is the limiting event.

CHAIR MARCH-LEUBA: If you want to be quick and dirty, yes, run the overcooling event with this methodology and see what you get. This is just saying who's worse but the DCS points to the problem of what is important here.

MR. SCHMIDT: Point taken.

MR. NOLAN: I'll add that the current analysis for this event obviously does not include a single failure of the IAB. So, this is just one valve.

CHAIR MARCH-LEUBA: I'm sorry, my brain works while you are talking, I apologize.

But the VIPRE methodology is supposed to be more conservative than the RELAP methodology and the RELAP methodology seems to agree with the TRACE numbers. And TRACE I trust, RELAP I'm trying to evaluate.

MEMBER CORRADINI: Be real careful. I don't mean to interject but you're looking at CHFR ratio and we don't know what TRACE is using for that.

They probably used Biasi or CSIE or essentially the lookup tables, the Groeneveld lookup tables. So, we haven't looked at that. We've looked at -- what did you guys show us?

I think just collapsed liquid level.

MR. THURSTON: Right.

MEMBER CORRADINI: So, if you want to really get yourself wrapped around the axle, ask them what their MCHFR is starting off there with TRACE. It's probably neither two-point-something neither one-point-whatever, it's probably something else.

CHAIR MARCH-LEUBA: Don't you have the CHFR limit in there?

MR. THURSTON: We do have a CHFR limit. The limiting case for the LOCA was 1.8 but I don't know if TRACE has a method to calculate minimum CHFR.

MEMBER CORRADINI: I think it's a

division.

MR. THURSTON: Yes, it's just a division but you have to have the coding in place to do that.

MR. LIEN: In TRACE we did not calculate the ratio but we did calculate the CHFR value based on the AECL table lookup.

MEMBER CORRADINI: The lookup table, the 2006 version? The latest one?

MR. LIEN: I have to look up the manual. Yes, but it's a table lookup. But we didn't calculate the ratio.

CHAIR MARCH-LEUBA: And I promised this morning I would not raise this issue again and I forgot I had promised that.

MEMBER CORRADINI: That's okay. I think you raise a valid point about trying to make sure that we understand what you start with and how you end up.

I think what got Dr. March-Leuba upset is limiting.

CHAIR MARCH-LEUBA: You can also say professor.

MEMBER CORRADINI: Right.

CHAIR MARCH-LEUBA: It's one of those minor universities.

MEMBER CORRADINI: It's orange, that's the only issue.

CHAIR MARCH-LEUBA: That's the important thing. The point is I was surprised a little bit to see these results that I thought we had a lot of margin and the collapse level is getting close to the top of active fuel.

There is plenty of two-phase levels above it and there is plenty of margin but it's not as much margin as --

MR. THURSTON: Momentary, right. Very momentary. 1020 seconds.

MR. NOLAN: And this is a non-LOCA event?

CHAIR MARCH-LEUBA: Correct, I was still dealing with the COL presentation.

MR. NOLAN: I understand.

CHAIR MARCH-LEUBA: I told you, my brain works in delayed time, especially after lunch.

MR. NOLAN: You can go to the next slide. And so there's two open items here, placeholder open items, one related to the application of single failure and then the other because of the LOCA topical.

MEMBER CORRADINI: These are everywhere?

MR. NOLAN: Yes, that's why I kind of called it a placeholder open item.

MEMBER CORRADINI: They're universal, they're not uncommon.

MR. NOLAN: Well, specific to 15.6.6, the reason we care about 15.6.6 in the LOCA is because Appendix B is there specifically for this event.

MEMBER CORRADINI: I understand. And I guess you guys have educated us but just to repeat, the reason it is an AOO is why? Not an accident?

MR. NOLAN: Because that's how NuScale classified it.

MEMBER CORRADINI: Aha. So, they came in with that as a classification as an AOO?

MR. NOLAN: Mm-hmm.

CHAIR MARCH-LEUBA: The only way I can justify this accident happening is if the disc itself on the big valve breaks. There will have to be lots of failures for it to come from the electrical system.

MR. NOLAN: I don't know. If they want to clarify, certainly they can come in with any classifications.

MEMBER CORRADINI: I didn't realize.

CHAIR MARCH-LEUBA: And the Applicant is allowed to use a higher frequency than...

MR. NOLAN: Next, I'll be talking about 15.4.6, inadvertent decrease in boron concentration.

For the NuScale design, this primarily occurs through the demin water system makeup via the CVCS to the RCS.

The way that it's mitigated is through isolation of the safety-related demineralized supply isolation valves. And they receive an isolation signal which would terminate the event on any reactor trip and high subcritical multiplication and low RCS flow.

They used two methods for evaluating, primarily modes one through three. The first one is the perfect mixing model, so at higher flows it would assume a perfect mixing and it's conservative because it's a slower reactivity insertion and it delays detection.

The other is a wave front model which would assume sort of slug flow cycling through the core.

CHAIR MARCH-LEUBA: Do you remember what subcritical multiplication censors they used? Because that would have to be surge level. Are they outside the vessel?

Where are they, do you know?

MR. NOLAN: Do you recall?

MS. SIWY: So, I didn't look in terms of this event specifically but the source rate high count rate trip as well as subcritical multiplication.

CHAIR MARCH-LEUBA: Yes, but its needs to be able to count first. So, you need to use different detectors. Are those excore?

MS. SIWY: Yes.

CHAIR MARCH-LEUBA: And they're always active?

MS. SIWY: Source range wouldn't always be active because they'd be shut off when you reach --

CHAIR MARCH-LEUBA: So, the operator will have to turn them on?

MS. SIWY: I mean when you're going up in power there's --

CHAIR MARCH-LEUBA: No, you just had a scram.

MS. SIWY: Okay.

CHAIR MARCH-LEUBA: If it was my detector, we'd turn it off when it's approved flux because I don't want to burn it.

MS. SIWY: Right, I agree.

MR. NOLAN: Well, after a scram the isolation valves are closed so you wouldn't have a dilution event occurring.

CHAIR MARCH-LEUBA: Well, you're saying that you're taking credit for a high subcritical multiplication factor? I'm asking how does the source level detectors get enabled after a scram?

Does it require operator action?

MR. NOLAN: Well, I mean it would be when you're starting up and then it's driven by your tech spec.

CHAIR MARCH-LEUBA: When you're starting up they're on but when you're going down and then you have a -- or if you already scram...Okay, think about it.

MR. SCHMIDT: This is Jeff Schmidt. So, after the scram you can establish you're going to be shut down so then you'll approach criticality again.

So, as you approach criticality, you will have your source reigns detectors on during that time period.

CHAIR MARCH-LEUBA: This is various low transient and you assume you have established a down, mode two or three, and the source reign detectors are enabled at that time.

MR. SCHMIDT: Right.

MR. NOLAN: Typically, this transient is really looked at from the perspective of how long does it take to lose shutdown margin and can the operators detect it and take action in a timely manner?

And in this case, NuSCale doesn't credit any operator action, and so they're just relying on the safety-related isolation signals.

CHAIR MARCH-LEUBA: Order of magnitude, is it five minutes or five days?

MR. NOLAN: We'll go to the next slide. So, for mode 1 hot zero power, the dilution event is bounded by the uncontrolled CRA withdrawal. The limiting dilution event occurs in either modes two or three at a higher flow.

The reason is anything less than 763 the system is isolating. And so I believe that limiting transient is about 88 seconds -- sorry, 88 minutes.

And then I think the calculation assumes you would lose shutdown margin at 124 minutes. And so it isolates well before you --

CHAIR MARCH-LEUBA: But do you mean by mistake I turned on demineralized water to maximum flow and I turned on the return to clean up, which extracts boron from the core and it takes 120 minutes to lose shutdown margin? Roughly?

MR. NOLAN: Yes.

CHAIR MARCH-LEUBA: Okay, so it's --

MR. NOLAN: But one makeup pump. It's controlled by tech spec, how many pumps you can have in operation. And I think the pump is by design 24 GPM, the analysis assumes 25 GPM.

CHAIR MARCH-LEUBA: It's not five minutes, it's two hours.

MR. NOLAN: Yes, it's 88 minutes, most likely operators are going to realize something is occurring.

And I think you'll see similar placeholder open items in this section as well, but once those are closed there's no other issues in this area.

MEMBER CORRADINI: Thank you.

CHAIR MARCH-LEUBA: Would this have any effect if you have the demineralized water injected before you scram? Then eventually you hit high power, you scram.

Will that have any effect on the return to power? Because you will start with high reactivity. It would be an additional component.

MR. SCHMIDT: Yes, it would because you would be coming down effectively from a higher point and your defect would be greater.

CHAIR MARCH-LEUBA: So the scram of power will be on high power or would it be --

MR. SCHMIDT: There is a high-power trip, yes.

CHAIR MARCH-LEUBA: Would there be another one before that?

MR. SCHMIDT: There could be a rate trip depending on --

CHAIR MARCH-LEUBA: It wouldn't be that fast.

MR. SCHMIDT: I honestly don't know but there's two --

CHAIR MARCH-LEUBA: I'm just suggesting for your return to power, maybe this is your limiting event.

MR. BRISTOL: This is Ben Bristol. So, as part of the analysis, we actually look at in mode one how much dilution and what the reactivity insertion is from the dilution and it's very small.

So, bear in mind dilutions are detected quickly such that there isn't really an additional consideration for the purposes of the overall MTC, the defect in the return to power analysis.

CHAIR MARCH-LEUBA: So, if you have the wrong boron concentration you will detect it early?

MR. BRISTOL: So, when the core is critical, it rapidly detects very small dilutions.

CHAIR MARCH-LEUBA: Right, but who detects it, the operator?

MR. BRISTOL: No, the excore. So, we get a high-power rate or an overpower trip.

CHAIR MARCH-LEUBA: The overpower trip is 118? How high is it?

MR. BRISTOL: The non-LOCA limit is 120.

CHAIR MARCH-LEUBA: 120 so it's 180. That's a lot of reactivity. If you assume the operators are on vacation in Bermuda, you would put a lot of positive reactivity for your scram.

If you have 118 power, you're a much higher roll line. Sorry, I'm talking BWR again.

MR. BRISTOL: So, it takes very small dilutions to insert that reactivity. It's not a factor of ten.

CHAIR MARCH-LEUBA: But you have that reactivity when you scram and then you're now going to return to power.

It's something else that you are going to have to compensate for when you return to power. Your power level is still going to be 2.5 percent and if you start at 120 it may now be 5 percent, 6 percent.

MR. BRISTOL: Sure.

CHAIR MARCH-LEUBA: You have additional positive reactivity that you should consider. My point is in my mind this is potentially the limiting scenario for return to power.

MR. BRISTOL: And I think our analysis shows, and obviously we're still reviewing this, but the end of cycle where there's no boron in the system is limiting from the overall temperature defect perspective.

CHAIR MARCH-LEUBA: I'm saying two weeks before end of cycle you remove all the boron so now you are without boron but have more U-235. So, the demineralized water comes in two weeks, three weeks before the end of cycle.

So, then your scram happens when your boron is zero, maximum MTC. In my mind that's potentially the limiting case. It's still going to be okay but in my mind it is potentially a limiting case. Just putting it out there.

MR. BRISTOL: Okay.

CHAIR MARCH-LEUBA: I think we have plenty of margin anyway.

MR. BARRETT: So, the Staff reviewed Section 15.2.8, which is the feedwater line break.

MEMBER CORRADINI: You've got to get closer.

CHAIR MARCH-LEUBA: That one doesn't go more. Keep it away from the paper or again, our court reporter will hate you.

MEMBER CORRADINI: Understood, now go for it.

MR. BARRETT: All right, so the Staff reviewed Section 15.2.8 of the DCA which is the feedwater line break analysis.

For the areas of review with respect to the identification of causes, the Applicant analyzed a wide and large range of feedwater line breaks in different locations, which includes different locations and different break sizes to cover every different possible potential cause.

As far as the methodology, that Applicant used a non-LOCA methodology report that we talked about earlier, which is still under staff review.

With respect to model assumptions, inputs, and boundary conditions, the Applicant provided conservative initial parameters in the analysis consistent with their topical report, including considerations for the availability of power, loss of AC power, loss of DC power with AC power.

The Applicant had also considered and analyzed single failures for each limiting case of the feedwater line break event.

For the minimum critical heat flux ratio case and the limiting RCS pressure case there were no single failures that would cause any sort of adverse impact.

For the limiting steam generator pressure case, the single failure of a safety-related check valve was found to be more limiting. So, it was only slightly more limiting though, a slight uptake in pressure.

Something that's interesting about this case is that they also considered a feedwater line break inside containment, which credits a non-safety check valve.

And the reason why they credit this non-safety check valve is you'll run out all of -- the faulted steam generator will basically be empty and then there's a containment isolation valve, and if it doesn't close fast enough you may end up losing too much inventory.

But they credit this check valve so based off of the closing characteristics, in order to ensure that you don't lose too much inventory they credit this non-safety check valve. And we've already talked about the crediting of non-safety check valves.

And also, the Staff looked at the results which included the reactor power, reactor and steam generator pressures, core temperatures, RCS break flow rates, minimal critical deep flux ratio, and the DHRS heat removal rates.

Next slide. So, the Staff confirmed that for the minimum critical heat flux ratio case, the RCS and steam generator pressurization cases, they all remain within limits.

From a radiological standpoint, it was confirmed that the steam line break radiological analysis bounds the feedwater line break analysis. The DHRS is safety-related which is one of the DHRS requirements so that was adequate.

And the Staff is holding back on making any general conclusions about the overall acceptability until the open items are completed.

The open items are the similar open items that we've already seen before, which is the non-LOCA topical report which is under review, going to NRELAP Version 1.4, the IAB as a single failure, and then also the boron dilution topic.

MS. SIWY: Okay, so let's move on to 15.1.2, which is the increase in feedwater flow event.

We're kind of highlighting this one because the unique aspects here are the potential steam generator overfill scenarios which could affect

DHRS performance, as well as crediting the non-safety-related feedwater red valve for that overfill case.

But we already kind of touched on that this morning. So, the methodology used is the non-LOCA EM, which is currently under Staff review, and VIPRE for the downstream subchannel analysis.

In terms of the input parameters, the initial conditions and assumptions, the Applicant considered a spectrum of different feedwater flow increases.

The Applicant also selected suitably conservative initial conditions and biases that maximized consequences with respect to the acceptance criteria. And the Staff confirmed this by looking at the Applicant's sensitivity calculations in audit.

And the Applicant also considered single failure and loss of power scenarios.

For this event there are no adverse single failures with respect to MCHFR but as I mentioned, the limiting single failure with respect to the steam generator overfill case is the failure of a feedwater isolation valve to close.

And in terms of loss of power, no loss of power scenario would be limiting because it would effectively terminate the cooldown event.

As far as the analysis results, the Applicant provided figures and numerical results for all of the figures of merit, as well as a figure for steam generator overfill.

MEMBER BROWN: This is just a bunch of tubes with water being pumped up through it and the only thing to do is the water can come out on the other end.

There's no water level measurement, at least I don't remember the details of it if there is.

MS. SIWY: So, just to be clear, there are no pumps, it's a natural circulation-driven system and it's based on boiling and condensing so if there's too much inventory there's the potential to not have enough surface area.

MEMBER BROWN: Inventory where? You mean in the tubes?

MS. SIWY: Yes, in the heat exchanger.

MEMBER BROWN: Within the tubes?

CHAIR MARCH-LEUBA: She's talking about DHRS --

MS. SIWY: Yes, DHRS.

MEMBER BROWN: You were talking about steam generator overfill, increase in feedwater flow.

MS. SIWY: So, when the DHRS is used, you're still using steam generators but you're just transferring the heat now to the reactor pool. So it's still using the same --

MEMBER BROWN: I understand that but they're outside the reactor vessel, the RPV, DHRS heat exchangers.

MEMBER CORRADINI: What she's getting at is once you isolate the system, you've got to make sure that you don't put too much water in the system or else you will essentially degrade the performance of the DHRS.

MEMBER BROWN: I thought the feedwater valves were isolated.

MEMBER CORRADINI: But if I isolate the steam and feedwater, then I've got remove the heat. To remove the heat I have to start on DHRS, which is on the inside of the isolation valves.

And if I put too much water in, they won't work as effectively.

MS. SIWY: Yes, and as I mentioned, the overfill case assumes a single failure of a feedwater isolation valve to close so there's that additional time during which inventory can still be added before the feedwater-regulating valve can close.

MEMBER BROWN: Okay.

MEMBER CORRADINI: Let me ask a question to follow Charlie's. So, now I've shut down the feedwater, the whatever it is, the non-safety-related feedwater valve.

It shuts down, I shut main steam, I've isolated, I've turned on DHRS. If I were to overpressurize DHRS, is there a pressure relief valve inside the isolation valves?

CHAIR MARCH-LEUBA: No, because you can only go to the saturation pressure on the primary and DHRS is designed at the same pressure at the primary.

MS. SIWY: Exactly.

MR. THURSTON: And the secondary, it's all the same.

CHAIR MARCH-LEUBA: All the steam you can produce in the DHRS is at the same temperature as the primaries so that's the maximum pressure you can possibly achieve on DHRS.

We went through that.

MEMBER CORRADINI: And I essentially start degrading at 100 percent? I'm eventually going to get to the degradation threshold.

MS. SIWY: I don't know how many detail we can get into here.

MEMBER CORRADINI: That's fine, if we can do it later, I'm interested.

CHAIR MARCH-LEUBA: See, on concern I have in this is that the DHRS heat exchangers are very low, they're not in the top, they are very low. So, if you fill it above the DHRS heat exchangers, you're only condensing it on the pipe and it's a very limited surface.

So, the moment you overfill to above the DHRS heat exchangers, there is still some condensation but very little. Do you understand that?

MEMBER BROWN: No. I mean, you've isolated everything so --

(Simultaneous speaking.)

MEMBER BROWN: -- whatever the inventory is --

MEMBER CORRADINI: I think all Jose is saying is that the heat comes in here, the heat goes out here. If I fill up the water so that the water level is above this, I'm toast.

MEMBER BROWN: But the DHRS heat exchanger is above the level of the top of the core, right?

MEMBER CORRADINI: Think of this. This is the steam generator, this is the DHRS. It's sitting in cold water, it's sitting getting heat from the core.

MEMBER BROWN: No, but it's a closed system.

MEMBER CORRADINI: Right, it's like an isolation condenser in a BWR. It's no different.

MEMBER BROWN: I'm trying to figure out how you can --

(Simultaneous speaking.)

MEMBER BROWN: The volume of water and mass of water in there is fixed once you close the valves.

MEMBER CORRADINI: Yes, but if I close the valves too late I have a little bit of a problem.

MEMBER BROWN: Why? There's just more water?

MEMBER CORRADINI: Because I put too much water in and I filled it up above the activation level so I've degraded performance that I can't remove the decay heat.

CHAIR MARCH-LEUBA: I'll explain.

MEMBER BROWN: Well, you've always got the heat down. It's lower, heating that water before it goes up. I'd have to go back and look at the diagram of the physical placement.

MEMBER CORRADINI: Think of it this way.

It's not that you don't transfer zero energy, you've degraded it to the point that essentially you're not really in decay heat.

It's that simple.

CHAIR MARCH-LEUBA: I'll explain it to him offline.

MEMBER BROWN: Don't worry. Go ahead. I'll go back and look at a picture.

MEMBER CORRADINI: Have I mischaracterized? I don't think so.

MS. SIWY: I think that's a pretty fair characterization.

MEMBER CORRADINI: Keep on going.

MS. SIWY: So, as far as the Staff review and findings, the non-LOCA EM is still under review so that's kind of an open item.

The Staff notes that the Applicant's input parameters, initial conditions, and assumptions are acceptable with the exception of the open item about the IAB as well as some of the conditions for the overfill analysis.

MEMBER SUNSERI: Alex, why don't you hang on a second? We've got three meetings going on.

MEMBER CORRADINI: We're just trying to educate our colleagues. Go ahead.

MS. SIWY: Would you like me start at the top of this slide?

MEMBER CORRADINI: Go ahead.

MS. SIWY: So, the Applicant's analysis result suggests that acceptance criteria are met but the Staff cannot positively confirm this until the open items are resolved.

And the open items, there's one about the potential for steam generator overfill because the Staff identified that the case analyzed in Revision 2 of the DCA may not be the most limiting case in terms of how full the steam generators can get.

MEMBER CORRADINI: And why is that?

MS. SIWY: It has to do with the biasing of initial conditions.

There are certain combinations of initial conditions that can lead to a higher level and there's a difference between the initial level versus the transient level that's achieved.

And that also depends on the initial conditions because they influence the course of the transient.

MEMBER CORRADINI: But at time zero there's essentially a water level above the DHRS cold site heat exchanger, right?

MS. SIWY: Mm-hmm.

MEMBER CORRADINI: So, you're worried that it essentially would rise from there after the re-equilibrium of the levels?

MS. SIWY: Yes, there's potential for that depending on the --

(Simultaneous speaking.)

MEMBER CORRADINI: So the initial conditions are assessed and you've not maximized the amount of initial water inventory? Is that really what you're getting at?

MS. SIWY: Yes. What I'm getting at is has the most limiting case in terms of hindering DHRS performance been considered?

And so we had some discussions with the Applicants about this and they expect to include additional sensitivity studies in their revised calculations for this event as they're updating to incorporate NRELAP5 Version 1.4 as well as the revised base model.

And the rest of the open items are the most generic open items so I won't go over those again.

Okay, should I move on to the next event? Okay, so the next event I'll talk about is 15.2.6, the loss of non-emergency AC power 3 the station auxiliaries.

So, the unique aspects with regards to the NuScale design is that there's no credit for emergency -- well, there are no emergency diesel generators and backup diesel generators are not credited.

So, all the decay heat is being removed through passive DHRS heat transfer. And this also credits the non-safety-related feedwater red valve for the limiting steam generator pressure case, which assumes a single failure of feedwater isolation valves in place.

CHAIR MARCH-LEUBA: Is that valve DC power or AC power?

MS. SIWY: Sorry?

CHAIR MARCH-LEUBA: The valve, is it driven by AC power or DC power? You assume the regulating valve works.

I would use a constant if it was driven by DC where you had the batteries because a loss of AC power would happen like that and then you lose everything. Just think about it.

MS. SIWY: Okay, so the methodology used here is the same as the last transient, non-LOCA EM with the downstream VIPRE analysis.

In terms of input parameters, initial conditions, and assumptions, I think this might help. Well, maybe, because you asked about DC.

So, the assumption here is that there's loss of the low-voltage AC power distribution system and the Staff went back and looked at the whole AC power distribution system and confirmed that's a conservative and acceptable assumption.

The Staff also confirmed that the Applicant selected suitably conservative initial conditions and biases that maximize the consequences with respect to all of the acceptance criteria. And the Applicant did consider single failure scenarios.

In terms of the analysis results, it was a similar set of figures and results for limiting RCS pressure, steam generator pressure, and MCHFR cases.

And this is sort of a similar set of findings as the previous event, so the non-LOCA methodology is still under review but the subchannel has been found acceptable.

The Applicant's input parameters, initial conditions and assumptions are acceptable.

The only open item with regards to that is consideration of single failure criteria with respect to the IAB and the Applicant's analysis results suggest that the acceptable criteria are met but we can't confirm this until the open items are resolved.

MEMBER CORRADINI: These are the similar open items?

MS. SIWY: Yes, they're the common open items I guess.

MEMBER CORRADINI: Okay, fine, thank you. Now to 6.2.1?

Are you guys not going to do 6.2.1 for us? A new set but you're the ever-constant.

MS. SIWY: I'm still driving, yes.

MEMBER CORRADINI: Give us one minute, we're still tutoring.

MS. SIWY: All right. Okay, why don't you go ahead?

MR. HAIDER: Thanks, everyone. Good afternoon, my name is Syed Haider, I'm the Lead Technical Reviewer at NRO for the NuScale FSAR Section 6.2.1.1.

On containment structure, which is mainly related to the containment thermal hydraulic design and peak containment pressure and temperature response is the subject of today's presentation.

Due to the novel nature of the NuScale design and the phenomenal logical complexities in the world, the Staff review of the containment thermal hydraulic design required a team effort of several reviewers from the Office of New Reactors, Office of Research, as well as contractors from ERI and Numark.

The review also required a close collaboration among the Chapter 6 and LOCA topical report reviewers due to several overlapping review areas that involved several audit activities as spanning over the last two years.

I would like to acknowledge the contribution made to the review by various co-authors.

This slide captures the presentation outline that I'll follow for the current open session slides that are intended to cover the non-proprietary material related to the Staff review as documented in the Phase 2 SER Sections, 6.2.1.1.

First, I will briefly introduce the scope of the Staff review, then I will describe the various conservatisms reviewed by the Staff in the NRELAP5 model for the design basis safety analyses of the NuScale power module, or NPM, containment.

As for that, it's initial boundary conditions. I will also explain why Staff found those conservatisms to be acceptable.

During the review the Staff requested

NuScale to perform several containment normalization studies for the NPM as well as the NIST-1 test facility at Corvallis.

I will also present a summary of the Staff review of the supplemental normalization information submitted recently by the Applicant and how it helped resolve the Staff concerns about NRELAP5 predictive capability for liquid thermal stratification and subcooling inside the containment.

And then I will summarize the NuScale containment design and the Staff confirmatory calculations for the limiting design basis event. However, most of the related detail will be presented during the closed session.

The open session presentation will be finished with a summary of the Staff review for NuScale FSAR Section 6.2.1.1.

So, this slide captures the overall scope of the Staff review. Basically, I represent the Staff review of the containment design basis safety analysis on NuScale FSAR Section 6.2.1.1.

That incorporates by reference the CRAM Technical Report, or TR, that describes the containment response analysis methodology. This is an additional report in addition to the LOCA and non-LOCA topical reports.

The Staff conducted the review following the NuScale design-specific review standard, or DSRS, Section 6.2.1.1.

One key regulatory requirement of the DSR is, a section that is rooted in GDC16 and 50, to ensure that the heat containment pressure and containment wall temperature are calculated by the limiting design basis safety analyses are bounded by the design pressure and temperature with sufficient margin.

The second key requirement is driven by GDC38 to ensure that the containment pressure reduces by 50 percent from its peak radio within 24 hours.

NuScale uses the NRELAP54, which is their custom version of the NRELAP5-30 baseline code to perform the integrated design basis safety analysis of the RPV mass energy release during the blowdown and the resulting containment pressurization and heat removal to demonstrate sufficient margin in NPM to meet the above two regulatory requirements.

The qualification of the NRELAP5 code is reviewed under the NuScale LOCA topical report for the primary site pipe breaks and reactor valve opening events. And non-LOCA topical report for the main steam line break and feedwater line breaks.

As there are no significant differences in physical phenomena between the pipe breaks and the valve opening events, the LOCA and non-LOCA parts or phenomena identification ranking tables are also applicable to the valve opening events, as well as the containment response analysis methodology or CRAM.

Therefore, the containment response analysis methodology or CRAM is an extension of the NuScale LOCA valve opening event and non-LOCA methodologies.

The containment response analysis methodology technical report references the LOCA and non-LOCA methodologies and justifies any differences from those methodologies, such as the initial or boundary condition changes that were needed to conservatively bias the mass energy release in order to maximize the containment pressure and temperature response.

That is why the review of qualification of NRELAP5 under the LOCA and non-LOCA topical reports would also cover the NRELAP5 application to containment response analysis methodologies, which will be an open item in Chapter 15.

MEMBER CORRADINI: Still under review.

MR. HAIDER: It's still under review.

MEMBER CORRADINI: I have a question. Under pressure reduction by 50 percent from peak value in 25 hours, that's not in GDC38. I have it right here. So where did that come from?

MR. HAIDER: It is in GDC --

MEMBER CORRADINI: I'm reading GE38 and it never says anything about 24 hours and 50 percent. So, is that a Reg Guide interpretation?

MR. HAIDER: It is a Reg Guide interpretation but GDC38 requires that there should be adequate cooling and the system should be remain cooled.

MEMBER CORRADINI: Okay, but the numbers are not in the criteria, they're in a Reg Guide. I just want to make sure that I'm not off base there because to me, that speaks of AP1000 and past life, it doesn't speak to the design criteria.

Am I remembering wrong?

MR. ASHLEY: No, this is Clint Ashley, you're remembering correctly. It's specified in the standard review plan.

MEMBER CORRADINI: Okay, that's fine.

MR. ASHLEY: You're reading GDC38 properly. It just says --

MEMBER CORRADINI: I've got it. You don't have to read it, I've got it.

MR. HAIDER: So, during about two years of the technical review, 9 FSAR Chapter 6.2.1.1 RAIs were issued with 21 questions.

In addition to the related RAIs issued under the LOCA topical report review, there are no uncommon open items left in this SER section, however, there are still several open items outstanding that related to Chapter 15 and Chapter 3.

So, the next two slides describe the NuScale containment design conservatisms and why they were accepted by the Staff. The Staff determined that NuScale used conservative initial conditions for the spectrum of scram system release events.

The Staff made sure that the fuel inputs were consistently used by the LOCA evaluation models and were conservative with respect to the mass energy release for the containment.

The Staff made sure that for the containment safety analysis the core power also included the uncertainties and the decay heat model was conservative.

The Staff determined that NuScale also maximized the RCS volume in fuel stored energy as well as the secondary system stored energy.

The Staff also established that NuScale had used a conservative containment free volume of 6000 cubic feet that was lower than the net containment free volume that was calculated by NuScale based on the latest containment geometry and design changes while duly accounting for the RCS thermal expansion, piping valves, cabling and miscellaneous components such as platform ladders into the containment.

NuScale also added a COL item to use the as-built containment free volume to confirm its peak calculated containment pressure, temperature are bounded by the containment design pressure and temperature values.

The Staff recommended that the COL item did address the Staff concerns regarding any unaccounted for manufacturing uncertainties in the containment free volume.

The Staff also reviewed the NuScale methodology to initialize the containment wall temperature and determined that NuScale had ensured a conservative initialization of the containment inside surface temperature for normal operation that would minimize condensation during a transient.

NuScale assumed a zero temperature drop across the reactor pressure vessel wall to maximize the radiator heating of the containment wall.

They also assumed an adiabatic boundary condition for the exterior surface of the containment upper head and above the cooling pool level.

And performed a conservative, finite analysis of the two initial temperature distribution along the entire containment wall using ANSYS.

For the containment safety analysis, NuScale applied the maximum temperature value of 240 degrees Fahrenheit, found by the two re-analysis on the entire containment upper head and in the surface, as well as the containment above the pool, which the Staff found conservative.

So, they found 240 degree at one point and they applied it on the entire surface above the pool level, which the Staff found conservative.

As an additional conservatism, NuScale used the high initial containment internal pressure of 3 psia for the containment safety analysis.

NuScale had provided a plot of the operational limits of acceptable RCS lead detection as a function of the containment pressure and pool water temperature which should define the normal operation limit.

MEMBER CORRADINI: That's what I guess I...Let me ask you it differently. So, the high temperature limit of 110 F is related to a saturation temperature of what? Pressure I mean.

MR. HAIDER: The plot, the horizontal axis, the vertical axis is the temperature.

MEMBER CORRADINI: Okay, fine.

MR. HAIDER: So, for 1.2 psia the corresponding pool temperature is 110. 1.2 corresponds to 110 degree Fahrenheit.

So, if the pressure goes above 1.2, then it's out of the operational limit so it would have to trip. While conservatively, the Applicant is using 3 psia which is 1.8 psia or 2 psia higher so it's conservative.

MEMBER CORRADINI: Okay, got it, thank you.

MR. HAIDER: So, based on the plot, the Staff found using 3 psia to be conservative as the operational plot showed that the containment internal pressure had to be maintained below 1.2 psia to ensure an acceptable in-containment leakage detection performance or the high pool temperature limit of 110 degrees Fahrenheit.

So, 3 psia was conservative as the initial containment pressure as it was greater than the limit of the operability of the leakage detection system that was 1.2 psia.

So, the peak containment pressure predicted by assuming an initial internal pressure of 3 psia is about 2 psia higher.

So, first of all, it is going to lead to higher, about 2 psia higher pressure, and secondly, it's also going to maximize the initial non-condensable mass, which is presented in the containment under normal operation.

Which is also conservative due to its adverse impact on the condensation heat transfer.

MEMBER CORRADINI: If I remember correctly, they also then took a decrement by adding additional non-condensables due to more gas space in the pressurizer.

Am I remembering correctly?

MR. HAIDER: That's right, and I'll be giving more detail on all that during the closed session.

MEMBER CORRADINI: Okay, fine.

MR. HAIDER: So, while addressing some of the Staff RAI questions, NuScale made several containment and design modeling changes.

One of the changes was to revised a technical specification by raising the cooling pool level from 55 feet to 65 feet.

However, the increased 65-foot pool level is only credited for containment wall temperature initialization and is not credited for containment heat removal.

So, the containment safety analysis is still based on the lower 55-foot level which the Staff found to be a conservative option.

The Staff closely reviewed the Applicant's modeling of the condensation heat transfer on the containment inside surface that also includes the adverse effects of non-condensable gases.

The Staff found the condensation modeling to be conservative. As NuScale was requested and they were able to demonstrate that it had used a conservative condensation heat transfer coefficient correlation.

The Staff also determined that the NuScale had used a maximum mass of non-condensables present in the containment volume during normal operation and released into the containment from RPV during the design basis transient.

The Applicant had duly accounted for the deterioration of the interfacial heat transfer to the condensing film on the containment wall caused by the insulating presence of non-condensable gases.

More details will be provided about the Staff review of the condensation heat transfer modeling during the closed session. So, during the review, the Staff had requested NuScale to demonstrate RELAP5's capability to predict the thermal stratification and subcooling of the liquid water.

That would accumulate at the lower containment due to the boron and condensation of the flashing steam on the containment wall.

The Staff was concerned that the old predicting the containment liquid water temperatures due to nodal stratification could lead to an underprediction of the containment pressure.

Because boron will get accounted for in the liquid space but less might get accounted for in the vapor space and that may lead to lower pressure.

So, the Staff needed a demonstration of lack of sensitivity of the simulation reserves to containment and liquid volume nodalization in order to establish NRELAP5's capability to accurately predict the containment pressure response and the peak containment pressure for the limiting design basis event of the inadvertent opening of the RRV, which is the largest liquid space discharge event in the NuScale design.

In response, NuScale submitted nodalization studies for the NPM as well as the NIST-1 test facility for the HB49 test.

The Staff reviewed both nodalization studies submitted for the containment volume and liquid space axial nodalization and concluded that the Applicant has demonstrated sufficient credibility for thermal stratification and subcooling of liquid water in the lower containment of the nuclear power module.

The Applicant has also shown that the peak containment pressure and temperature did not have any significant sensitivity to the containment wall radial nodalization and reactor pool axial nodalization.

MEMBER CORRADINI: So, this open item is closed?

MR. HAIDER: We finished reviewing and we believe that we have --

MEMBER CORRADINI: Okay, I just wanted to make sure --

(Simultaneous speaking.)

MR. HAIDER: -- reasonable assurance to

close this open item.

MEMBER CORRADINI: Okay.

MR. HAIDER: So, theoretically, yes.

The issue is an open item in Phase 2 SER, however, Applicant recently submitted the supplemental information and the Staff concludes that it has reasonable assurance to close it.

And more details will be provided about it during the closed session.

Okay, the Revision 0 for NuScale FSAR and CRAM technical report had documented a 951 psia peak containment pressure and 1000 psia containment design pressure. That's where the RVV started from.

So, while addressing some of the Staff RAIs about the peak containment pressure calculations and the containment design pressure, NuScale submitted certain containment design and modeling changes.

Those changes have resulted into an updated calculated pressure, containment pressure of 986 psia for the pressure limiting inadvertent reactor recirculation valve opening event.

NuScale also raised the containment design pressure from 1000 psia to 1050 psia that now shows about a 6 percent margin with respect to the updated peak calculated containment pressure of 986 psia.

The information regarding raising the containment design pressure is cracked under an open item in Chapter 6 which has not been closed yet.

Likewise, the peak calculated containment wall temperature was slightly increased from 523 Fahrenheit to 526 Fahrenheit from a double-ended break of the RCS injection line, which shows a 24 degrees Fahrenheit margin with a containment design temperature of 550 degree Fahrenheit.

So, the Staff also performed independent confirmatory calculations using MELCOR and TRACE codes that show that NuScale containment peak pressure and temperature predicted by NRELAP5 are conservative.

Considering the critical nature of the peak pressure review, the Staff also performed a large number of sensitivity studies using the safety analysis in RELAP5 decks that were obtained from NuScale.

So, the containment-related safety findings are pretty much being finalized with the help of three independent computer codes. That means MELCOR, TRACE, and NRELAP5.

A summary of the Staff confirmatory calculations will be present during the closed session. So, it looks like one slide was deleted.

Okay, so anyway, now the SER Section 6.2.1.1 does not have any specific open items directly related to containment thermal hydraulic design.

However, the SER still does reference several open items such as experimental validation of NRELAP5, scale and distortion issues that are reviewed under LOCA topical report.

The Staff has these indirect open items, we cannot finalize the safety findings about the NPM, however, once the Staff completed the review of those standalone issues in the DCA the Staff will be able to conclude that there is a 6.1 safety margin in the design pressure, there is a 24 degrees Fahrenheit margin in containment wall temperature.

The Staff can also conclude that the containment pressure dropped by 50 percent of its peak value well within 24 hours of the initiation of the transient in all DBAs.

This would essentially mean that all the regulatory requirements would have been met by the NPM once the open items have been closed.

Let's go to the next slide. So, here is a slide that I'll present for -- so, that was the conclusion of Section 6.2.1.1.

Here is a single slide that I'll present for Section 6.2.1.3 and Section 6.2.1.4 on mass energy release analysis for postulated LOCA and secondary pipe ruptures inside containment.

Boyce Travis is the reviewer of these two sections and he's on travel this week so I'm presenting. Basically, GDC50 requires the containment to be designed to accommodate the calculated pressure and temperature conditions resulting from a postulated break.

Staff evaluated the break spectrum blowdown conditions in energy sources for the NPM and determined that the NuScale either conformed directly with the guidance provided by NuScale DSR Section 6.2.1.3 and 6.2.1.4 or used appropriately bounding initial conditions to maximize initial RPV mass energy release.

However, the Staff's evaluation is dependent on the review of the LOCA topical report. Open item, as the reference analysis rely on NRELAP5 LOCA evaluation model.

This pretty much concludes my presentation for 6.2.1.1. Any questions?

MEMBER CORRADINI: I don't think so. Why don't we try to finish this before we do take a break or anything? We're close.

MR. ASHLEY: Very close. Good afternoon, I'm here to present the Staff's review of NuScale containment heat removal system or CHRS.

I'd like to inform the Committee that the technical reviewer that generated the safety evaluation and prepared the slides is not here today.

You'll notice at the bottom of the slides, unfortunately it's says proprietary but these slides are not proprietary. We'll need to clean that up.

MEMBER CORRADINI: We're always in trouble so thanks.

MR. ASHLEY: In the 6.2.2 review area, the Staff has one open item that does have a clear path to resolution and I'll talk about that in an upcoming slide.

To meet the regulatory requirements shown on this slide, traditional light-water reactors are generally equipped with active systems like pumps and valves.

In the past NuScale design these components are not necessary because the containment heat removal function is an inherent characteristic of the containment given that the containment vessel is immersed in a large reactor pool.

Also, because there's a tight coupling

between the containment heat removal system and the evaluation of limiting containment transients, the Staff reviewed containment heat removal requirements associated with general design criteria 35 and 38 in conjunction with the containment peak pressure and temperature analysis that you just heard presented by Syed Haider.

As a result, the Staff's presentation on this section will focus on NuScale's key design considerations and features related to the GDC40 exemption request and the long-term cooling evaluation.

During this week's ACRS Subcommittee meeting held on Tuesday, June 18th, the subject matter of this slide, long-term cooling, was discussed at length, both by the Applicant and the Staff. And that was done during the presentation on Section 6.3.

So, based on the interactions between ACRS Members and the Applicant and Staff, I believe the pertinent information on this slide has already been discussed. Assuming there aren't any questions, I would recommend we proceed to the next slide.

On this last slide, I'll discuss the Staff's review of the Applicant's GDC40 exemption request and the Staff's open item in Section 6.2.2.

As we already discussed GDC40 requires a nuclear power-plant design have provisions to test the containment heat removal system such that operability is demonstrated for the full spectrum of components.

The intent behind GDC40 is to ensure the continued operability of the containment heat removal system and verify the system meets the performance specifications assumed in the safety analysis.

In past practice, compliance with GDC40 has been achieved by testing the active system safety performance, characteristics such as pump flow rate for a containment spray system.

The NuScale design obviously uses passive containment cooling and does not include or require active components to provide the containment heat removal function.

To address the underlying performance of GDC40, the NuScale design requires two things: periodic in-service inspections of the containment heat removal surfaces, which will assess surface fouling, or degradation that could potentially impede heat transfer from the containment.

As well as inspection and testing of the other components that are necessary for the containment heat removal system to function such as the emergency core cooling system.

So, as a result of these NuScale inspection and testing requirements, the Staff found that the underlying purpose of GDC40 will still be accomplished with this exemption request.

MEMBER CORRADINI: I want to make sure because I've written down here, but the in-service inspection will still require a cleanliness program?

MR. ASHLEY: That is correct. And that's subject to the open item.

MEMBER CORRADINI: Okay, I got it. This is actually a straightforward open item. There's a claimless program and there were discussions in the FSAR that could be clarified.

There wasn't a close or explicit link between the debris conditions in the containment associated with the claimless program so they wanted to have a tighter link.

So, there was a call held with NuScale and NuScale agreed to update the DCA to make sure that there was a close coupling between the debris assumptions in the FSAR and the COL item.

That way, the COL Applicant will have -- it won't be as subjective, it'll be a very explicit link to the performance criteria.

This issue is tracked by open item 622-1, and this concludes the Staff's presentation on containment heat removal.

MS. SIWY: This also concludes the Staff's overall open presentations. So, are there any additional questions for the Staff at this time before we go for public comments and questions?

MEMBER CORRADINI: I don't hear any so why don't we go to public comment? So, can we get the public line open?

MR. SNODDERLY: Yes. I'm sorry, this is Mike Snodderly. If include ask the Staff to take an action item to correct those slides with the proprietary stuff so it's fixed.

I don't want to put them in the way they are now. Thanks.

MEMBER CORRADINI: So, is there anybody in the room that wants to make a public comment, please? And the public line should now be open so can anybody out there that's on the public line acknowledge that you're out there, please?

MS. FIELDS: Yes, this is Sarah Fields.

MEMBER CORRADINI: Go ahead with your comment.

MS. FIELDS: Yes. My comment is -- all of the sudden I hear a whole bunch of shuffling and everything, probably from someone on the open line.

My comment is in both the LOCA and non-LOCA loss of power events, there's no AC power and with or without DC power, it is the same thing that ECCS valve would open but it's not clear to me what would happen if an ECCS valve does not open rapidly or does not open at all.

And I don't understand how that has been evaluated and maybe in subsequent correspondence with the NRC I can get a handle on that because it's really important.

Okay, well, thank you for your time today and all the time that's gone into the work on this project by the NRC and NuScale.

MEMBER CORRADINI: Thank you. Anybody else on the open line that would like to make a comment, please? Okay, hearing none, why don't we close the public line?

Let's take a break until 2:55 p.m. and we'll pick up with the closed session then. And I'd like NuScale and the Staff to make sure the room is cleared of anybody that doesn't need to be here.

(Whereupon, the above-entitled matter went off the record at 2:38 p.m.)

Safety Evaluation with Open Items: Chapter 15, "Transient and Accident Analyses"; Containment Performance

NuScale Design Certification Application

ACRS Subcommittee Meeting
June 20, 2019

Agenda

- NRC Staff Review Team
- Summary and Status of the NRC Staff's Review
- Key Issues Related to Chapter 15 Analysis Methodology, Input Assumptions, Event Categorization, and Generic Evaluation
- Key Chapter 15 Events
- Containment Performance

NRC Staff Review Team

- Technical Reviewers:
  - Antonio Barrett, NRO/SRSB
  - Andrew Bielen, RES/CRAB
  - Tim Drzewiecki, NRO/ARTB
  - Jim Gilmer, NRO/SRSB
  - Syed Haider, NRO/SRSB
  - Michelle Hart, NRO/RGRB
  - Andrew Ireland, RES/CRAB
  - Shanlai Lu, NRO/SRSB
  - Ryan Nolan, NRO/SRSB
  - Jeff Schmidt, NRO/ARTB
  - Alex Siwy, NRO/SRSB
  - Ray Skarda, RES/CRAB
  - Matt Thomas, NRO/SRSB
  - Jason Thompson, RES/CRAB
  - Boyce Travis, NRO/ARPB
  - Carl Thurston, NRO/SRSB
  - Chris Van Wert, NRO/ARTB
- Project Management
  - Rani Franovich, NRO/DLSE

Summary of the Staff's Review

- Safety evaluation report (SER) is based on design certification application (DCA) Revision 2
- SER contains 32 open items and 9 confirmatory items
  - Significant progress has been made on several of these items since drafting the SER
- Two requested exemptions evaluated in Chapter 15
  - Portions of Appendix K (Section 15.0.2)
  - General Design Criterion (GDC) 27 (Section 15.0.6)
- 4 audits conducted for Chapter 15

Key Issues Related to Chapter 15:
Analysis Methodology, Input Assumptions, Event Categorization, and Generic Evaluation

- Analytical Methodology Topical Reports (TRs)
  - Loss-of-Coolant Accident (LOCA) Analysis
  - Non-LOCA Analysis
  - Rod Ejection
  - Accident Source Term
- Reanalysis of Chapter 15 Events with NRELAP5 Version 1.4 and Revised Base Model
- Design Changes to Decay Heat Removal System (DHRS) Logic and Emergency Core Cooling System (ECCS) Actuation Signals
- Credit for Non-Safety-Related Valves
- Treatment of Inadvertent Actuation Block (IAB) Single Failure
- GDC 27 Exemption
- Return to Power Analysis
- Long-Term Cooling Analysis

LOCA Topical Report Scope

- Topical Report Application Scope
  - Provide a methodology to analyze NuScale LOCA scenarios
  - Provide a methodology to analyze the inadvertent opening of an ECCS valve
  - Support the methodology for predicting the peak containment pressure
- ACRS Subcommittee meeting on LOCA Evaluation Model (EM) TR scheduled for Fall 2019

LOCA Regulatory Basis, Exemptions, Acceptance Criteria

- Regulatory Basis For LOCA Analysis - 10 CFR 50.46 Requirements
  - Peak cladding temperature < 2,200 °F
  - Local maximum fuel cladding oxidation < 17%
  - Maximum hydrogen generation < 1%
  - Long-term cooling addressed by a separate technical report in DCA Section 15.0.5
- NuScale uses an Appendix K model except for the following exemptions
  - Swelling and rupture of the cladding and fuel rod thermal parameters
  - Reflood & Refill
  - Pump Model and BWR specific items
- NuScale Licensing Acceptance Criteria for LOCA and Inadvertent Opening of an ECCS valve
  - Collapsed liquid level (CLL) remains above top of active core
  - Maintain margin to critical heat flux (CHF): CHF ratio (CHFR) > $CHFR_{-limit}$

LOCA Topical Report Topical Review Focus

Areas of staff focus

- Evaluation Model Development
  - Phenomena Identification and Ranking Table
  - NuScale Integral System Facility And Test Program
  - Scaling and Distortion
  - Code Assessment Against Integral Test Data and Separate Effects Data
- Applicability of NRELAP5 to LOCA Analysis
  - Event Progression
  - NRELAP5 Numerics and Condensation Model
  - Evaluation Model Input Development
  - Appendix K Requirement Models
  - Bounding input values

LOCA Topical Report
Major Issues and Status

Phenomena Identification and Ranking Table (PIRT)

Significant Finding:

- PIRT did not address the scenario with loss of coolant at the lowest possible elevation (through reactor recirculation valve (RRV))

NuScale Response

- Performed RRV opening test (HP-49) to confirm the existing PIRT
- Perform nodalization study on the NIST NRELAP5 model to evaluate stratification for peak containment pressure analysis

Issue Resolution Status

- HP-49 test results have been reviewed. No core uncovery is observed
- HP-49 benchmark case nodaliaztion study is ongoing

LOCA Topical Report Major Issues and Status

Containment Wall Condensation

Significant Findings:

1. NRELAP5 wall condensation correlation applicability range did not cover NuScale Power Module (NPM) LOCA conditions
2. The actual implementation of the correlation in NRELAP5 used different hydraulic diameter from the original correlation

NuScale Response:

1. Revised NRELAP5 code and plan to submit newer version of code (V1.4)
2. Activated conservative laminar flow condensation correlation
3. Demonstrated conservative peak containment pressure in comparison of test data

Issue Resolution Status:

The correlation and the NRELAP5 computational framework are considered conservative.

LOCA Topical Report
Major Issues and Status

CHF Correlation

Significant Finding:

During an inadvertent opening of RRV event, a momentary core flow stagnation and reversal is expected. NuScale applies a combination of high- and low-flow CHF correlations to cover the range of flows encountered during LOCA and LOCA-like events. NRC staff had questions regarding:

1. CHFR limit(s) for specific events
2. Application of a low-flow CHF modifier
3. Interpolation between the high and low flow CHF correlations

NuScale Response:

NuScale is developing LOCA TR markups to address the items identified above.

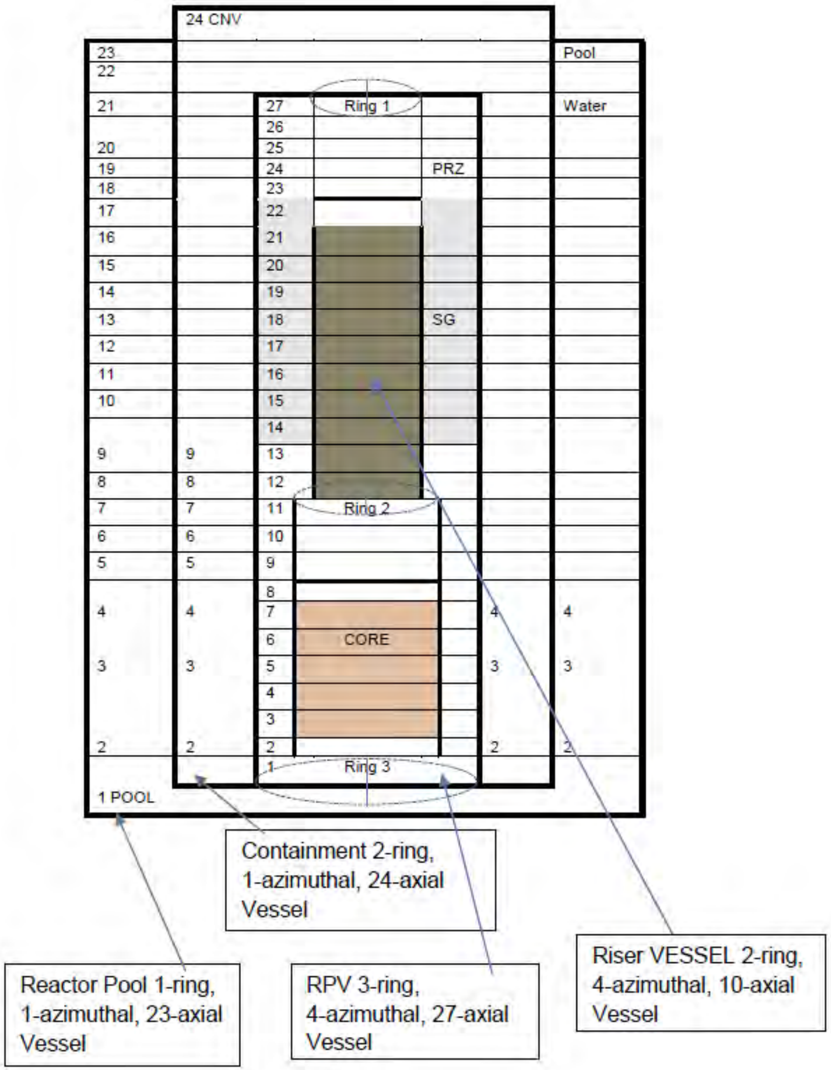
Issue Resolution Status:

Staff is waiting on LOCA TR markups

Staff LOCA Confirmatory Analysis

Both NRELAP5 and TRACE codes have been used to analyze LOCA TR in support of issue identification and resolutions.

Over 70 LOCA cases have been performed.



TRACE Model

Analytical Methodology Topical Reports: Non-LOCA

<u>Scope of Non-LOCA Methodology using NRELAP5</u>

- Non-LOCA design-basis events except for static events (e.g., control rod misalignment), control rod ejection, and inadvertent operation of ECCS
- Time frame covered: while mixture level is above top of riser and natural circulation is maintained
- Interface with other methodologies
- Certain event-specific assumptions and conservative bias directions for initial conditions

<u>Status of Non-LOCA Review</u>

- Review is ongoing - most technical issues resolved, but continued audits and public teleconferences to address remaining technical issues
- ACRS Subcommittee Meeting scheduled for Fall 2019.

Analytical Methodology Topical Reports: Non-LOCA

Areas of staff focus

- Conformance to existing guidance
  - Standard Review Plan (SRP) Section 15.0.2, “Review of Transient and Accident Analysis Methods”
  - Regulatory Guide 1.203, “Transient and Accident Analysis Methods”
  - Design-Specific Review Standard (DSRS) Section 15.0,“Introduction – Transient and Accident Analyses”
  - Event-specific SRP/DSRS sections
- Applicability of NRELAP5 to Non-LOCA Analysis
  - Considerable overlap with LOCA EM – graded approach
  - Separate effects tests: SIET, KAIST, NIST HP-03, HP-04
  - Integral effects tests: NIST NLT-02a, NIST NLT-02b, NIST NLT-15p2* *provided in request for additional information (RAI) response
  - Code-to-code benchmark against RETRAN-3D for reactivity-initiated events
  - Bounding input values
  - Other methodologies (e.g., subchannel)

Analytical Methodology Topical Reports: Non-LOCA

Areas of staff focus

- NRELAP5 facility models
- Inputs and modeling assumptions
  - Use of “suitably conservative” input in lieu of uncertainty evaluation
  - Adequacy of sensitivity studies to determine conservative bias directions for initial conditions
  - Consideration of single failures and loss of power scenarios
  - Treatment of plant control systems

Analytical Methodology Topical Reports: Non-LOCA

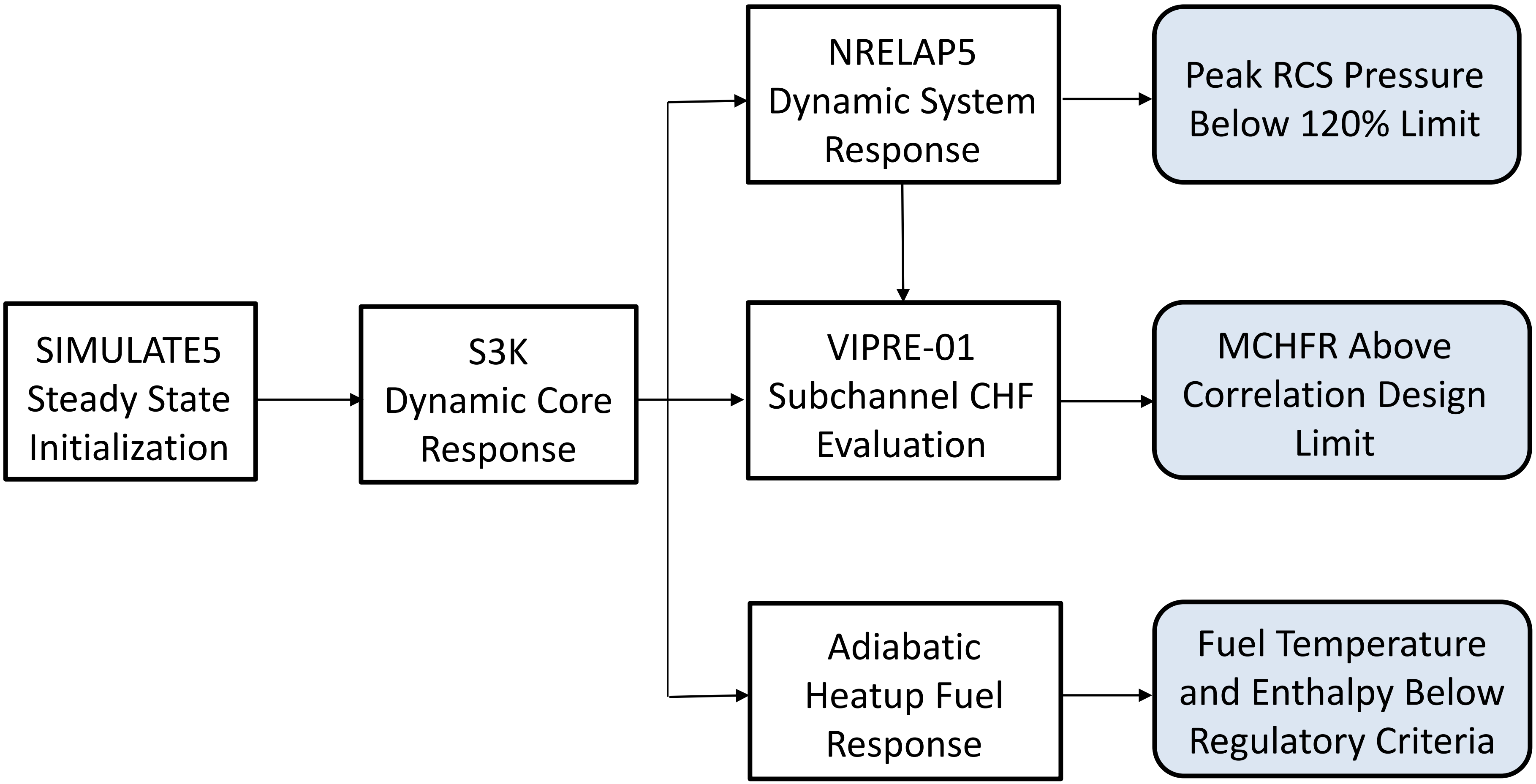
<u>Challenging Review Areas</u>

- Adequacy of steam generator (SG) heat transfer uncertainty
- Highly ranked phenomena and transient multi-dimensional effects
- Nodalization and scaling for DHRS
- Assessments against NIST test data

Analytical Methodology Topical Reports: Rod Ejection

- Rod Ejection Methodology Acceptance Criteria
  - Meets or exceeds staff guidance
    - Hot zero power (HZP) clad failure criteria based on the more limiting value associated with the highest differential pressure
    - No minimum CHFR (MCHFR) clad failure is allowed
    - No fuel centerline melt is allowed

Analytical Methodology Topical Reports: Rod Ejection



Analytical Methodology Topical Reports: Rod Ejection

Areas of staff focus

- Validation of SIMULATE-3K
- Input assumptions and uncertainties
- Transmittal of inputs between codes
- Adiabatic fuel response hand calculation

- Review Status
  - Staff is currently completing SER
  - ACRS subcommittee meeting scheduled for fall 2019

Analytical Methodology Topical Reports: Accident Source Term

- FSAR references NuScale topical report, TR-0915-17565, "Accident Source Term Methodology," for source term and accident analysis methods
- Dose analyses for:
  - Failure of small lines carrying primary coolant outside containment
  - Steam generator tube failure (SGTF)
  - Main steam line break (MSLB) outside containment
  - Fuel handling accident
  - Iodine spike design basis source term
  - Core damage event

Analytical Methodology Topical Reports: Accident Source Term

Open Item 15.0.3-1

- Review not complete because TR Revision 3 and related FSAR changes received April 19 and 21, 2019:
  - NuScale first proposed changes to methodology framework January 2018
  - Core damage event categorized as special event
    - To meet 10 CFR 52.47(a)(2)(iv) requirement for safety analysis, including footnote describing postulated fission product release to containment
    - Design-specific source term based on core damage scenarios in severe accident analysis
  - Iodine spike design basis source term added
    - Intended to be bounding surrogate design-basis accident radiological release to containment (without core damage) and radiation source for environmental qualification of equipment
    - Assumes release of entire coolant activity inventory to the containment

Analytical Methodology Topical Reports: Accident Source Term

- Accident Source Term Methodology TR review status:
  - RAIs related to core damage event and aerosol deposition, iodine spike design basis source term
  - Audit started on June 3, 2019
  - ACRS subcommittee tentatively scheduled December 2019

Reanalysis of Chapter 15 Events with NRELAP5 Version 1.4 and Revised Base Model

- NRELAP5 was revised (mid-2018) to address a minor non-conservatism in condensation heat transfer (RAI 8990)
- Additionally, other minor improvements implemented
- Detailed model changes deferred to closed session

Reanalysis of Chapter 15 Events with NRELAP5 Version 1.4 and Revised Base Model

- NRELAP5 base model (2015) used up to current DCA Revision 2
- Staff noted NRELAP5 models were not consistent with ANSYS solids model design changes (RAI 9325)
- NRELAP5 base model subsequently updated (2017) and will be used for DCA Revision 3 in conjunction with NRELAP5 code version 1.4
- NuScale to rerun Chapter 15 events using updated base model and NRELAP5 version 1.4
  - Revised analyses scheduled for July 31, 2019

Changes to DHRS/ECCS logic

- NuScale submitted a plan to revise the module protection system (MPS) logic for operational considerations:
  - Existing DHRS actuation signal will be split into two signals:
    - A secondary side isolation (steam and feedwater isolation)
    - DHRS actuation (opening the DHRS actuation valves)
  - NuScale further plans to remove the ECCS actuation signal on low reactor pressure vessel level, which the applicant states is not credited

- Staff has received the proposed changes, with the exception of planned changes to Chapters 15 and 19

- Staff review of these changes is ongoing, and will not be complete until after the planned re-analysis of Chapter 15 transients, which will incorporate the logic changes above

Credit for Non-Safety-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 non-safety-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 Non-Safety-Related Valves

- Relying on the non-safety-related secondary main steam isolation valves (MSIVs) in SGTF 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

Single Failure Criterion (SFC)

- Important element of the NRC's defense-in-depth safety philosophy
- A mechanism to promote reliability and confidence in the safety systems to perform their safety functions
- Applied at system design and plant safety analysis level
- Required by regulations

Treatment of IAB Single Failure

Requirements, Guidance, and Policy

- GDC in Appendix A to 10 CFR Part 50 states the design basis accident analysis includes consideration of single failure for active components.
- SRP/DSRS Chapter 15 states the performance of each credited protection or safety system is required to include the effects of the most limiting single active failure.
- SECY-77-439 specifies that single failure should be applied for components that rely on mechanical movement (with exception of simple check valves).
- SECY-94-084 clarified that check valves are subject to SFC when operating in passive systems with low driving forces but allowed for exemption to SFC if high reliability could be demonstrated.

Treatment of IAB Single Failure

- The NuScale Chapter 15 analysis does not assume the potential for single failure of the IAB valve closing function.
- The IAB valve is a safety-significant, first-of-a-kind design feature, and more complex than components currently excluded from the SFC.
- The staff requested Commission direction on several proposed options to continue the DCA review in SECY-19-0036.
- This is identified as Open Item 15.0.0.5-1.

Treatment of IAB Single Failure

<u>SECY-19-0036 Options</u>

1. Extension of existing SFC policy to evaluate the reliability of the IAB valve combined with system design and programmatic elements that reduce frequency of challenge to safety function.
2. Traditional application of the SFC.
   - Additional analysis
   - Alternate acceptance criteria
   - Design change
   - Exemption request
3. Commission determination that NuScale has adequately demonstrated the proper functioning of the IAB valve and failure of the IAB valve closing function is not credible.

Treatment of IAB Single Failure

Path Forward

- The staff and NuScale have aligned on a path forward using an Option 1 approach.
- The IAB is only required to perform its closing function if:
  - Highly reliable dc power (EDSS) is lost to the ECCS trip valve; and,
  - Reactor coolant system (RCS) is at high pressure
- The staff is evaluating the likelihood of challenge to the IAB due to loss of power in combination with the reliability of the IAB.
  - EDSS augmented design, quality, testing, and maintenance
- If it is concluded that the combined reliability of the EDSS and IAB is sufficiently small, then SFC does not need to be considered, commensurate with SECY-77-439.

GDC 27 Exemption

- GDC 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 principal design criterion (PDC), PDC 27
  - Exemption evaluation includes the lack of ECCS injection

SECY-18-0099

- SECY-18-0099 (ML18065A540) used the following three criteria used 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 specified acceptable fuel design limits (SAFDL) 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 supported the proposed staff criteria with the addition of evaluating the overall facility risk, which is reflected in the third criterion above (ML18052A532)
- Satisfying the three criteria in SECY-18-0099 would ensure no undue risk to public health and safety

Proposed PDC 27

- PDC 27 in FSAR Section 3.1.3.8 states:
  - “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, 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”
- Staff noted the proposed PDC 27 is unclear about whether the SAFDL are met, including margin for a stuck rod
  - This is Open Item 15.0.6-1

Return to Power Scenarios

- Three scenarios can potentially lead to a return to power
  - DHRS cooldown with dc power (EDSS)
  - DHRS cooldown without dc power (EDSS)
    - ECCS actuation at IAB setpoint
  - ECCS cooldown
- Can occur as a result of most Chapter 15 events
- Key assumptions assumed 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 out of the core consistent with current GDC

DCA Return to Power Analysis

- DCA Part 2, Tier 2, Section 15.0.6, Revision 2, presents a DHRS cool down opening the ECCS valves at the maximum return to power, which is thought to bound all three cooldown scenarios
  - Analysis inputs maximize the potential return to power
    - RCS mass maintains single phase natural circulation to maximize DHRS cooldown
    - Two DHRS trains in service
    - Lowest reactor building pool temperature (ultimate heat sink temperature)
    - Minimum shutdown margin
    - A conservative stuck rod hot channel peaking factor
    - A conservative, negative end-of-cycle (EOC) moderator temperature coefficient (MTC)
  - Analysis methods
    - Uses Inadvertent Operation of the ECCS methodology of DCA Section 15.6.6
    - Core modeling and nodalization are consistent with the LOCA topical report
    - Uses NRELAP5 version 1.4
    - MCHFR determined using CHF correlation consistent with the LOCA analysis
  - Results
    - Maximum, core average return to power of approximately 10% rated thermal power (RTP)
    - Equilibrium return to power of approximately 2.5% RTP
    - MCFHR limit met

Return to Power Open Items

- DCA does not adequately evaluate a potential return to power when water level drops below the riser
  - Response to RAI 9508 indicated RCS temperature reaches a value which could lead to a return to power
- DCA does not adequately address the potential return to power following a postulated rod ejection
  - Public meeting to be scheduled to address the path forward
- DCA does not adequately address ECCS return to power
  - EOC, ECCS return to power with decay heat greater than 100 kW is possible with the same assumptions as the DHRS cool down scenario
  - Non-EOC, ECCS return to power due to boron plate-out (boron redistribution) on regions outside the core

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Boron Redistribution

During the long-term cooling after ECCS actuation

- NuScale predicted return to power under EOC conditions
- RAI 8930 was issued to address potential boron redistribution for the entire cycle and the potential for a return to power within 72 hours

Safety Concern:

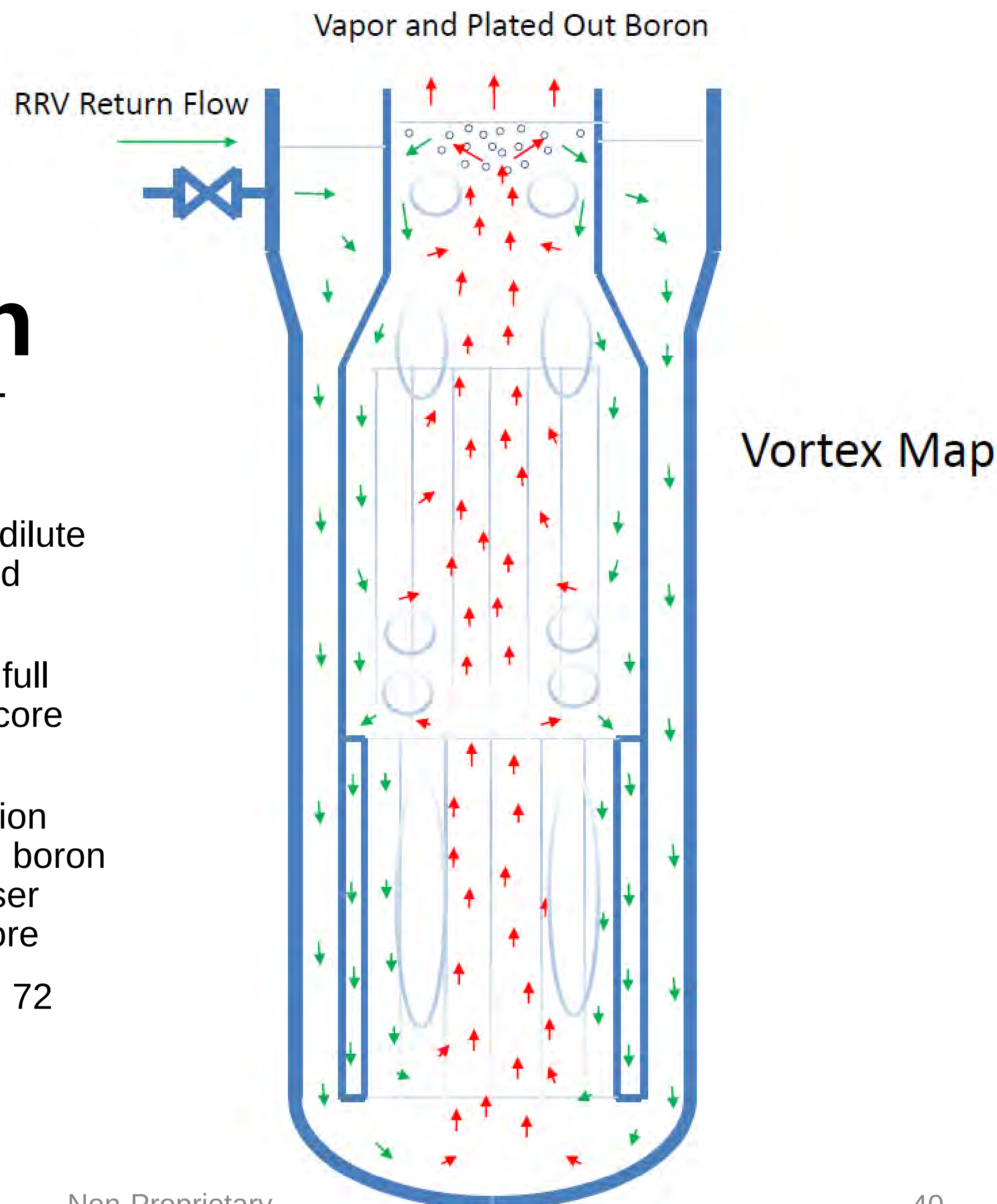
A potential return to power could challenge CHF. The return to power could further accelerate boron plate-out resulting in higher power level, and this cycle could continue.

Boron Relocation:

- Initial blowdown relocates large amounts of borated water to the containment some of which might not return
- Diluted condensate return can be transported to the core
- Boron plate-out due to volatility could further reduce the core boron concentration in the long term
- In the early part of the transient boron can concentrate in the core potentially offsetting the loss terms

Boron Redistribution

- In-Vessel Boron Mixing
  - Diluted condensate return could dilute the downcomer water volume and reach the reactor core
  - When boiling occurs in the core, full mixing is expected between the core and riser fluid
  - When boiling migrates to the region below the two-phase water level, boron could concentrate in the lower riser water volume and dilute in the core
- A potential return to power within 72 hours could exist



Return to Power Confirmatory Runs

- The Office of Research performed the following return to power scenarios
  - DHRS, EOC cool downs
    - With and without biasing feedbacks and reactivity components
  - ECCS, EOC cool downs
    - With and without biasing feedbacks and reactivity components
  - Beginning-of-cycle (BOC), ECCS cool downs
    - With and without biasing
    - With a positive reactivity insertion to simulate the core response for a potential bounding boron redistribution
      - Details of the analyses to be provided in the close session

Long-Term Cooling Analysis

- Addresses residual heat removal for LOCA and non-LOCA analyses when on the ECCS
  - Focuses on reduced inventory events which would challenge ECCS cooling capability
  - Assumes shutdown
  - Evaluates out to 72 hours
- Figures of merit include:
  - Decreasing clad temperature
  - Minimum RCS temperature to prevent boron precipitation
  - CLL remains above the top of active fuel
- Staff review focused on
  - Validation to NIST test data
  - NRELAP5 model
  - Assumed analysis conditions

Key Chapter 15 Events

- Decrease in Coolant Inventory
  - LOCA
  - Inadvertent Operation of ECCS
- Inadvertent Decrease in Boron Concentration
- Feedwater Line Break
- Increase in Feedwater Flow
- Loss of Non-Emergency AC Power to the Station Auxiliaries

15.6.5: LOCA

- LOCA break spectrum limited to 2” chemical and volume control system (CVCS), injection or discharge, high point vent, or pressurizer spray
- Two distinct phases: 1a and 1b, which differ if liquid or steam space break, 1b ends when stable ECCS recirculation is established, and long-term cooling begins
- Figures of merit are MCHFR and CLL above the core

15.6.5: LOCA

- Limiting case is a 10% CVCS injection line break
  - Incorrect RCS T_{ave} used 548 °F (RAI 9481)
  - Loss of AC power, reactor trip on high RCS pressure and no single failure applied
  - DHRS not credited
  - IAB delays ECCS main valves from opening (~6200s), maximizing inventory loss

15.6.5: LOCA

- Minimum CLL above top of active fuel < 5”, safety analysis limit = 0”
- MCHFR 1.8, CHFR design limit 1.29
- New ECCS logic not reflected
  - Riser level ECCS actuation eliminated
  - Containment vessel (CNV) level ECCS actuation changed from 220” to 264”
  - ECCS logic changes not likely to affect these results

15.6.5: LOCA

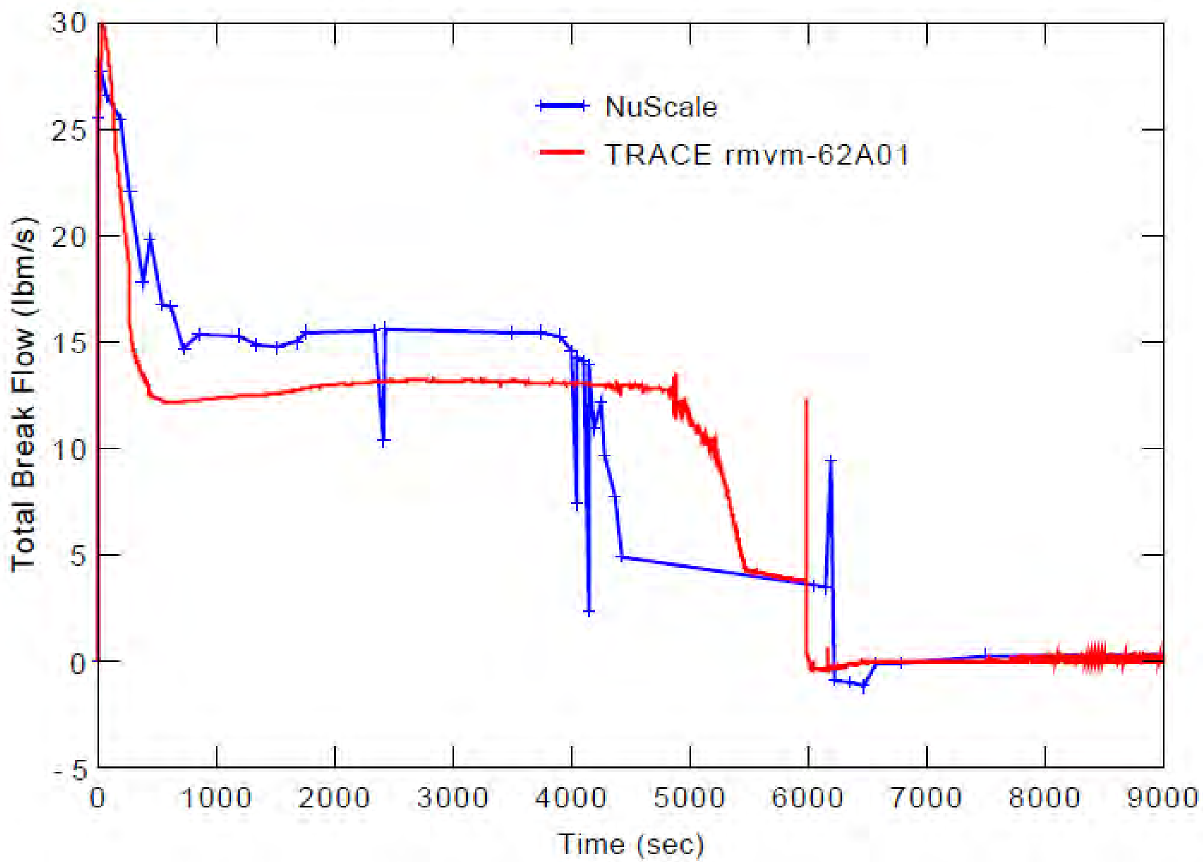
Sequence of events comparison of 10% CVCS injection LOCA (FSAR Rev. 2 – Table 15.6-12)

Event	NRELAP time (sec)	TRACE time (sec)
Line break	0	0
Loss of normal AC	0	0
High pressurizer pressure (2000 psia)	6	10
Reactor Trip System actuation signal	8	10*
Reactor trip	10	12
High containment pressure signal (9.5 psia)	16	10
Containment isolation	20	Not simulated
Low pressurizer level (35%)	323	504
Low pressurizer pressure (1600 psia)	504	151
Low Low pressurizer level (20%)	606	953
High containment water level (220 inches)	2238	3288
Low RCS Level (370 inches)	3100	3306
ECCS injection begins	6181	5981

*Note: high pressure trip delay not modeled per basemodel rev0

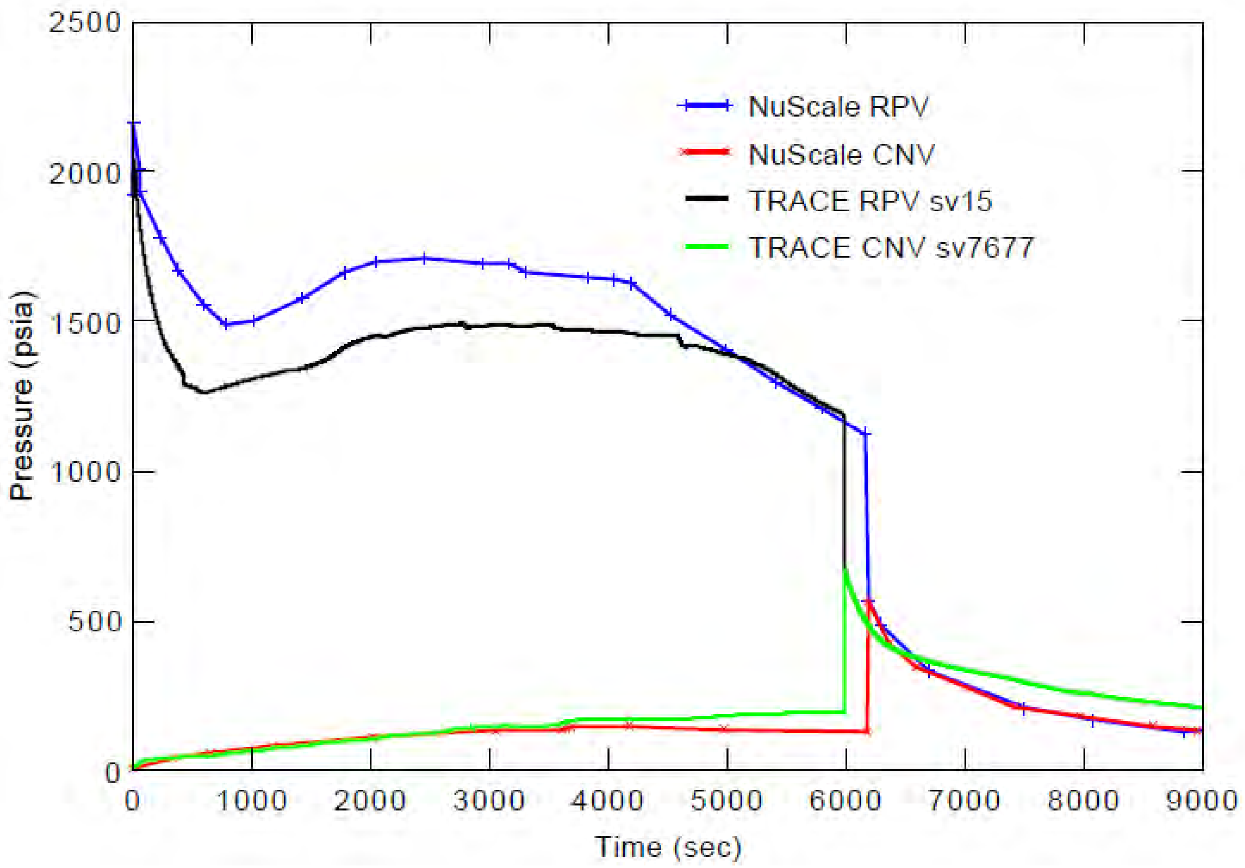
15.6.5: LOCA

Figure 15.6-41: Loss of Coolant Accident - Break Flow



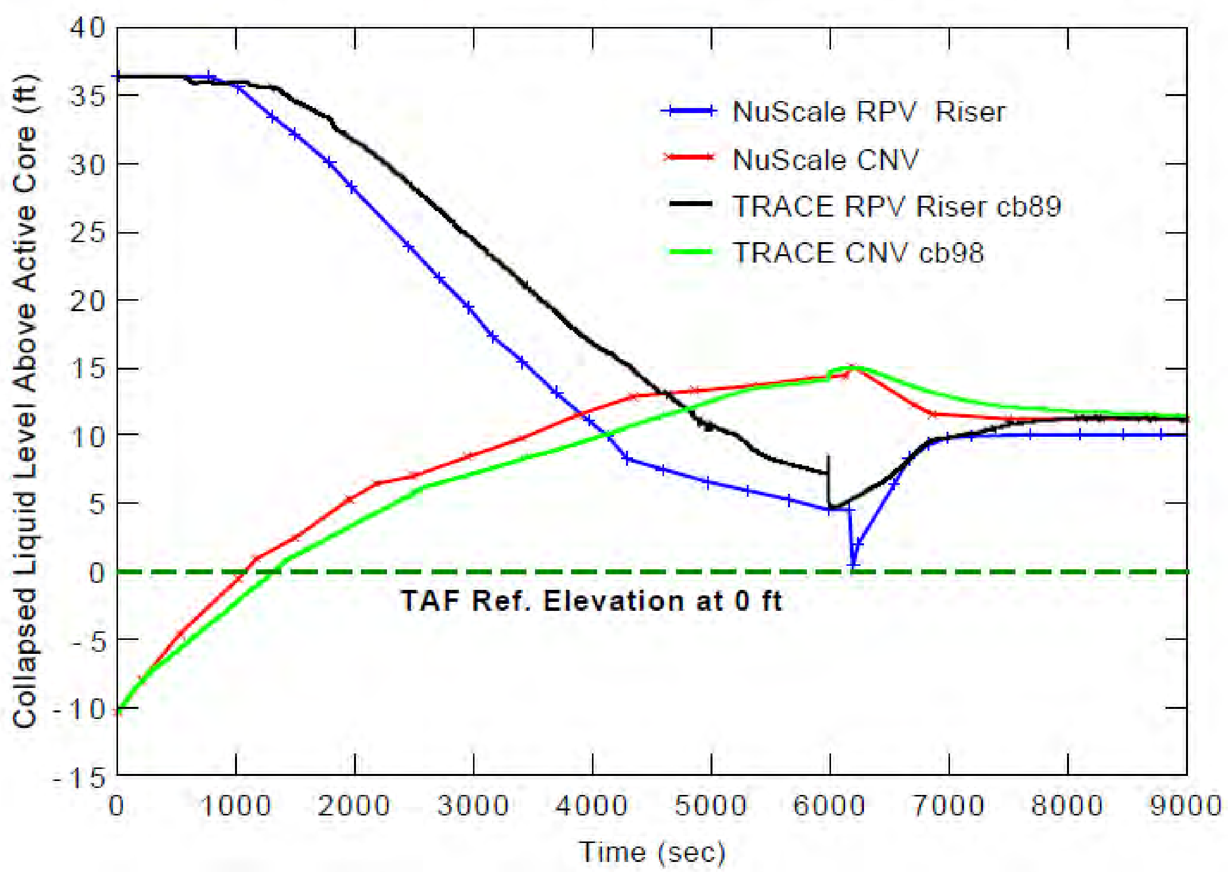
15.6.5: LOCA

Figure 15.6-42: Loss of Coolant Accident - Reactor Pressure Vessel and Containment Pressure



15.6.5: LOCA

Figure 15.6-45: Loss of Coolant Accident - Collapsed Liquid Level Above Active Core



15.6.6: Inadvertent Operation of ECCS

<u>Areas of Review</u>

- Unique Aspects:
  - Limiting Chapter 15 event for MCHFR.
- Methodology
  - LOCA Topical Report, Appendix B: currently under staff review
- Input Parameters, Initial Conditions, and Assumptions
  - The applicant selected suitably conservative initial conditions and biases that maximize consequences with respect to the acceptance criteria.
  - The applicant considered single failure scenarios.
- Analysis Results
  - The applicant provided figures and numerical results for limiting RCS pressure, SG pressure, and MCHFR cases. All are within acceptable limits.

15.6.6: Inadvertent Operation of ECCS

Open Items

- Open Item 15.0.0.5-1 related to whether the IAB is subject to the single failure criterion
- Open Item 15.0.2-2 related to the evaluation of the LOCA Topical Report

15.4.6: Inadvertent Decrease in Boron Concentration

- To mitigate inadvertent dilution events, the CVCS incorporates two redundant safety-related demineralized supply isolation valves. MPS signals include:
  - reactor trip system actuation
  - high subcritical multiplication
  - low RCS flow
- Two evaluations methods used for modes 1-3:
  - Perfect mixing model (slow reactivity insertion, and delays detection)
  - Wave front model (maximum reactivity through slug flow)

15.4.6: Inadvertent Decrease in Boron Concentration

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Results

- Mode 1 (HZP) dilution event is bounded by uncontrolled CRA withdrawal from a subcritical or low power startup condition.
- The limiting dilution event occurs in Modes 2 or 3 with RCS flow greater than 763 gpm. 517 pcm of margin remains after the demineralized water system isolation

Preliminary Staff Assessment

The staff determined that the analysis meets the guidance in SRP Section 15.4.6; however, cannot confirm whether the acceptance criteria for this event are satisfied due to Open Items.

15.2.8: Feedwater Line Break

<u>Areas of Review</u>

- Identification of Causes
  - The applicant analyzed a range of feedwater line breaks (FWLBs) in different locations throughout the system.

- Methodology
  - The applicant used the non-LOCA methodology (Topical Report TR-0516-49416) currently under staff review.

- Model Assumptions, Input, and Boundary Conditions
  - The applicant applied conservative initial parameters in the analysis consistent with the methodology including (un)availability of power systems.
  - The applicant considered and analyzed single failures for each limiting case of the FWLB event.
    - The limiting SG pressure case assumes the single failure of a safety-related check valve. FWLB inside containment credits a non-safety check valve. Credit for non-safety valves has been addressed and is a confirmatory item.

- Evaluation of Analysis Results
  - The applicant provided the reactor power, reactor and SG pressures, core temperatures, RCS and break flow rates, MCHFR, and DHRS heat removal rates.

15.2.8: Feedwater Line Break

Staff Review and Findings

- The staff confirmed that for the worst MCHFR case, the MCHFR remained above the 95/95 DNBR limit.
- The staff confirmed that for the worst RCS and SG pressure cases, the RCS and SG pressure remained below 110 percent of the design pressure.
- The staff confirmed that the radiological analysis of the MSLB bounds the radiological consequences for the FWLB.
- The staff confirmed that the DHRS is safety-related and provides adequate heat removal.
- The staff's conclusion as to the overall acceptability of DCA Part 2, Tier 2, Section 15.2.8, remains incomplete pending satisfactory resolution of the open items.

Open Items

- **Open Item 15.0.2-4:** TR-0516-49416, "Non-Loss-of-Coolant Analysis Methodology", (under review)
- **Open Item 15.0.2-1:** The applicant indicated its intent to update its transient and accident analyses to use NRELAP5 Version 1.4.
- **Open Item 15.0.0.5-1:** NRC staff questioned whether the IAB is subject to the single failure criterion.
- **Open Item 15.0.6-5:** NRC staff questioned the distribution of soluble boron during long-term cooling (RAI 8930, Question 15-27).

15.1.2: Increase in Feedwater Flow

<u>Areas of Review</u>

- Unique aspects:
  - Potential SG overfill scenarios, which could affect DHRS performance
  - Credits non-safety-related feedwater regulating valve (SG overfill case)
- Methodology
  - Non-LOCA EM, currently under staff review
  - VIPRE-01 subchannel analysis
- Input Parameters, Initial Conditions, and Assumptions
  - The applicant considered a spectrum of feedwater flow increases.
  - The applicant selected suitably conservative initial conditions and biases that maximize consequences with respect to the acceptance criteria.
  - The applicant considered single failure and loss-of-power scenarios.
- Analysis Results
  - The applicant provided figures and numerical results for limiting RCS pressure, SG pressure, and MCHFR cases, as well as one figure for SG overfill.

15.1.2: Increase in Feedwater Flow

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Staff Review and Findings

- The non-LOCA methodology is still under review; the subchannel analysis methodology has been found acceptable.
- The applicant's input parameters, initial conditions, and assumptions are acceptable except for the open items below.
- The applicant's analysis results suggest that acceptance criteria are met; however, the staff cannot confirm this until the open items are resolved.

Open Items

- Open Item 15.1.1-1: Potential for SG overfill – staff identified that the case analyzed in DCA Revision 2 may not be the most limiting case. The applicant is revising the calculation to incorporate more limiting cases.
- Open Item 15.0.2-4: Non-LOCA EM (TR-0516-49416) under review
- Open Item 15.0.2-1: Update to use NRELAP5 Version 1.4
- Open Item 15.0.0.5-1: Whether IAB is subject to the single failure criterion

15.2.6: Loss of Non-Emergency AC Power to the Station Auxiliaries

<u>Areas of Review</u>

- Unique Aspects:
  - Diesel generators not credited; decay heat is removed through passive DHRS heat transfer
  - Credits non-safety-related feedwater regulating valve (limiting SG pressure case)
- Methodology
  - Non-LOCA EM, currently under staff review
  - VIPRE-01 subchannel analysis
- Input Parameters, Initial Conditions, and Assumptions
  - The applicant assumed loss of the low-voltage ac power distribution system (ELVS).
  - The applicant selected suitably conservative initial conditions and biases that maximize consequences with respect to the acceptance criteria.
  - The applicant considered single failure scenarios.
- Analysis Results
  - The applicant provided figures and numerical results for limiting RCS pressure, SG pressure, and MCHFR cases.

15.2.6: Loss of Non-Emergency AC Power to the Station Auxiliaries

<u>Staff Review and Findings</u>

- The non-LOCA methodology is still under review; the subchannel analysis methodology has been found acceptable.
- The applicant's input parameters, initial conditions, and assumptions are acceptable except for the open items below.
- The applicant's analysis results suggest that acceptance criteria are met; however, the staff cannot confirm this until the open items are resolved.

<u>Open Items</u>

- Open Item 15.0.2-4: Non-LOCA EM (TR-0516-49416) under review
- Open Item 15.0.2-1: Update to use NRELAP5 Version 1.4
- Open Item 15.0.0.5-1: Whether IAB is subject to the single failure criterion

Containment Performance

- Containment Structure
  - Peak Containment Pressure/Temperature Review
- Containment Heat Removal

6.2.1.1: Containment Structure

Presentation Outline

- Scope of the Staff Review
- Conservatisms in the NPM CNV Model/Initial & Boundary Conditions
- Containment Nodalization Studies for the NPM & NIST-1 Test Facility
- Summary of the NuScale CNV Design and Staff Confirmatory Calculations
- Staff SER Summary for Section 6.2.1.1.

6.2.1.1: Containment Structure

Scope of the Staff Review

- NuScale FSAR Section 6.2.1.1 (DCA Part 2, Tier 2)
  - Containment Response Analysis Methodology (CRAM) Technical Report (TeR)

- Key regulatory requirements
  - Sufficient margin in containment design pressure/temperature (GDCs 16 & 50)
  - Pressure reduction by 50% from its peak value within 24 hours (GDC 38)

- Qualification of the NRELAP5 code to predict the NPM containment response
  - LOCA TR for primary side pipe breaks and reactor valve opening events
  - Non-LOCA TR for MSLB and FWLB events

- No significant differences in physical phenomena b/w the LOCA and valve opening events. LOCA and non-LOCA PIRTs also applicable to the valve opening events and CRAM
  - CRAM is an extension of the NuScale LOCA, valve opening event, and non-LOCA methodologies.
  - CRAM TeR references these methodologies and justifies differences for the CNV response analysis.

- 9 FSAR Chapter 6.2.1.1 RAIs issued with 21 questions in addition to related LOCA TR RAIs

6.2.1.1: Containment Structure

Conservatisms in the NPM CNV Model/Initial & Boundary Conditions [1/2]

- Conservative initial conditions for the spectrum of primary system release events
  - Conservative fuel inputs consistent with those used by the LOCA EM
  - Core power equal to rated thermal power plus uncertainties
  - Conservative decay heat model
  - Maximized RCS volume and fuel stored energy
  - Maximized secondary system stored energy
- Conservative containment free volume (6,000 ft^3)
  - Latest CNV geometry and design changes-> Net CNV free volume allows for RCS thermal expansion, piping, valves, cabling, and miscellaneous components, e.g., platforms and ladders
  - A combined license (COL) item to use the as-built CNV free volume to confirm that the peak calculated CNV pressure/temperature are bounded by the CNV design pressure/temperature values
- Conservative initialization of the CNV inside surface temperature to minimize the condensation rate
  - No temperature drop across the reactor pressure vessel (RPV) wall to maximize radiative heating
  - Adiabatic boundary condition for the CNV upper head above the pool level
  - A conservative ANSYS analysis of the 2-D initial temperature distribution of the CNV wall
  - Peak temp assumed for the entire CNV upper head in the safety analyses

6.2.1.1: Containment Structure

Conservatisms in the NPM CNV Model/Initial & Boundary Conditions [2/2]

- High initial CNV internal pressure of 3.0 psia
  - A maximum CNV internal pressure of 1.2 psia ensures acceptable in-containment leakage detection performance at the revised pool high temperature limit of 110 ^{o}F
  - The peak CNV pressure determined assuming an initial internal pressure of 3.0 psia will be approximately 2 psi higher.
  - Maximizes the initial noncondensable gas mass in the CNV

- Low reactor pool level used in safety analysis
  - Technical Specification 3.5.3 revised to raise the pool level from 55 ft to 65 ft. The minimum 65-ft pool level is only credited for CNV wall temperature initialization and not credited for containment heat removal.

- Conservative condensation heat transfer modeled on the CNV inside surface including the effects of noncondensable gas
  - Conservative condensation heat transfer correlation (More details in Closed Session)
  - Maximum mass of noncondensables in the CNV during operation, and released into the CNV from RPV during an event
  - Deterioration of the interfacial heat transfer to the condensing film on CNV wall caused by the presence of noncondensable gases

6.2.1.1: Containment Structure

<u>Containment Nodalization Studies for the NPM & NIST-1 Test Facility (HP-49 test)---Resolution Status</u>

- Demonstrate NRELAP5's capability to predict the thermal stratification and subcooling of the liquid water accumulating at the lower CNV due to blowdown and condensation of the flashing steam on the CNV wall, and, thus, peak CNV pressure accurately for the limiting design-basis event.
- NuScale submitted nodalization studies for the NPM & NIST-1 test facility (HP-49 test)
  - CNV axial nodalization to demonstrate sufficient fidelity for thermal stratification and subcooling of liquid water in the lower CNV
  - CNV wall radial structure and reactor pool nodalizations
- More details will be provided in the Closed Session about this open item (Open Item 6.2.1-4).

6.2.1.1: Containment Structure

<u>A Summary of the NuScale CNV Design and Staff Confirmatory Calculations</u>

- NuScale DCA
  - Peak calculated CNV pressure of 986 psia results from an inadvertent reactor recirculation valve opening event.
  - NuScale raised the CNV design pressure from 1000 psia to 1050 psia that shows an about 6% margin in the peak calculated CNV pressure of 986 psia. The information regarding raising the CNV design pressure is tracked under Open Item 03.08.02-1.
  - Peak calculated CNV wall temperature of 526 °F results from a double-ended break of the RCS injection line that shows an about a 24 °F margin to the CNV design temperature of 550 ° F.

- Staff's confirmatory calculations show that NuScale CNV peak pressure and temperature are conservative.

- A summary of the staff confirmatory calculations will be presented in the Closed Session.

6.2.1.1: Containment Structure

Staff SER Summary

Once the closure of the current open items is attained:

- 6.1% margin in design pressure (1050 psia vs. 986 psia)
- 24 °F margin in containment wall temperature (550 °F vs. 526 °F)
- CNV pressure drops by 50% of its peak value, well within 24 hours of the initiation of the transient in all design-basis events.

Sections 6.2.1.3 and 6.2.1.4 - Mass and Energy Release Analysis for Postulated LOCA and Secondary Pipe Ruptures inside Containment

Regulatory Basis

GDC 50 requires the containment be designed to accommodate the calculated pressure and temperature conditions resulting from a postulated break.

Staff Review

Staff evaluated the break spectrum, blowdown conditions, and energy sources and determined NuScale either conformed directly with the guidance provided in DSRS Section 6.2.1.3 or used appropriately bounding initial conditions, thereby maximizing the energy release.

Open Item

The staff’s evaluation dependent on the review of the LOCA topical report (Open Item 15.0.2-2), as the referenced analyses rely on the RELAP LOCA model.

Section 6.2.2 – Containment Heat Removal

Regulatory Basis

GDC 35 requires that abundant emergency core cooling to transfer heat from the reactor core following any loss of reactor coolant be provided, including in the presence of any debris generated (including 50.46(b) for long-term cooling)

GDC 38 requires the containment heat removal system be capable of rapidly reducing the containment pressure and temperature following a LOCA and to maintain these parameters at acceptably low levels

GDC 39 and 40 relate to the inspection and testing of the containment heat removal system

Key Design Considerations and Features

NuScale has requested an exemption from GDC 40, which requires testing of the containment heat removal system. For NuScale, containment heat removal is accomplished via passive conduction and convection heat transfer from the CNV to the reactor pool

For long-term cooling, the NuScale containment limits the consequential debris to latent debris, allowing for circulation and fluid communication between the containment and RCS following an ECCS actuation; does not resemble traditional long-term cooling concerns (GSI-191)

Section 6.2.2 – Containment Heat Removal

Staff Review - long-term cooling

NuScale Approach

- Limit containment debris to latent only
- No strainers subject to clogging
- Low fiber limit precludes a filtering bed at the fuel inlet
- Based on WCAP-16530-NP-A
- Exclude material-chemistry combinations expected to cause chemical effects
- Limit total particulate and chemical precipitate to 14 kg (30 lbm)
- Consider deposition on the fuel with 123 kg (271 lbm) chemical precipitates
- WCAP-16793-NP-A approach

Finding

- Using the specified debris limits, including the total fiber and total particulate debris allowed for the design, the staff determined that there is reasonable assurance that consequential debris as specified in the DCA will not impair long-term core cooling functionality.
- The type and amount of assumed chemical effects are acceptable for fuel inlet blockage and deposition on the fuel based on: staff's approval of WCAP-16793-NP-A for testing of fuel inlet blockage and analysis of deposition on the fuel and conservative debris limit with respect to the amount analyzed

Section 6.2.2 – Containment Heat Removal

Staff Review (cont.)

GDC 40 exemption

- Periodic inspections of the containment heat removal surfaces (as required by GDC 39) will assess surface fouling or degradation that could potentially impede heat transfer from the containment
- The other systems that act to remove heat from the containment, such as ECCS are also inspected and tested
- Therefore, the underlying purpose of the rule, to verify that the performance characteristics of the CHRS remain with acceptable parameters and to ensure operability, will still be accomplished

Open item

The limits imposed on the cleanliness program were not clearly identified in the FSAR, and staff cannot make a finding regarding a COL item; as such, staff is tracking Open Item 6.2.2-1, pending a submittal from NuScale to address the lack of clarity in the FSAR to explicitly identify the debris limits in Tier 2.

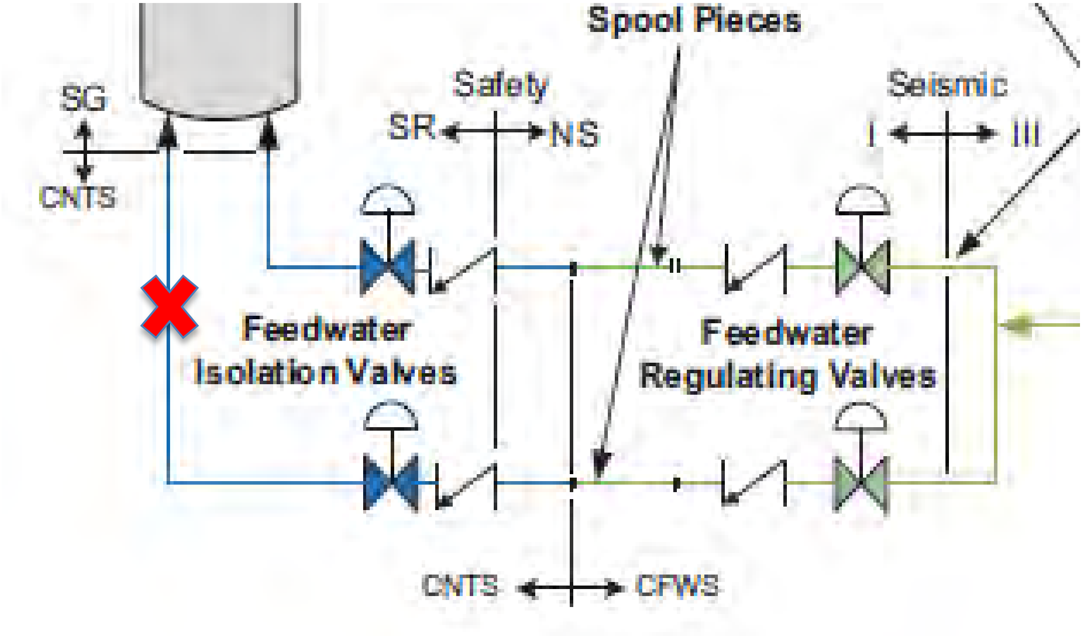
Acronyms

- AC alternating current
- ACRS Advisory Committee on Reactor Safeguards
- BOC beginning of cycle
- CFR *Code of Federal Regulations*
- CHF critical heat flux
- CHFR critical heat flux ratio
- CLL collapsed liquid level
- CNV containment vessel
- COL combined license
- CVCS chemical and volume control system
- DCA design certification application
- DHRS decay heat removal system
- DSRS design-specific review standard
- ECCS emergency core cooling system
- EDSS highly reliable dc power system
- ELVS low-voltage ac power distribution system
- EM evaluation model
- EOC end of cycle
- FSAR final safety analysis report
- FWLB feedwater line break
- GDC general design criterion/criteria
- HZP hot zero power
- IAB inadvertent actuation block
- LOCA loss-of-coolant accident
- MCHFR minimum critical heat flux ratio
- MPS module protection system
- MSIV main steam isolation valve
- MSLB main steamline break
- MTC moderator temperature coefficient
- NPM NuScale Power Module
- PDC principal design criterion/criteria
- PIRT phenomena identification and ranking table
- RAI request for additional information
- RCS reactor coolant system
- RPV reactor pressure vessel
- RRV reactor recirculation valve
- RTP rated thermal power
- SAFDL specified acceptable fuel design limits
- SER safety evaluation report
- SFC single failure criterion
- SG steam generator
- SGTF steam generator tube failure
- SRP Standard Review Plan
- TR topical report

Questions/comments from members of the public before the closed session starts?

Backup Slides

15.2.8: Feedwater Line Break



Michael J Derivan comments/concerns on NuScale FSAR Chapter 15 Safety Analysis

Dear Sirs,
I have been reviewing the NuScale FSAR Chapter 15 Safety Analysis (submitted for the Design Certification) for the 'Reduction in MFW Flow'. It is described in **FSAR 15.2.7.2**. I have some questions and concerns about this analysis in particular and in general how these NuScale Safety Analysis transient results will fit into the use of the NuScale Control Room Simulator used for the NuScale HFE program. The HFE program, in conjunction with the NuScale Simulator, is part of the NuScale plan to justify reduced Licensed Operator manning from what is required in NRC regulations. This is much discussed in other NuScale/NRC documents, and the HFE Program is described in other NuScale submittals.

The crux of my concern with the Reduction in MFW Flow analysis is this:

1. The analysis states the transient is terminated with a reactor high temperature trip (but the specific Th value at time of Reactor Trip is not stated or shown in this analysis. Is it the **Analytical Limit** of 610F as shown in **FSAR Table 15.0-7**?). The Reactor Trip is shown to occur in 690 seconds **(FSAR Table 1.2-21)** after a reduction in MFW Flow from 100% to 97.7% (in 0.1 sec). 690 seconds is 11.5 minutes!
   I want to see this analysis provide enough information in this FSAR analysis section/discussion so I can calculate an RCS water expansion volume (in ft-cu), to independently evaluate the expected increase in Pressurizer Level and RCS Pressure. (Or better yet, NRC Staff do the calc, they just might learn something).
   NOTE: Needed RCS volumes for a calculation are given in FSAR Section 5.
   My concern about this issue is based on my history as a Licensed Operator, and how this type event, coupled with FSAR Safety Evaluation transient understanding and training played into the TMI2 accident.

2. The analysis states conservative assumptions (initial conditions) are used. This analysis states it starts at a T-ave of 555F **(FSAR Table 15.0-6)**. This value is the plus side of the stated 'Uncertainty/Bias' value of plus or minus 10F considering a normal T-ave of 545F. It is not obvious the 'plus' uncertainty is conservative the way this transient has been run. It may intuitively seem starting hotter than normal is conservative with respect to thermal-hydraulic limits of the core, conditions at the core, or even Steam Generator maximum secondary pressure. But this transient basically 'cooks up' the RCS T-ave for 690 seconds until it trips on high Th. With the information provided, it is not obvious to me a worse case would be starting at the opposite 'bias' value with T-ave at 535F because T-ave has farther to run to the Th Trip so the time to Reactor Trip is longer, and the RCS water expansion volume would be larger. (This basically plays into another concern… where is all the RCS water expansion volume going during this RCS heat-up? I can't even find a plot of Pressurizer Level vs Time in the FSAR for this analysis. This expansion of Reactor Coolant is a very significant player in this event progression and final event results. Think TMI2. Further… **(FSAR Table 1.0-2**) under **Event Progression** states:

   > "An AOO should not develop into a more serious plant condition without other faults occurring independently. Satisfaction of this criterion precludes the possibility of a more serious event during the lifetime of the plant."

This particular event is obviously an AOO and I'm not convinced (nor should NRC be) it has been analyzed correctly with respect to increasing RCS Pressure and Pressurizer Level effects on the transient results. Not enough information is provided in the FSAR results to calculate an RCS expansion volume to get a feel for the Pressurizer Level increase. (It's also not my job to do this hand calc, but there is not enough info provided to do it anyway).

3. **FSAR Table 15.0-9 Assumed Single Failures and Credited Nonsafety-Related Systems** states for this event:

15.2.7 Loss of Normal Feedwater Flow… No adverse single failures.

However no discussion (or FSAR reference to) of the Single Failures actually considered for this event is provided (reminds me of TMI2), so again it is just a "CLAIM."

Further from **FSAR 15.2.7.5 Conclusions**:

5) The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified as assumed in the analysis and shall satisfy the positions of RG 1.53.

• No single failures were identified that have adverse impact on the acceptance criteria. Results from this scenario do not challenge the identified limiting parameters as described in this Section.

<u>NuScale should provide a FSAR reference to where in the FSAR this claim is proven.</u>

4. Generically, NuScale hasn't presented enough evidence, in FSAR Chapter 15, to prove to me, (an ex-licensed SRO) nor NRC, that any of their Safety Analysis (SA) assumptions are conservative; they just "claim it". When I was taught SA in the early '70s, Sensitivity Studies for input assumptions were done, and also reported in FSARs in the form of plots of key result vs. input value. Basically several transients obviously had to be run, using varying input assumptions showing the sensitivity (of the results) to particular input assumptions, but only the sensitivity vs. variable plot was shown in the FSARs, not all the actual computer run results. This is key to an Operator understanding the whole process. NuScale needs to either show all the runs in the FSAR or provide sensitivity studies in the FSAR to PROVE their analysis actually use conservative assumptions.

5. In the FSAR Sensitivity Study Analysis discussion section for this event **FSAR 15.9.3.2** it is stated:

   "Stability following reduction of feedwater flow is addressed in this section. A hypothetical rapid decrease in feedwater flow occurs because of feedwater pump speed change, valve alignment changes, or other causes. However, complete loss of feedwater is not considered because it would result in actuation of the MPS and a trip."

   And

   "The response to a rapid reduction in feedwater flow at 32 MW with initial decay heat consistent with 32 MW is stable."

   Why is NuScale limiting the Sensitivity study to a minimum of 32MWt (20% Rx Power) when operation below 20% RTP is part of the design. Additionally below ~15% RTP the Nuscale T-ave (and Pressurizer Level) are on a "ramp", and it is not obvious to me (nor considered by NuScale in the FSAR) that somewhere down the T-ave ramp (as a start

point) would not be a worse case overall scenario for this event. It can also be a player in the Initial Power Escalation Test Program for this plant for ex-core indicated power (and controls) as ex-core Nuclear Instruments require N leakage from the core which is influenced by the water T the leaking Ns pass through to get to the detector. I wonder about this initial condition because a similar event actually happened to me at Davis Besse on Sept 24, 1977. Not only did we end up with no auto Rx Trip, and a stuck open PORV, a full indicated Pressurizer level, a saturated RCS, with HPI off. It is certainly not clear to me, as stated by NuScale, the plant result for the NuScale design is "STABLE." **In fact I don't even believe it!!!**
I am concerned about this plant condition at event initiation because I can add my previous plant SA training was of no use to me, my EOPs based on that SA were of no use to me, and apparently the significance of that DBNPP event was not considered significant enough to provide a warning to the TMI2 Operators. The reason this point IS significant is because the initial crew of NuScale Licensed Operators will have no actual Operational Plant data (or benchmarked Simulator Training), to learn integrated plant response characteristics other than the fictional SA transients fed to the Training Simulator.

I also don't understand NuScale's statement that for this study Decay Heat is consistent with Decay Heat for 32MWt when a core DH is dependent on the total EFPD history of the core, not the core power level at the time of an event.

As an aside, the NuScale SA never uses a positive moderator reactivity coefficient. Does this design never have one, even at BOL as all new "old" design PWRs did (because of the high RCS Boron required on a brand new fresh core)?

6. I'd like NuScale to provide the RCS Loop Transport time at 100% RTP. It is important to know, with respect to how quickly installed T instruments (and controls) can "see" RCS T changes that result from Heat Transfer perturbations initiating in OTSGs. It can explain why PORVs could lift on original B&W 177 FA designs (with original plant setpoints) even before T changes are seen on RCS T instruments, even with an automatic Reactor Trip co-incident with Loss of MFW.

Changing the subject of my concern. It is well understood NuScale plans to ask for changes (reductions) to the required Licensed Operator manning as required in current NRC regulations. It is also well understood NuScale plans to do this based on using the NuScale Simulator for their HFE Program… basically using Simulator runs to show the "Task Analysis" functions, previously identified, required for events (DBAs, AOOs, etc) can be completed with the proposed NuScale Licensed Operator manning. My specific questions/concerns are below:

7. The last NuScale/NRC correspondence I saw concerning the SCOPE of the Task Analysis and this HFE investigation, was it would be limited with-in the scope of the NuScale NSSS Supply. This might imply events such as the **FSAR 15.2.7.2** MFW transients might not be included in this program because the MFW System is NOT with-in the scope of NuScale NSSS supply. Is my assumption correct??? (I still remember TMI2). I know it is not my shot to call…. But if it was "THIS PIG DON'T FLY."

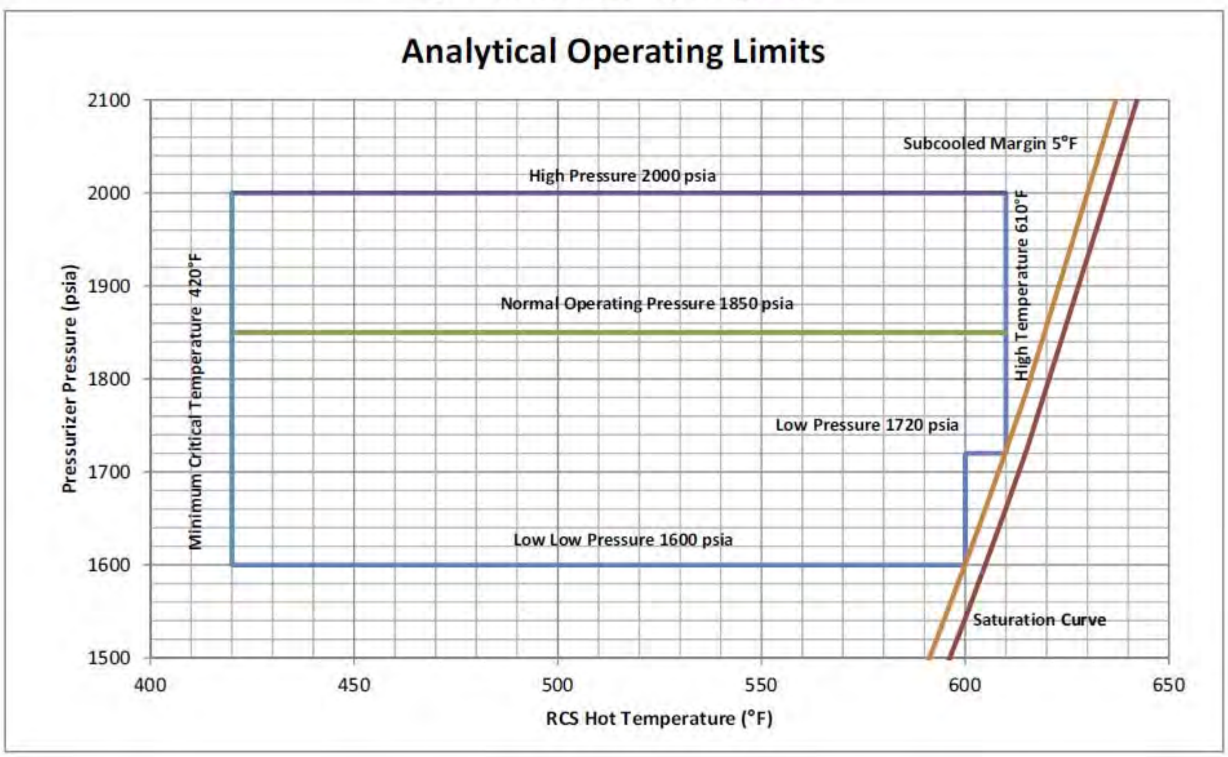
8. The ‘work-horse’ code for the NuScale SA is NRELAP. Those codes don’t run in ‘real time’. What code is running on the NuScale Simulator, which has to run in real time to be a “Training Simulator”, or of any value at all in evaluating Operator performance. (You know where this is going….). Is the NuScale Simulator running NRELAP? The whole point is there is no plant operations/transient data even available for this paper reactor other than what is a figment of someone’s (good) imagination (I acknowledge NRC Staff are reviewing it). What code exactly is this Simulator running that can be used as a basis for reducing Licensed Operator manning below current NRC regulations??? And just why does NRC trust it for something this important?

9. Has the NuScale Simulator been “benchmarked” against anything? If it runs its own “modified” code has it been benchmarked against the FSAR SA runs? If the Simulator is just running the FSAR SA code… you have nothing useful to “justify” anything, much less reduced Licensed Operator manning.

10. Has the NuScale Simulator OTSG (code) performance been benchmarked against the test runs of the OTSG done at the test facility in Italy?

11. FSAR Figure 15.0-9 has this P vs T graph:

Figure 15.0-9: Analytical Operating Limits



A typical PWR (at least a B&W plant) has an SPDS display such as this where RCS P and T plot in real-time and maintain the ‘history trace’. It is an extremely handy tool for Operators to monitor and diagnose plant up-sets.

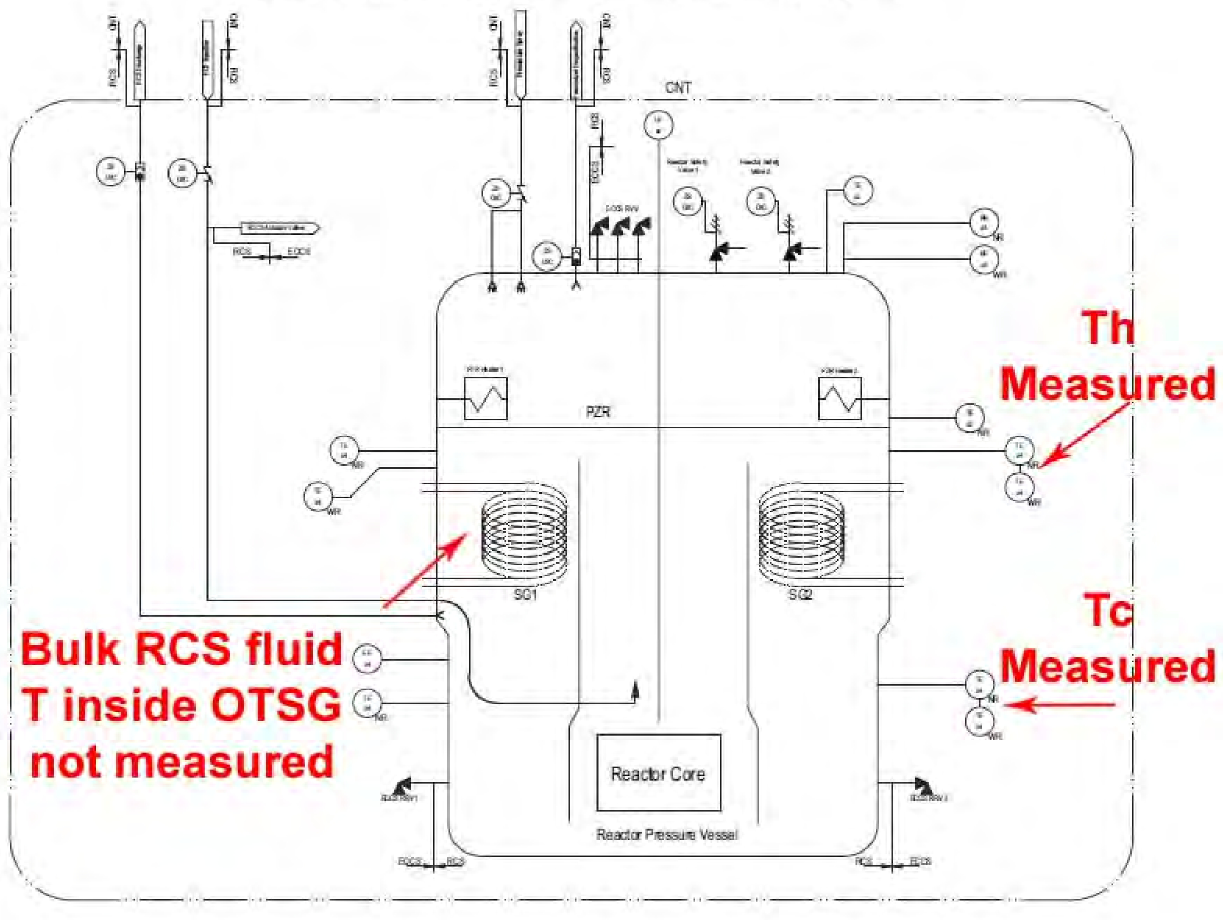
a) I’d like to see the NuScale FSAR ‘Reduction in MFW Flow’ SA transient results plotted on such a graph.
b) Then also add (to the same graph) the same transient results run at the same Initial Conditions as the FSAR SA, for a run on the NuScale Simulator. (Provides a “sanity check” for the 2-codes/model comparison).
c) Then I’d like to see the results for a Simulator run using all the same Input Values… except start with a T-ave using the negative bias value of 535F vice 555F. It would show if the NuScale SA transient IC is indeed “conservative.”

My conclusion and recommendation relative to the needed number of Licensed Operators.
The current published path has the “cart before the horse” on the needed number of Operators issue. This is a FOAK (and strange PWR with OTSGs) unit with no actual operational data available. If NRC can certify the design, independent of the Operator issue… so be it. Then build it, test it, gather the actual operational data (including easily run AOO events, etc.) during the Start-up Test Program and feed that actual plant response data back into the Simulator model/configuration. Then after it is known the Simulator has been benchmarked to actual plant response, and its performance is accurate, run the HFE Program to assess a lowing the number of needed Operators (for additional modules up to the proposed 12-Units). At that point NuScale and NRC can go back and re-address the required number of Operators needed for additional added modules in the long-term with confidence.

I also note that NuScale’s proposed Licensed Operator manning for a proposed 12-Unit full installation (3-ROs and 3-SROs) does in fact meet the current Operator manning NRC regulations if considering just a 1-module NuScale plant, which is all that will be operating at that point in time. Thus there is no “risk” (or delay) proceeding with my suggested path.

Discussion of RCS response to a Reduction in MFW Flow, for a PWR with OTSGs.
In a nut shell:

Figure 5.1-2: Reactor Coolant System Simplified Diagram



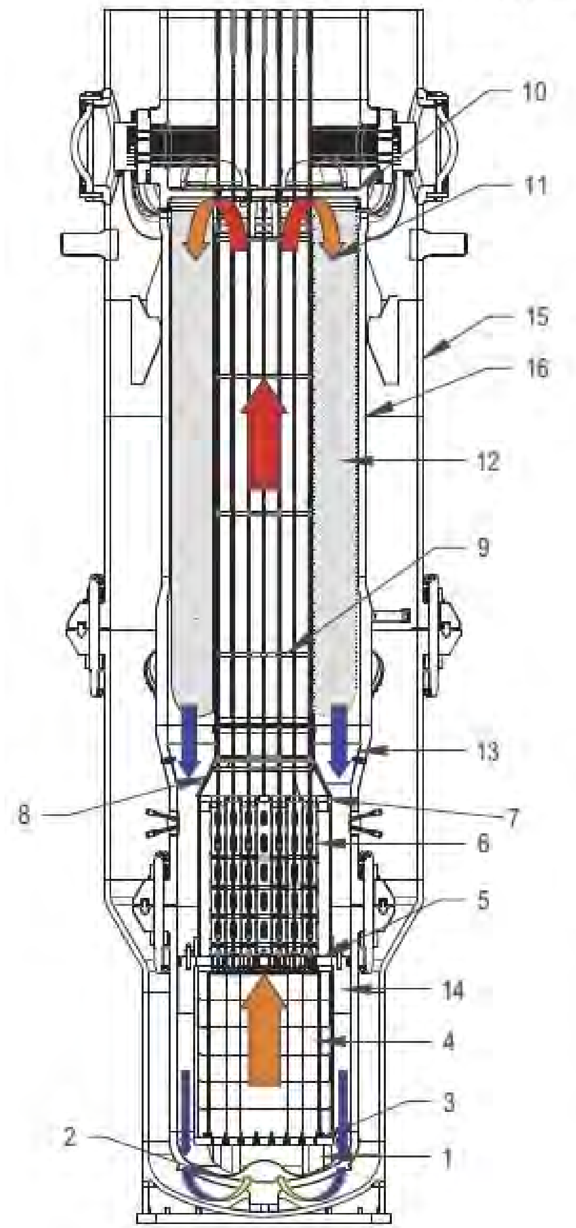
When the MFW Flow is reduced to an OTSG, and the Secondary Side water level starts to decrease (for NuScale OTSG, inside the tubes), the OTSG HT-X superheat region size starts to immediately increase in length. This means less heat is transferred **FROM** the bulk RCS fluid surrounding the OTSG tubes in that larger length HT-X region. Bottom line is the “cold side” T for natural circulation goes up. From just that affect, natural circulation flow rate has to go down. An extreme case example would be secondary side “dry out” from total Loss of MFW. In that case the RCS bulk fluid T around the OTSG tube bundle (inside the OTSG shell) would enter the OTSG shell at Th and exit the shell bottom also at Th since no heat would be removed inside the OTSG (the only OTSG HT-X region remaining would be a full-length superheat region…. I’ll acknowledge it can transfer some heat).
NOTE WELL this RCS T change happens inside the OTSG shell before that hotter RCS water ever moves outside the OTSG to the Tc measuring instruments. In a B&W PWR with OTSGs

this is why RCS P and Pressurize L start immediately increasing (and I mean IMMEDIATELY) even before the Tc instruments (or an Operator) can see it… and is why a B&W PWR is so sensitive to decreases in MFW Flow. The actual T-ave increases as fast as the OTSG water L decreases. But it can't be seen until the water reaches the measuring instruments. It is also why (with the "old" pre-TMI2 setpoints) on B&W plants a PORV would lift EVEN IF the Reactor Tripped simultaneous with a TLOMFW signal.

This is one reason I am curious about the RCS Loop Transport Time for the NuScale Unit…. hotter RCS water has to get all the way to the Th measurement to trip the Reactor on a 'reduction in MFW Flow'. The hotter water also has an effect on the ex-core NI reading as soon as it travels to the core region of the Pressure Vessel because neutron leakage will increase, depending on plant conditions when the NIs were calibrated to a calorimetric.

Another view:

Figure 5.1-3: Reactor Coolant System Schematic Flow Diagram

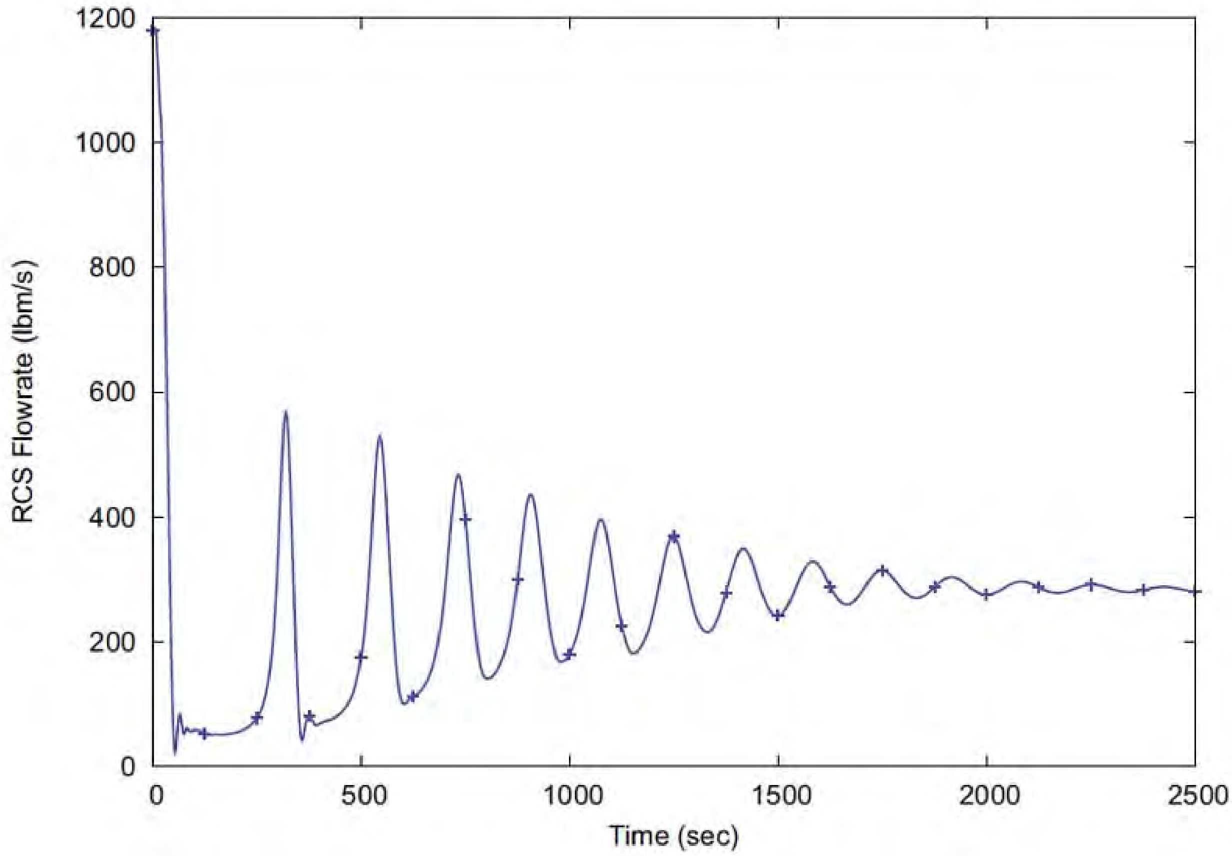


No.	Stage
1	Core support blocks in downcomer
2	Downcomer to lower plenum turn
3	Lower core plate
4	Core
5	Upper core plate
6	Control rod assembly guide tubes
7	Control rod assembly guide tube support plate
8	Riser transition
9	Control rod drive shaft support
10	Pressurizer baffle
11	Upper riser turn to annulus
12	Downcomer through steam generator
13	Downcomer transition
14	Upper core support blocks
15	Containment vessel
16	Reactor vessel

Th is measured up by #11, Tc is measured down by #13, the “cold side’ T for natural circ is the T at #12 (not measured), the “hot side” T for natural circ is the T at the fat red arrow in the riser section. When MFW Flow decreases (or is totally lost as in the SA) and the RCS T at #12 starts to increase, the driving force for natural circ decreases and that flow must go down! Yet it is not discussed in the SA for “Reduction in MFW Flow’.

Further, FSAR Figure 15.2-30:

Figure 15.2-30: Reactor Coolant System Flowrate - Peak RCS Pressure Case (15.2.7 Loss of Feedwater)



From **FSAR 15.2.7.3.3**:

"RCS flow (Figure 15.2-30) drops initially due to the reactor trip and stabilizes as DHRS flow is established.

This statement implies (and shows) RCS flow only decreases due to the Reactor Trip, 18 seconds after the event initiator **(FSAR Table 15.2-20)**, not the loss of the natural circ driving head due the bulk fluid RCS T increase within the OTSG shell heat sink from the Secondary side dry-out after the 100% TLOMFW initiator. Eighteen seconds is more than enough time to change the Ts driving the delta-P for natural circ, and affecting natural circ flow well before the Reactor Trip. After all… some RCS Ts must be changing in these 18 seconds…. as the Pressurizer in-surges enough from RCS heat-up expansion to terminate the event on an HP trip!

I don't think THIS OPERATOR believes RCS natural circ flow doesn't change before the Reactor Trip, based on just the discussion in this FSAR Section. Either the event is not being thoroughly explained in the text, or the NRELAP code is bogus.

In my wildest imagination I might conjure up an explanation claiming that as RCS bulk T inside the OTSG shell increases due to loss of Secondary heat sink, and as natural circ flow "tries" to decrease, it causes Reactor core flow to decrease causing a corresponding increase on the "hot side" T… and then "a miracle occurs" and magically the delta-P driving force for natural circ stays constant so the flow stays constant. If true an Operator needs to know that.

But I don’t think Occam would believe that explanation either.

Thanks for your time and consideration,
Mike Derivan.