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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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665TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

OPEN SESSION

+ + + + +

WEDNESDAY

JULY 10, 2019

+ + + + +

ROCKVILLE, MARYLAND

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The Advisory Committee met at the Nuclear
Regulatory Commission, Two White Flint North, Room
T2B10, 11545 Rockville Pike, at 8:30 a.m., Peter
Riccardella, Chairman, presiding.

COMMITTEE MEMBERS:

- PETER RICCARDELLA, Chairman
- MATTHEW W. SUNSERI, Vice Chairman
- MICHAEL L. CORRADINI, Chairman
- DENNIS BLEY, Member
- CHARLES H. BROWN, JR. Member
- VESNA B. DIMITRIJEVIC, Member

1 WALTER L. KIRCHNER, Member
2 JOSE MARCH-LEUBA, Member
3 DAVID PETTI, Member
4 JOY L. REMPE, Member
5 GORDON R. SKILLMAN, Member

6

7 ACRS CONSULTANT:

8 STEPHEN SCHULTZ

9

10 DESIGNATED FEDERAL OFFICIAL:

11 MICHAEL SNODDERLY

12

13 ALSO PRESENT:

14 CLINTON ASHLEY, NRO

15 RUFINO AYALO, NuScale

16 ANTONIO BARRETT, NRO

17 BRUCE BAVOL, NRO

18 AMBER BERGER, NuScale*

19 MARTY BRYAN, NuScale

20 KEVIN COYNE, NRO

21 TIM DRZEWIECKI, NRO

22 YUSUF FARAWILA, NuScale*

23 RANI FRANOVICH, NRO

24 ANNE-MARIE GRADY, NRO

25 SYED HAIDER, NRO

1 STEVE HAMBRIC, NRC
2 OLIVIA HAND, NuScale*
3 JOHN HONCHARIK, NRO
4 BHAGWAT JAIN, NRO
5 NADJA JOERGENSEN, NuScale
6 REBECCA KARAS, NRO
7 SELIM KURAN, NuScale*
8 RENEE LI, NRO
9 TIM LUPOLD, NRO
10 GREG MAKAR, NRO
11 MEGHAN McCLOSKEY, NuScale*
12 RYAN NOLAN, NRO
13 REBECCA NORRIS, NuScale
14 MATTHEW PRESSEN, NuScale
15 ZACKARY RAD, NuScale
16 HANNAH ROOKS, NuScale*
17 SUJIT SAMADDAR, NRO
18 PRAVIN SAWANT, NuScale*
19 THOMAS SCARBROUGH, NRO
20 JEFF SCHMIDT, NRO
21 RAY SKARDA, RES
22 OMID TABATABAI, NRO
23 BOYCE TRAVIS, NRO
24 ANDREA VEIL, ACRS
25 MARIELIZ VERA, NRO

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ROBERT WEISMAN, OGC
BRIAN WOLF, NuScale*
YUKEN WONG, NRO
PETER YARSKY, RES

*Present via telephone

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P R O C E E D I N G S

8:37 a.m.

1
2
3 CHAIR RICCARDELLA: The meeting will now
4 come to order. This is the first day of the 665th
5 meeting of the Advisory Committee on Reactor
6 Safeguards. I'm Peter Riccardella, Chairman of the
7 Committee.

8 The ACRS was established by the Atomic
9 Energy Act and is governed by the Federal Advisory
10 Committee Act, or FACA. The ACRS section of US NRC
11 public website provides information about the history
12 of the ACRS and provides FACA-related documents, such
13 as our charter, bylaws, Federal Register notices for
14 meetings, letter reports, and transcripts of all full
15 and subcommittee meetings, including all slides
16 presented at the meetings.

17 The Committee provides its advice on
18 safety matters to the Commission through its
19 publically available letter reports. The Federal
20 Register notice announcing this meeting was published
21 on July 1, 2019 and was revised on July 8, and
22 provided an agenda and instructions for interested
23 parties to provide written documents on request, or
24 request opportunities to address the Committee as
25 required by FACA.

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1 In accordance with FACA, there is a
2 Designated Federal Office, DFO, for today's meeting.
3 The DFO for this meeting is Mr. Michael Snodderly.

4 (Off-microphone comments.)

5 CHAIR RICCARDELLA: Okay, the DFO for this
6 meeting is Mr. Michael Snodderly. During today's
7 meeting, the Committee will consider the following:
8 NuScale's design certification application Chapters 3,
9 6, 15, 20, and the Stability Topical Report, and two,
10 preparation of ACRS reports.

11 As reflected in the agenda, portions of
12 the sessions on NuScale design certification
13 application and stability report made be closed in
14 order to discuss and protect information designated as
15 sensitive or proprietary. There is a phone bridge
16 line. To preclude interruption of the meeting, the
17 phone will be placed in a listen-only mode during the
18 presentations and Committee discussions.

19 We have received no written comments or
20 requests to make oral statements from members of the
21 public regarding today's sessions. There will be an
22 opportunity for public comment as we have set aside
23 ten minute in the agenda for comments from members of
24 the public attending or listening to our meetings.
25 Written comments may be forwarded to Mr. Snodderly,

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1 the DFO.

2 A transcript of the open portions of the
3 meeting is being kept, and it is requested that the
4 speakers use one of the microphones, identify
5 themselves, and speak with sufficient clarity and
6 volume that they can be readily heard. And I would
7 also ask everyone to silence cellphones and other
8 devices to avoid interruption of the meeting.

9 And with that, I will turn the meeting
10 over to Michael Corradini.

11 MEMBER CORRADINI: Thank you, Mr.
12 Chairman. So to remind the members where we are, this
13 is the final set of chapters from the draft -- I'm
14 sorry, from the DCA, as long, as well as the draft
15 safety evaluation reports. So we're going to go
16 through first the Stability Topic Report, and then we
17 will proceed on to the other chapters.

18 PARTICIPANT: Anyone here?

19 PARTICIPANT: Yes, NuScale Corvallis can
20 hear you.

21 PARTICIPANT: Okay.

22 (Off-microphone comments.)

23 MEMBER CORRADINI: All right, so let's try
24 again. So today we have a number of things on the
25 agenda. We first want to talk about the topical

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1 report which was submitted by NuScale and the
2 associated SE, and then we'll follow up with the final
3 four chapters of the DCA Revision 2.

4 Okay, so why don't I turn this over to
5 Zack. I'm sorry, excuse me, I'm going to turn it over
6 to Matthew from NuScale, excuse me. Matthew.

7 MR. PRESSEN: Quite all right. Thank you
8 all and good morning, I am Matthew Pressen, Licensing
9 Specialist with NuScale Power and Project Manager for
10 this topical report.

11 If we could move to the next slide. We
12 are here today to discuss the evaluation methodology
13 for stability analysis of the NuScale power module.
14 Slide 2, sorry.

15 MEMBER REMPE: You guys, is this the only
16 way we can see the slides? We're going through
17 Internet Explorer. Did they get downloaded to this
18 computer?

19 MEMBER CORRADINI: No, it's being run by
20 the control room.

21 MEMBER MARCH-LEUBA: But the control room
22 needs to download the file, not run it on Acrobat.

23 MEMBER CORRADINI: If we can suffer
24 through this, let's just keep on going.

25 MEMBER BLEY: Well, the two side screens

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1 are working.

2 MEMBER CORRADINI: We have hard copy.

3 MEMBER SUNSERI: Is your mic close to you?
4 Because with this background noise, it's hard to hear.
5 You're quiet.

6 MR. PRESSEN: So presenting today will be
7 Dr. Yusuf Farawila over the phone. He is the NuScale
8 System Thermal Hydraulic Stability Expert. He will be
9 supported by myself, Licensing, and Zack Rad. With
10 that, we will pass this presentation off to Dr.
11 Farawila. And if we could move to slide 3.

12 MEMBER SUNSERI: Ask your presenter to
13 speak.

14 MR. PRESSEN: Yeah, can Corvallis hear us?
15 So looks like the Corvallis line cannot hear us
16 anymore.

17 MEMBER SUNSERI: Did they mute as
18 requested? They need to unmute.

19 MR. PRESSEN: They did, but they are not
20 hearing us.

21 CHAIR RICCARDELLA: Let's take a five-
22 minute recess. We'll reconvene at ten to nine.

23 (Whereupon, the above-entitled matter went
24 off the record at 8:45 a.m. and resumed at 8:52 a.m.)

25 MR. FARAWILA: Good morning, everyone,

1 this is Yusuf Farawila here from NuScale. And I'm
2 going to try to, we are on slide 3, so we are going to
3 try to do this as fast as we can, maybe we can make up
4 some of the lost time.

5 Our agenda today is just to make an
6 introduction in the new methods and introduce our
7 stability solution and provide and for us to start
8 work that led us to that point. Both critical,
9 numerical, and experimental benchmarking. Some of
10 that will sure have to wait till the closed session.
11 And we'll introduce also our procedure methodology and
12 take your questions.

13 Slide 4, our main message is that the
14 NuScale Power model design was investigated for
15 stability and found to be stable in its entire range
16 of normal operation. Even outside of normal
17 operation, the reactor can be de-stabilized when the
18 riser flow is boiling. So we have identified
19 stability thresholds.

20 However, this unseasoned flow is limited
21 by non-linear effects so that even operations of that
22 type when the stability threshold is crossed is also
23 denied.

24 That stability threshold is protected by
25 scram, a full loss of riser in with the cooling. And

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1 that constitutes region exclusion. So not a detector
2 of resolution type. Because scram was already there
3 for other reasons other than stability, there is no
4 action required for cooling in the hardware that is
5 indicated (inaudible) instability.

6 PARTICIPANT: Can't hear anything.

7 MR. FARAWILA: These conclusions are based
8 on extensive analysis with first principles,
9 experimental and computational studies and we'll try
10 to include some of that in the next slide.

11 Go to slide 5.

12 (Simultaneous speaking.)

13 MR. FARAWILA: We looked at the literature
14 and found out that (inaudible) stability was reported
15 in the literature. And you can see certain types
16 there.

17 PARTICIPANT: I'm sorry. We're being
18 asked to pause and have everyone go on mute.

19 MR. FARAWILA: Okay, all right. All
20 right, this is Yusuf Farawila again on slide 5.
21 Presenting an outline of an instability --

22 (Simultaneous Speaking)

23 MR. FARAWILA: -- literature. You see the
24 location of the heater and the cooler can affect the
25 stability, you can see four regimes there. Only the

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1 first one could go unstable. And our modeling looks
2 like one of the most stable configurations. But --

3 (Off-microphone comments.)

4 MEMBER CORRADINI: So if could make a
5 suggestion. My suggestion to the Chairman is we
6 essentially adjourn till 9:15. We get this settled.
7 Keep Farawila on the line as the test case and get it
8 cleared up and not come back until we're --

9 MEMBER REMPE: And a test case with a
10 staff member acting as the public would be a good idea
11 too.

12 MEMBER CORRADINI: That's fine. So that's
13 my suggestion so we can clear this up, otherwise we're
14 just going to be fussing around.

15 CHAIR RICCARDELLA: Okay, the meeting is
16 recessed until 9:15, and hopefully we will be able to
17 reconvene at that time.

18 MEMBER CORRADINI: I volunteer to be the
19 public if Dr. Farawila --

20 PARTICIPANT: Okay, NuScale Corvallis is
21 back on the line and we're being requested to ask
22 everyone else to go on mute. We think we heard some
23 things from the meeting.

24 CHAIR RICCARDELLA: This is Pete
25 Riccardella, ACRS Chairman.

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1 PARTICIPANT: But now we're not hearing
2 anything again.

3 CHAIR RICCARDELLA: Okay, so tell her
4 we're going to recess the meeting until 9:15 East
5 Coast time. And we'll convene at that time, and
6 hopefully everything will be working.

7 (Whereupon, the above-entitled matter went
8 off the record at 8:57 a.m. and resumed at 9:30 a.m.)

9 MR. FARAWILA: All right, this is Yusuf
10 Farawila again, sorry for the problem. You are on
11 slide 4.

12 Presenting you with the main message, and
13 that is the NuScale Power module has been examined for
14 instability and how to be stable in the entire range
15 of its normal operation. Also outside of normal
16 operations, the reactor can be destabilized if the
17 riser flow (inaudible).

18 But even then, this unstable flow
19 oscillation amplitude is limited by nonlinear effect
20 and there is no critical heat flux ratio that is,
21 could actually improve. So that kind of instability
22 is also benign.

23 That stability threshold, which is voiding
24 the riser, is protected by scram, which is actuated
25 upon loss of inlet risers of cooling with a margin.

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1 So that's contextually equivalent to the traditional
2 region exclusion. The region is defined only that
3 line of the riser inlets of cooling, not a two-
4 dimensional one like BWRs.

5 And because this protection action is
6 there for other purposes, it's not made specially for
7 stability. So there is no hardware solution that is
8 required for stability. So the way we reached these
9 conclusions using first principles for that experiment
10 and computation studies are prudent.

11 We are in slide 5. I will present you with
12 an example to be found in the literature where metric
13 instabilities were reported in configurations of where
14 the cooling the heat sink and heat source are placed
15 around the loop. And that affects the stability.
16 Only the first kind with the heater in the extreme
17 bottom and the cooler in the very top is the most
18 unstable and instabilities were reported.

19 And our model is more like the last figure
20 there with a cooler on the upper side and the heater
21 is on the lower position of the other side. But we
22 could not just take this for granted and we went ahead
23 and made this extensive evaluation.

24 You go to slide 6, and we see how we
25 investigated this starting from the gray area. We

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1 just had a design and operational domain. We are
2 fortified with first principles and theory and
3 experience. We have been done a preliminary purge to
4 start with. And then we moved into constructed models
5 for the main code, which is code 10.

6 And on both sides of that box you will see
7 that we also did independent models using reduced
8 order modeling and also conducted experimental
9 investigation.

10 So using thus qualified code to
11 (inaudible) with a steady state of deviations and
12 stability during trending. And we identified where
13 the threshold is and reached the conclusion that the
14 model is stable within operating domain. So that
15 comes, brings us to the bottom box, where we
16 identified the stability solution area, just simply
17 protecting the riser's cooling margins and the
18 hardware is already in place.

19 Go to slide 7. The theoretical
20 investigation started with the extensive first
21 meeting. And we did a scope and review of thermal-
22 hydraulic instabilities to see which ones applied.
23 And we identified what possible mechanisms of
24 instabilities are, we used direct first principles to
25 study parts of the model, like riser-only mode, what

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1 will happen. And when you put the steam generator,
2 what kind of trends this can be for stability.

3 And so we could also show, and that part
4 was kind of raised in the Subcommittee meeting, that
5 we could also show that steam generator dynamics are
6 decoupled from the stability of that primary loop.
7 And I think that's important for -- important outcomes
8 from our theoretical investigations.

9 And these theoretical investigations also
10 used to inform the design of the stability experiment.
11 Mentioned in the third, there's always a subjective
12 judgement which a phenomena of how we rank each
13 phenomena are ranked medium or less. So we, as a
14 rule, rated all medium-ranked phenomena in modeling as
15 if they were highly ranked.

16 Go to slide 8. Just giving you an outline
17 of what we mean by instability, what are the required
18 conditions from first principles for a system to go
19 unstable. First, we have to have to have -- identify
20 negative feedback processes are there. Positive
21 feedbacks are not allowed.

22 So the negative feedbacks are present, but
23 they need to be delayed and they need to be
24 sufficiently strong. Because when a negative delay,
25 a negative feedback is delayed, then the system can

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1 overshoot in the other direction. And if that's
2 sufficiently strong, we get oscillatory instability.

3 So in our particular case, the feedback is
4 negative because any perturbation increase in core
5 flow will give you a force that will reduce it,
6 essentially defeating the initial perturbation. But
7 that feedback is delayed because it takes time for
8 this perturbation, like a changed temperature at the
9 core exit, to fill the riser before it will have an
10 impact on the (inaudible) that drives the flow.

11 So if the feedback is strong enough then
12 that is when the quantitative analysis is required.
13 So we need to look at all the characteristics of, you
14 know, the riser lens and the possibility of
15 (inaudible) change. Because if you receive a change,
16 then you have a stronger response that may become
17 unstable and this is something -- really something
18 actually.

19 Let's go to the next slide, slide 9. Just
20 introducing you to the main stability analysis tools,
21 PIM, which was optimistically named PIM, when means
22 pendulum in molasses. Because this module has proven
23 to be very stable, particularly under normal
24 operation.

25 You see a picture of the module, then see

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1 a more abstract picture, and then that's a diagram
2 that describes the geometry of the code. The core in
3 the middle and risers in the riser. And then in the
4 cold leg, you see the steam generator where the second
5 side acts as a heat sink.

6 We can go through this equation quickly,
7 we don't have to recite them in slide 10. So the
8 model equations are essentially having the thermal-
9 hydraulic conservation equations for liquid and vapor
10 mass balance and mixture momentum and energy
11 conservation, also assuming that the vapor is always
12 saturated.

13 So go to the next one, slide 11. We
14 applied (inaudible) kinetics in the core. So because
15 the core is rather small and very much in the bottom,
16 so there are no three-D effects that are of interest.
17 And we also modeled the reactivity input to that --
18 kinetics, which come from the thermal-hydraulic model
19 and come from also the Doppler Effect.

20 So there's always this interaction between
21 the thermal-hydraulics and the dynamics in the core.
22 Go to the slide 12. There are other models also that
23 need to be implemented in order for all these
24 phenomena to get coupled. And that's heat conduction
25 in the fuel rods, and we have heat transfer models.

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1 Particularly important is what happens in the steam
2 generator.

3 Slide 13. For any thermal-hydraulic model
4 like that, (inaudible) relations and correlation for
5 frictional pressure drop. We used the drift flux
6 formulation for the flip between the vapor and liquid
7 stages. Remember here that we are mostly (inaudible).
8 There is the possibility of subcooled boiling, but
9 since the code is applicable to beyond the normal
10 operation, we allow for vapor formation at any amount.

11 We can go with the riser to any amount.
12 So a general model for two-phase flow is required and
13 that's what is there. And so we have a equilibrium
14 evaporation and condensation model to account for all
15 of that. And the rest of the properties for water and
16 other physical materials and all the reactivity
17 coefficients that also should come from approved
18 neutronics codes.

19 Important to note that when you model, you
20 have to essentially not model everything. Some of the
21 things here are not modeled. We are not modeling the
22 pressurizer because of the permanent effect on the
23 momentum balance. And we make assumptions also about
24 heat transfers through the riser wall if we assume
25 it's (inaudible).

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1 And so all these assumptions that we use,
2 and for the -- to model something, it's either shown
3 to be concerned with its full stability, or the
4 auxiliary (inaudible).

5 Next slide, 14. We examined also the
6 effect of the flow stability in the steam generator
7 tubes. Because now you look at the secondary side and
8 see you have boiling channels inside the steam
9 generator, and will these be unstable or not.

10 And we could perhaps talk more about that
11 in the closed session, but the flow in these tubes is
12 subject to this fluid instability. And we have
13 experiments, and these experiments succeed to
14 demonstrate that these instabilities are possible.

15 However, the most important part is that
16 the observations in different tubes. You have over a
17 thousand tubes. They are not phase-locked, so with
18 random phase or out of phase, they do not feed back to
19 the primary.

20 Actually, the primary does not feed back
21 to that secondary either, because you cannot have a
22 perturbation in the flow in the cold leg in one
23 direction that affects two different tubes in opposite
24 directions. So that's not also possible.

25 That makes a case for having no impact on

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1 the primary flow for any possible instabilities that
2 can happen in the steam generator tubes.

3 PARTICIPANT: Before you go, when you're
4 ready to go to the next slide, Matthew says there's a
5 question.

6 MR. FARAWILA: Okay, I'll take the
7 question.

8 MEMBER MARCH-LEUBA: That's my cue --

9 MR. FARAWILA: Slide 14.

10 MEMBER MARCH-LEUBA: And you will relay it
11 to him, right? The third bullet says that the
12 experiments demonstrate for oscillations and they're
13 generally low flow. However, with -- is this a true
14 statement? Because we've seen other data that
15 indicates that this unstable in the full operating --

16 (Off-microphone comments.)

17 MEMBER MARCH-LEUBA: There. If you want
18 to simplify the question is do the experiments
19 demonstrate instabilities outside free flow?

20 MR. FARAWILA: I'm not sure I understood
21 the question, but it may be a question about when --
22 okay. I'm going to ask you what the question is.
23 When is the steam generator tube stable or unstable?

24 MEMBER MARCH-LEUBA: Yeah.

25 MR. FARAWILA: So basically, the way we set

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1 it in general, is that the question?

2 PARTICIPANT: Yeah.

3 MR. FARAWILA: Okay.

4 PARTICIPANT: That's the question.

5 MR. FARAWILA: Hold on.

6 (Off-microphone comments.)

7 MR. FARAWILA: Can we defer the answer to
8 that question to the closed session?

9 MEMBER MARCH-LEUBA: Okay. Okay, so we'll
10 wait for the closed session.

11 MR. PRESSEN: And yes, we do have a number
12 of slides covering the experiments in the closed
13 session, so.

14 MR. FARAWILA: Let me speak in the general
15 sense right now and tell you that for basically wave
16 oscillations in a boiling channel, you need to have a
17 generally high power to flow ratio. So everything
18 else being equal.

19 All right, may I proceed to slide 15?

20 (Off-microphone comments.)

21 (Pause.)

22 MR. FARAWILA: All right, this is Yusuf
23 Farawila again in Corvallis. We are on slide 15, and
24 we wanted to present in this slide how PIM is used to
25 calculate stability. So what we do is we perturb a

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1 (inaudible) solution and see how the cold responds.
2 So for several power levels and different modes of
3 operation at different exposures, things like this, we
4 --

5 PARTICIPANT: We want to move to slide 18.

6 MR. FARAWILA: All right. Oh right, slide
7 18, that's summary and conclusions.

8 All right, so to conclude, the stability
9 of the nuclear module was evaluated using a dedicated
10 code and supported by first principles and
11 experimental data benchmarking, and was found to be
12 unconditionally stable with the normal operation. And
13 the safety boundary was identified and predicted. And
14 best protection is a scram when the risers' subcooling
15 margin is reached within a margin.

16 And that essentially concludes the
17 presentation. And if you have any questions.

18 MEMBER KIRCHNER: I have a question. Have
19 you used this set of models to look at ATWS and what
20 is the impact of an ATWS on the stability of the
21 system?

22 MEMBER MARCH-LEUBA: While you type it in,
23 he means a beginning of cycle ATWS with a very low
24 density of activity coefficient.

25 MR. FARAWILA: Yeah, I mean it's a

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1 question about the anticipated plan without scram. So
2 essentially, the way we reach the stability threshold,
3 its protected by scram. So we do not have the scram
4 in order to allow the oscillations to proceed. So
5 that's considered essentially a transient without
6 scram. And I refer to that in slide 3 or something.

7 I think that is in slide 4, when we stated
8 that unstable flow oscillations' amplitude is limited
9 by non-linear effects. And their amplitude does not
10 result in any deterioration of the critical power
11 performance. And so that's considered an ATWS because
12 the real system with a scram, and the code did not
13 scram.

14 MEMBER KIRCHNER: Well, that doesn't quite
15 answer the question.

16 MR. FARAWILA: Yeah, there was that
17 question about the moderator feedback, the reactivity
18 feedback. In the beginning of cycle when the
19 moderator feedback, it's least stabilizing with
20 specific oscillation for end of cycle, when the
21 moderator coefficient is very negative. In some
22 calculations we don't get any instabilities.

23 Or if we do get instabilities when you
24 reach highly voided riser state, the outcome is the
25 same because the stabilization is really in the

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1 hydraulics. Because the swing of oscillations, in the
2 upper part of the oscillation, the void tube
3 collapsed. Once the void collapsed in that part, the
4 downswing cannot exceed that. So that is limiting
5 what we can measure for the oscillation.

6 MEMBER KIRCHNER: Well, the BWR fleet has
7 a problem with ATWS and operator procedures to respond
8 to that. I, and that's for circulation.

9 MEMBER CORRADINI: He can't hear you,
10 Walt. You're going to have to formulate it for --

11 MEMBER KIRCHNER: I'm trying to think of
12 a shorthand way. BWRs have this kind of issue under
13 ATWS conditions, and they are force circulation. I
14 intuitively expect it will be amplified in a natural
15 circulation.

16 MR. FARAWILA: I just mentioned that ATWS
17 with instability is problem with BWRs. And I will be
18 coming next month to address your committee on one of
19 those. Fortunately, NuScale reactor is not a BWR, and
20 if you allow instabilities to happen, they are so
21 benign that critical heat flux ratio actually
22 improves.

23 Because when you start moving the riser,
24 the steady state average flow increases so much
25 because you have so much more things to be had that

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1 the operations around that, you're not consumed the
2 increase that you get. So you get an improvement of
3 CHFR (phonetic).

4 I know that this kind of improvement that
5 we obtained from a generic combination that's not part
6 of the approval process, but it's generic enough. And
7 we know from first principles that this not be
8 correct. That we do not have any challenge to CPR.

9 In boiling water reactors, for example, we
10 know that severe oscillations can happen if not, if we
11 fail to scram, and you agree to have dry-out
12 definitely, and dry out in cyclical, dry out and
13 waiting.

14 And the operator has to ask to lower the
15 water level in order to avoid having the clad
16 temperature exceed limits. But we have absolutely no
17 issue in, of similar nature in the NuScale reactor.

18 MEMBER CORRADINI: Thank you. That's
19 good, that's good enough. Walt, you have a follow-up?
20 Are you okay for the moment?

21 MEMBER KIRCHNER: Okay for the moment.

22 MEMBER MARCH-LEUBA: I wanted to apologize
23 to all the members and to the readership of this
24 transcript that we have technological difficulties.

25 MR. FARAWILA: If there are no other

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1 questions.

2 MEMBER CORRADINI: Tell him thank you,
3 we'll move on to the staff.

4 MEMBER REMPE: I think we should also
5 think him and NuScale for helping us deal with the
6 situation. It's more than normal.

7 MEMBER CORRADINI: Okay, so we can turn to
8 the staff who are actually in the room.

9 MR. FARAWILA: Well, thank you very much.

10 MEMBER CORRADINI: Bruce, are you guys
11 ready? Okay.

12 MR. YARSKY: This is Dr. Peter Yarsky from
13 the Office of Research staff. I suppose we'll just do
14 this presentation in an analog format with the
15 handouts. Today I'm here to talk to you all about the
16 staff review of the NuScale Stability Topical Report.

17 If we go to slide 2, the review team
18 included Dr. Ray Skarda, myself, and Rebecca Karas
19 from the Office of New Reactors. If we go to slide 3,
20 this presents an outline that is, that shows the
21 content of what we presented at the Subcommittee
22 meeting.

23 And a few bullets here are highlighted.
24 These are items that we're going to cover in today's
25 presentation with the full Committee. And this

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1 includes discussion of the PIM evaluation model and
2 the impact of secondary side instabilities on primary
3 side stability. Other secondary side stability
4 concerns, and finally to wrap up with Stability
5 Topical Report conclusions.

6 If we go to slide 4, slide 4 presents a
7 figure that describes the exclusion region-based long-
8 term stability solution approach. And this was
9 described in the earlier session by NuScale that the
10 stability solution works by preventing instabilities.
11 Instabilities are possible only when there's riser
12 voiding.

13 So the module protection system precludes
14 the occurrence of riser voiding by tripping the
15 reactor if there's a loss of riser subcooling. If we
16 go to slide 5, calculations --

17 MEMBER CORRADINI: Just for the members
18 that weren't here, so the limit is what you state as
19 the allowed operation. So once they get close to
20 subcooling, the trip is incurred.

21 MR. YARSKY: I'm sorry, I don't think I
22 understand. So if you look at the figure, I can't
23 point to anything. So the allowable operation is in
24 this bluish area on the figure, and that's where there
25 is subcooled margin in the riser. So there's a trip

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1 subpoint where once the subcooled margin is smaller
2 than that trip subpoint, the module protection system
3 will insert control rods and shut the reactor down.

4 MEMBER CORRADINI: Thank you.

5 MR. YARSKY: Okay, if we go to slide 5.
6 The stability analyses are performed using the PIM
7 evaluation model. PIM, as it was described in this
8 morning's presentation, includes simple models for
9 thermal-hydraulics, reactor kinetics, fuel thermal
10 mechanic response and steam generator tube heat
11 conduction heat transfer.

12 And the PIM evaluation model has been
13 validated against tasks performed at the NIST-1
14 facility.

15 The key output from PIM, moving on to
16 slide 6, is the decay ratio. The decay ratio
17 acceptance criterion is determined to account for
18 model uncertainties and model biases such that
19 calculating a decay ratio less than the acceptance
20 criterion indicates that the power module is stable.

21 The decay ratio itself is insensitive to
22 variations in most of the important phenomena over the
23 PIM application range. And as a part of the staff
24 review, we confirmed that the acceptance criterion
25 affords sufficient margin to account for the bias and

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1 uncertainty in the method. As part of determining
2 that bias and uncertainty, numerical effects were
3 considered.

4 If we go to slide 7. This is a follow-up
5 to the issue of secondary side instability. As part
6 of the review, the staff determined that the secondary
7 side flow oscillations are not safety-significant with
8 respect to the primary side. And so while there is
9 the possibility for secondary side instability, found
10 that has no impact on the relevant regulatory criteria
11 as they apply to reactor stability.

12 MEMBER MARCH-LEUBA: Pete, are we going to
13 have a closed session with you? Because I cannot find
14 the enclosed slides.

15 MR. YARSKY: Oh, the staff is not planning
16 to present in the closed session on the topic of
17 stability.

18 MEMBER MARCH-LEUBA: So let me ask on the
19 secondary side, instability, generic question.
20 Because we all saw the closed Subcommittee meeting, or
21 some of us saw the closed Subcommittee meeting. It
22 was your evaluation there that the secondary side
23 would, is likely to be unstable under the complete
24 operating domain on the, for NuScale? Basically on
25 low flow?

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1 Because we just heard from NuScale that
2 the test indicated that would only happen at low
3 flows.

4 MR. YARSKY: I don't know how much in open
5 session we can talk about the specific conditions
6 under which the tubes would be expected to be
7 unstable. However, the staff based its review
8 assuming that the secondary side would be unstable.
9 So the conservative approach there of course is to
10 assume that it would be unstable and potentially
11 achieve like the maximum value of its oscillation.

12 MEMBER CORRADINI: So you didn't look for
13 where it's unstable, you assume it's unstable and see
14 the effect on the core.

15 MR. YARSKY: Right. So one thing the
16 staff did was to compare the operating conditions
17 expected during normal operation, including shutdown
18 and startup maneuvering, and compare that to test
19 conditions that were evaluated by NuScale. And from
20 that comparison, the staff was not able to conclude
21 that the secondary side would be stable.

22 So there are these, of course the
23 potential for certain conditions for the secondary
24 side to be unstable. In that case, the conservative
25 approach and the approach adopted by the staff was to

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1 assume that the secondary side would be unstable.

2 MEMBER MARCH-LEUBA: Is it safe to say
3 that, at least I haven't seen a calculation by NuScale
4 of the stability range in the secondary?

5 MR. YARSKY: Correct. So the staff did
6 not pursue requesting information from the applicant
7 to specifically demonstrate at what conditions the
8 secondary side would be stable or unstable. The
9 approach adopted by the staff was to just assume the
10 secondary side would be unstable and to evaluate that
11 instability as it relates to the primary side
12 acceptance criteria.

13 MEMBER MARCH-LEUBA: And one concern,
14 because ACRS sees all the chapters. Chapter 15?

15 MR. YARSKY: Yes.

16 MEMBER MARCH-LEUBA: One concern I
17 personally has is the impact on thermal fatigue. If
18 you have significant oscillations of, in temperature
19 at the inlet of the tube because the boiling boundary
20 goes up and down every three to ten seconds, you will
21 have thermal fatigue. So that has to be translated to
22 the Chapter 3 guys.

23 MR. YARSKY: Right so the issues
24 associated with like the mechanical performance and
25 thermal fatigue as a result of secondary side flow

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1 oscillations has to be addressed by the Chapter 3
2 review.

3 MEMBER MARCH-LEUBA: And just for the
4 record, we have NuScale saying on the record that they
5 will fix this instability. My concern is I don't know
6 how they're going to do it.

7 MEMBER BLEY: We had a fairly lengthy
8 discussion on this. But I don't think when we did
9 Chapter 3 we addressed this issue at all.

10 MEMBER MARCH-LEUBA: We did not.

11 MEMBER CORRADINI: I think if, so we'll
12 come back. We can ask questions of the applicant, but
13 I think they have to address the thermal fatigue as
14 part of Chapter 3, and this is one of the inputs that
15 have to be considered.

16 MEMBER BLEY: But what I was getting at is
17 we didn't raise it in our letter because we did not
18 talk about that I think that time, I don't think.

19 MEMBER CORRADINI: Well, and Chapter 3 is
20 part of this letter that you're going to see, and it
21 is mentioned. Okay.

22 MR. YARSKY: Right. And the cognizant
23 reviewers in Chapter 3 that are doing the staff review
24 for mechanical and thermal fatigue-related issues are
25 aware of this issue and are incorporating that of

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1 course into their review.

2 MEMBER MARCH-LEUBA: And now let me
3 summarize. The staff assumes the secondary side
4 oscillations will happen, and then propagates that
5 stable condition in the secondary to the core. And
6 because of the multiple (inaudible) close to a
7 thousand do not phase lock, they're kind of --

8 (Simultaneous speaking.)

9 MR. YARSKY: So they're, the staff in
10 conducting the review asked a number of questions
11 about the secondary side in relation to the claim that
12 the tubes would not phase lock.

13 And we weren't able to reach the same
14 conclusion that NuScale reached that it would be
15 impossible for the oscillations to be in phase,
16 because we're just, we have an absence of information
17 about the specific nature of the control system and
18 how the secondary side will be controlled, either with
19 different valves or different control system schemes.

20 So therefore when we assume the secondary
21 side is unstable and we wanted to propagate that to
22 the primary side, the staff requested an analysis
23 where the secondary side tubes all oscillate in phase.
24 Because that would produce the maximum consequence for
25 the primary side.

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1 MEMBER MARCH-LEUBA: And that's
2 conservative.

3 MR. YARSKY: That's conservative, and it
4 would bound any potential considerations for the final
5 design of the secondary side control systems.

6 MEMBER MARCH-LEUBA: And using that very
7 conservative assumption that the control, it fails and
8 locks them. You still don't see propagation to the
9 primary.

10 MR. YARSKY: So there of course you would
11 see some degree of oscillation on the primary side,
12 but it isn't significant from the standpoint of
13 thermal margin.

14 MEMBER MARCH-LEUBA: And that is because
15 of the different frequencies of oscillation?

16 MR. YARSKY: So that is an excellent
17 question. The frequencies are different between the
18 primary and secondary sides.

19 So in the bounding analysis that was
20 performed to address the impact on the primary side
21 from the secondary side, the secondary side maximum
22 for oscillation was actually adjusted so that it was
23 not at the natural frequency of the steam generator,
24 but rather the natural frequency of the primary
25 system.

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1 So it attempts to put that maximum
2 oscillation magnitude also at the worst possible
3 frequency. And so it's, the analysis that was
4 performed is a perfect storm of what could possibly be
5 the worst condition for the primary side. And we
6 found that it is not significant -- it doesn't
7 significantly impact thermal margin.

8 MEMBER BLEY: Did the staff do that
9 analysis or did you do it?

10 MR. YARSKY: No, the staff requested that
11 NuScale perform that analysis, so that analysis was
12 performed by NuScale.

13 MEMBER KIRCHNER: And then -- and so the
14 steam generator is just treated as one node heat sink?

15 MR. YARSKY: No. So there are multiple,
16 you need to treat the steam generator with multiple
17 nodes.

18 MEMBER KIRCHNER: Yeah.

19 MR. YARSKY: Because there's different
20 heat transfer regimes as you progress axially through
21 the steam generator. However, PIM has the capability
22 to treat -- I think it would be best if we wanted to
23 talk specifically about the calculation method that it
24 be deferred to the closed session.

25 I don't know how much we can talk about

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1 what PIM does in terms of the specifics of
2 nodalization and subnodal treatments in the open
3 session.

4 MEMBER KIRCHNER: Even if they have
5 different boiling regimes?

6 MR. YARSKY: I think in the open session
7 we can say that the methodology does treat that there
8 are different heat transfer regimes as you progress
9 axially on the secondary side of the steam generator
10 tube.

11 MEMBER KIRCHNER: Okay, that's really the
12 heart of my question. Okay.

13 MR. YARSKY: And that phenomena is
14 captured. But I think if we want to talk specifically
15 about how that's done, it would be best to do so in
16 closed session.

17 If we move to slide 8 -- oh, we have the
18 slides now.

19 (Laughter.)

20 MR. YARSKY: So in the review of the
21 topical report we found that while PIM is a simple
22 model, its models are anchored to some more high
23 fidelity upstream methods, and this improves its
24 overall accuracy. Because the decay ratio is highly
25 insensitive to variations in most of the important

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1 phenomena, the overall uncertainty in the decay ratio
2 is relatively small.

3 PIM predictions in the steady state and
4 transience have been confirmed by the staff with
5 independent trace confirmatory calculations that show
6 very good agreement between PIM entries. And so we
7 have concluded that PIM is acceptable for performing
8 the stability analysis for the NuScale power module.

9 If we move on to slide 9. The topical
10 report not only discusses PIM but also the nature of
11 the long-term stability solution. We found that this
12 solution is also acceptable as the applicant has
13 identified the correct primary instability mechanism,
14 we've confirmed that mechanism through our own
15 independent analysis.

16 During normal at-power operation, we've
17 concluded that the NuScale power module is very
18 stable, and that the exclusion region-based long-term
19 stability solution is effective in preventing the
20 reactor from becoming unstable during normal operation
21 and including the effects of AOOs.

22 We also considered worst rod stuck out
23 conditions. This was discussed in greater detail at
24 the Subcommittee meeting. But we concluded that
25 potential instability during return to power with

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1 worst rod stuck out is not an issue that needs to be
2 considered as part of that analysis.

3 And so therefore GDCs 10, 12, 13, 20, and
4 29, which are the applicable regulatory criteria, are
5 all met.

6 MEMBER MARCH-LEUBA: Pete, you performed
7 a calculation with trace of ATWS with a conservatively
8 low MTC model for the temperature coefficient, in
9 which you are in ATWS and now you have boiling for an
10 extended period of time in the riser. Could you
11 describe the stability results you saw with trace
12 within that transient?

13 MR. YARSKY: Oh, so this is really outside
14 of the scope of the stability evaluation here because
15 the long-term stability solution of course credits the
16 reactor trip. However, if you were to have an ATWS
17 event, there are sort of two possible ATWS cases one
18 could consider. One could consider like an
19 overpressurization-type ATWS scenario.

20 In that scenario, you don't have a
21 significant decrease in primary system pressure and
22 inventory. In those kinds of scenarios, the high
23 reactor power and the ATWS leads to a loss of
24 inventory through the reactor safety valves, which
25 will just burp off excess pressure, transferring

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1 inventory from the primary side to the containment.

2 In those scenarios, there's a slow
3 reduction of the level, but the pressure's really
4 high. So because of that high pressure, you don't
5 necessarily see voiding in the riser until you get to
6 a condition where the level drops below the top of the
7 riser.

8 And when the level drops below the top of
9 the riser, there's a significant reduction in the core
10 flow rate because the natural circulation pattern
11 breaks. At that point, the nature of any flow
12 oscillations, that mechanism that creates those flow
13 oscillations is very different than the mechanism that
14 we're discussing as part of this topical report,
15 because you don't have that same loop, right.

16 You now have something more akin to like
17 a YouTube manometer, where you have like a liquid
18 level that's separated by the riser and you have a
19 level above the core. Under those conditions, the
20 core flow rate can get quite low. You can have
21 boiling and you can have the level kind of sloshing
22 back and forth between the --

23 MEMBER MARCH-LEUBA: But before the water
24 level dropped to the top of the riser and went into
25 what I call the percolator mode, when the door was

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1 high and you had circulation, did you see significant
2 boils in the riser or did you see any flow
3 instability?

4 MR. YARSKY: In our calculations we didn't
5 observe significant flow instability. There is the
6 possibility for there to be conditions where flashing
7 may occur in the riser under certain conditions. That
8 may occur prior to the level dropping below the top of
9 the riser because the RSV is cycling between its lift
10 and its set pressure.

11 So if the core power remains high enough
12 that it heats up the riser to a relatively high
13 temperature and then the RSV lifts, there'll be a
14 moment where the system pressure may drop below the --
15 they drop sufficiently low that the temperature is
16 above the saturation temperature in the riser. Now,
17 that would be like a longer period-type phenomena, so
18 that would like over the course of minutes.

19 MEMBER MARCH-LEUBA: But that would not be
20 --

21 MR. YARSKY: Where the RSV cycling is
22 happening. But the voiding itself would be, would
23 occur in a much shorter timeframe.

24 MEMBER MARCH-LEUBA: But that would not be
25 an intrinsic instability, it would be driven by the

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1 SRV or RSV.

2 MR. YARSKY: Yeah, so you could
3 potentially have this short-term flow response to the
4 RSV cycling. But the RSV cycling frequency is going
5 to be divorced from the natural frequency of the flow
6 loop. So it wouldn't necessarily create a feedback
7 loop that would create a growing flow instability.

8 MEMBER MARCH-LEUBA: It's safe to use and
9 you used a state-of-the-art code trace.

10 MR. YARSKY: Yes.

11 MEMBER MARCH-LEUBA: To analyze this
12 phenomena, and you did not see large oscillations.

13 MR. YARSKY: Correct, correct. So trace
14 we did an extensive look at the trace applicability to
15 analyze these phenomena and concluded that trace is
16 able to model these phenomena for NuScale, including
17 doing additional trace assessments to address
18 potential information gaps.

19 So that is something that we've had done,
20 concluded traces applicable and performed those
21 analyses, and we didn't see any issues associated with
22 flow oscillation during our ATWS analysis.

23 MEMBER BLEY: Peter, this isn't exactly a
24 question for you, but maybe somebody on the staff can
25 address it. And we didn't ask NuScale about it. This

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1 ATWS scenario sounds kind of benign, but I'm not
2 completely sure of that. I don't think it's analyzed
3 in the PRA. It seems like, as if it ought to have
4 been. But I don't, the PRA, your PRA folks didn't
5 come to you about this issue, did they?

6 MEMBER CORRADINI: We had Pete's
7 presentation, I don't remember which Subcommittee
8 meeting, in closed session about these results.

9 MEMBER BLEY: On a --

10 MR. YARSKY: Yeah, so on the chapter, as
11 part of Chapter 19 review.

12 MEMBER BLEY: On Chapter 19. Okay, I
13 didn't remember that, okay.

14 MR. YARSKY: One thing to keep in mind is
15 that the acceptance criteria for beyond design basis
16 events like ATWS are different than the acceptance
17 criteria for stability and AOOs. So for an ATWS
18 event, there's the possibility that the, you may lose
19 thermal margin, that the fuel could go into CHF. That
20 would allowable so long as the core remains in a
21 coolable geometry.

22 MEMBER BLEY: That's why I was asking
23 about the PRA, which should have looked at --

24 MR. YARSKY: Right so the --

25 MEMBER BLEY: -- the likelihood of damage

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1 under the specifications.

2 MR. YARSKY: Right, so in the ATWS
3 scenario, you're looking at possibility for fuel
4 damage, you're not looking at possibility for fuel
5 failure due to CHF considerations.

6 MEMBER BLEY: Right.

7 MR. YARSKY: So because of that difference
8 in the acceptance criteria, the event that we analyze
9 for ATWS, we'd say well it looks relatively benign
10 from the standpoint of those acceptance criteria,
11 which are of course very different than the acceptance
12 criteria considered for AOOs.

13 MEMBER CORRADINI: But just to remind
14 everybody, he did identify and we have the docketed,
15 his complete report on ATWS.

16 MEMBER BLEY: Oh, I do remember that now.

17 MEMBER CORRADINI: I sent it to you.

18 MEMBER BLEY: Yes.

19 MEMBER CORRADINI: You asked me where is
20 it and I sent it to you.

21 MEMBER BLEY: You sent it to me, you are
22 absolutely correct.

23 MEMBER CORRADINI: Just wanted to make
24 sure we're clear.

25 MEMBER BLEY: And Peter was the right one

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1 to ask that question.

2 MR. YARSKY: I'm happy to help.

3 MEMBER CORRADINI: Anything else for
4 Peter? Okay.

5 MR. YARSKY: Thank you.

6 MEMBER CORRADINI: Thank you very much.
7 So you don't have anything in closed session to
8 present, but you'll be around if we have questions.

9 Did we have, I want to make sure, want to
10 ask the members, do we want to now go into closed
11 session, as if we're not already, and talk about
12 details that Yusuf Farawila said he'd have to wait and
13 talk about in closed session? Is that what the
14 members would like to do?

15 MEMBER MARCH-LEUBA: Yes.

16 MEMBER CORRADINI: Okay. So I'm going to
17 need to get NuScale folks back up in front and then
18 get Yusuf back on the line.

19 MEMBER MARCH-LEUBA: The transcript needs
20 to go into closed and we need to verify that a room is
21 available.

22 (Whereupon, the above-entitled matter went
23 off the record at 10:19 a.m. and resumed at 12:59
24 p.m.)

25 MEMBER RICCARDELLA: We will resume the

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1 meeting.

2 MEMBER CORRADINI: Okay, so, Marty, I
3 think you're going to take us through Chapter 3?

4 MR. BRYAN: Yes, sir. Thank you.

5 MEMBER CORRADINI: Okay.

6 MR. BRYAN: Good afternoon. I'm Marty
7 Bryan, the licensing project manager for Chapter 3 and
8 today we're going to cover three main subjects.

9 Amber Berger is going to take us through
10 turbine missiles. There was a lot of questions at the
11 last meeting, but I will note this is the open
12 session. Most of the detail for that will be in the
13 closed session.

14 Olivia Hand is going to discuss CVAP and
15 TF3. Again, it will be open, and more detail in the
16 closed session. Then Hannah Rooks is going to take us
17 through the DWO. I know there was a lot of discussion
18 earlier. We'll try to address some of those
19 questions.

20 MEMBER CORRADINI: And we'll hold the
21 closed session until we hear after the staff.

22 MR. BRYAN: Yes, correct, yeah. Okay, so
23 with that, Amber, will you take us through the turbine
24 missiles?

25 MS. BERGER: Sure, Amber Berger, NuScale.

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1 Chapter 3 or Slide 3 is just a title for the missiles.
2 Slide 4 is basically an overview of what we'll discuss
3 today -- Whoever is shuffling paper, could you put
4 your phone on mute, please? Thank you.

5 Okay, again Amber Berger. The slide 4 of
6 this slide deck is just the agenda of the turbine
7 missile discussion. We're going to talk about and
8 we'll show an illustration of the turbine missile
9 trajectory and the barriers that we're crediting in
10 our analyses.

11 We'll go over some additional details of
12 the turbine barrier analyses that we performing,
13 focusing on local analyses and global analysis. We'll
14 also present the turbine missile barrier results and
15 discuss the conservatism in our approach, and that's
16 all I have on turbine missiles right now.

17 MR. BRYAN: Thank you, Amber, and again,
18 we'll cover it in a lot more detail in the closed
19 session. I know there's a lot of interest in the REI
20 response, and so, previously, so we've extracted a lot
21 of that information.

22 So now we're going to transition to Olivia
23 taking us through CVAP and TF-3.

24 MS. HAND: Hi, so this is Olivia Hand. If
25 we move onto slide number 6, that has a brief

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1 discussion of what TF-3 is and the purpose of that
2 testing.

3 So this TF-3 test specimen is a five-
4 column full height assessment. It has prototypic
5 steam generator tube support. It's a two-part test
6 program. The first part is to characterize the
7 response of the test specimen, specifically looking at
8 the response of the steam generator tubes, and we're
9 going to be performing testing both in air and in
10 water.

11 Then once we have the (inaudible) testing
12 complete, we'll be moving onto flow testing. Flow
13 testing for this is just focused on the primary site
14 flow. So this is the test specimen and we do not have
15 flow inside of the steam generator tubes.

16 And the purpose again, we're looking to
17 characterize the frequency response of the steam
18 generator tubes, the mode shapes that we're seeing,
19 and also the damping (phonetic) values, and that's
20 because all of three of those parameters are inputs to
21 our design analysis.

22 As a part of the flow testing, we'll be
23 looking to characterize the critical velocities for
24 onset of (inaudible) phenomena, which are elastic
25 instability and vortex shedding, and then also we're

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1 going to be looking at what square vibration
2 amplitudes are as we change our primary coolant flow
3 rates.

4 And we've done some preliminary
5 demonstration testing. The NRC was out to view this
6 last month in Italy, and so far, our results are
7 showing very good agreement in both the frequency and
8 dampening with our design analysis predictions.

9 But again, those results are just
10 preliminary. We still have to actually complete
11 commissioning of the test facility, and then we're
12 hoping to kick off testing for our first column in
13 September.

14 MEMBER CORRADINI: So let me ask a
15 question at this point. The flow restrictor tests are
16 a separate set of tests, is that correct?

17 MS. HAND: Yes, that's correct. So we
18 have actually some validation testing to do before the
19 program for the final flow restrictor design to look
20 for the lack of a leakage flow instability
21 characteristic for that component, but, yeah, there
22 are no flow restrictors in this design. There is
23 nothing inside of the steam generator to (inaudible)

24 MEMBER CORRADINI: So, I'm sorry, go
25 ahead.

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1 MEMBER MARCH-LEUBA: No, you go ahead.
2 Okay, so this is the TF-3 test. What's the status of
3 the TF-2 test? Because we are going to -- if we go on
4 (inaudible), we're going to strongly encourage that
5 you rerun the heated tube flowing side tests with the
6 final solution. How much of a problem would that be
7 to resuscitate TF-2 or am I asking the wrong person?

8 MS. HAND: Hannah might be able to --
9 Hannah Rooks might be able to field that question a
10 bit better because as far as flow and vibration is
11 concerned, we've already used the TF-2 results for our
12 benchmarking, and we got acceptable results, so.

13 MEMBER MARCH-LEUBA: Yeah, this is a
14 completely different topic. It's not flow and
15 vibration. It's inside the tube circulation, so which
16 is what we tested on the TF-2 test. What I'm asking
17 you is does the TF-2 reading still exist? Because
18 apparently TF-3 will not be able to do this.

19 MS. ROOKS: Yeah, this is Hannah Rooks.
20 So the TF-2 test facility is currently, it's still in
21 existence, but it's, you know, it's closed down and we
22 weren't planning on doing any more testing at that
23 facility.

24 MEMBER CORRADINI: But can you remind us
25 what TF-2 is? It's a heated test facility, is that

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1 correct?

2 MS. ROOKS: Yes.

3 MEMBER CORRADINI: Okay, and --

4 MS. ROOKS: It has primary and secondary
5 side flow, and the main purpose of it was to
6 characterize (inaudible).

7 MEMBER CORRADINI: But it was that test
8 facility that at least exhibited flow, two-phase flow
9 oscillatory behavior on the secondary side, is that
10 not correct?

11 MS. ROOKS: Yes, so we had tests that were
12 specifically designed to reduce oscillations, and then
13 we also had other tests that had stable operations, so
14 we had kind of a variety of tests in that facility.

15 MEMBER CORRADINI: Okay, thank you. That
16 helps me.

17 MEMBER MARCH-LEUBA: Are you documenting
18 those? Which conditions were the ones that were
19 stable? Are those -- I don't think -- are they on the
20 area response? I forget the number. I don't think
21 I've seen it. It would be very interesting for me to
22 see what condition resulted in stable flow.

23 MEMBER CORRADINI: I think what Dr. March-
24 Leuba is asking for really is to get some sort of
25 mapping of where things were stable versus where

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1 things were not stable on the secondary side, and the
2 flow conditions that essentially drove those
3 differences.

4 MEMBER MARCH-LEUBA: Yeah, basically this
5 issue of the tube flow instability, which has nothing
6 to do with vibrations. It's a two-phase flow
7 oscillation has gotten so much visibility that you
8 have to solve it, and you have to prove that it's
9 solved.

10 So I'm just putting it in there that it
11 needs to -- you need to start thinking about how to
12 fix it and then how to prove that you fixed it, and
13 I'm thinking here that TF-3 will not be the proper
14 vehicle. Resuscitating TF-2 might be a better path to
15 a solution.

16 MR. BRYAN: We are going to talk about DWO
17 here in the third presentation as part of this.

18 MEMBER MARCH-LEUBA: Okay, so we'll
19 postpone it then.

20 MR. BRYAN: Yeah.

21 MEMBER MARCH-LEUBA: You guys know --

22 MR. BRYAN: Yeah.

23 MEMBER MARCH-LEUBA: -- where we're going.

24 MR. BRYAN: Yeah.

25 MEMBER RICCARDELLA: Regarding CVAP and

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1 TF-3, I'll point out that, you know, we addressed this
2 at our last meeting and actually included it on our
3 letter, in our prior letter, and one of our
4 recommendations related to that is that there be an
5 ITAAC regarding TF-3.

6 MR. BRYAN: Yes, and we have proposed some
7 wording to be put in tier one as opposed to an ITAAC
8 to address and ensure TF-3 is done, so we'll probably
9 be working with the staff to evaluate that, so we did
10 see that in the letter, yes.

11 MEMBER RICCARDELLA: Okay, great.

12 MEMBER CORRADINI: Go ahead.

13 MR. BRYAN: Go ahead, Olivia.

14 MS. HAND: That was it for CVAP and TF-3,
15 so we can move on to Hannah.

16 MR. BRYAN: So Hannah is going to cover
17 the DWO discussion.

18 MS. ROOKS: Okay, so we'll move onto slide
19 8. So this is an overview of the work that has been
20 done so far in our design base. One of the design
21 objectives for the steam generator is to withstand all
22 loadings for the design type of the components, and
23 that includes oscillation loading.

24 So what we've done is we've done RELAP,
25 NRELAP5 simulations to compare to the TF-1 and TF-2

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1 test data, and we compared it to both the cases with
2 stable results and cases with (inaudible).

3 And so the test data shows that the
4 frequency of the oscillations are much slower than
5 natural frequencies of the components, so there is no
6 resonance of (inaudible) concerns, but they do show
7 that we have (inaudible) oscillations at our low power
8 operating conditions.

9 So the design objective of our flow
10 restrictor is to create pressure drops in the
11 (inaudible) to limit those oscillations (inaudible)
12 set the lower limits.

13 So the analysis that we've done with our
14 full 21-column steam generator model is to analyze the
15 conditions at (inaudible) readings of power levels and
16 to (inaudible) on that last coefficient in the unit
17 flow restrictor to get below our second criteria for
18 the flow.

19 MEMBER CORRADINI: So let me just ask a
20 question at this point. The flow restrictor that you
21 now are testing is not what was used in the TF-2
22 testing, is that not correct?

23 MS. ROOKS: Yes, that's true. So the TF-2
24 test, the last coefficient was lower than our current
25 design, and it was also just like an orifice, and our

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1 current design has several steps of constricting area
2 (inaudible) in it, so it's a different design.

3 MEMBER CORRADINI: Thank you.

4 MEMBER MARCH-LEUBA: Okay, and I see you
5 using NRELAP5 for these models. Did you benchmark it,
6 I mean, the TF-2 experimental data, and how were --

7 (Simultaneous speaking.)

8 MEMBER MARCH-LEUBA: I'm telling you
9 because my experience with RELAP is, you know, very
10 bad.

11 MS. ROOKS: It showed very stable results
12 at the 100 percent power operating conditions, but we
13 did see some oscillations in our lower power levels,
14 so we've done some, you know, combination of
15 operational changes. So we've increased our steam
16 pressure at the lower power levels to provide more
17 subcooling, and we are also looking into some other
18 options for, you know, operational changes as well.

19 MEMBER CORRADINI: So can I just follow up
20 on Dr. March-Leuba's question? Is there on the docket
21 a report that compares the calculation to the TF-2
22 experiments or is that internal to NuScale?

23 MEMBER MARCH-LEUBA: And since I know the
24 answer is internal, can we see it in a couple of
25 weeks?

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1 MEMBER CORRADINI: Well, let her confirm
2 what you're thinking. If Hannah could at least, if
3 you could please at least explain? You said you did
4 a comparison of NRELAP to the TF-2 data. Is that in
5 some form of a report or can it be discussed with us?
6 That's what I think we're having some interest in.

7 MS. ROOKS: Yes, we do have those written
8 calculations. They're proprietary class one
9 calculations because they have, you know, test data in
10 them.

11 MEMBER CORRADINI: Okay, so we'll take
12 that up and see what, between the staff and NuScale,
13 if we're allowed, if we can see those in some manner
14 or form because that confirmatory calculations we were
15 looking for relative to you predicting it under full
16 power conditions.

17 MS. ROOKS: Okay.

18 MEMBER CORRADINI: Go ahead. I'm sorry
19 for interrupting.

20 MS. ROOKS: All right, so part of our
21 revision three was just going to clarify some
22 terminology that's in our SR section 5412, so we just
23 wanted to make clear that the design intent of the
24 inward flow restrictor is not to fully preclude
25 (inaudible) oscillations, and that there will be

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1 naturally some small amount of oscillations, but to
2 ensure that they're within acceptable limits.

3 And then the other ongoing activities
4 right now are we are developing a stability map based
5 on the TF-1 and TF-2 data for comparison with the
6 stability map that the NRC has presented as part of
7 the stability work.

8 So moving onto the next slide, I want to
9 go over the activities that are in our ITAAC related
10 to the oscillations, so we have a simulator tube. The
11 tube (inaudible) and the tube sheet are class one
12 components.

13 Their value was evaluated to the rules of
14 subsection (inaudible) of the ASME code, and these
15 components will be included in the design report, so
16 will be submitted and reviewed by the NRC, and these
17 are actions that are currently in our ITAAC.

18 So the analysis has been completed and is
19 under NRC review. The tube sheet (inaudible) were not
20 included and the (inaudible) was not considered yet,
21 but maximum result usage in the vicinity of the tube
22 sheet weld is very low.

23 So the next step is the (inaudible)
24 analysis. And we'll consider all of the (inaudible)
25 and that will help kind of circle back to the design

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1 work that was done to confirm that the (inaudible) to
2 reduce those oscillations to acceptable levels, so
3 that was it.

4 MR. BRYAN: Yeah, I'm starting to take
5 questions. Is that all? That's all we had for the
6 open session for Chapter 3.

7 MEMBER SKILLMAN: Yes, I'd like to ask
8 this question. The presenter communicated that
9 secondary pressure was raised to increase subcooling,
10 and the question is did that increase in pressure
11 result in any magnification of the pressure waves at
12 the mouth of the tube where the restrictor orifice is
13 located?

14 It seems to me that driving up the
15 pressure will increase subcooling, but it could also
16 enlarge a rarefaction wave if there's a pulse.

17 MR. WOLF: This is Brian Wolf,
18 development supervisor at NuScale. I'm not sure I
19 understand the question, but here (inaudible)
20 oscillations (inaudible) waves, not pressure waves.
21 What we know is that obviously inlet flow restriction
22 helps to reduce oscillations.

23 If you increase exit losses, that will
24 potentially increase oscillations. Increasing
25 pressure typically helps DWO, as well as increasing

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1 the inlet subcooling to some extent, depending on
2 where you're at, and then also increasing power is
3 destabilizing, and decreasing flow rate can be
4 destabilizing depending on the operating conditions.

5 So I'm not sure I understand the question,
6 but increasing the pressure, considering all other
7 things constant, would typically be used to help
8 reduce DW oscillations.

9 MEMBER SKILLMAN: Thank you.

10 MEMBER CORRADINI: So are we -- so we'll
11 move on to Chapter 6?

12 PARTICIPANT: Yes.

13 MEMBER CORRADINI: Okay.

14 PARTICIPANT: I think we're going to --

15 PARTICIPANT: Fifteen.

16 PARTICIPANT: Yeah.

17 PARTICIPANT: NuScale doesn't plan to
18 present on Chapter 6 publicly, but they will in the
19 closed session.

20 MEMBER CORRADINI: So we will talk about
21 six in the closed session?

22 PARTICIPANT: Yes.

23 MEMBER CORRADINI: All right, thank you,
24 to Chapter 15.

25 MR. PRESSEN: And just to confirm, do we

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1 have Meghan on the line?

2 MS. McCLOSKEY: Yes, I'm here.

3 MR. PRESSEN: Excellent.

4 MEMBER CORRADINI: Go ahead.

5 MR. PRESSEN: All right, so thank you all
6 again. I'll introduce myself. I'm Matthew Pressen,
7 licensing specialist with NuScale Power and project
8 manager for Chapter 15.

9 We're here today to discuss Chapter 15 of
10 the NuScale design certification application, and the
11 presentation today will be provided primarily by our
12 technical experts in Corvallis.

13 Meghan McCloskey, supervisor of system
14 thermohydraulics, will be our speaker today, and
15 joining us as needed will be Dr. Pravin Sawant, Dr.
16 Brian Wolf, Dr. Selim Kuran, and Mark Shaver also on
17 the phone in Corvallis.

18 So we wanted to provide a quick scope of
19 the information that we will be presenting today as
20 there is a lot of information summarized in Chapter 15
21 where we deal with postulated transients and events,
22 and a number of topical reports that define
23 methodologies that we use in Chapter 15 but have not
24 yet presented on.

25 So in our presentation today, we focus on

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1 material provided in the FSAR and the results of those
2 methodologies. So we have the summarized topical
3 reports here, but any detailed discussion will wait
4 until we get those in November, I believe.

5 A quick overview of what we'll be going
6 over today, as mentioned, will be the high level of
7 the NuScale module design, including an overview of
8 Chapter 15 assumptions and analyses.

9 We'll cover the limiting cases seen in
10 Chapter 15, and for the Chapter 6.2.1, containment
11 response, and the last portion of our presentation
12 will be focused on long-term cooling and how Chapter
13 15 assumptions look in 72 hours.

14 So this is a high level map of the
15 technical and topical reports that developed methods
16 used in Chapter 15 and Section 6.2.1, containment.
17 NRELAP5 was developed from the RELAP5 3D code to
18 address NuScale specific phenomenon and levels used to
19 evaluate the NuScale power module when developed
20 following Reg Guide (inaudible), evaluation model
21 development and assessment process, which is that
22 acronym there.

23 These topicals are sorted by several
24 additional methods, which the ACRS had a chance to
25 review, which are the nuclear analysis codes and

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1 methods, the critical heat flux, and the subchannel
2 analysis listed there.

3 And while we will discuss this in more
4 detail in the later portion of our presentation, we
5 did want to kind of put up front our exemption to GBC
6 27 since it has an impact on the acceptance criteria
7 for the NuScale BCA essentially to account for the
8 postulated return to power correction that can occur
9 under Chapter 15 assumptions as part of that NuScale
10 design, assure fuel cladding integrity for all design
11 basis events, so we're decking our fuel for all of
12 those Chapter 15, well, Chapter 15 pretty much.

13 So again, this is a fairly dense slide.
14 The point of that is to open our discussion with
15 Chapter 15 with a quick summary of our dose
16 consequences.

17 The bold numbers on the right-hand of the
18 slide represent our limiting cases for the postulated
19 design basis and beyond design basis events, and we
20 felt it was a good slide to show we are creating a
21 safe, and passive, and functional nuclear design.

22 MEMBER MARCH-LEUBA: Could you describe
23 the scenario for what you call the core damage event?

24 MR. PRESSEN: Yeah, do we have anyone in
25 Corvallis that can talk to core damage events?

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1 MR. SHAVER: Sure, this is Mark Shaver,
2 supervisor of radiological engineering at NuScale.
3 The core damage event is what's typically called the
4 maximum hypothetical act or MHA, and the guidance used
5 to develop it was the NEA white paper on mechanistic
6 source terms for small modular reactors, so it has
7 substantial core damage.

8 MEMBER CORRADINI: But give us a little
9 bit --

10 (Simultaneous speaking.)

11 MEMBER CORRADINI: Can you give us a
12 little bit more about the initiating event and how it
13 progresses?

14 MEMBER MARCH-LEUBA: And what fraction of
15 the core is assuming failure?

16 MR. SHAVER: It's actually a surrogate
17 event that's representative of a spectrum of severe
18 accident beyond design basis events, so it's not one
19 particular scenario.

20 MEMBER CORRADINI: But it's a radio --

21 (Simultaneous speaking.)

22 MEMBER CORRADINI: It's a radiological
23 accident, so I'm assuming you have essentially a gap
24 release? I'm still trying to understand because I
25 don't think -- we're thinking thermohydraulic, but I'm

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1 thinking it's more a radiological design basis event,
2 is that not correct?

3 MR. SHAVER: It's a beyond design basis
4 radiological event, but the inputs are thermohydraulic
5 inputs, and if you'd like the details of the different
6 events that go into that, we can get our PRA group on,
7 but we do have a gap release and release timing, and
8 those are informed by different scenarios that are
9 severe accidents.

10 MEMBER CORRADINI: Okay, go ahead. We'll
11 catch up. Keep on going.

12 MR. PRESSEN: All right, so on slide
13 seven, this was primarily provided here for reference
14 for any questions that we had --

15 OPERATOR: The chairperson has
16 disconnected. The conference will now end.

17 MEMBER MARCH-LEUBA: Do we want to break
18 off the record?

19 MEMBER CORRADINI: Well, obviously we're
20 on break for right now.

21 (Whereupon, the above-entitled matter went
22 off the record at 1:27 p.m. and resumed at 1:31 p.m.)

23 MEMBER CORRADINI: Okay. Let's keep this
24 going. That was pretty quick.

25 MR. PRESSEN: All right. So back to slide

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1 seven. It's provided here as a reference for use for
2 any questions. It's a quick diagram reminder of what
3 the MPN looks like.

4 And it provides a schedule of highlights
5 on the unique aspects of the NuScale design, such as
6 the integrated reactor design, passive safety decay
7 heat removal, and automated module protection system
8 for handling event mitigation.

9 So, and with that, we are on slide eight.
10 Which is where I will pass this over to Megan
11 McCloskey.

12 MS. MCCLOSKEY: Thank you. This is Meghan
13 McCloskey with NuScale. Before we get into the
14 overview of the Chapter 15 events and results, we have
15 two slides on the module protection system functions
16 and the types of functions expected in response to the
17 Chapter 15 design basis event.

18 The module protection system actuations
19 for the NuScale plant were designed with a focus on
20 our events and our specific module response. And the
21 actuations needed to support a passive plant response
22 to design basis events without operator action.

23 So, when we consider the scope of events,
24 the functions needed from the module protection
25 systems can be broadly classified in four areas.

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1 Reactivity control, through reactor trip, you know, or
2 -- and isolation of the blind emission source from the
3 demineralized water systems.

4 RCS and secondary inventory controls where
5 containment isolation assures that position inventory
6 is maintained for ECCF cooling. And containment
7 isolation also mitigates loss of inventory outside of
8 the module for breaks of small lines to limit dose
9 consequences and mitigate the event.

10 So, secondary isolation also mitigates
11 dose consequents for events such as the steam
12 generator tube test layer. And assures that second --
13 appropriate secondary site inventory is maintained and
14 at least one train of DHRS for the decay heat removal.

15 A dual secondary side, if cooling is
16 unavailable, the module protection system actuates
17 DHRS. If necessary the emergency core cooling system
18 to provide heat removal.

19 And finally, as was discussed this
20 morning, primary side sub-cooling and stability are
21 protected by a reactor trip. Going to slide nine.

22 MEMBER MARCH-LEUBA: Okay. Can we hold
23 up, please? I don't know which particular revision
24 you have been -- NuScale has been changing the number
25 of trips, or the number of initiation signals that

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1 result on a trip, eliminating some of the redundancy.

2 Could you explain what the philosophy was
3 on -- for example this actuation only happens on high
4 level in the containment.

5 MEMBER CORRADINI: And there was a second
6 trip that you indicated to us in the subcommittee
7 meeting that you were going to eliminate.

8 MEMBER MARCH-LEUBA: That's correct. And
9 you mistake that for losing DC power where that is not
10 an automatic trip.

11 Could you --

12 MS. MCCLOSKEY: Right. Right, so we --

13 MEMBER MARCH-LEUBA: Could you explain the
14 philosophy behind, I mean, we don't -- I don't feel
15 comfortable eliminating the redundancy. I would like
16 to have as much redundancy as possible.

17 Could you explain what your philosophy
18 was?

19 MS. MCCLOSKEY: In the original design as
20 submitted, we had ECCS actuation by the module
21 protection system on either low level in the riser
22 inside the vessel.

23 MEMBER MARCH-LEUBA: Um-hum.

24 MS. MCCLOSKEY: Or high containment
25 levels.

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1 MEMBER CORRADINI: That makes sense.

2 MS. MCCLOSKEY: And the change that we had
3 made is to eliminate the trip on the high, I'm sorry,
4 on the low riser level. And so we are relying on the
5 containment level trip to actuate ECCS.

6 The way these signals were originally
7 designed, in most cases that riser level and
8 containment level were happening around similar times.
9 And we determined that we wanted to delay the ECCS
10 actuation.

11 And so we focused on the containment level
12 signal. And it's a little bit higher than it was
13 previously.

14 From Chapter 15 perspective, the module
15 protection system is our safety-related system. There
16 are redundant instruments measuring the containment
17 level.

18 And so from that perspective, we felt
19 there was sufficient robustness in the riser to
20 achieve ECCS.

21 MEMBER CORRADINI: So, I don't think I
22 understand your timing argument. Can you repeat that?
23 Because that -- I remember you saying that to us
24 during the subcommittee meeting.

25 And I'm not following, because I think I'm

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1 with Dr. March-Leuba that the redundancy on two
2 different locations gives you a diversity that you now
3 have removed.

4 And I guess I'm not understanding your
5 timing argument. Can you repeat that please?

6 MS. MCCLOSKEY: When we have those two
7 different signals, in terms of the event progression
8 -- in terms of the progression, removing the riser
9 signal doesn't significantly change the timing of the
10 event progression, because we're getting the actuation
11 signal on the high containment level around the same
12 time that you, we would have been getting the
13 actuation on the low RCS riser level.

14 MEMBER MARCH-LEUBA: Yeah. Well, your
15 point though of why we're trying to talk about
16 diversity and redundance is that this is correct on a
17 simulation when everything works perfectly.

18 But in real life, having back up, I mean,
19 in my car I like to have four breaks instead of only
20 one. You know, just in case one breaks, one fails,
21 you have the -- the ones in the back still work.

22 So, your only excuse for the lack of
23 available water is that in the simulation you were
24 always tripping on containment. So, why have the
25 inside the vessel level?

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1 I mean, it is a perfectly acceptable
2 answer, only that that assumes that all those other
3 instruments are going to work perfectly. And all
4 events will progress the way you modeled them.

5 MEMBER BLEY: This is Dennis Bley. I sort
6 of got a hint you were implying there was some kind of
7 interference you could have if these trips are coming
8 in about the same time.

9 But, I don't understand that. This -- is
10 the only reason you're getting rid of it is because
11 they're replicates and you don't need to -- you don't
12 think you need them both?

13 MS. MCCLOSKEY: From Chapter 15
14 perspective, we didn't think we needed them both. I
15 don't think -- we have our I&C Group here.

16 And we can provide some additional
17 information with respect to diversity, the diversity
18 aspects with the containment level signals.

19 MR. AYALO: Thank you Meghan. So my name
20 is Rufino Ayalo, I'm with the I&C Review Group. And
21 just one of the things that isn't specifically, you
22 know, modeled and taken into account within Chapter 15
23 is some of the diversity that we already have within
24 some of our measurements, our sensor measurements.

25 So for example, the containment water

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1 level, within, you know, regarding redundancy, you
2 know, we have four separate containment water level
3 sensors. And the -- it would specify that two of the
4 four has to be diverse, from different vendors.

5 So that provides a different level of
6 diversity. So, we're not relying on a single sensor
7 from the same vendor across all four separations.

8 As we said, we have two from one
9 manufacturer and two from another. And that provides
10 an additional level of diversity.

11 MEMBER BLEY: Can I -- before you leave
12 that one, can I push you a little bit? I remember in
13 the past we had different kinds of DP cells, we
14 thought from different manufacturers.

15 But when you really dug into it, you found
16 that the actual operable unit was only by one
17 manufacturer, and put in different housings and had
18 other names on them.

19 So, are you sure you have that diversity
20 you're claiming?

21 CHAIR REMPE: Well, I'd even go further.
22 Let's remind ourselves that this is close to a first
23 of a kind application in a containment or a reactor
24 vessel.

25 And I'm real curious, I mean, I wasn't

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1 here when you guys discussed this, because I had to
2 leave for the subcommittee. But, at the time somebody
3 from NuScale popped up and said oh, yeah, Chapter 7
4 talks about diversity.

5 And I looked it up before this meeting.
6 So I'm glad you at least acknowledged your diversity
7 definition is two different vendors.

8 But why didn't you explore something like
9 a multipoint thermal couple on the side of the
10 containment? Or within the containment or within the
11 vessel as a really truly diverse sensor that's not a
12 first of a kind application?

13 MR. AYALO: So, that's a lot -- there's a
14 lot of questions there. So let me see if I can try to
15 answer a few of those.

16 And just as a reminder as well, for later
17 this month, there will be a presentation related to
18 sensors as well. But, regarding your -- the first
19 comment about the use of the different level
20 measurements.

21 You know, one of the approaches that we're
22 taking is it was a phased approach that we started and
23 we described in our technical report associated with
24 sensors.

25 You know, we focused with the --

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1 essentially a down place of various concepts. And you
2 know, based on the conditions that were, based on our
3 scenarios, based on environmental conditions, we
4 decided to use the levels selected.

5 And while it does appear to be, you know,
6 first of a kind in terms of a nuclear application,
7 there are applications such as Dean Drum's (phonetic)
8 where you see the use of level sensors in this
9 particular application.

10 So our -- in sort of this style of
11 environment, so --

12 CHAIR REMPE: Well, it has a radiation, a
13 high dose radiation level in the other applications?
14 I mean, I know you -- they've been used maybe in a
15 split fuel pool or something.

16 But, within the reactor vessel you see --
17 I don't know what your design has. But some of them
18 have some sort of glass that will darken and would
19 recalibrate.

20 And you've seen some sort of application
21 where they've done this, and they don't have to worry
22 about the radiation levels, fluences?

23 MR. AYALO: Yeah. I mean, I guess that --
24 I guess maybe I should clarify. I'm not saying it's
25 an exact match of our situation with the high

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1 radiation, the high pressurized temperature.

2 But there are, and we will demonstrate as
3 part of our program, the environmental qualification
4 of the center with the radiation, with the pressure,
5 with the temperature, but from a -- from a purely that
6 there is no examples in other industries where this is
7 being used in a similar application.

8 And that -- and by that I mean, the
9 pressure and temperature, I would say that that --
10 that there are examples of that. And that would be
11 Dean Drum where there is a level measurement taken
12 with the use of radar. And in particular, guided wave
13 radar.

14 CHAIR REMPE: Well, it will be interesting
15 to hear more about it when we travel to Corvallis.
16 But again, now we hear you're going to have two
17 vendors helping you do this first of a kind technology
18 or when you finally deploy it.

19 And again, you could use other
20 technologies that are different that would be more
21 diverse and maybe provide more confidence that you
22 don't have to worry about common cause failures, which
23 is, I believe, what Jose was emphasizing in the
24 transcript when I was reading through it.

25 MEMBER MARCH-LEUBA: Well, at a minimum

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1 the diversity from two vendors gets you away from --
2 if it truly is diverse, and they're not just two
3 different labels on the same instrument. It gets away
4 from the digital common cause blue screen of death.
5 That's good.

6 CHAIR REMPE: But now we have to worry
7 about fluence not only for design basis conditions,
8 which is what is in that, the open version of your
9 technical report.

10 It clearly says that is what you are going
11 to do for the radiation levels. But now, we might
12 want to worry about it from MND type events which
13 might have a different radiation level too.

14 But anyway, it's just a question that I
15 keep asking.

16 MEMBER MARCH-LEUBA: My personal sense in
17 how they -- it's up to the seven guys up there on the
18 microphone, is that this sensor will measure a level
19 during initial startup as you empty the containment.

20 And then it will be out of the scale for
21 two years. It will not be giving you a signal. It
22 will be giving you noise. Hopefully not much noise.

23 But, there won't be any bounce back for
24 two years. So, if there is a failure of those
25 sensors, it's likely it will go unnoticed for two

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1 years until you need it.

2 So, that's like, I mean, you have to
3 increase their liabilities significantly. When you're
4 -- every second you're measuring and getting a signal
5 that this is changing and it's oscillating, you have
6 confidence that the sensor is alive.

7 When for two years you have it completely
8 off scale, you've got to make sure that thing is
9 working. Just something for you to think about.

10 MR. AYALO: So what we -- so I would like
11 to respond to that. But I think that in terms of some
12 of the details involved, I'd like to discuss that in
13 the closed session.

14 But, I would disagree with your statement
15 that it would be out of calibration. Because what
16 you're essentially saying is if I have an empty tank,
17 you know, a level sensor is always out of calibration,
18 which is not necessarily true.

19 But in the closed session I'd like to go
20 into more details on how you can demonstrate that the
21 level sensor itself, it's still working, even with no
22 level in any tank, or even in our containment.

23 MEMBER MARCH-LEUBA: And you can save that
24 for our visit to Corvallis in a couple of weeks if you
25 want to. You know, I just want to bring -- the fact

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1 that you thought about it, that's -- you're 90 percent
2 there.

3 MR. AYALO: Yeah, thank you.

4 MS. MCCLOSKEY: Okay. So this is Meghan
5 McCloskey again. Moving onto slide nine. When we
6 look at the functions of the module protection system
7 and look at our design basis events mitigation, this
8 slide summarizes broadly the types of actuation that
9 we expect to see for the different event types.

10 Of course reactor trip is common. To
11 provide reactivity control and reduce heat loads to
12 the decay heat level.

13 For the vents that are increasing heat
14 removal transients, secondary site isolation isolates
15 the module from the malfunction causing most of the
16 event. And then BHRS heat removal can be provided as
17 necessary.

18 For the decrease in heat removal events,
19 the decay heat removal system provides any removal
20 following malfunctions that indicate normal --
21 secondary type cooling is unavailable.

22 The reactivity and power transients, the
23 reactivity control is most important. It's provided
24 by reactor trip and the demineralized water isolation.

25 The increase in RCS, in reactor coolant

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1 system inventory transients, is mitigated by
2 containment isolation. Which controls the inventory
3 -- which controls the additional inventory from the
4 chemical and volume control system.

5 And finally, the decrease in reactor
6 coolant system inventory transients is mitigated by
7 containment isolation to control the inventory and
8 mitigate dose consequences.

9 For LOCA and valve opening events, and
10 ECCS, emergency core cooling system actuation
11 establishes the cooling path. And finally, quality of
12 the light water reactor trip protection.

13 Going to slide 10, the Chapter 15 design
14 basis initiating events consider internal events that
15 could affect a module while operating at power
16 conditions from hot zero power to hot full power
17 conditions.

18 And the Chapter 15 analysis are performed
19 for a single module response. For the NuScale we
20 primed chill systems, the reactor pool ultimate heat
21 sink is the key safety related shared system.

22 And it's most relevant in the long-term
23 cooling analysis. And therefore the long-term cooling
24 analysis would consider the effects of up to 12
25 modules rejecting heat to the reactor pool ultimate

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1 heat sink.

2 In order to identify these Chapter 15
3 design basis events, we used a systematic process to
4 evaluate the NuScale system. And components that
5 could affect the module response and cause reactor
6 trip or other type of plant transients as well as
7 considering industry experience and other advanced
8 module PRAs in order to identify the skillful design
9 basis events.

10 And we evaluated the heat scale systems
11 and components to identify failures that could affect
12 the module process, the module response, focusing as
13 well as on NuScale system designing system -- design
14 features such as the containment evaluation system and
15 the emergency core cooling system.

16 When we had identified the failures we
17 categorized them based on their effect in the module.
18 And those categories are consistent with the operating
19 white water plants.

20 And we also have NuScale specific
21 phenomena and event progression that are analyzed as
22 part of Chapter 15, namely feedwater flexibility that
23 was discussed this morning, and return to the power
24 analysis that we have later in this presentation.

25 Going to slide 11, this slide highlights

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1 the NuScale unique Chapter 15 events. Because our
2 other events are consistent with light water reactors,
3 or are not applicable to our design because of the
4 natural circulation operation of the plant.

5 So any unique events, the first is
6 discussed in Section 15.1.6, the loss of containment
7 vacuum. During normal operation, the containment is
8 operated at vacuum conditions in order to minimize
9 heat transfer to the coolant and allow for leakage
10 detection.

11 And the loss of vacuum is postulated due
12 to failures in the evacuation system. Containment
13 flooding is postulated due to a break in piping that
14 can carry its reactor component cooling water to the
15 control rod drive mechanism.

16 And either of this loss of vacuum or
17 flooding increases the heat transfer from the reactor
18 pressure vessel to the containment in terms of normal
19 operating conditions. And so it's a -- we're hoping
20 a very slow increase in a heat removal event that's
21 really non-limiting compared too other over cooling
22 events analyzed tentative to inspection.

23 The inadvertent operation of the decay
24 heat removal system is another unique NuScale AOO. A
25 place for the heat removal system is inside for decay

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1 heat removal if an inadvertent operation causes a
2 decrease in heat removal while that power operation.

3 And there are several variations of this
4 event. It may be a single valve opening, which
5 results in a diversion of feedwater flow from the
6 steam generator too over the decay heat removal system
7 condenser.

8 Or it may be an inadvertent signal that
9 actuates the DHRS valve and closes the secondary
10 isolation valve on one or both frames in the system.

11 In the event of 15.6.6 in the decreased
12 inventory events, the inadvertent operation of the
13 emergency core cooling system is a NuScale unique
14 event. It also works without the valve opening event.

15 And then although opening of a recirculation
16 valve, or a vent valve is not expected to occur during
17 the life of the plant, this was conservatively
18 classified as an anticipated operational occurrence.

19 And we extended the LOCA evaluation model
20 to the valve opening events and used the inner related
21 pod code to demonstrate that margin two critical heat
22 flux is maintained during the event.

23 Now moving on to slide 12. For Chapter 15
24 analysis, there are a number of assumptions that are
25 deterministically applied.

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1 For the -- as a passive plant in the
2 NuScale design, we assume no operation actions to
3 demonstrate that none are required to achieve safety
4 functions for at least 72 hours after the limiting
5 event.

6 In addition to the -- during the
7 initiating event, the worst single failure of a
8 safety-related component assumes, and then the
9 scalability of, and power considered to determine
10 whether loss of power is more conservative with
11 respect to margin two acceptance criteria.

12 In terms of the scope of the event
13 progress, the Chapter 15 analysis are performed until
14 a safe, analyzed condition is reached, by which we
15 mean that the module protection system responses to
16 the initiating events have occurred.

17 We've demonstrated that there's margin to
18 the acceptance criteria in system parameters such as
19 inventory, temperatures, pressures, are trending in a
20 favorable direction.

21 After the end of that short term response
22 and demonstration of the safe stabilized condition,
23 the longer term progression is considered. And the
24 long-term cooling analysis addresses emergency core
25 cooling systems, long-term decay, and residual heat

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1 removal.

2 The return to power analysis demonstrates
3 that this provides acceptable fuel design limits are
4 met. This hopefully continues in the module with
5 running control water is struck out of the core, and
6 a return to power occurs.

7 And if power is available but no operator
8 actions are assumed, then extended DHRS operation is
9 good and considered.

10 Going to slide 13, this slide and the next
11 slide illustrate the effect of loss of power, AC and
12 DC power functions on the Chapter 15 event
13 progression.

14 This slide is for the non-LOCA events that
15 are initially mitigated by decay heat removal system.

16 And so occurring these events relatively early in the
17 transient, the module protection system will detect
18 the upset condition and actuate reactor trip, and if
19 necessary the decay heat removal system.

20 On the search side, with AC and DC power
21 available, that point is the end of the module
22 protection system actuation. And typically stable,
23 the completely mobile system is established by a half
24 hour into the event.

25 If both trains of decay heat removal

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1 system are operating and that cooling is assumed, a
2 wide recovery may occur in the first two hours. If
3 this occurs, it's a self-limiting condition that is
4 maintained over the 72-hour time frame.

5 In the schematic in the middle of the
6 slide, its AC power loss, but DC power remains
7 available. And the main difference in the event
8 progression occurs at 24 hours when the module
9 protection system timer actually BCCS in module
10 transition from the decay heat removal system to the
11 emergency core cooling system cooling.

12 And then finally at the bottom of the
13 slide, if neither AC nor DC power are available, then
14 the emergency core cooling system valves are actuated
15 early in the transient progression due to the assumed
16 loss of power.

17 And this occurs while the reactor coolant
18 system is at high pressure conditions, and the
19 inadvertent actuation block actuates to hold the
20 emergency core closed and valves closed while decay
21 heat removal system cooling is established.

22 MEMBER MARCH-LEUBA: Sorry, can you go
23 ahead and tell us, the last one, no AC, no DC, what
24 tapes the ACC is about opening?

25 MS. MCCLOSKEY: The assumed loss of DC

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1 power -- actually it's the ECCSL.

2 MEMBER MARCH-LEUBA: So it's an
3 instantaneous opening of the solenoids and the I --
4 whenever you reach the IAD pressure it will open.

5 MS. MCCLOSKEY: Yes. Yes. So the loss of
6 DC power, or power in the solenoid --

7 MEMBER MARCH-LEUBA: Okay. Got it.

8 MS. MCCLOSKEY: Yeah. Okay. And the
9 decay heat, still on that last one, the decay heat
10 removal system cools the module until -- until the
11 module is sufficiently depressurized that the IAD
12 releases. And then the ECC itself opens and the
13 module transition.

14 So, on slide 14, this covers the same
15 scenarios as previously, but for a look at PSI. And
16 in those cases with power available, the module
17 protection system actuates a reactor trip contained in
18 isolation.

19 And the normal response of the design is
20 to actuate the decay heat removal system as well.
21 When the containment level is reached, then the
22 emergency core cooling system is actuated and in a
23 realistic event progression, the emergency core
24 cooling system valves are expected to open at that
25 time.

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1 The -- if AC power is lost, but DC power
2 is available, then the event progression is
3 essentially the same. If DC power is lost in addition
4 to the AC power, then the ECCS valves are actuated
5 while the reactor cooling system is still at high
6 pressure condition.

7 And again, the inadvertent actuation block
8 acts to hold the valves, the ECCS main valves closed
9 until the reactor coolant system is sufficiently
10 depressurized through the break to which the
11 inadvertent actuation block relief comes.

12 We're ready to do to --

13 MEMBER MARCH-LEUBA: What we're asking --
14 sorry. I don't remember, what is the IAB actuation
15 pressure?

16 MS. MCCLOSKEY: The range is the -- the
17 threshold pressure range is -- and the DCA is 1,000 to
18 1,200 PSI. And the release pressure range is 1,000 to
19 1,200 PSI.

20 MEMBER MARCH-LEUBA: Yeah. So the
21 releases are Delta P, between the core and the
22 containment, correct? Yeah.

23 MS. MCCLOSKEY: Not technically.

24 MEMBER MARCH-LEUBA: Okay. Thank you.

25 MS. MCCLOSKEY: Okay. Slide 14. The

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1 eliminating transient results. I'm sorry, slide 15.
2 Because some of the limiting Chapter 15 transient
3 results for the various acceptance criteria that are
4 addressed in the Chapter 15 analysis.

5 In the NuScale design, the maximum reactor
6 coolant system pressure is mitigated by the reactor
7 safety valve. And so there are several events that
8 result in safety valve lift and have similar maximum
9 pressures around 21 steps to get PSIA.

10 And that's analyzed considering the safety
11 valve set points, and accounting for direct. And it
12 remains well below the acceptance criteria of 2,315
13 PSIA or 2,520 PSIA depending on the event
14 classification.

15 In the maximum steam generator pressure,
16 the steam generator design pressure is equal to the
17 reactor coolant, with some design pressure at 2,100
18 PSIA. And the maximum pressure in our transient
19 analysis occurs after decay heat removal system
20 actuation as the heat transfer for the primary too
21 secondary generate steam and begin condensing in the
22 decay heat removal condenser.

23 So from a physical standpoint, the
24 secondary pressure is limited to the saturation
25 pressure for the primary site hot temperature. And as

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1 a result in all of our cases, the margin to second
2 criteria is on the order of about 700 PSI.

3 And for the minimum critical heat flux
4 ratio, the limiting non-LOCA event that's analyzed
5 with VIPRE and the subchannel analysis methodology is
6 the single rod withdrawal case.

7 And reactivity events, the submitting
8 result is 1.61, which remains above the 9595
9 acceptance criteria of 1.284.

10 For the inadvertent opening of a valve,
11 our limiting case is the opening of a recirc valve
12 which then launch with NRELAP5. And the results are
13 shown on the slide here, 1.41 compared to the limit of
14 1.13.

15 And then finally for a LOCA we also, the
16 NuScale evaluation model demonstrates that margin to
17 critical heat flux remains and level remains above the
18 top of the core. And so in both of these cases, we
19 continue to show margin.

20 MEMBER MARCH-LEUBA: And both the IRV
21 opening, and the LOCA, use the LOCA methodology?
22 Which does not use the VIPRE subchannel CHF
23 correlation, correct?

24 MS. MCCLOSKEY: So -- correct. The VIPRE
25 subchannel methodology is not used for either the

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1 valve opening event or the LOCA analysis.

2 The Appendix D of the LOCA topical report
3 describes the valve opening methodology and how the
4 margin of critical heat flux is demonstrated there.

5 MEMBER MARCH-LEUBA: Yeah. One concern
6 that I --

7 MS. MCCLOSKEY: And it continues to PSI.

8 MEMBER MARCH-LEUBA: One concern I raised
9 during the subcommittee meeting, and I assume we're
10 still having a closed session Chapter 15, correct,
11 Mike?

12 MEMBER CORRADINI: Yes.

13 MEMBER MARCH-LEUBA: The answer is yes.
14 I will talk more about that then, was the differences
15 in MCHFR estimates for this various state by the two
16 methodologies, the VIPRE channel and the LOCA
17 methodology, which were very large.

18 And indeed this number that you show her,
19 1.796 for LOCA is higher than the steady state MCHFR
20 that you show later.

21 So we'll -- just remember that we'll talk
22 about this during the closed session. Keep going.

23 MS. MCCLOSKEY: Yeah. Okay. All right,
24 that finishes the overview of Chapter 15 for the
25 committee's presentation.

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1 And the next key slides summarize the
2 Chapter 621 containment response analysis. So, going
3 to slide 17, the containment pressure and temperature
4 analysis is performed for a spectrum of primary and
5 secondary sides, mass and energy release events to
6 demonstrate that there's margin to the design pressure
7 and temperature limits considering the worst single
8 loss of failure and loss of the power scenarios.

9 The containment analysis technical report
10 describes the methodology that we use to analyze the
11 con -- pressure and temperature response. And due to
12 the design of the module with the small high pressure
13 containment and the design of the emergency core
14 cooling valves opening on high containment level or
15 the release pressure, the mass in energy released from
16 the primary is very tightly coupled to the containment
17 pressurization response.

18 And therefore, we use NRELAP5's model, the
19 integrated response of the reactor pressure vessel
20 blow down and the containment pressurization and heat
21 removal.

22 The qualification of NRELAP5 for this
23 analysis was based on the qualification and the scale
24 plant modeling approaches that are described in the
25 LOCA and the non-LOCA evaluation models. Because the

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1 evaluation models were developed considering
2 containment pressure as a figure of merit from the
3 codes developed for VDM.

4 And therefore for the containment sent
5 out, the overviews are based on the LOCA and the non-
6 LOCA event analysis models with the initial boundary
7 condition modified to biased mass and energy releases
8 from the reactor pressure vessel to maximize the
9 containment pressure and temperature response.

10 Our limiting results for peak pressure is
11 a recirculation valve opening event. And the maximum
12 pressure of 986 PSIA is just below the design pressure
13 limit of 1050.

14 And the maximum temperature event is a
15 break in the reactor coolant system injection line,
16 with a maximum core temperature of 526 degrees
17 Fahrenheit less than the 550-degree design
18 temperature.

19 The slide 18 summarize the peak heat of
20 events. And shows the containment pressure for the
21 maximum pressure case.

22 So at time, this is initiated by the
23 inadvertent research valve opening at time zero with
24 assumed loss of both AC and DC power. Which results
25 in a reactor trip and secondary site isolation.

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1 Very rapidly you get a containment
2 pressure signal. And the plot shows the containment
3 pressure increases rapidly due to the blow down
4 through the valve.

5 And when the IAD release pressure is
6 reached, the ECCS valves open at 74 seconds into the
7 event. And the containment pressure increases rapidly
8 from that point including the additional mass and
9 energy release.

10 And peaks around 91 seconds after the
11 event initiation. That pressure is rapidly turned
12 around as the energy is absorbed into the containment
13 wall.

14 After 700 seconds in this case, heat
15 transfer to the reactor pool is established. And the
16 heat removal through the containment wall to the pool
17 exceeds the decay and residual heat transfer through
18 the emergency core cooling valves to contain them.

19 It's after this point there's no driver
20 for any subsequent increase in keeping the heat times
21 crossed. And the pressure is reduced below 50 percent
22 of the peak value within the first hour of the event
23 initiation.

24 All right, hearing no questions, we'll
25 move onto slide 20 to summarize the long-term cooling

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1 analysis results.

2 In the NuScale design there are different
3 events and analysis assumptions that can result in
4 emergency core cooling systems cooling. Obviously the
5 LOCA and the valve opening events initially transition
6 to ECCS cooling.

7 And for long-term cooling, the full bright
8 spectrum is evaluated, as well as the valve opening
9 event. If a loss of power is assumed, the non-LOCA
10 event will transition from decay heat removal system
11 to ECCS cooling.

12 And the non-LOCA events specifically
13 considered are steam generator tube failure, because
14 that event results in a decrease in RCS inventory.
15 And therefore there is a lower inventory event on the
16 reactor coolant system when emergency core cooling
17 system valves open.

18 And so we evaluate that to demonstrate
19 that there is sufficient inventory maintained to
20 support ECCS cooling.

21 We analyze a loss of feedwater flow
22 because that's a heat shift event. And so there's a
23 small increase in reactor coolant system energy due to
24 the loss of cooling.

25 And certain events didn't have any

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1 significant impact on the long-term cooling through
2 ECCS. And we also considered a generic decay heat
3 removal system cool down.

4 To find the long-term cooling calculations
5 we -- the transient calculations were performed for 12
6 and a half hours after emergency core cooling system
7 actuation. Which is sufficient to capture the minimum
8 level of what occurs during the long-term phase, and
9 demonstrates a long-term level of recovery.

10 To evaluate the 72-hour design basis
11 coping period, the state point calculations, we ran
12 state point calculations to establish the quasi-steady
13 condition. And that's really a function of decay
14 heat.

15 For those state point calculations the
16 decay heat was set to a constant value in the RELAPS
17 model. And the models run until the temperatures are
18 converged.

19 There are several different acceptance
20 criteria that are of interest in the long-term cooling
21 analysis. And in general to evaluate these different
22 acceptance criteria, we have two different biasing
23 approaches.

24 In the minimum level cases, these are
25 performed to demonstrate that liquid level remains

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1 above the top of the core. Because in the long-term
2 phase, the minimum level occurs in scenarios where the
3 containment heat transfer is maximized, and the
4 reactor coolant system energy is maximized.

5 And this results in the maximum amount of
6 vapor generated in the core that needs to be vented
7 through the vent valves, increasing the Delta P over
8 the vent valves.

9 And this requires the highest liquid level
10 in containment to maintain sufficient driving tests to
11 balance since that pressure drop results in a minimum
12 level over the top of the fuel.

13 The minimum temperature cases are
14 evaluated to demonstrate margin to boron precipitation
15 limits. And the maximum temperature cases are
16 performed to demonstrate that the under conditions of
17 high decay heat and high cool temperatures, and lower
18 cool levels, the containment provides sufficient heat
19 removal to maintain decrease in climbing temperatures.

20 MEMBER MARCH-LEUBA: Let me interrupt you
21 here. You keep talking in this slide about boron
22 precipitation, like boron plating. A different
23 mechanism that has been proposed, doesn't result in
24 plating or precipitation, but it results in
25 homogeneous distribution of boron between the core and

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1 the containment.

2 And what I mean by -- and this applies
3 more to Chapter 20 when the modules will be under ECCS
4 operation for weeks at a time. And so.

5 So, when you're in ECCS operation, you're
6 steaming steam through the upper valves, the RSVs or
7 RVVs. And that steam is essentially the steam water.
8 It does not have any boron or any boron to speak of
9 with them.

10 So you are dumping distilled water into
11 the containment. And one place or another, you are
12 concentrated boron in the core.

13 Now, after a couple of weeks of this
14 operation, you will have essentially pure distilled
15 water in the containment, which makes the down comer
16 to be pure distilled water.

17 The riser will have a higher concentration
18 of boron, whether liquid or solid, I don't know. And
19 then the possibility exists that that cold distilled
20 water in the riser will, during one of these passes,
21 will make it into the core and regain criticality.

22 So, a concern I have, I'm not as much
23 concerned about precipitation as the homogeneous
24 distribution of the boron concentration or -- in the
25 core is what I'm going to worry about.

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1 But, clearly after you operate for two or
2 three weeks on the distilled mode, you have clean
3 water on the containment and the down cover. And it's
4 just waiting for any transient to make it into the
5 core.

6 That's the concern I have more than
7 plating. I know you guys have done some boron
8 transport analysis that you showed us.

9 MS. MCCLOSKEY: Um-hum.

10 MEMBER MARCH-LEUBA: I just wanted to put
11 into the record that the homogeneity is on the boron
12 distribution after a couple of weeks of distilled
13 operation, need to be a list of, a list for that.

14 And you are -- you -- you're driving too
15 much precipitation as being the only problem. But,
16 there are other ways to put high reactivity into your
17 core. Which are unlikely to happen.

18 MS. MCCLOSKEY: Well, if we're operating
19 in ECCS cooling conditions for long-term, because our
20 decay heat rates are very low, then the recirculation
21 rates from the containment is the down comer into the
22 bottom of the core are also quite low.

23 And so under those conditions, we have not
24 identified any realistic mechanism that can result in
25 a large slug of dilute water entering the bottom of

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1 the core and causing a return to power.

2 MEMBER MARCH-LEUBA: But then yes,
3 describe one for you. I mean, it -- granted decay
4 heat is not that much. But in all the safe reactions
5 we had, it was sufficient to melt the core.

6 So there's energy in there. And you are
7 operating at one --

8 MS. MCCLOSKEY: Well, we have water over
9 the core. So, --

10 MEMBER MARCH-LEUBA: Right. You're not
11 going to melt the core. But you have 1 percent at the
12 beginning and then you have 0.5 percent. And then you
13 establish at some level, which is non-reachable,
14 you're boiling a lot of water.

15 And all that water ends up in the
16 containment, is distilled water. Okay. And you're
17 then driving it into the down comer over weeks, right.

18 And this is not two hours. This is a week
19 or two. And eventually the steady-state condition, if
20 you continue to operate distilled water and put it in
21 the containment, the containment will be distilled
22 water.

23 The down comer will be distilled water.
24 And the riser will have an increased concentration of
25 boron. Now hopefully that boron is concentrated in

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1 the core.

2 But, if you've been flushing on the top of
3 the riser like some calculations show, you might be
4 concentrating the boron in the top of the riser and it
5 stays there, dissolved.

6 So, there are -- there are -- go ahead.

7 MS. MCCLOSKEY: What we've seen in the
8 literature for large scale tests is that there's no
9 significant gradient of boron concentration in large
10 scale tests.

11 We've also considered the -- if you
12 postulate the boron that's concentrating in the riser
13 above the core, then the fluid density in that region
14 would increase due to the boron concentration.

15 And we considered the really tailored
16 instability mechanism to evaluate how large of a
17 concentration gradient could develop with the type of
18 geometry in from the riser.

19 And based on that analysis, we are seeing
20 only a very small grad -- a concentration gradient
21 that could develop.

22 MEMBER CORRADINI: I don't understand the
23 last thing you said. Could you repeat that please?

24 MS. MCCLOSKEY: We considered -- if you're
25 concentrating -- if you postulate that boron is

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1 concentrating above the core region some place, then
2 the fluid density would increase slightly due to the
3 boron concentrate, the difference in boron
4 concentration there.

5 And you have either postulating heavier
6 fluid, a more dense fluid over the top of lower dense
7 fluid with a slightly lower boron concentration. And
8 that's an inherently unstable situation that would
9 tend to resolve itself based on the raised tailored
10 instabilities.

11 MEMBER CORRADINI: So, you're basically
12 saying that you would create a circulation pattern
13 that would remix?

14 MS. MCCLOSKEY: Yes.

15 MEMBER CORRADINI: Okay. All right.
16 Thanks, I didn't appreciate what you were saying.
17 Thank you.

18 MEMBER MARCH-LEUBA: But my point is, let
19 me give you the core and riser will be uniformly
20 mixed. And through the raise stability you have
21 circulation.

22 And your boron in the core and riser will
23 be uniform. The boron in the down comer and
24 containment will eventually go down to zero. Because
25 there is nothing there to replenish it.

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1 And there are mechanisms to remove it.
2 Which is making it -- driving it through the core. If
3 you have any reservation, like for example a return to
4 criticality, when you suddenly increase the flow
5 through the core, you will sack, cold, and break the
6 water from the down comer. And that will give you a
7 positive feedback.

8 MS. MCCLOSKEY: I think the following
9 scenario --

10 MEMBER MARCH-LEUBA: I'm just saying that
11 there is --

12 MS. MCCLOSKEY: The scenario that's being
13 postulated that would cause such a large probation,
14 because --

15 MEMBER MARCH-LEUBA: A return to power?

16 MS. MCCLOSKEY: We're shut down at the --
17 we're shut down at this point and we're concentrating
18 boron in the core and riser regions, which will
19 prevent a return to power.

20 MEMBER MARCH-LEUBA: Yeah. I'm just -- my
21 -- I just wanted to put it on the table. That there
22 is a possibility.

23 I mean, there is clearly, the design of
24 the ECCS results in a nonhomogeneous distribution of
25 boron across the vessel and containment. And high

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1 reactivity water concentrates at the core inlet.

2 It may just be the down comer. It may not
3 be the lower plenum. The potential exists that it has
4 to be looked at. That's what I'm saying there.

5 You're going to just dismiss it that's
6 impossible because it is there. It's waiting to
7 happen.

8 MR. RAD: Yeah, well we have looked at it.

9 MS. MCCLOSKEY: I think that I --

10 MR. RAD: Yeah, I'm sorry. Meghan, this
11 is Zack. Yeah, so we're sensitive to your concern.
12 And I don't think anybody is saying it's not something
13 that should be considered or looked at.

14 In fact we've spent a lot of time looking
15 at it. And a lot of discussion with the staff on it.
16 I don't think we're quite resolved on agreeing on all
17 of the assumptions.

18 But it's certainly something that we're
19 looking in. So, it may not show up on this slide, but
20 it's certainly a topic of discussion with the NRC, and
21 something that we've been spending a lot of time on.

22 MEMBER MARCH-LEUBA: As I said before, if
23 you thought of the problem, you're 90 percent over the
24 solution. The problems happen when you have not
25 thought of what would happen.

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1 MR. RAD: right.

2 MEMBER MARCH-LEUBA: And we need to be
3 thinking about that. And this, we'll discuss this
4 issue during the Chapter 20, a little longer, because
5 I certainly believe that reactivity, subcriticality,
6 now there's a criticality if there's a down should be
7 one of the safety goals.

8 And we keep saying it's not. I mean, you
9 keep saying it's not that concerned, because this
10 cannot possibly happen.

11 There are physical forces driving you
12 towards a return to power. There is cold un-braided
13 water in the region of your core.

14 We'll discuss more of that. As I said,
15 this happens when you're abating in these conditions
16 for two, three, four, or five weeks. You start
17 concentrating un-braided water, a lot of it.

18 All right. So we'll follow up later.

19 MS. MCCLOSKEY: Okay.

20 MEMBER MARCH-LEUBA: Keep on going.

21 MS. MCCLOSKEY: I think we are -- we go to
22 slide 21. This slide presents results for the minimum
23 level case for the injection line break.

24 And the plot shows the riser collapse
25 level as a function of time for both the 100 percent

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1 injection line break, and a 5 percent injection line
2 break.

3 And for the 100 -- for the larger break
4 the emergency core coolant system actuates early in
5 the transient. And then as the inventory
6 redistributes the biasing conditions to maximize the
7 reactor coolant system energy and minimize the
8 emergency core cooling system valve capacity, results
9 within a higher pressure difference between the
10 reactor pressure vessel and the containment vessel.

11 And that requires a higher driving head
12 and containment to maintain circulation and continue
13 venting vapor from the reactor pressure vessel to
14 containment.

15 So for this case, in long-term, the
16 minimum level occurs between three to five hours after
17 the emergency core cooling system actuation.

18 For the smaller break, per the LOCA
19 evaluation model that NuScale has developed, the decay
20 heat removal system is not credited. And therefore
21 there's an extended time of break flow into the
22 containment before the ECCS valves are calculated to
23 open.

24 And that minimum level, that's contained
25 in the 5 percent break occurs during the short term

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1 LOCA phase as the emergency core cooling system
2 actuates. And then longer term, the decay heat is
3 lower. And the long-term cooling level disk is less
4 pronounced compared to the large break scenario.

5 But for both break scenarios, the long-
6 term level recovers towards equilibrium as the
7 pressure difference between these reactor pressure
8 vessels and the containment reduces with decay heat.

9 Going to slide 22, this slide presents
10 results from the injection line maximum temperature
11 case to demonstrate that even under conditions with
12 low reactor pool level assumed and high pool
13 temperature.

14 The containment provides effective
15 cooling. And the clotting temperatures continue to
16 decrease long-term.

17 In these cases, the system pressure is
18 running higher than they do in the minimum level
19 cases. And the level depression during the long-term
20 is not observed.

21 So, slide 24, the last section of the
22 presentation discusses Section 15.06 and return to
23 power analysis. The NuScale design includes an
24 exemption to the GDC-27.

25 So we have principal design criteria 27 to

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1 ensure that safety related reactivity control system
2 is designed to achieve and maintain a subcritical
3 core. And ensure that fuel integrity for an extended
4 over cooling in combination with a partial cell of
5 reactivity system, namely of stuck rods as assured.

6 The compliance with PDC-27 is demonstrated
7 with three parts of analysis. One the immediate
8 shutdown from hot conditions is -- the control rod
9 self-sufficient work to protect the reactor coolant
10 pressure boundary. And SAFDLs takes the margin for
11 the worst rod stuck out of the core.

12 Two that cold shut down conditions are
13 achieved with all control rods fully inserted. And
14 third, that the loss of shutdown margin consequences
15 for a case where the highest worth control rod is
16 stuck out of the core, that those consequences are
17 benign.

18 And that's demonstrated by showing that
19 the return to power does not challenge SAFDL, and the
20 critical power level is sufficiently low that the
21 decay heat removal system, or the emergency core
22 cooling system continues to provide sufficient removal
23 capacity.

24 And then finally, the NuScale exemption
25 request describes the probability that the combination

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1 of conditions that results in a loss of shutdown and
2 return to power with a single rod stuck out of the
3 core is a small less stained one E to the minus six
4 per reactor year.

5 MEMBER MARCH-LEUBA: And that's -- hello,
6 hold on a minute. Does your PDC-27, we learned the
7 other day that there is a requirement that every
8 single transient that could result in sort of pow --
9 in a return to power, which is all of them, cannot
10 have -- must not have CHF violations, even at the
11 beginning of the transient.

12 Do you remember that discussion?

13 MS. MCCLOSKEY: Yes.

14 MEMBER MARCH-LEUBA: Yes. That was a
15 surprise to me. Because I was not aware of it. So,
16 is that part of our PDC-27? Or is that part of your
17 analysis? Second one raised.

18 MR. RAD: I can answer that question for
19 you Meghan. Yeah, so that's built into our licensing
20 basis. It's a function of the acceptance criteria and
21 the methodologies we've chosen.

22 So at this point we don't have
23 methodologies that analyze transients post CHF to
24 demonstrate that there is no fuel failure.

25 So that's the natural limit based on the

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1 methodologies that we use. And so it is a functioning
2 of our licensing basis.

3 There's the potential that we could
4 analyze further in the future, and understand what
5 happens to the fuel post-CHF. And then analyze any
6 subsequent consequences of a potential postulated
7 return to power.

8 MEMBER MARCH-LEUBA: You can also go into
9 my next question. Which is, you're happy today with
10 this analysis, because you can do it.

11 But let's say you do a power upgrade next
12 year. Then you have -- you will have to come back and
13 get another exception, because I believe the exception
14 includes that requirement.

15 MR. RAD: Our exempt -- or our goal is to
16 not have that specific requirement be in the PDC. But
17 to have it be part of the licensing basis subject to
18 NRC review it changes in the future.

19 MEMBER MARCH-LEUBA: And what does the
20 staff think about it?

21 MR. RAD: I'm not sure what the final
22 wording is. But I think we're pretty close to
23 alignment.

24 MEMBER MARCH-LEUBA: Okay.

25 MEMBER CORRADINI: We should ask the staff

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1 when they're up.

2 MEMBER MARCH-LEUBA: Yes. Because it
3 really should be part of the PDC. And if it is a
4 requirement, it's like one.

5 And it should be harder for you to remove
6 it in other words.

7 MR. RAD: And it is an administrative
8 exercise to request a separate exemption. But,
9 there's also a lot of work to do to change our
10 methodologies, to get NRC approval for that as well.

11 MEMBER MARCH-LEUBA: Well, okay.

12 MS. MCCLOSKEY: Okay. Going to slide 25
13 now. In the DCA submittal, we had a bounding dynamic
14 return to power analysis to demonstrate that there was
15 no challenge to minimum critical heat flux ratio
16 limits, the reactor coolant pressure boundary, or the
17 containment boundary.

18 And as we've worked through the DCA
19 review, we've worked to develop an alternate approach
20 to analyze the return to power analysis that allows
21 for a clearer demonstration that the range of
22 conditions under which a return to power could occur,
23 are analyzed.

24 And so this slide and the next demonstrate
25 that the new app -- the approach and it's consistent

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1 with what was presented to the subcommittee last
2 month.

3 The mechanisms that lead to a loss of
4 shutdown margin and return to power are first
5 moderator cool -- overcooling, which can be postulated
6 from either decay heat removal systems cooling, or
7 emergency core cooling systems cooling.

8 And under cold reactor pool conditions
9 particularly, either of these systems can cause a
10 fairly rapid temperature decrease in the first several
11 hours after the event. Which causes the reactor
12 reinsertion from the increased moderation.

13 And then second, xenon decay after
14 shutdown cause a slow post-shutdown reactivity
15 insertion. So for the return to power
16 characterization, NRELAP5 is run for several state
17 points with a specified power level to characterize
18 the steady-state reactor coolant system temperature
19 and a function of that power level.

20 And we've run those state points for decay
21 heat removal systems, cooling with the riser remaining
22 covered, decay heat removal system cooling with the
23 riser uncovered, and core emergency core cooling
24 system cooling.

25 And then simulate is run to calculate the

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1 critical power level assuming the worst rods stuck out
2 as a function of reactor coolant system temperature.
3 And where these state points from the NRELAP5 and
4 simulate calculations intersect, gives the power level
5 and reactor coolant system temperature where a return
6 to power could occur.

7 And the conclusions from this analysis are
8 summarized here. Currently based on best system, it
9 simulates results. But a loss of shutdown margin may
10 occur around 40 hours with, if you're assuming the
11 third one in the RCS. And so end of cycle conditions
12 and moderator temperature feedback when this occurs
13 after xenon decay.

14 The resistant coolants in temperatures are
15 quite low. Based on best estimates, simulate results
16 less than about 200 degrees Fahrenheit.

17 ECCS cooling can drive the module to lower
18 temperature conditions. And therefor slightly higher
19 power results.

20 There are -- if you have a sufficiently
21 high pool temperature, this limits -- it is a self-
22 limiting nature of the design limit goal the module
23 temperatures will reach. And that can preclude a
24 return to criticality with worst stuck rods.

25 But our power levels remain very low,

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1 around 1 to 2 percent. And at pool boiling CHF
2 evaluation demonstrates the parts into minimum
3 critical heat flux ratio.

4 MEMBER MARCH-LEUBA: How confident are we,
5 or basically let me rephrase the question. How we run
6 some uncertain analysis on that 1 to 2 percent power
7 level, could it be 15 to 25 percent?

8 How do we know it's not 15 percent power?

9 MS. MCCLOSKEY: We are placing analysis
10 that consider uncertainties in the simulated results.
11 And what that shows, is it shifts the critical power
12 level.

13 It shifts the temperature up about 25
14 degrees. Which results in full power level that's
15 still less than 2 percent.

16 So we're nowhere near 15 or 20 percent
17 power level. Even the bounding dynamic return to
18 power analysis that was performed in the DCA system,
19 I think the peak power was approximately 14 percent
20 under that, the conditions of those analyses.

21 And that was based on a RELAP analysis and
22 point kinetics modeling.

23 MEMBER MARCH-LEUBA: Yes. Which use a
24 constant MPC value for the whole analysis, correct?

25 MS. MCCLOSKEY: Yes.

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1 MEMBER MARCH-LEUBA: Yes. Which is very,
2 very valid core submission. So, the simulate is a
3 much better approximation.

4 Okay. Thank you.

5 MS. MCCLOSKEY: Okay. Slide 26 shows the
6 results from this return to power characterization.
7 And again, this is what was presented to the
8 subcommittee.

9 The X axis is the critical power level.
10 And the Y axis is the reactor coolant system average
11 coolant temperature.

12 And the orange, blue, and purple dots are
13 the results of the NRELAP5 state point analysis, where
14 we specified the power level in NRELAP5, and
15 calculated the steady-state reactor coolant system
16 temperature for either DHRS or ECCS cooling.

17 And the black line is the critical power
18 level calculated by simulate as a function of the
19 reactor coolant system temperature. And so for ECCS
20 cooling, the critical power level in these cases was
21 about 1 percent power, because the lower temperatures
22 are reached.

23 But there's -- at the cold, low power
24 conditions there are significant margins of CHF and
25 the emergency core cooling system or the decay heat

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1 removal systems have sufficient capacity to continue
2 cooling the module.

3 MEMBER MARCH-LEUBA: Yeah. The concern we
4 have with this figure is related to my previous
5 question of uncertainties.

6 That black line, if you have chosen to put
7 temperature in the X axis and power in the Y axis,
8 that black line is almost vertical. Or in this case,
9 almost horizontal.

10 And a very tiny change in temperature
11 results in a very large change in critical power. If
12 you've run any sophisticated analysis with that type
13 of sensitivity, you will find that -- it's a very
14 large degree with it. You better not miss it by much.

15 Anyway, just an observation.

16 MS. MCCLOSKEY: We've got a -- yeah, we've
17 got a fairly tight frame on the X axis as well here.
18 And what we've seen is if you -- if you shift that
19 simulate line up by about 25 degrees, then the ECCS
20 critical power level increases to about 2 percent
21 power.

22 MEMBER MARCH-LEUBA: Okay. Keep going.

23 MS. MCCLOSKEY: That concludes the NuScale
24 Chapter 15 presentation.

25 MEMBER CORRADINI: Questions by the

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1 members?

2 MEMBER RICCARDELLA: What is the relative
3 peaking then for the bundles that don't have the --
4 for the stuck rod assembly?

5 What's the relative peaking radially
6 primarily for those assemblies versus the average? I
7 think what you're presenting here is the average
8 power.

9 MS. MCCLOSKEY: We'd like a couple of
10 minutes to discuss, please.

11 MEMBER CORRADINI: You can also come back
12 -- we can also come back in closed session or later
13 on.

14 MEMBER RICCARDELLA: Yeah.

15 MEMBER CORRADINI: You don't have to sit
16 here and wait. Why don't you take time to look for
17 the response and we'll come back to it.

18 MS. MCCLOSKEY: Thank you.

19 MEMBER CORRADINI: Okay. We're moving
20 onto Chapter 20, is that correct? And then we'll take
21 a break as a reward.

22 MS. MCCLOSKEY: Okay. Thank you.

23 MEMBER BLEY: Are we taking a break?

24 MEMBER MARCH-LEUBA: Not officially I
25 don't think.

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1 MEMBER BLEY: Okay. I'm going to see if
2 we can get some coffee then.

3 MEMBER CORRADINI: If you guys are ready.
4 Go ahead.

5 MS. JOERGENSEN: Good afternoon. My name
6 is Nadja Joergensen. I'm a licensing engineering with
7 NuScale Power.

8 Next to me is Zack Rad, Director of
9 Regulatory Affairs. And with me I have Chris Maxwell
10 to do the majority of the presentation on Chapter 20,
11 mitigation strategies and beyond design basis external
12 events.

13 With that I'll turn it over to Chris.

14 MR. MAXWELL: Good afternoon. The new
15 Rule 10 CFR 51.55 separated into requirements for
16 mitigation strategies for design basis events, which
17 is covered by paragraph 501. And requirements for a
18 spent fuel pool level indication, which is covered by
19 paragraph Echo.

20 The initiating event is not defined by the
21 Rule, but the outcome is. And that is that the
22 results in an extended loss of AC power concurrent
23 with the loss of normal access to the normal heat sink
24 for a passive design such as NuScale's.

25 And the Rule requires that the three key

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1 safety functions of core cooling, containment, and the
2 spent fuel pool cooling will be maintained throughout
3 the ELAP.

4 To evaluate the NuScale power plant scope
5 and capability, it's necessary to establish first what
6 the minimum installed coping duration is. NuScale
7 considers a 14-day minimum coping period to be
8 sufficient to establish an alternate means of removing
9 heat.

10 And this duration is based on the
11 operating experience from Fukushima where at that
12 event, without a preplanned strategy or staged
13 equipment, or a hardened makeup line and with access
14 to the site significantly restricted by the earthquake
15 and tsunami, they were able to add water to the Unit
16 Four spent fuel pool after nine days using offsite
17 resources.

18 And were able to restore onsite capability
19 to inject water to the fuel pool using the fuel pool
20 cooling system at the 14-day point.

21 Beyond this minimum coping duration, the
22 continued use of installed plan equipment, as well as
23 ad hoc resources and equipment repairs, could be used
24 to continue coping indefinitely.

25 Now, I'd like to talk about the NuScale

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1 power plant's response to an ELAP current with the
2 loss of normal access to the normal heat sink, and the
3 status of the three key safety functions during that
4 event.

5 As we just mentioned, NuScale power plant
6 was informed by the events at Fukushima and sought to
7 provide a design that would cope during an ELAP
8 without the use of AC or DC electrical power, without
9 a need for inventory addition, or supplemental onsite
10 equipment, without the use of offsite resources, and
11 without a need for operator action. And therefore
12 without a need for operator monitoring during the
13 event.

14 And in short, the extended coping duration
15 is provided by the design through automatic response
16 of installed plant equipment alone. As I mentioned in
17 the subcommittee, the strategy provides a significant
18 advantage of allowing the operators to focus on
19 restoring site capabilities, including electrical
20 power rather than the deployment of resources,
21 equipment, and strategies, just to address the three
22 key safety functions.

23 So, now we'll look at the response of the
24 NuScale power plant to the extent of loss of AC power,
25 concurrent with the loss of the normal heat sink,

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1 without operator action and without the use of offsite
2 resources.

3 So, within one minute of the event
4 initiation, a reactor trip, decay heat removal system
5 initiation and containment isolation are received by
6 all modules. This makes the reactor subcritical and
7 places in service the decay heat removal necessary to
8 establish safe shutdown conditions.

9 This establishes the core cooling and the
10 containment functions. Twenty-four hours into the
11 event, the module protection system de-energizes the
12 trip solenoids for the ECCS valves, allowing those
13 valves to open and placing the emergency core cooling
14 system into service, again, at that 24-hour point.

15 With this, each of the modules is in safe
16 shutdown with passive decay heat removal and
17 containment cooling, while the spent fuel in the spent
18 fuel pool continues to be passively cooled by the
19 inventory of the ultimate heat sink.

20 During this entire period, the ultimate
21 heat sink is receiving heat and is heating up. And
22 after more than five days, the ultimate heat sink will
23 begin to boil. At this point, the ultimate heat sink
24 level is lowering.

25 And again, with no operator action, so no

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1 operator actions will lower to 45 feet in the ultimate
2 heat sink after more than 50 days. Now 45 feet is a
3 level that's just below the top of the DHRS passive
4 condensers.

5 And during this entire period again, the
6 core cooling and containment and safety functions are
7 maintained. Forty-five feet, the significance of this
8 is, there's no significance regarding the ability to
9 remove core heat or to cool the containment. This is
10 just where the long-term cooling analysis stopped. It
11 was suspended at 45 feet.

12 MEMBER MARCH-LEUBA: And just for the
13 record, the ultimate heat sink during all this 50
14 days, is really the environment, because you're
15 venting steam out of the building. Is that correct?

16 MR. MAXWELL: That's correct.

17 MEMBER MARCH-LEUBA: Through a, can you
18 say in open session, how you vent it?

19 MR. MAXWELL: We have rupture, a safety-
20 related rupture disk in the reactor building.

21 MEMBER MARCH-LEUBA: Thank you.

22 MEMBER BLEY: It just starts boiling at
23 five days.

24 MR. MAXWELL: That's right.

25 MEMBER BLEY: So, how many days later

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1 before you actually pop this blow out test?

2 MR. MAXWELL: Oh, we didn't analyze that
3 for this piece of it. Instead we assumed that into
4 our passive cooling calculations we assumed that we
5 retained all of it because it's a worst case scenario
6 to retain that heat in the building, causing the rooms
7 adjacent, the module protection system rooms to heat
8 up.

9 MEMBER BLEY: To heat up. Okay.

10 MEMBER MARCH-LEUBA: And because in
11 reality, it's a very large building that is cold.

12 MEMBER BLEY: Yeah. It would take quite
13 a while.

14 MEMBER MARCH-LEUBA: Yeah. My concern is
15 not losing the steam. It's what would happen to all
16 the instrumentation and all the electricity that is
17 not flowing through there?

18 MEMBER BLEY: Right.

19 MR. MAXWELL: And again, the DC equipment
20 is in rooms that are outside of the pool area.
21 They're in rooms that are adjacent to the pool area,
22 but not in communication with the pool area
23 themselves.

24 MEMBER MARCH-LEUBA: Are they isolated?
25 I mean, they're -- I mean, I really want condensing

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1 humidity on all my electric systems.

2 MR. MAXWELL: Right. But the --

3 MEMBER MARCH-LEUBA: But since you lost
4 power, you don't really care that much.

5 MR. MAXWELL: Right. So again, the core
6 cooling and the containment functions are maintained
7 during this period.

8 But with respect to the spent fuel cooling
9 function during this, we said that heat continues to
10 be added to the ultimate heat sink, and the level
11 continues to lower.

12 And after about more than four months,
13 level will reach the bottom of the opening in the weir
14 wall between the spent fuel pool and the other pools
15 of the ultimate heat sink.

16 And that point's significant, because
17 before this point the reactor pool and the refuel pool
18 are acting as a makeup source to this spent fuel pool
19 as it's evaporating. And the, again, up until this
20 point they act as a single volume.

21 Beyond this, now only the spent fuel pool
22 volume remains to cool the spent fuel in the spent
23 fuel pool. And it requires still another month, or
24 more than five months total for a level to lower to
25 the spent fuel -- to the top of the spent fuel in the

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1 spent fuel pool. Again, assuming no operator --

2 MEMBER KIRCHNER: Excuse me, for the
3 record Chris though, is that sent fuel in the pool at
4 the maximum capacity? Or for this calculation?

5 MR. MAXWELL: This calculation goes beyond
6 our actual maximum capacity.

7 MEMBER KIRCHNER: Okay. Right.

8 MR. MAXWELL: That's 18 years' worth of
9 spent fuel. This assumes that the reactors were at
10 102 percent power, all of them when the event
11 initiated.

12 None of the inventory that evaporates off
13 returns to the pool. All very conservative conditions
14 when calculating this pool boil off.

15 MEMBER MARCH-LEUBA: And just a reminder
16 that 18 years of spent fuel is according to nine cores
17 of spent fuel per core. So you have nine times more
18 fuel in the spent fuel then you do in the cores.

19 And those are real old. But still, this
20 spent fuel pool will produce more heat --

21 MR. MAXWELL: That's correct.

22 MEMBER MARCH-LEUBA: Then the fuel cores
23 themselves, right?

24 MR. MAXWELL: That's absolutely correct.

25 And the pool boil off reveals that. And that's why it

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1 was suggested that the other two pools of the ultimate
2 heat sink act as the makeup source. Because it is the
3 spent fuel pool that's driving the reduction in level.

4 MEMBER MARCH-LEUBA: Okay.

5 MR. MAXWELL: The significant driver. And
6 the next slide, please?

7 During the subcommittee there was some
8 discussion about a shutdown margin. And we wanted to
9 present this information.

10 I want to point out that this data doesn't
11 come from a -- it's a preliminary calculation. And it
12 doesn't come from a docketed analysis.

13 But what this is showing us here on the Y
14 axis is we have boron concentration. And on the X
15 axis is a time to fuel cycle.

16 So the blue line represents the nominal
17 boron concentration during operation over the cycle.
18 And I'll go straight to the green line, the lower
19 line, which is criticality at a temperature of 420
20 degrees, or safe shutdown.

21 So you see the margin from between the
22 blue line and the green line to reach criticality.
23 And the red line represents 4-0 degrees, or 40 degrees
24 to reach criticality.

25 So you can see at the beginning of the

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1 cycle you have to dilute more than 600 ppm to reach
2 criticality.

3 MEMBER MARCH-LEUBA: And you didn't do a
4 very good job explaining what that is. Is time to
5 deliver.

6 MR. MAXWELL: Okay. Y axis, boron
7 concentration. X axis time in the cycle. And my blue
8 line is my normal operating boron concentration.

9 MEMBER MARCH-LEUBA: That's an AF15 of
10 1.0?

11 MR. MAXWELL: That's correct.

12 MEMBER MARCH-LEUBA: Okay. So what's the
13 green line?

14 MR. MAXWELL: The green line is with all
15 rods in, and cooled down to 420 degrees.

16 MEMBER MARCH-LEUBA: Okay. So the
17 difference between green and blue is the role off and
18 whatever the temperature is. And that will depend on
19 whether it is the beginning or end in the cycle.

20 MR. MAXWELL: That's correct.

21 MEMBER MARCH-LEUBA: What is this?

22 MR. MAXWELL: You see at the beginning of
23 the --

24 MEMBER MARCH-LEUBA: Oh, is this time of
25 cycle?

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1 MR. MAXWELL: It is. That's our X axis.
2 And again, the difference there, the red line is,
3 instead of 420 degrees or safe shutdown, the red line
4 is 4-0 degrees, 40 degrees in RCS picture.

5 MEMBER MARCH-LEUBA: And this is with all
6 rods up, or all rods in.

7 MR. MAXWELL: Yes, all rods in. That's
8 correct. Which is one of the boundary assumptions in
9 ELAP space.

10 MEMBER BLEY: Your blue line to the right
11 side of the curve is that's essentially the end of
12 cycle?

13 MR. MAXWELL: That's correct. Although we
14 -- the X axis is from beginning of cycle to all the
15 way to the end of cycle.

16 And you can see that the other two lines
17 stop because the boron concentration essentially would
18 become, you know, give you a negative boron
19 concentration. That's why those lines terminate.
20 Okay, next slide.

21 The next portion of the Rule is paragraph
22 Echo, which address spent fuel pool level, or
23 correction, spent fuel pool monitoring.

24 The requirement is to provide reliable,
25 remote indication for five years after the last fuel

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1 that's in the spent fuel pool was used in an operating
2 reactor.

3 The NuScale ultimate heat sink system
4 includes four remote level indications for the
5 following, for the reactor pool, the refueling pool,
6 and two for the spent fuel pool.

7 These instruments are seismically
8 qualified and environmentally qualified for the
9 conditions in the pool area. And they're normally
10 powered by our highly reliable DC power system via the
11 plant protection system.

12 They also include a replaceable battery
13 power source, which is independent of the plant AC and
14 DC power systems. And that replaceable battery source
15 has a known capacity of 14 days.

16 MEMBER MARCH-LEUBA: And we discussed
17 this, this -- the readout is accessible during a
18 severe accident, correct?

19 MR. MAXWELL: That's correct.

20 MEMBER MARCH-LEUBA: We still don't know
21 where they are, but they're not in the pool.

22 MR. MAXWELL: Yes, sir. They're just --
23 that's the one thing we know for certain, is they
24 aren't in the pool area. They're remote from that
25 location.

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1 MEMBER PETTI: As well as the battery.

2 MR. MAXWELL: That's correct. Both the
3 battery and the indication, correct.

4 MEMBER MARCH-LEUBA: Oops, forgot that
5 one.

6 MR. MAXWELL: So, in summary again, the
7 two parts of the Rule, the first part is the
8 mitigation strategy.

9 NuScale's mitigation strategy is to rely
10 on the automatic response of permanently installed
11 safety-related plant equipment to establish and
12 maintain those three key safety functions of core
13 cooling, containment, and spent fuel pool cooling.

14 And to provide an extended coping
15 capability superior in the 14 days. Again, just to
16 point out that the -- question?

17 CHAIR REMPE: Well, I want to finish what
18 you were saying. And then --

19 MR. MAXWELL: That just a reminder that
20 the mitigation strategy itself doesn't rely on AC or
21 DC electrical power inventory addition or require any
22 operator action to establish and maintain those safety
23 functions.

24 And regarding paragraph Echo and our spent
25 fuel pool level indication strategy, we're relying

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1 upon our installed instrumentation with a 14-day
2 batter back up power supply.

3 CHAIR REMPE: So, I'm thinking back on the
4 days from when we were dealing with the Near Term Task
5 Force. And Steve was around then too. The intent I
6 thought, the stuff I've been reading has reminded me
7 of the fact that the intent of the rule was to try and
8 give the plants an option to deal with something they
9 didn't expect.

10 And because there's uncertainties. And so
11 we've got this stylized event. And you seem to have
12 a lot of confidence, and we didn't try to explore, you
13 know, where's the big uncertainties in some of the
14 things that you've presented for this scenario.

15 But, I think at the beginning of your
16 discussion you were saying you were trying to satisfy
17 the intent of the rule. And I'm just kind of
18 wondering, and again, I seem to harp a lot about
19 instrumentation and operators not having data if we're
20 taking away some flexibility to the operator.

21 So, I mean, you say yeah, this stuff will
22 be there. But they're not supposed to do anything.
23 But again, how much confidence do we have after 72
24 hours, until the 14 days that this instrumentation
25 will really be there?

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1 Because that's really, you meet the
2 requirements of the Rule, but have we met the intent,
3 of the Rule which hasn't been published by the way,
4 you know?

5 MR. MAXWELL: I would step it back a step.
6 And say that what we were looking to do, and we're
7 talking about FLEX and the strategies that went into
8 that.

9 Was that what we saw was that that was the
10 response to the operating experience at Fukushima. So
11 we said, we don't want to be in a situation where we
12 require this FLEX equipment.

13 And I don't mean to be flippant about
14 this, but rather than a Band-Aid, we want the design
15 to address the conditions. We went through the use of
16 safety-related equipment, and only safety related
17 equipment, to be able to respond to the conditions
18 that are postulated by this event.

19 So, again, not knowing the initiating
20 events, but knowing the results of that that must be
21 addressed, that's what the design did. It provided a,
22 without an operator action, without a requirement for
23 operator response, the passive ability to shut down
24 the reactor.

25 To establish and maintain core cooling.

1 To cool the containment and to cool the spent fuel and
2 spent fuel pool for an extended period of time,
3 extended coping duration.

4 I'm very sensitive to the other piece of
5 it, what the operators will want to do as well.

6 CHAIR REMPE: Right. You said you were an
7 operator.

8 MR. MAXWELL: Absolutely. And so I have
9 a high level of confidence in the beyond the 72 hours.
10 We've talked about 72 hours.

11 But we have redundant, 100 percent
12 redundant, highly reliable DC power system, which is
13 another 72 hours on top of it just from our installed
14 highly reliable DC power system.

15 We've got redundant back up diesel
16 generators that can supply all of the charges for that
17 system. We've got an auxiliary AC power source.

18 The instrumentation, I can't speak into
19 too much detail about it, but I can say that the
20 Chapter 15 events are more bounding then the ELAP
21 event as far as the conditions that the
22 instrumentation experiences.

23 And that they're required to be qualified
24 for a far greater period then 14 days. And beyond.
25 So, with that, I have a high level of confidence that

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1 the operators will have information available to them.

2 But what they -- what I want to emphasize
3 is that it's just a check that the key safety
4 functions are continuing to be maintained while they
5 address the initiating event and establish normal site
6 function and capability.

7 That's what's really the important part of
8 this strategy to me, is that while those safety
9 functions are maintained, it's getting back to normal.
10 And that's what this allows.

11 MEMBER RICCARDELLA: Is there any concern
12 about this return to power scenario, under the ELAP
13 event?

14 MR. MAXWELL: We have our -- let me ask
15 Meghan to respond to that question. And NuScale,
16 Meghan, can you answer that question for me?

17 MS. MCCLOSKEY: Sure. I think one of the
18 key things is to consider in terms of return to power,
19 under the ELAP conditions, our operating in ECCF
20 cooling mode, which as we discussed earlier today,
21 will tend to concentrate boron in the core and riser
22 region.

23 Which will be, with the function of all
24 control rods inserted, will likely prevent any return
25 to power scenario.

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1 MEMBER KIRCHNER: They don't assume a step
2 though.

3 MEMBER RICCARDELLA: I understand that.
4 Let me --

5 MR. MAXWELL: Let me -- can I just answer
6 in a licensing perspective?

7 MEMBER RICCARDELLA: Yes.

8 MR. MAXWELL: So, our PDC-27, principal
9 design criteria, and the related exemption note that
10 with all rods in, we maintain the reactor subcritical,
11 period.

12 So, --

13 MEMBER RICCARDELLA: Regardless of the
14 boron con -- or regardless of where you are in the
15 cycle? And --

16 MR. MAXWELL: That's right.

17 MEMBER RICCARDELLA: And the boron
18 concentration?

19 MR. MAXWELL: That's correct. So the
20 boron concentration is a reflection of where you are
21 in the cycle. And that's why we showed you that graph
22 to demonstrate how much margin we have.

23 So the postulated scenarios of boron
24 dilution or concentration and relocation, which are a
25 little more complicated than simply distilling. But

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1 the margin to that is exceptional.

2 So, it -- we demonstrated at 40 degrees,
3 well that's not even the postulated scenario, right?
4 We'll postulate that the spent -- the ultimate heat
5 sink heats up.

6 And so while we don't have a specific
7 analysis for this, this scenario under ELAP, we wanted
8 to demonstrate what we have based on our Chapter 14
9 postulations.

10 MEMBER RICCARDELLA: As long as all rods
11 are in, there's no condition that gives you a return
12 to power.

13 MR. MAXWELL: That's correct.

14 MEMBER DIMITRIJEVIC: Can you go to the --
15 back to slice four so we can discuss your 14 days and
16 the 72?

17 Because then that's where I have a couple
18 of questions. One is what means ad hoc resources?
19 What do you mean by ad hoc resources?

20 MR. MAXWELL: That we can procure without
21 prearranged contracts.

22 MEMBER DIMITRIJEVIC: So basically how you
23 designed that, you don't want to rely on some outside
24 source of agreement. You want to be completely
25 independent of the rest of the world, so.

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1 MR. MAXWELL: That's right.

2 MEMBER DIMITRIJEVIC: All right. And so
3 my question now, can you arm the water to your pool
4 from fire truck?

5 MR. MAXWELL: We have an externalized --
6 the short answer is yes. But the way we do it is we
7 have an assured makeup line to the spent fuel pool,
8 seismically qualified with an external connection on
9 the reactor building, with a standard fire protection
10 thread.

11 MEMBER DIMITRIJEVIC: Okay. And let's say
12 that you don't get any power. And this with the
13 diesel generator which you mentioned a couple of
14 times, how they cool? Your -- those back up diesel
15 generators?

16 MR. MAXWELL: There's skid mounted diesel
17 generators with their own cooling water and I believe
18 they're -- they haven't been selected yet, but I
19 believe they're actually air cooled.

20 MEMBER BLEY: I was going to say, that's
21 perfect.

22 MEMBER DIMITRIJEVIC: All right. But my
23 other question is, so you don't really get any power
24 at, you know, after 14 days back.

25 And maybe there is another calculation.

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1 Maybe you splash some water somewhere. Maybe you get
2 some leaks somewhere.

3 And then you have to worry about -- your
4 main thing here is this pool.

5 MR. MAXWELL: Right.

6 MEMBER DIMITRIJEVIC: This is the thing
7 that keeps all of this together. So, if you don't get
8 any power, don't anything change. So, how would you
9 makeup to the pool?

10 MR. MAXWELL: Well, one of the principal
11 assertions is that our pool is seismically qualified
12 as well. So that -- within the bounds of this event,
13 we don't assume a pool leak.

14 However, we do have, we've got that
15 external connection. It's a gravity feed to the pool.
16 And though they're not credited, we have onsite
17 resources, fire protection, water.

18 We've got a pool surge tank that's a
19 million gallons, essentially of water, sitting in a
20 berm. And then there's the --

21 MEMBER DIMITRIJEVIC: So you will have
22 some like diesel driven fire pump or something?

23 MR. MAXWELL: We do have a diesel driven
24 fire pump that we don't credit. But we do have one
25 onsite.

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1 MEMBER DIMITRIJEVIC: And you can just
2 pump the water in?

3 MR. MAXWELL: Because it's gravity, it's
4 gravity feed. So, as long as we're able to deliver to
5 that -- to that assured makeup line, we can. Pumper
6 trucks or trucks were procured ad hoc with water. We
7 can add water to the pool that way as well.

8 MEMBER BLEY: I hadn't thought about it,
9 but you've talked about this gravity feed so often, on
10 the outside of the building, at what elevation is that
11 hook up?

12 MR. MAXWELL: At ground level.

13 MEMBER BLEY: Ground level. So you really
14 could drain from a truck with no assistance.

15 MR. MAXWELL: Yes, sir.

16 MEMBER BLEY: And then it's piped to the
17 spent fuel pool?

18 MR. MAXWELL: To the spent fuel pool. And
19 it's sized to ensure the worst pool boil off rate that
20 it has the capacity in excess of worst boil off rate.

21 MEMBER BLEY: That's gravity flow?

22 MR. MAXWELL: Correct.

23 MEMBER DIMITRIJEVIC: So how is this line
24 normally isolated?

25 MR. MAXWELL: It has a manual isolation

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1 valve. Again, external to the reactor.

2 MEMBER DIMITRIJEVIC: And so that was so
3 it can be manually opened?

4 MR. MAXWELL: Correct. Without accessing
5 the pool.

6 MEMBER DIMITRIJEVIC: And this would be
7 accessible, you know, even though you've got steam
8 coming out?

9 MR. MAXWELL: Right. That being external
10 assures that.

11 MEMBER DIMITRIJEVIC: All right.

12 MEMBER RICCARDELLA: You might not be the
13 right person to ask, but somebody will. When your
14 seismic PRA, at whatever point it's done, look at the
15 fragility of your pool building, and confirm that the
16 probability of a serious, a serious rupture of that is
17 very, very low.

18 MR. MAXWELL: Can you answer that
19 question?

20 MR. RAD: I think once you -- sorry. I
21 believe the answer to your question is yes. But I
22 don't have that right in front of me.

23 MEMBER RICCARDELLA: Okay.

24 MEMBER KIRCHNER: Chris, I was looking at
25 your diagrams that you used for the heat sink and the

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1 height of the weir. How did you come about
2 determining that height?

3 MR. MAXWELL: The height of the weir wall
4 itself?

5 MEMBER KIRCHNER: Yeah.

6 MR. MAXWELL: I believe that it's ten feet
7 above the top of the spent fuel rack, the weir wall
8 is. And that's the shielding requirement. Ten feet
9 of water above the top of the spent fuel pool rack at
10 all times.

11 But that's not an official answer. That's
12 just what I recall.

13 MEMBER KIRCHNER: No. I was just thinking,
14 you're going to be doing refueling 24/7. The height
15 of that makes the lift and then the transit and drop.

16 MR. MAXWELL: So we tried --

17 MEMBER KIRCHNER: And actually you have to
18 be underwater obviously when you're doing a fuel
19 element transfer, but.

20 MR. MAXWELL: And we calculated that as
21 well. And what we saw was that with the fuel bundle
22 in the weir, in the opening, so its highest point, --

23 MEMBER KIRCHNER: Right. But you still
24 have ten feet under normal conditions.

25 MR. MAXWELL: Right. And ten feet for the

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1 pool level to lower within ten feet of the top of that
2 bundle that's in the weir wall is about 74 days.

3 MEMBER KIRCHNER: Yeah. No, I was just
4 thinking minimalizing the amount of up and down on the
5 crane.

6 MEMBER PETTI: So, I have a different
7 question. It sounds like there's still some
8 conservatism is built into that cartoon graphic time
9 line.

10 What's your best estimate?

11 MR. MAXWELL: We only use the conservative
12 values.

13 MEMBER PETTI: You've got to really looked
14 at --

15 MEMBER CORRADINI: They have numbers. I
16 thought you were going to quote, I thought you were
17 going to quote 50/50/150?

18 MR. MAXWELL: Well he says that's
19 conservatism built into that. And that's true.

20 MEMBER CORRADINI: At 50/50/150?

21 MR. MAXWELL: Yes, sir.

22 MEMBER PETTI: But he talked about, you
23 know, more fuel then would ever be in there. Some of
24 the boiling.

25 And he had to do -- just all these

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1 consumptions, if you took all those out, --

2 MR. MAXWELL: Right.

3 MEMBER PETTI: You know, how much time do
4 you really -- do you think you have in a best
5 estimate?

6 MR. MAXWELL: And --

7 MEMBER PETTI: What's the sense of margin?

8 MR. MAXWELL: Because we looked at what
9 our installed coping duration needed to be, you have
10 a minimum duration of 14 days in.

11 With all that conservatism built in, we
12 saw that again, nothing changes. We don't lose core
13 cooling at 50 days. It's just the calculation stops
14 there at 45 days.

15 But we said with that margin there was no
16 reason to go further.

17 MEMBER KIRCHNER: Just Chris, out of
18 curiosity more than anything, when does the estimated
19 boiling of the heat sink start? How many days out?

20 MR. MAXWELL: It's about a little more
21 than five days.

22 MEMBER KIRCHNER: Just beyond five. I was
23 looking at the schematic and couldn't tell. Okay.

24 MR. MAXWELL: Okay.

25 MEMBER DIMITRIJEVIC: Based on this graph,

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1 you're okay for five months actually, before you even
2 start to worry about the spent fuel pool, right?

3 Or that you would never worry about
4 anything else?

5 MR. MAXWELL: Well, let me address that in
6 a couple of different stages. The first one is that
7 we have a tech spec requirement ultimate heat sink
8 level.

9 So our operators are still bound by tech
10 specs and working to maintain the ultimate heat sink
11 level within tech specs.

12 MEMBER DIMITRIJEVIC: But what's measuring
13 this? What's --

14 MR. MAXWELL: The spent fuel pool level
15 indication.

16 MEMBER DIMITRIJEVIC: And that will be
17 available during this?

18 MR. MAXWELL: That's correct.

19 MEMBER CORRADINI: And that's the new
20 battery there.

21 MEMBER BLEY: They have those replaceable
22 batteries every 14 days they can stop it now.

23 MR. MAXWELL: But the -- the two safety,
24 key safety functions of core cooling and ultimate --
25 and I'm sorry, core cooling and containment are

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1 assured to a level, ultimate heat sink level of 45
2 feet.

3 Beyond that, we're talking just about the
4 spent fuel pool key safety function. And it's, you
5 know, again well past four months, 150 days before we
6 reach the top of the spent fuel pool and the spent
7 fuel rack.

8 MEMBER DIMITRIJEVIC: Well, you're going
9 to lose the job as and the efficiency of this. But
10 you will still have ACCS cooling, right?

11 MR. MAXWELL: That's correct.

12 MEMBER DIMITRIJEVIC: So, basically --

13 MR. MAXWELL: We have unofficial
14 calculations, non-docket information that suggests air
15 cooling beyond 45 -- beyond 50 days is sufficient.

16 MEMBER DIMITRIJEVIC: Right.

17 MEMBER BLEY: And from what you've told
18 us, the level will be going down faster in the spent
19 fuel pool then it will out in the main pool.

20 MR. MAXWELL: That's correct.

21 MEMBER CORRADINI: Further questions?
22 Okay. Let's take a break. 3:20.

23 (Whereupon, the above-entitled matter went
24 off the record at 3:10 p.m. and resumed at 3:20 p.m.)

25 MR. TABATABAI: Goof afternoon, everyone.

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1 Thank you very much for your time.

2 We'll be presenting to you Chapter 20 and
3 Chapter 6 with this crew up here. We're going to
4 start with Chapter 20. This is the full Committee
5 presentation. We had the Subcommittee presentation
6 yesterday and, basically, you asked to come back with
7 to just talk about mitigating strategies and the
8 staff's approach to review NuScale's mitigating
9 strategies. For today, we will be talking about a
10 little bit of background information for members who
11 were not present yesterday and, also, after that,
12 we'll get right into the review strategies.

13 I don't want to spend too much time on
14 this slide, just in the interest of time. But the
15 purpose of this slide is to highlight for the members
16 that the review of Chapter 20 has been kind of a
17 dynamic review. We received Revision 2 last year in
18 October. But, since then, there have been quite a few
19 developments. The bottom line is that our SER is
20 based on Revision 2 of the DCA, and the findings that
21 we have in Chapter 20, SER, are limited to 72 hours
22 after initiation of a beyond-design-basis. We will
23 get into more details as Ryan will take over to talk
24 about the staff's regulatory framework and the staff's
25 review.

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1 But, first, Ryan?

2 MR. NOLAN: Okay, yes. Thanks, Omid.

3 So, the recently-approved Regulation 5155
4 for Mitigation Strategies for Beyond-Design-Basis
5 Events does not apply to design cert applicants.
6 However, NuScale is voluntarily seeking the NRC
7 approval to use some of the installed design features
8 for the mitigation strategies.

9 Go to the next slide.

10 Again, this is the rule text for your
11 reference.

12 MEMBER CORRADINI: Just to slow you down,
13 since I know you're going to once we get you on a
14 roll -- but there are other Design Certification
15 Applicants that have done this.

16 MR. NOLAN: That's correct. We have in
17 the past reviewed Design Cert Applicants and provided
18 finality on the installed equipment when performing
19 certain capabilities.

20 MEMBER CORRADINI: All right. Thank you.

21 CHAIR REMPE: Can I go further? I know
22 the APR1400 was one, but there have been others beyond
23 it? There's more than one?

24 MR. NOLAN: It's primarily the APR1400.

25 MEMBER CORRADINI: I think all the rest

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1 predated it.

2 MR. NOLAN: Exactly.

3 CHAIR REMPE: Right. So, there's just
4 one.

5 MR. NOLAN: So, when you look at the
6 orders, say for the AP1000, there were statements --
7 and I don't recall if it was in the orders or within
8 the SECY paper itself -- that made statements to the
9 effect that AP1000 for 72 hours for the onsite coping
10 capability is sufficient, and that the COL-holder, the
11 licensee is responsible basically for the offsite
12 resources and, then, development of the strategies
13 beyond 72 hours.

14 CHAIR REMPE: So, there's wasn't a Chapter
15 20 for the AP1000, but, somehow or another, it got
16 approved for 72 hours, if I ever get asked on this.
17 So, thank you. Okay.

18 MR. NOLAN: Yes.

19 MEMBER CORRADINI: I think the AP1000,
20 just to be clear on the record, is the AP1000 had
21 already been certified prior to Vogtle and --

22 MR. NOLAN: That's correct. So, Vogtle --

23 MEMBER CORRADINI: They were in between
24 the two operating --

25 MR. NOLAN: Well, Vogtle got an order.

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1 MEMBER CORRADINI: Correct.

2 MR. NOLAN: Yes.

3 CHAIR REMPE: It's really just one CDA
4 Applicant, and that's the APR1400. So, maybe you do
5 have a singular word in my draft letter. Thank you.

6 MR. NOLAN: Yes, oh, yes.

7 So, the staff's approach for the Chapter
8 20 review, there's a few considerations. The primary
9 consideration was that we maintain consistency with
10 the level of detail that was performed for the
11 operating reactors as well as the previous Design
12 Certification where we reviewed this.

13 So, for an operating reactor, where, say,
14 phase 1, coping with installed plant equipment was on
15 the order of magnitude of eight hours, you know, the
16 use of onsite, portable equipment up to 24 hours may
17 be a little bit beyond that. And so, that's where my
18 bullet here that talks about, you know, the staff
19 really focused on the most critical time-sensitive
20 actions. There wasn't an extensive review performed
21 really beyond 72 hours for the operating fleet.

22 And so, for NuScale, the staff, what we're
23 planning on doing is looking at the installed design
24 features, reviewing their capabilities and capacities,
25 as well as addressing any transient, credible

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1 transient phenomena, for the first 72 hours. So,
2 that's a review of the design aspects of the SSCs.
3 Any of the operation aspects of mitigating strategies
4 would be deferred to the COL.

5 And so, speaking of the COL, if there is
6 no credible transient phenomena, we do not expect the
7 COL would have to address any additional design
8 aspects beyond 72 hours. What we would expect to see
9 is the development of the strategies, identification
10 of any resources, and the level of detail of those
11 resources would be commensurate with the time
12 available.

13 And so, yesterday what I mentioned was, if
14 makeup to the spent fuel pool is needed, we would
15 expect to see identification of where the water is
16 coming from, but we wouldn't expect to see necessarily
17 preplanned contracts in place to acquire that, because
18 of the time available for the site to acquire that
19 resource.

20 MEMBER MARCH-LEUBA: You might have
21 already said that, but what I'm missing in this review
22 is a set of emergency operating guidelines. At the
23 COL, you will be reviewing the EOPs, the actual
24 procedures for that particular applicant. But we are
25 seeing how a reactor might behave if we decide not to

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1 do anything until it is completely passive, which has
2 nothing to do with reality. In reality, all these
3 transients are going to be significantly better.

4 I think we all would have been served much
5 better if we said, yes, we have a passive, but here's
6 what really is going to happen at here and my EPGs.
7 And I think that's what you're saying. Is that what
8 you're saying, that to go beyond the 72 hours, you
9 would really like to see the EOPs or, in our case,
10 RPGs?

11 MR. NOLAN: Yes, that's certainly part of
12 it, but any incredible transient phenomena is a large
13 piece of it as well. We're resolving a lot of those
14 issues as part of Chapter 15, but we're only reviewing
15 it out to 72 hours. I'm not saying, or I can't say,
16 whether it's an issue or not. It's not something that
17 we have looked at. And to do so, to go beyond 72
18 hours is not necessary to make a finding at this
19 point. Or it could be seen as an unnecessary
20 regulatory burden on a future COL. It could limit
21 their ability to come up with their own strategies.
22 On top of it, we don't know where the site is located.
23 But, as you get further out in time, you get more
24 uncertainties.

25 MEMBER BLEY: You've got help me with this

1 unnecessary burden. I don't buy that one.

2 (Laughter.)

3 MEMBER BLEY: I come in with a COL.
4 They've already done it for me. That's great. If
5 they haven't or they did and I want to change it, I
6 can still do that. COLs do that. They don't have all
7 that many, but the few we've had do that without any
8 extra burden.

9 CHAIR REMPE: Well, we went through this
10 a little bit yesterday, the fact that we explored with
11 the staff, have you identified some things in addition
12 you would need? Yes, they have the right to say, "I'd
13 like you to go further." I mean, they can say that.
14 But, on the other hand, the staff can say, if you want
15 that, I'd like to see X, Y, and Z and documentation;
16 I've got some RAIs to think about.

17 MEMBER BLEY: That has nothing to do with
18 the question I just asked.

19 CHAIR REMPE: Yes, it's a --

20 MEMBER BLEY: No.

21 CHAIR REMPE: -- burden that they will
22 have to assume, if they want to do that, is what I
23 would contend.

24 MEMBER BLEY: No. These guys will do it
25 or won't do it, however it turns out. But when a COL

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1 comes along, it's going to be less burden on the
2 breadth of all the COLs, assuming they sell lots of
3 these things, if it's already resolved. If it's not,
4 they would have to do the same thing. If it is
5 resolved and they want to do something different,
6 they're back kind of where they started from. You've
7 really got to explain why that's extra burden on them.

8 MR. NOLAN: I think we understand the
9 comment. Our position at this point is we have the
10 information we need to make a regulatory finding
11 that's needed --

12 MEMBER BLEY: I get that, but --

13 MR. NOLAN: -- and to go beyond that --

14 MEMBER BLEY: But I don't understand this
15 point that you've raised, and it's been raised
16 elsewhere, that it's additional burden on the COL
17 applicant.

18 MEMBER CORRADINI: Keep on going. I think
19 Dennis has expressed himself.

20 MR. NOLAN: I understand.

21 So, the last bullet here is on the
22 instrumentation piece. In that, NuScale's position is
23 that instrumentation is not needed to support the
24 strategies. And so, they do not rely on the
25 instrumentation, and to verify certain parameters is

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1 not a direct requirement in the rule, and that it's
2 only there as like an added benefit or additional
3 assurance.

4 MEMBER CORRADINI: Let me ask you the
5 question, let me ask you a different question. Is 14
6 days indefinite?

7 MR. NOLAN: So, it's an interesting
8 question.

9 MEMBER CORRADINI: Well, if I understand
10 the crux of NuScale's argument, it is they want to go
11 there for that reason.

12 MR. NOLAN: Yes.

13 MEMBER CORRADINI: And so, my question is
14 that versus all this other folderol.

15 MR. NOLAN: I think if we were to go down
16 that road, we would need additional engagement with
17 NuScale to better understand their position. Because
18 when you look at the rule --

19 MEMBER CORRADINI: I'm sorry to pick on
20 you, but you're the one up there. If that's the case,
21 they're willing to go through the extra engagement.
22 What I'm sensing from staff is you don't even want to
23 go there. I'm sorry.

24 MR. NOLAN: And I think that, personally,
25 I need clarity on what the position is. Is it just to

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1 acquire offsite resources? That's one thing. Or is
2 it to demonstrate sufficient site functional
3 capability? That's a very different thing. Are you
4 assuming that you are fixing and recommissioning
5 normal systems to bring the site function capability
6 back? Because that, I don't think it would be
7 reasonable to assume non-seismic systems could be
8 repaired in that short amount of time.

9 MEMBER CORRADINI: Okay, but what I guess
10 I'm hearing from you is there are things that your
11 guys are going to have to think about and ask and get
12 clarification, either by analysis or commitment, if
13 it's beyond the 72 hours. And at this point in time
14 -- I'm picking my words carefully -- you don't want to
15 claim that anything is indefinite until you check into
16 the questions you would ask and the answers you would
17 get?

18 MR. NOLAN: Yes. I don't believe that 14
19 days would be indefinite. Now can 14 days be used to
20 justify ad hoc use of offsite resources, that's
21 another question.

22 MEMBER CORRADINI: Okay. So, it's not so
23 much a matter of that it's more than three days? It's
24 a matter of whatever it is -- call it "X" -- how is
25 "X" arrived at and what happens after "X"?

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1 MR. NOLAN: Right. I mean, if you're
2 looking at Fukushima as the example, 14 days is
3 essentially where they've transitioned approximately
4 from phase 2 to phase 3. And one can argue they're
5 essentially still in phase 3 indefinitely.

6 MEMBER CORRADINI: One could argue that.

7 MR. NOLAN: Right. They haven't --

8 MEMBER CORRADINI: That's fair.

9 MR. NOLAN: Yes.

10 MEMBER CORRADINI: That's a fair comment.

11 CHAIR REMPE: I want to go back to what
12 Dennis said, just to make sure that I can find a way
13 through this. If one says that, if the staff had said
14 that it could place additional burden for a finding at
15 the design certification stage -- for example, they
16 may need to provide additional analysis testing or
17 mitigating measures -- is that a problem with you, Dr.
18 Bley? Because it could, for a design certification
19 finding --

20 MEMBER BLEY: I don't have a problem. So,
21 I'm not sure what the question means. Try me again.

22 CHAIR REMPE: Okay. So, if the staff
23 would say that a finding for performance beyond 72
24 hours could place unnecessary additional regulatory
25 burden at the design certification stage because they

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1 may require additional analysis testing or mitigating
2 measures, for, again, a finding at the DCA stage, I
3 think that's a true statement. It does give them more
4 burden for a finding at the DCA stage. It could if
5 they decide they need that.

6 MEMBER BLEY: I don't get that. I don't.
7 So, that doesn't convince me.

8 CHAIR REMPE: Why does that not make sense
9 to you? Because, to me, it makes sense.

10 MEMBER BLEY: Well, it's --

11 CHAIR REMPE: And they say something --

12 MEMBER BLEY: Why does it make sense to
13 you?

14 CHAIR REMPE: Because they could get a
15 finding, if they go with 72 hours, perhaps based on
16 the available information -- if they want to go beyond
17 72 hours, the staff may say, well, I've got a lot of
18 RAIs on what you're going to do.

19 MEMBER BLEY: But that's at this stage.
20 That's at the design cert stage.

21 CHAIR REMPE: Right.

22 MEMBER BLEY: Yes.

23 CHAIR REMPE: And so, this could be
24 additional --

25 MEMBER BLEY: Oh, it could be. It's

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1 additional burden for this applicant who said we want
2 to take on that burden.

3 CHAIR REMPE: Yes.

4 MEMBER BLEY: Now we don't manage the
5 staff. But if the argument is it will cost the NRC
6 more to evaluate this 14-day issue than it would not
7 to do it, I mean, that's true, but that's true for
8 what we're already doing for Chapter 20 which isn't
9 required at this stage.

10 CHAIR REMPE: They've never had to do
11 anything.

12 MEMBER BLEY: Well, we've taken on the
13 burden. The staff has taken on the burden of looking
14 at Chapter 20. They've stopped it where they want to
15 stop it. The Applicant has asked to go further. It's
16 going to cost the Applicant more.

17 CHAIR REMPE: It's going to take more
18 time.

19 MEMBER BLEY: It's going to take more
20 time.

21 CHAIR REMPE: And they've got all this
22 push to try to do this expeditiously.

23 MEMBER DIMITRIJEVIC: It's their choice.
24 It's their choice, Joy. That's where the difference.
25 It's Applicant's choice.

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1 CHAIR REMPE: Yes, I understand it's their
2 choice.

3 MEMBER BLEY: But what I didn't understand
4 is why this would make the COL more difficult.

5 CHAIR REMPE: Oh, I didn't --

6 MEMBER BLEY: I think it will make the COL
7 easier.

8 CHAIR REMPE: Oh, I definitely agree with
9 you on that, but I didn't know that the staff would be
10 saying that this is more burden for the COL applicant.
11 I thought it was the same as --

12 MEMBER BLEY: Go back and read the
13 transcript. It was said.

14 CHAIR REMPE: Well, what's in the Safety
15 Evaluation and what is in the SECY that we're not
16 supposed to know? I thought it was pretty vague. Or
17 it has not been publicly released. It just said it
18 could place additional regulatory burden. I don't
19 think they said "on the COL applicant," did they?

20 MR. TABATABAI: That's true. Actually,
21 the NuScale position is to resolve as much during the
22 DCA stage with the installed equipment, rather than
23 posing it to --

24 CHAIR REMPE: To reduce regulatory
25 uncertainty for the COL applicant. That's been pretty

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1 clear. And the staff response back saying that this
2 is more regulatory burden, I thought was basically
3 saying for the NuScale folks, not the COL applicant.
4 Did you say for the COL applicant? I don't think you
5 said that.

6 MR. TABATABAI: No, I think that this
7 comment was that you were talking about DCA stage.
8 And I think Dr. Bley was referring to the COL stage
9 burden. I think that that was the disconnect.

10 CHAIR REMPE: But, okay --

11 MEMBER BLEY: Which is what I thought I
12 heard Ryan say.

13 MEMBER DIMITRIJEVIC: Right, yes, he did.

14 CHAIR REMPE: Did you say that, Ryan?

15 MR. NOLAN: Yes. I mean, I probably
16 should have used a different word.

17 CHAIR REMPE: Does the SECY say that,
18 though we can't --

19 MR. NOLAN: No.

20 CHAIR REMPE: The SECY does not say that?

21 MR. NOLAN: I should have said
22 flexibility.

23 MEMBER MARCH-LEUBA: I don't want to read
24 it, but --

25 CHAIR REMPE: I'm right or he's right? Or

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1 Dennis is right?

2 MEMBER MARCH-LEUBA: You're right.

3 CHAIR REMPE: That's what I thought.
4 Thank you. Because, for other reasons, I just want to
5 make sure I've got it clear. Thank you.

6 MR. NOLAN: We're at a point now in the
7 review where we can make a finding; no additional
8 information is needed from NuScale. It's just
9 documenting in the Safety Evaluation. To go beyond
10 that would require certainly more resources on our
11 end, possibly more resources on you folks.

12 MEMBER CORRADINI: I know you're near
13 done, so I don't want to hold you back. You're
14 getting excited now.

15 (Laughter.)

16 MEMBER CORRADINI: But, on the other hand,
17 it's a cost-benefit question. If the benefit is you
18 would term that indefinite, they might be willing to
19 do the cost. If you're going to term it, well, it's
20 just four times more days, but it's not indefinite
21 because I don't like the term, it may not be worth the
22 cost. But you're not even willing to entertain --

23 MR. NOLAN: Well, yes, like I said, that
24 term "indefinite," I think we would have to look at
25 that closer and better understand what it actually

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1 means. Is it, do you mean just an alternate means to
2 remove heat, ad hoc use of offsite resources, or do
3 you actually mean mitigation strategies ends at 14
4 days?

5 MEMBER CORRADINI: So, I know -- I think
6 I saw the draft SECY, but I don't remember it anyway.
7 So, it doesn't matter. It seems to me, if it doesn't
8 address the concept of what's indefinite and what
9 terms, what is necessary to determine indefinite --

10 MR. NOLAN: It doesn't, no.

11 MEMBER CORRADINI: -- then there's no
12 point, because it's just a cost-benefit process
13 question that just sits out there.

14 MR. NOLAN: Yes, the SECY does not address
15 what "indefinite" means.

16 MEMBER BLEY: We haven't reviewed the
17 letter that NuScale has sent asking for the 14 days.

18 MEMBER CORRADINI: No, but, I mean, based
19 on the --

20 MEMBER BLEY: At least personally, I don't
21 have a clue what it actually says.

22 MEMBER CORRADINI: Correct. No. But,
23 based on the discussion we had yesterday, that's what
24 we were told was the key.

25 MEMBER BLEY: And if they wanted to argue

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1 that at that point they wouldn't need anything
2 special, they could have responded through normal
3 sources and --

4 CHAIR REMPE: But, Mike, I would contend
5 that the draft SECY has a lot of information regarding
6 that, beyond 72 hours, the staff is going to need
7 information that is site-specific to make that
8 determination. They have, in their own way, answered
9 the question you're raising: is 14 days enough or
10 not? Because they've raised -- perhaps the draft SECY
11 would point out several examples where site-specific
12 information is needed.

13 MEMBER BLEY: That doesn't seem to be in
14 dispute.

15 CHAIR REMPE: Yes.

16 MEMBER BLEY: The staff has said it would
17 need more information and NuScale has said, yes, we'd
18 have to provide more information.

19 CHAIR REMPE: Yes, but it may require
20 something that's site-specific, and NuScale can't do
21 that because they don't have a firm site. And so,
22 this thing about 14 days -- and I think that was what
23 Ryan was trying to say back to you -- is, was 14 days
24 enough or not? Well, it depends on the site.

25 MR. NOLAN: That's certainly part of it,

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1 yes.

2 MEMBER DIMITRIJEVIC: It may not be the
3 case.

4 MEMBER CORRADINI: He thanks you for that
5 reasoning.

6 (Laughter.)

7 CHAIR REMPE: Well, I spent a lot of time
8 last night thinking about it, okay?

9 MEMBER CORRADINI: I'm sure it's true,
10 but --

11 (Laughter.)

12 MEMBER CORRADINI: Go ahead.

13 MR. NOLAN: I'm done.

14 MEMBER CORRADINI: Oh, you're done? Oh,
15 okay.

16 MR. TABATABAI: Okay. Well, the last
17 slide for the Chapter 20 presentation, as you're
18 aware, we have issued a SECY paper that describes our
19 plan for competing the review. The SECY paper will be
20 publicly available on Friday, and NuScale plans to
21 submit Revision to the DCA in August of this year.
22 So, our plan is, basically, to review Revision 3 based
23 on SECY papers, guidance, and approach, and compare
24 the information, evaluate the information compared to
25 the requirements of the new rule 10 CFR 51.55.

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1 So, this actually concludes our Chapter 20
2 presentations. If there are any comments?

3 (No response.)

4 MR. TABATABAI: Okay. So, we are going to
5 go to Chapter 6.

6 MEMBER CORRADINI: So, the Chapter 20
7 person escapes.

8 MR. TABATABAI: Okay. Thank you. We're
9 going to start our Chapter 6 presentation. We don't
10 have any closed session, any slides for the closed
11 session. So, we are going to talk about Chapter 6
12 here.

13 During the June 18th Subcommittee for
14 Chapter 6 presentation, we presented Section 621 --
15 I'm sorry -- 6364 and 65. And after the Subcommittee
16 presentation, we were asked to focus our full
17 Committee presentation on, basically, exemption
18 requests in Chapter 6. So, this is, basically, the
19 topic of our full Committee presentation.

20 Section 6.3 will be discussed as part of
21 Chapter 15. ECCS valves will be discussed as part of
22 Chapter 3, and other topics that the ACRS members had
23 expressed interest in will be discussed by NuScale as
24 part of their presentations during the closed session.

25 And I think this morning when you

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1 discussed some of the issues, I think they referred to
2 the visits at Corvallis for that additional
3 information.

4 So, with that, I'm going to turn the
5 microphone over to Clint Ashley to talk about
6 exemptions related to 6.24.

7 MR. ASHLEY: Next slide, please.

8 I'll be presenting the staff's review of
9 Section 6.24, exemption requests related to
10 containment isolation requirements. There's four
11 requirements that NuScale is seeking an exemption
12 from. And these four are GDC-55, 56, 57, and a TMI-
13 related requirement at the bottom.

14 The provisions in GDC-55 require, in part,
15 that each line that is part of the reactor coolant
16 pressure boundary and penetrates primary reactor
17 containment shall be provided with isolation valves.
18 And GDC-55 specifies that these valves will be located
19 one inside containment and one outside containment.

20 The provisions in GDC-56 are nearly
21 identical to GDC-55, except that GDC-56 applies to
22 lines that penetrate containment and connect directly
23 to the containment atmosphere. For both of these
24 GDCs, redundant valve barriers required to account for
25 a single act of failure in the isolation provisions,

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1 and this is achieved by putting those isolation valves
2 in series.

3 NuScale's GDC-55 and 56 penetration lines
4 are designed with two automatic isolation valves, but
5 instead of one inside and one outside containment,
6 they're both located outside containment. And as a
7 result, the Applicant is seeking an exemption for
8 those GDC-55 and 56 elements. And it's just those
9 elements, and it's just a part of the requirement.

10 The provisions in GDC-57 require, in part,
11 that each line that penetrates primary containment is
12 neither part of the reactor coolant pressure boundary
13 or doesn't connect directly to the containment
14 atmosphere shall have at least one isolation valve.
15 And as specified in GDC-57, the valves shall be
16 outside containment.

17 The containment isolation function for
18 GDC-57 lines is generally accomplished by a closed
19 system inside containment and that isolation valve
20 outside containment. And that gives you your two
21 barriers.

22 In the NuScale design, there are two
23 independent decay heat removal systems. Each decay
24 heat removal system has a steam line and a condensate
25 return line, and the Applicant proposes to use that

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1 closed-loop DHRS system outside containment as an
2 alternative to the isolation valve requirement. And
3 as a result, they are seeking an exemption.

4 And the last item on this slide,
5 10 CFR 5034(f)(2)(xiv)(E) -- and hopefully, I won't
6 say that too many times; I'll just say TMI-related
7 item, and you'll know what I'm talking about.

8 (Laughter.)

9 MR. ASHLEY: But it requires containment
10 isolation systems that include automatic closing on a
11 high radiation signal for all systems that provide a
12 path to the "environs". That's the word they use in
13 the rule. I'd prefer to say "environment". So, I
14 might skip between "environs" and "environment".

15 But, in this case, NuScale design provides
16 signals to automatically close all systems that
17 provide a path to the environment without using a high
18 radiation signal. And a result, they're seeking an
19 exemption.

20 Next slide, please.

21 MEMBER RICCARDELLA: This business of the
22 two isolation valves outside containment versus one
23 outside and one inside, is that -- it's kind of
24 related to the break exclusion, the significance of
25 the break exclusion zone, isn't it?

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1 MR. ASHLEY: That is correct.

2 MEMBER RICCARDELLA: Because, at
3 Subcommittee, I asked you, had this been abandoned
4 before, and you said, yes, it has, but it's not that
5 common. And as I understand it -- and correct me if
6 I'm wrong -- if you assume a break in that exclusion
7 zone, if you have valves both inside and outside, then
8 you can still postulate one active failure. But if
9 they're both outside and you have a break, you can't
10 postulate one active failure. Okay?

11 MR. ASHLEY: Well, that's a good question.

12 MEMBER RICCARDELLA: Yes.

13 MR. ASHLEY: If I postulate the single
14 active failure is the one that's inside containment,
15 and I have a break in that piping between the
16 containment and the outside isolation, I still have
17 that same flow path.

18 MEMBER RICCARDELLA: Yes. Yes.

19 MR. ASHLEY: So, I do think in NuScale's
20 exemption request they do acknowledge that -- and this
21 is also consistent with the design-specific Review
22 Standard 624 guidance, which leverages some ANS
23 standards as well. They talk about, if you've got
24 severe conditions inside containment -- and the
25 NuScale design does have severe or harsh conditions --

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1 you can have both valves outside containment, but you
2 need to take into account, you want some high
3 confidence that that piping system -- I've heard some
4 people refer to it as "super pipe" -- but you're going
5 to apply Standard Review Plan 362 and Branch Technical
6 Position 3-4, which gives you additional confidence
7 that you can prevent a break from happening in that
8 passive pipe system.

9 MEMBER BLEY: Can you give me a one-minute
10 tutorial on this phrase "break exclusion zone"? Is
11 that an ASME term that says, if you do it this way, it
12 won't break there, or what is that?

13 MR. ASHLEY: Is Renee Li in the audience?
14 (Laughter.)

15 MR. ASHLEY: Maybe Renee could speak to
16 that better than I could.

17 MEMBER BLEY: Thank you.

18 MR. ASHLEY: She's a 362 reviewer and is
19 well-steeped in Branch Technical Position 3-4.

20 MEMBER BLEY: Or is it an NRC term? I'm
21 not even sure.

22 MS. LI: Yes, I'm Renee Li.

23 You are correct, the staff position, as
24 described in Branch Technical Position 3-4, the break
25 exclusion actually is defined from the inside

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1 containment isolation valve through the containment
2 penetration up to the outside containment penetration.
3 And this region, portion of piping, people refer to
4 "super pipe" because the design and the inspection
5 requirement is upgraded to preclude a pipe break in
6 this portion of piping.

7 So, for the NuScale design, as it expands
8 -- they have different configuration. There is no
9 inside isolation valve, but, rather, the two isolation
10 valves is built in one single valve body. And that
11 portion of the valve actually is welded to -- I think
12 the configuration is you have the containment vessel,
13 and then, the nozzle, and then, the safe-end, and
14 then, the valve.

15 So, the break exclusion as defined for the
16 NuScale configuration is different from what the staff
17 guideline in BTP 3-4. So, with that, we have asked a
18 very detailed question. It is that, for the NuScale
19 configuration, with the way that they define the break
20 exclusion, how you can demonstrate with the NuScale
21 design the probability of piping failure within the
22 break exclusion is still extremely low, such that it
23 satisfies the requirement. It says you don't have to
24 evaluate many effects if one can demonstrate the
25 failure is extremely low.

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1 So, in that, NuScale has demonstrated
2 their piping stress design, stress as well as fatigue
3 meet the intent of the BTP 3-4, such that the stress
4 is about 80 percent -- and I'm talking about the
5 stress -- 80 percent of the SME's event stress limit,
6 and the fatigue is .1, which is well below the SME1.
7 But most important is that, for all the welds that
8 they talk about within the break exclusion, they are
9 all subject to augmented ISI requirement such that
10 their volumetric examination is for every 10 years.

11 And there are other design upgrades, but
12 those are the two main points. So, with that, the
13 staff thought NuScale has demonstrated appropriately
14 that the failure would be extremely low.

15 MEMBER RICCARDELLA: So, it's super pipe?

16 MS. LI: Yes. Yes, it's different, but --

17 MEMBER RICCARDELLA: It's different?

18 MS. LI: I mean --

19 MEMBER RICCARDELLA: Extra-super pipe?

20 (Laughter.)

21 MS. LI: No, the configuration is
22 different.

23 MEMBER RICCARDELLA: Yes, I understand.

24 MS. LI: But the essential point of low
25 stress, low fatigue, augmented inspection, yes, it's

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1 the same.

2 MEMBER RICCARDELLA: Okay.

3 MS. LI: Okay?

4 MEMBER RICCARDELLA: Okay. Thank you.

5 MS. LI: You are welcome.

6 MR. ASHLEY: Thank you, Renee.

7 So, for GDC-55, 56, and 57, on this
8 summary slide, we found that the NuScale design
9 accomplishes the containment safety function by
10 providing a containment isolation capability
11 comparable to that required by the GDCs, and
12 therefore, the underlying purpose of the GDCs is met.
13 So, we found that in this case the underlying purpose
14 of the rule was being satisfied, which supported an
15 exemption request. And the staff recommends that the
16 Commission approve the exemption.

17 Next slide, please.

18 MEMBER BLEY: Let me ask you one question.
19 This should have been directed at NuScale, and I think
20 maybe we did during the Subcommittee, but I don't
21 remember for sure.

22 What is the main reason they don't want to
23 have any valves inside on these lines?

24 MR. ASHLEY: I'm sorry, were you directing
25 it to NuScale?

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1 MEMBER BLEY: No, I was directly it to
2 you.

3 MR. ASHLEY: Okay.

4 MEMBER BLEY: I already missed NuScale.
5 So, I thought maybe you could help me.

6 (Laughter.)

7 MEMBER RICCARDELLA: Well, we could ask if
8 anybody --

9 MEMBER BLEY: We could, but did they
10 justify it to you?

11 MR. ASHLEY: Well, I think the exemption
12 request outlined a scenario that made a lot of sense,
13 especially given the harsh environment, a thousand
14 pounds of pressure, really high temperatures, and
15 you've got a design of a single valve body, two
16 isolation valves. You've got hydraulic controls.
17 You've got a non-1E power system. So, all these
18 things, if you've looked at other ways to achieve
19 isolation, I think that harsh environment inside
20 containment poses a significant challenge.

21 MEMBER BLEY: I guess I'm not sure. When
22 is the environment that extreme inside containment?

23 MR. RAD: This Zac Rad, NuScale.

24 Harsh environment was certainly one of the
25 considerations, but if you think about how you would

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1 have to operate those containment valves, in order to
2 protect containment we have to put additional
3 penetrations in containment with actuation controls
4 and things like that. So, there's also --

5 MEMBER BLEY: Oh, no, I'm sorry. Yes,
6 okay. So, it's not only that, it's where do you put
7 them, too?

8 MR. RAD: That's correct.

9 MEMBER BLEY: Yes, yes.

10 MR. RAD: That's correct, yes. And
11 there's also --

12 MEMBER BLEY: I was thinking up in the top
13 you have some room, but maybe not.

14 MR. RAD: Right. Space considerations are
15 also a factor. So, there's quite a few considerations
16 that weren't part of the initial requirement
17 considerations.

18 MEMBER BLEY: Yes. Thanks, Zac.

19 MR. ASHLEY: So, this particular slide
20 summarizes the staff's findings for the Three-Mile-
21 Island-related requirement. And I'll speak to this a
22 little bit more.

23 The NuScale Design Certification
24 Application requests an exemption from this TMI-
25 related requirement as applied to the containment

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1 evacuation system, or the CES. That's the system that
2 ties directly to the environment during normal
3 service.

4 In the DCA, the Applicant describes that
5 the design meets the underlying purpose of the rule by
6 isolating the containment evacuation system using two
7 automatic containment isolation signals. And these
8 two signals are high pressure in containment and low
9 pressurizer level.

10 The Applicant explains that the NuScale
11 design differs from traditional large lightwater
12 reactor designs because the reactor core uncover and
13 resulting core damage cannot occur without reaching a
14 low pressurizer level containment isolation setpoint.
15 And therefore, an event similar to Three Mile Island
16 Unit 2 is precluded from the NuScale plant design.

17 The Applicant also states that the
18 pressurizer is located well above the level of the
19 reactor core and not connected to the reactor vessel
20 by piping. Any decrease in reactor vessel inventory
21 to the level of the core would result in complete
22 emptying of the pressurizer and operation of the level
23 containment isolation signal.

24 So, as such, they believe that they meet
25 the underlying purpose of the rule with these

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1 alternate containment isolation signal provisions, and
2 the staff found that acceptable. And we also
3 recommend that the Commission approve the exemption
4 request.

5 That concludes my discussion.

6 MR. TABATABAI: Any questions for Clint?

7 (No response.)

8 MR. TABATABAI: No? So, the next
9 presentation is by Anne-Marie Grady, and she will be
10 talking about the exemption requests in 6.25.

11 Anne-Marie?

12 MS. GRADY: For combustible gas control,
13 there is one exemption request. And the exemption
14 request is to 10 CFR 50.44(c)(2), which requires that
15 a plant accommodate hydrogen generation up to 100
16 percent fuel clad-coolant reaction while limiting
17 containment hydrogen to less than 10 percent and
18 maintain containment structural integrity and other
19 accident-mitigating features. That's the requirement.

20 NuScale has requested an exemption from
21 that to not provide a system that would control the
22 hydrogen concentration to less than 10 percent.
23 Pursuant to 10 CFR 50.12, the Commission is authorized
24 to grant exemptions, and the exemptions that NuScale
25 has requested, as long as there are special

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1 circumstances. And there are six different
2 circumstances, any one of which is sufficient.

3 NuScale has requested the exemption based
4 on two of the special circumstances. One, the
5 application of the regulation 10 CFR 50.44(c)(2) is
6 not necessary to achieve the underlying purpose of the
7 rule. And secondly, that the compliance would result
8 in undue hardship or other costs that are
9 significantly in excess of those incurred by others
10 similarly situated.

11 The staff review on 10 CFR 50.44(c), which
12 applies to advanced lightwater reactors, focused on
13 the hydrogen conditions in the containment during
14 postulated accident scenarios, demonstration of
15 adequate mixing in containment post-accident,
16 equipment survivability following postulated hydrogen
17 combustion. NuScale's analyses of these scenarios
18 demonstrated that the containment would survive a
19 bounding combustion event inside the containment at 72
20 hours.

21 Staff did confirmatory calculations and
22 confirmed that the calculation for limiting the
23 pressure pulse inside the containment resulting from
24 the hydrogen combustion -- actually, detonation -- at
25 72 hours was such a very short duration, the

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1 containment integrity would be maintained. And the
2 containment integrity is the requirement, is what
3 we're trying to protect in 10 CFR 50.44(c).

4 MEMBER BLEY: Anne-Marie, in looking at
5 these things, where does one assume the oxygen comes
6 from to it?

7 MS. GRADY: Well, as you know, there is
8 some oxygen to begin with, but very little --

9 MEMBER BLEY: Very little, yes.

10 MS. GRADY: -- because, of course, it's
11 being operated at a vacuum.

12 MEMBER BLEY: Yes.

13 MS. GRADY: If there's a break in the
14 containment, ECCS line break, for example, the
15 hydrogen would be released into the containment, but,
16 also, radiolysis would begin. And radiolysis
17 generates not only hydrogen, but also oxygen. And
18 NuScale did an analysis and showed how much hydrogen
19 will be generated, and they got up to, I think, about
20 49 percent of the hydrogen that would be generated
21 from the failed fuel.

22 MEMBER BLEY: Okay.

23 MS. GRADY: And coupled that with the
24 oxygen we produced from the radiolysis, it would just
25 barely make 5 percent oxygen in containment, which is

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1 just enough to support combustion. And then, they
2 postulate that that combustion could lead to flame
3 acceleration, and the flame acceleration could be
4 reflected, and the reflection could, therefore, lead
5 to DDT, which means deflagration -- which is burning
6 -- to denotation transition. So, they took a truly
7 conservative approach and --

8 MEMBER BLEY: With that little bit of
9 oxygen? I'm surprised.

10 MS. GRADY: -- assumed that they could
11 survive all of those things and come up with a very
12 high-pressure pulse in containment. Now the high-
13 pressure pulse in containment lasts for a very short
14 duration, and other branches in the agency have looked
15 at the pressure pulse and the duration and have
16 determined that that pressure pulse would not exceed
17 the strain on the containment vessel.

18 MEMBER BLEY: How short is that time
19 period?

20 MS. GRADY: Pardon?

21 MEMBER BLEY: How short, roughly, is that?

22 MS. GRADY: .445 milliseconds.

23 MEMBER BLEY: Milliseconds? Yes, that's
24 fine. Okay.

25 MEMBER RICCARDELLA: Half a millisecond.

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1 MEMBER CORRADINI: Doesn't even know it's
2 there.

3 MS. GRADY: I beg your pardon?

4 MEMBER CORRADINI: It wouldn't even know
5 it's there.

6 MS. GRADY: I don't know. But, based on
7 that, based on the fact that there is, to begin with,
8 there's the containment vessel that is oxygen-starved,
9 and even though there could be, if you wanted to
10 postulate a combustion event actually occurring at 72
11 hours, with all of the conservatism that would lead
12 that to detonation, the containment integrity is
13 maintained. And on that basis, we believe that
14 NuScale's analysis and position satisfies the
15 underlying principle of the rule, which is to maintain
16 containment integrity during a combustion event in
17 containment.

18 Okay. Do you have any questions on that?

19 (No response.)

20 MS. GRADY: All right. Then, there's
21 containment leak rate testing. And containment leak
22 rate testing also has an exemption request, and
23 actually has a request to not meet 10 CFR 50, Appendix
24 A, GDC-52, which is to provide in their design the
25 capability for containment integrated leak rate

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1 testing, which is often referred to as Type A testing,
2 at the containment design pressure.

3 They've also asked for an exemption for
4 10 CFR 50, Appendix J, which is the requirement to
5 perform the Type A testing. Now, typically, a COL
6 applicant would be asking for that exemption because
7 they're the ones who have to ask for that. But, since
8 NuScale is asking not to provide the means in their
9 design to perform the testing, they're also asking if
10 we would consider that they don't have to do the
11 testing. Since they're logically connected, we
12 certainly accepted the premise of their argument that
13 they could ask for them together.

14 NuScale makes their argument that it's not
15 necessary to meet the intention of the rule by
16 performing this Type A test. And they base it on many
17 factors. One of them, because the NuScale containment
18 design is an ASME Section 3, Class 1, pressure vessel;
19 because they have done an analysis for leak tightness
20 of the containment vessel, which we have reviewed;
21 that they have, unlike the operating fleet, 100
22 percent inspectability of the containment vessel, both
23 inside and outside the containment. And, of course,
24 they're required to inspect it for the ASME Code.

25 And they have also, in addition to that,

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1 in addition to doing their analysis to show that in
2 their belief the containment vessel at accident
3 pressure will not leak, the vessel itself at the
4 mechanical joints, at the bolted flanges, they have
5 proposed to provide a preservice design pressure test,
6 which is in addition to the ASME Code hydro test
7 that's performed on every ASME vessel.

8 And they have done that for a couple of
9 reasons. One, the ASME Code hydrostatic test is done
10 really to see if the vessel is strong and to see if
11 there are any flaws in the material in the vessel.
12 However, the mechanical joints, if they leak, the
13 vessel still passes the Code hydro. So, that test
14 wouldn't demonstrate an essentially leak-tight
15 containment vessel. So, they propose this other
16 preservice design pressure test, and the pressure test
17 that they have proposed would be in addition to the
18 Code hydro. It would be a hydrostatic test, not
19 pneumatic. It would be pressurized up to the design
20 pressure. It would be held for 30 minutes. And any
21 leakage in the containment vessel would be a failed
22 test, and therefore, the vessel would not go into
23 operation until that situation had been remedied.

24 The preserve design pressure test is going
25 to be ITAACed. And that, of course, means every

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1 containment vessel will have this one-time preservice
2 design pressure test.

3 MEMBER RICCARDELLA: But the bolted joint
4 is taken apart --

5 MS. GRADY: I'm sorry?

6 MEMBER RICCARDELLA: The bolted joint is
7 taken apart and put back together at every refueling,
8 right?

9 MS. GRADY: I couldn't hear you. I'm
10 sorry.

11 MEMBER RICCARDELLA: The bolted joint --

12 MS. GRADY: Yes.

13 MEMBER RICCARDELLA: -- in the
14 containment, isn't that taken apart and, then, put
15 back together at every refueling?

16 MS. GRADY: Absolutely. There are 21
17 bolted joints in the containment actually. You're
18 thinking of the main flange? Yes, it's opened and
19 reclosed after every refueling, but there are 20
20 others, and about half of them are also opened and
21 reclosed.

22 MEMBER BLEY: But you don't have to do
23 another hydro? Just one in the beginning?

24 MS. GRADY: Pardon?

25 MEMBER BLEY: I thought you said you only

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1 have to do the hydro test once on a containment when
2 it's brought in.

3 MS. GRADY: The ASME Code hydro test,
4 once.

5 MEMBER RICCARDELLA: Yes. No, but this
6 leak test --

7 MS. GRADY: The preservice design pressure
8 test, once, that's correct.

9 MEMBER BLEY: But they do a leak test of
10 some sort every time?

11 MS. GRADY: But I haven't gotten to the
12 other tests that are part of the Appendix J
13 requirements.

14 MEMBER RICCARDELLA: Okay.

15 MS. GRADY: On the mechanical joints,
16 NuScale is going to produce, provide the results of
17 the Type B tests, which are on the mechanical joints,
18 as they are required by Appendix J. They're going to
19 do that at every refueling.

20 MEMBER RICCARDELLA: Okay.

21 MS. GRADY: When they start to refuel,
22 they've got a Type B, which is a pneumatic test of the
23 design pressure. They're going to be do a Type B
24 test, an as-found test, which means they're going to
25 test the vessel before they take it apart. Then,

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1 they're going to take it apart, do their refueling,
2 put it back together again. And for any joints that
3 have been opened, mechanical joints, they will do as-
4 left testing as well. So, they will confirm that the
5 mechanical joints are tight.

6 They will do, as expected, Type C testing,
7 which is the leakage test on the piping penetrations,
8 which has the containment isolation valves in it.
9 They will do that, per the ASME -- per the Appendix J
10 requirements and at the periodicity that's required by
11 Appendix J.

12 So, no change in those local pressure
13 tests, those local pneumatic pressure tests at all.
14 They're just asking not to test the overall vessel --

15 MEMBER RICCARDELLA: Got it.

16 MS. GRADY: -- pneumatically.

17 MEMBER RICCARDELLA: But, they mean parts
18 of Appendix J, but not all of it?

19 MS. GRADY: That's correct. That's
20 correct.

21 Okay. So, as I said before, it's an ASME
22 Section 3, Class 1, pressure vessel. They performed
23 a leakage analysis. They have design specifications.
24 They have capability for 100 percent vessel inspection
25 and examination and testing. And all of these things

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1 provide assurance that the leakage integrity of the
2 containment is maintained.

3 NuScale analyzed the flange bolts to the
4 flange openings using ANSYS. And based on the seal
5 design and specification, they calculated flange
6 contact pressures at every one of these and
7 corresponding flange gaps, and based on the internal
8 accident pressure and temperatures.

9 Staff reviewed all of these calculations
10 during an audit. Based on the fact that the design
11 and the analysis and the Types B and C testing would
12 confirm the leak tightness of the containment vessel,
13 staff recommends that the exemption request did not
14 require Type A test, nor to require the design
15 capability for the Type A testing.

16 MEMBER KIRCHNER: Anne-Marie, you know,
17 once they are ready to go, whether it's a new module
18 or refueled module, then they're going to pull a
19 vacuum on the containment. What happens next if they
20 have water ingress?

21 MS. GRADY: I'm not exactly sure what your
22 question is, and I think maybe somebody --

23 MEMBER KIRCHNER: Well, it's an inverse
24 leak test. It's not necessarily a design loading
25 internally, but by pulling a vacuum on the containment

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1 when they're ready to operate, if they had a seal
2 problem, that would show it.

3 MS. GRADY: They do the as-found -- the
4 as-left testing before they would pull the vacuum.

5 MEMBER KIRCHNER: Yes, I understand that.
6 But when they pull a vacuum, if, for whatever reason,
7 some seal somewhere flexed or something, then they're
8 going to suck in water from the heat sink.

9 MEMBER RICCARDELLA: But that's only 15
10 psi.

11 MEMBER KIRCHNER: I know. I know that.
12 But it's --

13 MR. ASHLEY: So, the containment
14 evacuation system would be able to pick up that
15 inventory.

16 MEMBER RICCARDELLA: They should pick that
17 up.

18 MR. ASHLEY: And you don't necessary know
19 where that inventory came from. So, you assign to the
20 reactor coolant system, and you would have to meet
21 those leakage criteria, which are very tight. So, you
22 would enter into a tech-spec condition very quickly,
23 I would imagine.

24 MEMBER KIRCHNER: Right. Okay.

25 MEMBER RICCARDELLA: But that's at much

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1 lower pressure.

2 MEMBER KIRCHNER: No, I know that.

3 MEMBER RICCARDELLA: That's an Appendix J
4 test, right?

5 MEMBER KIRCHNER: I know that.

6 MR. ASHLEY: Well, certainly the vacuum
7 from a differential is -- I think their vacuum was 1
8 psi, roughly. But if there was a leak into
9 containment from any source, your containment
10 evacuation system leakage detection would pick that
11 up.

12 MEMBER CORRADINI: Keep on going.

13 MR. ASHLEY: Well, that actually concludes
14 our presentation. Are there questions?

15 MEMBER BLEY: I wanted to go back to
16 hydrogen.

17 MS. GRADY: Yes.

18 MEMBER BLEY: I think you told me that one
19 of the potential sources of oxygen is if you have a
20 line break.

21 MS. GRADY: No, I didn't.

22 MEMBER BLEY: I thought you said --

23 MS. GRADY: I said from radiolysis.

24 MEMBER BLEY: Well, you did say that, but
25 I thought you also said --

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1 MS. GRADY: Oh, you said hydrogen?

2 Absolutely.

3 MEMBER BLEY: No, oxygen getting in.

4 MS. GRADY: No, radiolysis is the source
5 of the --

6 MEMBER BLEY: Is the only source?

7 MS. GRADY: -- additional oxygen in the
8 containment.

9 MEMBER BLEY: Hydrogen gets through --

10 MS. GRADY: I'm sorry, there is a small
11 amount of hydrogen in the RCS, and if that gets out,
12 yes, that would be accounted for, too. But that's not
13 a significant amount --

14 MEMBER BLEY: No, where I was going,
15 because I thought you said you had to also think about
16 pipe breaks, it's a very rich mixture. I'd be
17 surprised if it burns in containment. But if it could
18 somehow leak out into the reactor building, that's a
19 different issue.

20 MS. GRADY: Out into the reactor?

21 MEMBER BLEY: Yes. Reactor building.

22 MS. GRADY: Oh --

23 MEMBER BLEY: But outside of containment.
24 If it gets outside of containment outside of the
25 building, I don't care, but if it gets out inside

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1 of --

2 MS. GRADY: While it's burning, are you
3 talking about or what?

4 MEMBER BLEY: No. Before it burns.

5 MS. GRADY: Well, it --

6 MEMBER BLEY: I don't think it's going to
7 burn. With 95 percent hydrogen, it would be pretty
8 lucky to get anything burning. But if it could
9 somehow get out -- we don't have any way it can get
10 out?

11 MS. GRADY: Are you thinking of the
12 scenario where NuScale evaluated the containment
13 bypass accident where there was, in fact, a CVCS line
14 break outside containment?

15 MEMBER BLEY: I didn't know they did that.
16 I didn't remember that.

17 So, did they assume all of the hydrogen
18 goes out into the reactor building?

19 MS. GRADY: They did a containment bypass
20 scenario. CVCS has line break outside containment.

21 MEMBER BLEY: Okay.

22 MS. GRADY: They did a MELCOR analysis,
23 and they did determine that, yes, hydrogen would leave
24 outside containment and would be under the bioshield.

25 MEMBER BLEY: Ah, and this was the thing

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1 that could disrupt the bioshield? All right.

2 MS. GRADY: Yes, that's correct.

3 MEMBER RICCARDELLA: So, they redesigned
4 the bioshield to address that. Remember?

5 MEMBER BLEY: I do remember that, but you
6 can't get enough of a burn out there to do anything
7 else.

8 MS. GRADY: Their analysis said that there
9 would be sufficient hydrogen and oxygen to support
10 combustion, but that --

11 MEMBER BLEY: Out there, I'm sure there
12 would be, yes.

13 MS. GRADY: But they didn't go beyond that
14 to characterize what that really meant. We looked at
15 that further and did determine that there would be
16 enough analysis to support combustion.

17 MEMBER BLEY: Okay.

18 MS. GRADY: But now there's a new design
19 for the bioshield, and we're addressing how that would
20 contain the combustion event and prevent it from going
21 into the greater reactor building. However, the new
22 design for the bioshield means it's not air-tight
23 anymore. There are fixed openings in the bioshield.

24 MEMBER BLEY: Okay. So, it could leak
25 hydrogen?

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1 MS. GRADY: Yes. Yes, it could. But it's
2 a big building and it's a small amount of hydrogen.
3 So, the concentration, I have not seen a number in the
4 reactor building, but I would suspect it's really low.

5 MEMBER BLEY: If it thoroughly mixed. If
6 it didn't --

7 MS. GRADY: Yes. Yes.

8 MEMBER BLEY: -- it would burn where it
9 comes out. Okay. Thank you.

10 MS. GRADY: You're welcome.

11 Any questions?

12 (No response.)

13 MR. TABATABAI: Okay. This concludes our
14 presentation for Chapters 20 and 6.

15 MEMBER CORRADINI: Now you're moving on to
16 Chapter 3?

17 MR. TABATABAI: Yes.

18 MEMBER CORRADINI: Are you ready?

19 MS. VERA: We're ready.

20 MEMBER CORRADINI: All right. Proceed.

21 MS. VERA: Good afternoon, everyone.

22 My name is Marieliz Vera. I'm the Project
23 Manager for Chapter 3 of the NuScale DC Application
24 review. Today, we're going to present Chapter 3,
25 design of structure, components, equipment, and

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1 systems.

2 And the first couple of slides is the
3 staff review team. And we have the list of the
4 sections that have no open items. And this slide
5 here, we have the summary of the open items in Chapter
6 3 without 3.92 that was already presented and 3.11
7 that has 18 open items. Nine of those open items have
8 been resolved up to date. And then, you can see the
9 summary of the open items that are still open.

10 MEMBER BLEY: The turbine missile barrier
11 analysis, you haven't reviewed that as yet, right?

12 MS. VERA: So, yes, we're going to present
13 a little bit of it.

14 MEMBER BLEY: So, you have reviewed it?

15 MS. VERA: We're still on the review.

16 MEMBER BLEY: Started to review it? Okay.
17 You haven't finished your review?

18 MS. VERA: Exactly.

19 MEMBER BLEY: Okay.

20 MS. VERA: So, we're going to focus our
21 presentation today in the turbine missile, turbine
22 missile barriers, TF3 testing, steam generator tube
23 density, wave isolation, thermal fatigue, and ECCS
24 valve design.

25 For the turbine missile, we're still

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1 reviewing the information NuScale submitted and we're
2 going to discuss with NuScale the information we
3 presented and the questions in a public meeting that
4 is scheduled for next week, July 16th.

5 MEMBER BLEY: Is that going to be webcast?

6 (Laughter.)

7 MS. VERA: No. You're welcome to call in.
8 It's a public meeting.

9 MEMBER BLEY: Right.

10 MS. VERA: So, we're going to start with,
11 now I'm going to go to John Honcharik.

12 MR. HONCHARIK: Yes. I was a reviewer for
13 turbine missiles and I'll just go over it real quick.
14 The regulatory basis, GDC-4 requires that central SSCs
15 be protected from missiles.

16 And right now, the safety-related and
17 risk-significant SSCs in the NuScale design are
18 located in the reactor building and the control
19 building. The turbine generator rotor shafts are in
20 unfavorable orientation with respect to the reactor
21 building and control building.

22 So, they are within the low-trajectory
23 hazard zone for turbine missiles. So, to meet the
24 requirements of GDC-4, NuScale proposed to use
25 barriers, basically, the walls of the existing

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1 structures. Next slide.

2 So, I'll be talking about the turbine
3 missile spectrum, the spectrum of missiles. And Sujit
4 here will talk about the actual missile barrier
5 analysis.

6 With that, we had one open item. And
7 basically, requested that the staff determine that the
8 NuScale had not used a full spectrum of the turbine
9 missiles, basically, the size, weight, and speed,
10 which included up that half-stage of the last rotor.
11 So, NuScale provided some additional information, just
12 before the Subcommittee meeting. At that time, we had
13 not reviewed it. So, currently -- next slide.

14 As Marieliz stated, we'll be discussing
15 some of these issues --

16 MEMBER SUNSERI: Hey, John. John, could
17 you hold on a second, we have three meetings going on
18 and I want -- I think we deserve to pay you respect
19 here. So, are we ready? Okay. Please continue.

20 MR. HONCHARIK: All right, thank you. So,
21 as Marieliz stated, we'll discuss some of these issues
22 with NuScale in a meeting next week. So, I just
23 wanted to give you some highlights of some of the
24 things that we wanted to discuss with them.

25 First item would be the destructive

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1 overspeed of 160 percent. Basically, what was your
2 basis for that, when other things, especially the EPRI
3 report, has different destructive overspeed? And
4 also, that destructive overspeed should be the
5 bounding speed that you use for all your analysis for
6 all the different types of missiles that you have.

7 MEMBER BLEY: Yes, the one thing I worry
8 about is, if it's better than assumed, so it goes to
9 even higher overspeeds before it comes apart --

10 MR. HONCHARIK: Right.

11 MEMBER BLEY: -- you get bigger --

12 MR. HONCHARIK: Right.

13 MEMBER BLEY: -- not bigger, but you get
14 faster missiles.

15 MR. HONCHARIK: Right. And that's why, I
16 think, I'm asking the question about the basis for the
17 160, 180 is probably more realistic, from the
18 materials standpoint and such. Okay. So, that's why
19 I wanted to discuss a little bit more about this,
20 because in the RAI responses, I didn't see enough
21 information in there about that.

22 MEMBER BLEY: At Subcommittee, we talked
23 about it. To date, I don't know if we've had any
24 monoblock rotors come apart anywhere.

25 MR. HONCHARIK: No, well, there have been

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1 monoblock rotors that were done previously, earlier.

2 MEMBER BLEY: That have --

3 MR. HONCHARIK: That have failed.

4 MEMBER BLEY: -- come apart? Okay.

5 MR. HONCHARIK: Right, due to inclusions
6 and such.

7 MEMBER BLEY: Okay.

8 MR. HONCHARIK: So, it's mainly sulfide
9 inclusions. And part of the issue here is that,
10 NuScale is using an off-the-shelf turbine, okay?
11 Therefore, there's no testing done and everything
12 else, okay?

13 I mean, yes, they're not going to buy
14 something that's -- but part of the, if we do the
15 turbine missile probably analysis, part of that is
16 also turbine rotor integrity, which includes material
17 chemistry, what you do with tramp elements, how you
18 process it, fracture toughness, and such. If you
19 don't have that, fracture toughness in particular, you
20 could fracture even a monoblock rotor.

21 MEMBER BLEY: I see.

22 MR. HONCHARIK: And the next issue that
23 we'd like to --

24 MEMBER BLEY: I guess, I wasn't necessarily
25 arguing you couldn't --

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1 MR. HONCHARIK: Right.

2 MEMBER BLEY: -- I was arguing, it might be
3 going a whole lot faster than we expect when it
4 actually comes apart.

5 MR. HONCHARIK: Yes, right. But I think,
6 typically, I think that probably would be a higher
7 range. I mean, it could go higher, but I'm not sure,
8 I mean, because, here, we're talking about 200 percent
9 of the -- that's pretty fast. I'm not sure exactly
10 even the design of the actual rotor either.

11 The next would be the need to discuss the
12 methodology that they used to determine the speed of
13 the half-rotor. Because, basically, we wanted to make
14 sure it included half of the -- with the blades,
15 because part of it would be that some of the speed
16 would come from that outer portion of the blades,
17 right?

18 So, we wanted to look at also to see what
19 the dimensions were, just to have some confirmation.
20 So, we want to discuss that with them a little bit
21 more. I know they'll be presenting later, but what
22 I'm talking about is basically what they're going to
23 be presenting later, so, we --

24 MEMBER BLEY: Are you looking --

25 MR. HONCHARIK: -- didn't want to discuss

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1 --

2 MEMBER BLEY: -- at an exchange of more
3 RAIs or you thinking you'll go audit some of the
4 actual calculations they've done?

5 MR. HONCHARIK: Well, right now, I think
6 the plan is, next week, we're going to have --

7 MEMBER BLEY: This public meeting.

8 MR. HONCHARIK: -- a meeting with them to
9 discuss these things.

10 MEMBER BLEY: Okay.

11 MR. HONCHARIK: From that meeting, I think
12 we'll discuss whether or not we do another audit or
13 ask more RAIs or such. So, I think that's more of, to
14 get this out. So, this is the first time, I think,
15 they are also hearing this, too. But we wanted to let
16 you, since I think ACRS requested that we talk about
17 responses, so this is what we found so far, up to
18 date.

19 And the next item that we want to discuss
20 with them is the -- they had a couple of reports in
21 there, what we looked at during the audit, and I think
22 what we're looking for is that they include some of
23 the analysis and information that we requested from
24 the audits and also, the RAI responses to be factored
25 into those reports.

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1 And then, also, have those be part of the
2 DCD, because that's basically part of their basis for
3 saying that the turbine missiles are prevented, based
4 on this missile barrier.

5 MEMBER BLEY: That is their basis.

6 MR. HONCHARIK: Right. Yes. This is their
7 basis and should be documented as such. So, so far,
8 that is what we have for the turbine missile spectrum.
9 If there's no question, I'll turn it over to Sujit,
10 who will talk about the turbine missile barrier
11 analysis.

12 MR. SAMADDAR: I'm Sujit Samaddar and I'm
13 speaking on behalf of the staff who's actually
14 performing the review, but feel free to ask me the
15 questions. I'll skip the first slide, because John
16 has already presented the regulatory requirements and
17 the SRP section that we are using as the guidance for
18 the barrier.

19 So, essentially, NuScale has characterized
20 the turbine missiles as A, B, and C, which have the
21 weight, the velocity, and equivalent damage and the
22 associated overspeed. So, these are the three types
23 they have actually looked at.

24 Then, the acceptance criteria for the
25 concrete barrier itself is pretty standard. We have

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1 looked at the back face scabbing, which is the
2 limiting criteria for the barrier. So, there should
3 nothing be spalling up from the back of the barrier.

4 And the acceptance methodology that we
5 have allowed, or we allow, is the empirical equations
6 based on test results, which is in our SRP. And also,
7 the use of the finite element analysis, with
8 validation.

9 MEMBER CORRADINI: Can the analysis --

10 MEMBER BLEY: That's what --

11 MEMBER CORRADINI: I'm sorry.

12 MEMBER BLEY: That's what they're doing, is
13 finite element analysis?

14 MR. SAMADDAR: That's what they've chosen.

15 MEMBER BLEY: Yes.

16 MR. SAMADDAR: Yes.

17 MEMBER CORRADINI: Can the finite element
18 analysis or empirical equations determine spalling off
19 the backside?

20 MR. SAMADDAR: Yes. They use penetration
21 as a criteria. And on that, they have additional
22 factors that they use to quantify that.

23 MEMBER CORRADINI: As I remember these from
24 long ago, they're usually penetration correlations.
25 But the penetration correlation then has some sort of

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1 empirical connection to back spalling?

2 MR. SAMADDAR: Correct.

3 MEMBER CORRADINI: Okay.

4 MR. SAMADDAR: They do it in two steps.

5 One is for perforation, can be checked --

6 MEMBER CORRADINI: Sure.

7 MR. SAMADDAR: -- based on penetration, and
8 so is the scabbing part.

9 MEMBER CORRADINI: Okay.

10 CHAIR REMPE: I hate to interrupt you, but
11 periodically, you put your papers on the microphone
12 and that causes the transcriber to lose his hearing.
13 So, thank you.

14 MR. SAMADDAR: Oh, sorry. Sorry.

15 MEMBER BLEY: Mike, as I recall, is that
16 those empirical equations, some of them came from
17 military projectile tests.

18 MR. SAMADDAR: Yes. And now, there are a
19 wide range of these types of empirical relationships.

20 MEMBER BLEY: Yes, they're part --

21 MR. SAMADDAR: It's not just the NDRC, but
22 there are plenty others.

23 MEMBER BLEY: Yes, okay.

24 MR. SAMADDAR: So, the NuScale has used
25 NDRC to start their evaluation and they have used, for

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1 limited cases, the Mann and Whitney relationship. And
2 both these empirical relationships have a limitation
3 based on the test data. And both assumes the missile
4 and the target are both hard, or rigid, that's the
5 term really to be used, not hard, but rigid.

6 The finite element method which NuScale
7 has chosen allows for a variation of the stiffness of
8 both the missile, the hardness of the missile, as well
9 as the target. Okay. But the criteria for acceptance
10 is still continuing to be the same, which is based on
11 the scabbing. Next slide, please.

12 MEMBER CORRADINI: Does the FEM analysis
13 actually predict the scabbing or is it a correlation
14 given the penetration?

15 MR. SAMADDAR: It's a correlation given the
16 penetration.

17 MEMBER CORRADINI: So, in other words, they
18 do a more mechanistic penetration, but then use a
19 correlation to determine if I have scabbing?

20 MR. SAMADDAR: They do a more realistic, I
21 would say, they use the finite element to find the
22 penetration and then, they use the penetration and
23 proceed to the scabbing.

24 MEMBER CORRADINI: Okay, thank you.

25 MR. SAMADDAR: So, the FEM procedure was

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1 not previously reviewed by the staff for a high-speed
2 turbine missile impact. So, this is the first time we
3 are trying to do that, so there are limitations
4 associated with that.

5 The staff was looking for benchmarking of
6 the computer results. So, they want a low end for
7 benchmarking, because we don't have the high end test
8 data. So, we want to see that the FEM produces
9 similar results as would have been from one of these
10 empirical results.

11 We also expect that the deformable missile
12 will give lesser penetration depth than hard missiles.
13 And larger the kinetic energy of the impact, the
14 greater will be the penetration depth. So, these are
15 the types of things we want to see, once they do the
16 validation between the FEM and the empirical methods.

17 MEMBER SKILLMAN: Sujit, how is the missile
18 orientation addressed? For instance, a portion of the
19 low pressure, last stage low pressure blading plus a
20 portion of that rotor entering perpendicularly, with
21 the blades going in first, will probably result in a
22 different outcome than if the rotor portion enters
23 first versus the blade. So, how is the geometry of
24 the missile addressed?

25 MR. SAMADDAR: Actually, in the empirical

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1 relationships, there is a factor put in if the missile
2 is very pointed to very blunt. So, we use a
3 coefficient to account for what you just said.

4 MEMBER SKILLMAN: Okay, thank you.

5 MR. SAMADDAR: Okay. So, this -- we are
6 still continuing with the review. And so, the next
7 step from the local part of the missile impact, we
8 look at the global assessment, of what it does
9 globally when a larger body or mass hits the target.
10 So, that is all captured in the global analysis.
11 Okay.

12 And the global assessment, the staff
13 review is inconclusive. So, right now, the analogy is
14 that since the barrier has been designed for
15 automobile impact, from tornado-born --

16 MEMBER SKILLMAN: Yes, this is a tornado
17 analysis.

18 MR. SAMADDAR: -- that logic is not quite
19 sitting right with us, because the mass that the
20 assumptions for the missile, currently for the turbine
21 blade, is about 3,000 pounds. And a normal automobile
22 engine should not weigh more than a few hundred
23 pounds. It's a big mismatch there.

24 So, that, we have to look at the analysis
25 really what they performed and we are looking into

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1 what they have given us to make sure that we can
2 convince ourselves that we are looking at equivalent
3 things.

4 MEMBER BLEY: I wanted to follow up on Mr.
5 Skillman's question. I don't know if you've been able
6 to look at their actual finite element analysis, but
7 do they just assume a head-on impact or do they
8 consider it could come from many different angles and
9 be of different sizes and sort of do a probabilistic
10 look at those different alternatives?

11 MR. SAMADDAR: No, to answer your question,
12 I have not looked at their --

13 MEMBER BLEY: Okay, you

14 MR. SAMADDAR: -- finite element analysis.

15 MEMBER BLEY: -- haven't seen that?

16 MR. SAMADDAR: But for them to compare
17 apples-to-apples, they will have to look at
18 perpendicular strike. Because that's how --

19 MEMBER BLEY: Say that again, I'm sorry?

20 MR. SAMADDAR: Perpendicular strike.

21 MEMBER BLEY: Okay, yes.

22 MR. SAMADDAR: Because that's how all the
23 test missiles have happened.

24 MEMBER BLEY: Okay.

25 MR. SAMADDAR: So, to compare with values

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1 from the testing, they will have to do a strike.

2 MEMBER BLEY: Okay, thank you.

3 MEMBER RICCARDELLA: Wasn't there an RAI
4 response that had a bunch of analytical curves
5 compared to data? I thought I saw something like
6 that.

7 MR. SAMADDAR: There were plots, there are
8 plots, I mean, that has been presented, but we have to
9 analyze it and we are still in the process --

10 MEMBER RICCARDELLA: Okay.

11 MR. SAMADDAR: -- it is very difficult for
12 me to openly --

13 MEMBER RICCARDELLA: Okay, sure.

14 MR. SAMADDAR: -- talk about it. The other
15 point to note is, for the global damage or for the
16 large piece, that the Item Number C Missile, okay, the
17 velocity is around, what?, 350 miles per hour, of the
18 missile impact, while tornado and hurricanes are about
19 90 to 200 miles an hour.

20 MEMBER BLEY: Yes, 200 is completely
21 inappropriate.

22 MR. SAMADDAR: So, these are the two major
23 differences in the global analysis part that the staff
24 has right now seen.

25 MEMBER BLEY: Okay. Will you just be

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1 reviewing their analysis or will you be doing some of
2 your own?

3 MR. SAMADDAR: We will, for reasonable
4 assurance, we will do something on our own, since we
5 will be looking at rational approaches, we will be
6 looking at things as engineers. If it does not hurt
7 our senses, we will most probably accept it, but we
8 will not look into numerical details of everything.

9 MEMBER KIRCHNER: It seems to me that
10 there's a lot of experience in doing this kind of
11 analysis and you could just go for, bound their
12 analysis by just looking at past work done on this
13 with appropriate -- an automobile is not an
14 appropriate surrogate. But there -- you see where I'm
15 going with this? I mean, there's been a lot of
16 analysis on turbine missiles.

17 MEMBER BLEY: Actually, from what I've
18 seen, it's been pretty simplistic analysis.

19 MEMBER KIRCHNER: Oh, simplistic.

20 MEMBER BLEY: There was -- I only know of
21 one analysis like this was ever done and it was a long
22 time ago.

23 MEMBER KIRCHNER: Okay.

24 MR. SAMADDAR: I think that was not a
25 question, but --

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1 MEMBER KIRCHNER: I was just --

2 MR. SAMADDAR: -- we agree with that
3 statement, there is quite a number of, amount of
4 literature available. And we are already looking into
5 those to bound our conclusions.

6 MEMBER KIRCHNER: There have been a lot of
7 things thrown at concrete walls, of all different
8 shapes and sizes, including armor.

9 MEMBER BLEY: That's true.

10 MR. SAMADDAR: Yes. Some of that data is
11 kind of restricted. Okay. So, that's a problem,
12 trying to bring it out into the open, some of them are
13 quite restricted. Okay. Carrying on.

14 So, in conclusion, the staff is continuing
15 with this review and we have a meeting with NuScale,
16 it's a public meeting, on July 16, where we convey our
17 findings that we have so far to them and hear their
18 responses on it. Okay. And we'll take it from that
19 point on.

20 So, that concludes my presentation. If
21 there are no further questions, I'll turn it back to
22 --

23 MR. WONG: My name is Yuken Wong.

24 MR. SAMADDAR: That was me again.

25 MR. WONG: Dr. Steve Hambric and I

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1 previously had presented to the full Committee Section
2 3.9.2, including the comprehensive vibration
3 assessment program, and we were request to report
4 feedback on the TF-3 demonstration modal testing, as
5 well as the steam generator tube regarding density
6 wave oscillation and thermal fatigue. Next slide.

7 The purpose of TF-3 is to obtain fully
8 induced vibration data for prototypical helical coil
9 steam generator for validation of fluid elastic
10 instability, vortex shedding, and turbulence
11 buffeting design analysis. And NuScale had previously
12 discussed about the objectives, in the earlier
13 session, so I'm not going to go over it again. Next
14 slide, please.

15 There are some figure here, show the
16 configuration of the helical coils and the support
17 SOLs, those tabs. And for the TF-3 demonstration
18 modal testing, there are five -- two columns will be
19 installed for TF-3. The two outer columns were
20 installed. And performing the modal testing on Column
21 12.

22 And from inspecting the tubes, we find not
23 all spans of the support configuration are equal.
24 That's because the supports are not spaced equally,
25 they are eight supports, they're not on the

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1 circumferential direction, they are not spaced
2 equally.

3 So, there are some longer spans and there
4 are some shorter spans. On the longer spans,
5 generally, half of them appear loose. And what we
6 call loose is, they're not fixed supports. And the
7 other half, the long span, they're generally tight,
8 they're fixed supports. So, half are loose and half
9 are tight. For the shorter spans, they are generally
10 tight, or what we call, have a fixed support
11 configuration. Next slide, please.

12 The TF-3 was originally intended to
13 replicate the thermal expansion of the tubes and
14 support systems, and a stiffer boundary condition at
15 the support. So, tighter boundary conditions will
16 lead to higher frequencies and lead to higher fluid
17 elastic stability also.

18 As we show on the figures, there are gaps
19 between the tubes and the supports and the tabs. And
20 the preload system cannot close those gaps between
21 tubes and the tabs and the supports.

22 So, what the preloading system can do is
23 successfully tighten the fit between the supports, the
24 vertical support column, but not press the tubes into
25 the supports.

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1 So, however, as I mentioned, it does
2 tighten the supports, so increase of tube frequencies
3 by less than ten percent. And TF-3 will now test the
4 looser bounding conditions, without simulating the
5 thermal expansion of the tubes.

6 MEMBER MARCH-LEUBA: And do I understand
7 correctly, the tubes will be -- there won't be any
8 flow through the tubes and they will be air filled?

9 MR. WONG: For TF-3, there will be primary
10 flow. However, there --

11 MEMBER MARCH-LEUBA: Secondary?

12 MR. WONG: -- are no secondary side flow.

13 MEMBER MARCH-LEUBA: And they'll be air
14 filled, there won't be any mass in there, will that
15 change the frequencies? I've heard air filled, water
16 filled, what is it going to be?

17 MR. WONG: Yes, it's empty.

18 MEMBER MARCH-LEUBA: Empty?

19 MR. WONG: It's empty.

20 MEMBER MARCH-LEUBA: So, then, the mass is
21 lower, right? Will that change the frequency?

22 MR. WONG: Yes, it will, I believe it will,
23 for the analysis, they will compensate.

24 MEMBER MARCH-LEUBA: Oh, so this will be
25 used to benchmark some analytical tool and then, you

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1 extrapolate results? Okay, got you.

2 MR. HAMBRIC: Yes, this is Steve Hambric.
3 So, yes, they're going to have to account for the fact
4 that they're missing the secondary coolant, in their
5 margins that they infer from the TF-3 test. But --
6 and that should be defined in their test plan when
7 it's finalized.

8 MEMBER MARCH-LEUBA: Okay, thank you.

9 MR. WONG: Yes, so --

10 MEMBER RICCARDELLA: Excuse me, would you
11 repeat what you said earlier about the looser tubes
12 lead to lower natural frequencies than if they were in
13 there tight? Is that what you're saying?

14 MR. WONG: That's correct.

15 MEMBER RICCARDELLA: And that's
16 conservative?

17 MR. WONG: And it will be conservative.
18 So, we're hoping, with the preloading system, it can
19 replicate the tube condition in the normal operating
20 condition, because the test is performed at room
21 temperature. So, in a reactor operation, it heat up,
22 to expand.

23 MEMBER RICCARDELLA: Yes.

24 MR. WONG: However, the preloading system
25 will not duplicate that thermal expansion of the

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1 tubes.

2 MEMBER RICCARDELLA: Yes.

3 MR. WONG: So, therefore, the TF-3 will not
4 be a bounding test. It will have a looser --

5 MEMBER RICCARDELLA: Because we expect the
6 excitation --

7 MR. WONG: -- support condition.

8 MEMBER RICCARDELLA: We expect the
9 excitation frequencies to be low relative to the
10 natural frequencies, is that what you're saying?

11 MR. WONG: For TF-3, it will -- the
12 measured frequency will be lower than --

13 MEMBER RICCARDELLA: Yes.

14 MR. WONG: -- what would be in --

15 MEMBER RICCARDELLA: Which separates it
16 further --

17 MR. WONG: -- normal operation.

18 MEMBER RICCARDELLA: -- from the expected
19 excitation frequency?

20 MR. HAMBRIC: No, no, they get closer.

21 MEMBER RICCARDELLA: Okay.

22 MR. WONG: Yes.

23 MR. HAMBRIC: So, it's a conservative test,
24 they're more likely to experience FEI because of the
25 loose supports.

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1 MEMBER RICCARDELLA: Understood, because
2 the excitation frequencies are even lower than
3 expected natural frequencies?

4 MR. HAMBRIC: Well, that's to be decided.
5 They're going to tell us where the excitation
6 frequencies actually occur when they run the test.

7 MEMBER RICCARDELLA: Okay.

8 MR. HAMBRIC: So, we don't know where they
9 are now, we have estimates based on historical
10 literature and experiments, but this is a new --

11 MEMBER RICCARDELLA: Yes.

12 MR. HAMBRIC: -- structure that hasn't been
13 tested before, so we don't know where the excitation
14 frequency is.

15 MEMBER RICCARDELLA: But if those
16 excitation frequencies turned out to be higher, then
17 the looser test would be non-conservative, right?

18 MR. HAMBRIC: No, no, no, then they'll find
19 fluid elastic instability, right? They'll --

20 MEMBER RICCARDELLA: Yes, okay.

21 MR. HAMBRIC: Yes, they'll experience it
22 and they'll tell us at what speed that happens.

23 MEMBER RICCARDELLA: Okay. All right,
24 thank you.

25 MR. WONG: Okay. Next slide, please. So,

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1 now, NuScale will analytically demonstrate the tube
2 and support conditions. They presented thermal
3 expansion calculations. Thermal expansion will push
4 the tubes downward and radially out onto the tube
5 supports.

6 During the audit, the staff asked how
7 other forces will affect the tube to support contact?
8 Such as the primary flow and gravity will also push
9 the tubes down. Secondary flow will push the tubes
10 radially out, due to the Coriolis effect. NuScale
11 will quantify these effects of gravity and flow and
12 the staff will review the thermal expansion and
13 gravity and flow effects. Next slide.

14 During the audit, we also examined the
15 steam generator tube, due to vortex shedding. In this
16 figure, it shows a shot of the feedwater transition
17 tubing.

18 Only the lower tubes are susceptible to
19 vortex shedding, because there's no downstream
20 structure to disrupt the formation of the vortex. So,
21 the feedwater transition tubing is at the bottom. And
22 those tubing are bent in two planes. So, they are
23 stiffer and so, the frequency is higher.

24 And the measured frequencies are much
25 higher than previously calculated. And NuScale will

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1 soon submit an updated steam generator tube vortex
2 shedding assessment, which should show increased
3 margin against vortex shedding.

4 MEMBER KIRCHNER: Yuken, are you referring
5 now to what looks like Span E on this diagram, or are
6 you talking about the bottom of the bundle, if I'm
7 looking at this correctly, which is Span D?

8 MR. WONG: Near the top of this picture,
9 you see where the cursor shows?

10 MEMBER KIRCHNER: Yes.

11 MR. WONG: Those are the feedwater
12 transition --

13 MEMBER KIRCHNER: Yes, so those would be
14 stiffer, yes.

15 MR. WONG: Yes, stiffer, because they are
16 bent more --

17 MEMBER KIRCHNER: Sure.

18 MR. WONG: -- than the remaining of the
19 tubing.

20 MEMBER KIRCHNER: And they're closer to the
21 tube sheathe.

22 MR. WONG: So, the measured frequency is a
23 lot higher than previously --

24 MEMBER KIRCHNER: Sure.

25 MR. WONG: -- estimated.

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1 MEMBER MARCH-LEUBA: And when you say the
2 frequencies that are measured are much higher than
3 calculated, you're talking a factor of a thousand or
4 a factor of ten percent? I mean, how much higher?

5 MR. HAMBRIC: I'd say on the order of a
6 factor of four or five.

7 MEMBER MARCH-LEUBA: Okay.

8 MR. HAMBRIC: It's substantial.

9 MEMBER MARCH-LEUBA: Substantial? Okay.

10 MR. HAMBRIC: Yes. Yes, they were very
11 conservative with their initial estimates.

12 MEMBER MARCH-LEUBA: And does it put the
13 frequencies in resonance with something that it's not
14 supposed to? Or is it not high enough?

15 MR. WONG: Previously, it was about 20
16 percent margin. So, right now, it should improve --

17 MEMBER MARCH-LEUBA: Better?

18 MR. WONG: -- by those factors.

19 MEMBER MARCH-LEUBA: Oh, so, it went from
20 a 20 percent margin to a 400 percent margin, is that
21 what you're saying?

22 MR. WONG: No, this --

23 MEMBER MARCH-LEUBA: I wouldn't understand
24 the figure, let's keep going.

25 MR. HAMBRIC: Yes, they're going to submit

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1 the new margins shortly and they will be much
2 improved, that's all we can really say now.

3 MEMBER MARCH-LEUBA: Okay, thank you.

4 MEMBER CORRADINI: So, we're going on to
5 thermal fatigue.

6 MEMBER RICCARDELLA: So, again, we reviewed
7 this last meeting as part of our review of 3.9.1, I
8 think it was, and -- whatever, 3.9.something. And one
9 of our comments in our letter was, we recommended that
10 -- you said the staff was considering requiring an
11 ITAAC for this and we recommended that we do that.
12 But then, I heard this morning that they're working on
13 perhaps something different than an ITAAC to address
14 this.

15 MR. WONG: Correct. NuScale's position is
16 that ITAAC is required for every module. And since
17 this TF-3 testing is only a one-time test --

18 MEMBER RICCARDELLA: Okay.

19 MR. WONG: -- maybe there's some other
20 vehicle that's more suited to this need, such as on
21 the statement in Tier 1 and Tier 2 of the SEA.

22 MEMBER RICCARDELLA: Okay.

23 MEMBER CORRADINI: Oh, I see. Otherwise,
24 it would be for everybody.

25 MR. WONG: Correct.

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1 MEMBER CORRADINI: Okay, I see your point.

2 MR. WONG: Yes.

3 MEMBER CORRADINI: Okay, thank you.

4 MEMBER RICCARDELLA: Okay, thank you.

5 MEMBER CORRADINI: Thank you.

6 MR. WONG: Okay. Next slide. I'm going to
7 talk about the density wave oscillation. The steam
8 generator inlet flow restricters will limit the
9 density wave oscillation during normal operation, by
10 limiting the flow to within plus/minus ten percent.
11 The TF-1 and TF-2 test --

12 MEMBER MARCH-LEUBA: Wait, you said they
13 were limited to plus/minus ten percent, guaranteed?
14 Can you repeat what you said, I was sleeping?

15 MR. WONG: According to the DCD, Chapter
16 5.4, the flow restricters will limit the flow to
17 within plus/minus ten percent.

18 MEMBER MARCH-LEUBA: The flow oscillation
19 will not be more than ten percent and that's a
20 commitment?

21 MR. WONG: The flow through the steam
22 generator tubes.

23 MEMBER MARCH-LEUBA: But is that a
24 commitment in the DCD? And so, somebody then will
25 measure it and make sure that happens? Because it

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1 didn't, I mean, when it was measured in TF-2, it was
2 not limiting to ten percent.

3 MR. WONG: Well, TF-2 did not have the
4 inlet flow restricters installed.

5 MEMBER CORRADINI: So, I think what he's
6 asking is, how are you going to verify, how is the
7 applicant or the staff with review, going to verify
8 that they meet what they just said they're going to
9 meet? Granted, they want it plus or minus ten
10 percent, but how do we know that it's going to be plus
11 or minus ten percent?

12 MR. WONG: For the inlet flow restricters
13 concept testing, they selected the design that
14 provides the adequate pressure drop to limit the
15 density wave oscillation. And the final design is
16 based on the selected design and there's some minor
17 differences between the final design and the selected
18 design. However, the differences are small and should
19 not affect the pressure drop.

20 MEMBER KIRCHNER: May I ask a question?
21 I'm trying to put two things together here. So, in
22 the previous section, you talked about how the helical
23 tube bundles are going to lock themselves in as you
24 bring the unit up to power.

25 One of the things that's happening is

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1 expansion, if I got this right, the thermal expansion
2 of the tube bundle will kind of lock it in place with
3 the tube support things. But as soon as you put a
4 cold water slug in, it's going to be the other
5 direction. Is it going to sit there and rattle and
6 vibrate?

7 If you have density wave oscillations,
8 what we talked about this morning is the concern about
9 thermal mechanical fatigue, but also, if it's changing
10 the flexure of the tubes against the support tabs,
11 then you're going to have vibration there associated
12 with the thermal cycling of the tube, or am I missing
13 something?

14 MR. WONG: From the TF-1 and TF-2 data,
15 because they did not install the inlet flow
16 restricters, those tests did show density wave
17 oscillation. However, the frequency of the density
18 wave oscillation is so low, compared to the steam
19 generator tube frequencies, and the density wave
20 oscillation --

21 MEMBER KIRCHNER: No, I understand that,
22 but when you couple the thermal effect and the fact
23 that you're cooling the tube and heating it, so to
24 speak, by changing the thermal conditions, then it's
25 going to change how well it sits in the support

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1 mechanism, in terms of the tabs.

2 So, if all things are going well, and it's
3 stable at full power, and the system heats up and it's
4 just going along, then as I understand it, the design
5 is such that the expansion of the tubes will kind of
6 help it lock into its support tabs.

7 But as soon as you start thermally
8 changing that, you're going to induce -- you're going
9 to loosen up its support, then you're going to have
10 periods of cooling and heating of the tube bank and
11 that's going to change how well it sits in the tabs.

12 MR. WONG: The feedwater will be preheated.

13 MEMBER KIRCHNER: Yes.

14 MR. WONG: And this test is conducted at
15 room temperature.

16 MEMBER KIRCHNER: Okay, what's the --

17 MR. WONG: It's still bounding.

18 MEMBER KIRCHNER: What's the inlet
19 feedwater temperature?

20 MR. WONG: I do not --

21 MEMBER RICCARDELLA: It's way above room
22 temperature.

23 MEMBER KIRCHNER: Oh, I know that.

24 MEMBER RICCARDELLA: so, it's not going to
25 loosen it that much.

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1 MEMBER KIRCHNER: No, no, you -- it's
2 heated up, it's running, you design it for whatever
3 the normal inlet feedwater temperature is, that's
4 several hundred degrees, I don't know what it is off
5 the top of my head --

6 MEMBER CORRADINI: Two hundred and fifty C.

7 MEMBER KIRCHNER: -- and the thing locks in
8 because the tubes expand.

9 MEMBER RICCARDELLA: Yes.

10 MEMBER KIRCHNER: But now, if you start
11 having oscillations, the tube will cool down and the
12 tab, the way it's locked into the tabs is going to
13 loosen up.

14 MEMBER RICCARDELLA: But I assume those
15 oscillations aren't going to take you down close to
16 room temperature, you would have --

17 MEMBER KIRCHNER: I didn't say that.

18 MEMBER RICCARDELLA: Well, but the locking
19 in occurs between room temperature and 400 degrees, if
20 --

21 MEMBER KIRCHNER: I don't know that.

22 MEMBER RICCARDELLA: -- you come down to
23 350, you're still, I would assume, still pretty well
24 locked in, right?

25 MEMBER PETTI: It's in a limited number of

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1 tubes, the boiling --

2 MEMBER KIRCHNER: Yes.

3 MEMBER PETTI: -- right? So, it's not the
4 entire flange, it's just --

5 MEMBER KIRCHNER: No.

6 MEMBER PETTI: -- somewhere in the middle.

7 MEMBER CORRADINI: Your microphone is off.

8 MEMBER KIRCHNER: I'm just speculating, I
9 don't know what that differential thermal expansion
10 and how well the tube will then sit there in the
11 course of the density wave oscillation. Maybe it's
12 not enough --

13 MEMBER CORRADINI: Depends on how big the
14 delta T is, right?

15 MEMBER KIRCHNER: Yes. Yes, maybe it's not
16 a big enough delta T to make a difference --

17 MEMBER CORRADINI: The design delta T --

18 MEMBER KIRCHNER: -- as that boiling front
19 moves.

20 MEMBER CORRADINI: The design delta T from
21 inlet feed water to saturation is 80 C.

22 MEMBER KIRCHNER: Yes, so 100-something
23 degrees, 160, okay. I just --

24 MEMBER CORRADINI: You're sitting with an
25 inlet temperature about 250 C. So, you're going from

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1 250 C to saturation and then, you're coming back down.
2 But I think what, Walt I think, is trying to get at
3 is, he's asking a quantitative question of a
4 qualitative thing.

5 Sure, you're going to expand out, but
6 then, you're going to get to some sort of oscillatory
7 behavior somewhere in the region that is going to be
8 -- it's going to see an oscillation in temperature and
9 that oscillation in temperature is going to cause an
10 oscillation of the tube temperature, which is going to
11 cause it to essentially start shrinking and swelling,
12 shrinking and swelling. And the question is, can you
13 do this in perpetuity and no problem or not?

14 MEMBER RICCARDELLA: Well, I mean, fatigue
15 is a different question than whether you lose the
16 firmness of the support or not, right?

17 MEMBER CORRADINI: Right.

18 MEMBER RICCARDELLA: I mean, right?
19 They're two different questions.

20 MEMBER KIRCHNER: Two different questions,
21 if you lose the firmness of the support and the
22 ability for tube to rattle --

23 MEMBER RICCARDELLA: Yes.

24 MEMBER KIRCHNER: -- and vibrate and fret
25 is going to increase.

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1 MEMBER MARCH-LEUBA: But the solution that
2 we check is, NuScale has committed to is, we will not
3 have these oscillations, so we don't have to worry
4 about it.

5 MEMBER CORRADINI: Within some --

6 MEMBER RICCARDELLA: That's what the ten
7 percent means?

8 MEMBER MARCH-LEUBA: I was going to make
9 the second comment. I'm an expert on density wave
10 oscillations, that's how I made my career, there is no
11 such thing as a small density wave oscillation. If
12 it's unstable, it will grow until you hit reverse
13 flow, and that's the first non-linearity you hit on
14 that instability is reverse flow.

15 So, when you have a thermal hydraulic
16 density wave oscillation, you always have at least 100
17 percent oscillation, because it will grow until it
18 gets reverse flow. There is no such thing as a small
19 DWO, it just doesn't exist.

20 So, plus, another thing, when we reviewed
21 the stability topical, we were told that the TF-2 had
22 very significant inlet restricters. They have
23 modified it a little bit, but the inlet restriction in
24 TF-2 was fairly large. I cannot say the number, but
25 I call it infinity.

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1 MR. WONG: Yes, they had orifices for TF-2.

2 MEMBER MARCH-LEUBA: Well, I'll look at the
3 record, but they told us what the velocity losses were
4 and they were equivalent to what they're designing
5 now. So, this is an issue that has to be resolved.

6 And my concern, I'm going to just put it
7 here out in the open, is that NuScale is going to tell
8 you, we have a RELAP model that predicts we don't have
9 any oscillations, get out of here. And you'll say,
10 okay, they have a model, they have calculated it, and
11 it's going to be okay. And it's not going to be okay.

12 And it goes back, Mike, to what we were
13 talking about for the -- let me finish my comment.
14 You need to have the right people reviewing the right
15 things.

16 And this particular item is a cross-
17 discipline item, you have to have a thermal hydraulic
18 guy that knows density waves review what they're doing
19 and a mechanical vibration guy that tells you what the
20 consequences of that is. So, it's a cross-discipline,
21 you cannot do a Chapter 3 review and a Chapter 15
22 review independently.

23 MR. MAKAR: Thank you. This is Greg Makar
24 from Materials and Chemical Engineering Branch. I
25 just wanted to clarify that the information in the

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1 design certification on the limitation of the
2 oscillation, I think it's a limitation on the amount
3 of mass flow, on the mass flow rate, not on the
4 oscillation itself, not the density wave oscillation,
5 but the mass --

6 MEMBER RICCARDELLA: That ten percent
7 number --

8 MR. MAKAR: -- mass flow rate.

9 MEMBER RICCARDELLA: -- is just on mass
10 flow rate, yes.

11 MEMBER MARCH-LEUBA: So, you're predicting
12 the steady state mass flow that you expect to have?

13 MR. MAKAR: I'm not, but that's what the --

14 MEMBER MARCH-LEUBA: No, that's what --

15 MR. MAKAR: -- I think that's --

16 MEMBER MARCH-LEUBA: -- plus/minus ten
17 percent --

18 MR. MAKAR: -- the flow restricters design
19 is intended to limit the mass flow rate --

20 MEMBER MARCH-LEUBA: No, it's not.

21 MR. MAKAR: -- to within a --

22 MEMBER MARCH-LEUBA: The flow restricters
23 is there to prevent oscillations. I mean, otherwise,
24 you wouldn't pay for the pressure drop. It is to
25 prevent oscillations.

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1 MR. MAKAR: Agreed, but that's -- the
2 number in the DCD is not --

3 MEMBER MARCH-LEUBA: Okay.

4 MEMBER CORRADINI: But I think -- let's
5 just clarify. I think what you're saying is the total
6 feedwater flow, you're worried about the individual
7 tube behavior? Am I getting it correctly? Because
8 they have no way of knowing what the tube flow is.

9 MR. MAKAR: I don't have the DCD with me,
10 but that was -- I was just trying to --

11 MEMBER CORRADINI: Okay.

12 MR. MAKAR: -- explain what I --

13 MEMBER MARCH-LEUBA: That would make sense,
14 the error on my steady state flow that I predict is
15 plus/minus ten percent.

16 MEMBER CORRADINI: But I think -- I want to
17 make sure that the staff understands with Dr. March-
18 Leuba is worried about.

19 He's not worried about the total mass
20 flow, he's worried about the fact that, if I have the
21 density wave oscillation, I'm going to, in certain
22 channels, cause a reverse flow, and he's worried about
23 then -- that may only happen in one channel every once
24 in a while, and may parade around the thousand tubes,
25 but on the other hand, you're causing some sort of

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1 temperature swing, which would then cause a thermal
2 fatigue swing. A thermal fatigue, which you've got to
3 make sure you can survive. That's your point.

4 MEMBER MARCH-LEUBA: And if they're not
5 sufficiently clamped --

6 MEMBER CORRADINI: Yes, that part I'm not
7 --

8 (Simultaneous speaking.)

9 MEMBER CORRADINI: I guess, personally, I'm
10 not worried about that, I'm more worried about the
11 fact that they minimize what you're worried about.

12 MEMBER MARCH-LEUBA: I'm not worried about
13 it, I like to make sure that you can look at it and
14 say, yes, it does not detach when it cools down, which
15 it probably won't.

16 MEMBER CORRADINI: Keep on going.

17 MR. LUPOLD: This is Tim Lupold, from the
18 Mechanical Engineering Branch. One thing I'd like to
19 say, you're asking questions that the staff's not
20 going to be able to answer regarding --

21 MEMBER CORRADINI: We're not expecting a
22 answer, as much as just make sure you understand our
23 question.

24 MR. LUPOLD: We understand the question.

25 MEMBER CORRADINI: Okay.

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1 MR. LUPOLD: I'll also say that, when you
2 look at the steam generator, you do have to look
3 holistically on everything. And there are
4 requirements for inspection of the steam generators.
5 Yuken already reviewed, a month ago, what the
6 inspection requirements would be for the initial
7 inspections for the steam generators.

8 And then, there's ultrasonic -- well,
9 there's not. There are examinations of the tubes,
10 which can identify if there is a wear associated with
11 the tubes vibrating in their supports and things like
12 that. So, also consider that those things do exist.

13 MEMBER CORRADINI: Keep on going.

14 MR. WONG: Okay. Next slide, please.
15 Regarding steam generator tube thermal fatigue, the
16 steam generator tubes are ASME Section III, Class 1,
17 and fatigue analysis is required.

18 The staff has not reviewed a specific
19 steam generator tube fatigue analysis. The staff is
20 reviewing the steam generator feedwater plenum fatigue
21 analysis. NuScale has provided an ASME fatigue
22 screening report identifying most critical fatigue
23 locations. Those selected locations have been
24 reviewed, as the analysis there.

25 In the DCD ITAAC 2.1-4.2, we will confirm

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1 the ASME data reports have been prepared for ASME
2 Class 1 components listed in Table 2.1-2. This table
3 includes RCS components, including the steam
4 generator. And this is the end of my presentation.

5 MEMBER CORRADINI: Great, thank you.

6 MR. SCARBROUGH: Hello, this is Tom
7 Scarbrough. I'm going to give you an update on the
8 emergency pool cooling system valve review that we
9 have ongoing. In the open session, of course, we have
10 quite a bit more detail to talk about in a closed
11 session, but I'll give you a high level discussion.

12 There were three reactor vent valves that
13 are five-inch in size and two reactor recirculation
14 valves that are two-inch in size, and they allow that
15 natural circulation that we've been talking about.

16 Each RVV and RRV has a first-of-a-kind
17 design arrangement of a main valve, the inadvertent
18 actuation block, IAB valve, a solenoid trip valve, and
19 a solenoid reset valve, all connected by 20 feet of
20 tubing or so.

21 The NuScale is conducting an ECCS valve
22 design demonstration testing at Target Rock right now,
23 to satisfy 10 C.F.R. 50.43(e), which requires that for
24 new designs, like passive reactors, that they
25 demonstrate their safety features that's referenced in

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1 52.47(c)(2).

2 We have an ongoing audit of this whole
3 process, designed for the ECCS valves. We were onsite
4 the week of June 3 and watched some of their testing.
5 And we'll be going back next week to look at some of
6 the other testing that's going to take place.

7 Overall, with the ECCS valves, there's
8 several portions that should provide confidence in the
9 valves once they're fully designed and demonstrated by
10 50.43(e). First, you have an ASME Standard QME-1 2007
11 qualification requirement. We've accepted that in Reg
12 Guide 1.100 Rev 3.

13 There's ITAAC for equipment, qualification
14 for safety-related valve, and it points to the
15 qualification report, which is the QME-1 standard
16 qualification report.

17 They're also required to meet the code of
18 record that's specified in the ASME Code 2012 edition,
19 that's currently incorporated by reference in 10
20 C.F.R. 50.55a. Of course, the COL applicant is
21 required to use the most latest version, it's probably
22 going to be like an 18-month time period, but they
23 have to implement the most latest version of the OM
24 code that's incorporated by reference in 50.55a.

25 The other aspect is that NuScale submitted

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1 an alternative to 50.55a, to apply the Appendix 4
2 requirements for their ECCS valves. And these
3 include, the Appendix 4 requirements include stroke
4 time testing every quarter, but they can't do it every
5 quarter, so they'll be doing it every outage.

6 But also, initial and periodic diagnostic
7 performance assessment testing and it's up to a ten-
8 year interval, but they have to demonstrate
9 diagnostically that they can out to those longer
10 intervals.

11 But based on discussions with NuScale,
12 what they've planned is to bench test, like the IAB,
13 take it off, put it on a bench, make sure that it's
14 operating within the regime of the pressure release
15 and engagement pressures and such as that. But then,
16 you still need to finalize the design first, and
17 that's still ongoing with the 50.43(e) test.

18 So, our next steps. We have to complete
19 our audit, that's ongoing. We have an open item in
20 the SER regarding ECCS valve design. And we'll be
21 reviewing the final test report for all the testing
22 that's taking place right now. We need to review the
23 design specification updates.

24 We had an audit previously, they've
25 notified now that those specs have been updated and

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1 they're ready for us to review, so we'll be conducting
2 a follow-up audit to confirm that those changes took
3 place. And then, we'll update the SER as necessary to
4 come up to speed with Phase 4 of the SER.

5 So, that's where we are right now and we
6 can provide more detail during the closed session
7 about the ECCS valves.

8 MEMBER MARCH-LEUBA: From the reliability
9 point of view, Chapter 19 and the PRA analysis claims
10 very good performance for this first-of-a-kind valves.
11 Is there any plan for confirming that untested
12 performance?

13 MR. SCARBROUGH: Well, this testing that's
14 going on right now for 50.43(e) is a design
15 demonstration. So, there's not enough tests to come
16 up with a reliability value for that.

17 MEMBER CORRADINI: They're functionality
18 tests.

19 MR. SCARBROUGH: Yes, they're
20 functionality testing for 50.43(e), for design
21 certification. So, that gets them to that point.
22 Then, for the COL phase, they have to satisfy QME-1,
23 which is also a qualification.

24 But once again, they won't be doing enough
25 testing to demonstrate a specific reliability of the

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1 valves. And then, they're going to have OM code
2 testing and things of that nature. We're metering
3 that.

4 Usually, it takes several years of -- in
5 the NUREG reports that Idaho puts together, and such,
6 and research, it take years to come up with a value
7 that's reasonable for a valve. But to me, if you look
8 at all the numbers in those NUREGs, they're on the
9 range of 1E-3 per demand.

10 MEMBER MARCH-LEUBA: There are some that
11 have 14 percent failure probability.

12 MR. SCARBROUGH: Yes, it's --

13 MEMBER MARCH-LEUBA: Some types of valves.

14 MR. SCARBROUGH: Yes, it's a -- valves have
15 sort of a range right around that, little bit more,
16 little bit less. But that's --

17 MEMBER MARCH-LEUBA: You would expect a
18 more complex one to be in the lower range?

19 MR. SCARBROUGH: Yes, yes, right. And it's
20 going to -- once they finish all of this and then,
21 actually perform the qualification testing and have
22 some experience, I would expect it to be a typical
23 valve.

24 It is sort of a typical valve, but it's
25 the application that's so unusual, how they're

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1 actually applying it to this situation. So, I think
2 down the road, they will.

3 But I don't think, right now, we can say
4 there's a specific reliability, but other than that
5 fact that once they finish all this and they get years
6 and years of experience, it's probably going to be a
7 typical valve that we see, from that perspective.
8 Nothing spectacular and probably nothing terrible,
9 it's going to be somewhere in the typical range of 1E-
10 3 or something like that. That's what I would expect.
11 That's where they are.

12 MEMBER KIRCHNER: But qualify what you mean
13 by 1E-3, failure to function or does that include
14 opening inadvertently?

15 MR. SCARBROUGH: Typically --

16 MEMBER KIRCHNER: Does it mean half open?
17 Does it mean -- what does that mean? What are you
18 saying 1E-3?

19 MR. SCARBROUGH: Well, when I -- there's --
20 in the studies that are done on valve performance,
21 they look at failure to close, failure to open,
22 spurious opening, they look at all the various
23 factors.

24 And from a failure to open, failure to
25 close, it's whether or not it will perform its

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1 function right, without leakage. And that's what I
2 expect it to be able to do and that's what they'll
3 have to qualify for. So, it's going to be in that --

4 MEMBER KIRCHNER: Well, 1E-3 is an awfully
5 low number for a breach of the primary system.

6 MR. SCARBROUGH: Well --

7 MEMBER KIRCHNER: It's a really high number
8 in terms of frequency, is what I'm trying to say.

9 MR. SCARBROUGH: Yes, well, this valve is
10 -- once they have designed it and they install it,
11 it's -- other valves, like safety relief valves,
12 right?, they --

13 MEMBER KIRCHNER: I know.

14 MR. SCARBROUGH: -- also have that sort of
15 similar type 1E-10, like on pressurizer safety valves,
16 right? They're sort of the same range as well. And
17 so, you might have that, right?, you might still have
18 that same potential for that type of failure.

19 But it's consistent with the other type
20 valves that you have used for these safety valve
21 applications, things of that nature. So, it's not out
22 of the range of sort of typical for safety valves, as
23 such, in terms of the reliability.

24 So, I think that's what we'd expect them
25 to achieve, once they finish the design and do all

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1 this testing we're talking about and they have some
2 years of experience. I think that's -- that would be
3 the best we would look for for this.

4 It might be a little more reliable, but
5 it's probably going to be a typical valve. It's a
6 spring-loaded valve that works under differential
7 pressure, so it's going to be along that line. Its
8 unusual nature is its application, in terms of
9 (inaudible). But it's first-of-a-kind, so we'll have
10 to see where they go.

11 MEMBER DIMITRIJEVIC: So, because it's
12 first-of-a-kind, they didn't assume it's actually
13 usual valves, so that kind of get a little back to
14 reliability performance. I just want to talk to
15 Committee, there is two different failure modes.

16 One failure mode is fails on demand, and
17 the relief valve and the circulation valve will
18 perform differently in the assumptions of the PRA on
19 this thing.

20 And then, there is a failure to remain
21 open or failure to remain closed, or spurious
22 operation, which is the failure rate per hour. And
23 those are completely different, they usually 1E-7 or
24 1E-8 per hour.

25 So, they're dependent on exposure period,

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1 how long you have to remain closed or open. So, this
2 is what you call sort of like failure rate between the
3 test or something. And they will take forever, given
4 that we will not have hundreds of those valves to get
5 any meaningful data on those.

6 MR. SCARBROUGH: It'll take longer.

7 MEMBER DIMITRIJEVIC: Because they will not
8 be challenged so often, and there is -- I don't even
9 know what type of test is proposed to be done during
10 the refueling. So, yes.

11 MR. SCARBROUGH: And there's different
12 aspects. There's solenoid valves, which have sort of
13 history of their performance --

14 MEMBER DIMITRIJEVIC: Right.

15 MR. SCARBROUGH: -- and then, there's the
16 main valve, which is hydraulically closed, right?, I
17 don't want to get into too much detail, we're in open
18 session, but it is hydraulically closed and it's held
19 there by that. So, just very -- it's probably going
20 to very reliable in terms of not inadvertently
21 opening, right?, because you have a pressure holding
22 it.

23 And then, you have the IAB valve, which is
24 kind of this new valve, but typically, it's sitting
25 there, right?, it's sitting there ready to perform its

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1 function. So, it's typically just sort of hanging
2 out, right?, it's not doing much at that point, until
3 it's called up --

4 MEMBER DIMITRIJEVIC: There may not --

5 MR. SCARBROUGH: -- to operate.

6 MEMBER DIMITRIJEVIC: -- be some standby
7 failure mechanism --

8 MR. SCARBROUGH: Yes.

9 MEMBER DIMITRIJEVIC: -- we were discussing
10 moisture or some type of thing which may affect.

11 MR. SCARBROUGH: Right.

12 MEMBER DIMITRIJEVIC: Also, the PRA assumed
13 that this valve may not need a solenoid and may open
14 on some small delta P itself. They will not be able
15 to test that, so I mean, if this test is different
16 than what is needed to confirm reliable performance of
17 this valve, it just will confirm performance

18 MR. SCARBROUGH: Yes. And -- yes, this
19 type of test they're doing now is just to demonstrate
20 the design for the certification. Is the design
21 credible?

22 MEMBER DIMITRIJEVIC: Right.

23 MR. SCARBROUGH: Can this combination of
24 valves operate properly in this application? That's
25 what they're trying to show right now. They still

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1 have to go through all the qualification steps and
2 they're going to have to develop their in-service
3 testing program.

4 And they're going to operating these
5 valves. When they shut down for every refueling
6 outage, they will be opening these valves. And then,
7 when they go into an outage, they're going to have to
8 develop their in-service testing program to stroke
9 them and make sure they operate properly and do an
10 inspection. There's going to be a, these valves are
11 so important, there's going to be quite a bit of
12 activity regarding these valves very outage.

13 MEMBER BLEY: Just two points, for
14 everybody here. We have been promised, when we go
15 visit in a couple weeks, that they're going to walk us
16 through how they've combined the various failure rates
17 from existing valves to cover this complex of valves
18 that kind of go by single names. So, we look forward
19 to seeing that.

20 Although it was a completely -- in other
21 design certs, there have been cases where kind of a
22 valve that we've seen before, but expanded greatly in
23 size, where they did substantial numbers of tests,
24 thousands, to actually kind of demonstrate the failure
25 rate of these new valves.

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1 We're not seeing any push to do that here.
2 And I'm not sure if in that other design cert, there
3 was a push from the staff that led to it or if it was
4 just something the vendor decided to do on their own.

5 MR. SCARBROUGH: Yes, I think, if we go
6 back to the AP1000 and the squib valve discussion and
7 such, I think they were making some assumptions on
8 reliability of those particular valves, which may or
9 may not --

10 MEMBER BLEY: And there were some vacuum
11 breakers in ESBWR --

12 MR. SCARBROUGH: Yes.

13 MEMBER BLEY: -- that had extensive
14 testing, too.

15 MR. SCARBROUGH: Right.

16 MEMBER CORRADINI: I think Dennis is
17 remembering that as the example case.

18 MR. SCARBROUGH: Okay.

19 MEMBER BLEY: Yes, because they did lots of
20 tests on those.

21 MR. SCARBROUGH: And we're not asking to do
22 that, because I'm not looking for a specific
23 reliability number, I'm looking to see if they
24 satisfied 50.43(e), which is a demonstration of the
25 safety feature.

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1 And then, once they get past that, the COL
2 applicant is going to have to deal with all the QME-1
3 qualification and things of that nature. But I'm not
4 looking for a specific value for the reliability of
5 the valve.

6 MEMBER BLEY: I'm concerned about that,
7 because of these, the real significance of these
8 potentially to the risk of the plant. And the PRA
9 right now isn't showing much, because they're using
10 very strict failures rates for these valves.

11 MEMBER MARCH-LEUBA: Let me ask you a
12 question. In your mind, do you think it is somebody
13 else's job to look at the reliability and it's the PRA
14 guy's job, not mine?

15 MR. SCARBROUGH: Well, if --

16 MEMBER MARCH-LEUBA: That's what you said,
17 you're looking at, does it work?

18 MR. SCARBROUGH: Yes, right. That's my
19 side of it, right? In terms of the reliability
20 values, in terms of the PRA view of, if they come to
21 me and ask me what I think reliability is, I would
22 say, if you want a number, it's 1E-3 per domain. But
23 --

24 MEMBER MARCH-LEUBA: You gave me the answer
25 I wanted.

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1 MR. SCARBROUGH: -- they haven't come to
2 me. So, that's --

3 MEMBER MARCH-LEUBA: But -- yes. I take
4 yes for an answer. And the other thing is, this valve
5 is very complex and it has orifices and restrictions
6 and the flows have to be matched very accurately.
7 Indeed, we were told that the first time they built
8 one, they messed up and they had to go over it with a
9 file and make the orifice larger. So, there are
10 failure mechanisms or fouling of these orifices.

11 MEMBER CORRADINI: If we're getting into
12 closed session, they're --

13 MR. SCARBROUGH: Yes, I think --

14 MEMBER MARCH-LEUBA: This is not closed
15 session, this is --

16 MR. SCARBROUGH: I think we're getting into
17 that area. I think, I appreciate your question and I
18 agree, but I think we should talk about that in closed
19 session, if you don't mind.

20 MEMBER MARCH-LEUBA: Okay.

21 MEMBER CORRADINI: Any questions of the
22 current group? Which will probably become the next
23 group also, since they'll be going first in closed
24 session.

25 MEMBER MARCH-LEUBA: What's the plan for

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1 Chapter 15?

2 MEMBER CORRADINI: There is no open
3 presentation for Chapter 15.

4 MR. SNODDERLY: I'm sorry, Mike, there is
5 a note in Chapter 15.

6 MEMBER CORRADINI: Oh, I thought there
7 wasn't.

8 MR. SNODDERLY: There it is.

9 MEMBER DIMITRIJEVIC: I know I look
10 everywhere.

11 MEMBER CORRADINI: I was told there wasn't.

12 MEMBER BLEY: I think everything that could
13 go wrong has gone wrong today.

14 MEMBER DIMITRIJEVIC: Yes, I think we
15 should --

16 MR. SNODDERLY: We could go -- so, Mike, do
17 you want to just go into closed? If that's okay with
18 the staff, we could go right to just closed session
19 for 15 and discuss.

20 MS. KARAS: The presentation is an open
21 presentation.

22 MR. SNODDERLY: Right. So, do you want to
23 do it in an open session then?

24 MEMBER CORRADINI: Do we have --

25 MS. KARAS: I mean, it should be open

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1 session, there's nothing that's --

2 MR. SNODDERLY: Okay, that's fine.

3 MEMBER BLEY: We shouldn't close unless we
4 have to.

5 MR. SNODDERLY: Right.

6 MEMBER CORRADINI: Okay. My apologies, I
7 thought --

8 MR. SNODDERLY: Well, when --

9 MEMBER CORRADINI: So, thank you, and we'll
10 get the next group up.

11 MR. SNODDERLY: So, Mike, if I could make
12 a suggestion? I'm sorry, can I have the floor,
13 please? I -- Bob Weisman from OGC is here and I think
14 that it would be a very good idea for ACRS leadership
15 and Mike Corradini to speak to him, with regards to
16 the status of the section.

17 MEMBER CORRADINI: Is this open?

18 MR. WEISMAN: No, this is Bob Weisman from
19 OGC and if we could take a short break and meet in the
20 ACRS chambers for just a few minutes, I would
21 appreciate it.

22 MEMBER CORRADINI: Okay. So, let's take a
23 15-minute break, we'll be back a quarter of.

24 (Whereupon, the above-entitled matter went
25 off the record at 5:27 p.m. and resumed at 5:43 p.m.)

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1 MEMBER CORRADINI: Okay. So it's my
2 apology that somehow I was under the impression that
3 there was no open part of 15; so that's my mistake.
4 So we did it out of order. So, Jeff, are you going to
5 lead us?

6 MR. SCHMIDT: I'm Jeff Schmidt with
7 reactor systems, and I'm going to be going through the
8 safety evaluation with open eyes for Chapter 15.

9 These are the reviewers, the same one that
10 we showed in subcommittee, so I'm going to pass this.
11 So this is the agenda for today. So we're going to
12 have unclear open items which we'll try to define as
13 best we can; staff requirements for SECY-19-0036;
14 Chapter 15 limiting cases; we'll have a slide on that.

15 We'll talk about critical heat flux ratio
16 variance; long-term cooling, Chapter 15 exemptions,
17 return to power, and then finish up with containment
18 structure and containment (inaudible).

19 So unclear open items: items without a
20 mutual understanding and clearly-defined path for its
21 resolution. So I think these are the important
22 bullets, because that seems like it's kind of
23 nebulous. It includes items for which there may be an
24 agreement between NuScale and the NRC, but
25 documentation supporting information is pending. I

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1 think there is probably a lot that falls under that
2 category.

3 OUI lists changes frequently as new
4 material is submitted by NuScale; that's also very
5 true. A list of OUIs has been reduced since Chapter
6 15 SCR draft which is out a month or so.

7 So here is the list of unclear open items.
8 I think a lot of these are somewhat repetitive, so
9 they kind of fall under the same category, let's put
10 it that way.

11 Recriticality: again, this is a return to
12 power with natural circulation interrupted, so that's
13 a loss of the riser that we talked about before and
14 earlier today.

15 Excluding event scenarios of ejected rod
16 with margin for stuck rod. So this is whether you
17 have to include two stuck rods in the rod ejection
18 analysis. That was an open item.

19 NuScale RAI response pending, expected to
20 support decision at CRDM housings, robust with
21 enhances features precluding the need to evaluate
22 long-term effects such as recriticality within the
23 design basis of returning the short-term analysis.

24 So if we proceed down this path which is
25 where we're headed right now, the N minus 2 regarding

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1 recriticality would not be evaluated. This event
2 would really focus on the short-term energy deposition
3 and basically keeping the core intact for coolable
4 geometry over, say, a very short time period.

5 MEMBER MARCH-LEUBA: But then for return
6 to power you would not evaluate --

7 MR. SCHMIDT: We will not evaluate, yes.
8 That's right. This is more of a core disassembly
9 calculation, just as it would GDC-28.

10 Boron redistribution: we've talked about
11 that. This is just a potential return to power
12 scenario. We're having continuing dialogue regarding
13 volatility correlation and mixing assumptions. You
14 kind of alluded to before, the mixing and where the
15 boron goes, so that is still ongoing.

16 MEMBER MARCH-LEUBA: Yes, the way I
17 describe it is, there are physical mechanisms that
18 result in inhomogeneous distribution of boron.

19 MR. SCHMIDT: Right.

20 MEMBER MARCH-LEUBA: I don't know where it
21 is.

22 MR. SCHMIDT: So we are trying to, from
23 the staff, we are trying to determine what that mixing
24 might look like between the core and the riser and
25 effectively, what the changeover rate is.

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1 MEMBER MARCH-LEUBA: Yes.

2 MR. SCHMIDT: So far, we've been
3 unsuccessful, but we're still attempting.

4 Nuclear analysis parameters: return to
5 power; again, this is a return to power situation.
6 What do you assume for, say, MTCs or stuck rod where
7 it's basically feedback terms. Decay heat levels
8 precluding a return to power and addressing EOC-ECCS
9 return to power.

10 Again, a lot of these are return to power.
11 I think we're making good progress on the EOC-ECCS.

12 Long-term cooling: there's a statement in
13 the long-term cooling report that the analysis
14 demonstrates adequate cooling for 30 days. A NuScale
15 re-evaluating the analytical basis for that statement
16 and potential revisions to that statement.

17 Just so you know, the long-term cooling
18 report only really goes to 72 hours. That's a basic
19 assumption. But there is a statement in there that
20 discusses a 30-day coping period, and I think we're
21 trying to resolve whether that should really be in
22 there or not.

23 MEMBER MARCH-LEUBA: You need to
24 coordinate with the Chapter 20 guys, because the Rule
25 72 NuScale 114, and now you bring in 30?

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1 MR. SCHMIDT: I did not bring in 30, to be
2 clear. That's in the long-term cooling report.

3 MEMBER MARCH-LEUBA: From NuScale?

4 MR. SCHMIDT: That's correct.

5 MEMBER MARCH-LEUBA: So they changed their
6 mind?

7 MR. SCHMIDT: Well, this was done well
8 before, I think, the 14 days. So this proceeds the 14
9 days.

10 MEMBER MARCH-LEUBA: We have enough
11 problems with 72 and 14.

12 MR. SCHMIDT: Yes, yes. But he's right
13 that the fact that the finish line may be changing.

14 MEMBER MARCH-LEUBA: Okay.

15 MEMBER KIRCHNER: Let me just ask a
16 question. Would your review of this matter, long-term
17 cooling after 30 days, provide sufficient information
18 to your colleagues doing Chapter 20 to make a
19 determination? That 14 is half of 30.

20 MR. SCHMIDT: So just to be clear, the
21 long-term cooling review is only out to 72 hours
22 currently. That's a base assumption. The staff is
23 unsure what to do with this. A single statement in
24 the long-term cooling report that addresses 30 days,
25 mostly regarding ultimate heat sink level --

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1 MEMBER MARCH-LEUBA: That's typically
2 true. There's always a simple limitation.

3 MR. SCHMIDT: Well, it's a technical
4 report, not a topical report, so there's no
5 limitations.

6 MEMBER MARCH-LEUBA: There's no
7 limitations?

8 MR. SCHMIDT: No, no. There either has to
9 be a justification in the long-term cooling report
10 that staff can review or the sets could be --

11 MEMBER MARCH-LEUBA: A return to holding?

12 MR. SCHMIDT: Well, the sets could be
13 struck, and it's really a long-term cooling after 72
14 hours. That's a base assumption. You read the
15 executive summary, you'll see that.

16 MEMBER CORRADINI: So just one thing for
17 the members who weren't here on the subcommittee; we
18 saw in closed session, which we'll probably have that
19 if we want it tomorrow, just because of the time of
20 day -- but in closed session of the subcommittee we
21 heard from the staff that their -- I'll call it audit
22 calculation analysis went out to 72 hours, right?
23 Because that's what I remember you showing us in
24 various shapes and forms.

25 MR. SCHMIDT: For return to power or long-

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1 term cooling? See, I separate those --

2 MEMBER CORRADINI: I think it was return
3 to power.

4 MR. SCHMIDT: Yes, I think it was return
5 to power also.

6 So to be clear, the long-term cooling
7 report is just residual and decay heat, but it also
8 only extends to 72 hours. And the return to power
9 also only goes to 72 hours.

10 MEMBER BROWN: I got it; it was long-
11 term cooling that we were talking --

12 MEMBER CORRADINI: That's what he said.

13 MEMBER BROWN: No, he said --

14 MR. SCHMIDT: In return to power we were
15 showing graphs to 72 hours. long-term cooling we
16 just had a slide and no graphs.

17 MEMBER BROWN: No, but you were talking
18 about the emphasis from -- yesterday. They only do
19 their assessments for --

20 MEMBER MARCH-LEUBA: That was yesterday?

21 (Simultaneous speaking.)

22 MEMBER CORRADINI: All I said was, the
23 subcommittee meeting from the staff, when they showed
24 us detailed audit calculations for return to power was
25 for 72 hours. Jeff corrected me that the long-term

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1 cooling was just a slide, nothing beyond that, and
2 that was Tuesday.

3 MEMBER MARCH-LEUBA: But you're talking
4 about two different subcommittees, he is talking about
5 yesterday's RELAP.

6 MEMBER CORRADINI: No, I'm not talking
7 about yesterday; I'm talking about the subcommittee
8 meeting in June.

9 MEMBER BROWN: Okay. All right. I was
10 talking about whether they were talking about the
11 longer-term, they only have to do an audit or
12 evaluation up to a certain time by rule or something
13 like that. That's what I was remembering. I got
14 confused; sorry about that.

15 MEMBER DIMITRIJEVIC: And this is
16 completely different conditions than for Chapter 47.
17 For RELAP this one for long-term cooling they have
18 something government commissions.

19 MR. SCHMIDT: So the other unclear open
20 items are topical report methodologies; we talked
21 about those in subcommittee as well as RELAP version
22 1.4 and revised base model. Staff has those and are
23 actively reviewing it.

24 Steam generator heat transfer uncertainty,
25 which goes to the non-LOCA topical report; still

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1 awaiting revised NuScale analysis, expected to
2 demonstrate that the results are insensitive to the
3 assumed heat transfer coefficient. Really, the 30
4 percent, I think, is being removed from the non-LOCA
5 topical report uncertainty.

6 MEMBER CORRADINI: This is secondary site;
7 heat transfer?

8 MR. SCHMIDT: This is heat transfer from
9 the primary to the secondary.

10 MEMBER CORRADINI: So, integrated?

11 MR. SCHMIDT: Integrated.

12 MEMBER CORRADINI: Okay.

13 MR. SCHMIDT: Chapter 15 re-analysis from
14 design and methodology changes: we talked about this
15 before. ECCS logic changes; you guys hit that up a
16 little bit today on the RCS level, ECCS actuation
17 going away.

18 Those ECCS logic changes are expected by
19 August 30th. Decay heat removal logic changes we've
20 also discussed. Those analyses are expected August
21 30th as well.

22 And then the RELAP Version 1.4 is also
23 August 30th.

24 MEMBER CORRADINI: So the first bullet
25 under the last underline --

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1 MR. SCHMIDT: First bullet under the last
2 underline; got it. ECCS logic changes?

3 MEMBER CORRADINI: Yes. Are we going to
4 hear about that today?

5 MR. SCHMIDT: Other than we have to
6 evaluate, and it's coming August 30th. No, you're not
7 going to hear any details.

8 MEMBER CORRADINI: All right. Because I
9 think some of us are still struggling to understand
10 the logic of the logic change. So we're expecting to
11 ask that of NuScale when we go visit them in
12 Corvallis.

13 MR. SCHMIDT: Yes, I can't speak to those.

14 MEMBER CORRADINI: That's fine. I just
15 wanted to make sure.

16 MEMBER KIRCHNER: Jeff, refresh my memory.
17 What's the DHRS logic change?

18 MR. SCHMIDT: They're separating out, I
19 think -- and I have a backup slide to remind me -- but
20 I think they're basically changing out when you would
21 isolate secondary side versus activate the decay heat
22 removal system.

23 Actually, if I could ask Boyce, who is the
24 reviewer of that section, he's probably better to
25 speak to it.

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1 MR. TRAVIS: So originally, all the DHRS
2 actuations were integrated, and now that they've
3 revised the DHRS logic, not only for operational
4 concern, but it also resulted in -- there's isolation
5 signals in DHRS actuation signals that have been
6 separated. And so the isolation signals will occur
7 first. Sometimes they occur at the same time as the
8 actuation signals, but not always.

9 It preserves the inventory in the DHRS
10 with the isolation signal, and then they actuate the
11 DHRS with the actuation valves to start the
12 circulation on the --

13 MEMBER MARCH-LEUBA: If I remember
14 correctly, DHRS will trip, because the isolation
15 happened, which will cause the pressure to rise and
16 the DHRS will actuate.

17 MR. TRAVIS: For some transients, yes.

18 MEMBER MARCH-LEUBA: It's similar to
19 bullet number 1 under the last underline.

20 MEMBER CORRADINI: Okay, but since we
21 probably won't discuss this again, I want to make sure
22 this -- I want to get the reasoning. In this case the
23 reason is to preserve the inventory.

24 MR. TRAVIS: So that's the net result, but
25 the reasoning for the change, and you'd have to speak

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1 to NuScale for the exact reasoning, but I believe it's
2 an operational concern related to startup
3 considerations for how their logic interlocks, and
4 this is a consequence of that. But the functional
5 effect of the change in the DHRS system is not --

6 MEMBER CORRADINI: So it's a startup logic
7 issue.

8 MR. TRAVIS: Yes, and so the --

9 MEMBER CORRADINI: Okay.

10 MR. TRAVIS: -- the effect on -- we expect
11 the effect on the Chapter 15 analyses to be negligible
12 for the purposes of this analysis because in all
13 cases, the DHRS will be isolated at a similar time
14 when it was previously, but it may not actuate the
15 DHRS if it's not required; for instance, if it's an
16 over pool event.

17 MEMBER CORRADINI: All right; I appreciate
18 it.

19 MEMBER MARCH-LEUBA: A concern I also have
20 is that by going in this circuitous route, you
21 increase the probability of not doing something you're
22 supposed to do.

23 Because it makes not much sense unless
24 you're doing (inaudible) power maneuvering that you
25 isolate the heat sink, isolate the secondary and do

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1 not turn DHRS on. And that's what they do; they
2 isolate the secondary, and they wait to see if the
3 secondary will heat up to turn on the emergency.

4 It works; it works in the computer very
5 well, but it only works if everything works as
6 designed.

7 MEMBER CORRADINI: You were going to say
8 something?

9 MR. TRAVIS: No, I think you've captured
10 the -- that's not how I phrased it, but that does
11 capture the --

12 MEMBER CORRADINI: But I hear what? I'm
13 sorry.

14 MR. TRAVIS: That's now how I would have
15 phrased it, but that does capture what is being done,
16 yes.

17 MEMBER CORRADINI: Okay. Sorry.

18 MEMBER MARCH-LEUBA: Okay, you beat me to
19 it again.

20 MEMBER BROWN: Before you shift slides,
21 there were these unclear open items, which is a new
22 revised definition of how we do things, which is just
23 fine. You explained the basis, and you said there
24 were three categories, items for which there may be
25 agreement, but documentation supporting information is

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1 pending.

2 And I guess my question is, there's five
3 bullets under return to power analysis. Is that the
4 first bullet includes items where you all totally
5 agreed with it, but there's documentation pending? Or
6 is that one where we're still looking for something
7 else, and other disagreements are being put aside?

8 MR. SCHMIDT: So I think NuScale mentioned
9 in their open presentation that some of the return to
10 power analyses are being redone with different
11 methodology than what is currently in the DCA right
12 now.

13 Those analyses -- and they showed the same
14 results of those analyses -- the staff needs to review
15 those analyses still. So that is work that NuScale is
16 doing that the staff still needs to review, but I
17 think conceptually, you still have to look at it. We
18 can't make a definitive statement at this point, but
19 at least conceptually I think, regarding the EOC ones
20 --

21 MEMBER CORRADINI: EOC?

22 MR. SCHMIDT: End of cycle. So it would
23 be a first bullet. I think the second bullet is just
24 rod ejection in lines 2. So the first and fourth and
25 fifth bullets are heading towards resolution.

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1 MEMBER BROWN: Did we do Chapter 15 this
2 morning? I was --

3 MEMBER MARCH-LEUBA: No.

4 MEMBER BROWN: The NuScale?

5 MEMBER MARCH-LEUBA: Oh, yes.

6 MEMBER BROWN: Yes, I was missing for
7 other reasons. I got here late, so I apologize.
8 That's why I wanted to ask the question.

9 MR. SCHMIDT: So I think it's fair to say
10 that while we're still calling these unclear open
11 items, some have more clarity than others.

12 MEMBER BROWN: That we will revisit. I
13 presume it will be revisited to finalize it.

14 MR. SCHMIDT: Yes.

15 MEMBER BROWN: At some point with us as
16 well, right?

17 MR. SCHMIDT: Yes, it has to be, right.

18 MEMBER KIRCHNER: Okay. Help me a little,
19 though, because your prior slide said items in bigger
20 font, mutually understood and clearly defined path.
21 Which of these bullets on the subsequent page are
22 without mutually understanding and clearly defined?
23 Are there any show-stoppers here, or is it just a
24 question of edit material?

25 MR. SCHMIDT: So I would say the one at

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1 least in my mind that is probably the most unclear is
2 the boron redistribution which would be ECCS potential
3 return to power.

4 MEMBER KIRCHNER: Okay.

5 MR. SCHMIDT: If you wanted to try and
6 flavor these.

7 MEMBER MARCH-LEUBA: Because all the
8 others I would not call them unclear. We have called
9 them we haven't had time to review them yet, or mostly
10 because of a late submittal.

11 MEMBER KIRCHNER: Right.

12 MEMBER MARCH-LEUBA: Because they changed
13 their mind and send us something new.

14 MR. SCHMIDT: Yes, some of these are, I
15 think, on a relatively clear path forward.

16 MEMBER MARCH-LEUBA: Those happen in every
17 review. There are a few where we are waiting for the
18 RAI, and we just keep it in line, and we have to keep
19 moving.

20 MR. SCHMIDT: Okay. Staff requirements
21 for SECY-19-0036: the inadvertent actuation block
22 valve IAB is part of the emergent core cooling system
23 valve and is designed to prevent high pressure RPB
24 blowdown when the DC power supply is lost to the ECCS
25 trip valves.

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1 So that's just a function of the block
2 valve. I think we probably know that already at this
3 point.

4 Chapter 15 analyses don't assume single
5 failure of the IAB block, and the key is to block. It
6 does include the failure to open, but at high
7 pressures we need it to block first and then open, so
8 it's a two-step phase.

9 The Commission has determined that the
10 staff should review Chapter 15 of the NuScale DCA
11 without assuming a single failure of the inadvertent
12 actuation block valve to close.

13 So what that means to Chapter 15 is
14 there's no change to the DCA relative to the IAB
15 single failure.

16 MEMBER MARCH-LEUBA: IAB valve to close --

17 MR. SCHMIDT: To block.

18 MEMBER MARCH-LEUBA: To block?

19 MR. SCHMIDT: To block.

20 MEMBER MARCH-LEUBA: So if it failed to
21 block, you should not assume failure to block.

22 MR. SCHMIDT: We should not assume failure
23 to block.

24 MEMBER MARCH-LEUBA: Should you assume a
25 failure to open?

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1 MR. SCHMIDT: Yes, we do assume a failure
2 to open.

3 MEMBER MARCH-LEUBA: No, no; to allow
4 when the pressure is high.

5 MR. SCHMIDT: Allowed -- no.

6 MEMBER MARCH-LEUBA: Well, that would be
7 a failure to block.

8 MR. SCHMIDT: That's a failure to block.

9 MEMBER MARCH-LEUBA: Not open when the
10 pressure is low.

11 MR. SCHMIDT: Not open when --

12 MEMBER MARCH-LEUBA: The needle goes up,
13 blocks, and it never comes back.

14 MR. SCHMIDT: And it never comes back.

15 MEMBER MARCH-LEUBA: So the valve never
16 opens when --

17 MR. SCHMIDT: No, we only take single-
18 failure of some valves to open. We still take that.

19 MEMBER MARCH-LEUBA: You take it as a
20 global for the whole valve.

21 MR. SCHMIDT: Yes, I think that's captured
22 by the --

23 MR. RAD: Hey, Jose, this is Zack Rad. I
24 want to clarify one thing. Treating the valve of IAB
25 as passive relative to the single failure criteria

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1 means that a failure of that device in addition to the
2 initiating event is not assumed. It could be the
3 initiating event.

4 MEMBER MARCH-LEUBA: Understood, thank
5 you.

6 MR. SCHMIDT: So this is the Chapter 15
7 limit cases. What we tried to do here is just give an
8 overview of Chapter 15, so we looked at basically the
9 figures of merit we use in Chapter 15, RCS pressure,
10 steam generator pressure, critical heat flux, and then
11 we have two critical heat flux; one VIPRE, one RELAP,
12 and then collapsed liquid level above the top of the
13 active fuel.

14 So you can basically go down; again, we're
15 using AOO criteria. I think that's important to keep
16 in mind. And you can basically look at the
17 percentages, go to the right-hand column and look at
18 the percentages of margin that are available. Any
19 questions on that?

20 And then I'm going to turn it over to
21 Antonio to discuss CHF correlations.

22 MEMBER MARCH-LEUBA: We want to talk about
23 this in closed session tomorrow so we can talk
24 numbers?

25 MR. SCHMIDT: No, no. We hadn't planned

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1 on it, not that I'm aware of.

2 MEMBER MARCH-LEUBA: So I cannot ask
3 numbers?

4 MEMBER CORRADINI: You can ask difference
5 in numbers; you can't ask exact numbers.

6 MEMBER REMPE: Is there a concern that you
7 (inaudible)?

8 MEMBER MARCH-LEUBA: Very significant --
9 I have a very significant concern, but Antonio, go for
10 it.

11 MR. BARRETT: Okay. For Chapter 15 CHF
12 correlations, for non-LOCA events where CHF is
13 calculated, the NSP4-CHF correlation is used in the
14 VIPRE subchannel analysis, this has been approved by
15 the NRC and the ACRS has already seen it.

16 For LOCA and LOCA-like events where CHF is
17 calculated, there's a high-flow correlation and a low-
18 flow correlation which is used in the NRELAP5 systems
19 analysis code, and that methodology is still currently
20 under staff review.

21 MEMBER CORRADINI: I was -- do you have
22 more to say?

23 MR. BARRETT: Yes. This is just an intro
24 to get everybody on the same page.

25 MEMBER CORRADINI: Thank you.

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1 MR. BARRETT: Okay. Now, with regard to
2 the critical heat flux ration variances, there were a
3 couple of different things that we looked at to try
4 and explain where these variances come from,
5 especially in the initial minimum critical heat flux
6 ratios.

7 So first we looked at the correlation
8 which is the NSP-4 correlation for non-LOCA and then
9 the high-flow and low-flow for the LOCA and LOCA-like
10 events.

11 And if you take these correlations and
12 apply the exact same conditions, barring anything
13 else, you'll find that the NSP-4 actually provides a
14 lower CHF.

15 MEMBER MARCH-LEUBA: By lower, meaning
16 what?

17 MR. BARRETT: So if you had the same --

18 MEMBER MARCH-LEUBA: Kilowatts?

19 MR. BARRETT: Yes. And so -- I can't
20 remember the --

21 MEMBER MARCH-LEUBA: Kilowatts per minute.

22 MR. BARRETT: Yes. And if you use the
23 same exact efflux, you would get a lower minimum
24 critical heat flux ratio for NSP-4.

25 So the takeaway is that the variance that

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1 you would see between a LOCA and non-LOCA event is not
2 squarely on the correlation that's used. It provides
3 some variance, but that's not the key takeaway.

4 So eventually, if you look at the initial
5 conditions, the analysis inputs, there's some biasing
6 going on that are for specific events: the RCS
7 temperature, the RCS flow, the pressurizer pressure;
8 this all has an impact on your critical heat flux
9 ratio and also for NuScale for some events, they
10 looked at different power levels, which would also
11 have an impact.

12 And finally, when you look at the
13 methodology and you look at the analysis codes that
14 are being used for VIPRE as a subchannel analysis, and
15 it basically only covers the period of interest.

16 So if you have the minimum critical heat
17 flux ratio starting at time zero, you'll get the first
18 50 seconds including time zero. If it occurs at time
19 100 seconds, you'll get 75 seconds to 125 seconds.

20 So if you look at that plot, you may see
21 what you think is a difference. But as for RELAP,
22 which is for the LOCA-like events, it continuously
23 calculates a minimum critical heat flux ratio.

24 So finally, when you look at the modeling
25 itself for how you would model something in VIPRE

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1 versus how you would model something in RELAP, VIPRE
2 is a subchannel analysis code, so you can't inherently
3 model it the exact same way as you would in RELAP,
4 especially in terms of peaking factors and the heat
5 flux assumptions.

6 So if you took RELAP, for example, and you
7 tried to do something consistent with what was done in
8 VIPRE, but you did something that was consistent, but
9 you used more conservative peaking factors or more
10 conservative modeling of those peaking factors, you
11 could really drive down your minimum critical heat
12 flux ratio.

13 MEMBER CORRADINI: So let me ask the
14 question a different way. This all makes sense, so
15 has staff taken an example and walked through these to
16 show you can replicate within the variance of the two
17 correlations the same ratio?

18 I mean, what you're saying to me is,
19 there's reasons; here are the reasons. And my next
20 question is, have you done the detective work to show
21 for an example that these reasons all wash out, and
22 you do get the same result for a condition?

23 MR. BARRETT: Well, I'm not saying you --
24 for the correlation, yes.

25 MEMBER CORRADINI: Okay.

1 MR. BARRETT: For the correlation, I've
2 looked at both correlations and some conditions that
3 are close to the initial normal operating conditions.

4 MEMBER MARCH-LEUBA: So you plug in
5 pressure, flow, cooling, on both of them, and you get
6 similar CHF values?

7 MR. BARRETT: Somewhat similar, with NSP-4
8 being a little bit lower.

9 MEMBER MARCH-LEUBA: Okay. That's even
10 more concerning. Let me tell you what my complaint
11 is, then I'll tell you what my concern is.

12 The complaint is, this is a steady-state.
13 I'm not worried about transients; it's time
14 (inaudible) on the scale.

15 MR. BARRETT: Correct.

16 MEMBER MARCH-LEUBA: They're off by a very
17 significant margin, and I'm not allowed to say how
18 much it is here because we don't have a closed
19 session. But the amount is scary.

20 MR. BARRETT: Right. So the point I'm
21 trying to get at is, that is not necessarily related
22 to the CHF correlation. It's more related to how it
23 was modeled.

24 MEMBER MARCH-LEUBA: And that's why I'm
25 saying I'm even more scared now. It is not the

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1 correlation; it is the model.

2 MR. BARRETT: Right.

3 MEMBER MARCH-LEUBA: What makes you think
4 that the model, that is off by a factor of X with
5 respect to the other, in a steady-state, that's any
6 good during the transient if it misses the steady-
7 state by that much?

8 MR. BARRETT: I hear you. So I guess the
9 point --

10 MEMBER PETTI: Can you help the
11 uninformed? It misses it because we're in a natural
12 circulation regime.

13 MEMBER MARCH-LEUBA: No.

14 MEMBER PETTI: I mean --

15 MEMBER MARCH-LEUBA: No, no.

16 MEMBER PETTI: Yes, we've modeled CHF in
17 reactors many, many times. What's unique here that's
18 causing the model --

19 MEMBER MARCH-LEUBA: I don't know. I know
20 they have two models, and the starter time CO, they
21 initialize the steady-state. And the answer is a
22 factor of X, X being a large number, not 10 percent.

23 MEMBER KIRCHNER: That should be of
24 concern, because the CHF correlations, by and large,
25 are not developed with transient data; they're

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1 developed by running the rig, the test rig. You just
2 gradually go until you get a burnout or CHF, whatever
3 you want to call it.

4 And the thermocouples pick that up, and
5 that's how they do it. So it's a quasi-steady-state
6 correlation, and they should be close. They're off by
7 a significant --

8 MR. BARRETT: Yes. So the correlations,
9 I think when you look at just the correlations it
10 looks somewhat reasonable.

11 MEMBER MARCH-LEUBA: So the methodology is
12 incorrect.

13 MR. BARRETT: So the methodology for
14 VIPRE, the subchannel methodology has been reviewed by
15 the staff, has been approved by the staff, and the
16 ACRS has already reviewed that.

17 MEMBER MARCH-LEUBA: Based on the heat
18 fluxes of this code would expect the VIPRE numbers to
19 be more representative of reality than the RELAP
20 numbers. RELAP numbers are --

21 MR. BARRETT: Correct. That is a
22 reasonable observation. So when you go to something
23 different, go to RELAP, and you are using something
24 that is acceptable but more conservative, that also
25 could be okay.

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1 So it doesn't necessarily mean that
2 there's anything wrong, but that is still under review
3 by the staff.

4 MEMBER MARCH-LEUBA: Let me put something
5 else in the record, a concept. If I make a wrong
6 calculation, I compare it as my benchmark, I have a
7 50-50 chance of being conservative. You fall to the
8 left, you fall to the right.

9 And if you a highlight for every time I
10 heard that; We didn't hit the benchmark, but we were
11 conservative; I say, Well, you were wrong. You were
12 not conservative. You were wrong. And in this case,
13 I am asking you, why are they conservative? Can you
14 put your hand on (inaudible) and say, I want to be
15 conservative anytime for every condition and during
16 the transient.

17 MR. BARRETT: Okay. So this is what I'm
18 trying to say. So for VIPRE and the subchannel
19 analysis in NSP-4, that is acceptable, and it's good.
20 For the RELAP, what was done for RELAP is still under
21 staff review, but that is the one that looks to be
22 more conservative.

23 MEMBER MARCH-LEUBA: That one is the one
24 I don't believe.

25 MEMBER CORRADINI: Yes, but I think -- I

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1 want him to finish, because what I'm hearing him say
2 is, it's the starting conditions, the correlation you
3 check, there are differences, but not big differences.
4 It's the starting conditions which could be different.
5 It could be a hot channel assumption versus an average
6 channel. It could be how I tend to do what are
7 traditionally accidents versus AOOs. That's what I'm
8 hearing you say.

9 What I'm still thinking the only way to
10 satisfy him is, get to a root cause on an example to
11 show that they all come in line if you could do that.
12 That's the only way to satisfy his concern.

13 MEMBER MARCH-LEUBA: I want you to tell
14 me, we're using a hollow peaking factor of 3.0 as
15 opposed to asserting 1.4 because we want to be
16 conservative. That won't be conservative.

17 MR. BARRETT: That is the what I'm getting
18 at. That's what I'm trying to say.

19 MEMBER MARCH-LEUBA: But we need to know
20 what it is. We don't license reactors based on what
21 the (inaudible) says.

22 MR. BARRETT: I agree.

23 MEMBER MARCH-LEUBA: It may be a peaking
24 factor; it may be the (inaudible). It may be --

25 MR. BARRETT: Because it's still under

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1 review is why I don't necessarily want to get too far
2 into the details.

3 MEMBER MARCH-LEUBA: We will see all of
4 this in October-November.

5 MR. BARRETT: Okay.

6 MEMBER MARCH-LEUBA: I will give you a
7 pass now, but I won't give you a pass then.

8 MR. BARRETT: Understood. The point that
9 I'm trying to make is, the peaking factor is what
10 seems to be, when they try to line it up is making it
11 more conservative.

12 MEMBER MARCH-LEUBA: Right. And the
13 concern I have. That was called recreational
14 complaining. The concern is that by using this
15 conservative assumption, we reach the wrong
16 conclusions. I heard you guys say that the limiting
17 transient is an event opening of (inaudible).

18 But that's only because the steady-state
19 was too low. The transient was nothing. I mean, you
20 have to get a microscope to see the drop in CHFR for
21 that one.

22 MR. SCHMIDT: So if you go back to slide
23 7; this is Jeff Schmidt. Let's go back to slide 7.

24 MEMBER MARCH-LEUBA: Let me finish. I
25 venture to say that if we stopped at the high CHFR

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1 level with a LOCA methodology, opening two IABs will
2 be perfectly acceptable. We will have to have the
3 Commission issue an order that you cannot operate. We
4 are shooting ourselves in the foot by adding those
5 peaking factors, whatever they are.

6 MR. BARRETT: So I understand what you're
7 saying, but we review what we're provided, so if they
8 are being --

9 MEMBER MARCH-LEUBA: An acceptable outcome
10 of this review is to say, No, you did a bad job. Do
11 it again. On occasion, we should do it.

12 (Simultaneous speaking.)

13 MR. SCHMIDT: The staff's job is to
14 determine if these are conservative calculations for
15 Chapter 15.

16 MEMBER MARCH-LEUBA: What I'm asking you
17 is --

18 MR. SCHMIDT: Not that they could be done
19 better.

20 MEMBER MARCH-LEUBA: Prove to me that it's
21 conservative. Why is it conservative instead of --

22 MR. SCHMIDT: I understand.

23 MEMBER MARCH-LEUBA: If it was wrong, and
24 I happen to be lucky.

25 MR. SCHMIDT: Yes, right. So just to go

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1 back one point though is, what we try to do in slide
2 7 when we talk about the VIPRE CHF and the RELAP CHF
3 is, the way I look at really what the margin is, it's
4 the margin relative to its design limit. They all
5 have different design limits, right?

6 MEMBER MARCH-LEUBA: Right.

7 MR. SCHMIDT: So this gives you some idea
8 -- and there's really not a lot of spread here between
9 25 and 24, so I think that's a point well taken, is
10 that that's not overly limiting, relative to single
11 rod withdrawal.

12 But when you look at the margins, it's not
13 just the starting value; it's how close you get to
14 your design limit for that correlation that matters.

15 MEMBER MARCH-LEUBA: Yes, but if your
16 biased in your initial CHFR, and one of them is an
17 American utility but much larger than any other one,
18 if you start with a CHFR of 5, then you will get close
19 to the limit.

20 The delta; if you look at a delta for the
21 1566, this is the same; it's 0.05. If you didn't --
22 it didn't drop. I mean, if you look at the figure
23 with your arms extended, it's a flood light.

24 MR. SCHMIDT: So if you're saying you
25 could redo that one with a different correlation, you

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1 would have --

2 MEMBER MARCH-LEUBA: Same correlation,
3 proper initial conditions, or adjusting the right
4 methodology.

5 MR. SCHMIDT: Or using a less conservative
6 methodology.

7 MEMBER MARCH-LEUBA: My concern is not the
8 methodology. My concern -- the methodology does not
9 benchmark the data. We haven't done a benchmark, but
10 they would apply the methodology --

11 MEMBER CORRADINI: I'm not going to try to
12 cut you off, but you told me you were going to talk
13 about it in November-December, so have we talked about
14 it enough now?

15 MEMBER MARCH-LEUBA: No.

16 MEMBER KIRCHNER: May I ask a question of
17 clarification? So when you got to the fourth event,
18 the inadvertent operation, the ECCS, you're now
19 switching to the RELAP code to do the analysis.

20 MR. SCHMIDT: Yes.

21 MEMBER KIRCHNER: Why is the acceptance
22 criteria 1.13 there?

23 MR. SCHMIDT: That is the -- probably 9595
24 plus anything you put in for --

25 MEMBER KIRCHNER: For the RELAP

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1 correlation?

2 MR. SCHMIDT: Right. And so --

3 MEMBER KIRCHNER: Because 1.284 is the
4 NSP-4 NuScale correlation.

5 MR. SCHMIDT: Right. That's correct.

6 MR. BARRETT: I think you can look at it
7 as the uncertainty from CHF of critical heat plus
8 ratio of one, so that's the uncertainty; 0.013 is the
9 uncertainty associated with that correlation; 0.284 is
10 the uncertainty associated with the NSP-4 correlation,
11 and then wherever you start out with.

12 So once you get close to it, since it's
13 your uncertainty, once you hit that you basically
14 assume that --

15 MEMBER KIRCHNER: I guess intuitively what
16 I'm reacting to is that I would expect the uncertainty
17 to be greater for that correlation than for the NSP-4
18 correlation. And the reason is, that correlation is
19 being used primarily for steady-state operations for
20 the NuScale correlation, and VIPRE is being mainly
21 used for subchannel analysis which is the simpler
22 problem with less uncertainty.

23 You've got more uncertainty in transient
24 CHF than you do in steady-state. I just don't get it.

25 MR. SCHMIDT: Just to be clear, VIPRE is

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1 using transients too. It's a transient --

2 MEMBER KIRCHNER: I know that, but you
3 benchmark the VIPRE NuScale CHF correlation basically
4 on steady-state operations.

5 MEMBER CORRADINI: I think Jeff has a
6 lifeline.

7 MR. SCHMIDT: Yes, I'm going to call a
8 lifeline.

9 MR. DRZEWIECKI: This is Tim Drzweiecki,
10 (inaudible) staff. I have looked on these
11 correlations. One reason why it's lower; both are
12 based on a 9595 limit --

13 MEMBER KIRCHNER: I know that.

14 MR. DRZEWIECKI: -- and they're both based
15 on the same test data set. Now, you're right; the
16 NSP-4 is actually a best fit to data and its benchmark
17 as well, but the (inaudible) correlation, which is
18 then fit to this data account with a limit.

19 One reason why this limit is lower has to
20 do with the same reason why you're probably seeing a
21 more conservative result when you run RELAP is the
22 fact that that methodology and your peaking factors
23 are going to bias your results in a conservative way,
24 such that when you have 9595 limit, you are actually
25 getting a more conservative result, such that you're

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1 going to predict a lower CHF or lower CHFR, usually,
2 so with the base correlation so you can have a lower
3 limit.

4 MEMBER MARCH-LEUBA: Because your
5 correlation is making an incorrect prediction of the
6 CHF, you give it a lower uncertainty?

7 MEMBER CORRADINI: No, no.

8 MR. DRZEWIECKI: I'm saying it's biased
9 lower. It's biased lower; I'm not saying that the
10 error is lower; I'm saying that it's biased lower.

11 MEMBER MARCH-LEUBA: Okay.

12 MR. DRZEWIECKI: I think this will be more
13 clear once you walk through the whole CHF review as
14 part of LOCA, but you'll have to see --

15 MEMBER MARCH-LEUBA: Right now I want to,
16 when we get off the microphone, I'll tell Walt what
17 the difference between the two predictions is. It's
18 unacceptable in my point of view. It's unacceptable
19 in the 21st century. Now, we get in a steady-state a
20 prediction of the critical heat flux that is off by a
21 factor of X, X being a large number.

22 MR. DRZEWIECKI: I understand what you're
23 saying, but I think if we go and look at how they're
24 setting what their peak PIM power is, and the powers
25 that would surround it, if that is done in a very

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1 conservative way inside of RELAP, that is what we're
2 seeing here.

3 If you look at what's inside of VIPRE,
4 that's also conservative, but it's not biased nearly
5 as much. In fact, your peak PIM factor is set by your
6 tech specs, where in RELAP it's more.

7 MEMBER MARCH-LEUBA: What I said earlier,
8 we're not licensing reactors based on a gut feeling or
9 what it may be. We license reactors because we know
10 what it is. So it is the responsibility of the staff
11 to know what it is.

12 (Simultaneous speaking.)

13 PARTICIPANT: Antonio's still digging and
14 is trying to make comparisons to understand, is it the
15 peaking factor? Is it? Are you trying to --

16 MR. TRAVIS: Yes, that's a fair statement.
17 It's still under review. I intend to get to --

18 MEMBER MARCH-LEUBA: I can do this too.
19 It's not your responsibility. The first thing you
20 discuss do tomorrow morning is issue -- tell me what
21 the difference is. You're the one that wants to do
22 this correlation. It's not your job to be Sherlock
23 Holmes; it's their job.

24 As I say, it's a perfectly acceptable
25 result of this is, if you give me such disparity in

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1 results, then you don't know what you are doing. Tell
2 me why or go back to the drawing board.

3 MEMBER BLEY: We've raised the questions;
4 we don't manage the staff.

5 MEMBER MARCH-LEUBA: I was giving you
6 friendly advice.

7 (Laughter.)

8 MEMBER CORRADINI: Okay. Let's move on.

9 MR. SCHMIDT: All right. ECCS long-term
10 capability and analysis. Analysis addresses residual
11 heat removal for LOCA and non-LOCA events; focuses on
12 reduced inventory events which challenge the ECCS
13 cooling capability; assumes shutdown, which I think is
14 important; evaluates Chapter 72, which we discussed
15 earlier; figures of merit are decreasing clad
16 temperature, minimum RCS temperature will prevent
17 (inaudible) precipitation, and collapsed liquid level
18 remains above the top of the active fuel.

19 Staff review focused on validation of NIST
20 test data and RELAP model, assumed analysis
21 conditions. In-vessel downstream effects evaluated
22 was formed in DCA Section 6.3, and the staff found the
23 NuScale evaluation acceptable. That refers to the in-
24 vessel downstream effects.

25 So anything on long-term cooling?

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1 Appendix K exemption; so ANS-71 is replaced with ANS-
2 73 and RELAP5. That's fairly common in RELAP5, that
3 you typically have the 73 in there.

4 Post-LOCA PWR reflood and refill
5 phenomenon are not to be encountered, so there's the
6 post-CHF regime heat transfer correlations. Baker-
7 Just water metal-water reactions; obviously, if you
8 don't have uncover, you don't have a significant metal
9 -water reaction; clad swelling and rupture.

10 GDC-27; I'm going to go pretty quick
11 through that. We talked about this before. The staff
12 took the position that GDC-27 meant that you were in
13 a shutdown configuration.

14 So following an initial shutdown, the
15 NuScale reactor can return to criticality during a
16 cooldown event on either decay heat or lower ECCS.
17 NuScale submitted an exemption to 27, requested
18 approval of a principal design criteria.

19 Exemption of (inaudible) also included a
20 lack of ECCS injection, which we should know by now.

21 So we wrote a SECY-180099, which basically
22 had these three success criteria to determine if the
23 exemption was going to be approved, and the first one
24 is basically preserve the SAFDLs four-year events.

25 The combination of circumstances and

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1 conditions leading to actual post-reactor trip, return
2 to criticality is not expected to occur in the
3 lifetime of the plant.

4 And there's incremental risk to the public
5 health and safety from hypothetical return to
6 criticality at the NuScale facility with multiple
7 reactor modules does not adversely erode the margin
8 between the Commission's goals for new reactor designs
9 related to estimated frequencies of core damage or
10 large releases and those calculated for the NuScale
11 design.

12 So that's largely met; if you can meet the
13 SAFDLs, you're going to be meeting that.

14 ACR supported the proposed staff criteria
15 with the addition of evaluating the overall risk,
16 which was the third bullet above. Satisfying the
17 three criteria in the SECY would ensure no undue risks
18 to public health and safety.

19 So we talked about this before; three
20 scenarios can lead to return of power, KD removal of
21 cooldown with keeping DC power. The KD removal
22 cooldown without DC power, which is when you're going
23 to actuate the ECCS at the set point, and then just a
24 valid ECCS signal and cooldown can occur from most
25 Chapter 15 events.

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1 Key assumptions: no operator action; only
2 safety-related equipment is used to mitigate the
3 event. The worst stuck rod is assumed out of the core
4 consistent with the current GDCs.

5 Return to power analysis: that's in
6 Section 1506 of the DCA. It presents a decay heat
7 removal cooldown, retaining single-phase natural
8 circulation with the ECCS valves opening at the
9 maximum return to power, which is thought to bound all
10 the three scenarios that I listed in the previous
11 slide.

12 In the DCA analysis, maximum core average
13 return to power is 10 percent rated thermal power,
14 with the equilibrium value at about 2.5 percent, and
15 the CHF limits are met.

16 A potential decay heat system return to
17 power when water level drops below the riser still
18 needs to be evaluated to ensure the SAFDLs are met.
19 That was just uncovered as part of the review.

20 EOC-ECCS return to power still needs to be
21 evaluated to ensure SAFDLs are met, and the boron
22 redistribution of potential return to power at times
23 other than EOC still needs to be resolved. Those are
24 the ones we talked about earlier for the unclear open
25 items.

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1 MEMBER CORRADINI: And this is the one
2 where you guys had done some separate analysis?

3 MR. SCHMIDT: That's correct.

4 MEMBER CORRADINI: We heard about it in
5 the closed session, in subcommittee.

6 MR. SCHMIDT: That's correct. So and then
7 the staff still had an action to complete,
8 confirmatory analysis. We presented some results that
9 were single bay, so the bay heated up, which can limit
10 the return to power. And the staff is currently
11 running with basically almost in a cool calculation to
12 see what the return to power looks like. Those are
13 running currently.

14 And now I'm going to turn it over to
15 containment.

16 MR. HAIDER: Next slide? My name is Syed
17 Haider. I am going to give a high-level overview of
18 the staff review for NuScale FSAR Section 6.2.1.1 on
19 containment structure which is ameliorated to the peak
20 containment pressure and temperature that is reserved
21 from the limiting containment design, this is events.

22 This slide captures some of the staff
23 review of the NuScale FSAR Section 6.2.1.1 in the
24 containment response analysis and methodology
25 technical report that is incorporated by reference.

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1 At present the Applicant's design and the
2 staff's confirmatory analysis reserves for the
3 limiting containment design (inaudible).

4 As presented during the June 20th ACRS
5 subcommittee meeting, the staff reviewed various
6 conservatisms used in the Applicant's RELAP model. As
7 with its initial boundary conditions to ensure that
8 the containment safety analysis of the NuScale power
9 module was conservative.

10 The staff also performed confirmatory
11 analysis to make sure that NuScale containment design
12 meets the two key regulatory requirements. One
13 requirement that is rooted in GDCs 16 and 15 to ensure
14 that the peak containment pressure and wall
15 temperature can create it for the limiting mass energy
16 of these events are bounded by the NuScale design
17 values with sufficient margin.

18 The second requirement given by GDC-38 for
19 containment (inaudible) is interpreted by the SRP as
20 to ensure that the containment pressure reduces to 50
21 percent of its heat value within 24 hours.

22 So a NuScale application was presented,
23 and we have five modeling (inaudible) for seven
24 containment DBS scenarios that include three LOCA
25 breaks and two anticipated operational occurrences

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1 involving RRV and RVV rads inadvertent of the events;
2 one main steam line break, and one fatal water line
3 break.

4 The staff performed confirmatory analysis,
5 and using the MELCORE code for the same spectrum of
6 seven NPM DBS scenarios, using the primary and
7 secondary mass and energy of this data supplied by the
8 Applicant, which has been the standard practice in the
9 containment design reviews.

10 The confirmatory analyses were performed
11 with the worst case single failure and loss of normal
12 AC and DC power. We evaluated the NuScale pressure
13 and temperature response and confirmed that the
14 required regulatory safety criteria were met by the
15 NuScale containment design.

16 Both the Applicant and staff also ran
17 numerous sensitivity cases of the peak containment
18 pressure and temperature to investigate several key
19 factors such as the condensation heat transfer
20 modeling on the containment inside surface, effective
21 non-condensable gases, release timing, and composition
22 normalization of containment volume and heat
23 structures and liquid (inaudible) effects inside the
24 containment, as well as the cooling pool.

25 In general, the peak containment pressure

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1 and temperature were found to be not much sensitive to
2 these factors to significantly reduce the safety
3 margins.

4 This slide compares the staff MELCORE
5 confirmatory analysis results with NDF-5 results
6 submitted by the Applicant for the limiting
7 containment pressure and temperature design basis
8 events.

9 The limiting pressure event for NPM is an
10 inadvertent opening of area per the circulation valve,
11 while the limiting temperature event is an RCS
12 injection line (inaudible) break.

13 The initiation of each of these events is
14 followed by subsequent ECCS actuation, while the key
15 pressure and temperature occurs after the ECCS was
16 opened. On the graph on the left, the two MELCORs in
17 blue and green are based on an eleven node and a
18 single node containment volume analysis.

19 The graph shows a peak containment
20 pressure of 967 psia, predicted by the staff, using
21 MELCOR versus 986 psia that was predicted by NuScale
22 using NDF-5.

23 Compared to the 1050 psia containment
24 design pressure value, the licensing basis peak
25 pressure value of 986 has six percent margin, while

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1 the staff considers the margin to be sufficient.

2 The staff confirmatory value of 967 psia
3 with about eight percent margin is even lower than the
4 NuScale predicted value of 986 psia.

5 So the graph on the left also shows that
6 the containment pressure drops for 50 percent of the
7 peak value well within an hour after the initiation of
8 the event, which is much earlier than 24 hours that
9 was stipulated by the SRP.

10 On the graph on the right, the MELCOR
11 predicted a peak containment wall temperature of 521
12 degree Fahrenheit that is even lower than the NDF-5
13 predicted value of 526 degrees Fahrenheit for the
14 temperature limiting scenario.

15 So there is a sufficient margin of 24
16 degrees Fahrenheit in the licensing basis value of 526
17 degree Fahrenheit to the containment wall, our
18 temperature design value of 550 degrees Fahrenheit.

19 So I would also like to mention that the
20 staff also performed dependent confirmatory study
21 using the trace code, and the resulting peak
22 temperature and pressure results were even lower than
23 the ones that were predicted by MELCOR.

24 The staff also performed a number of
25 sensitivity studies using the NDF-5 containment safety

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1 analysis texts that were obtained from NuScale.

2 So the containment-related safety findings
3 are informed by three independent computer cores;
4 that's MELCOR, NDF-5, and trace.

5 In summary, the agreement between the
6 NuScale licensing basis calculations and the MELCOR
7 confirmatory analysis results is a reasonable,
8 independent, confirmatory calculation using MELCOR and
9 trace cores show that NuScale containment peak
10 pressure and temperature predicted by Applicant are
11 using NDF-5 are conservative; now SCR Section 6.2.1.1
12 and does not have any unresolved issues regarding the
13 containment response analysis and methodology. They
14 are either resolved, closed, or settled as
15 confirmatory items.

16 However, the SCR section still has several
17 open items related to Chapter 15 and Chapter 3, such
18 as the NIST 1 facility scaling distortion review,
19 experimental validation of NDF-5, and the
20 acceptability of increasing the containment design
21 pressure from 1,000 psia to 1,050 psia.

22 These open items are reviewed under
23 Chapter 3 and LOCA topical report under Chapter 15 and
24 will be resolved there. Once these open items are
25 resolved, the staff will be able to conclude that all

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1 mandatory requirements for the NuScale CSR Section
2 6.2.1.1 have been met.

3 So this pretty much concludes my
4 presentation, and since we don't have any major
5 issues, all open items essentially belong to Chapter
6 15 and Chapter 3, and they are still being reviewed.

7 MEMBER CORRADINI: Questions by the
8 committee? So this is the -- -so, thank you to the
9 staff. Given our starting conditions, we've ended in
10 a flurry.

11 PARTICIPANT: We've got one more slide.

12 MEMBER CORRADINI: I thought --

13 MEMBER KIRCHNER: Not there, Mike.

14 MR. TRAVIS: I'll try to go through this
15 as quickly as possible. There's one exemption related
16 to containment; it's GDC-40 exemption. We've
17 discussed long-term cooling already; I won't discuss
18 the first few bullets.

19 One exemption related to GDC-40 requested
20 by NuScale for containment heat removal; that's
21 testing of the containment heat removal system as the
22 containment heat removal system in this case is the
23 exterior of the pressure vessel submerged in the pool.

24 NuScale requesting an exemption to the
25 testing requirement in GDC-40. Staff reviewed the

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1 exemption request and determined that the periodic
2 inspections that are required by GDC-39, in addition
3 to the other periodic testing and inspections required
4 of other systems that will transfer heat to the
5 containment via the containment heat removal system
6 will satisfy the requirements associated with GDC-40,
7 and therefore the underlying purpose of the rule,
8 which is to verify that the performance
9 characteristics of the containment heat removal system
10 would remain within acceptable parameters to ensure
11 operability. That will still be accomplished, and
12 therefore the plan to recommend to the Commission
13 grant the exemption to GDC-40.

14 There's one open item in 622 related to
15 the performance requirements associated with the
16 containment cleanliness program. Those requirements
17 were deferred to the cleanliness program. Because
18 it's a DC-stage, we cannot make a finding on the
19 cleanliness program itself; that would be for COL. So
20 NuScale explicitly identified the limits in 622. We
21 received that response.

22 MEMBER CORRADINI: This is really to the
23 cleanliness going forward for this --

24 MR. TRAVIS: That's correct. The degree
25 limits associated with the cleanliness program for the

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1 COL are now explicitly identified in the DC as limits
2 rather than --

3 MEMBER CORRADINI: Okay. All right, thank
4 you.

5 MR. TRAVIS: -- allowable values for the
6 (inaudible).

7 MEMBER CORRADINI: Now, excuse me.
8 Questions by the Committee? Okay. So I think at this
9 point we'll take public comments, and then I have some
10 questions about the need for a closed session and what
11 we do in the closed session.

12 So can we open the phone line? Is there
13 anybody left in the room that wants to make a public
14 comment?

15 So the public line should be open. If
16 anybody is out there, can you please acknowledge by
17 speaking up for a moment, please? Anybody on the
18 public line? Anybody on the public line who wants to
19 make a comment?

20 Okay. Hearing nothing, why don't we close
21 the public line, and now let me make a few
22 suggestions. I've spoken to NuScale. They are ready
23 to present things if we have questions on Chapter 3
24 and Chapter 6.

25 And their anticipation is, a lot of the

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1 other proprietary things we might have with them would
2 be covered in two weeks when we're physically visiting
3 Corvallis.

4 But I wanted to give the Committee a
5 chance to ask questions on Chapter 3 and 6, as they
6 were ready, potentially, with slides if you have
7 questions.

8 So I'm looking at you, Bob; missiles? I'm
9 looking at --

10 MEMBER BLEY: I was expecting them to go
11 through the missile discussion when we're out there,
12 so I haven't prepared by studying the last document
13 they sent.

14 MEMBER CORRADINI: Which we now have.

15 MEMBER BLEY: Which we do have.

16 MEMBER CORRADINI: We have.

17 CHAIR RICCARDELLA: There were actually
18 two RAI responses. Did you see that? One was kind of
19 general, and --

20 MEMBER BLEY: We got two; there's actually
21 six or seven.

22 CHAIR RICCARDELLA: Okay.

23 MEMBER CORRADINI: So my question is, so
24 let me just lay out the grand plan. The grand plan
25 is, it's very late. We're going to lose some of the

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1 staff, because there's a train leaving for their
2 potential homes, and they're not going to get anywhere
3 else.

4 So if we want a closed session with the
5 staff, it's got to be tomorrow. If we want a closed
6 session with NuScale, they are ready and willing, but
7 they have stuff only if we have questions. They're
8 not going to come up and start giving us a
9 presentation.

10 MR. MELTON: So for Corvallis, we were
11 anticipating doing turbine missiles here. This is
12 where we wanted to talk about it. I think we would
13 like to present, if at all possible, tomorrow, on
14 turbine missiles.

15 Chapter 6, I think we're okay with going
16 with a closed session on 6. But 3, we do want to
17 engage in 3 and present some more information. So if
18 we can do that tomorrow, that would be super.

19 MEMBER CORRADINI: That's fine.

20 PARTICIPANT: And that will be in closed
21 session.

22 MR. MELTON: Yes. That's closed session,
23 and there's details we'd like to present and talk
24 about.

25 PARTICIPANT: Okay. That sounds good.

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1 MEMBER CORRADINI: I knew that you had
2 something; I didn't know if you wanted to present or
3 wanted to ask questions about.

4 MEMBER REMPE: Do you know that we can't
5 do this at 8:30? We have a PMP that's going to last
6 for a couple of hours before we can do that. Is that
7 okay still, with your plans for travel?

8 MR. MELTON: We'll be fine with that.
9 We're good with that, so later in the morning is good.

10 MS. KARAS: And the Chapter 15 staff, I
11 think we're good until about 7:15, if you have
12 questions.

13 (Whereupon, the above-entitled matter went
14 off the record at 6:44 p.m. and resumed at 6:45 p.m.)

15 MEMBER CORRADINI: Well, I haven't
16 finished. Do we have questions of the staff on 3 or
17 6 that we want to drag them up in closed session now,
18 since we have 30 minutes before we're going to lose
19 them, or do we want to wait for tomorrow?

20 CHAIR RICCARDELLA: On 3, shouldn't the
21 staff go after NuScale?

22 PARTICIPANT: Right.

23 MEMBER BLEY: Well, the staff has told us
24 all they know on the --

25 MEMBER CORRADINI: The staff is going to

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1 review what they've got --

2 MR. COYNE: They haven't reviewed it all
3 yet, for missiles.

4 MEMBER BLEY: I don't think you're looking
5 to tell us anything more on missiles.

6 MR. MELTON: So our turbine missile
7 analysis staff are not here. But they just recently
8 got an information submittal from NuScale. They've
9 done a preliminary review. They have questions --

10 MEMBER BLEY: And they told us about that.
11 I don't think they have anything more to tell us at
12 this time.

13 MEMBER CORRADINI: So unless I hear great
14 objections, we can either do, in the next 30 minutes,
15 before I start losing a lot of people, we can do a
16 closed session on topics on Chapters 3 and 6 if the
17 members want, or we wait until tomorrow at 10:15.

18 Because I have another assignment for the
19 members now, and that's to take a look at the letter,
20 because we have it ready. We thought we were going to
21 get to it, but I have it printed and ready, based on
22 input by Dennis -- well, hang on. I'm just trying to
23 tell everybody.

24 So what's your pleasure? My thought is to
25 let go everybody here unless they want to watch us

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1 make sausage.

2 PARTICIPANT: I think that's a good idea.

3 MEMBER CORRADINI: All right. And we
4 would pick it up tomorrow in closed session, because
5 then we're going to get some of the input from the
6 members tonight and have a different letter, or
7 hopefully not a too different letter by tomorrow.

8 MEMBER REMPE: You're planning on doing an
9 entire read-through starting at 6:45?

10 MEMBER CORRADINI: I'm asking.

11 MEMBER REMPE: Okay. But that's what
12 you're suggesting.

13 MEMBER CORRADINI: That's what I'm --

14 CHAIR RICCARDELLA: Yes, why don't we just
15 read it? Yes.

16 MEMBER MARCH-LEUBA: You'd have to do this
17 tomorrow. It's mandatory to read it. It's part of --

18 MEMBER BROWN: He's a speed reader; don't
19 worry. You won't understand a word he says.

20 MEMBER REMPE: I think we could do a read-
21 through tomorrow. It's 6:45 --

22 MEMBER CORRADINI: Fine. So you want to
23 get a copy of the letter tonight?

24 (Simultaneous speaking.)

25 MEMBER CORRADINI: I've gotten changes

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1 through the day on a continuing basis. So we're now
2 at version 7. I've got input from Dennis and Jose and
3 Joy and Dick and somebody else; I can't remember who.

4 MEMBER BROWN: Can we get a printed copy
5 now?

6 MEMBER CORRADINI: Yes, and you can take
7 it home and read it. I've read it a number of times.
8 I'm sick of looking at it.

9 PARTICIPANT: Let's have it.

10 MEMBER BROWN: I won't read it until I get
11 it my hot little hands.

12 MEMBER CORRADINI: So at this point I'm
13 going to turn this over to the Chairman.

14 CHAIR RICCARDELLA: Okay. So we are in
15 recess until 8:30 tomorrow morning.

16 (Whereupon, the above-entitled matter went
17 off the record at 6:48 p.m.)

18

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July 9, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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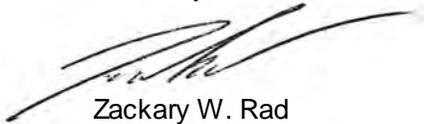
SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Full Committee Presentation: NuScale Topical Report-Evaluation Methodology for Stability Analysis of the NuScale Power Module," PM-0719-66233, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) NuScale Full Committee Meeting on July 10, 2019. The materials support NuScale's presentation on the "Evaluation Methodology for Stability Analysis of the NuScale Power Module" topical report.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Matthew Presson at 541-452-7531 or at mpresson@nuscalepower.com.

Sincerely,



Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Robert Taylor, NRC, OWFN-7H4
Michael Snodderly, NRC, TWFN-2E26
Samuel Lee, NRC, OWFN-8H12
Gregory Cranston, NRC, OWFN-8H12
Bruce Baval, NRC, OWFN-8H12

Enclosure: ACRS Full Committee Presentation: NuScale Topical Report-Evaluation Methodology for Stability Analysis of the NuScale Power Module, PM-0719-66233, Revision 0

Enclosure:

ACRS Full Committee Presentation: NuScale Topical Report-Evaluation Methodology for Stability Analysis of the NuScale Power Module, PM-0719-66233, Revision 0

ACRS Full Committee Presentation

NuScale Topical Report Evaluation Methodology for Stability Analysis of the NuScale Power Module

July 10, 2019



Presenters

Dr. Yousef Farawila (*)
System Thermal Hydraulics

Matthew Presson
Licensing Project Manager

(*) On the phone

Agenda

- Introduction and Main Message
- Stability Solution Type
- Stability Investigation Description
 - Theoretical
 - Numerical Using New Code PIM
 - Experimental Benchmark
- Procedure and Methodology
- Summary
- Questions and Discussions

The Main Message

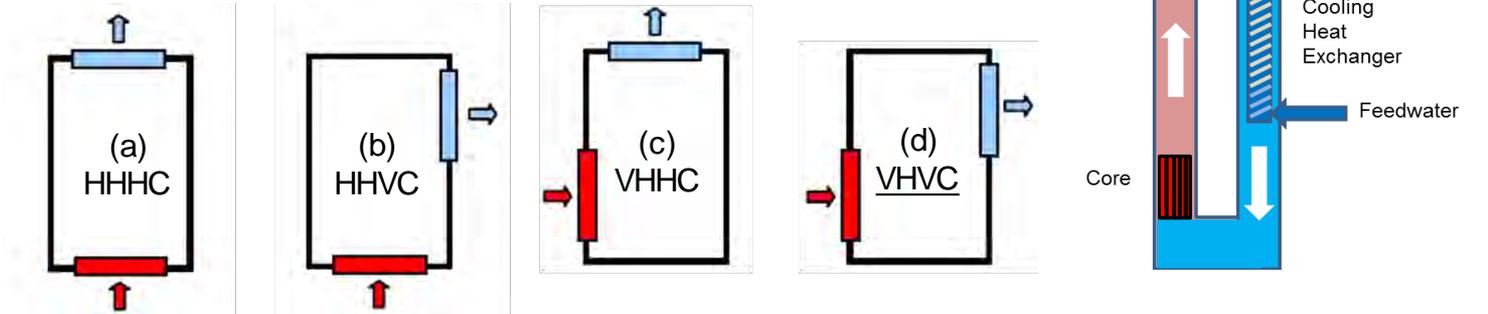
- The NuScale power module design was found to be stable in the entire range of normal operation
- Outside of normal operation, the reactor is destabilized when the riser flow is voided, however
 - Unstable flow oscillation amplitude is limited by nonlinear effects and the critical heat flux ratio actually improves
- The stability threshold is protected by scram upon loss of riser inlet subcooling
 - Conceptually equivalent to a “region exclusion” not a “detect and suppress” solution type
 - No action required to implement a stability solution hardware
- These conclusions are based on extensive first principles, experimental, and computational studies. Details next.

Stability Evaluation

- Natural circulation instabilities were reported in literature

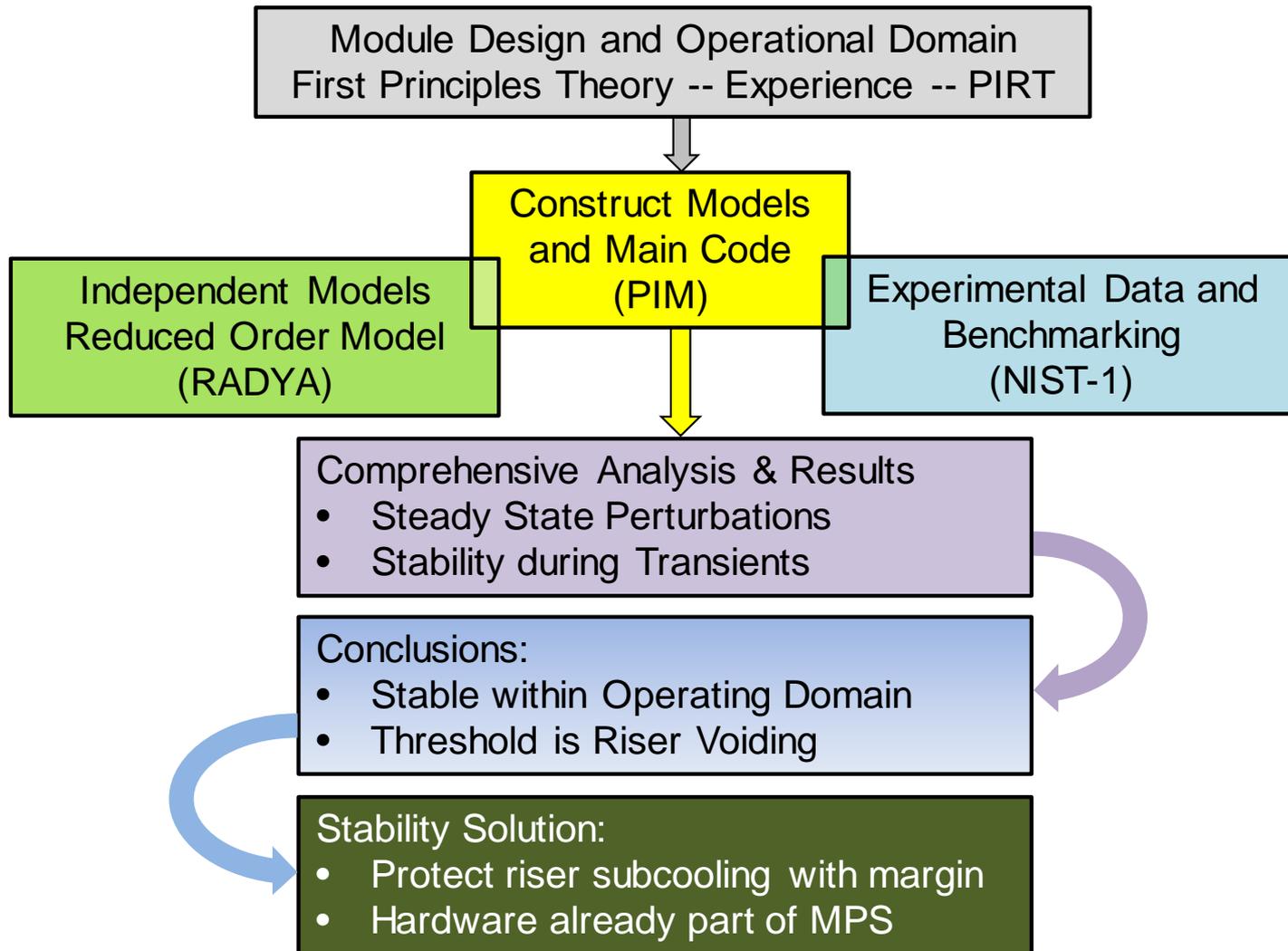
- See for example D.S. Pilkhwal et al., "Analysis of the unstable behaviour of a single-phase natural circulation loop with one-dimensional and computational fluid-dynamic models," *Annals of Nuclear Energy* 34 (2007) 339–355.

- a) HHHC: horizontal heater and horizontal cooler (the only unstable configuration);
- b) HHVC: horizontal heater and vertical cooler;
- c) VHHC: vertical heater and horizontal cooler;
- d) VHVC: vertical heater and vertical cooler (qualitatively like NuScale module)



- Investigation of the NuScale module stability commenced to demonstrate stability, identify threshold conditions, and license stability protection methodology

Stability Investigation Elements



Theoretical Investigation

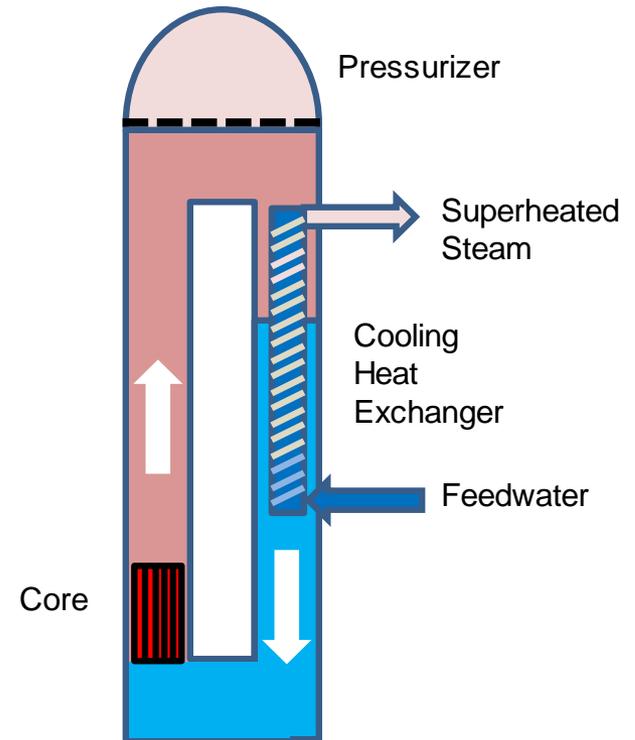
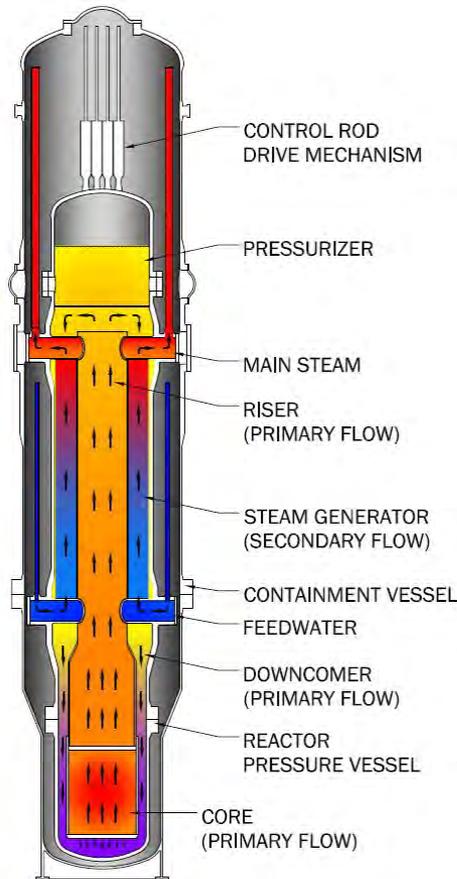
- Kick off with an expert committee to generate a first PIRT
- Scoping review of thermalhydraulic instability modes and contrasting with the NPM design features
- Identification of the possible instability mechanism
- Analysis from first principles
 - Riser-only mode (separate from cold leg)
 - Stability trend with power using a simple SG model
 - Shows decoupling of possible oscillations in SG tubes from primary flow and core
 - Informs design of stability experiments
- All medium ranked phenomena treated as highly ranked

Theoretical and First Principles

- A system with feedback processes may undergo oscillatory instability if the feedback is:
 - Negative (positive feedback is unconditionally unstable)
 - Delayed
 - Sufficiently strong
- NuScale natural circulation mode is examined
 - Feedback is negative. A perturbation increasing core flow decreases exit temperature thus decreases riser density head
 - Feedback is delayed. Transport delay for core exit condition to fill the riser and reach maximum density head effect.
 - Feedback strength is related to liquid thermal expansion and possibility of phase change, riser length, SG characteristics, reactivity feedback... Requires detailed modeling

Main Stability Analysis Tool: PIM

- Transient 1-D 2-phase non-equilibrium primary loop flow



Model Equations of the PIM code

- Thermalhydraulic conservation equations

- Liquid and vapor mass balance

$$\frac{dM_{l,n}}{dt} = \dot{m}_{l,n-1} - \dot{m}_{l,n} - \Gamma_n$$

$$\frac{dM_{g,n}}{dt} = \dot{m}_{g,n-1} - \dot{m}_{g,n} + \Gamma_n$$

- Mixture momentum conservation with drift flux
(integrated momentum)

$$\frac{dI}{dt} = \Delta P_{grav} - \Delta P_{friction} - \Delta P_{local} + \Delta P_{resid}$$

- Energy conservation (assume saturated vapor)

$$\frac{d}{dt} (M_{l,n} h_{l,n}) = \dot{m}_{l,n-1} h_{l,n-1} - \dot{m}_{l,n} h_{l,n} - \Gamma_n h_{fg} + \dot{Q}_n$$

t	time
M_l	liquid mass
M_g	vapor mass
\dot{m}_l	liquid mass flow rate
\dot{m}_g	vapor mass flow rate
Γ	rate of evaporation
I	integrated momentum
ΔP_{grav}	gravitational press. drop
$\Delta P_{friction}$	friction pressure drop
ΔP_{local}	local pressure drop
ΔP_{resid}	residual pressure drop
h_l	liquid enthalpy
h_{fg}	latent heat
Q	power
n	control volume index
$n-1$	upstream index

Model Equations of the PIM code

- Point Nuclear Kinetics

$$\Lambda \frac{d\Phi}{dt} = \beta(\rho - 1)\Phi + \lambda C$$

$$\frac{dC}{dt} = \beta\Phi - \lambda C$$

C	Concentration of the delayed neutron precursors
λ	Decay constant of the delayed neutron precursors
Φ	Neutron flux amplitude
β	Delayed neutron fraction
Λ	Prompt neutron lifetime
ρ	Reactivity

- Thermalhydraulic model provides reactivity input
 - Moderator density reactivity feedback model (equivalent to moderator temperature reactivity under single-phase flow)
 - Doppler fuel temperature reactivity feedback
- Heat source from neutron kinetics feeds back to thermalhydraulics
 - Energy deposited in fuel pellets (proportional to neutron flux)
 - Fraction of fission energy deposited directly in coolant
 - Decay heat: input by the user as fraction of initial power

Model Equations of the PIM code

- Heat conduction in fuel rods
 - Pellet conductivity is function of temperature and burnup
 - Driven by energy deposited in fuel pellets
 - Heat flux at outer rod surface as power source to coolant
 - Pellet temperature needed for Doppler reactivity
- Heat transfer models for heat sink (steam generator)
 - Secondary side flow is driven by user-provided inlet forcing function
 - Secondary flow is subcooled, 2-phase equilibrium, and superheated
 - Primary flow parameters calculated from transient conservation equations
 - Heat transfer between primary and secondary flow
 - Heat transfer correlations
 - Transient heat conduction in tube walls

Model Equations of the PIM code

- Closing Relations and Correlations
 - Frictional pressure drop (single- and two-phase friction and local losses)
 - Drift flux parameters
 - Non-equilibrium evaporation and condensation model
 - Thermodynamic properties for water
 - Physical material properties (pellets, cladding, SG tubes)
 - Pellet-clad gap conductance
 - Reactivity coefficients as functions of exposure and moderator density
- What is not modeled
 - Pressurizer; pressure is input provided constant or forcing function
 - Heat transfer through riser wall, adiabatic riser is default option
 - Heat capacity of structures; only ambient heat losses through vessel

Flow Stability in Steam Generator Tubes

- Secondary side flow changes from single-phase liquid, to two-phase mixture, ultimately to superheated steam
- Flow in the tubes is subject to density wave instability
- Experiments demonstrate flow oscillations under certain conditions (generally low flow)
 - Oscillations in different tubes are not phase-locked
 - Effect on primary flow cancels out, confirming first-principle finding
- No impact on the primary flow and core stability
 - No feedback loop between primary and secondary oscillations

PIM Results of Perturbing Steady State

- Purpose is to calculate stability parameters of decay ratio and period at different conditions of power and exposure
 - Following a user-applied small perturbation flow will oscillate
 - Oscillations will grow with time if system is unstable
 - Oscillations will decay eventually returning to the pre-perturbation state if the system is stable
- Stability parameters, decay ratio and period, are extracted from the transient output. Observations:
 - Unconditional stability in the entire operational range
 - DR decreases with power and exposure
 - Period also decreases with power
 - Observations agree with the independent Reduced Order Model

PIM Application Methodology

- For perturbations of steady state to get DR
 - Vary power within 5-100% of rated
 - BOC and EOC, and any point in between if warranted
 - Conservative assumptions for MTC and decay heat fraction
- For a depressurization transient (scram not credited)
 - Verify that unstable oscillations limit cycle without CHF decrease
- Stability conclusion is generic, but confirmation is needed
 - For plant upgrades such as power uprates
 - Plant operation changes such as operating temperatures and maximum boron concentration
 - Changes in fuel design that would change natural circulation flow

Long Term Stability Solution

- Region Exclusion for NuScale
 - Unstable region defined by a single parameter (core exit subcooling)
 - Monitor and protect margin to riser exit subcooling (with temperature margin below saturation point at pressurizer pressure)
 - Operator alarm when subcooling margin is approached
 - Riser exit subcooling will be controlled by the reactor protection system as part of normal operating limits – not only for preventing instabilities
 - Generic solution: there are no fuel or cycle design elements

Summary and Conclusions

- Stability of the NuScale module was evaluated using a dedicated code (PIM) and supported by first principles analysis and experimental data benchmarking
- The module was found unconditionally stable within normal operation domain using conservative criterion
- Stability boundary identified as associated with riser voiding (loss of riser inlet subcooling)
- Stability protection methodology protects riser inlet subcooling with a margin to define the exclusion region enforced by the module protection system with scram

Portland Office

6650 SW Redwood Lane,
Suite 210
Portland, OR 97224
971.371.1592

Corvallis Office

1100 NE Circle Blvd., Suite 200
Corvallis, OR 97330
541.360.0500

Rockville Office

11333 Woodglen Ave., Suite 205
Rockville, MD 20852
301.770.0472

Charlotte Office

2815 Coliseum Centre Drive,
Suite 230
Charlotte, NC 28217
980.349.4804

Richland Office

1933 Jadwin Ave., Suite 130
Richland, WA 99354
541.360.0500

Arlington Office

2300 Clarendon Blvd., Suite 1110
Arlington, VA 22201

London Office

1st Floor Portland House
Bressenden Place
London SW1E 5BH
United Kingdom
+44 (0) 2079 321700

<http://www.nuscalepower.com>

 [Twitter: @NuScale_Power](https://twitter.com/NuScale_Power)



Presentation to the ACRS Full Committee
Staff Review of NuScale Topical Report

TR-0516-49417-P, REVISION 0
“EVALUATION METHODOLOGY FOR STABILITY ANALYSIS
OF THE
NUSCALE POWER MODULE”

Presenters:

Peter Yarsky, Ph.D.- Senior Reactor Systems Engineer, RES
Ray Skarda, Ph.D.- Reactor Systems Engineer, RES
Bruce Baval - Project Manager, Office of New Reactors

July 10, 2019
(Open Session)

NRC Technical Review Areas/Contributors

- **Ray Skarda** - RES/Division of Systems Analysis (DSA)/Code and Reactor Analysis Branch (CRAB)
- **Peter Yarsky** - RES/DSA/CRAB
- **Rebecca Karas** (BC) - NRO/Division of Engineering, Safety Systems and Risk Assessment (DESR)/Reactor Systems, Nuclear Performance, and Code Review Branch (SRSB)

Outline

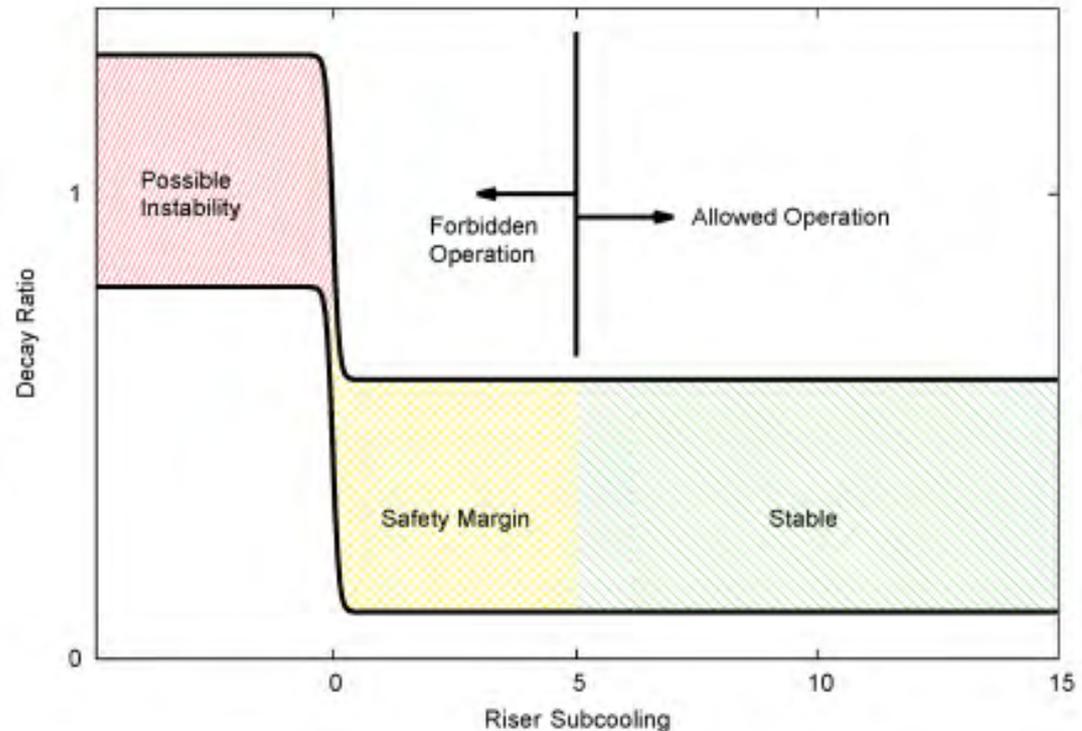
Primary Review Areas

- Regulatory Criteria
- Long Term Stability Solution
- Instability Modes and Phenomenology
- PIM Evaluation Model
- Impact of Secondary Side Instabilities on Primary Side Stability
- Other Secondary Side Stability Concerns
- Uncertainty and Acceptance Criteria
- Stability with Worst-Rod-Stuck-Out (WRSO)
- Stability Topical Report Conclusions
- Design Certification Document (DCD) 15.9 Stability

Long Term Stability Solution (LTSS)

Exclusion Region Based Solution

- The NuScale LTSS is based on an exclusion region principle. GDC 12 and GDC 10 are met by preventing instabilities that could challenge specified acceptable fuel design limits (SAFDLs).
- The Module Protection System (MPS) precludes instability by enforcing riser subcooling margin and tripping the reactor. GDC 13, GDC 20, and GDC 29 are met by operation of the MPS to sense adverse conditions and trip the reactor.



PIM Evaluation Model

The PIM Evaluation Model is Simple but Acceptable

- The PIM evaluation model includes simple models for thermal-hydraulics, reactor kinetics, fuel thermal-mechanical response, and steam generator tube heat conduction and heat transfer.
- Integral validation provided against NIST-1 stability tests.

Decay Ratio (DR) Acceptance Criterion

- DR is insensitive to variations in most of the important phenomena over the PIM application range.
- DR Acceptance Criterion affords sufficient margin to account for bias and uncertainty.
- Numerical effects were considered as part of the DR bias.

ACRS SC Follow-up on SG Secondary Side Instability

Secondary side oscillations are not safety significant with respect to reactor (primary side) stability

- Staff finds that secondary side flow oscillations will not challenge the thermal margins.
- Applicant's analysis demonstrates that severe secondary flow oscillations would result in a MPS protective trip on reactor power or superheat.
- Analyses performed by the applicant demonstrate flow oscillations that are not safety significant.
 - Chapter 3 technical staff will discuss SG thermal fatigue concerns as part of the DCD Chapter 3 review.

Stability Topical Report Conclusions

PIM-based Stability Analysis Method is Acceptable

- PIM is a simple model, but its models are anchored to upstream, high-fidelity models to improve accuracy.
- The DR is highly insensitive to variations in important phenomena and their models, leading to relatively small uncertainty in the DR.
- PIM predictions in steady-state and transients have been confirmed by the staff with independent TRACE confirmatory calculations.
- PIM is acceptable for performing stability analysis for the NuScale power module.

Stability Topical Report Conclusions

LTS Solution is Acceptable

- Primary instability mechanism properly identified by the applicant and confirmed by independent staff TRACE analysis.
- During normal, at power operation, the NuScale power module is very stable.
- The exclusion region based LTS solution is effective in preventing the reactor from becoming unstable during normal operation including the effects of AOOs.
- Potential instability during return to power with WRSO is not a safety concern.
- GDCs 10, 12, 13, 20, and 29 are met.

Questions?

Backup Slides

Staff Review Timeline

- NuScale submitted the Topical Report (TR) TR-0516-49417-P, “Evaluation Methodology for Stability Analysis of the NuScale Power Module,” on July 31, 2016, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16250A851). Applicant provided supplemental information by letter dated December 3, 2016 (ADAMS Accession No. ML16340A756).
- Staff issued 62 requests for additional information (RAIs) with NuScale providing responses – all responses were resolved/closed.
- Staff plans to issue its final SER in late August 2019.
- Staff plans to publish the “-A” (approved) version of the TR in late November 2019.

Regulatory Criteria

General Design Criteria (GDCs) from Design-specific Review Standard 15.9.A

- GDC 10, “Reactor Design,” requires that specified acceptable fuel design limits (SAFDL) not be exceeded during any condition of normal operation, including conditions that result in unstable power oscillations with the reactor trip system available.
- GDC 12, “Suppression of Reactor Power Oscillations,” requires that oscillations be either not possible or reliably and readily detected and suppressed.
- GDC 13, “Instrumentation and Control,” includes requirements for the hardware implementation of long term stability (LTS) solution.
- GDC 20, “Protection System Functions,” requires the reactor protection system to initiate automatic action so unstable power oscillations are avoided.
- GDC 29, “Protection against Anticipated Operational Occurrences,” requires stability LTS solution design for an extremely high probability of accomplishing safety functions.

Instability Modes

- Dynamic and static instability modes were considered.
- Applicant identified and evaluated many modes.
- The applicant's findings in terms of modes are consistent with staff findings from an independent Phenomena Identification and Ranking Table development process.

Stability with WRSO

NuScale is stable at intermediate pressures

- Applicant conservatively analyzed stability for intermediate pressures (i.e, before emergency-core-cooling-system (ECCS) actuation).
- Strong moderator feedback increases likelihood of recriticality with WRSO, but strong moderator feedback is stabilizing.
- Applicant's analysis demonstrates stability margin at intermediate pressures.

Stability with WRSO

NuScale will experience mild flow oscillation at low pressure

- After ECCS actuation, level drops below the riser, natural circulation flow pattern is broken and flow oscillations occur where core flow is driven by density head differences provided by void formation in the core region.
- Analyses performed by the applicant demonstrate flow oscillations that are not safety significant.

DCD 15.9 Stability Review

Stability Performance during Steady State Conditions is Acceptable

- Stable under steady-state conditions
 - Analyses demonstrate at all power levels > 5 percent of rated that the DR remains well below the acceptance criterion.
- Transients analyzed
 - Certain events result in new stable, steady state conditions.
 - Certain events result in reactor trip due to MPS enforcement of the exclusion region prior to the onset of instability.

DCD 15.9 Stability Review

Stability Performance during Transients is Acceptable

- All AOO classes considered in the applicant's analysis.
 - Increase in heat removal by the secondary system
 - Decrease in heat removal by the secondary system
 - Decrease in reactor coolant system flow rate
 - Increase in reactor coolant inventory
 - Reactivity and power distribution anomalies
 - Decrease in reactor coolant inventory
- LTS Solution is effective in preventing the occurrence of instability.
- Therefore GDCs 10, 12, 13, 20, and 29 are met.

DCD 15.9 Stability Review

Increase in Heat Removal by the Secondary System

- Analysis consistent with the stability analysis methodology topical report.
- Applicant analyzed maximum feed flow increase that does not produce an automatic, prompt MPS trip.
- PIM calculations confirm that the reactor remains stable

DCD 15.9 Stability Review

Decrease in Heat Removal by the Secondary System

- The staff reviewed the feedwater (FW) flow reduction event analyzed in the DCA.
- The DCA demonstrates that even mild FW flow events will progress in similar manners, eventually leading to a MPS trip based on hot-leg temperature (i.e., the LTSS).
- Therefore, the staff finds that the LTSS is effective in preventing the reactor from reaching an unstable condition.
- The staff review of the DCA analysis does not impact the staff review of the Stability TR.

DCD 15.9 Stability Review

Decrease in Reactor Coolant System Flow Rate

- CVCS pump over-speed would be considered an AOO.
- The staff found that a CVCS flow reduction or a reduction in secondary side heat removal event would bound this class of events.
- Analysis of CVCS pump over-speed is not required to demonstrate compliance with GDC 12.

DCD 15.9 Stability Review

Increase in Reactor Coolant Inventory

- Events that increase inventory but also increase reactor coolant system pressure are non-limiting because higher pressure increases stability margin.
- Staff considered events such as CVCS or spray malfunction that could increase inventory but maintain pressure, and these would be bounded by events that increase or decrease secondary side heat removal.
- Analysis of this class of event is not required to demonstrate compliance with GDC 12.

DCD 15.9 Stability Review

Reactivity and Power Distribution Anomalies

- A bounding event is analyzed based on a maximum control rod withdrawal that does not produce an automatic, prompt MPS trip based on high flux or high flux rate.
- PIM calculations confirm that the reactor remains stable.

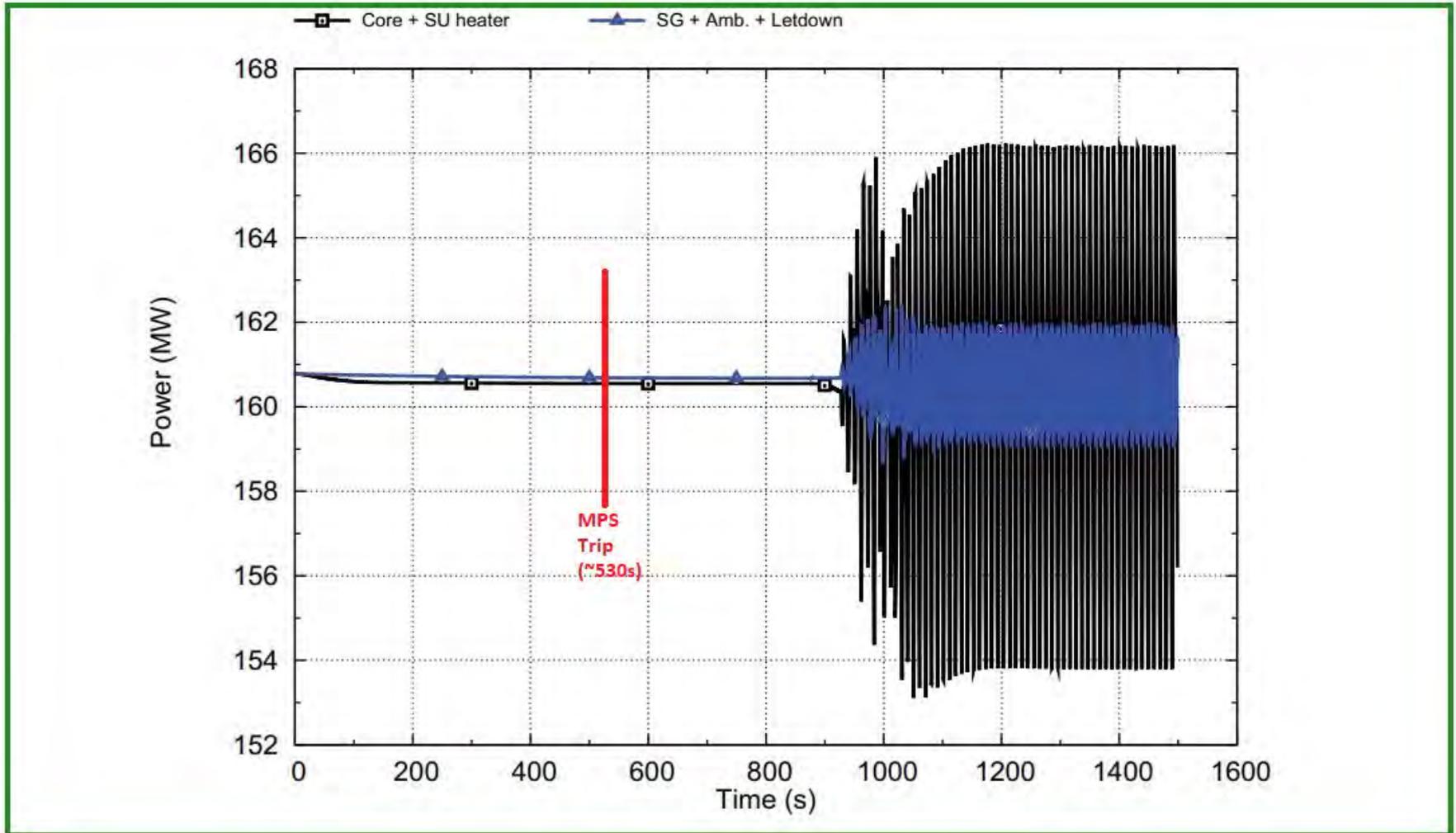
DCD 15.9 Stability Review

Decrease in Reactor Coolant Inventory

- The limiting event is a slow depressurization where the low pressure trip is not credited.
- PIM calculations show that, eventually, the reduction in pressure leads to low riser subcooling, which initiated the LTS solution MPS trip to enforce the exclusion region.
- PIM calculations demonstrate that the onset of instability would occur well after the reactor is shutdown by control rods.

DCD 15.9 Stability Review

LTS Solution MPS Trip Timing



July 9, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Titled “ACRS NuScale Full Committee Presentation: NuScale FSAR Chapter 3, Design of Structures, Systems, Components and Equipment,” PM-0719-66231, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on reactor Safeguards (ACRS) NuScale Full Committee Meeting open session on July 10, 2019. The materials support NuScale’s presentation of Chapter 3, “Design of Structures, Systems, Components and Equipment,” of the NuScale Design Certification Application.

The enclosure to this letter is the nonproprietary version of the presentation titled “ACRS NuScale Full Committee Presentation: NuScale FSAR Chapter 3, Design of Structures, Systems, Components and Equipment,” PM-0719-66231, Revision 0.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Marty Bryan at 541-452-7172 or at mbryan@nuscalepower.com.

Sincerely,



Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Robert Taylor, NRC, OWFN-8H12
Michael Snodderly, NRC, OWFN-8H12
Samuel Lee, NRC, OWFN-8H12
Gregory Cranston, NRC, OWFN-8H12
Marieliz Vera, NRC, OWFN-8H12

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NuScale Nonproprietary

ACRS Full Committee Presentation

NuScale FSAR Chapter 3

Design of Structures, Systems,
Components and Equipment



July 10, 2019

PM-0719-66231
Revision: 0

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Presenters

Marty Bryan

Licensing Project Manager

Amber Berger

Civil/Structural Engineer

Olivia Hand

Mechanical Engineer

Hannah Rooks

Supervisor, NSSS Systems Analysis

Turbine Missiles

ACRS Full Meeting on Turbine Missiles

- Illustrations of turbine missile trajectory and barriers
- Turbine missile barrier analyses details
 - Local analyses
 - Global analysis
- Turbine missile barrier results
- Conservatisms in NuScale turbine missile barrier approach

CVAP and TF-3

CVAP and TF-3 Discussion

- CVAP with particular emphasis on the purpose of TF-3 testing, including any possible feedback from recent testing:
 - TF-3 is a 5-column, full height, prototypic support test specimen. Modal testing (both in-air and in-water) and flow testing (unheated, primary side flow only) is planned.
 - The purpose of TF-3 testing is to validate the frequencies, mode shapes, and damping values used in the design analyses. The flow testing will provide the critical velocities and RMS vibration amplitudes of the SG tube bundle due to primary coolant flow. The testing will validate the safety margins for vortex shedding, fluid elastic instability, turbulence
 - Preliminary modal testing has shown good agreement with design analysis frequencies and damping.

SG DWO

SG DWO – FSAR Design Work

- Design objective for SG is to withstand all loadings for design life of the component.
- NRELAP5 SG model compared to TF-1 and TF-2 test data for predicting onset of density wave oscillation.
- DWO frequency much lower than natural frequencies of SG components. No resonance/FIV concerns.
- SG stability analysis shows limited flow oscillations due to DWO at low power operating conditions.
- Design objective for SG inlet flow restrictor (IFR) is to produce necessary pressure drop in the tube subcooled region to limit oscillation magnitudes to acceptable limits.
- Revision to FSAR Section 5.4.1.2 to clarify terminology; the design intent of the IFR is not to fully preclude DWO, but to ensure oscillations are within acceptable limits.
- DWO stability map based on TF-1/TF-2 data is under development for comparison with stability map developed by NRC.

SG DWO – ITAAC Closure Activities

- The SG tubes, tube-to-tubesheet welds, and the tubesheet are ASME Class 1 components evaluated to the rules of Subsection NB. These components will be included in an ASME Design Report that will be reviewed by the NRC. These actions are being tracked as part of ITAAC closure.
- SG FW plenum analysis is under NRC review.
 - The tubes and tube-to-tubesheet welds were not included and DWO loading was not considered, however maximum resultant usage in the vicinity of the tube-to-tubesheet welds is very low.
- SG tube and tube-to-tubesheet weld fatigue analyses will be completed; will consider all transient and DWO loading, and will confirm if IFR sizing is adequate.

Acronyms

AR	Acoustic Resonance	MCS	Module Control System
BC	Boundary Condition	NDRC	National Defense Research Council
CNT	Containment	RMS	Root Mean Square
CRB	Control Building	RXB	Reactor Building
CVAP	Comprehensive Vibration Assessment Program	SG	Steam Generator
DCR	Demand to Capacity Ratio	SSC	Structures, Systems, and Components
DWO	Density Wave Oscillation	TGB	Turbine Generator Building
FEI	Fluid Elastic Instability	VS	Vortex Shedding
FIV	Flow Induced Vibration		
FW	Feedwater		
FEM	Finite Element Model		
IFR	Inlet Flow Restrictor		
ITAAC	Inspections, Tests, Analysis, and Acceptance Criteria		

Portland Office

6650 SW Redwood Lane,
Suite 210
Portland, OR 97224
971.371.1592

Corvallis Office

1100 NE Circle Blvd., Suite 200
Corvallis, OR 97330
541.360.0500

Rockville Office

11333 Woodglen Ave., Suite 205
Rockville, MD 20852
301.770.0472

Charlotte Office

2815 Coliseum Centre Drive,
Suite 230
Charlotte, NC 28217
980.349.4804

Richland Office

1933 Jadwin Ave., Suite 130
Richland, WA 99354
541.360.0500

Arlington Office

2300 Clarendon Blvd., Suite 1110
Arlington, VA 22201

London Office

1st Floor Portland House
Bressenden Place
London SW1E 5BH
United Kingdom
+44 (0) 2079 321700

<http://www.nuscalepower.com>

 [Twitter: @NuScale_Power](https://twitter.com/NuScale_Power)



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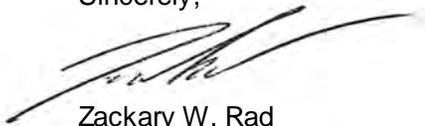
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Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Robert Taylor, NRC, OWFN-7H4
Michael Snodderly, NRC, TWFN-2E26
Samuel Lee, NRC, OWFN-8H12
Gregory Cranston, NRC, OWFN-8H12
Rani Franovich, NRC, OWFN-8H12

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ACRS Full Committee Presentation

NuScale FSAR

Chapter 15

Transient and Accident Analyses



July 10, 2019

Presentation Team

- Dr. Pravin Sawant (*)** - Supervisor, Code Validation and Methods
- Dr. Brian Wolf (*)** - Supervisor, Code Development
- Dr. Selim Kuran (*)** - Thermal Hydraulic Software Validation
- Meghan McCloskey (*)** - Supervisor, System Thermal Hydraulics
- Mark Shaver (*)** - Supervisor, Radiological Engineering
- Matthew Presson** - Licensing Project Manager

(*) On the phone

Scope of Chapter 15

Evaluation of the safety of a nuclear power plant requires analyses of the plant's responses to postulated equipment failures or malfunctions

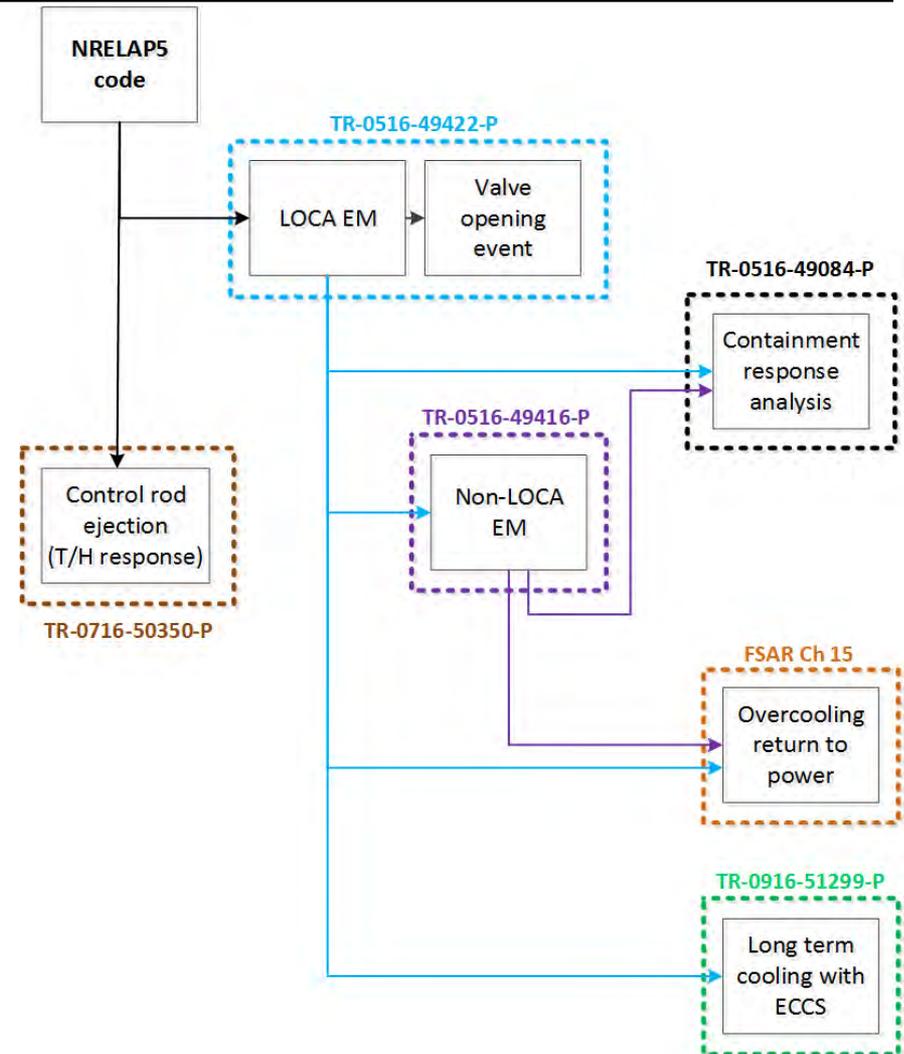
- FSAR Ch 15 addresses deterministic design basis safety analyses
- FSAR Ch 15 summarizes results of the NuScale design basis events identified, event classification, methodology for analysis, event results and margin to acceptance criteria
- FSAR Ch 15 provides results demonstrating radiological dose consequences remain below acceptance criteria

Chapter 15 Overview Agenda – July 10th

- NuScale design overview
- Ch 15 overview and analytical assumptions
- Limiting case results
- Radiological analysis
- Ch 6.2.1 Containment response analysis limiting results
- Long term cooling

System T/H Analysis Basis

- NRELAP5 code developed from RELAP5-3D
 - Modified to address NuScale-specific phenomena/systems
- LOCA Evaluation Model (EM) developed following RG 1.203 EMDAP
 - LOCA EM extended to derive EMs for other events as shown in this figure.
- Additional supporting EMs include
 - Nuclear Analysis Codes – TR-0716-50350-P-A
 - Critical Heat Flux – TR-0116-21012-P-A
 - Subchannel Analysis – TR-0915-17564-P-A
- Additionally, the NuScale design implements an exemption to GDC 27 to account for its passive assumption - No operator actions are required to achieve safety functions for 72 hours under Chapter 15 postulated conditions.
 - In implementing this exemption, the NuScale DCA assures fuel cladding integrity for all design basis events, including under postulated accident conditions.



Chapter 15 Radiological Dose Consequences

- Doses remain below applicable limits for all events
- Methods for determining dose consequences consistent with RG 1.183
- Design Basis Accident Radiological Consequences – TR-0915-17565
- Core Damage Event (CDE) Radiological Consequences – TR-0915-17565

EAB: Exclusion Area Boundary
 LPZ: Low Population Zone
 CR: Control Room

Event	Location	Acceptance Criteria (rem TEDE)	Dose (rem TEDE)
Iodine Spike Design Basis Source Term ⁽¹⁾ (pre-incident iodine spike)	EAB	25.0	<0.01
	LPZ	25.0	<0.01
	CR	5.0	<0.01
Iodine Spike Design-Basis Source Term ⁽¹⁾ (coincident iodine spike)	EAB	25.0	<0.01
	LPZ	25.0	<0.01
	CR	5.0	<0.01
Core Damage Event ⁽²⁾	EAB	25.0	0.63
	LPZ	25.0	1.37
	CR	5.0	2.14
Main Steam Line Break (pre-incident iodine spike)	EAB	25.0	<0.01
	LPZ	25.0	<0.01
	CR	5.0	0.01
Main Steam Line Break (coincident iodine spike)	EAB	2.5	<0.01
	LPZ	2.5	<0.01
	CR	5.0	<0.01
Steam generator tube failure (pre-incident iodine spike)	EAB	25.0	0.08
	LPZ	25.0	0.08
	CR	5.0	0.20
Steam generator tube failure (coincident iodine spike)	EAB	2.5	<0.01
	LPZ	2.5	<0.01
	CR	5.0	<0.01
Primary coolant line break	EAB	6.3	0.02
	LPZ	6.3	0.04
	CR	5.0	0.08
Fuel handling accident	EAB	6.3	0.55
	LPZ	6.3	0.55
	CR	5.0	0.89

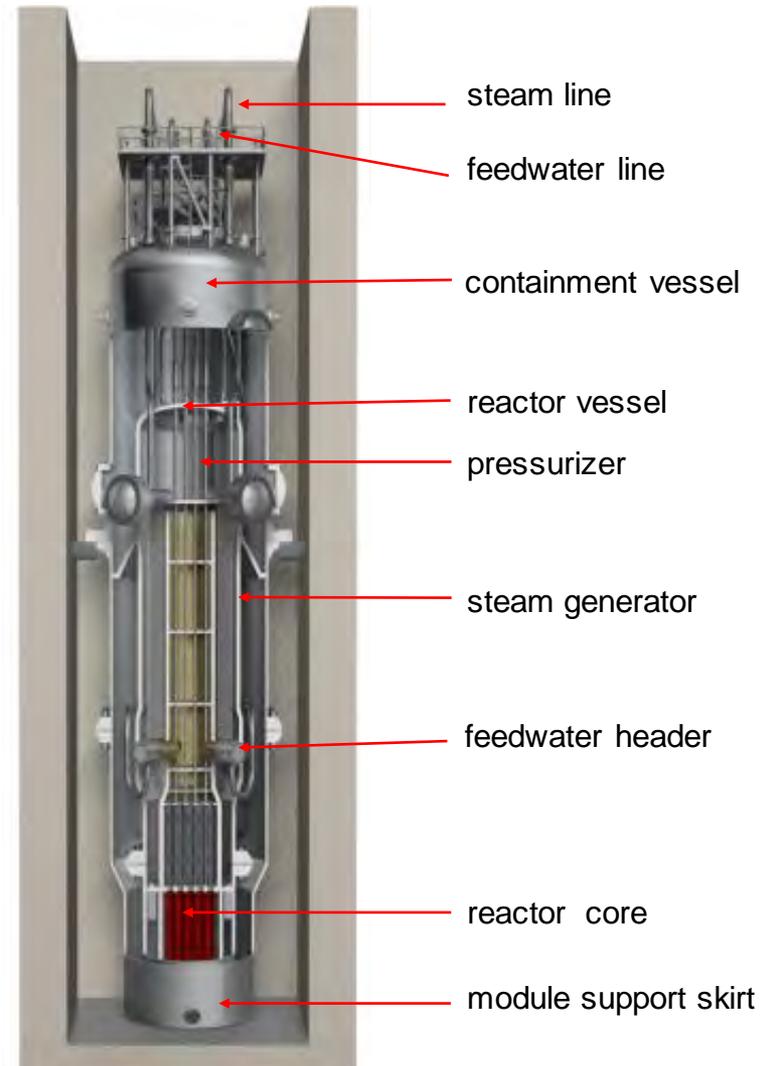
(1) The iodine spike DBST is not an event, rather it serves as a bounding surrogate for design-basis loss of primary coolant into containment events.

(2) The CDE is a beyond-design-basis special event.

Power Module Overview

Integral Pressurized Water Reactor

- Core, steam generator and pressurizer in one vessel
- Integrated reactor design, no large-break loss-of-coolant accidents
- Reactor coolant system operated in single phase (liquid) density driven flow
- Safety decay heat removal systems are passive and fail safe
- Module protection system designed to automate event mitigation actuations (no operator actions)



Module Protection System Functions

- Reactivity Control
 - Reactor trip
 - CVCS/Demineralized Water Isolation
- RCS and Secondary Inventory Control
 - Containment Isolation
 - Secondary Isolation
- Heat Removal
 - DHRS Actuation
 - ECCS Actuation
- Subcooling
 - Reactor trip

Event Mitigation

Increase in heat removal transients

- Reactor trip
- Secondary Isolation

Decrease in heat removal transients

- Reactor trip
- DHRSActuation

Reactivity and power transients

- Reactor trip
- Demineralized water isolation

Increase in RCS inventory transients

- Reactor trip
- CVCS Isolation

Decrease in RCS inventory transients

- Reactor trip
- CNV Isolation
- ECCS actuation

Stability

- Reactor trip

NPM Design Basis Event Identification

- Ch 15 initiating events considered for internal events in a single NPM while at power
- Systematic process used to evaluate NPM systems to assure appropriate scope of design basis events identified
- Design basis events categories consistent with operating LWRs
 - Increase in heat removal, decrease in heat removal, reactivity and power distribution anomalies, increase in RCS inventory, decrease in RCS inventory, radioactive release from a subsystem or component
- NuScale-specific phenomena/event progression:
 - PWR stability
 - Return to power

NuScale Unique Ch 15 Events

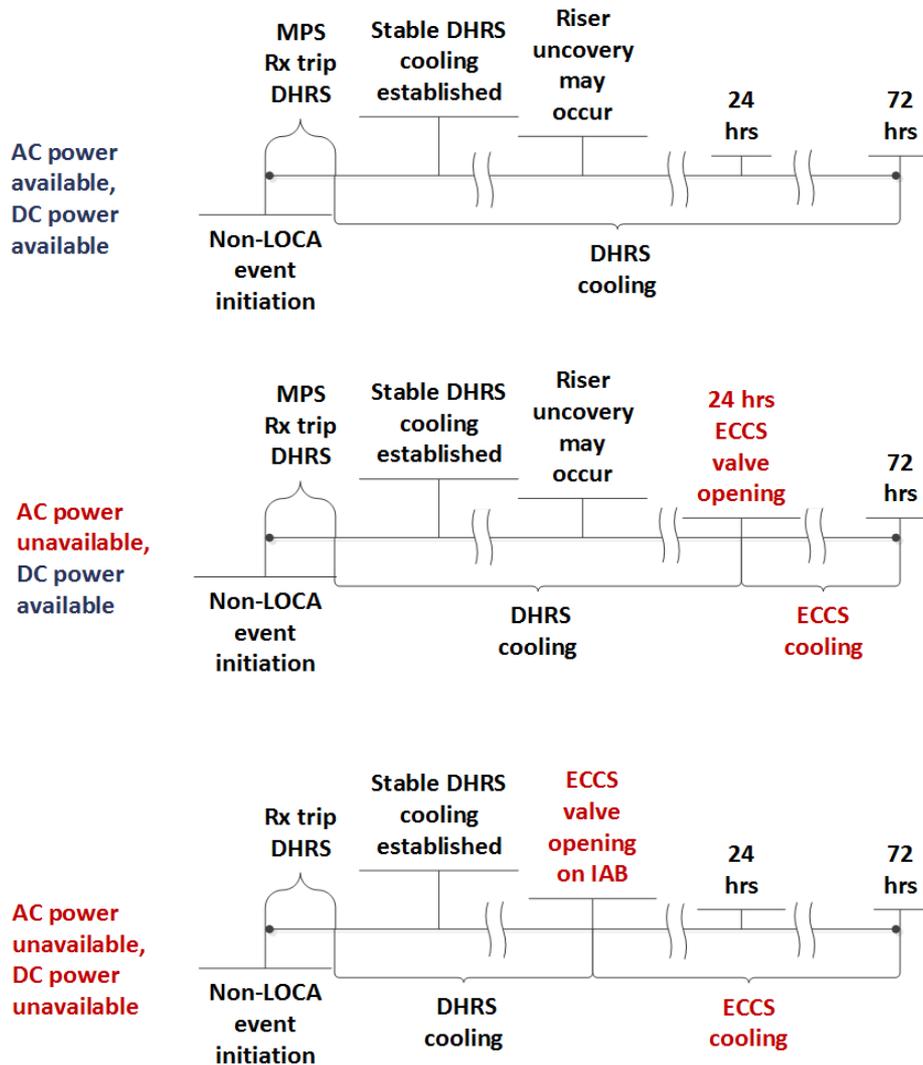
Sec.	Event	Classification	System T/H (code)	Subchannel (code)	Radiological
15.1.6	Loss of containment vacuum/containment flooding	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a
15.2.9	Inadvertent operation of the decay heat removal system	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a
15.6.6	Inadvertent operation of emergency core cooling system	AOO	Valve opening event (NRELAP5)	n/a	Yes ⁽¹⁾

(1) See FSAR Ch 12

Analytical Assumptions for Ch 15 Analysis

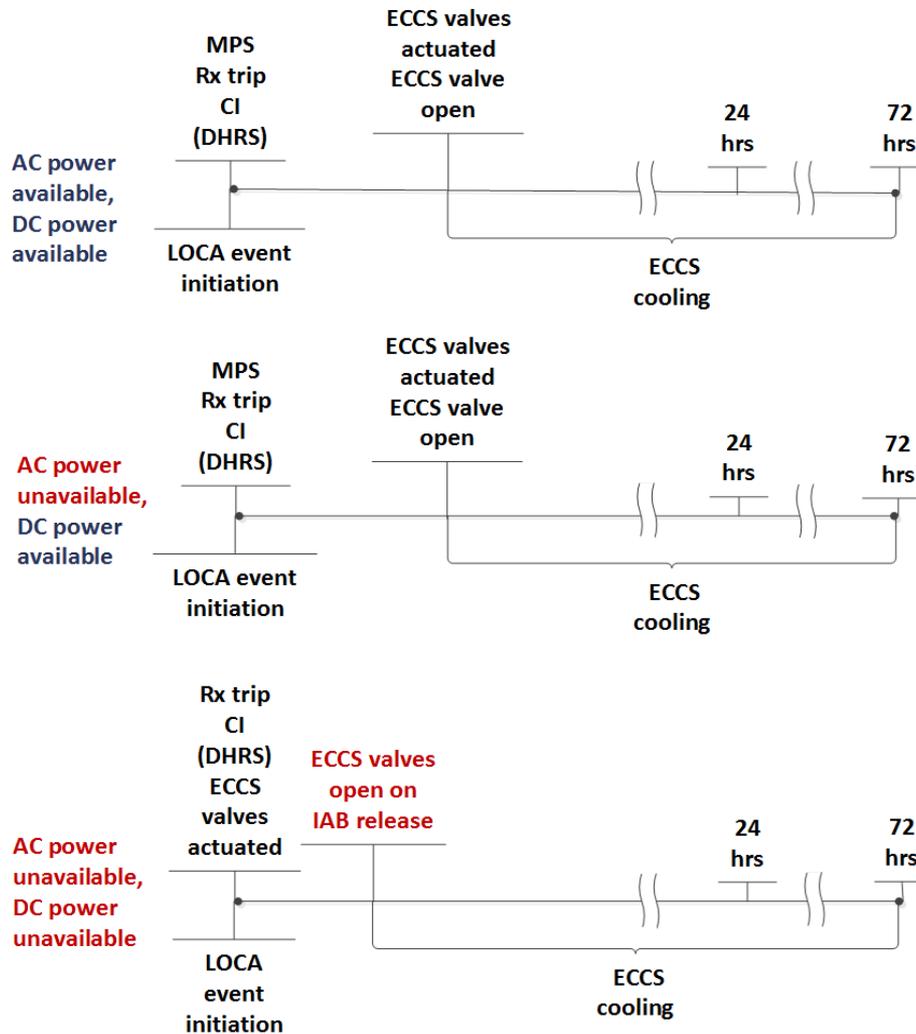
- **Operator action**
 - No operator actions required to achieve safety functions for 72 hours after an initiating event occurs
- **Single failure**
 - Worst single failure of safety-related component assumed
- **Loss of power**
 - Chapter 15 event analyses consider availability of AC power and DC power
- **Scope of event progression**
 - Safety analyses of design basis events are performed from event initiation until a safe, stabilized condition is reached
 - After safe, stabilized condition is reached:
 - ECCS long term decay and residual heat removal
 - Return to power
 - Extended DHRS operation

Loss of Power – Non-LOCA Event



- Availability of AC, DC power affects whether ECCS valves actuate, and what time they open

Loss of Power –LOCA Event



- Availability of DC power affects whether ECCS valves open on level actuation or on IAB release
- Break size and credit for DHRS operation also affect whether ECCS valves open on level or on IAB release

Limiting Transient Results

Parameter	Event	Acceptance Criterion	Limiting Result
Maximum RCS Pressure	Several	< 2315 psia (110% P _{design}) or < 2520 psia (120% P _{design})	~ 2170 psia
Maximum SG Pressure	Inadvertent Operation DHRS	< 2315 psia (110% P _{design})	1582 psia
	SG tube failure	< 2520 psia (120% P _{design})	1806 psia
MCHFR	Single rod withdrawal	> 1.284	1.614
	Inadvertent Opening RRV	> 1.13	1.41
	LOCA	> 1.29	1.796
Level above top of core	LOCA	> 0 ft	1.5 ft

Chapter 6.2.1

Containment Response Analysis

July 10, 2019

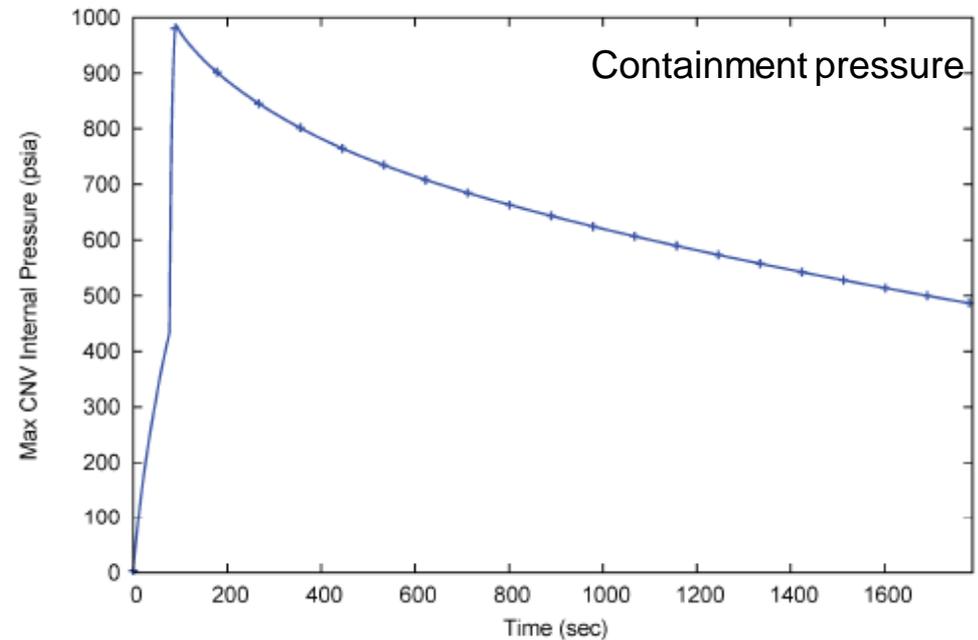


6.2.1 CNV Response Analysis

- CNV designed to withstand full spectrum of primary and secondary system M&E releases with worst case single active failure and loss of power
- CNV response analysis methodology based on NRELAP5 described in TR-0516-49084 Rev 0
 - NRELAP5 used to model integrated response of the RPV blowdown and CNV pressurization/heat removal
 - Containment response analysis based on models from LOCA EM or non-LOCA EM, with initial and boundary condition changes necessary to conservatively bias the mass and energy release and maximize the CNV pressure and temperature response
- Limiting results:
 - Peak pressure: 986 psia < 1050 psia design pressure
 - Peak wall temperature: 526 °F < 550 °F design temperature

Containment Response Limiting Pressure Case

Time (sec)	Event
0	Inadvertent RRV opening Loss of normal AC and DC power FW/MS isolation Reactor trip
0.4	High CNV pressure – CNV isolation
74	ECCS valve opening on IAB release <i>- Single failure of 1 RRV to open</i>
91	Peak CNV pressure 986 psia < 1050 psia design pressure
~1800	CNV pressure < 50% peak pressure



Chapter 15

Long Term Cooling

July 10, 2019

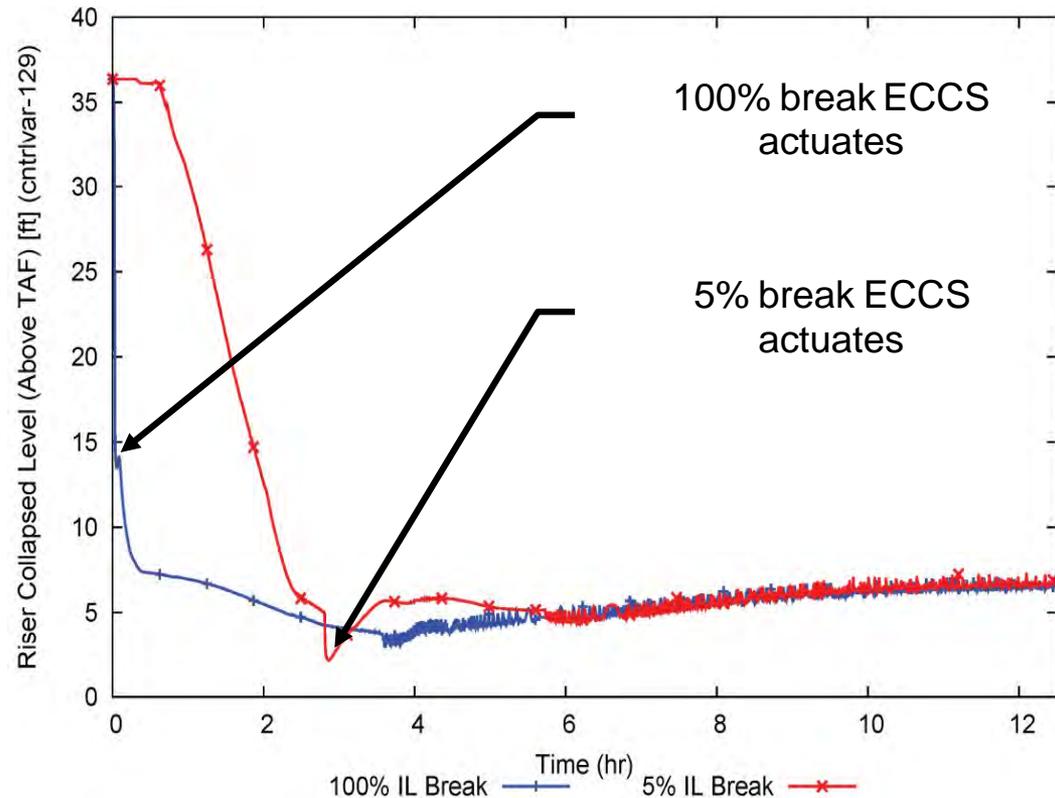


Long Term Cooling Analysis

- Events considered:
 - LOCA events transition to ECCS cooling; full break spectrum evaluated
 - Non-LOCA events transition from DHRS to ECCS cooling
 - Events considered are SGTF, loss of FW flow, and a generic evaluation of DHRS cooldown
- Biasing approach and results:
 - **Minimum level:** All minimum level cases showed the core remained covered at all times during the LTC phase, and that boron precipitation is precluded during the time of minimum level when boron concentration is maximized
 - **Minimum temperature:** All minimum temperature cases showed margin to boron precipitation
 - **Maximum temperature:** All maximum temperature cases showed decreasing cladding temperature, with final clad temperature remaining well below operating temperature at full power

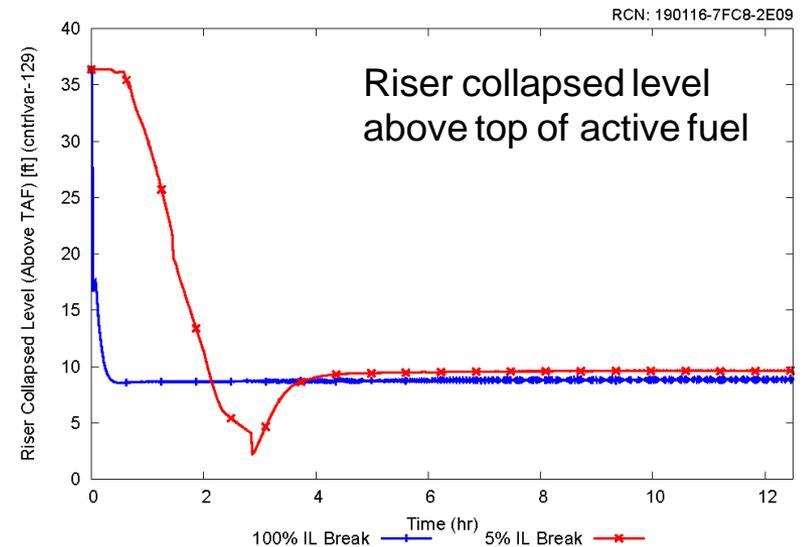
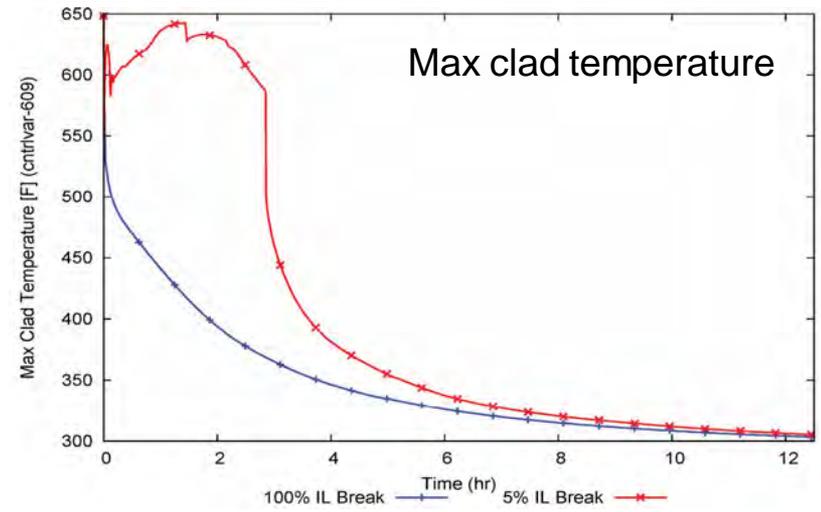
LTC Results – Injection Line Min Level

- Minimum level case maximizes RPV energy and minimizes CNV energy
- Higher pressure difference between the RPV and CNV increases coolant accumulation in the CNV which reduces RCS level
- Minimum level during LTC phase occurs 3-5 hours after ECCS actuation
- Long term, level recovers towards equilibrium as the pressure difference between the RPV and CNV is reduced with decreasing decay heat



LTC Results – Injection Line Max Temp

- Maximum temperature case maximizes initial RCS energy and decay heat while minimizing heat transfer to the reactor pool
- Results demonstrate clad temperature follows a decreasing long term trend
- Long-term level remains at equilibrium condition



Chapter 15.0.6

Return to Power

July 10, 2019



Principle Design Criteria 27

- Principle Design Criteria 27 → Passive Reactor GDC-27 equivalent

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.

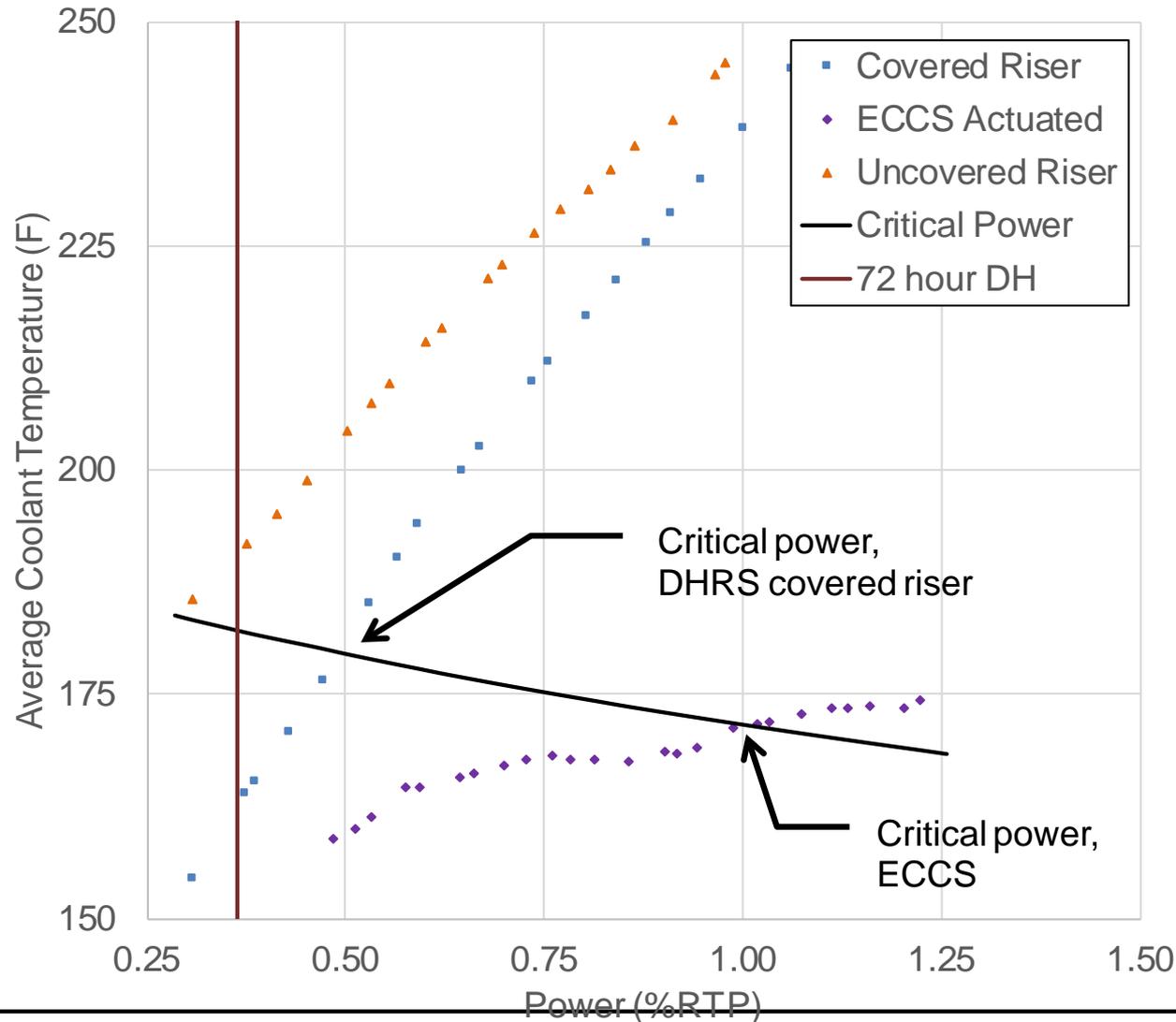
Following a postulated accident, the control rods shall be capable of holding the reactor core subcritical under cold conditions, without margin for stuck rods.

- Compliance with PDC-27:
 - Immediate shutdown margin is sufficient to protect RCPB and SAFDLs with appropriate margin for the worst rod stuck out of the core
 - Cold shutdown is achieved with all control rods fully inserted
 - Loss of shutdown margin consequences
 - Worst case OCRP not challenging to SAFDLs
 - Critical power level does not challenge DHRS or ECCS heat removal capabilities
 - Probability of the combination of conditions that results in a loss of shutdown return to power with a single rod stuck out of the core is small

Return to Power Characterization

- Characterize the steady state RCS temperature a function of available decay heat for both DHRs and ECCS heat removal.
- Calculate the worst rod stuck out critical power level as a function of RCS temperature.
- Conclusions:
 - Loss of shutdown margin may occur around 40hrs with zero Boron in RCS after Xenon decay
 - RCS temperature low (< 200F based on best estimate SIMULATE results)
 - ECCS cooling is more limiting
 - Higher pool temperature can preclude worst rod stuck out criticality (140F based on best estimate SIMULATE results)
 - Simple pool boiling CHF analysis demonstrates large margin

Return to Power Characterization



Acronyms

- **AOO – Anticipated Operational Occurrences**
- **ASME – American Society of Mechanical Engineers**
- **ASTM – American Society for Testing and Materials**
- **BPVC – Boiler Pressure Vessel Code**
- **CES – Containment Evacuation System**
- **Ch – Chapter**
- **CIV – Containment Isolation Valve**
- **CNV – Containment Vessel**
- **COL – Combined License**
- **CRDM – Control Rod Drive Mechanism**
- **CVCS – Chemical and Volume Control System**
- **DBST – Design Basis Source Term**
- **DHRS – Decay Heat Removal System**
- **ECCS – Emergency Core Cooling System**
- **EFPY – Effective Full Power Years+D51**
- **EM – Evaluation Methodology**
- **EPRI – Electric Power Research Institute**
- **FIV – Flow Induced Vibration**
- **FSAR – Final Safety Analysis Report**
- **ft – feet**
- **FW – Feedwater**
- **FWIV – Feedwater Isolation Valve**
- **GDC – General Design Criteria**
- **HZP – Hot Zero Power**
- **IAB – Inadvertent Actuation Block**
- **ISI – Inservice Inspection**
- **LOCA – Loss of Coolant Accident**
- **LTC – Long Term Cooling**

Acronyms

- **LTOP – Low Temperature Overpressure Protection**
- **LWR – Light Water Reactor**
- **M&E – Mass and Energy**
- **MCHFR – Minimum Critical Heat Flux Ratio**
- **MPS – Module Protection System**
- **MSIV – Main Steam Isolation Valve**
- **MSS – Main Steam System**
- **MWt – Megawatts thermal**
- **NDE – Non-destructive Examination**
- **NEI – Nuclear Energy Institute**
- **NPM – NuScale Power Module**
- **NPS – Nominal Pipe Size**
- **NRELAP – System Thermal Hydraulic Code**
- **OCRCP – Overcooling Return to Power**
- **OD – Outside Diameter**
- **OE – Operations Experience**
- **oF – degrees Fahrenheit**
- **PDC – Principle Design Criteria**
- **psia – pounds per square inch absolute**
- **P-T – Pressure and Temperature**
- **PTS – Pressurized Thermal Shock**
- **PWR – Pressurized Water Reactor**
- **PWSCC – Primary Water Stress–Corrosion Cracking**
- **PZR – Pressurizer**
- **RCCWS – Reactor Component Cooling Water System**
- **RCPB – Reactor Coolant Pressure Boundary**
- **RCS – Reactor Coolant System**

Acronyms

- **RG – Regulatory Guide**
- **RPV – Reactor Pressure Vessel**
- **RRV – Reactor Recirculation Valve**
- **RSV – Reactor Safety Valve**
- **RTNDT – Reference Temperature for Nil-ductility Transition**
- **RVV – Reactor Vent Valve**
- **Rx – Reactor**
- **SAFDLs – Specified Acceptable Fuel Design Limits**
- **SDM – Shutdown Margin**
- **SG – Steam Generator**
- **SGTF – Steam Generator Tube Failure**
- **T/H – Thermal Hydraulic**
- **TRV – Thermal Relief Valve**
- **TS – Technical Specifications**
- **TT – Thermally Treated**
- **UHS – Ultimate Heat Sink**
- **USE – Upper Shelf Energy**
- **VIPER – Subchannel Code**

July 9, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled “ACRS Full Committee Presentation: NuScale FSAR Chapter 20, Mitigation of Beyond-Design-Basis Events,” PM-0719-66253, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) NuScale Full Committee Meeting on July 10, 2019. The materials support NuScale’s presentation of Chapter 20, “Mitigation of Beyond-Design-Basis Events,” of the NuScale Design Certification Application.

The enclosure to this letter is the nonproprietary version of the presentation entitled “ACRS Full Committee Presentation: NuScale FSAR Chapter 20, Mitigation of Beyond-Design-Basis Events,” PM-0719-66253, Revision 0.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Nadja Joergensen at 541-452-7338 or at njoergensen@nuscalepower.com.

Sincerely,



Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Robert Taylor, NRC, OWFN-7H4
Michael Snodderly, NRC, TWFN-2E26
Samuel Lee, NRC, OWFN-8H12
Gregory Cranston, NRC, OWFN-8H12
Omid Tabatabai, NRC, OWFN-8H12

Enclosure: ACRS Full Committee Presentation: NuScale FSAR Chapter 20, Mitigation of Beyond-Design-Basis Events, PM-0719-66253, Revision 0

Enclosure:

ACRS Full Committee Presentation: NuScale FSAR Chapter 20, Mitigation of Beyond-Design-Basis Events, PM-0719-66253, Revision 0

ACRS Full Committee Presentation



NuScale FSAR

Chapter 20

Mitigation of Beyond-Design- Basis Events

July 10, 2019

Presenters

Chris Maxwell
Senior Reactor Operator

Zackary Rad
Director, Regulatory Affairs

Nadja Joergensen
Licensing Specialist

Mitigation Strategies for Beyond- Design-Basis External Events 10 CFR 50.155(b)(1)

Indefinite Coping

- NuScale considers a minimum coping period of 14 days using only installed plant equipment to be sufficient time to establish “alternate means of removing heat.”
 - In the Fukushima Daiichi accident, without pre-planning or a hardened pool makeup connection, and with limited access to off-site resources, personnel began
 - adding water to the Unit 4 SFP with fire and concrete pump trucks after 9 days; and
 - injecting water via the fuel pool cooling system at 14 days.
- Beyond minimum installed equipment coping period, the continued use of installed plant equipment, ad hoc resources, and equipment repairs can be used to continue coping indefinitely.

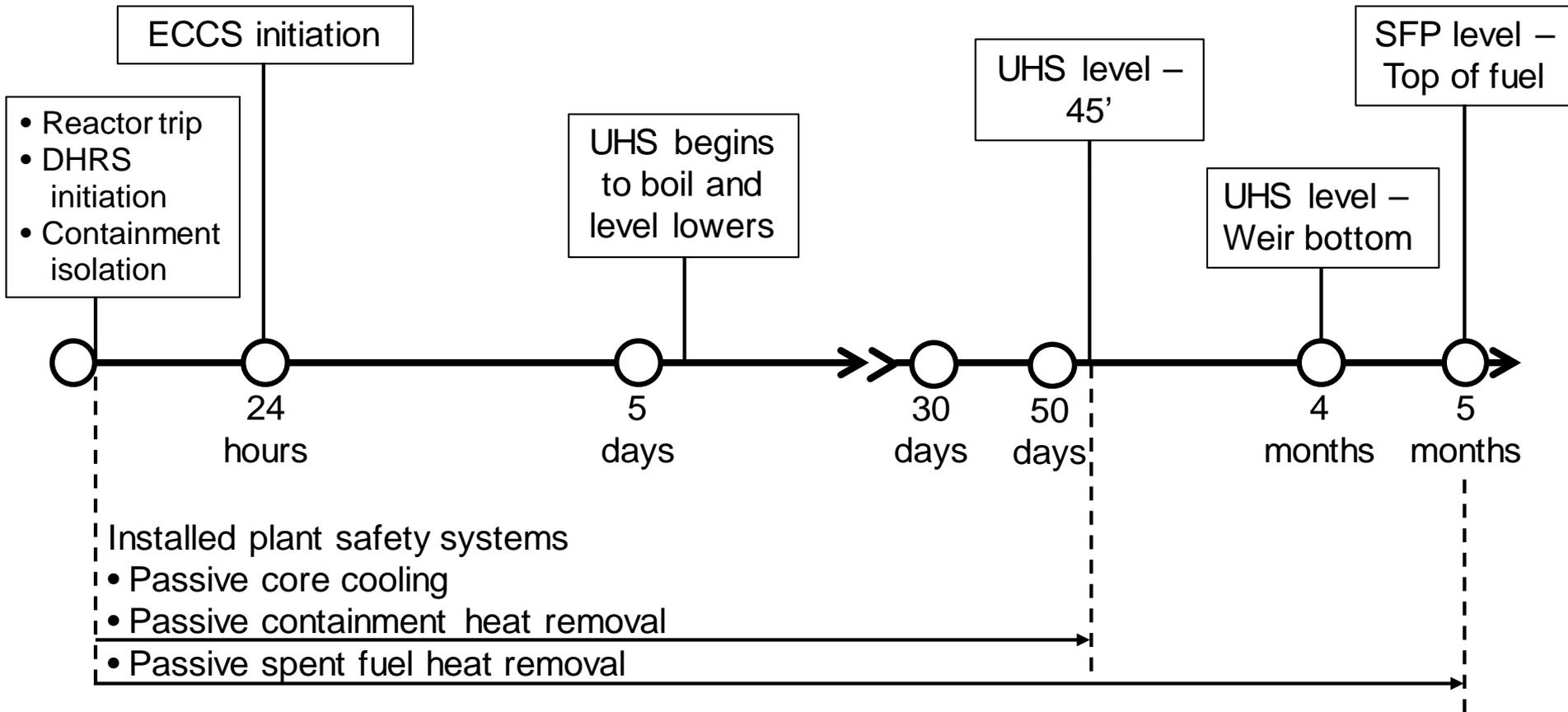
NuScale Coping Capability

NuScale Power Plant Design

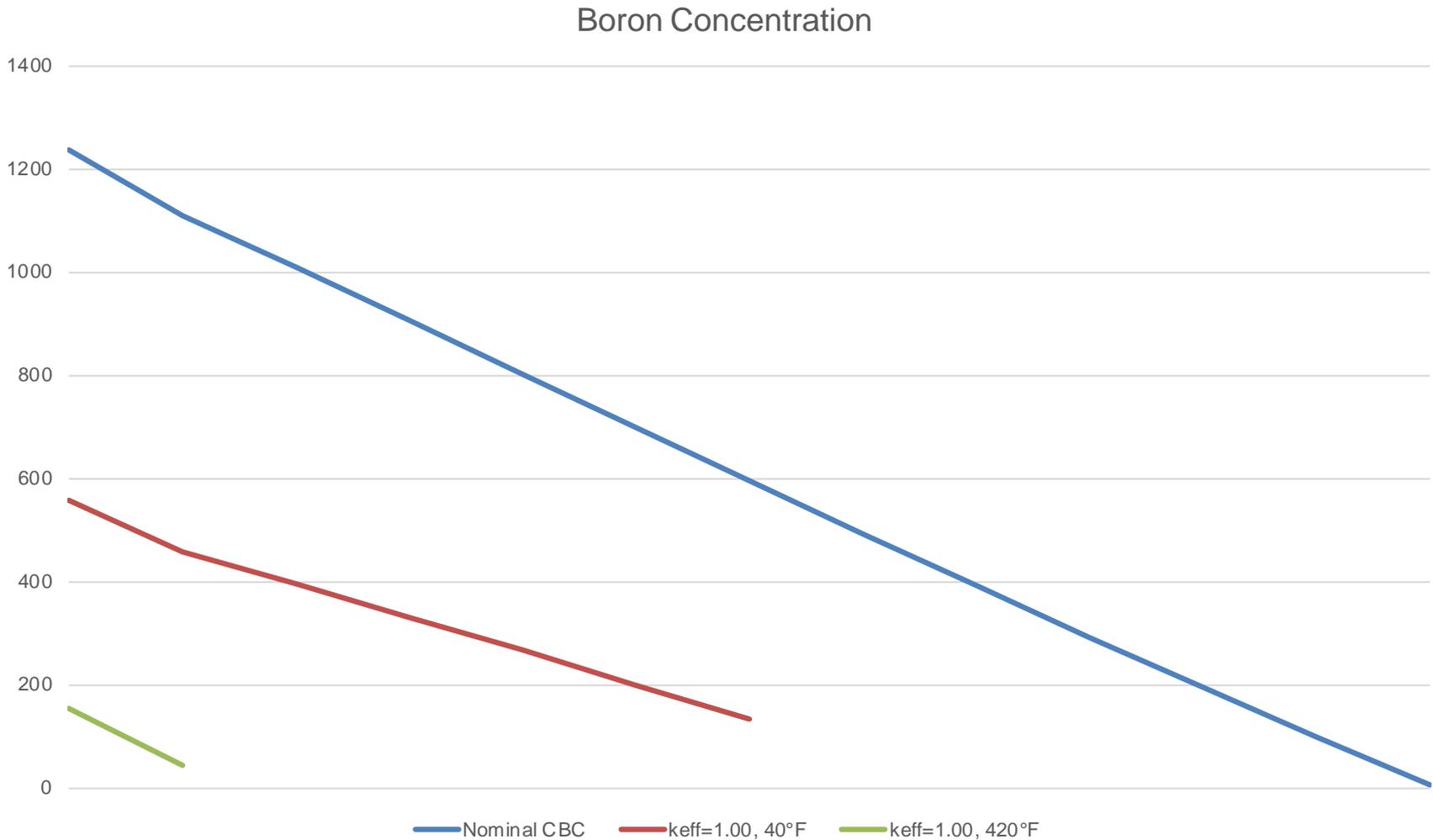
- The NuScale Power Plant was designed to provide coping during an ELAP concurrent with a LUHS without:
 - AC or DC electrical power
 - Inventory addition
 - Supplemental equipment
 - Off-site resources
 - Operator action (monitoring)
- Extended coping duration is provided by the automatic response of installed plant equipment alone.
- This strategy permits plant staff to focus on addressing the initiating event and restoring normal functional capabilities.

NuScale Power Plant Response

Plant response without operator action or off-site resources



Reactivity Control



Spent Fuel Pool Monitoring 10 CFR 50.155(e)

Spent Fuel Pool Level Indication

- The UHS system includes remote level indication for the following:
 - Reactor Pool
 - Refueling Pool
 - Spent Fuel Pool (2)
- Normally powered by the highly reliable DC power system via the plant protection system.
- Include a replaceable battery power source, independent from the plant AC and DC power systems, with a minimum capacity of 14 days.

Conclusion

- NuScale Power Plant Mitigation Strategy
 - Rely on the automatic response of permanently installed, safety-related plant equipment to establish and maintain the three key safety functions and provide extended coping capabilities of greater than 14 days.
- The NuScale Power Plant mitigation strategy does not require:
 - AC or DC electrical power
 - Inventory addition
 - Operator action
- NuScale SFPLI Strategy
 - Installed instrumentation with 14 day battery backup power supply.



United States Nuclear Regulatory Commission

Protecting People and the Environment

Preliminary Safety Evaluation with Open Items: Chapter 20, “Mitigation of Beyond- Design-Basis Events”

NuScale Design Certification Application

ACRS Full Committee Meeting
July 10, 2019

Agenda

- Background and Overview
- Regulatory Framework for the Staff Review
- Staff Review of NuScale's MBDBE strategy
- Phase 4 Review Plan
- Abbreviations

Background and Overview

- 10/30/2018: NuScale submitted its DCA, Rev 2
- 01/24/2019: NRC approved Final MBDBE Rule (10 CFR 50.155)
- 03/28/2019: NuScale informed the staff that it was revising its Ch. 20
- 06/10/2019: NuScale submitted revised Ch. 20
- 06/14/2019: NuScale submitted Rev. 1 to ELAP Technical Report
- 06/26/2019: NRC issued Information SECY 19-0066¹
- 07/12/2019: SECY 19-0066 will be publicly available (ML19148A443)
- Staff's (preliminary) Phase 2 SER for Chapter 20 is based on Rev. 2 of NuScale's DCA
- Based on the docketed information, the Staff's findings in Phase 2 SER are limited to the first 72 hours after initiation of a BDBE.

¹SECY 19-0066: "Staff Review of NuScale Power's Mitigation Strategy for Beyond-Design-Basis External Events"

Regulatory Framework for the Staff Review

- The recently approved regulation, 10 CFR 50.155, for mitigation of beyond-design-basis events (MBDBE) does not apply to applicants for a design certification.
- NuScale is voluntarily seeking the NRC's approval of its proposal to use installed design features to mitigate beyond-design-basis external events.
- NuScale design incorporates several design features that provide enhanced capabilities for mitigating an extended loss of ac power compared to currently operating reactors.

Regulatory Framework (Cont'd)

TEXT OF 10 C.F.R. § 50.155(b), (c), and (e) APPROVED BY THE COMMISSION

(b) Strategies and guidelines. Each applicant or licensee shall develop, implement, and maintain:

(1) Mitigation strategies for beyond-design-basis external events. Strategies and guidelines to mitigate beyond-design-basis external events from natural phenomena that are developed assuming a loss of all ac power concurrent with either a loss of normal access to the ultimate heat sink or, for passive reactor designs, a loss of normal access to the normal heat sink. These strategies and guidelines must be capable of being implemented site-wide and must include the following:

(i) Maintaining or restoring core cooling, containment, and spent fuel pool cooling capabilities; and

(ii) The acquisition and use of offsite assistance and resources to support the functions required by paragraph (b)(1)(i) of this section indefinitely, or until sufficient site functional capabilities can be maintained without the need for the mitigation strategies.

....

(c) Equipment. (1) The equipment relied on for the mitigation strategies and guidelines required by paragraph (b)(1) of this section must have sufficient capacity and capability to perform the functions required by paragraph (b)(1) of this section.

(2) The equipment relied on for the mitigation strategies and guidelines required by paragraph (b)(1) of this section must be reasonably protected from the effects of natural phenomena that are equivalent in magnitude to the phenomena assumed for developing the design basis of the facility.

....

(e) Spent fuel pool monitoring. In order to support effective prioritization of event mitigation and recovery actions, each licensee shall provide reliable means to remotely monitor wide-range water level for each spent fuel pool at its site until 5 years have elapsed since all of the fuel within that spent fuel pool was last used in a reactor vessel for power generation.

SRM-M190124A, Enclosure 1, Federal Register Notice at 140-41.

Staff Review of NuScale MBDBE

- Maintain consistency with scope of review performed for operating reactors and other design certifications.
 - Focus on the initial response coping period (first 72 hours) where the most critical and time-sensitive actions are projected to occur.
- Verify the design capabilities and capacities of the permanently installed SSCs satisfy the required safety functions, including the effects of credible transient phenomena, for 72 hours following initiating event.
 - Review will focus on design aspects of SSCs as described in the FSAR.
 - Operational aspects (e.g., procedures, training) deferred to COL stage.

Staff Review of NuScale MBDBE (Cont'd)

- COL applicant would need to describe how mitigating strategies (or sufficient site functional capabilities) are maintained for an indefinite time period.
 - SSC design aspects would only need to be addressed if there are credible transient phenomena (e.g., return to power) that could challenge core cooling, containment, or SFP cooling beyond 72 hours
 - Level of detail expected is commensurate with time available to implement actions.
- Staff plans to document that instrumentation is not relied on to support the mitigation strategies; however:
 - Instrumentation is expected to be available for the initial 72 hours,
 - Provides additional assurance that systems have responded as designed.

Phase 4 Review

- The staff will follow its plans, as described in SECY 19-0066, to complete its review of NuScale's Ch. 20 in Phase 4 of the DCA review
- The Staff will evaluate NuScale's DCA, Rev. 3, information for compliance with the requirements of 10 CFR 50.155
- NuScale DCA, Rev. 3, is expected to be submitted in late August 2019

Abbreviations

ACRS	Advisory Committee on Reactor Safeguards
CFR	Code of Federal Regulations
CNV	Containment Vessel
COL	Combined License
BDBA	Beyond-Design-Basis Accident
CNV	Containment Vessel
DCA	Design Certification Application
DHRS	Decay Heat Removal System
ECCS	Emergency Core Cooling System
EDSS	Highly Reliable Electrical System
ELAP	Extended Loss of AC Power
FSAR	Final Safety Analysis Report
JLD	NRC Japan Lessons Learned Directorate
MBDBE	Mitigation of Beyond-Design-Basis Events
MPS	Module Protection System
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	NRC Office of Nuclear Reactor Regulation
NSIR	NRC Office of Nuclear Security and Incident Response
NRO	NRC Office of New Reactors
SER	Safety Evaluation Report
SRM	Staff Requirement Memorandum
UHS	Ultimate Heat Sink



Safety Evaluation with Open Items: Chapter 6, “Engineered Safety Features”

NuScale Design Certification Application

ACRS Full Committee Meeting
July 10, 2019

Agenda

- Overview of the NRC Staff's Presentation
- Exemption Requests Contained in FSAR Section 6.2, "Containment Systems"
- Abbreviations

Overview of the NRC Staff's Presentation

- The staff briefed the ACRS Subcommittee On June 18, 2019
- Members asked the staff to cover the following topics at Full Committee:
 1. Section 6.3, “Emergency Core Cooling System,” with emphasis on ECCS valves and their performance out to 72 hours
 - *Staff will cover as part of its Ch 15. briefing*
 2. ECCS valves description, test plan, and results
 - *Staff will cover as part of Ch. 3, and 15, briefings, as needed*
 3. 6.2.1 and 6.2.2 with emphasis on results of limiting case
 - *NuScale will address as part of its presentation*
 4. Exemption requests associated with Chapter 6
 - *Staff will address in the following slides*
 5. Containment level measurement and the challenge that containment is dry during normal operation
 - *NuScale will address as part of its presentation*



NuScale Exemption Requests Regarding Containment Isolation Systems (Section 6.2.4 of the SER)

Clint Ashley
Reactor Systems Engineer, NRO

Exemption Requests

- NuScale's DCA (Part 7) contains exemption requests associated with the following containment isolation requirements:
 - GDC 55
 - GDC 56
 - GDC 57
 - 10 CFR 50.34(f)(2)(xiv)(E)

10 CFR 50.12 - Specific Exemptions

- Pursuant to 10 CFR 50.12 in part, the Commission may grant exemptions when special circumstances are present.
- According to 10 CFR 50.12(a)(2)(ii), special circumstances are present whenever, “application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.”

Underlying Purpose

- The underlying purpose of GDCs 55, 56, and 57 is to provide containment isolation capability that supports the safety function of containment to provide a barrier to the uncontrolled release of radioactivity to the environment.
- The underlying purpose of 50.34(f)(2)(xiv)(E) is to limit radiological releases by ensuring containment isolation for systems that provide paths to the environs.

Staff's Review

- For GDC 55, 56 and 57, the staff finds that the NuScale design accomplishes the containment safety function by providing a containment isolation capability comparable to that required by the GDCs and therefore, the underlying purpose of the GDCs is met.
- The staff find's NuScale's exemption request meets the requirements for an exemption as defined in 10 CFR 50.12.

Staff's Review

- For 50.34(f)(2)(xiv)(E), the staff finds that the NuScale design isolates systems that provide a path to the environs before core damage or degradation occurs, preventing significant releases from the containment, and therefore, the underlying purpose of the rule is met.
- The staff find's NuScale's exemption request meets the requirements for an exemption as defined in 10 CFR 50.12.

NuScale Exemption Request Regarding “Combustible Gas Control in Containment” (SER Section 6.2.5)

*Anne-Marie Grady
Reactor Systems Engineer, NRO*

Regulatory Basis

- 10 CFR 50.44(c)(2) – requires that the plant accommodate hydrogen generation up to 100 percent fuel clad-coolant reaction while limiting containment hydrogen to less than 10 percent, and maintain containment structural integrity and other accident mitigation features.
- NuScale has requested an exemption from 50.44(c)(2), which would require the design provide a hydrogen control system to limit hydrogen concentrations below 10 percent.
- Pursuant to 10 CFR 50.12 in part, the Commission may grant exemptions when special circumstances are present.
 - *Special circumstances are present (10 CFR 50.12(a)(2)(ii))* in that application of the regulation is not necessary to achieve the underlying purpose of the rule
 - *Special circumstances are present (10 CFR 50.12(a)(2)(iii))* in that compliance would result in undue hardship or other costs that are significantly in excess of those incurred by others similarly situated.

Exemption Request

- Staff review focused on:
 - Hydrogen conditions in the CNV during the postulated accident
 - Demonstration of adequate containment mixing
 - Equipment survivability following postulated hydrogen combustion
- NuScale analyses demonstrated the containment would survive a bounding combustion event inside the containment at 72 hours
- Staff confirms the calculation for a limiting pressure pulse inside the containment resulting from a hydrogen combustion event at 72 hours for a very short duration, and containment integrity would be maintained.
- Providing a system to control the hydrogen concentration is not necessary to serve the underlying purpose of 10 CFR 50.44(c)(2), which is to prevent a loss of containment integrity
- Staff recommends granting the Exemption Request

NuScale Exemption Request Regarding “Containment Leakage Testing” (SER Section 6.2.6)

*Anne-Marie Grady
Reactor Systems Engineer, NRO*

Exemption Request Regarding Containment Leakage Testing

- NuScale requested an exemption from the following regulations:
- 10 CFR 50, App. A, GDC 52, Capability for Containment Leakage Rate Testing, requires the capability to perform containment periodic integrated leakage rate testing (ILRT) (Type A) at containment design pressure.
- 10 CFR 50, App. J—Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, requires Type A tests preoperationally and periodically. thereafter.
- NuScale asserts that the CNV ASME design, analysis for leak tightness, 100% vessel inspectability, and pre service design pressure hydrostatic leakage test would satisfy the underlying goal of demonstrating CNV allowable leakage.

Exemption Request Regarding Containment Leakage Testing (Cont'd)

- NuScale CNV is an ASME Section III, Class 1 pressure vessel
- CNV leakage analysis, design specifications, capability for 100% vessel inspection, examination, and testing, provide assurance that the leakage integrity of the CNV is maintained.
- NuScale analyzed the CNV bolt design for the flanged openings, using ANSYS, based on the seal design and specification.
- NuScale calculated flange contact pressures and corresponding flange gaps, based on CNV internal accident pressure and temperatures.
- Staff reviewed these calculations during an audit.

Exemption Request Regarding Containment Leakage Testing (Cont'd)

- NuScale asserts that the CNV design combined with Types B and C test results are sufficiently representative of accident conditions to demonstrate that the TS leak rate, L_a , would not be exceeded.
- In addition to testing required by ASME, NuScale proposes a preservice design pressure test to confirm the expected performance of the CNV design. This would be verified by ITAAC.
- The CNV ASME design, analysis for leak tightness, 100% vessel inspectability, and pre service design pressure hydrostatic leakage test would satisfy the underlying goal of demonstrating CNV allowable leakage
- The staff recommends approving the exemption request to not require Type A testing, nor to require design capability for ILRT.

Abbreviations

ACRS	Advisory Committee on Reactor Safeguards
ASME	American Society of Mechanical Engineers
CES	Containment Evacuation System
CFDS	Containment Flooding and Drain System
CFR	Code of Federal Regulations
CNV	Containment Vessel
CRHS	Control Room Habitability System
CRVS	Control Room Ventilation System
COL	Combined License
DBA	Design-Basis Accident
DCA	Design Certification Application
DSRS	Design Specific Review Standard
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GL	Generic Letter
ILRT	Integrated Leakage Rate Testing
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
MSS	Main Steam System
NRC	Nuclear Regulatory Commission
NRO	NRC Office of New Reactors
PDC	Principal Design Criteria
RG	Regulatory Guide
SER	Safety Evaluation Report
SSC	Structures, Systems, and Components
TEDE	Total Effective Dose Equivalent



Presentation to the ACRS Full Committee

NuScale Design Certification Application Review

Safety Evaluation Report

Chapter 3: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Project Manager: Marieliz Vera

July 10, 2019

Staff Review Team

● Technical Staff

- ◆ **Renee Li**, Senior Mechanical Engineer
- ◆ **Tuan Le**, Mechanical Engineer
- ◆ **Yiu Law**, Mechanical Engineer
- ◆ **Alexander Tsirigotis**, Mechanical Engineering
- ◆ **Yuken Wong**, Senior Mechanical Engineer
- ◆ **Jason Huang**, Mechanical Engineer
- ◆ **Nick Hansing**, Mechanical Engineer
- ◆ **Michael Breach**, Mechanical Engineer
- ◆ **Kaihwa Hsu**, Senior Mechanical Engineer
- ◆ **Tom Scarbrough**, Senior Mechanical Engineer
- ◆ **Jorge Cintron- Rivera**, Electrical Engineer
- ◆ **Sheila Ray**, Senior Electrical Engineer
- ◆ **Vaughn Thomas**, Structural Engineer
- ◆ **Robert Roche-Rivera**, Structural Engineer
- ◆ **Sunwoo Park**, Structural Engineer
- ◆ **Pravin Patel**, Structural Engineer
- ◆ **Ata Istar**, Structural Engineer
- ◆ **George Wang**, Structural Engineer
- ◆ **Maryam Khan**, Structural Engineer
- ◆ **Manas Chakravorty**, Senior Structural Engineer
- ◆ **Bhagwat Jain**, Senior Structural Engineer
- ◆ **John Ma**, Senior Structural Engineer
- ◆ **Edward Stutzcage**, Health Physicist
- ◆ **Raul Hernandez**, Reactor System Engineer

Staff Review Team

- **Technical Staff**

- ◆ **Nicholas McMurray**, Materials Engineer
- ◆ **John Honcharik**, Senior Materials Engineer
- ◆ **Eric Reichelt**, Senior Materials Engineer
- ◆ **Seshagiri Tammara**, Physical Scientist
- ◆ **Vladimir Graizer**, Seismologist
- ◆ **John Wu**, Mechanical Engineer
- ◆ **Jim Strnisha**, Mechanical Engineer

- **Project Managers**

- ◆ **Gregory Cranston**, Lead Project Manager
- ◆ **Marieliz Vera**, Chapter Project Manager

Sections with no Open Items

- ◆ 3.2.1 – “Seismic Classification”
- ◆ 3.2.2 – “System Quality Group Classification”
- ◆ 3.3.1 – “Severe Wind Loads”
- ◆ 3.3.2 – “Extreme Wind Loads (Tornado and Hurricane Loads)”
- ◆ 3.4.1 – “Internal Flood Protection for Onsite Equipment Failure”
- ◆ 3.4.2 – “Analysis Procedures”
- ◆ 3.5.1.1 & 3.5.1.2 – “Internally Generated Missiles (Outside and Inside Containment)”
- ◆ 3.5.1.4 – “Missiles Generated by Tornadoes and Extreme Winds”
- ◆ 3.5.1.5 – “Site Proximity Missiles (Except Aircraft)”
- ◆ 3.5.1.6 – “Aircraft Hazards”
- ◆ 3.5.2 – “Structures, Systems, and Components to be Protected from Externally Generated Missiles”

Sections with no Open Items (Contd)

- ◆ 3.6.1 – “Plant Design for Protection against Postulated Piping Failures in Fluid Systems Outside Containment”
- ◆ 3.7.1 – “Seismic Design Parameters”
- ◆ 3.7.4 – “Seismic Instrumentation”
- ◆ 3.8.5 – “Foundations”
- ◆ 3.9.1 – “Special Topics for Mechanical Components”
- ◆ 3.10 – “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment”
- ◆ 3.12 – “ASME BPV Code Class 1, 2, and 3 Piping Systems and Associated Support Design
- ◆ 3.13 – “Threaded Fasteners—ASME BPV Code Class 1, 2, and 3”

Summary of open items

- Chapter 3 (without 3.9.2 and 3.11) has 18 Open items on the P2 SE
- 9 items have been resolved
- Items that are still open
 - 03.05.01.03-1 – turbine missile
 - 03.05.03-1 – turbine missile barrier
 - 03.06.02-1 – RVV and RRV bolted connection break exclusion justification
 - 03.07.03-1 – Bioshield
 - 03.08.02-1 – Updated CNV stress due to new seismic loads
 - 03.08.02-2 – Stress analysis of CNV RPV support
 - 03.08.03-3 – CNV fatigue evaluation
 - 03.09.03-1 – stress and fatigue analysis Reactor vessel and reactor vessel internals
 - 03.09.06-1 – ECCS Valve Design Demonstration Testing



Section 3.5.1.3 – Turbine Missile

John Honcharik

3.5.1.3 Turbine Missiles

Regulatory Basis and Use of Barriers

- 10 CFR Part 50, Appendix A, GDC 4, requires SSCs important to safety to be appropriately protected against environmental and dynamic effects, including the effects of missiles that may result from equipment failure.
- Safety-related and risk-significant SSCs for the NuScale design are located within the RXB and CRB.
- Turbine generator rotor shafts are unfavorably oriented such that the RXB and CRB are within the turbine low-trajectory hazard zone.
- To meet the requirements of 10 CFR Part 50, Appendix A, GDC 4, NuScale proposes to use installed or existing structures for protecting essential SSCs that meet the acceptance criteria in DSRS Section 3.5.3.

3.5.1.3 Turbine Missiles

Determining Turbine Missile Parameter on Barriers

- Open Item 03.05.01.03-1 : The staff determined that NuScale had not used the full spectrum of turbine missiles (size, weight and speed) which includes up to half of the last stage of the rotor in the barriers analysis.
- NuScale provided additional information in June which responded to staff RAIs and is currently under review.

3.5.1.3 Turbine Missiles

Current Staff Review of NuScale RAI Responses

- Need basis for destructive overspeed of 160% when other sources, including the EPRI Report 1006451, states 180 to 190%.
 - This will be the destructive overspeed used in barriers analysis as the bounding speed.
- Need methodology of determining speed of half of rotor based on NRC RAI No. 9596, Question 03.05.03-4 with the blades attached.
- Need to update previous reports in electronic reading room (Report ER-F010-6250, Revision 0 and Report ER-F010-6488, Revision 0) to include the analysis with the half of the rotor and other changes made based on audit and RAI responses. These reports are basis for turbine missiles and therefore need to be submitted to the NRC and documented in the DCD.



3.5.3 – Barrier Design Procedures

BP Jain

Use of Barriers

- To meet the regulatory requirements of 10 CFR Part 50, Appendix A, GDC 4, NuScale proposes:
 - to use the existing structures as barrier for protection of essential SSC against turbine missiles
 - the structures meet the acceptance criteria in SRP Section 3.5.3.

Turbine Missile Barrier

NuScale considered the following information in evaluating the barriers against potential turbine missile impact.

- NuScale considered the following missiles for evaluating barriers:
 - (A) Turbine Blade : Weight: 32.6 lbs.; Velocity 784 mph ; 1.41” equivalent diameter ; (at 120% overspeed)
 - (B) Turbine Blade with rotor fragment : Weight 52.6 lbs.; velocity 996 mph; rotor fragment width, 4.5”; (at 160% overspeed)
 - (C) Half of last stage rotor: weight 3079 lbs.; velocity 350 mph; 12” wide by 48” in diameter; (at 160% overspeed)
- Acceptance Criteria – Concrete barriers should be thick enough to prevent:
Back face scabbing
- Acceptable Impact analysis methods:
 - Empirical equations based on test results (SRP 3.5.3)
 - Finite Element analysis with validation

Turbine Missile Barrier

Staff Review of applicant information Summarized

- Using Empirical relations:
 - NDRC
(Based on Equivalent Dia.)
 - Penetration
 - Scabbing
- Other Formula:
 - Penetration
 - Scabbing
- Limitations of the empirical methods:
 - Range limitations (test data limitations)
 - Missile and target both considered as hard
- Finite Element Method:
(Based on blade Geometry)
 - Penetration
 - Scabbing

(Based on Empirical Formula)

Turbine Missile Barrier

Staff Review of applicant information Summarized

- FEM procedure is not previously reviewed by the staff for high speed turbine missile impact
 - Benchmarking of computer results
 - Reported computer results against relevant solid bullet test data
 - Expected results
 - Deformable missiles expected to provide less penetration depth
 - Larger Kinetic energy of impact expected to provide larger penetration depth



Section 3.5.3 - Turbine Missile Barrier

Open Item 03.05.03-1

- Global damage assessment
 - Staff review inconclusive
 - Need additional information regarding the missile model, analysis methodology and procedure
 - Assumption of an automobile impact as a surrogate to half of the last stage rotor missile impact (3079 lbs) is inappropriate due to significant dissimilarities in:
 - weight of the hard missile (auto engine weighs much less than 3000 lbs)
 - Rotor foot print is less than auto foot print
 - Missile impact velocity (350 MPH) is much higher than auto crash MPH

Conclusions

- Staff will discuss its findings and path forward with Nuscale in a public meeting on July 16, 2019.
- NuScale to provide additional summarized information to demonstrate the consistency of the results between empirical and FEM approaches.



TF-3 Testing Purpose and Feedback on TF-3 Demonstration Modal Testing

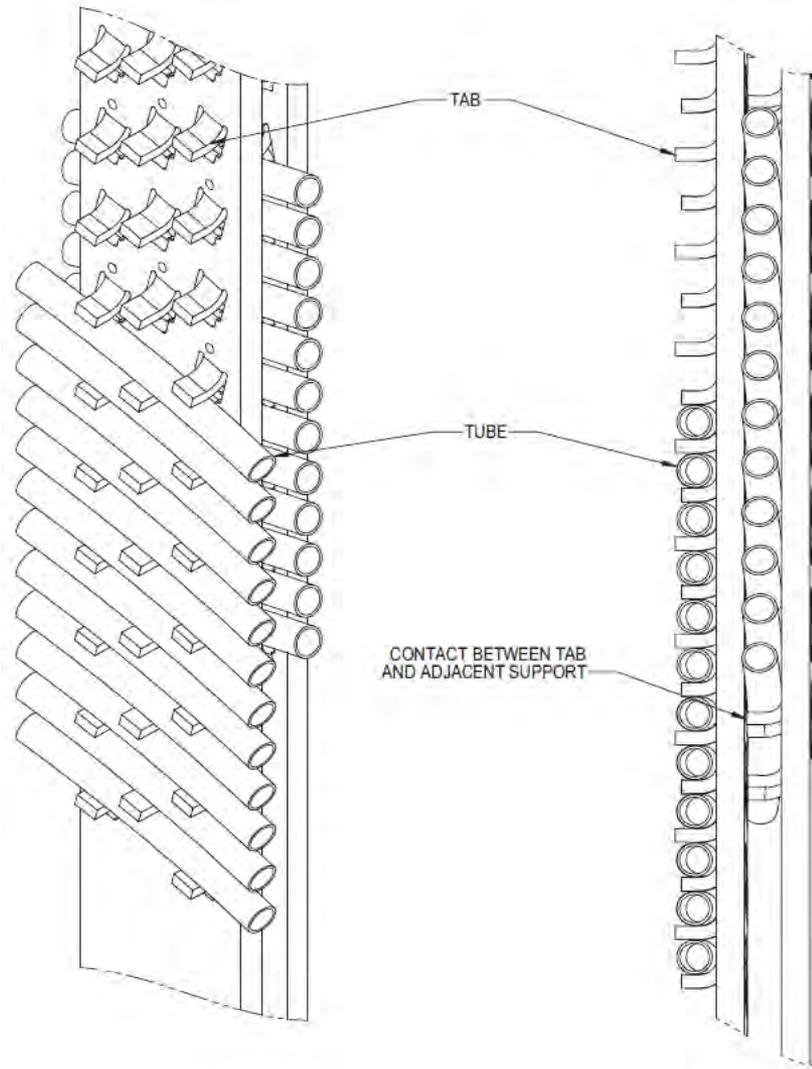
Yuken Wong

TF-3 Testing Purpose

- Obtain flow-induced vibration data for prototypical helical coil steam generator (HCSG) for validation of Fluid-elastic instability (FEI), vortex shedding (VS), and turbulence buffeting (TB) design analysis
- Specific objectives are:
 - Determine in-air and in-water natural frequencies and mode shapes of prototypic HCSG construction
 - Determine in-air and in-water damping values
 - Obtain data to characterize primary flow dynamic pressure fluctuations, SG tube and tube support vibration amplitudes
 - Obtain high flow rate vibration amplitudes to demonstrate margins to FEI and VS

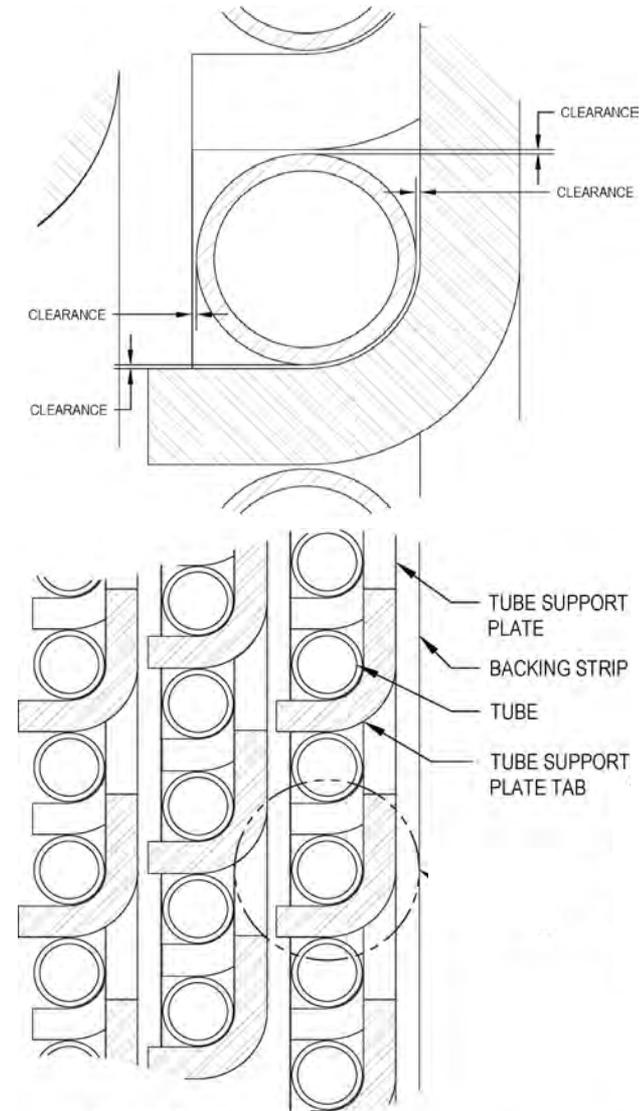
TF-3 Demonstration Modal Testing - FEI

- Outer 2 (Columns 12 and 13) of 5 tube columns were installed
- Half of long tube spans appear tight (clamped/fixed) and half appear loose (pivoted)
- Short tube spans appear tight



TF-3 Demonstration Modal Testing – FEI (cont'd)

- TF-3 was intended to replicate the thermal expansion of the tubes and support system and stiffer (clamped) boundary conditions at the supports
- However, TF-3 preloading system does not clamp all tubes into the supports due to clearances between tubes and supports
- Preloading system increases tube frequencies by less than 10%
- TF-3 will now test the looser (and therefore bounding) boundary condition without simulating the thermal expansion of tubes

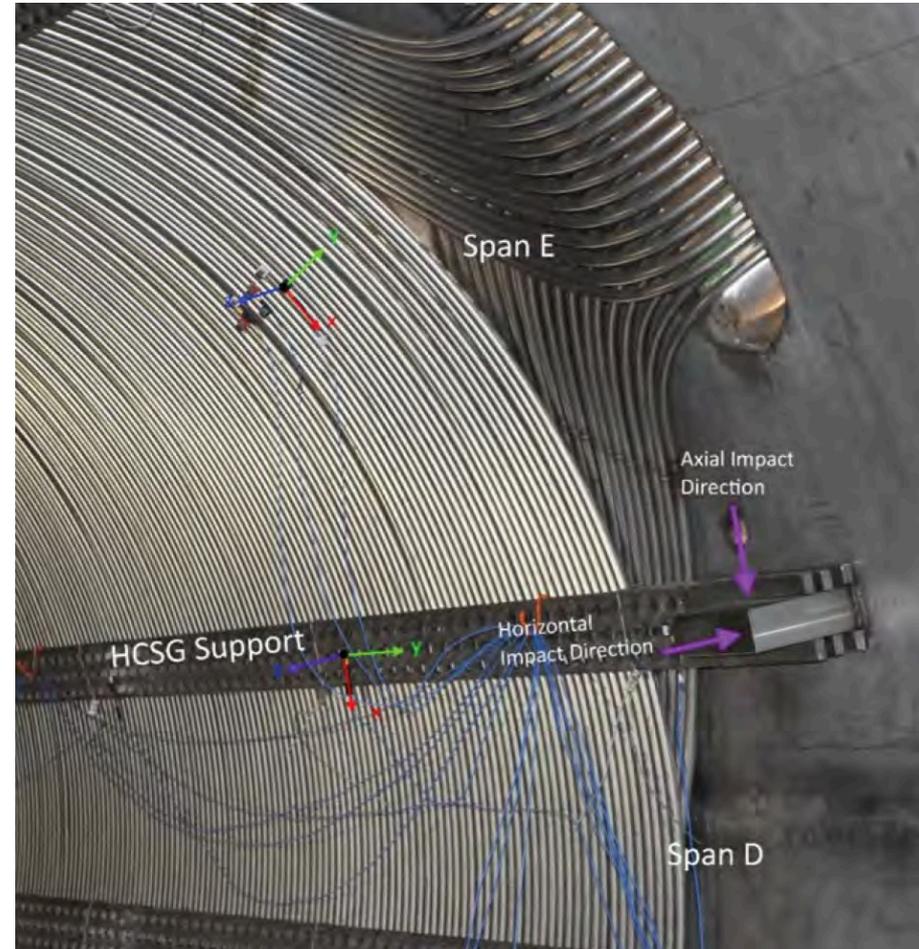


Analytical Prediction of Thermal Expansion - FEI

- NuScale presented thermal expansion calculations of selected tubes and supports
 - Thermal expansion will push the tubes down and radially out onto the tube supports
- Staff asked how other forces will affect the tube to support contact
 - Primary flow and gravity will also push tubes down
 - Secondary flow will push tubes radially out
- NuScale will quantify the effects of gravity and flow
- Staff will review the thermal expansion and gravity/flow effects

SG Tube Vortex Shedding

- Only lower tubes are susceptible to VS
- The feedwater transition bends have stiff geometry and support conditions
- Measured frequencies are much higher than previously calculated frequencies
- NuScale will soon submit an updated HCSG VS assessment which should show increased margin against VS



SG Tube Density Wave Oscillation and Thermal Fatigue

SG Tube DWO

- SG inlet flow restrictors (SGIFR) limit density wave oscillation (DWO) during normal operation
- TF-1 and TF-2 tests show DWO frequencies are very low compared to the SG tube resonance frequencies
- The DWO observed in TF-1 and TF-2 is not a concern for SG flow-induced vibration
- Minor differences in final SGIFR design and tested design will have insignificant effect on differential pressure

SG Tube Thermal Fatigue

- SG tubes are ASME Class 1 and fatigue analysis is required
- Staff has not reviewed a specific SG tube fatigue analysis
 - Staff is reviewing the SG tube plenum fatigue analysis
 - NuScale has provided a ASME Fatigue Screening Report identifying most critical fatigue locations.
 - Those select locations are being reviewed as the analyses are prepared.
- ITAAC 2.1-4.2 exists to confirm that ASME data reports have been prepared for ASME Class 1 components in Table 2.1-2.
 - Table 2.1-2 includes RCS integral RPV/SG/Pressurizer

Section 3.9.6 – Functional Design and Qualification, and Preservice Testing and Inservice Testing of Pumps, Valves, and Dynamic Restraints

Thomas Scarbrough

ECCS Valve Design Review

- 3 Reactor Vent Valves (RVVs) and 2 Reactor Recirculation Valves (RRVs) allow natural circulation for emergency core cooling.
- Each RVV and RRV has first-of-a-kind (FOAK) design arrangement of a main valve, inadvertent actuation block (IAB) valve, solenoid trip valve, and solenoid reset valve connected by hydraulic tubing.
- NuScale conducting ECCS Valve Design Demonstration Testing at Target Rock to satisfy 10 CFR 50.43(e).
- Ongoing NRC audit with onsite test observation during June 3 week.

ECCS Valve Functional Design and Qualification, and IST

- ASME Standard QME-1-2007 for qualification of ECCS valves as accepted with conditions in RG 1.100 (Rev. 3).
- ITAAC acceptance criteria for functional qualification of safety-related valves require Qualification Report (specified in QME-1 standard).
- Code of record is ASME OM Code (2012 Edition) as incorporated by reference in 10 CFR 50.55a
- NuScale to apply Appendix IV to OM Code (2017 Edition) for preservice and inservice testing of ECCS valves as 50.55a alternative.

ECCS Valve Review Next Steps

- Complete NRC audit of ECCS Valve Design Demonstration Testing to close SER Open Item.
- Review Design Specification audit follow-up items to close SER Confirmatory Item.
- Update SER as necessary for Phase 4.

Abbreviations

- ACRS Advisory Committee on Reactor Safeguards
- AOV Air-Operated Valve
- ASME American Society of Mechanical Engineers
- BPV Boiler & Pressure Vessel
- BTP Branch Technical Position
- CIV Containment Isolation Valve
- COF Coefficient of Friction
- COL Combined License
- CNV Containment Vessel
- CRB Control Building
- DC Design Certification
- DSRS Design Structure Response Spectra
- DCA Design Certification Application
- DWO Density Wave Oscillation
- ECCS Emergency Core Cooling System
- FEM Finite Element Method
- FWS Feedwater Piping
- FEI Fluid-Elastic Instability
- FOAK First of a Kind
- FSAR Final Safety Analysis Report
- HCSG Helical Coil Steam Generator
- HOV Hydraulic-Operated Valve
- IAB Inadvertent Actuation Block
- IST Inservice Testing
- ITAAC Inspections, Tests, Analyses, and Acceptance Criteria
- LBB Leak Before Break
- MSS Main Steam Piping
- MOV Motor-Operated Valve
- NDRC National Defense Research Committee

Abbreviations

- NRO NRC Office of New Reactors
- OM Operation and Maintenance
- PST Preservice Testing
- QA Quality Assurance
- RAI Request for Additional Information
- RG Regulatory Guide
- RCS Reactor Coolant System
- RPV Reactor Pressure Vessel
- RRV Reactor Recirculation Valve
- RVV Reactor Vent Valve
- RXB Reactor Building
- SER Safety Evaluation Report
- SRP Standard Review Plan
- SSCs Structures, Systems, and Components
- SG Steam generator
- SGIFR Steam Generator Inlet Flow Restrictor
- VS Vortex Shedding
- TB Turbulent Buffeting



United States Nuclear Regulatory Commission

Protecting People and the Environment

Safety Evaluation with Open Items: Chapter 15, “Transient and Accident Analyses”; Containment Performance

NuScale Design Certification Application

ACRS Full Committee Meeting
July 10, 2019

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

Agenda

- Unclear Open Items (UOIs)
- Staff Requirements for SECY-19-0036
- Chapter 15 Limiting Cases
- Critical Heat Flux Ratio (CHFR) Variance
- Long Term Cooling
- Chapter 15 Exemptions
- Return to Power
- Containment Structure
- Containment Heat Removal

Unclear Open Items

- Items without a mutually understood and clearly defined path towards resolution
 - Includes items for which there may be an agreement between NuScale and NRC, but documentation and supporting information is pending
 - UOI list changes frequently as new material is submitted by NuScale
 - List of UOIs has been reduced since Chapter 15 SER drafted

Status of Current Chapter 15 Unclear Open Items

Return to Power Analysis

- Recriticality while natural circulation is interrupted (15.0.5-1): Awaiting calculation and revised RAI response from NuScale
- Excluding event scenario of ejected rod with margin to stuck rods (15.0.6-6): NuScale RAI response pending, expected to support decision that CRDM housings robust with enhanced features, precluding need to evaluate long term effects such as recriticality within design basis, but retaining short term analysis
- Boron redistribution during ECCS (15.0.6-5): Continuing dialogue regarding volatility correlation and mixing assumptions
- Nuclear analysis parameters in Return to Power analysis (15.0.6-3): Expected to be addressed in revised calculations
- Decay heat level precluding a return to power and addressing EOC ECCS return to power (15.0.6-4): NuScale analysis of EOC ECCS return to power pending

Long-Term Cooling

- Statements in LTC technical report that analysis demonstrates adequate cooling for 30 days (15.0.5-2): NuScale reevaluating analytic basis for statement and potential revisions to statement

Topical Report Methodologies (15.0.2-2, LOCA; 15.0.2-4, non-LOCA)

- NRELAP5 v1.4 and revised base model (LOCA TR): Staff recently received NRELAP5 v1.4 and revised base model and is evaluating
- SG heat transfer uncertainty (non-LOCA TR): Staff awaiting revised NuScale analyses expected to demonstrate results insensitive

Chapter 15 reanalysis from design and methodology changes

- ECCS logic changes (15.0.0.4-1, 15.6.5-1): Submittal of analysis revisions expected August 30
- DHRS logic changes (15.0.0.4-1, 15.6.5-1) Submittal of analysis revisions expected August 30
- NRELAP5 v1.4 (15.0.2-1): Submittal of analysis revisions expected August 30

Staff Requirements for SECY-19-0036

- The inadvertent actuation block (IAB) valve is part of the emergency core cooling system (ECCS) valve and is designed to prevent high-pressure RPV blowdown when the dc power supply (EDSS) is lost to the ECCS trip valve.
- Chapter 15 analyses don't assume single failure of the IAB to block
- The Commission determined:

“The staff should review Chapter 15 of the NuScale Design Certification Application without assuming a single failure of the inadvertent actuation block valve to close.”

Chapter 15 Limiting Cases

Figure of Merit	Limiting Event	Limiting Value	Acceptance Criteria	Margin
RCS Pressure	15.2.8, Feedwater System Pipe Breaks Inside and Outside Containment	2164 psia	AOO: 2310 psia PA: 2520 psia*	AOO: 6.32% PA: 14.1%
SG Pressure	15.6.3, Steam Generator Tube Failure	1806 psia	AOO: 2310 psia PA: 2520 psia*	AOO: 21.8% PA: 28.3%
MCHFR – VIPRE	15.4.3, Control Rod Misoperation (Single Rod Withdrawal)	1.614	1.284	25.7%
MCHFR – NRELAP5	15.6.6, Inadvertent Operation of ECCS	1.41	1.13	24.8%
Collapsed Liquid Level (CLL) Above Top of Active Fuel	15.6.5, LOCA	< 5 inches	0 inches	N/A

* Although these events are classified as postulated accidents (PAs), they meet the acceptance criteria for anticipated operational occurrences (AOOs).

Chapter 15 CHF Correlations

Non-LOCA Events (Non-Loss of RCS Coolant)

- NSP4 CHF Correlation
- VIPRE-01 Sub-Channel Analysis Code
- NRC Approved Methodology

LOCA and LOCA-Like Events (Loss of RCS Coolant)

- High-Flow CHF Correlation and Low-Flow CHF Correlation
- NRELAP5 Systems Code
- Methodology Under NRC Staff Review

Chapter 15 CHF Correlations

Critical Heat Flux Ratio Variance

Correlation

Critical Heat Flux

- NSP4 correlation for non-loss of RCS coolant analyses
- High-flow and low-flow correlation for loss of RCS coolant analyses

Initial Conditions

Analysis Input Condition Biasing

- RCS temperature, RCS Flow, Pressurizer Pressure, etc.

Methodology

Analysis Code

- VIPRE-01 sub-channel code calculates MCHFR during the period of interest only
- NRELAP5 systems code calculates MCHFR continuously throughout the event

Modeling

- Difference in peaking factor and heat flux assumptions

ECCS Long-Term Cooling Capability and 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
- In-vessel downstream effects evaluation performed in DCA Section 6.3
 - Staff found NuScale evaluation acceptable

LOCA Exemption

Appendix K – LOCA Exemptions

- ANS-71 replaced with ANS-73 with RELAP5 standard precursors
- Post LOCA – PWR reflood and refill (phenomena are not to be encountered)
 - post-CHF regime heat transfer correlations
 - Baker-Just Metal Water Reactions
 - Clad swelling and rupture

GDC 27 Exemption

- General Design Criterion (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

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

Return to Power Analyses

- DCA Part 2, Tier 2, Section 15.0.6, Revision 2, presents a DHRS cool down, retaining single phase natural circulation with ECCS valves opening at the maximum return to power, which is thought to bound all three cooldown scenarios
 - Maximum, Core average return to power 10% rated thermal power (RTP), equilibrium return to power of approximately 2.5% RTP
 - MCFHR limit met
- A potential DHRS return to power when water level drops below the riser (UOI 15.0.5-1) still needs to be evaluated to ensure SAFDLs are met
- EOC, ECCS return to power case (UOI 15.0.6-4) still needs to be evaluated to ensure SAFDLs are met
- Boron redistribution and a potential return to power at times than EOC still needs to be resolved (UOI 15.0.6-5)
- Completion of staff's confirmatory analyses

Containment Performance

- Containment Structure
 - Peak Containment Pressure/Temperature Review
- Containment Heat Removal

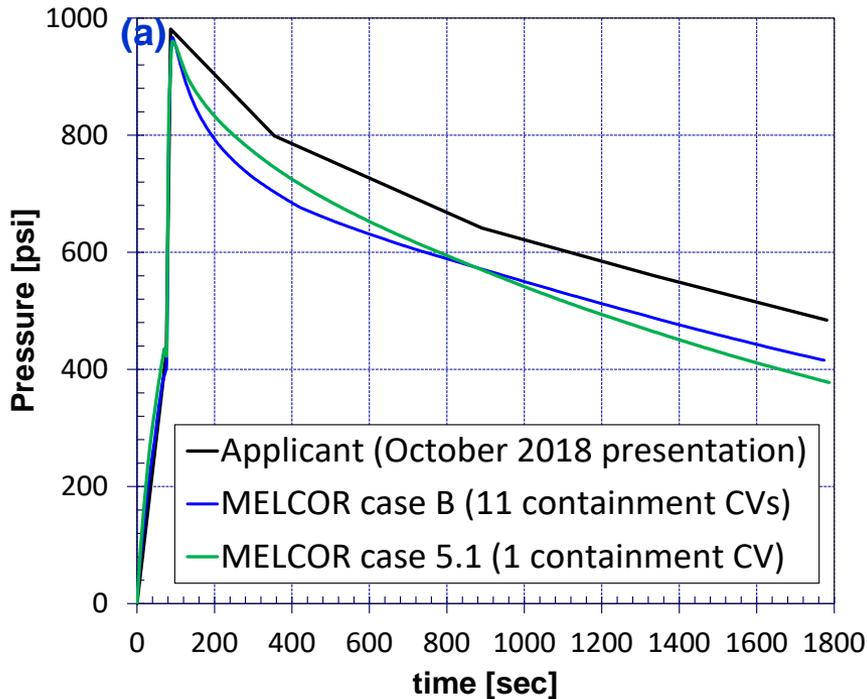
6.2.1.1: Containment Structure

Scope of the Staff Review

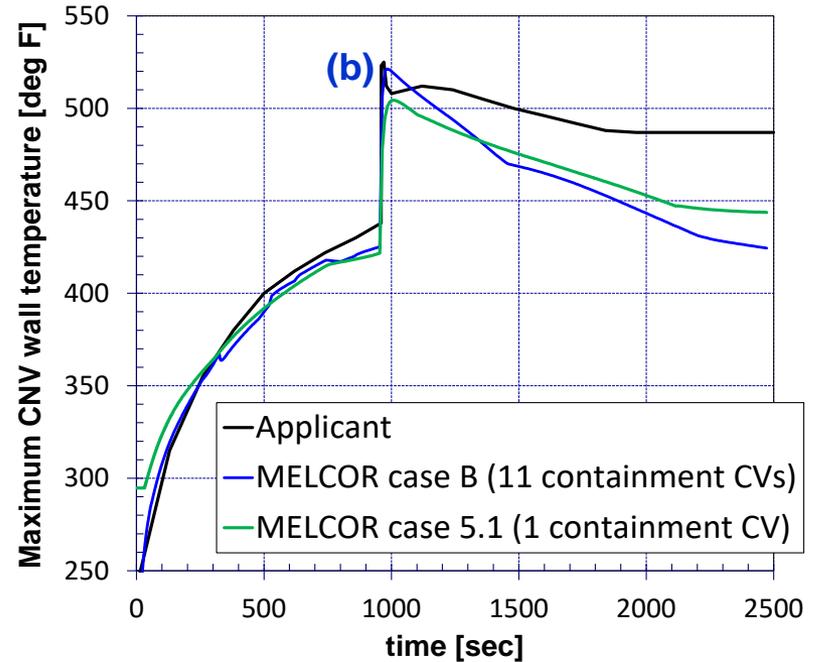
- NuScale FSAR Section 6.2.1.1 (DCA Part 2, Tier 2)
 - Containment Response Analysis Methodology (CRAM) Technical Report (TeR)
- Key regulatory requirements
 - Sufficient margin in containment design pressure/temperature (GDCs 16 & 50)
 - Pressure reduction by 50% from its peak value within 24 hours (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

Staff & Applicant's Results for the Limiting Pressure & Temperature DBA Scenarios



(a) Pr. Limiting: Inadvertent RRV Opening



(b) Temp. Limiting: RCS Injection Line Break

Applicant Peak Values by NRELAP5	986 psia	526 °F
Staff Confirmatory Peak Values by MELCOR	967 psia (a)	521 °F (b)
FSAR CNV Design Values	1050 psia	550 °F

Section 6.2.2 – Containment Heat Removal

NuScale Approach for Long-term Cooling

- Limit containment debris to latent only
- No strainers subject to clogging
- Low fiber limit precludes a filtering bed at the fuel inlet
- Exclude material-chemistry combinations expected to cause chemical effects

Finding

- Using DCA specified debris limits, staff determined there is reasonable assurance that consequential debris 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 and conservative debris limit

GDC 40 Exemption

- Periodic inspections of the containment heat removal surfaces (as required by GDC 39, in accordance with ASME BPVC) will identify 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
- 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 could not 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

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