Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title:	Advisory Committee on Reactor Safeguards NuScale Subcommittee: Open Session
Docket Number:	(n/a)
Location:	Rockville, Maryland
Date:	Wednesday, June 19, 2019

Work Order No.: NRC-0389 Pages 1-257

NEAL R. GROSS AND CO., INC.
Court Reporters and Transcribers
1323 Rhode Island Avenue, N.W.
Washington, D.C. 20005
(202) 234-4433

DISCLAIMER

UNITED STATES NUCLEAR REGULATORY COMMISSION'S ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The contents of this transcript of the proceeding of the United States Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards, as reported herein, is a record of the discussions recorded at the meeting.

This transcript has not been reviewed, corrected, and edited, and it may contain inaccuracies.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

+ + + + +

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

NuSCALE SUBCOMMITTEE

OPEN SESSION

+ + + + +

WEDNESDAY

JUNE 19, 2019

+ + + + +

ROCKVILLE, MARYLAND

+ + + + +

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

COMMITTEE MEMBERS:

JOSE MARCH-LEUBA, Chair

RONALD G. BALLINGER, Member

DENNIS BLEY, Member

CHARLES H. BROWN, JR. Member

MICHAEL L. CORRADINI, Member

VESNA B. DIMITRIJEVIC, Member
JOY L. REMPE, Member
PETER RICCARDELLA, Member
GORDON R. SKILLMAN, Member
MATTHEW W. SUNSERI, Member

ACRS CONSULTANT:
STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:
MIKE SNODDERLY

CONTENTS

P R O C E E D I N G S

(8:30 a.m.)

MEMBER CORRADINI: Why don't we get started. So this is the second day of our three-day committee meeting on NuScale, I'm not going to run through the pro forma discussion except to mention that we have a closed line for NuScale subject matter experts that should be open.

Could the NuScale -- back in Corvallis?

MEMBER BLEY: That you're on the line in Corvallis.

MEMBER CORRADINI: That you're on the line.

PARTICIPANT: NuScale's on the line.

MEMBER CORRADINI: Okay, all right. And then we have the public line which is on mute mode. And then I just want to remind everybody since I was the offender last time, turn down your volumes. Turn down your cell phones. Put them on mute, something, so they don't talk back to you.

MEMBER BROWN: Are you going to do that?

MEMBER CORRADINI: And then other than that I want to just proceed. We have an agenda today where we are going to first go over the topical report, and then transition into Chapter 15 from the applicant. And tomorrow we'll have the whole day to talk with staff about Chapter 15.

So I'll turn it over to Dr. March-Leuba who is going to lead us through the two days.

CHAIR MARCH-LEUBA: Thank you, Dr. Corradini. So we're going to start with the stability report, topical report, which -- but before you start, let me give a summary for my colleagues of the way I think this is going to go.

And so the stability report is an excellent technically speaking. It is -- I found a microphone, yes.

It's expanding the methodology for BWR stabilities that we've known for 40 years now, and 40 years ago we would not be able to have done this. But now we know all the problems and issues and we know how to arrest them.

And better than that we have a particular mass of people that can review your work, understand it and intuitively say yes or no, whereas, it's not a one person saying something, it's a community of people saying.

So based on that methodology, we have identified that the limited stability concern for NuScale is natural circulation oscillation inside the core because of buoyancy concerns. So when you generate a lighter mass on the riser you get more flow, which then cools down the riser and you can have oscillations. They become the normative oscillation. And they analyzed it using all this methodology that we know and they have decided there is no safety significance to it.

And there could be a possibility of instabilities if we get into boiling in the riser and they prevent that with an exclusion region. It's one that accepted a long time one of the solutions in BWRs in which you don't allow boiling in the riser.

So, I think that it's perfectly covered and I would like it to go fast so we can we move to the difficult part of which are the next one on Chapter 15.

Now one thing I'm concerned about is on top of the primary system instability we have secondary system instabilities. So you have these long thin tubes in the steam generator where the steam, the bubbles are inside of the tube and what is all the experience we have is with the bubbles outside of the tube.

So there's a large, very large friction two phased flow of friction we calling that stability domain that is very large and very destabilizing. So to try to stabilize that NuScale has put, as we always do, an inlet restriction, and we will talk about those details in the closed session. But the possibility exists with non negligible likelihood. We're not talking 10 to the minus 7, we're talking maybe 50/50, that the steam generator will operate with flow oscillations in the secondary.

And I'm not sure we covered the solution on the open session or the closed session, but the staff on NuScale have reached a solution and concluded that this would not be of safety significance to the core and would not violate GDC 12. And during the closed session when we have our discussion, I would like to reach an agreement among yourselves and that is we agree with the conclusion.

MEMBER BLEY: What's the potential from, I would imagine there would be problems with steam generator supports and vibrations and that.

CHAIR MARCH-LEUBA: That's what I want to discuss with them. So I noticed in the presentation, NuScale has maybe one bullet in the whole presentation, so I will ask you to speed up through the next four slides, so we can get to the question and answer area.

And we may have to do it in the closed session, so it's up to NuScale to tell us when to stop and move to the closed session, because things get proprietary real fast.

NuScale, please go ahead.

MR. PRESSON: Thank you for that opening, Dr. March-Leuba. And good morning, I'm Matthew Presson with licensing at NuScale Power, project manager for this topical report. Presenting today will be Dr. Yoursef Farwila, our stability expert, and supported by myself in licensing, and Ben Bristol, the supervisor of system thermal hydraulics.

And with that as requested, we'll get on with the presentation.

DR. FARWILA: Thank you. Good morning, everyone, Mr. Chairman and all the members. I was introduced already and I already know most of you. So let me speak through the presentation.

Our agenda is very simple today. The topic is simple. Just make a quick introduction and tell you about the stability solution type and describe how we investigated it with theoretical, numerical and experimental benchmarks and how we use these tools in a procedure and methodology to demonstrate the stability of the module, summarize, and try to save as much time for your questions.

Start with the main message, the NuScale power module design was found to be stable in its entire range of normal operation. This is unconditional stability as long as we are in normal operation.

CHAIR MARCH-LEUBA: But you are referring to the primary system, correct?

DR. FARWILA: The primary system, yes, anything that affects the core. So it is not going to be unstable or growing power oscillations or flow oscillations that go through the core. Outside of that, other systems are essentially outside of the scope or touching in the scope and we'll be having a chance to discuss how much that connection is.

CHAIR MARCH-LEUBA: Can you remind us what GDC, the general design criteria 12 says about oscillations?

DR. FARWILA: GDC 12 says you're not allowed to have oscillations unless you can detect and suppress them so that the SAFDLs cannot be violated with whatever thermal limits or other limits, cannot be violated.

CHAIR MARCH-LEUBA: The way I read it is you can either detect and suppress them or they cannot violate SFADLs. So you may, according to GDC 12, you may allow oscillations that do not violate SAFDLs. Specified acceptable fields and limits.

DR. FARWILA: Right. That's what I thought I said.

CHAIR MARCH-LEUBA: Mm-hmm, yeah. Okay.

MEMBER BLEY: I know our role here is reactor safety. On the other hand, if you sell me a plant with steam generators that starts coming apart because of this, I'm not going to be very happy with you; neither is anybody else. Although that's not what we delve into, maybe sometime you can talk about what the effects on steam generators could be.

CHAIR MARCH-LEUBA: Yeah, I think we would want to save that discussion for the closed meeting, so I'm going to speed you up so we can have more time for that one.

DR. FARWILA: All right. Okay, part of the main message also is to stress that outside of normal operation, for any reason, the reactor can be destabilized when the riser flow is voided, but even then these unstable flow oscillations are limited by nonlinear effect, so the magnitude of those oscillations is going to be limited and there is still not going be any critical heat flux violations.

So now we identified where the stability threshold is. That's already protected by scram upon loss of riser inlets of cooling. We put a margin for that eventuality, so, essentially, the hard stability solution is to do nothing.

CHAIR MARCH-LEUBA: Going back to bullet number two, you say that even if unstable flow oscillations were to develop SAFDLs wouldn't be violated. Is this because of a specific characteristic of NuScale or is it because NuScale operates with so much CHF margin that even if the CHF oscillation never hit limits?

DR. FARWILA: Actually, the critical heat flux ratio is going to increase. You are running a lot of ahead of the presentation, but that's all right. When you void the riser that gives you a much greater feedback.

So for a unit of enthalpy added you get a lot of density difference and that density contrast is what drives the flow and the instability. So the flow itself starts to become higher, so the average flow gets higher and the magnitude of the oscillation is bounded by in the upper part of the flow the voids would collapse.

So once the voids collapse you cannot

increase more than that and the overshoot in the other side would be about the same magnitude.

CHAIR MARCH-LEUBA: My question was a little loaded in the sense that if in the future we have a 20, 30, 50 percent power increase, power upgrade that would eat into the CHF margin, you would still have this effect.

DR. FARWILA: Yes, of course.

CHAIR MARCH-LEUBA: But it would have to be evaluated.

DR. FARWILA: Definitely. Any change in the way the reactor is operated is specified that we will have to repeat the stability analysis. But the same tools that we have with the phenomena in it would be covered.

CHAIR MARCH-LEUBA: Thank you.

DR. FARWILA: So these conclusions that are presented in the main message, they are based on extensive first principles experimental and computational. We just don't believe codes blindly. We have to verify them by other ways, other methods. We have to have first principles understanding.

MEMBER BLEY: Is this, the trip on loss of subcooling in the riser, is that based, was that always there or is that something that's been added and might have been moved into Chapter 7 as part of the instrumentation?

DR. FARWILA: It's always there.

MEMBER BLEY: Okay, thank you.

DR. FARWILA: So just going back, when we first started with this it was a surprise. I mean why do you need to have a stability analysis at all? It's a PWR and PWRs, usually in the FSARs you have a paragraph talking about stability and dismiss it.

But in our particular case you can see that we have published reports with the experiments that you can have instabilities in natural circulation flow. And you can see here that there are several configurations with horizontal cooler, horizontal heater and you have vertical cooler and vertical heater and vertical both.

And experiments could show that only in the first one that you can have instabilities but none of the others, but you could say the configuration of the experiment is different from what we have. I mean you can, not scale it properly and we don't know much about it, but anyway this last configuration is very similar to the NuScale instability.

But instead of taking that for granted, of course an investigation of the modular stability is in order and in order to really verify that and understand it. So the way we started with this, that first box in there in the original if you see it in the printout it was painted gray.

So we were in the gray area when we know only that we just have a design. We have expertise. We know the theory, but it's still in the gray area until we go to more colorful boxes with developing a code and have independent models to verify and experimental data to benchmark it and understand it.

And once we are confident with the tools we have, we go to the next step and analyze the module in different modes of operation, different powers, just all the range that you have and essentially do perturbations to see if perturbations would grow. And also I'll tell you more about that later, look at stability during transients, not just from a steady state.

And so we concluded from this that reactor is stable within normal domain and most important thing, identified what is the threshold for instability which is riser voiding. That gets us to the green area where we have a stability solution which, luckily, does not introduce any additional hardware as the threshold is already protected by the module protection system.

Like any new reactor with new phenomena we start with a PIRT. That PIRT committee met for a week and was extra conservative, everything counts, everything, a lot of times were ranked high.

MEMBER CORRADINI: When did this occur?

DR. FARWILA: That occurred in 2015 or so, so well, well before anything started.

MEMBER CORRADINI: Is that on the docket, the PIRT?

DR. FARWILA: I don't think so.

MEMBER CORRADINI: Okay, right.

DR. FARWILA: Right. It's probably not very interesting reading.

MEMBER CORRADINI: Because you brought it up so I thought I'd ask.

DR. FARWILA: Yes, of course.

We have a PIRT, you could say a post-code PIRT. There is a pre-code PIRT and a post-code PIRT. The post-code PIRT is worth reading. It's part of the topical report.

MEMBER CORRADINI: Okay.

DR. FARWILA: Right.

MEMBER BLEY: It's a topical report?

MEMBER CORRADINI: That's what we have as a separate.

MEMBER BLEY: That is the one we have, okay.

DR. FARWILA: Yes. So that's you could say more the relevant one, not the one we started with, with always conservatively as human things.

So anyway, we looked at, first, the principles just to look at the riser only and see what happens considering the cold leg as boundary condition. And we look at the stability trends with power because we got surprising results that the higher the power, the more stable the reactor is. We did not see this before in PWRs.

So you could say do I believe my code? That's a taboo. We cannot believe a code without understanding, so the understanding comes from first principles and using other models and things like that.

So this analysis from first principles is one of the important legs of this project and it essentially, it informs the design of the stability experiments as well because you can go to the facility and you don't know what to measure and how to measure it. Unless you know something from first principles, there is always this organic relationship between an experiment and the theoretical understanding that comes with it. They feed off each other.

So anyway, regarding the PIRT, all medium ranked phenomena were treated as highly ranked. There was no complacency in this regard. So important thing in theoretical understanding of instability is that if you have positive feedback it's not allowed, of course we don't allow it, and you cannot operate system with positive feedback.

So we have negative feedback that's engineered in, but this negative feedback if it's too strong and delayed it can overshoot in the correction and that can be oscillatory. And so a negative feedback that is delayed and sufficiently strong will be, can become in principle unstable. And when you look at our module, the feedback is negative.

I mean if you have a perturbation increasing flow, it cools off the riser and that reduces the density head that drives the flow essentially correcting that perturbation.

MEMBER BLEY: On your last slide when you said you treated all ranked phenomena as highly ranked, what did that mean to you? Did you require experiments to verify those things, or what did you do?

DR. FARWILA: Mainly put models that I have high fidelity to represent these phenomena.

MEMBER CORRADINI: But you did do, you had done -- I want to, here, get to the experiments, so.

DR. FARWILA: We did experiments, yes. Actually, when the presentation was shortened, we are not presenting as much. I can talk to them though.

MEMBER CORRADINI: Okay.

DR. FARWILA: Yeah.

CHAIR MARCH-LEUBA: Now it would be nice if you'd give us a two-minute summary of the experiments.

DR. FARWILA: Of course, I will.

Okay, so we said that this feedback is delayed because it takes time to fill the riser with that different temperature in order to make an effect. And the strength of the feedback is of course is related to the thermal expansion and that's what creates the density difference.

So if you have a different fluid that expands more or if you have boiling that will be stronger or if you take the whole thing to a different planet where the gravitational constant is different, it would behave differently. I didn't hear a laugh, but we actually had models where I changed the g in order to find out -- I needed to change it by a factor of 10 in order to make things unstable.

So all right, getting to the main tool, it's a code called PIM. And you can see the geometrical simplification from this iconic picture to more of an advanced schematic, more accurate schematic to just the essential loop of really what happened. So it's not looping like this, of course, you know, it's looping like that.

But here are the essential element, a small core almost like a point in the bottom of a tall riser, and you have the cold leg filled with a helical steam generator acting as the heat sink.

CHAIR MARCH-LEUBA: I don't remember the details. Can you go back? Does PIM consider heat conduction from the riser to the heat exchanger directly through the wall?

DR. FARWILA: Yes, it does.

CHAIR MARCH-LEUBA: Okay, so it's capable of doing that when the water level drops below the top of the riser, it will still transfer heat?

DR. FARWILA: When the water level drops to --

CHAIR MARCH-LEUBA: Below the riser.

DR. FARWILA: -- below the riser, then PIM is no longer applicable because you don't have established continuity for natural circulation.

CHAIR MARCH-LEUBA: All right, but I think that's -- some of the calculations we've seen you still establish heat transfer from the pink to the blue, even though you don't have natural flow?

DR. FARWILA: That must be in NRELAP5.

CHAIR MARCH-LEUBA: Okay, thank you.

And while we have this here, could you describe how representative with experiments, the experiments that you were talking about and just tell us how representative they were with respect to this reality?

DR. FARWILA: The experiments were done in the NIST-1 facility --

CHAIR MARCH-LEUBA: Mm-hmm.

DR. FARWILA: -- which is a scaled hydraulic loop that looks very much like this except for the height is -- was it one-third?

PARTICIPANT: Mm-hmm.

DR. FARWILA: And the diameter is much smaller than one-third, so it's more of a long vessel with a riser and --

MEMBER CORRADINI: We'll see that in July.

DR. FARWILA: Oh, oh. Wonderful.

MEMBER BLEY: Out there? I thought you said they did that at NIST.

MEMBER CORRADINI: NIST is the name of the facility at Oregon State.

MEMBER BLEY: Oh, that's right. I had forgotten that.

MEMBER CORRADINI: NIST is not NIST. NIST is the name of the facility.

MEMBER BLEY: It's another NIST, got it.

MEMBER CORRADINI: NuScale Integral something or other.

MEMBER BLEY: Yeah. Yeah, we -- okay.

CHAIR MARCH-LEUBA: Yeah. So, basically, it's representative of the NuScale, you have different time constants. You have to scale on the results.

MEMBER CORRADINI: Just so the members remember, those that went, we won't ask for hands -- that there were prior experiments done in Corvallis with a non-scaled facility. This was then rebuilt as we were visiting in '15 or '16. I can't remember which summer. They were rebuilding the facility to have it based on some scaling discussion which we're going to actually talk about later today in terms of the scale.

DR. FARWILA: All right. We can address

the scaling in maybe in the closed session, the scaling of the experiments and --

CHAIR MARCH-LEUBA: But in summary, the results of the NIST instability experiments did it show it was stable or unstable?

DR. FARWILA: It showed it too stable, almost dead stable.

CHAIR MARCH-LEUBA: Okay.

DR. FARWILA: Okay.

CHAIR MARCH-LEUBA: He's disappointed.

This is what I will ask you to, is skip through the next four slides.

DR. FARWILA: Yes, I'm going to skip these.

MEMBER CORRADINI: Before you skip, so I want to understand. So PIM is not a homogeneous equilibrium model; it's non-equilibrium between the phases?

DR. FARWILA: Non-equilibrium mechanical and thermal.

MEMBER CORRADINI: So you have different temperatures and different velocities.

DR. FARWILA: Yes.

MEMBER CORRADINI: Okay. And you didn't first do a non -- a linear analysis, a simple linear analysis to show the regions of stability or instability?

DR. FARWILA: We did and we published it.

MEMBER CORRADINI: Was that the published -- is that the reference that you had referenced a couple slides ago, or is it some other reference?

DR. FARWILA: I did not put -- no, no. This reference here is not ours.

MEMBER CORRADINI: Okay, so I'd be curious about the reference, but you did do a linear analysis to begin with?

DR. FARWILA: Yes, we did.

MEMBER CORRADINI: Okay. And then my next question, you can do it when you want. I'm curious about the regimes of stability and instability between the linear analysis and the non-linear analysis. Because in other applications for even just gas instability flows in high-pressure gases, which was done for gas-cooled reactor designs, you find that the regions are hard to predict.

So I'm kind of curious how they laid on top of each other, or at least if you looked at these comparisons.

DR. FARWILA: Right. In the beginning we thought maybe you have a frequency domain code.

MEMBER CORRADINI: Right.

DR. FARWILA: But since we know how to write time domain codes that are not hampered by the problems we use to have with numerical dispersion diffusion and all these things, we thought we would just have a time domain code with the small perturbations it's totally equivalent to a frequency domain.

MEMBER CORRADINI: Sure.

DR. FARWILA: But outside of the normal operation we get limit cycles and non-linear circulation in everything.

MEMBER CORRADINI: Okay, fine. Thank you.

DR. FARWILA: So, yeah.

Okay, so I'm going to skip through the model since you already have the topical report and have read this presentation before. Just one little thing here. We looked at what was not modeled and we do not model the pressurizer. It does not contribute to the momentum.

And there are certain things that we anchor to the more detailed code, NRELAP5, so we don't have to model them directly. There are simple things also that we know is conservative not to model, like we take a pneumatic riser because whatever heat exchange is going to go into them, the transient and things like that, so anything that's not modeled is either conservative or very low ranking.

MEMBER BLEY: Conservative always bothers me because it's conservative with respect to something but maybe not some other things. So it's conservative with respect to the generation of oscillations?

DR. FARWILA: Yes.

MEMBER BLEY: Is that what you mean by conservative?

DR. FARWILA: Of course.

MEMBER BLEY: Okay. Well, of course is nice.

DR. FARWILA: Actually, things that we say conservative we've verified it to be conservative. Like for this methodology we say we do not model heat exchange through the riser, but it's modeled in the code and you can see that there's a little bit of stabilization when you have it, so we say, "Okay, don't worry about it. Let's just continue," and use the historical calculation when we did not have that feature.

DR. SCHULTZ: So with all of the features that you've listed in the previous slides then, in order to identify that conservatism you did sensitivity studies associated with each of the elements that you've described?

DR. FARWILA: This is why there's been a lot of sensitivity studies, yes.

DR. SCHULTZ: Okay, thank you.

DR. FARWILA: Yeah. We had three years.

DR. SCHULTZ: Okay, that helps.

DR. FARWILA: All right, so how the analysis was done, in the linear regime you have SS, meaning steady state, so we have a steady state at a certain power and for each power we have flow because it's natural circulation. You cannot move power and flow independently without a pump. So when we say in the range of power, we also mean in the range of flow.

So we perturb a steady state and look at the result. You will see the flow and power oscillating and the oscillations are decaying, and that by looking at the manner in which these decay we know what the decay ratio is. So that's how we get the stability parameters, the decay ratio and the oscillation period.

And we find that as you said before, we have unconditional stability in the entire operation range and a curiosity there that decay ratio decreases when you increase power and also when you increase exposure. And we find --

CHAIR MARCH-LEUBA: And by exposure you really mean the reactivity feedback is larger at the end of cycle?

DR. FARWILA: Yes.

CHAIR MARCH-LEUBA: So increasing reactivity feedback is good.

DR. FARWILA: When you have large negative feedback it's stabilizing. And that also was done from first principles in the paper we published in Nuclear Science and Engineering. It was also in the NURETH-16. It was invited for archiving later on.

MEMBER CORRADINI: The Chicago meeting.

DR. FARWILA: I don't remember where because I didn't manage to go.

MEMBER CORRADINI: Yeah, okay. Fine.

DR. FARWILA: All right. These observations all are in agreement with independent reduced older models so we have additional way to trust it.

All right, the application methodology, we apply these perturbations to steady state to get the decay ratios. We vary power widely from five to hundred percent. Actually, we do one percent also, to see what could happen. But you cannot get CHF response or worry about SAFDLs below 20 percent, but we say 5 since we started with that.

We do beginning of cycle and end of cycle and if we have any reason to suspect any more limiting with later feedback, maybe closer to the beginning of cycle but not the beginning of cycle itself, we examine that as well.

And we have considered assumptions and not only for the moderator temperature coefficient, but also for decay heat fraction because decay heat fraction is just like fission energy except for it has a zero feedback from the moderators. So if the moderator feedback is positive then it is more conservative to have zero decay heat.

But if you have negative moderator temperature coefficient then it is more conservative to assume the largest possible in decay heat, like you have been operating at high power for some time and then you drop the power.

And I have a question.

MEMBER RICCARDELLA: Yeah. On the previous slide, could you go back to that, please? And in the bottom sort of bullets you have decay ratio decreases with power, period decrease. Is that decreases with increasing power?

DR. FARWILA: Yes, with increasing power and increasing cycle exposure.

MEMBER RICCARDELLA: Okay, and period also decreases with increasing power.

DR. FARWILA: Period also decreases with increase in power, yes.

MEMBER RICCARDELLA: Okay, thank you.

DR. FARWILA: You're welcome.

All right, so outside the normal operation, the only transient we -- we did a scoping of all the AOOs in a generic kind of way and the only thing that became unstable was a depressurization transient because only then you can void the riser. And so we could get limit cycles and we could see that the CHFR, the critical heat flux ratio actually increases, not decreases.

So these stability conclusions are generic but we intend to, are committed to confirm them if needed, like if a plant upgrades to, or increased rate of power or the plant operation changes, you have anything different that may affect normal rate of temperature, the activity or the fuel design itself that could increase the rated flow or things of that sort, so anything that would affect the nature of natural circulation or feedbacks will be re-examined.

MEMBER REMPE: So you said you did some sensitivity studies with respect to power uprates. Do you have any feel for how much of power uprate might affect this conclusion? Like is ten percent enough, or did you do a sensitivity study in that area?

DR. FARWILA: Okay, what, fortunately, for this module, when you increase power you become more stable.

MEMBER REMPE: Okay.

DR. FARWILA: So power upgrades as long as it does not create voiding in the riser we are covered. But they have been analyzed. I did not analyze them myself, but they have been analyzed.

MEMBER REMPE: So you don't have other individuals who've done analyses to look at how much of a power uprate would cause voids in the riser?

DR. FARWILA: No, I don't have that number.

MEMBER CORRADINI: It can be calculated, but I'm more interested in your 50-megawatt machine than other machines.

DR. FARWILA: All right.

MEMBER BLEY: You've said that several times that it was kind of a surprise that as power goes up, the oscillations go down.

DR. FARWILA: Yes.

MEMBER BLEY: Because the code said so is one thing, physically how do you explain that phenomena?

DR. FARWILA: When the code said so, everything stopped. We had to understand it from first principles. And I probably should get into that in the closed session.

MEMBER BLEY: Closed session, that's fine.

DR. FARWILA: All right. With the long-term stability solution, you already said that before. It's a originally seclusion. It's one-dimensional not two-dimensional like PWRs. And we just protect against void in the riser and that's essentially that.

MEMBER CORRADINI: Okay. I think you have to, for the members, explain what you mean by one- versus two-dimensional.

DR. FARWILA: Okay. For boiling water reactors we have a power flow map, so essentially one acts as power, the other acts as flow. And the operator can take the state of the reactor, you can vary power by putting control rods. You can change flow by changing the pump or pump valves.

But in this reactor module we don't have a pump, so if you change power, you change flow with it. So essentially have, it's like a straight line. It's power versus flow is unique curve, so that makes it a one-dimensional seclusion.

MEMBER RICCARDELLA: You said a unique curve.

DR. FARWILA: Yes. I mean later on when we say we want to examine the transients, so we consider the power and flow as stage variables. So in the phase space you access points other than that unique line only through transients. So you have non-zero time derivatives that take you somewhere else and maybe there are islands of instability we haven't seen or anything like that, so that's really the rationale for examining transients.

All right, so do I speed through this one since we already know what the conclusions are and go to questions? Because I think I'm almost out of time.

MEMBER CORRADINI: Well, I want you to -- well, I'm not sure. I don't really care about that. I want to make sure I understand the experiments that were done. So where is the point that we want to talk about the experiments, in closed session or now?

DR. FARWILA: Best in the closed session.

MEMBER CORRADINI: Okay.

DR. FARWILA: There's a lot of more

proprietary information in the experiments.

MEMBER CORRADINI: Okay.

DR. FARWILA: Okay.

CHAIR MARCH-LEUBA: So anyone have questions by the members?

I'm trying to speed up through the open session so we can go into the closed session and talk freely.

MEMBER CORRADINI: Thank you all.

DR. FARWILA: Thank you.

CHAIR MARCH-LEUBA: Okay, Bruce. And rules apply, you guys are authorized to go five miles over the speed limit.

MR. BAVOL: Okay, good morning. My name is Bruce Bavol. I'm a project manager for the NRC. This is staff's presentation for the open session for the Stability Topical Report, Safety Evaluation along with 15.9, the Stability section of the Chapter 15.

To my right, Dr. Ray Skarda. To his right, Dr. Peter Yarsky will be presenting, Rebecca Karas is the branch chief for the Reactor Systems branch. I'll mention briefly that the staff had 62 RAIs, all responses were resolved and closed.

Our full committee is scheduled for July 10th, and the staff plans to issue the final SER in

August, late August, and then the dash A approved version of the SER and topical report in November 2019.

With that I'll turn it over to the authors to go over the outline and the rest of the presentation.

MR. SKARDA: I'm Ray and I'll just start. These are the eight items. The outline comprises these eight items, the regulatory criteria that were used to perform the stability topical, stability methodology topical report review as well as 15.9 DCD review, the long-term stability solution that was proposed by the applicant, the instability modes and the phenomena that are considered important to the NuScale design.

The applicant developed as you saw their own computer code PIM to perform the stability analysis. The stability acceptance criteria and uncertainty. Worst rod stuck out analysis which we'll talk about a little bit later was not part of the original submittal. We'll get to that. And then finally we'll summarize the staff's conclusion with respect to the review of the topical report as well as the DCD 15.9.

So the five regulatory criteria that are important to stability are shown here, and these are also the ones that are called out in the standard review plan 15.9A which is specific to NuScale. This is talked about a little bit before, so briefly GDC 10 says don't exceed the SAFDLs.

GDC 12, no uncontrolled out powers, uncontrolled power oscillations. GDC 13, 20 and 29, compressing this would be these relate to instrumentation controls, systems and functions that ensure that the long-term stability solution accomplishes its job of protecting against instabilities. And that long-term stability solution is shown on the next slide. That's the exclusion region.

The main thing here as you heard is that the exclusion region criteria is simple and straightforward. And if you look at the figure that the whole main point of that is that you have to maintain a five-degree submargin, a five-degree Fahrenheit riser subcooling margin for operation.

If that is not the case, the modular protection system precludes against instabilities and forces a trip, generally either a hot leg temperature trip or pressurizer pressure, low pressurizer pressure trip.

MEMBER CORRADINI: So let me ask, I don't remember. But where does NuScale operate normally in terms of riser subcooling compared to the forbidden region and the allowed region?

MR. SKARDA: Yeah. So it has to be -- their subcooling margin is the riser and it needs to be five degrees Fahrenheit of larger.

MEMBER CORRADINI: But what is it in normal operation. That's what I don't --

CHAIR MARCH-LEUBA: I seem to remember it was 35 Fahrenheit.

MEMBER CORRADINI: That's what I thought. I thought it was in the 10s that you used. Okay fine.

MR. SKARDA: Oh, what they're operating at. Okay, I'm sorry.

MEMBER CORRADINI: And then my second question is, when I get into the forbidden region and I lose my safety margin, I go to the possible instabilities, is it the riser that is avoiding or the possible instabilities are essentially a SAFDL issue?

What I'm trying to understand is am I worried about the void into the core or I'm reading about SAFDLs being violated because I have an unstable flow?

MR. SKARDA: The subcooling margin

protects the SAFDLs through minimum, the NCF --

MEMBER CORRADINI: Okay, fine. That's what I thought. Okay, thank you. Thank you very much.

MR. SKARDA: Next slide.

So okay, so there's in terms of excluding instabilities the applicant looked at several types of instability modes, identified those that were relevant to the NuScale design. In the DCD, I think they, as you saw earlier, the one that's more important to them is this natural circulation with density wave oscillation characteristics riser dominant and that will be discussed in more detail in the closed session.

The applicant's findings are consistent with the staff's findings based on an independent PIRT that was also performed by the staff. Next slide.

So as I mentioned, the applicant developed their own computer model and computer code to perform the stability analysis, PIM. PIM's evaluation model includes simple models for a thermal-hydraulics reactor, kinetics, fuel thermal mechanical response and steam generator to peak conduction and heat transfer, which you saw earlier.

And it's been validated against the NIST-1 stability tests which were integral tests. The decay ratio is the principle acceptance criteria with respect to steady state operation at various power levels. The main conclusion's here is that the K ratio is insensitive to variations in most of the important phenomena over, really, the range of analysis related to the NuScale design.

The decay ratio acceptance criteria affords adequate margin to account for biases and uncertainty and that includes the numerical effects. Primarily, dissipation is what I would call the main one, dispersion. It's highly dissipative at least with steady operation, so any change in phase is not, it's just not going to be an issue from what we're seeing. Next slide.

So I mentioned this worst rod stuck out analysis, and this is something that is in combination with an analysis that was done as part of 1506, the return to power. There was some combination with that instability analysis.

From the stability standpoint, it provided an analysis at intermediate pressures from operating pressure down to where ECC would actuate. You'll see both in the SER for 1506 and so forth, just to provide context that event is actually decay, DHRS overcooling event.

So you're really driving, you're trying to drive this thing into sort of a recriticality mode. So trying to make sure I say the right things here in terms of the -- we talked about the strong moderator feedback with in terms of recriticality which is important in driving that power or that recriticality.

But the other side of that is it really damps oscillations then. So in terms of an event, you'd ideally like something that's going to drive the initial event hard to that point and then you would like to have weak moderator feedback in order to have those oscillations persist longer.

And so that will be again discussed --

CHAIR MARCH-LEUBA: And we see those details in Chapter 15 this afternoon and tomorrow.

MR. SKARDA: Yes. That's correct. That's correct.

CHAIR MARCH-LEUBA: But just because we're seeing -- because it's an overcooling event because you cannot reach criticality unless you are, I believe it's under 200, I mean very cold, then your coolant has shrunk and you are below the riser and therefore you cannot have natural circulation anymore.

So you really your parenthetical 18-- we talk about this is the closed session for 15.

MR. SKARDA: Yeah. In fact, the next -- so basically before that occurs, before he sees actuation, for example, there's actually, actually plenty of subcooling margin in this particular case. Next slide.

And so as you were saying, I don't have to say it. Yeah, once he sees this actuates you've broken the natural circulation flow pattern and you're really in a pool boiling kind of mode, but at very, very low power.

So the bottom line is this analysis by the applicant demonstrated that flow oscillations are not safety significant. Even if they had large amplitudes, it's really low power. Next slide.

So the stability report conclusions with respect to PIM approval for performing stability analysis, PIM's a simple model but it's anchored in upstream, high fidelity models to improve accuracy. The decay ratio is highly insignificant to variations in important phenomena.

The PIM predictions, both steady state and transient, have been confirmed by staff through independent confirmatory calculations. And so PIM is -- staff's found PIM to be acceptable for performing the safety related stability analysis for the NuScale module. Next slide.

In terms of the long-term stability solution, the primary instability mechanisms were properly identified by the applicant and confirmed by independent staff TRACE analysis.

During power operation of the NuScale power module, NuScale power module is very stable. The exclusion region-based, long-term stability solution is effective in preventing the reactor from becoming unstable during normal operation and including effects of AOOs. And then as we mentioned, the potential instability during return to power with respect to worst rod out is not a safety concern and the five GDCs related to safety are met.

So briefly summarizing the review of section 15.9 of the DCD, analysis of perturbed steady state conditions demonstrate that decay ratio remains well below the acceptance criteria for power levels greater than five percent of rate of power.

The certain transient analyses result in new studies, stable state conditions, and in others that would exceed the hot leg trip or low pressurizer pressure trip, enforce the exclusion region.

With respect to the AOOs, those are the

AOO classes that you find in other sections of 15.9, increase in heat removal, decrease in heat removal from a secondary system and so forth. The first two, really, the bounding events that were performed there were feedwater flow increases and decreases.

I won't go through -- some of these were dispositioned, just questions, we can take those. But the bottom line is the long-term stability solution is effective in preventing the occurrence of instability and again the five GDCs important to stability are met. And I believe that's it.

We have some backup slides.

CHAIR MARCH-LEUBA: Probably the answer to my question is let's wait for the closed session, but anything you can say in open session with respect to the secondary side, the steam tube instabilities?

DR. YARSKY: Let's defer that to the closed session. But the staff has a slide specifically on that topic.

CHAIR MARCH-LEUBA: I know. And, but in open session can we say that you're satisfied with the proposed solution by NuScale or do you still have reservations?

DR. YARSKY: I would say in terms of secondary side instability the staff was able to determine that it does not pose any safety concerns.

CHAIR MARCH-LEUBA: Okay. That's what I wanted to put on the record for the open session. Thank you.

Any more questions -- okay. At this point we're going to transition to a closed session, but before that we're going to ask for comments from -- can we open the phone line?

Any members from the public that want to make a comment on the open session?

We're waiting to see if the phone line is open.

MEMBER CORRADINI: Is anyone on the public line? Please at least acknowledge that you're out there.

MEMBER RICCARDELLA: Mike is still trying to verify.

MR. LEWIS: That's the worst thing. I've never heard such dancing in my life trying to keep away from the major problems and listening to details, details, detail and then talking them into oblivion instead of answering them strongly.

Further, there was a heck of a lot of noise over this line. I presume everybody's shuffling papers and hoping to cover me over.

But this is exactly what I'm talking about. You're going into a secret session, a closed session, just to make it easier for the licensee to cover all the problems that should be out there in the public. I object strenuously and I plan to have some more legal objections down the pipeline.

Thank you for listening to me. Good bye.

CHAIR MARCH-LEUBA: Just a moment. You came on to the open line a little late so we didn't hear your name. Can you state it for the record?

MR. LEWIS: Marvin, M-A-R-V-I-N, Lewis, L-E-W-I-S. I live in northeast Philadelphia. My email is marvlewis@juno.com. No period between the V and the L and juno is spelled like the goddess, not the city in Alaska.

CHAIR MARCH-LEUBA: Thank you very much.

Any other comments from the open line?

MS. FIELDS: Yes. First of all, I'd like to know when the open session returns. Is that after lunch?

CHAIR MARCH-LEUBA: In principle, the open session will start at 11:00 a.m. if we can make up some time, for the open session for Chapter 11 and we'll go one hour before lunch. Chapter 15, my mistake. So in principle 11:00, but we may be late.

MS. FIELDS: That's not the time frame on the agenda. So you think you're coming back to open session at 11 o'clock?

CHAIR MARCH-LEUBA: That's what we're shooting for.

MR. SNODDERLY: This is Mike Snodderly. I'll send you a email when we're going to start back up in open session. But right now we plan to follow the agenda and it'll be sometime around 11:00 a.m. But I'll send you an email and I'll send it to all that -- to that interested members of the public list that I have.

MS. FIELDS: Okay, so I do have a comment. I'm glad you brought up the issue of an increased power and the possibility of a different type of fuel because you may be aware there's a parallel process.

You have the NRC design certification process which for the most part is based on a lot of documentation, a lot of review. I've listened to quite a few of the NRC's NuScale meetings and it's been very, very valuable. But there's a parallel process, the UAMPS process.

But I'd appreciate it if someone would close their -- go on mute so there's not a lot of background on the phone.

CHAIR MARCH-LEUBA: Thank you, Sarah.

MS. FIELDS: All I hear is shuffling of papers and I can hardly hear.

MEMBER CORRADINI: There's no shuffling here. It must be on other lines.

MEMBER BLEY: And you're very clear coming in.

MS. FIELDS: Oh, okay. So this parallel process, they anticipate a 20 percent of power uprate, but they haven't outlined how they would get to that power uprate and implying that it would be Utah Associated Municipal Power Systems, which is the only entity that currently plans to submit a COL application using the NuScale design.

So they're talking about this, they're not talking about how exactly they would get the power uprates whether it would be a design change or whether it would be part of the COL application or whether it would be some future license amendment request.

There are other things, if you go online there are articles about new types of fuel and NuScale anticipating using some of these new types of fuel.

UAMPS has already announced when they're going to get a NRC approval of their COL application, when they're going to start site preconstruction activities, when they're going to complete the construction, when they're going to start operating their first power module and when they're going to start operating the complete array of 12 power modules.

The Department of Energy, one of their staff people at a conference in Salt Lake already announced, "Hey, this is happening." So there's a lot of effort out there and also calling this a carbon-free power process.

And anyone who knows anything about the nuclear fuel chain knows that carbon is used from the moment that the uranium industry starts going out and exploring for uranium and produces uranium, and all along the fuel chain it uses fossil fuels to make the fuel.

And there's -- so calling it a carbon-free power project is disingenuous and disinformation. So on this you should be aware there's other parallel world out there of public relations and trying to get the public to accept this no matter what the cost might be.

I live 20 miles from a very low-income community in Utah where the ratepayers there still do not exactly how much this is going to cost. So whatever's going on in your departments it comes down to a ratepayer getting a bill for a project that over the long term no one has a real perception of how much it's going to cost each and every ratepayer over a period of 50 years. So that's some of the background of community concern.

Thank you.

MR. SNODDERLY: Thank you, Ms. Fields.

So we're going to go into a closed session now. So, Makeeka, could you please close the open phone line.

I'd like to ask Bruce and other members of the staff, Rebecca, if there's anybody here from the staff that does not have a need to know, we'd like to ask you to leave the room now. And the same thing --

MEMBER CORRADINI: Well, I think we had scheduled a break to allow this to happen easier.

MR. SNODDERLY: Okay, if you want, I think we're ready to go if you want to --

MEMBER CORRADINI: Can we at least take five minutes to make sure? That's what I would suggest. Let's just take five minutes, make sure the open line is closed.

MR. SNODDERLY: Okay.

MEMBER CORRADINI: All right. I think

we're going to lose members anyway to use --

MR. SNODDERLY: Yes, you are. Okay.

MEMBER CORRADINI: So five minutes.

(Whereupon, the above-entitled matter went off the record at 9:28 a.m. and resumed at 11:30 a.m.)

MEMBER CORRADINI: Okay. Why don't we come back in session. So, our plan is to start off with the Applicant in terms of Chapter 15.

And we'll go through an hour. And then break for lunch wherever we get in an hour. Jose?

CHAIR MARCH-LEUBA: Go for it.

MEMBER CORRADINI: Matthew, you're it.

MR. PRESSON: All right. Thank you again. To reintroduce myself, I am Matthew Presson, Licensing Specialist with NuScale Power, and Project Manager for Chapter 15.

We are here today to discuss Chapter 15 of the NuScale design certification application. And to cover the unique aspects of how NuScale approaches transient and accident analysis in our design.

The presentation today will be provided primarily by the people joining us here at the table. We have myself.

We have Megan McCloskey, Thermal Hydraulic Analyst. We have Ben Bristol, Supervisor of System

Thermal Hydraulics. And Paul Infanger, Licensing Specialist with Chapter 15.

And potentially joining us on the phone as needed, will be Dr. Pravin Sawant, Supervisor of Code Validation and Methods. Dr. Brian Wolf, Supervisor of Code Development.

Dr. Selim Kuran, Thermal Hydraulic Software Validation. Mark Shaver, Supervisor of Radiological Engineering. And Greg Myers, Licensing Specialist for Containment.

All right. For the scope of our discussion today, there's a lot of information that is summarized in Chapter 15, where we deal with postulated transients and events.

There are also a number of topical reports that define methodologies that we use in Chapter 15, that we have not yet presented to the ACRS.

Therefore, our presentation today will be focused on the material provided in the FSAR, citing Chapter 15 itself, and the results of those methodologies which are again, provided in that chapter.

We will discuss the high level details of these methods so that we have a basic understanding. And we are happy to summarize those topical reports here.

But any detailed discussion of those methods will need to wait until we present the topical reports for your review later this year.

CHAIR MARCH-LEUBA: Are we expecting those in the October time frame?

MR. PRESSON: I believe October. Yeah. So this is a big old slide. It is a copy of slide 77. Later on in this presentation, those acronyms and events will be covered in the presentation that follows.

But we did want to open this entire discussion on Chapter 15 with the summary of doses, just to cover the health and safety of the public. And how it applies to NuScale design.

As you can see, the dose consequences for our postulated events, and even the beyond design basis events listed there, remain very low in the NuScale power module.

So for all of our discussions on postulated events that follow, the NuScale power position is that we have a safe design.

MEMBER SKILLMAN: What is the purpose for the bolding on the rem TEDE for dose?

MR. PRESSON: So again, we'll get into

those details later. But those are bolded to show the highest doses that we had.

MEMBER SKILLMAN: Thank you. Okay, I thought that something, yes.

MR. PRESSON: This is a quick review of what we'll be going over today. Our design overview, as well as the Chapter 15.

The intellectual assumptions for the Chapter 15 analysis. So, some thermal hydraulics analysis, methodologies. Some selected transient results. Our radiological analysis.

Some on Chapter 6.2.1 containment response analysis. That's where we follow it up with the -- from the loco model it is built off of. And the long term cooling.

And with that, again, our overview of safety and non-safety systems with Megan McCloskey.

MS. McCLOSKEY: Thank you. All right, so before we dive into Chapter 15, I think for members of -- particularly after the discussion yesterday and this morning, are familiar with the NuScale power module design and the internal reactor design with the core and steam generator and pressurizer in one vessel.

When we get to the transient results later this afternoon, we'll see examples of module protection system actuations in response to events.

And for Chapter 15 design basis analysis, we're primarily concerned with confirming the effective operation of the safety-related systems for safety related decay heat removal. We have two systems, the emergency core cooling system, and the heat removal system.

So to set the stage, on those we have just a couple of slides that I think we can go through quickly on the operation.

CHAIR MARCH-LEUBA: I know we say this all the time, but people are on the phone, and they're probably not hearing you.

MS. McCLOSKEY: Okay.

CHAIR MARCH-LEUBA: Can you just speak or get the microphone, maybe -- people are -- away from the paper.

MS. McCLOSKEY: Yeah. I was concerned about the paper shuffling.

All right. For the emergency core cooling system design, we have two reactor recirculation valves on the side of the reactor pressure vessel. And three reactor vent valves on the top head of the containment.

And these -- when demanded, these valves open to establish a boiling condensing flow path to transfer decay and residual heat from the reactor pressure vessel by venting steam from the reactor pressure vessel into containment where it condenses on the inside surface of the containment wall.

And then is transferred through the containment wall to the reactor pool ultimate heat sink. The --

CHAIR MARCH-LEUBA: Yesterday we asked this question when we were talking about the ECCS valves. I had read into a figure, what is the water level after you open the ECCS valve, and assuming initial inventory normal now.

Where does the water level settle? I read ten feet above active fuel.

MS. McCLOSKEY: Yes. That's the --

CHAIR MARCH-LEUBA: That's usual?

MS. McCLOSKEY: The nominal equilibrium water level.

CHAIR MARCH-LEUBA: Okay. Thank you.

MS. McCLOSKEY: Yeah. Typically there's more than 20 feet above the top of active fuel when the ECCS valves open under a normal expected progression.

MEMBER CORRADINI: And how far -- how far above active fuel are the RRVs?

MS. McCLOSKEY: I don't have that number.

MEMBER CORRADINI: It's around eight feet, six feet.

MR. BRISTOL: It's six feet long.

MEMBER CORRADINI: Okay. All right.

CHAIR MARCH-LEUBA: Eight feet? You think it's eight feet?

MEMBER CORRADINI: No. They said six feet.

MR. BRISTOL: Six.

CHAIR MARCH-LEUBA: Oh. I thought it was four. But you know better.

MEMBER BLEY: And you said that if a coil breaks level at about ten feet?

MS. McCLOSKEY: Yes.

CHAIR MARCH-LEUBA: And the ECCS valves will open, based on the instrument, that level instrumentation on the containment. Is that correct?

MS. McCLOSKEY: Yes. The containment -- or sorry, the ECCS valves are actuated by the module protection system either due to a high containment level signal. Or after if there's a 24 hour loss of AC power supply.

CHAIR MARCH-LEUBA: But that -- that's not the MPL. I mean, it's --

MS. McCLOSKEY: Yes.

CHAIR MARCH-LEUBA: The timer -- you lose DC power to the solenoids at 24 hours.

MS. McCLOSKEY: If DC power is available, then after 24 hours the load for maintaining the ECCS -- to the ECCS valve solenoids is shed.

CHAIR MARCH-LEUBA: Yeah.

MS. McCLOSKEY: So that's the timer.

CHAIR MARCH-LEUBA: So is it -- really the activation is really on the water level.

MS. McCLOSKEY: Yes.

CHAIR MARCH-LEUBA: And we had a discussion, I wanted to put it on the record again, about the probability of failure of that level instrumentation in the containment.

Which is an advanced sensor, which has not been used in nuclear reactors before. And it is a complex sensor. It's a rhythm-based sensor which we haven't seen any details, but it's likely to have, be a digital instrument.

Probably microprocessor based. So, it would be subject to common core failures. So all four level sensors may fail if it's a digital system.

Or you have to consider a common core failure from the digital point of view. And to complicate things, the containment is always empty.

So you're not exercising that instrumentation. So, it can fail and you never know it's blind.

So, we are going to have to address that thoroughly in Chapter 7 next time it comes here.

MS. McCLOSKEY: Right. In the scope of the Chapter 15 analysis, since the module protection system and the instrumentation are safety related, we do assume that they function.

CHAIR MARCH-LEUBA: Right. But it's more of a Chapter 7, Chapter 19 issue.

MS. McCLOSKEY: Yes.

CHAIR MARCH-LEUBA: But, I just wanted to put the concept on the record again.

MS. McCLOSKEY: Yes.

MEMBER CORRADINI: Just so I remember, so there are four different level sensors for the containment water level?

MS. McCLOSKEY: Four -- there are four channels, I think.

MEMBER CORRADINI: Four channels.

MEMBER BLEY: And if this instruments are safety related, and I thought they were, do we have any source of safety-related electric power that supply them?

MS. McCLOSKEY: No. The --

MEMBER BLEY: You don't. Okay.

MS. McCLOSKEY: Because if the -- if the DC power that's supplied from the highly reliable supply is lost, then our -- it is postulated to be lost.

The safety systems actuate in the NuScale design. So, the safety --

MEMBER BLEY: Well, a level sensor won't actuate. It --

MS. McCLOSKEY: No. But the -- the power to the ECCS valve solenoids will wake up.

MEMBER BLEY: They will. Yeah. That's right.

MS. McCLOSKEY: And so then eventually the valves will open to establish cooling, so.

MEMBER BROWN: And just to make sure I understand. The radar-based system that we -- that I think Jose is talking about, is actually during normal operation, measuring the level up in the pressurizer.

Isn't that correct?

MEMBER CORRADINI: They're different

things.

MS. McCLOSKEY: Yeah. That's -- I think you're ---

MEMBER CORRADINI: Two different things. Inside vessel, outside vessel. You're talking --

MEMBER BROWN: I'm inside the reactor vessel.

MEMBER CORRADINI: They're -- but that's not what we're asking about.

MEMBER BROWN: But it's outside.

CHAIR MARCH-LEUBA: Yeah. I'm talking containment.

MEMBER BROWN: You're talking about -- oh, you're talking about the containment water.

CHAIR MARCH-LEUBA: So we have four sensors in the pressurizer, four sensors in containment.

MEMBER BROWN: Okay. I didn't know you were talking -- I just kind of had a disconnect.

MEMBER CORRADINI: I'm still -- I'm still awake. That's all it should be.

MS. McCLOSKEY: So actually it's on containment level.

MEMBER BALLINGER: I mean, we're also making an assumption here, you are anyway. That when the containment doesn't have any water in it, those sensors are not working.

MEMBER BLEY: That's true. We don't know the answer to that.

MEMBER BALLINGER: I mean, most electronics have diagnostics that run all the time and things like that.

So, we have to be careful when we talk -- when we get to Chapter 9 that we don't make the --

CHAIR MARCH-LEUBA: Chapter 7.

MEMBER BALLINGER: Chapter 7 rather.

MEMBER BROWN: Well, I mean, there -- the reason I'm asking, the con -- the radar base one is measuring the pressurizer level are above the reactor vessel.

And they're, at least based on the picture, I thought I remembered, they're going down to measure the pressurizer level up in that upper part of the reactor vessel.

For the containment ones, are they up on that upper head area outside the pool? Are they -- are they located somewhere else?

I mean, it's a --

MS. McCLOSKEY: I'm not sure of the sites.

MEMBER BROWN: Well, you said that the water level in the reactor within the containment is about ten feet above the reactor core.

MEMBER BLEY: After everything settles down.

MEMBER BROWN: After everything settles. Are you still depending on those? I mean, that's a pretty long shot for the radar detectors to be measuring the water level in the containment.

MEMBER BLEY: They're supposed to be measuring it all the way down at the bottom.

MEMBER BROWN: That's a -- what I'm saying. That's a long shot.

MR. BRISTOL: This is Ben Bristol. So, yeah. A couple of points. I think as the ACRS is -- we've described before, there's performance requirements that are established based on the safety analysis for the instrumentation.

We've documented what those performance requirements are. And it is yet to be demonstrated by the final sensor selection that, yes, those performance requirements can be met.

In addition to the common cause failure, there is consideration given to the sensor design as an input to the diversity and defense in depth analysis of the MPS system to ensure that common cause sensor failure is either analyzed within the bounds of the Chapter 15 analyses that are otherwise ensured that it's a low probability.

CHAIR MARCH-LEUBA: That's a very good answer. Thank you. We just put in a marker that whenever Chapter 7 comes back, or Chapter 19, we'll bring you back again.

MR. BRISTOL: Okay.

MS. McCLOSKEY: Can we go onto the next slide? The second system for decay heat removal is the Decay Heat Removal System that removes heat after a loss of normal secondary site cooling.

When the Decay Heat Removal System is actuated, the containment isolation valves, the main steam and feed water lines close.

And the actuation valves on -- for the Decay Heath Removal System open to establish an alternate heat removal path through the steam generators so that energy is transferred from the primary side to fluid inside the steam generator tubes.

That steam is transported through the Decay Heat Removal System actuation valves over to the condensers that are located in the reactor pool. Then the condensate is transferred back to the steam generat -- to the steam generators.

MEMBER BLEY: My copy of slide nine shows water in the containment. That is wrong. Right? Your copy doesn't. It's all white.

The copy they gave us shows water in the line.

MS. McCLOSKEY: No. There's no water in containment. That's the -- that's the pool.

MEMBER BLEY: That's slide nine.

MS. McCLOSKEY: That's the pool.

MEMBER BLEY: Oh. You're right.

MS. McCLOSKEY: That's the pool.

MEMBER BLEY: Never mind.

MEMBER CORRADINI: White is air. It's steam.

MEMBER BLEY: Never mind. Never mind.

MEMBER BROWN: Oh, talk about a shot.

MEMBER CORRADINI: White is like you.

MEMBER BALLINGER: But you are right in one sense. There's water -- our slides show water in the vessel.

MS. McCLOSKEY: Inside the reactor vessel.

MEMBER CORRADINI: It was supposed to be in --

MEMBER BROWN: In the reactor vessel, yes.

MEMBER BALLINGER: It's light blue here, dark blue in there.

MS. McCLOSKEY: Yes.

MEMBER BROWN: It's very light blue.

CHAIR MARCH-LEUBA: Through all the documentation there are references to the old and the new actuation logic for DHRS. Will you tell us what is the actuation logic for the DHRS?

What signals tip it? What signal tells MPS to trip the DHRS valve? To open it?

MS. McCLOSKEY: There are a couple of signals. One of them is high secondary side steam pressure.

Loss of power also actuates the valve. And --

MR. BRISTOL: High RCS temperature.

MS. McCLOSKEY: High RCS temperature.

MR. BRISTOL: High RCS pressure.

MS. McCLOSKEY: And pressure.

CHAIR MARCH-LEUBA: Oh, high pressure inside the vessel.

MS. McCLOSKEY: Um-hum.

CHAIR MARCH-LEUBA: High temperature inside the vessel, or high temperature outside in the secondary.

MS. McCLOSKEY: Right.

CHAIR MARCH-LEUBA: Okay.

MS. McCLOSKEY: Indications that secondary site cooling is not effective.

CHAIR MARCH-LEUBA: The language indicates that you use to trip on more things. Then I assume that you have identified that they're not necessary?

MR. BRISTOL: So, -- this is Ben. The signals used to be combined. There was two functions, and we'll get into this in a little bit more detail later.

But there was an isolation function that was required to mitigate some events for the secondary side. And then there was a DHR actuation that was required.

The analysis that we did showed that those two didn't necessarily need to come at the same time. And as part of the start up procedure that ended up being a consideration that was fairly limiting, particularly at the lower temperature conditions.

CHAIR MARCH-LEUBA: Um-hum.

MR. BRISTOL: So the decision was to decouple the secondary isolation as its own function, so that on conditions that we knew that we needed to preserve inventory, secondary isolation would occur.

But not immediately cause DHR actuation. And not continue to require DHR to be active for the MPS logic.

So what it does, is it allows for DHR to actuate and then clear once we have established cooling based on the trip signals.

MEMBER BROWN: So you could not actuate DHS, the Decay Heat Removal System totally. You just isolate the main steam isolation valves?

MR. BRISTOL: That's correct.

MEMBER BROWN: And the feed water valves?

MR. BRISTOL: Feed water isolation valves.

MEMBER BROWN: And the feed water isolation valves.

MR. BRISTOL: Yes.

CHAIR MARCH-LEUBA: And only if the secondary steam starts heating up, then you need additional cooling.

MR. BRISTOL: That's correct.

MEMBER BROWN: So the other path is not established until you met some other preconditions?

MR. BRISTOL: That's correct.

CHAIR MARCH-LEUBA: That's in the area.

MEMBER BROWN: Okay. All right. Thank

you.

CHAIR MARCH-LEUBA: And I assume all the analysis in Chapter 15 represent the new logic, not the old logic.

MR. BRISTOL: Well, it's a -- that's a great question. It's one of the design changes that's being implemented in the current analysis effort.

I think the NRC has asked us to describe that.

CHAIR MARCH-LEUBA: Are you redoing everything?

MR. BRISTOL: We are in the middle of updating the FSAR analysis. They're very similar from a transient progression to what we've seen before.

MEMBER CORRADINI: But this is -- this is what is identified in many cases as Rev 3.

MR. BRISTOL: That's correct.

MEMBER BROWN: That will also be reflected in Chapter 7 Rev 3 relative to this differentiation of what logic is required?

MR. BRISTOL: That's correct.

MEMBER BROWN: Okay.

CHAIR MARCH-LEUBA: Let me just stipulate here before we go any further. That NuScale has so much margin that for any error it doesn't matter what you analyze, you're always going to get the same answer, really good.

And so, but it is nice to do the paperwork properly. Okay.

MEMBER CORRADINI: And you'll continue to cheerlead that.

MS. McCLOSKEY: All right. So, in terms of the scope of Chapter 15, the design basis initiating events consider internal events that could affect a module operating at power.

The Chapter 15 analyses that we do are performed for a single module response. In terms of shared systems for the NuScale design, the reactor pool ultimate heat sink is the important shared system there.

And so our long term cooling analyses consider effects of the reactor pool temperature and the level on the module response that account for up to 12 modules rejecting heat to the reactor pool.

We went through a systematic process to assure that we had identified an appropriate scope of design basis events for Chapter 15. Particularly considering the systems and components that are unique to the NuScale design, such as the containment evacuation system for the ECCS valves.

MEMBER DIMITRIJEVIC: Excuse me. Could I ask you? Because I just want to make sure I understand what you just said before.

MS. McCLOSKEY: Um-hum.

MEMBER DIMITRIJEVIC: You said that you considered that changes in pool temperature are only for long term cooling for all units. Right?

Just for long term cooling?

MS. McCLOSKEY: That was an area of focus for long term cooling, to assure that we had bounded effects of multiple -- of potentially multiple modules rejecting heat to the pool.

MEMBER DIMITRIJEVIC: In long term cooling.

MS. McCLOSKEY: In long term.

MEMBER DIMITRIJEVIC: But you did not I suppose, analyze that one unit, you know, can be long -- or that even by the one unit just goes to transit.

MS. McCLOSKEY: We have considered -- the long term cooling analyses also consider potentially only one unit rejecting heat to a cold pool.

So that is -- that is covered.

MEMBER DIMITRIJEVIC: All right. Well, my question is, can we have 11 units injecting pool -- heating the pool, when one unit just going to DHRS or

ECCS operations?

So, you know, when -- I was wondering how did you model that pool temperature? And what case would you consider for it?

As I understood you only considered difference from the beginning only for long term cooling operations.

MS. McCLOSKEY: Well, and let me clarify here. We -- okay, in terms of multi-module effects, if one module were to experience a transient or an accident scenario, --

MEMBER DIMITRIJEVIC: Right.

MS. McCLOSKEY: That potentially affected the reactor pool temperatures for other modules, they would, I believe that they would still be under the control of the applicable technical specifications for those modules, to assure that to consider whether they would remain operating, or what actions operators can take.

I'm not sure if that gets at your question.

MEMBER DIMITRIJEVIC: Well, I mean, I was thinking about situations like where you have a common thing like loss of offsite power. Or, you know, you can have a loss of all AC power for all units.

And they may be entering the transients in the different phases. You know, because they would, you know, I'm not sure, but how we end operation that will go.

So that, you know, I mean, they will all three probably. But I'm not sure how they will go to different phases.

It's even, you know, will they go in the ECCS operation all in the same time? And how will actuate.

So, I was sort of wondering, can we have some normal occurrence like loss of offsite power where they will all be in the accident position of certain taking, you know, depending on the pool?

MS. McCLOSKEY: We -- we'll get to some of the example, some of the discussion in long term cooling later this afternoon.

We considered different scenarios for long term cooling, both maximum temperature cases as well as minimum temperature cases. And that considers a range of pool conditions.

And it's actually the minimum temperature case where you have the most heat removal outside of the containment vessel, where we have our most challenge in terms of level and margin to warm precipitation.

And so we've considered both ranges, both ends of that spectrum for a range of cooling that occur.

MEMBER DIMITRIJEVIC: Right. Well we will see as you present them then. And we will get a better idea of how does that fit and all

MR. BRISTOL: Yeah. And I'll just add that the pool temperature does a specified range in tech specs. So the analyses are initiated from, within that range.

MEMBER DIMITRIJEVIC: Um-hum.

MR. BRISTOL: So, we don't necessarily initiate a transient from something outside of that range. In the condition of the loss of offsite power to the, you know, to the entire plant, all the modules immediately go to DHR cooling conditions.

MEMBER DIMITRIJEVIC: Okay.

MR. BRISTOL: So they don't -- they wouldn't come in at stages kind of like you were talking about.

MEMBER DIMITRIJEVIC: Right. Well, I was trying to think, --

MR. BRISTOL: Sure.

MEMBER DIMITRIJEVIC: Because when we were have some discussion about operator actions through that. You know, because they cannot, they have enough time to go from one unit to units.

So, I will think of a type of accident I was thinking. But, even in these cases, they all come to DHRS when -- I mean, will temperatures stay in the -- okay. Well, we had to go to ECCS, right?

MEMBER BROWN: No.

MR. BRISTOL: Yes.

MEMBER DIMITRIJEVIC: You are having something goes wrong with the DHRS, I'm just trying to think about accidents which can cause that.

MR. BRISTOL: So the range of DHRS performance includes analysis of a range that's outside of the tech specs. The transients aren't necessarily initiated from outside of that pool temperature range.

But, there is consideration given to that. And I think we'll get into that maybe a little bit later, --

MEMBER DIMITRIJEVIC: All right. All right. Well, let's do that.

MR. BRISTOL: As we get into the details of the methodology.

MEMBER BROWN: During normal operation 12 modules, everything is working just fine. We're inside a vacuum inside the containments.

Is it expected that there's -- the pool temperature just stays constant? It doesn't heat up a little bit just due to some type of heat effects coming through the containment and out to the pool. And then stabilizers because the pool stabilizes after some point if you've got them all?

Is there a range of operation, 12 modules where if you start cold, start them all up, and the temperature would rise up just because of the reactor operations themselves? Even though they're within a vacuum?

MR. BRISTOL: Yes. Yeah, there is some heat loss, known heat loss from the module to the pool that's considered.

I think the spent fuel pool is one of the larger heat loads. I'm not extremely familiar with that analysis.

MEMBER BROWN: Yeah. Put the spent fuel pool aside. It's a -- I understand that cooling.

MR. BRISTOL: Sure. The pool has a cooling system that takes into consideration the normal heat loads and off-normal heat loads of DHR sometimes.

MEMBER BROWN: But there will be some heat up even though you've got the reactor vessel and everything within that vacuum area. There's going to be some, has to be, heat conducted through there.

MR. BRISTOL: Yeah. The pool definitely has a heat load.

MEMBER BROWN: Yeah. Okay. All right.

MR. BRISTOL: A normal heat load that's considered part of the systems then.

MEMBER BROWN: So, but the rest of your -- I'm trying to springboard off of Vesna's comment. And then you've also considered if you lost power, then you -- everybody goes to DHRS.

Then you'd have to be able to handle that in terms of the general -- and the pool has no cooling now. It's got the total, the total mass of water.

It has to be able to accomplish decay heat removal for all 12 modules. And not go outside of an acceptable band.

MR. BRISTOL: That's correct. And that's where we get pretty quickly into the short term transient response, pool temperature of the pool.

The pool is a big enough heat sink. It doesn't have a huge temperature transient where the long term cooling analysis is primarily where we have the consideration of the coping period, and what are all the heat loads.

And ensuring the --

MEMBER BROWN: Yeah. I'm not worried about an accident rate.

MR. BRISTOL: Sure.

MEMBER BROWN: I was interested in the 12 modules now are all in DHRS. Then you could have an accident. And you have to consider that as well?

MR. BRISTOL: No. We would consider the loss of power being the initiating event. And after DHR actuation then we're in long term cooling conditions.

So we wouldn't postulate a new initiating event during that event progression.

MEMBER BROWN: Even long term?

CHAIR MARCH-LEUBA: He's already dumping all of the heat from the core into the pool through DHRS. What worse can you make this?

MEMBER BROWN: I don't know.

CHAIR MARCH-LEUBA: You're dumping 100 percent --

MEMBER DIMITRIJEVIC: This is what I was trying to actually get to.

MEMBER BROWN: Yes. I thought that's

where you were trying to get to.

MEMBER DIMITRIJEVIC: I was trying to go to ECCS where something goes wrong when we -- actually we have additional accidents.

But then I, I realized that that's additional accident. And you would not look in that. But, I mean, it could be like some safety valve refuse to close or something, I mean, you know.

MEMBER BROWN: No idea. You're right.

MEMBER CORRADINI: I guess I'd ask a different question instead of asking all these things. What's the rate of rise if I had to assume the decay heat of all 12 modules into the pool?

MS. McCLOSKEY: It --

MEMBER CORRADINI: I calculate it to be less than a degree, substantially less than a degree an hour. On the order of a degree an hour.

That's what I would ask.

MS. McCLOSKEY: I don't know the specific rate of rise. But if you take realistic initial conditions for the pool level and temperature and decay heat loads for the 12 modules, then the -- assuming boil off of -- assuming the heat loads to the reactor pool, there's more then 30-days worth of level above the top of the Decay Heat Removal Systems.

MEMBER CORRADINI: Okay. Fine. All right. That's another way of doing it. Okay. Thank you.

MS. McCLOSKEY: Yeah.

MEMBER CORRADINI: I appreciate that.

MR. BRISTOL: It's very slow, yeah.

MS. McCLOSKEY: It's very slow.

MEMBER CORRADINI: I figure that.

MR. BRISTOL: Yeah.

MS. McCLOSKEY: Yeah.

CHAIR MARCH-LEUBA: Thirty days.

MEMBER CORRADINI: I mean, I didn't want to interject -- oh, I'm sorry. I didn't want to interject with their questions on the right.

But I have a different question. Are you guys done?

You were at initiating events, you were giving us a long list. In Table 15.01 and 15.02, you identify a thing called an IE.

And I didn't understand why you need an IE versus an AOO and a DBE, because in 15.02 your table for acceptance criteria is essentially, the IE's acceptance criteria is the same as the DBE.

So why did you make the distinction to begin with? I'm clueless.

MS. McCLOSKEY: Part of the distinction comes in the radiological dose acceptance criteria.

MEMBER CORRADINI: Not in the thermal hydraulic. So, --

MS. McCLOSKEY: Not in the thermal hydraulic dose acceptance criteria.

MEMBER CORRADINI: So, for the IE, the acceptance criteria is more like an AOO?

MS. McCLOSKEY: For -- in terms of the radiological dose, it's -- the radiologic -- the acceptance criteria are aligned with a small fraction of the acceptable dose for accidents.

MEMBER CORRADINI: But not the full dose?

MS. McCLOSKEY: But not the full dose.

MEMBER CORRADINI: And it's not the AOO which is essentially mainly just preserving the SAFTLs.

MS. McCLOSKEY: Right. So when we get --

MEMBER CORRADINI: Where would I find that? I guess I was looking, and I missed that.

MS. McCLOSKEY: You -- in the Table of the dose analysis results for the different acceptance criteria.

MEMBER CORRADINI: Oh.

MS. McCLOSKEY: For the steam generator

tube failure. And for small lines outside of containment, you'll see the acceptance criter -- yeah. Yes, that sample.

MEMBER BROWN: The acceptance criteria do change as the --

MS. McCLOSKEY: The acceptance criteria.

MEMBER CORRADINI: That's the one where it's 6.3 and not 5 or 25?

MS. McCLOSKEY: Correct.

MEMBER CORRADINI: Oh. Because the only two that you identified in 15.01 was the small line, and another one, which I can't remember.

MS. McCLOSKEY: The steam generator tube failure.

MEMBER CORRADINI: Thank you. Thank you very much. Okay. Thank you. Appreciate it.

MS. McCLOSKEY: So, and --

MEMBER CORRADINI: I'm done.

MS. McCLOSKEY: Okay. In terms of identifying the scope of initiating events for Chapter 15, we started with the PRA initiating events, because this was examined, then summarize the scope of events that could cause a reactor trip or transient.

But from there we examined the systems that were identified in the PRA as relevant to causing a reactor trip or transient in -- for additional detail in order to identify the specific impacts on the module, in order to categorize and classify the design basis events for Chapter 15.

CHAIR MARCH-LEUBA: Sorry to put you on the spot, but can you give me an example that you identify through PRA that was not on the SRP of Chapter 15?

MS. McCLOSKEY: An example might be, would be the containment flooding and loss of containment vacuum events. Because those are --

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

MS. McCLOSKEY: One potential cause of loss of containment vacuum is --

CHAIR MARCH-LEUBA: Well, that will not be on SRP 15. But it would be in the PRA.

MS. McCLOSKEY: Right. When we went through the design basis events that we identified and categorized them, our categories are consistent with those for operating light water reactors.

Except that the category of decrease in RCS flow events is not applicable due to the natural circulation design of the plant.

We also have NuScale specific phenomena

and -- or event progressions. PWR stability we've discussed already this morning.

And then return to power analysis are part of the Chapter 15 analysis. And we'll discuss that later this afternoon.

MEMBER CORRADINI: Where is a good time for me to ask about the evolution of NRELAP from 1.3 to 1.4? Since that appears in a lot of the open items as an addendum.

You decide where in the discussion today you want to explain that. And then the second one that I want to get explained is the scale distortion in NIST 1 and NIST 2 experiments.

MS. McCLOSKEY: Okay.

MEMBER CORRADINI: So, you don't have to answer it now. You just decide where you want to inject it. And then I can ask my questions. Okay?

MS. McCLOSKEY: Okay. I think -- all right. Well, probably the closed session is appropriate for any discussion of the details there.

MEMBER CORRADINI: Okay.

MS. McCLOSKEY: But, most of the discussion of scale distortions between the NIST 1 facility and the plant, we would defer detailed discussion of that to the topical report --

MEMBER CORRADINI: Which is yet to be --

MS. McCLOSKEY: Methodologies. Right. With the LOCA topical report methodology when we present that to the ACRS committee.

MEMBER CORRADINI: Okay. But that's nowhere in, nowhere in the short term future?

MS. McCLOSKEY: I believe it's this fall in the October time frame was the schedule for meetings.

MEMBER CORRADINI: At the earliest.

CHAIR MARCH-LEUBA: Did you say LOCA or non-LOCA?

MEMBER CORRADINI: Both are still out there as open.

CHAIR MARCH-LEUBA: I know. But which -- which were you talking about? You were talking about LOCA?

MEMBER CORRADINI: Both.

MS. McCLOSKEY: LOCA.

MEMBER CORRADINI: Okay. And then the RELAP 1.4 trans -- the transition from 1.3 to 1.4, because staff has yet to finish its review of that.

Or I'm not sure if you've yet to even issue the differences.

MS. McCLOSKEY: Yes. The differences are

issued. And the staff has also audited that information.

So, I think we can say a little bit --

MEMBER CORRADINI: In closed session.

MS. McCLOSKEY: In the closed session about the details of the code changes.

MEMBER CORRADINI: Okay. All right. That's fine. That's good. We'll just wait until closed session. Thank you very much.

MS. McCLOSKEY: And the effects. The design basis events, this has already been mentioned here. We are classified as anticipated operational occurrences, infrequent events or accidents.

Events that could potentially occur one or more times during the lifetime of the plant, are classified as anti -- as AOOs.

Events that are not expected to occur are classified as infrequent events or postulated accidents. Or in some cases they were conservatively classified as AOOs.

And one example of that is the inadvertent opening of an ECCS valve. We don't expect that event to occur during the lifetime of the plant.

But it was conservatively classified as an AOO. And we demonstrate that those acceptance criteria are met. Because that bounds a transition of other events to ECCS cooling if a loss of power were to be assumed.

And in some cases we also simplified the event classification by applying a deterministic criteria where the event was similar to other PWRs. Particularly where event consequences are small and calculating a NuScale specific event frequency is not warranted.

So again, failures such as in the containment evacuation system are classified as an AOO event.

MEMBER CORRADINI: So, there was an open item, or an RAI, it maybe an open item. I don't remember if I've got it down right, in terms of how you classified return to power.

So, how is that being classified?

MS. McCLOSKEY: That is a -- it's not an initiating event. It's an event progression.

MEMBER CORRADINI: Right.

MS. McCLOSKEY: And so it's not -- the event classification applies to the initiating events. But we demonstrate that the AOO acceptance criteria are met.

MEMBER CORRADINI: Ah. Okay. So, the way

you look at it is if it were an AOO.

MS. McCLOSKEY: Well, that's what defines the acceptance criteria. Yes.

MEMBER CORRADINI: Okay. Fine. Thank you.

CHAIR MARCH-LEUBA: Because the initiating event is loss of offsite power for more than 24 hours. And no non-safety grade power back up coming up.

But yet if you -- it's perfectly acceptable to make it an AOO.

MS. McCLOSKEY: Right.

MEMBER CORRADINI: I'm going to ask the same question of the staff. So I'm just trying to ask it here and get there and see if there's consistency.

MS. McCLOSKEY: Um-hum.

MEMBER CORRADINI: Because it was left out. Okay, an AOO. Thank you very much.

MS. McCLOSKEY: So next slide. This slide and the next several slides summarize the design basis events and their respective categories.

So we have the event, the event classification, the evaluation model used to do the system thermal hydraulic analysis if applicable. And the end reliant code used there.

Whether the event is analyzed for

subchannel analysis with the VIPRE-01 code. And then whether the event is part of the Chapter 15 radiological dose analysis.

The NuScale specific events are highlighted. And so here loss of containment vacuum or containment flooding is a unique event.

The loss of containment vacuum could be postulated due to a malfunction in the containment evacuation system. Containment flooding is postulated due to a break in piping that carries reactor component cooling water to the control rod drive mechanisms on top of the reactor head.

So although these events, these postulated events don't directly interface with the primary or the secondary side of the module, they would -- they could result in a slow increase in heat transfer from the reactor vessel compared to the evacuated -- the vacuum conditions normally in containment.

And so it's included here as an increase in the removal event.

CHAIR MARCH-LEUBA: And anywhere you say in RELAP 5 you mean you plan to do it with the newest version of the model, correct?

We have results for 1.3 and you plan to do it for 1.4?

MS. McCLOSKEY: Yes. We're in the progress of -- we're in the process of revising those analyses to 1.4.

CHAIR MARCH-LEUBA: And remind me again, when you use VIPRE in a transient, you use VIPRE for every entire step? I don't remember the -- I remember the methodology for steady state.

But for transient, how do you apply VIPRE during the transient?

MS. McCLOSKEY: The boundary conditions from the NRELAP 5 results are provided for power, fl -- total RCS flow, pressure, and core inlet pressure.

CHAIR MARCH-LEUBA: At every time slice?

Ms. McCLOSKEY: Well, I think --

MR. BRISTOL: No. So there's an edit frequency that's generated from RELAP and transmitted to, you know, around a second or half a second.

CHAIR MARCH-LEUBA: Okay. But then even time slice.

MR. BRISTOL: Yeah. That's right, yeah. Yeah.

CHAIR MARCH-LEUBA: And how many roles do you have play with VIPRE? How many roles of the, subchannels does VIPRE simulate? You don't remember?

MR. BRISTOL: I don't.

CHAIR MARCH-LEUBA: Okay. It's been a year since we reviewed it. But, just I was asking.

MR. BRISTOL: That's fair. We can follow up with that if that's of interest.

CHAIR MARCH-LEUBA: It would be nice to know. I know we know, but I don't remember.

MEMBER CORRADINI: But you have -- you do track the hot channel versus the average channel. That's what I would assume.

MR. BRISTOL: Um-hum.

MEMBER CORRADINI: At the time when we did this, you had a nodalization approach to as where you determined the hot channel. And then you essentially had less and less subchannel nodalization --

MR. BRISTOL: Um-hum.

MEMBER CORRADINI: as you went. It went away from the grouping of hot channels. So I assume that's what was done here.

MR. BRISTOL: Fifty-two rods. Is the answer.

MEMBER CORRADINI: Okay. Thank you.

MS. McCLOSKEY: All right.

MEMBER BROWN: Before you leave that, the last item, go back.

MS. McCLOSKEY: Um-hum.

MEMBER BROWN: How much of an effect overall is there if you just lose the vacuum and it's just air? Can -- does that significantly heat up the pool?

MS. McCLOSKEY: No. It's a --

MEMBER BROWN: So it's -- you could operate that way? Or do you require a shut down if that occurs?

MS. McCLOSKEY: We would be outside the --

MEMBER BROWN: But just air. It's air, not water. It's not -- it hasn't been flooded.

MS. McCLOSKEY: We would be outside the limits established for monitoring containment leakage.

MEMBER BROWN: Okay. So it would be a containment leakage issue then.

MS. McCLOSKEY: Um-hum.

MEMBER CORRADINI: They'd have to shut down.

MEMBER BROWN: That's what I guess I would translate. Okay. Thank you.

MS. McCLOSKEY: In Section 15.2 are the, you know, it results in a decrease in heat removal from the secondary side.

And here the NuScale specific event is inadvertent operation of the Decay Heat Removal

System. Because the Decay Heat Removal System is sized for decay heat removal, so inadvertent operation of -- actually causes a decrease in heat removal.

There are several different variations of this event that we analyze. That maybe a single valve, actuation valve opening.

It maybe an inadvertent signal that actuates the DHRS train valves and closes the secondary isolation valves on one train or on both trains.

And we have some example results on the transient for the single valve opening later this afternoon.

In terms of the reactivity and power distribution anomalies, these events are similar to the scope for light water reactors. The flow related events are not applicable to the design.

And for the 15.4.6, the inadvertent decrease in warm concentration, it's postulated due to a CVCS, chemical volume and control system malfunction that causes a boron dilution transient to occur. That's analyzed as part of the non-LOCA evaluation model.

And then in -- we analyzed the control rod ejection accidents. There's a separate topical report for the wad ejection analysis where the power response is calculated using the SIMULATE-3K code.

And the system response is analyzed with NRELAP-5. And then VIPRE is used to assess crit -- margin to critical heat flux ratio.

MEMBER CORRADINI: And I think we're going to see that topical in September.

MS. McCLOSKEY: I don't know.

MEMBER CORRADINI: Well, that's all right.

MS. McCLOSKEY: I mean, later -- yeah, we'll see the topical later in the --

MEMBER CORRADINI: Later in the fall time frame.

MS. McCLOSKEY: Later in the year. Yep.

MEMBER CORRADINI: All right. Thank you.

MEMBER BLEY: I know you answered Mike on this earlier, but I kind of didn't follow it then.

Most of these are AOOs. But, you have an IE in there. And I forget why you said that was.

MS. McCLOSKEY: And I forgot about that one when we were talking about IEs.

MEMBER BLEY: That's okay.

MS. McCLOSKEY: That's the inadvertent loading of a fuel assembly in it.

MEMBER BLEY: So that just means that's

when you picked up the most PRA? Or why is it falling under?

MS. McCLOSKEY: That is one where the radiological acceptance criteria are a fraction of those acceptable for the accident dose.

MEMBER BLEY: Okay.

MS. McCLOSKEY: It was that. It wasn't the steam generator tube failure. I was mistaken earlier.

MEMBER CORRADINI: That's okay. I just wanted to know what it was. And you answered it. That's fine.

MEMBER DIMITRIJEVIC: Well, you have, I mean usually we have a thorough, infrequently it's an accident. And I thought that your postulate an accident is something where we normally have an accident. And the IE will be what you call infrequent event.

But you actually classify everything with AOO, right?

MEMBER CORRADINI: No.

MS. McCLOSKEY: No.

MEMBER DIMITRIJEVIC: No, no. There is the one postulated accident where there is the one IE.

So, I mean, is there some reason why you decided to just classify everything as AOO and then have a couple of exceptions which don't correspond to this other usual division?

MEMBER CORRADINI: We haven't gotten to 15.6 yet. There's a lot of postulated events in 15.6.

MS. McCLOSKEY: In most cases the initiating events are similar to those for operating PWRs.

MEMBER DIMITRIJEVIC: Right.

MS. McCLOSKEY: And so we classified them consistently.

MEMBER DIMITRIJEVIC: But no. For the loss of offsite power is infrequent event. Which you call AOO here. I mean, and well also --

MS. McCLOSKEY: Loss of offsite power is typically analyzed as an AOO.

MEMBER DIMITRIJEVIC: Oh, really?

MS. McCLOSKEY: Yes.

MEMBER DIMITRIJEVIC: That's unusual. I mean, not from my time. But anyway -- well, maybe my time is, my time isn't right.

Well, the thing is so what is then like infrequent event, like steam generator tube rupture. What -- how would you classify that?

MS. McCLOSKEY: I was mistaken earlier as we -- a steam generator tube rupture is actually classified as a postulated accident. And that's on the 15.6 slide.

MEMBER DIMITRIJEVIC: And most LOCAs you classify right, as a postulated accident, right?

MS. McCLOSKEY: Yes. Yes.

MEMBER CORRADINI: They have a long list. We're only half way through their list.

MEMBER DIMITRIJEVIC: Okay. I'm going to check with the -- with the one with the specifications in the PLA specimen. I'm going to see.

I mean, for me this is like almost everything you've done is AOO. Which is like really spec.

MS. McCLOSKEY: And this is in the design basis event space. In terms of events that are unique to NuScale that we conservatively classified as an AOO, in some cases because we have sufficient margins, it wasn't valuable to do a specific analysis of the event frequency in order to justify classifying it as something other than an AOO.

And so we treated it as an AOO.

MEMBER DIMITRIJEVIC: All right. Because obviously some of those are not going to occur during the, you know, hopefully during the plant life, so.

MS. McCLOSKEY: Right.

MEMBER DIMITRIJEVIC: All right. Okay. I mean, and you think that this is conservative from your perspective. So, you know, what you need to so express this, that's all right.

MS. McCLOSKEY: We go to 15.5 in terms of events that increase reactor coolant system inventory, for the NuScale design as a chemical volume control system is the only system with capability to increase RCS inventory during normal operation.

Then in --

MEMBER SKILLMAN: Well, what about control rod drive cooling?

MS. McCLOSKEY: The control rod drive cooling doesn't interface with the primary system. And so it's -- that's a failure in the lines carrying cooling to the control rod drives, is treated in the containment flooding analysis.

MEMBER SKILLMAN: Okay.

MS. McCLOSKEY: In 15.1.

MEMBER SKILLMAN: Okay. Thank you.

MS. McCLOSKEY: In the -- in 15.6 then, we have the events that decrease reactor coolant system inventory.

The inadvertent operation of an emergency core cooling system valve or also -- we also call that inadvertent opening of a reactor safety valve, is addressed here.

And that's analyzed with the valve opening event methodology that's an extension of the LOCA analysis methodology. But we analyze it to demonstrate that AOO acceptance criteria are met.

CHAIR MARCH-LEUBA: Is that going to be on the same topical report as LOCA?

MS. McCLOSKEY: Yes. The methodology --

CHAIR MARCH-LEUBA: So there's like an appendix?

MS. McCLOSKEY: Is now described in appendix B of the LOCA topical report.

CHAIR MARCH-LEUBA: And has it always been an AOO? Or is this a recent modification?

I'm asking in conjunction with an IAB, you know, an inadvertent actuation block valve.

MS. McCLOSKEY: Um-hum.

CHAIR MARCH-LEUBA: If you consider this an AOO and you survive it, why do we need an IAB?

MS. McCLOSKEY: The -- since we -- since the NuScale design doesn't have a safety-related power supply, then in design basis event space, any event where you assume loss of power --

CHAIR MARCH-LEUBA: Oh.

MS. McCLOSKEY: To occur, could be postulated to transition to ECCS cooling.

CHAIR MARCH-LEUBA: But this one -- so this event, 15.6.6 is loss of AC power, which trips -- well, I mean you trip the solenoid on the ECCS valve. But IAB still holds it closed?

MS. McCLOSKEY: No. 15.6.6 postulates in an inadvertent opening of one ECCS valve.

CHAIR MARCH-LEUBA: And release of pressure.

MS. McCLOSKEY: While, the RCS is operating at normal operating pressure and 100 -- and 2 percent power.

CHAIR MARCH-LEUBA: And using AOO acceptance criteria.

MS. McCLOSKEY: Yes.

CHAIR MARCH-LEUBA: So then the question is, why bother with the IAB?

MS. McCLOSKEY: So --

MEMBER CORRADINI: I think the answer yesterday was they don't want to have too many invalid actuations. They want to have valid actuations.

So that was their protection from an investment standpoint.

CHAIR MARCH-LEUBA: I thought the answer is they were not sure. They were not -- that they could survive this.

MEMBER CORRADINI: Oh. We can -- I'm not sure, I can't remember who from the audience came up and helped us. But that's what I remembered.

MEMBER BLEY: It was Storm. And -- oh, you'll cover it. Go ahead.

CHAIR MARCH-LEUBA: Say your name.

MR. RAD: So, some of this information is in --

MEMBER BLEY: Your name?

MR. RAD: I'm sorry, this is Zachary Rad, Director of Reg Affairs, NuScale Power. Some of that information in detail would be in the proprietary session.

But, the objective here, so analyzing the inadvertent opening as an initiating event is basically just following the Chapter 15 protocol.

What we don't want is to have an inadvertent opening of one, and another one opens because of the basic design. I mean, so that's why the IAB is in there. Multiple openings would be negative.

MEMBER CORRADINI: Thank you.

MS. McCLOSKEY: The other event I'll highlight here is the small line break outside of containment. That event is traditionally analyzed in PWRs for radiological dose analysis.

And we also analyze it for dose analyzes here. The small line break event in the NuScale design is mitigated by closing the containment isolation valves typically on a low pressurizer level signal.

And then you have sufficient inventory remaining in the reactor coolant system to support cooling through the Decay Heat Removal System. And so that's why you'll see that's part of the non-LOCA event analysis.

MEMBER CORRADINI: A long list.

MS. McCLOSKEY: In terms of the thermal hydraulic and fuel acceptance criteria, this table summarizes acceptance criteria for minimum critical heat flux ratio, the primary and secondary site pressures, containment pressure, and event progression.

This is generally consistent with the standard review plan guidance, except that the NuScale analysis methodologies are developed to demonstrate that fuel cladding integrity is maintained, and MCHFR remains above the limit for AOOs, infrequent events, or postulated accident conditions.

The secondary side system pressure, design pressure is equal to the primary system design pressure at 2100 psi. And so you'll see in our analysis results we have, we retain significant margin to the secondary pressure limits.

The containment design pressure is 1050 psi. And we have those results in FSAR 621 to discuss later this afternoon.

So, given all of these different acceptance criteria and different event types, the next two slides summarize the NuScale topical and technical reports, describing the analysis methodologies, to just lay out what these connections are.

For a typical non-LOCA event, we built the plant model in NRELAP-5. And that's used to calculate the system from a hydraulic response to demonstrate that the primary and secondary pressure criteria are met. And that a safe stabilized condition is achieved.

And that's in the non-LOCA topical report. The boundary conditions from that system from a hydraulic analysis are then provided to for the subchannel analysis with VIPRE to demonstrate the field cladding integrity are met.

And boundary conditions are from the -- are provided for the radiological analysis. And the accident source term analysis topical report describes the methodology to establish the source terms for the radiological consequences.

MEMBER BLEY: Let me -- and Mike, I know we have the topical on source terms later this year. I think I know that.

MEMBER CORRADINI: We -- it has been delivered to staff and they are starting to review it.

MEMBER BLEY: Okay. You don't know when we're going to --

MEMBER CORRADINI: I'm not sure that we have scheduled it.

MEMBER BLEY: Okay. What about the other two topicals there? Have we --

MEMBER CORRADINI: We've already looked at the subchannel topical report on how they've analyzed it. We have not -- we -- it's somewhere in the fall that we'll see the non-LOCA and the LOCA.

MEMBER BLEY: Okay.

MS. McCLOSKEY: In terms of LOCA and valve opening events, we use NRELAP-5 to protect -- to predict the system and the hot channel response.

And in the NuScale methodologies the acceptance criteria for these events are to demonstrate margin too minimum critical heat flux ratio. And that water level is maintained above the top of the fuel.

The long term cooling analysis with ECCS is analyzed with RELAP to demonstrate that water level remains above the top of the core. And the core inlet temperature remains sufficiently high that boron precipitation is precluded.

And then the containment response is an extension of the LOCA EM and also the non-LOCA EM for the secondary side breaks to evaluate peak pressure and temperature levels.

MEMBER CORRADINI: So, let me get a clarification. Because I know the topicals are being submitted to us for review -- I'm sorry, submitted to the staff, excuse me, for review.

The technical reports are essentially addendums to Chapter 15 analysis?

MS. McCLOSKEY: Chapter 15 and Chapter 6, yes.

MEMBER CORRADINI: Okay. Which are not being asked for the staff to look at them, only from the standpoint of audit. And they want to do confirmatory calculations.

But they are not being submitted to the staff for separate review.

MS. McCLOSKEY: They are part of the DCA.

MEMBER CORRADINI: Okay. That's what I thought. I just wanted to make sure.

MS. McCLOSKEY: So this is part of the Chapter 6 and Chapter 15 analysis.

MS. KARAS: Yes, this is Becky Karas from Reactor Systems. We do review the technical reports in concert with the DCA review --

MEMBER CORRADINI: But we would find your evaluations buried inside the SEs?

MS. KARAS: That's correct.

MEMBER CORRADINI: Okay. Fine. Thank you.

MEMBER DIMITRIJEVIC: For the Chapters.

MEMBER CORRADINI: For the Chapters, yes.

MR. SCHMIDT: This is Jeff Schmidt. It's incorporated by reference to the DCA.

MEMBER CORRADINI: I'm telling this to the -- I'm saying it out loud so the members, if they get interested, because we felt that we didn't have enough paper to read so far.

So, just in case. The 700 pages was a quick read.

MEMBER BLEY: Half of it's pictures.

MEMBER CORRADINI: That made it quicker.

MS. McCLOSKEY: The module protection system actuations in the NuScale design were developed and designed with a focus on this scope of design basis events and the event response and the actuations that we need to support the passive full response to design basis events without crediting operator actions.

And so we consider the design basis events, the functions can be broadly classified into four areas. The reactivity control either through reactor trip or through isolation of the source of dilute water that could be causing an inadvertent dilution of event.

RCS and secondary site inventory control. The containment isolation assures that sufficient inventory is maintained to support ECCS cooling in the event of a postulated pipe break or valve opening.

It also mitigates the loss of inventory outside of the module, limiting dose consequences. And the secondary isolation limits dose consequences for events such as steam generator tube failure.

And also assures that appropriate inventory is maintained in at least one train of the Decay Heath Removal System for DHRS cooling.

If normal secondary side cooling unavailable, we've talked about the DHRS. And if necessary, the emergency core cooling system actuations.

And then finally primary side subcooling and stability are protected by reactor trip.

CHAIR MARCH-LEUBA: So would this be a good time to stop for lunch?

MS. McCLOSKEY: I think one more slide too just --

MEMBER CORRADINI: A good transition point?

MS. McCLOSKEY: It would be a good transition point.

MEMBER CORRADINI: Okay. Thank you.

MS. McCLOSKEY: Because when we take those overall functions of the module protection system and look at the design basis event mitigation for the different types of events that we've gone through, that's what's summarized on this slide.

MEMBER CORRADINI: Okay.

MS. McCLOSKEY: That the increase in heat removal, transients are most often cause by a postulated secondary side malfunction. So secondary isolation is important there.

Decrease in heat removal transients generate Decay Heat Removal System actuations. Reactivity in power transients rely on reactor trip, and demineralized water isolation.

Increase in RCS inventory events are mitigated with isolation from the increase in inventory source. Decrease in RCS inventory events rely on containment isolation to control the inventory available for ECCS cooling, and to mitigate dose consequences and reactor trip instability.

MEMBER DIMITRIJEVIC: Shouldn't you have like a measure control function also complication. Like the, you know --

MS. McCLOSKEY: The pressure in the reactor coolant system is limited by the reactor safety valves.

MEMBER DIMITRIJEVIC: Right.

MS. McCLOSKEY: And so those are passive, passively actuated valves. They aren't controlled. So it's not -- that's not a function of the module protection system.

The module protection system provides

actuations --

MEMBER DIMITRIJEVIC: But module systems have like inter-module protection function.

MEMBER CORRADINI: Is your green light on? I'm not sure anyone can hear you.

MEMBER DIMITRIJEVIC: Oh, well. I will just consider it as a multiple presentation fraction where I was not pressure control part of the function which we're looking at.

MS. McCLOSKEY: This was -- sorry, this was focused on the module protection system.

MEMBER DIMITRIJEVIC: System, I see. All right.

MEMBER CORRADINI: So this is a good time for us?

MS. McCLOSKEY: Yes.

MEMBER CORRADINI: Okay. So, we're going to try to catch up --

MR. BRISTOL: I have a quick follow up on the --

MEMBER CORRADINI: Okay. Go ahead Ben.

MR. BRISTOL: Common cause failure. For the level sensors, that's actually addressed in DCA Section 7.1.5.1.2. And in Table 7.1-13 kind of has the functions and how diversity and common cause failure is addressed for the very sensors.

CHAIR MARCH-LEUBA: Can you read us the section again?

MR. BRISTOL: Yes. So the DCA Section Number 7.1.5.1.2.

CHAIR MARCH-LEUBA: Level five, yes?

MR. BRISTOL: Yes. And the DCA Table is 7.1-13.

CHAIR MARCH-LEUBA: Thank you. You know, whenever you say that, we're going to look at it. And then come back and say it doesn't say anything.

(Laughter)

MR. BRISTOL: Yeah.

CHAIR MARCH-LEUBA: I'm only kidding you.

MEMBER CORRADINI: That's not a given. He's just being --

MR. BRISTOL: I'll be prepared for that follow up. Thank you.

MEMBER CORRADINI: Okay. So why don't we take a break until 1:15. So we'll have a very efficient lunch.

(Whereupon, the above-entitled matter went off the record at 12:33 p.m. and resumed at 1:17 p.m.)

CHAIR MARCH-LEUBA: For your reference we are starting on Slide 24, okay?

MS. McCLOSKEY: All right, so as we move into a summary of the analytical assumptions that are applied in the Chapter 15 analyses, this slide shows a map of the analytical pressure and temperature, operation limits, and module perfection system limits.

And Dr. Corradini, I think this morning you had a question about the normal T hot operation versus --

MEMBER CORRADINI: It was somebody, I don't remember. It doesn't matter.

MS. McCLOSKEY: Sorry, so there was a question this morning and the normal T hot is shown there in the green dot in the middle with the operating range that's analyzed.

And then the T cold and T hot is about 590 degree compared to the hot leg temperature, module protection system analytical limit of 610 degree to protect the margin to subcooler.

CHAIR MARCH-LEUBA: My understanding is if we make a mistake completing the flow, T hot will be maintained at that point and T average will oscillate.

MS. McCLOSKEY: The plant control is based on T ave and maintaining constant T ave.

CHAIR MARCH-LEUBA: What we were told is

if -- which we will -- you're assuming a flow, a flow rate. And we will have the flow rate the plant wants to have and if it's only 0.1 percent off there's no difference.

But if it's a significant difference, we will keep T hot where it is now or we will move T cool.

MEMBER CORRADINI: I think they've got to check on that.

MR. BRISTOL: So, for the consideration of operating margin, yes, there is some thought that goes into where the analytical limit, at which point a trip would come in, how much margin needs to account for censor uncertainty, how much additional margin needs to count for normal transience.

And so, yes, most likely the 59595 condition would be maintained as comfortable operating margin and then T ave would flow from there.

MS. McCLOSKEY: And we have the high and low pressure operating limits shown on there, the plus or minus 70 PSI from normal operating condition at 1850.

And then the margin to high pressure and low low-pressure analytical limits are also shown --

(Simultaneous Speaking.)

CHAIR MARCH-LEUBA: And the red lines are automatic scrams, correct? If a person goes below 1600, the protection system will scram?

MS. McCLOSKEY: Yes. I'll go on. In terms of the analytical assumptions for Chapter 15 analysis, we'll talk about operator actions, single failures, loss of power and then the scope of event progression.

In the NuScale design for the Chapter 15 analyses no operator actions are required to achieve the safety functions for 72 hours after an initiating event occurs.

We do consider operator errors in identifying the scope of initiating events, such as inadvertent signals that could occur.

But any operator actions that are allowed by procedure will make the consequences of an event less severe, and therefore are bounded by the Chapter 15 analyses.

The scope of multiple operator errors or errors that result in a common-mode failure are beyond design basis.

In the Chapter 15 analyses, we assume the limiting single failure of a safety-related component, and so we went through a systematic evaluation of our systems and the safety-related components to identify failures that could affect the transient progression.

And due to both the simplified design in terms of reducing the number of safety-related components and redundancy of the safety-related components, that reduces the overall scope of failures that need to be evaluated in the Chapter 15 events to determine the worst single failure for a particular analyses.

And this slide summarizes the results of that evaluation for the safety-related systems, relevant single failures that occur in those systems, and then some discussion.

So, in terms of redundancy in components, an example of that are the containment isolation valves on the CVCS piping.

Since there are two isolation valves in series, single failure of one isolation valve doesn't change the event progression and so it's not specifically part of a calculation analyzed because it doesn't affect the transient results.

CHAIR MARCH-LEUBA: One thing that we've discussed with respect to that in other meetings is what I call analytical redundancy. So, if this particular scram fails, indirectly we will catch it with this other one in scram 2.

And there's a lot of flows in NuScale that I don't think we're taking proper credit for. So, if the pressure fails and doesn't scram you, you will get a scram at high temperature.

MS. McCLOSKEY: The module protection system functions are modeled in the Chapter 15 analyses.

And I'm not sure if you're getting at considering common-cause failures of instrumentation, and I think as Ben mentioned earlier this morning, there's a summary of those results in Chapter 7.

CHAIR MARCH-LEUBA: Yes, and typically we consider failure of one control rod to insert in all Chapter analyses.

Is that in addition to this?

MS. McCLOSKEY: That's accounted for in the scram worth.

CHAIR MARCH-LEUBA: The one rod out is not a single failure, that's an assumption?

MS. McCLOSKEY: No.

CHAIR MARCH-LEUBA: On top of that you have a single failure.

MS. McCLOSKEY: Yes, and on top of that we consider power availability.

CHAIR MARCH-LEUBA: Power is not safety-grade. But you have to consider DC power to fail and is that because it was operating before so it should stay there?

It would be a failure to make it fail?

MS. McCLOSKEY: We consider the availability of DC power because it's not a safety-related Class 1E power supply system.

CHAIR MARCH-LEUBA: But in all transients you have the DC power on.

MS. McCLOSKEY: For many events that's more limiting for the transient response because the loss of DC power actuates the safety systems.

CHAIR MARCH-LEUBA: I didn't see that.

MS. McCLOSKEY: In terms of the single failures that are considered in the analyses, in the module protection system single failure of an instrument channel is considered.

And that's relevant for asymmetric reactivity events where a censor closer to the power asymmetry could be assumed failed and that would delay the response of the remaining instruments that are operating.

In the containment isolation valves we consider failure to close of a main steam isolation valve or a feedwater isolation valve. In those cases, we take credit for the non-safety-related back up valves in both of those lines.

Similarly, there are check valves in the feedwater that prevent backflow in the case of a postulated large feedwater line breaking inside of containment.

And in the case of failure to close those check valves, the non-safety-related back up check valve in the lines is credited.

CHAIR MARCH-LEUBA: So, if we state that in plain English, you're taking credit for non-safety-grade back up equipment?

MS. McCLOSKEY: And that's consistent with NUREG 0138 and the guidance in Reg Guide 1.206 that non-safety-grade equipment can be credited as back up in the case of a single failure to the safety-related component.

CHAIR MARCH-LEUBA: As long as the single failure is not the initiating event. So, let's talk about the two check valves. You have an initiating event, Check Valve 1 fails, you can take credit for 2.

Now, if Check Valve 1 fails as your initiating event, well, you need to have another one.

MS. McCLOSKEY: Well, if Check Valve 1

fails as the initiating event, that would result in a decrease in the feedwater flow, which is analyzed as an initiating event and doesn't demand response of the check valves.

The check valves are used in the case of large feedwater line breaks inside containment where you can get reverse flow from the intact steam generator over to the break if the check valves were not there.

CHAIR MARCH-LEUBA: I guess I will ask tomorrow from the Staff about this single failure. The first time I looked at it, it felt to me that we're taking credit for non-safety-grade stuff.

MS. McCLOSKEY: And that's consistent with the applicable regulatory guidance and the augmented quality that's applied to these valves. The feedwater reg valve and the backup means isolation valves are seismic Class 1 valves.

They're part of the in-service testing program and they're in tech specs with limiting conditions for operability and surveillance requirements.

CHAIR MARCH-LEUBA: So, other than not having an N stamp in the outside casing, it's treated as safety-grade?

MS. McCLOSKEY: I think effectively.

CHAIR MARCH-LEUBA: The most important thing is whether they're in tech specs in Appendix B.

So, whenever one of those equipments goes out to service, if it's safety-grade, you have an LCO and you have to start downgrading power and eventually shutting down.

If it's not in tech specs, you won't.

MS. McCLOSKEY: The back up main steam isolation valves and the feedwater valve are in tech specs.

CHAIR MARCH-LEUBA: With an LCO?

MS. McCLOSKEY: Mm-hmm. The backup check valve is also seismic Class 1 and part of the in-service testing program.

CHAIR MARCH-LEUBA: Because check valves in particular have a history of failing a lot and that's why we have --

MS. McCLOSKEY: The Chapter 15 analyses, they're only relevant for the postulated accident large feedwater line break.

CHAIR MARCH-LEUBA: Okay, we will ask the Staff about their opinion on this.

MS. McCLOSKEY: In the emergency core cooling system, we consider failure of one of the ECCS valves to open either event valve or a recirc valve.

We also consider a module protection system failure to actuate a division of ECCS, in which case one vent valve and one recirc valve would remain closed.

The IAB failing to close upon demand is not treated as one of the single failures in the accident analyses. That would occur in the case of a postulated loss of DC power that results in a demand to the ECCS valves.

CHAIR MARCH-LEUBA: So, you just told us this morning that you were going to analyze that.

MS. McCLOSKEY: We analyze the inadvertent opening of an ECCS valve, yes.

CHAIR MARCH-LEUBA: By itself?

MS. McCLOSKEY: By itself.

CHAIR MARCH-LEUBA: So, you don't want to analyze it in conjunction with another initiating event?

MS. McCLOSKEY: Correct.

CHAIR MARCH-LEUBA: Like loss of DC power?

MS. McCLOSKEY: Loss of DC power is part of the loss of power assumptions that are treated. So, loss of DC power is not a specific initiating event.

But if we go to the next slide on loss of power, we consider loss of AC power at the time of event initiation or at the time of reactor trip. And in the case of loss of AC power, we consider whether the DC power system is available or unavailable.

So, postulating a loss of AC power at the time of reactor trip with loss of DC power at the time of reactor trip, if the IAB were not there, then that could postulate a demand on the ECCS system in conditions far outside normal operation.

And this is all in the context of the deterministic design basis assumptions for Chapter 15.

CHAIR MARCH-LEUBA: But if I understand the system, AC power feeds the batteries and then the batteries feed the instrumentation and so on and on and everything.

The AC doesn't bypass. So, if the battery cable out fails, that would be a failure of DC power even though you have AC power.

MS. McCLOSKEY: In which case, that would actuate the reactor trip DHRS containment isolation.

CHAIR MARCH-LEUBA: I'm not saying it's a bad thing, it's an event that should have been analyzed.

MEMBER DIMITRIJEVIC: I thought I was so

loud.

MS. McCLOSKEY: And we consider that range as part of the loss of AC power initiating event, but since the safety systems are actuated, it doesn't progress to more severe condition for the core.

CHAIR MARCH-LEUBA: Let me see if I understand. One of the initiating events, 15.6.6, the RRB opens by itself and truly opens.

And there are other scenarios that you can imagine where the IAB would prevent that from opening. That's what we have analyzed?

MS. McCLOSKEY: The event in 15.6.6 postulates some sort of a mechanical failure in the valve. From the perspective of Chapter 15, worked on get into the details of exactly what that postulated failure is.

But in other scenarios we assume that the valves operate as designed except in the case of the single failures that we were talking about. Does that answer -- I'm not sure if that gets to your question.

CHAIR MARCH-LEUBA: I was trying to figure out the logic of assuming it fails on 15.6.6 and assuming it doesn't fail on everything else.

MS. McCLOSKEY: We treat the initiating events separately.

MR. RAD: I can provide some insights within the context of the regulatory framework. This is Zachary Rad, Director of Reg Affairs, NuScale Power.

Relative to the single failure criteria, a component treated as passive is assumed to fail as an initiating event but is not assumed as an additional failure.

So, following that regulatory framework, we've analyzed the inadvertent opening of the ECCS valve.

Continuing on, because it's not considered as an additional failure -- so under a single failure criteria in active components you have to consider the worst single failure of an active component in addition to the initiating event.

Because this device is treated as passive in our safety analysis, it is not considered to be an additional failure relative to the initiating event.

So, if you have a LOCA you don't assume an additional passive failure, an additional LOCA for instance. In this case, it's a valve that we've treated as passive relative to the single failure criteria. We don't consider that as an additional failure either.

CHAIR MARCH-LEUBA: So, when do you consider additional failures of check valves?

MR. RAD: We consider additional failures on active components as we've treated them relative to single failure criteria.

CHAIR MARCH-LEUBA: So, check valves is an active component?

MR. RAD: Check valve is an active component that is sometimes treated as a passive component relative to the single failure criteria.

MEMBER CORRADINI: Does that make sense?

CHAIR MARCH-LEUBA: No.

MR. RAD: So, active in the fact that it actually does move so technically it is active.

However, its treatment relative to the single failure criteria is as if it were passive as it relates to the information I just covered before, how it's treated relative to subsequent failures.

CHAIR MARCH-LEUBA: IAB is also active?

MR. RAD: That's correct, it moves.

CHAIR MARCH-LEUBA: It moves.

MEMBER CORRADINI: But it would be part of the initial failure under -- I'm looking for all of their -- it's 15.6.6, inadvertent ECCS opening so they assume a failure but then they're claiming that any subsequent IAB failures cannot be assumed because they're not an active component.

That's what I thought you just said.

MR. RAD: That's correct. So, to put it in a different context, how do we assume that was an active component and the initiating event was the inadvertent opening of an ECCS valve for whatever reason?

And we have to assume the worst-case single active failure, that evaluation would include the failure of the additional failure of that IAB and the potential opening, additional opening, of a second ECCS valve.

Does that make sense? Especially given that our assumptions are loss of AC power and because we don't have safety-related DC power, and correct me if I'm wrong, in our analysis we assume that's lost as well.

CHAIR MARCH-LEUBA: It might not be lost.

MR. RAD: Right, may or may not be lost.

MS. McCLOSKEY: Yes.

CHAIR MARCH-LEUBA: Okay, I just wanted to make sure it's on the record that it is an assumption that is passive. And the Commission has not agreed on that yet, correct?

There is a request for --

(Simultaneous Speaking.)

MEMBER CORRADINI: It's still under review. It's an open item also.

MEMBER BROWN: But there are a lot of items which appear to be passive, which you might think are active but are referred to as passive.

MEMBER CORRADINI: The logic, however unusual, makes sense to me. I can take a passive component and claim a failure but I can't then go on and say other passive components can also be assumed under single failure criteria.

That's what he said. I'm sure I said it wrong.

(Simultaneous Speaking.)

MEMBER BROWN: It's just the rules of doing this sort of thing.

MEMBER CORRADINI: Well, let me turn to the Staff, I'll probably muddle it up.

MR. NOLAN: Just one point of clarification, this is Ryan Nolan from the Staff. It's agreed that the IAB is an active component, it's whether it's subject to single failure or not.

So, I just want to make the clarification --

MEMBER CORRADINI: No, no, you said it better than I did. I apologize.

MEMBER BLEY: When you say the loss of DC power can either happen or not, I assume that means you look at it and if that's the worst single failure you can have you use it?

Or is there some other criteria?

MS. McCLOSKEY: It's in addition to the worst single failure.

MEMBER BLEY: In addition to?

MS. McCLOSKEY: Yes.

MEMBER BLEY: And you still might consider it failed. How do you decide which way you consider it?

MS. McCLOSKEY: Whichever is more conservative with respect to minimizing margin to the acceptance criteria.

MEMBER BLEY: So you go both ways?

MS. McCLOSKEY: Yes.

MEMBER BLEY: In all of Chapter 15?

MS. McCLOSKEY: Yes.

MEMBER BLEY: Okay, thank you.

MS. McCLOSKEY: So, the next couple of slides go into that loss of power assumption and how it's treated, and what effect it has on the NuScale design event progression.

Because for non-LOCA events, the power availability affects whether the ECCS valves actuate and then at what time they are postulated to open. For LOCA-type events, the power availability affects the time they actuate and when they open.

So, next slide.

MEMBER CORRADINI: Can I say it a little bit differently just so...So, the only thing in contention in terms of the assumptions, Staff and you are on the same page in terms of assumptions except for the open item relative to the IAB?

In terms of the initiating assumptions you just went through?

MS. McCLOSKEY: Yes.

MEMBER CORRADINI: Is that a correct statement?

MS. McCLOSKEY: Yes, I believe so. We had several RAIs related to the backup --

MEMBER CORRADINI: I figured that's why you made it green.

MS. McCLOSKEY: -- back up valves but I think that's closed.

MEMBER CORRADINI: I just wanted to make sure we summarized it in an appropriate manner, that's all. Thank you very much.

MS. McCLOSKEY: In the case for a non-LOCA-type event that the AC power is available and DC power is available, then very early on in the event progression the module protection system will actuate reactor trip and the decay heat removal system if necessary, if normal secondary side cooling is unavailable.

And then shortly thereafter, on the order of half an hour to an hour timeframe, stable DHRS cooling is established.

If the cooling through the decay heat removal system is very effective and depending on what the initial levels and temperatures in the reactor coolant system are, riser uncovery may also occur in that short timeframe.

If the power remains available and there's no credit for any operator actions to restore conditions, that cooling is maintained for the 72-hour duration.

If AC power is unavailable, then the beginning of the event progression is the same and the difference comes at 24 hours when the ECCS valves are signaled by the module protection system to open. And then the plant will transition to ECCS cooling.

If both AC power and DC power are assumed unavailable, then the ECCS valves are actuated at the time that the DC power is assumed to be lost.

But the event progression is the same because the ECCS valves are held closed by the IAB until the DHRS cooling is sufficient to depressurize the reactor coolant system below the release point, at which point the valves open and it transitions to ECCS-cooled.

CHAIR MARCH-LEUBA: And there is no issue with timing? And the IAB set point is going to be different on every valve so if an RV opens before an RVV or an RVV opens before an RV it doesn't make any difference?

Because when we lose AC power, we trip everything on a 24-hour timeframe at the same time. When you are now relying on the IAB spring to go clean, each valve will trigger at a different time.

Is there any possibility of messing up? Because I would prefer to open the RVVs first and then the RRVs.

MS. McCLOSKEY: In the case of the non-LOCA event progression, by the time that the RCS is depressurized sufficiently, you've been tripped, for a couple of hours you're at decay heat levels.

And a significant amount of the initial energy in the RCS has been transported to the cool. So, that's not a transition that --

(Simultaneous Speaking.)

CHAIR MARCH-LEUBA: The DHRS is cooling slowly, the pressure in the vessel is the saturation pressure at that temperature.

So, slowly the pressure is -- and there will be one valve that will open first and the others will still be closed.

And then there will be a sudden depressurization and all of them will open within a second or two. But does it make a difference, the order?

MS. McCLOSKEY: With respect to margin to the acceptance criteria, I'd say no and that event is bounded by the initiating event of the valve opening while it power-conditions.

CHAIR MARCH-LEUBA: The fact that once you open one, within a couple of seconds everything will open and you are in decay heat?

MS. McCLOSKEY: Yes, in decay heat you're depressurizing. All right, the next slide has the LOCA event progressions.

CHAIR MARCH-LEUBA: I'm sorry, for all of these you have assumed CVCS not to be operational because it's not safety-grade.

But if you keep it going and maintaining water level and pressure, which it could because it's non-safety grade, would it be a bad thing? IAB will never click?

So, you have this same progression event but now the operator didn't look, CVCS continued to maintain the water levels so you are adding water or how you isolate the -- you probably do not isolate.

MS. McCLOSKEY: Go ahead.

MEMBER CORRADINI: You would have a containment isolation I would think.

MR. BRISTOL: If there's no DC power then yes, the modules have been isolated so the reactor is tripped, DHR is actuated and containment is --

CHAIR MARCH-LEUBA: There has to be a loss of power.

MR. BRISTOL: -- containment isolated, yes.

CHAIR MARCH-LEUBA: But if you have a scram without a loss of power?

MR. BRISTOL: And we'll get into this in some of the transient results but yes, there are definitely cases where power available and normal control systems available is more limiting than not.

One specific event is reactivity events. Without pressure control, those trip very quickly on high pressure so it's actually more conservative to assume the spray comes on and maintains pressure control and we eventually progress to a trip condition on overpower over temperature.

CHAIR MARCH-LEUBA: Say you have a forced scram for no reason whatsoever, everything else is working.

CVCS continues to maintain water level in the pressurizer and DHRS comes on and you isolate the secondary. Will CVCS continue to maintain level and dilute the boron?

MR. BRISTOL: With reactor trip, demin water is isolated. So, there's certain conditions in which the demin water supply is available unisolated.

CHAIR MARCH-LEUBA: It has to be unisolated only? The reactor trip will isolate clean water?

MR. BRISTOL: Mm-hmm.

CHAIR MARCH-LEUBA: What I'm coming to hear is there are combinations of things, there are combinations of situations, that work and doesn't work.

I'm not sure that you have considered all of them in the recommendation.

MEMBER CORRADINI: I got the impression they were trying to consider the limiting. They sure haven't considered all combinations. But I thought you were trying to consider the limiting ones.

Am I misunderstanding?

MR. BRISTOL: The focus is definitely on the limiting ones, particularly for the purposes of Chapter 15. When we get into the topical reports, we can get into some of the sensitivities that led us to conclusions of the limiting events.

A lot of sort of the interesting thought exercises get into the progressions for the extended cooling events.

And I think when we get into that discussion later this afternoon, we'll kind of discuss how that progresses and why all the events sort of collapse into a pretty consistent trend.

But yes, we've given consideration to postulated CVCS malfunction, extended cold water injections, things of that nature.

And that's why we have some of these protection systems that are a little bit unique to NuScale, to ensure that we can have that walk-away safe story of MPS will isolate things if operators aren't there and paying attention to make sure that we don't just have an unmanaged or unoperated module.

And we're trying to make everyone safe.

CHAIR MARCH-LEUBA: I'll save it for this afternoon but I'm already thinking of a transient that you haven't done.

MR. BRISTOL: Okay.

MR. INFANGER: This is Paul Infanger. The inadvertent operation or misoperation of CVCS is an event that's analyzed in 15.5. We looked at all the combinations of what CVCS could --

MS. McCLOSKEY: For the increase in inventory event.

MR. INFANGER: Increase in inventory and we looked at boron dilution in 15.4.6.

MS. McCLOSKEY: The loss of power progressions in a LOCA-type event are typically early in the transient. The module protection system actuates the reactor trip, containment isolation because containment is isolated.

The decay heat removal system is also isolated in the design and then ECCS valves are actuated on high containment level in the case of all power available.

And for a realistic event progression, they would be expected to open at that time. And then ECCS cooling is established for the 72-hour duration.

If AC power is unavailable, that event progression is basically unchanged. If DC power is assumed to be unavailable as well, then the ECCS valves are actuated by the loss of DC power.

And we credit the IAB function to hold them closed until the RCS is sufficiently depressurized, and then the valves will open a little bit earlier to establish ECCS cooling.

In terms of the design basis event progressions, the short-term analyses are analyzed from the event initiation until a safe, stabilized condition is reached, by which we mean that the initiating event has been mitigated by the module protection system actuations that are expected to occur.

We've demonstrated that margin to our acceptance criteria has been met and system parameters such as inventory levels, temperatures, or pressures are trending in a favorable direction.

Either the inventory levels are going up or have stabilized, temperatures and pressures are trending down.

And then after the safe, stabilized condition is reached, we get into the longer-term analyses of ECCS long-term decay and residual heat removal, the return to power analyses, and extended DHRS operation.

CHAIR MARCH-LEUBA: I may need your help on this to put this in proper English. But I'm concerned by a little bit of paperwork and we have mode one, mode two, mode three, mode four.

Mode two is hot shutdown that requires you to have a temperature greater than 425 and mode three is safe shutdown which can have a temperature lower than 420.

But my suspicion is that to go to mode three to safe shutdown, tech specs will require a boron concentration before the operator can go from mode two to mode three.

At least Chapter 4 has a table that calculates how much boron you need to have to go into safe shutdown. The table says unless you have 300 PPM you are not in safe shutdown.

It's a function of exposure. So, my suspicion is that's going to propagate into tech specs. Now, by going into the passive cooling on DHRS, the operator is initiating a transfer from mode two to mode three.

You already scrammed, you are in mode two. It initiates a transfer from mode two to mode three without establishing the PPM requirements before he does that. And in Matt's plan that would result in a fine.

Is that correct?

MEMBER SUNSERI: I understand what you're saying but I'm looking at the table right now and the reactivity conditions are the same for both modes, so less than or equal to 99.

So, you don't have to go to the shutdown margin, the cold shutdown margin, if you know what the transition mode looks like.

CHAIR MARCH-LEUBA: If you look at the previous table, there is a PPM required to maintain 0.99 on Chapter 4. I'm just putting it out there.

There's a possibility that by turning passive cooling on, you're actually moving from mode two to mode three. If mode three has some requirements different than mode two, then you are in violation of procedures.

And if it does not have different requirements, why have two modes? I'm just putting it out there because if that happens in a control room today, you will get a fine.

MS. McCLOSKEY: I think I would defer that to Operations and our tech spec folks.

CHAIR MARCH-LEUBA: It's on the record, I can read it afterwards. But there is a mode two and a mode three. They're different. There has to be some difference between the two.

MR. INFANGER: Without operator action, you will transition to mode three and if all rods are in, you will be subcritical under all conditions.

If one rod is not fully inserted, there are some scenarios where you could never return to power and that is analyzed in 15.0.6.

CHAIR MARCH-LEUBA: Okay, you guys continue, let me look at the table in Chapter 4.

MS. McCLOSKEY: Let's go on to the next slide.

So, although the discussion of the topical reports are deferred to later in this qual, we did want to provide a high-level picture of the system's thermal hydraulic methodologies and the links between them, starting with NRELAP5, which is the engine of our system thermal hydraulic analysis for Chapter 15.

And NuScale procured the NRELAP5-3D code from INL and modified it to address NuScale-specific phenomena in systems. And that's what we call the NRELAP5 code.

The code description is provided in the LOCA topical report, along with most of the validation.

The LOCA EM was developed following Reg Guide 1.203 and was extended for analysis of other events, such as the valve-opening events and to focus on other acceptance criteria including containment and the long-term cooling analysis.

The non-LOCA topical report leverages that code description and validation described in the LOCA topical report based on considering differences between the high-ranked phenomena for the different types of events.

And then the containment response technical report is an extension of these evaluation models with focus on the containment pressure and temperature acceptance criteria for both the primary release events and the secondary side pipe-break events.

And then we have overcooling, return to power, and event progression from either the decay heat removal system operation or ECCS and long-term cooling with ECCS as an extension from the LOCA EM.

That's, big picture, how this fits together.

MEMBER CORRADINI: The way I read the chart, just to repeat it back to you so I'm understanding, NRELAP is being used in all of the various ways but in a different set of assumptions or procedures?

NRELAP is the tool but it's used with different assumptions instead of protocols depending upon the application?

MS. McCLOSKEY: Primarily, yes, some different protocols. Primarily different biasing of initial and boundary conditions and different requirements to model secondary side breaks, those sorts of --

CHAIR MARCH-LEUBA: So, the different boxes identify different conservatisms that you have to input on the input of NRELAP? Like, for example, you have to use Appendix K?

MS. McCLOSKEY: Right, that's part of the LOCA topical report. It's not part of the non-LOCA topical report.

CHAIR MARCH-LEUBA: But it tells you what input to provide to RELAP? Instead of providing best testing, you provide Appendix K?

MS. McCLOSKEY: Mm-hmm.

CHAIR MARCH-LEUBA: And are you going to describe the changes, the new RELAP versus the old RELAP? Are you going to describe those?

MS. McCLOSKEY: I think in the closed session this afternoon our subject-matter experts at Corvallis speak to a high-level summary of the changes.

CHAIR MARCH-LEUBA: I'm personally more interested in why they were made so late. Did we find an efficiency analysis that triggered the need to do it?

MS. McCLOSKEY: No, there were --

CHAIR MARCH-LEUBA: It's closed so let's wait.

MS. McCLOSKEY: Some of the changes were error corrections that we identified and needed to correct as part of the normal process of development and maintenance.

CHAIR MARCH-LEUBA: Typically, when you have an error correction you don't rerun the whole Chapter 15. You have outweighed Chapter 15 against the error which did not affect it.

MS. McCLOSKEY: As we're in the middle of the review process, revising the analyses is the path that we are taking.

CHAIR MARCH-LEUBA: If we start doing this, you'll never finish. We'll never finish.

MS. McCLOSKEY: Just one slide on the LOCA EM development, the topical report was developed following the Reg Guide 1.203 evaluation model development and assessment process.

We developed PIRT to identify high-ranked phenomena for the LOCA pipe-break events that is also applicable to the valve-opening events, focused on the short-term response.

We developed an assessment basis for NRELAP5, including the separate effects test and integral effects test to address these high-ranked phenomena. Unique phenomena were addressed by NuScale-specific tests in items such as the steam generators.

The code development, as I said previously, was developed in RELAP5-3D and we performed an applicability evaluation for the EM including bottom-up evaluation of the models and correlations in the code, and a top-down analysis focused on the performance against the IEDs.

The non-LOCA EM, we developed that to perform conservative analyses following the attentive

Reg Guide 1.203 and applying a graded approach, leveraging that code description and much of the validation that was described in the LOCA topical report.

We did a PIRT identifying high-ranked phenomena considering all the different types of non-LOCA events.

And then we did a GAP analysis for those high-ranked phenomena compared to the high-ranked phenomena addressed in the LOCA analyses to identify what had to be addressed in addition for the non-LOCA-specific analyses.

We did do some additional NRELAP5 code validations, focused primarily on the decay heat removal system and the integral non-LOCA response.

MEMBER CORRADINI: Can I take you back? You don't have to go back in the slide but the way you've applied NRELAP is using Appendix K assumptions, not using best estimate assumptions? Am I remembering correctly?

MS. McCLOSKEY: Yes, that's correct.

MEMBER CORRADINI: Does that influence some of the changes you made between 13 and 14?

MS. McCLOSKEY: No.

MEMBER CORRADINI: Okay.

MS. McCLOSKEY: All right, and now I'll turn it over to Ben to talk to several of the transient example results that are presented in Chapter 15.

MR. BRISTOL: Yes, thanks. So, we have these grouped. I'll try to take this at a little faster pace and just, obviously, you can slow me down when you'd like to.

So, analysis results, we have different event types. We'll start with just walking through the sections in 15, how they're categorized.

The first type is increase in heat removal by secondary or we call it coolant events. So, there's a variety of them there highlighted. We are going to walk through an example of an increase in feedwater flow event.

We've got the limiting analysis results summarized with the acceptance criteria in this table. Are you driving? You're driving.

PARTICIPANT: Unless you want to.

MR. BRISTOL: I can. There, transition complete. Okay, so for the case that we've presented in the FSAR, this is 100 percent increase in feedwater flow event. So, that's initiated at time zero.

In this particular event, that's detected by a low-steam super-heat trip so analytical limit is reached. That correlations pretty closely, this is why it's a limiting event, to the high reactor power trip being reached.

And we get a reactor trip signal, DHRS actuation, and peak pressure occurs a little bit of a while later.

CHAIR MARCH-LEUBA: What initiates DHRS?

MR. BRISTOL: Low-steam super heat.

CHAIR MARCH-LEUBA: And it's automatic?

MR. BRISTOL: That's correct.

(Simultaneous Speaking.)

CHAIR MARCH-LEUBA: In real life you would expect the operator to take over that reactor and control it? Or would you leave it like that?

This is the licensing basis, no hands off, but in real life would you expect the operator to defeat the containment isolation and take it off DHRS? Or just leave it on DHRS?

MR. BRISTOL: I think it depends on what the operators are doing but if this is the only module that's being influenced, one of the key things that they try to do is stop the cool-down.

And the reason for that is because, very quickly, the CVCS makeup capacity is very low relative to the shrinkage that is caused by the cool-down.

So, maintaining inventory in the pressurizer is one of the key parameters and the best way to do that is to maintain temperature.

CHAIR MARCH-LEUBA: So, you would expect the pressure to go there and defeat DHRS, open up the secondary and start controlling?

MR. BRISTOL: If the conditions are correct, they may consider closing the DHRS actuation valves if they can reestablish normal feed, in which case then the primary temperature can be controlled.

CHAIR MARCH-LEUBA: But what I'm talking about is, this is the perfect time to tell you on the record, that the table I was talking about before is Table 4.3-2, nuclear parameters for a cycle.

And at the bottom of the table it says boron concentration, for safe shutdown you need 1164 PPM at the beginning of the cycle, 240 at the end of cycle.

It feels to me that somebody when they made these calculations were planning to define all three requirements to have those PPMs. By doing what you just did now, you transfer the core to mode three without achieving those requirements.

So, I'm just telling you that it's

perfectly okay to not have those requirements but if you write your tech specs with this PPM for mode three, the transient you just ran now will serve you a fine, will put you in the yellow column of the evaluation of your plant and you'll have to pay $50,000.

So, somebody has to see the logic of what's the difference between mode two and mode three and is there a requirement for mode three, and is this mode three -- because you're going into mode three. You're going to go below 420 Fahrenheit.

MEMBER SUNSERI: I don't know about that, Jose. Just thinking about how the plants I run work, you have a reactor trip. The tech specs are laid out the same, 0.99 K effective or whatever, right?

So, you know you're shut down if you verify all your rods are in or if one's stuck or whatever. Now the question becomes what's my shutdown margin and how is xenon affecting that and all that kind of thing?

So, you do a shutdown margin calculation. You plug in the 0.99, you plug in your rod configuration, you plug in your temperature and all the reactivity configurations, and you compute what your boron concentration is.

And then you adjust it as necessary over time so you get xenon-free or an equilibrium scenario or whatever it is.

CHAIR MARCH-LEUBA: I don't think it's a real problem, I think it's a paperwork problem and somebody that does operations, ask him to read the transcript and see if he understands what I'm saying.

Let me repeat it one more, you guys have mode two and three, which are different.

They may have different requirements and if you follow what you just described, you jump from mode one to mode three and hands off, didn't touch anything, the possibility exists of you violating your procedures.

So it is incumbent on you to write the procedures correctly.

MEMBER SUNSERI: But they're going to go from mode one to mode two.

CHAIR MARCH-LEUBA: No, they'll go to mode three.

MEMBER SUNSERI: No, because mode three has to be less than 420 degrees. You're going to get there in time but you're not going to get there right away.

It's just kind of like a PWR, you go from mode one to mode three, you skip over two, and you can stay in mode three or you can go to mode four, which is a colder temperature, right?

You have to change your boron concentration if you're going to go colder.

CHAIR MARCH-LEUBA: The moment you scram you go to mode two. Within an hour to two hours --

MEMBER SUNSERI: On this plant.

CHAIR MARCH-LEUBA: On this plant, so the one here described. Within a couple of hours, you're going to be in mode three conditions.

Depending on how you write, how you define your modes, you might be in violation of something. Look at it.

MEMBER CORRADINI: I think we can move on and I think we've got that point.

MR. BRISTOL: Okay. All right, so we walked through the event sequence.

This figure here, we've got the increase in feedwater flow that the scales aren't the greatest but we go from 80 up to 160, quickly isolate. Here's reactor power as a function of time.

Takeaway events detected in the steam generators are isolated before both steam generators are filled, and in this figure DHRS is actuated. This is RCS flow, RCS temperature.

So we've established stable cooling. This is sort of the truncation criteria for the short-term transient response.

CHAIR MARCH-LEUBA: It's important in this transient that you stop -- you turn on DHRS before it overflows.

MR. BRISTOL: That's correct.

CHAIR MARCH-LEUBA: So, the timing of the scram, not the scram but the transfer to DHRS is important.

MR. BRISTOL: That's correct.

CHAIR MARCH-LEUBA: Yes.

MR. BRISTOL: This figure kind of illustrates that a little bit. We've got secondary pressure which isn't really the point here.

The figure is the unimpacted or the steam generator level as the increasing inventory is sort of on this time scale here.

This is where we get the low super heat isolation signal and then as DHR drains we have an additional filling of the steam generator, even at a collapsed level of 80 percent, DHR is still effective.

CHAIR MARCH-LEUBA: So, the steam generator only fills to about 75 to 80 percent?

MR. BRISTOL: That's correct.

CHAIR MARCH-LEUBA: So, that does not impede the DHRS operation?

MR. BRISTOL: No.

CHAIR MARCH-LEUBA: At that level. What level would it fail DHRS? Pretty high up?

MR. BRISTOL: Yes, I don't have that number immediately offhand.

I know in some of the cases we look at, not necessarily the limiting cool-down cases, but this is an event where the feedwater isolation failure is a limiting event from DHR performance perspective.

So, one train, if the feedwater isolation valve were to fail, we're waiting on the feedwater reg valve to close in order to mitigate that overfill.

That's a much slower timeframe valve so the one generator will end up with a higher level than the other generator and it degrades the performance accordingly.

CHAIR MARCH-LEUBA: Do you guys remember that if your steam generator secondary side is full with water, DHRS doesn't work? If the liquid level hits 100, DHRS will be rendered inoperable. And it hits 80 percent.

MEMBER CORRADINI: 100 percent of what?

100 percent of the active tube length? Or 100 percent of all the way up to the -- I want to understand what 100 percent is.

MR. BRISTOL: The tube length.

CHAIR MARCH-LEUBA: So there's still plenty of steam volume?

MR. BRISTOL: And that's primarily at high RCS temperature conditions.

So, if you were looking at this from the sort of minimum or maximum cool-down, as RCS temperature starts to drop, we would quickly see the impact of the increased inventory on performance.

CHAIR MARCH-LEUBA: But DHRS isolates everything?

MR. BRISTOL: Yes, that's right.

CHAIR MARCH-LEUBA: When you close it, whatever inventory you close, that's what you have.

MR. BRISTOL: That's right. Okay, so final figures here, here's DHR performance I think. Am I reading that right? And this is the VIPRE figure for the transient.

Minimum MCHFR is down here, right at reactor trip. So, the conclusion is DHR is still functionally removing decay heat and MCHFR margin exists.

Okay, heat-up events, decrease in heat removal events. Again, here's a summary table, you'll see the RCS pressure in all of these is fairly consistent. That's due to the capacity of the reactor safety valve.

These cases all have the reactor safety valve lifting. That mitigates the pressurization response. And secondary pressures, the heat-up events tend to be pressurization events as opposed to actual temperature-driven events in most cases.

So, the MCHFR is non-limiting for these event types. An example we're walking through I think is loss of AC power and inadvertent actuation of DHR.

So, loss of AC, this is an event we simulate where there's a simultaneous loss of feedwater and turbine trip.

So, the loss of flow to this steam generator as well as an increase in steam pressure or steam pressurization on the steam side quickly results in high-pressurized pressure.

With that big of a transient to the steam generators, the downcomer side actually heats up and swells pretty quickly. That causes a pressurization response that's detected rapidly.

We get a reactor trip and DHR actuation

because of those signals.

MEMBER CORRADINI: Which of the various AOOs fills the generator the most? Is it the one we just went through?

MR. BRISTOL: Increase in feedwater flow, yes.

MEMBER CORRADINI: And if one were to -- okay, I'll stop there and I'll come back. Thank you.

MR. BRISTOL: So, RCS pressure response over here and secondary side pressure response. So the sharp decline here is the pressure increases rapidly until the safety valve lists and then decreases.

As DHR cooling is established, we see sort of this stable cooling trend. One of the things about the way the DHR actuates all of our -- for most of the events, tube failure is an exception, the actual actuation of DHR is what's driving the pressure response.

So, as the secondary system is isolated, then it reaches equilibrium saturated condition with the RCS temperature. And so that's really what's driving this pressure response.

It will keep pressurizing until it's pretty close to the RCS temperature and so for the limiting events of that category, RCS temperature is the main driver, actually, of the secondary pressure response.

Again, flow, RCS flow, and temperature response with DHR actuation. And I sort of alluded to that in this particular case the slight heat-up pressurization actually creates the limiting MCHFR conditions as the initial condition as opposed to some transient CHF.

Inadvertent DHR, so when we're operating at full power conditions the pressure drop across the steam generator is actually a little bit higher than the head, the level head, of the DHR system itself.

So, if one of the valves at the top were to open inadvertently, there's actually a little bit of a feedwater bypass that occurs.

And so this creates a pretty minor loss of feedwater, a little bit of flow injecting into the steam outlet portion of the steam generator. It reaches an equilibrium condition fairly quickly.

In this particular case it takes 400 or 500 seconds for the high-temperature trip to be reached, at which point we get, again, reactor trip and DHR actuation. So, here's a little bit slower transient response in terms of the figure.

There's a slight increase in secondary pressure as temperature is increasing, and then DHR actuation. This kind of shows the steam level response.

The feedwater bypass creates a little bit of a level loss in the steam generators and the difference there in levels is the impact of steam generator versus the non.

Again, RCS pressure response, this is a case where we're actually assuming the modular control system is functioning to keep pressure maintained. That creates the limiting temperature response.

If this were not assumed, you can see by this progression we would reach the high-pressure trip pretty quickly, much more quickly than the high-temperature trip. So, you see the temperature figure over on the spot.

So, in summary, the DHR valve opens and diverts some of the feedwater flow around the steam generators but RCS pressures remain within acceptance criteria.

Here's our CHF figure in reactor power. So, you see over here there's a slight mismatch between the reactor power and the steam generator power which causes the heat-up event.

Reactivity events, again, here's just a quick summary table. Pressure response, these are primarily evaluated for CHF.

The increasing power events generate the limiting category of our CHF-limited events, in particular we'll walk through the single rod withdrawal.

That reacts a little bit like an inadvertent bank withdrawal but it has a unique peaking that's applied in the analysis and that's what causes the local heat flex to increase.

CHAIR MARCH-LEUBA: This is from low power?

MR. BRISTOL: These are from...I think the single rod withdrawal is initiated at 75 percent power. The peaking is just a little bit worse than the 100 percent power.

But the bank withdrawal is analyzed for sort of the startup transient that has the power --

CHAIR MARCH-LEUBA: But the single rod withdrawal is from 75?

MR. BRISTOL: That's right. So, in these cases we actually see that the slower reactivity insertions generate more of a thermal hydraulic transient and that's what generates the limiting conditions.

So, it's a little bit of a chase between the thermal hydraulic conditions getting to the trip conditions versus the overpower conditions. This particular case, as you can see, we reach high hot leg temperature at about 150 seconds.

RCS pressure limit is reached shortly after that, at which point the reactor is tripped and DHR is actuated.

So, here's a figure of the reactivity insertion as a function of time and then the power. So, we start at 75 percent power and increase up to just over 100 percent power.

The reactivity insertion overall and temperature feedback kind of create that cap there, at which point it would trip the reactor.

Pressure and temperature response for the events, there's a slight pressurization the way that the system is modeled. It's actually a simplification that's applied to the spread.

This is another one where the pressure control actually makes the event worse and so there is some pressure control applied until that's overcome by the increasing temperature.

And we've got RCS flow, MCHFR. So, 1.67

MCHFR for this event, or is that 62? I can't read that far.

Okay, increase in RCS inventory, this is actually a category of events that NuScale doesn't really have very severe transients. We look at the maximum, sort of an inadvertent actuation of the maximum makeup capacity.

That's what analyzed here. The consideration really is for a malfunction event where the pressurizer level is increased above the nominal condition to generate a reactor trip, at which point we would take a loss of AC power and a turbine trip.

Could we create a pressure response that looks somewhat different than the normal pressure transients that we look at? So, that's basically what is evaluated.

It's a pretty slow response, I don't have any of the specific results in this presentation. They're in the SR.

So, decrease in inventory events, I'll spend a little bit more time here. We've got a tube failure scenario we'll walk through and then an inadvertent ECCS actuation.

I think this is the RRV and then we'll compare that to the limiting LOCA scenario, kind of set up some of the other topics we'll get into later this afternoon.

So, tube failures are detected and mitigated by RCS level. It's a pretty slow event. The first thing that comes in is the low pressurizer level set point. That's actually a protection set point for the pressurizer heaters.

So, 35 percent is the pressurizer level at which the pressurizer heaters exist axially. And so once the heater starts to uncover, we want to protect them so there's a pressurizer heater trip and a reactor trip that comes with that.

The containment isolation doesn't come further on until 20 percent pressurizer level.

CHAIR MARCH-LEUBA: So, the reactor trip gets tripped, it doesn't isolate anything?

MR. BRISTOL: That's correct. So, upon reactor trip --

MEMBER CORRADINI: How much of it is a change in physical level to go to what you said, 20 percent from normal?

MR. BRISTOL: Normal is at 60 percent.

MEMBER CORRADINI: So what is that?

MR. BRISTOL: Pressurizer is ten feet tall I think.

MEMBER CORRADINI: So, it's almost a foot per...?

MR. BRISTOL: Yes.

MEMBER CORRADINI: Okay, thank you.

MR. BRISTOL: So, if the reactor trips and the heater's tripped, it's a bit of a race but we get to low low-pressurizer level and low-pressurizer pressure pretty quickly.

In this case, low-pressurizer pressure creates a DHR or containment isolation and DHR actuation. And ultimately, that's what mitigates the event.

So, we see here secondary pressure, primary pressure, slight drift down in primary pressure until reactor trip and then we see an increase in DHR actuation.

So, with the reactor trip we start to increase the secondary pressure. Here you can really tell the difference between the impacted steam generator and the non-impacted steam generator.

So, a scaled-out figure just with the pressures, the impacted steam generator quickly reaches equilibrium pressure with the primary system. The other steam generator is on DHR and providing coolant.

So, just a figure of pressurizer level, and steam generator level. Again, you can see here upon secondary isolation the level and the impacted steam generator dramatically increasing.

So, primary inventory remains well above the top of the core in DHR. The unimpacted train in DHR provides coolant.

Here is just a figure of the break flow rate into the integral mass release. We'll get into the radiological results but this is one of the inputs that's provided downstream to the radiological analysis for the dose consequences.

Inadvertent RRV opening, this is a very rapid event.

MEMBER CORRADINI: Is this the one that maximizes -- maximize is the wrong word. This is the one that is the one that's most severe in terms of depressurization?

MR. BRISTOL: No, the inadvertent RVV is a much more rapid pressure transient in terms of the RCS. This is the RRV so this is a liquid space, loss of inventory.

It looks a little bit like the discharge line break. This is the event that sets the limiting containment pressure response analysis, and we'll get into that later as well.

MEMBER CORRADINI: So, this maximizes the peak pressure in containment?

MR. BRISTOL: That's correct.

MEMBER CORRADINI: Whereas the RVV controls or is limiting in terms of the depressurization rate?

MR. BRISTOL: That's right, yes. Yes, the steam space release is a much faster factor loss than the liquid space release.

CHAIR MARCH-LEUBA: But it does not pressurize the containment as much?

MR. BRISTOL: That's right. And we'll get into the containment response, but it's a volume transient for the containment.

If we're increasing with liquid without de-energizing the RCS, then there's more energy once the ECCS actually actuates overall.

Okay, so in this case we get a flow. there is some pressure transient but there is a flow transient in response to the opening of the valve. We get an immediate high containment pressure signal in this case, and that's what causes the reactor trip.

This particular scenario actually simulates a loss of AC and DC power at time zero so with that, reactor trip comes immediately as well as DHR actuation and containment isolation.

And so this is a case we'll kind of see here. This is the case where one RRV is open, the rest go onto IAB immediately with the IAB close function until we reach the IAB release condition, which is the differential pressure.

And so that's where we see the rest of ECCS valves opening shortly into the transient. Level figures, I think the nominal level condition in this particular case doesn't look at failure of one of the other valves to open.

So, in terms of the top of the active core, the RCS level is right there at about ten feet. The containment level is a couple of feet above that. Here's the flow response I could mention.

There's a short flow transient and then with reactor trip we get -- oh, no, sorry, this is the RRV flow rate. So, as the valve opens, containment's pressurizing so the flow drops, and then after ECCS actuation equilibrium is quickly reached.

Zoomed in figure on RCS temperature and then this is the temperature response after ECCS actuation.

Short-term and longer-term RCS flow

response. So, the limiting CHF condition occurs to this little transient flow response there, it's actually driving it.

So, MCHFR occurs in the first couple of seconds and the margins increase after reactor trip. The technique for calculating this we'll get into further in later follow-up discussions, but we have margins to the acceptance criteria.

CHAIR MARCH-LEUBA: So, am I reading this correctly? At time zero the MCHFR is 1.5? Or is there a drop that I don't see? Because I thought it was closer to two.

MEMBER CORRADINI: I think it's higher.

MR. BRISTOL: So, this gets into the differences in the RELAP calculational approach versus the subchannel calculational approach.

MEMBER CORRADINI: -- using your approved NSP...

MR. BRISTOL: Four.

MEMBER CORRADINI: Four, thank you.

CHAIR MARCH-LEUBA: So, this RELAP MCHFR is bias low?

MR. BRISTOL: Mm-hmm.

CHAIR MARCH-LEUBA: Or simply the correlation that you're seeing is bias low?

MEMBER CORRADINI: This is Hench-Levy.

MR. BRISTOL: That's correct.

MEMBER CORRADINI: And you use that to again be conservative? I didn't catch that in the chapter.

MR. BRISTOL: I think we'll defer that discussion to -- are we getting into that later today?

MS. McCLOSKEY: We don't have slides on that later today because that's part of the LOCA topical report.

MR. BRISTOL: Yes, we're giving an overview of the LOCA topical in the closed session and that's bordering on the proprietary so we can discuss it --

CHAIR MARCH-LEUBA: At a high level, CHF correlations are fuel-specific because they depend a lot on the spacers? You're using a generic CHF correlation for the real calculations that do not care what the fuel is?

MR. BRISTOL: Not completely. The analytical limit or the exceptions criteria is based on --

CHAIR MARCH-LEUBA: But that's uncertainties.

MR. BRISTOL: But it's benchmarked to our fuel-specific CHF --

CHAIR MARCH-LEUBA: But the actual value of CHF that you predict, with a fuel-specific CHF correlation versus a generic can be off by 50 percent.

The generic ones tend to bound everything because they don't account for the spacer turbulence. And the fuel fabricator has spent a lot of money creating that turbulence.

MR. BRISTOL: Certainly.

CHAIR MARCH-LEUBA: And are charging you for it.

MEMBER CORRADINI: But just keep in mind, when we discussed this a year ago we agreed that the NSP4 had a limited range of applicability in terms of pressure and flow.

And they're not under these conditions within that range of applicability to pressure and flow. So, my assumption was this was a conservative application.

MR. BRISTOL: I think we are with --

MEMBER CORRADINI: Well, you fall out of it. I mean you start off there but you fall out of range.

MR. BRISTOL: Sure.

MEMBER CORRADINI: Okay, so we're going to wait until later?

MR. BRISTOL: Yes.

MEMBER CORRADINI: Okay.

CHAIR MARCH-LEUBA: So, now let's get into the recreational complaining again about the IAB.

MEMBER CORRADINI: They still have a lot more slides.

CHAIR MARCH-LEUBA: Yes, but let's do some recreational complaining. Filling one IAB gives you a 0.1 delta CHF, nothing compared to your margin where you have two to start with.

You could superimpose this transient on every other AOO and be perfectly okay and save a lot of hassle with that IAB that is so complicated. That's my ten cents. Keep going.

MR. BRISTOL: Thank you.

CHAIR MARCH-LEUBA: It's not worth the fight. 0.1 CHF is not worth the fight.

MEMBER CORRADINI: I don't think it's that straightforward but I understand what Member March-Leuba is saying.

It's the same thing I asked yesterday relative to how all these work in concert.

CHAIR MARCH-LEUBA: Yes. I thought this transient was a lot worse and that's why you didn't want to have it. But it's not bad, it's actually pretty good. Keep going.

MR. BRISTOL: Okay, so to contrast, we'll kind of walk through the LOCA scenario.

The limiting event from a level perspective as presented in the FSR's ten percent injection line break, this is a relatively slow transient although the initial detection's quite quick.

Again even very small breaks of high-energy lines in containment, containment's small, we quickly get a pressure response. And let's see, so this case also assumes loss of AC power at time zero. We get a reactor trip.

So, with the loss of AC power we actually get a pressurization response. The inventory loss is not actually driving the event detection in this particular case.

So, with the reactor trip, shortly after that we get the high containment response, containment isolation, and eventually low pressurizer level, low pressurizer pressure response. And this is a case on IAB.

So, one of the conservativisms in the LOCA EM is DHR heat removal is not credited so there's a slight pressurization that occurs.

Eventually, the inventory starts to turn the pressure around as inventory is increasing in the containment vessel.

That reaches kind of an equilibrium condition and eventually, we're waiting on IAB release out here in the about 6000 second timeframe.

That causes a pretty rapid pressure drop which is driving -- if we look over here, this is our level response. It's at the time of ECCS actuation.

CHAIR MARCH-LEUBA: So, you barely, barely hit core uncovery right there at 6000 seconds?

MR. BRISTOL: Yes, it doesn't actually get to the top of the core from a collapse-level perspective but it's starting to get close to it.

CHAIR MARCH-LEUBA: But you're likely to have some flashing so the actual liquid is higher on that?

MR. BRISTOL: Yes, there's a fair amount of flashing. RCS is at saturated conditions, obviously, at this point and so with this depressurization there's some liquid flashing that occurs.

CHAIR MARCH-LEUBA: And we are not publishing here the CHFR because for LOCA that's not a concern?

MR. BRISTOL: I think we do publish CHFR but it occurs at the initiating event. But CHF is screened as part of the methodology through the entire transient response but not specifically evaluated to --

CHAIR MARCH-LEUBA: But it's not a criteria?

MR. BRISTOL: That's right.

CHAIR MARCH-LEUBA: For LOCA.

MR. BRISTOL: It's evaluated. So, one of the mechanisms for the Appendix K evaluation is ensuring that CHF does not occur.

CHAIR MARCH-LEUBA: Sure.

MR. BRISTOL: As part of the LOCA --

(Simultaneous Speaking.)

-- so within the code there's a model that's built in and it's screened to --

CHAIR MARCH-LEUBA: Screened to criteria. If you didn't go over this, you'd have to do more.

MR. BRISTOL: That's right.

CHAIR MARCH-LEUBA: Before you go on, more recreational complaining. Can you go back to Slide 68? And we see the CHFR starts at 1.5, right? Now, go back to Slide 41?

MR. BRISTOL: 31?

CHAIR MARCH-LEUBA: 41. Yes, that's the one. CHFR starts at 2.8. What's the difference?

MR. BRISTOL: The initial conditions, the models that are applied.

CHAIR MARCH-LEUBA: 2.8 versus 1.5?

MR. BRISTOL: Well, okay, so let me see if I can do this --

CHAIR MARCH-LEUBA: Basically, what I'm saying is make up your mind. Which is it?

MR. BRISTOL: 2.8.

CHAIR MARCH-LEUBA: Wouldn't this be worse if you just have the 1.5? Indeed, I see an decrease of almost 1.0.

MR. BRISTOL: That's right.

CHAIR MARCH-LEUBA: If you just have the 1.5, it will be bad.

MR. BRISTOL: Certainly.

CHAIR MARCH-LEUBA: So, why do you have 2.8 here and not on the other one?

MR. BRISTOL: The conservativisms that go into setting up the channel that's evaluated in the RELAP model are quite conservative and that's really what drives the primary difference in the initial CHF calculation.

CHAIR MARCH-LEUBA: So, the DHRS heat removal event has less conservativisms than the ECCS inadvertent opening?

MR. BRISTOL: This calculation is performed using the VIPRE subchannel methodology and, therefore, it's going to calculate different local thermal hydraulic conditions than the RELAP approach.

CHAIR MARCH-LEUBA: I hope VIPRE will give you a lower ratio. This is rich, right?

MR. BRISTOL: Yes.

CHAIR MARCH-LEUBA: VIPRE is more conservative, it will give you a lower number. Maybe it's the other way around.

MEMBER CORRADINI: It's the other way around. This is using NSP4.

MR. BRISTOL: That's correct.

MEMBER CORRADINI: So this, in theory, has been done with their experiments at Stern with their essentially reactor-relevant bundle in difference to what is being used as a default within RELAP. Am I close?

MR. BRISTOL: That's correct.

CHAIR MARCH-LEUBA: So, this is not the RELAP CHFR?

MR. BRISTOL: This is not the RELAP CHFR, this is the VIPRE CHFR.

CHAIR MARCH-LEUBA: And the ECCS?

MR. BRISTOL: Is the RELAP CHFR.

CHAIR MARCH-LEUBA: And you used VIPRE here because otherwise you couldn't survive it?

MR. BRISTOL: This event we could use the RELAP approach as well and show a large margin.

CHAIR MARCH-LEUBA: It doesn't feel like it. If you start at 1.5 but you lose 1 -- you lose 1. You start at 2.8 and you scram at 1.8 or 1.9. You lose one all in all in CHFR.

MR. BRISTOL: Understood.

CHAIR MARCH-LEUBA: It would have been nice if Chapter 15 had been consistent or at least properly advertise what you're using. And I assume both methods are acceptable.

MEMBER CORRADINI: I think we've got to ask the Staff that separately.

CHAIR MARCH-LEUBA: It feels like cheating a little bit.

MR. LINGENFELTER: Andy Lingenfelter of NuScale. I think, Jose, to answer your question, part of that is wrapped into the topicals.

And while we would have loved to have done the topicals first and Chapter 15 second, it didn't work out that way.

And so maybe when we get to the proprietary section we can give you a little more color on CHFR --

(Simultaneous Speaking.)

That might help.

CHAIR MARCH-LEUBA: So did you use different evaluation methods --

MR. LINGENFELTER: Appendix K for LOCA and we didn't use Appendix K for non-LOCA.

CHAIR MARCH-LEUBA: So, the ECCS open valve you used the LOCA methodology?

MR. INFANGER: When we did the analysis in RELAP for the non-LOCA, we did calculate an MCHFR and we used that as a scoping for what would be the most limiting events.

And then we ran VIPRE on those events to fine-tune it and the VIPRE number is always a lot lower than MCHFR.

CHAIR MARCH-LEUBA: So, this is not VIPRE?

MR. INFANGER: That is VIPRE. Yes, and just ballpark, you had like two if you use the RELAP calculation. So, the RELAP is much, much higher.

CHAIR MARCH-LEUBA: So RELAP would give you 4.8?

MR. INFANGER: Yes. No, the minimum would be like 3.8.

MS. McCLOSKEY: And just to clarify, that screening technique that's applied as part of the non-LOCA EM is different than the CHF evaluation that's performed for the valve opening events using RELAP.

So, that's a little bit of a difference there.

MEMBER CORRADINI: We've got to go to closed session so we can talk this out.

CHAIR MARCH-LEUBA: Right, and I'll make a note to ask the question when we revisit and keep talking, because how come we have factors of two on CHFRs for different methodologies? It would be nice to be semi-consistent.

MR. BRISTOL: Okay.

MEMBER BLEY: I apologize, I was out for some of the time. I don't think you have any slides on that. 15A, you're not going to talk about that, right? Or are you?

MEMBER CORRADINI: It's exempt out. We haven't gotten there yet though.

MEMBER BLEY: But I'm going to leave early, and I apologize for that as well and I'd like to --

MEMBER CORRADINI: But I think similar to --

MEMBER BLEY: There are no slides on it.

MEMBER CORRADINI: Right, and similar to APR1400 it essentially is exempt for a Chapter 15 analysis.

MR. BRISTOL: It is beyond design basis.

MEMBER BLEY: There's an argument in Chapter 15 that is part of the submittal.

MEMBER CORRADINI: Right, but I think this is consistent with past DCs.

MEMBER BLEY: They had an argument in Chapter 15 about it? I don't remember that.

MEMBER CORRADINI: I've got the slides.

MEMBER BLEY: So, you're telling me to shut up?

MEMBER CORRADINI: I didn't say that. I didn't say that at all, I'm just simply saying it's consistent, that's all I said.

MEMBER BLEY: I want to make a comment and ask a couple short questions and then I'll be done with it. The argument at the end in here is that ATWS is covered by the PRA and it calculates a very low frequency of ATWS, which I believe for the electronics getting a signal.

But most of the likelihood of failure to scram in other systems is due to problems with the rods going in. And I don't know that that was covered. So, I'm suspicious of the 1.7-5 per year.

We're not going to argue the PRA here. You begin with the discussion that links back to the anticipatory turbine trip that you don't have, and I'm thinking back 40 years to when that came into play for PWRs with U-tube steam generators.

And the original ATWS calculations at least for some of those saw a very -- well, before it failed to scram, it kept running and used up the inventory for certain transients, the inventory in the steam generators.

Then all of the sudden the pressure goes up faster than the relief and safety valves can relieve and you were going to break something in the system, so they came up with this anticipatory turbine trip to prevent that.

In a design such as this where you don't have any real inventory in the steam generators, I don't think that would apply.

But should you have an ATWS -- and are we going to cover that somewhere else? Maybe -- in your design and the pressure starts taking off inside, you have a couple nice big vent valves.

Does that take the edge off of the ATWS if you actually have one? I'll go back at this in the PRA the next time they come around but if you can say anything about it --

MEMBER CORRADINI: So, can I help you a little bit?

MEMBER BLEY: -- I'd appreciate it. I don't know if you can or not.

MEMBER CORRADINI: There's two things happening.

One, we did have a Staff audit calculation that was presented last month by Dr. Yarsky that went over this and I think the answer to the vent valve, the pressure in the vent valve, is it does open, it does have that.

(Simultaneous Speaking.)

MEMBER BLEY: -- the top off the pressure, okay. I didn't remember that.

MEMBER CORRADINI: I also think there's now -- I'm going to look at the Staff -- a completed report by the Staff, a large report, very interesting, on ATWS which we can get.

MEMBER BLEY: I wouldn't mind getting that.

MEMBER CORRADINI: In fact, I thought Chris Brown sent us an email that that was available.

MEMBER BLEY: I don't know.

MEMBER CORRADINI: I can find the email and send it out to you. He basically sent us a download site --

MEMBER BLEY: For the ATWS?

MEMBER CORRADINI: For the proprietary, for what Staff has done in terms of a whole range of --

MEMBER BLEY: You have helped me, I assume, as long as I can go through it.

(Simultaneous Speaking.)

MR. SCHMIDT: This is Jeff Schmidt from reactor systems. So, that was done under Chapter 19 and Dr. Yarsky did that, that confirmatory. So, reactor systems may have it or don't have it.

MEMBER CORRADINI: But I think we requested it. Some time between last month and this month, Chris sent an email out with a location. I can resend the email.

MEMBER BLEY: I missed that if that came out. That probably covers my --

MEMBER CORRADINI: It has a complete --

(Simultaneous Speaking.)

MEMBER BLEY: -- except for the PRA stuff which I'm a little nervous about.

MEMBER CORRADINI: Okay.

MEMBER BLEY: Thank you.

MEMBER CORRADINI: I'll send the email. I'll make sure that Mike or I send the email to the Committee again.

MEMBER BLEY: I appreciate it.

CHAIR MARCH-LEUBA: With an ML number preferably?

MEMBER CORRADINI: Yes, there is an ML number.

CHAIR MARCH-LEUBA: And it points to the right place?

MEMBER CORRADINI: I was able to download the report.

MEMBER BLEY: So, you've seen it?

MEMBER CORRADINI: I 've seen it. I haven't read it, I had other homework.

CHAIR MARCH-LEUBA: All I know is the last time I saw it, I saw the cover page and they told me they couldn't give it to me.

MEMBER CORRADINI: We've got it.

MR. BRISTOL: Okay, I think that concludes the transient portion of the presentation. We're going to go into the radiological doses next.

MR. INFANGER: This is Paul Infanger.

CHAIR MARCH-LEUBA: Before you start, is this going to be similar to what we've done with the LOCA and non-LOCA methodology?

Because there is a source term methodology topical report that hasn't been reviewed yet.

MR. INFANGER: Right, and we're not going to get into the topical report.

CHAIR MARCH-LEUBA: You're using the results of that report assuming it gets approved?

MR. INFANGER: That's correct.

CHAIR MARCH-LEUBA: Is that what you're doing now?

MR. INFANGER: That's correct. Okay, so the radiological analysis, we used the standard radiological dose consequences that are used in the industry. We used Reg Guide 1.183 and we used that for the acceptance criteria.

And if you look at the events, this is right out of the guidance, and they talk about lost coolant, accidents, fuel-handling accidents, rod ejection accident. But the acceptance criteria is due to damaged fuel.

However, in our events only fuel-handling accidents resulted in damaged fuel. We don't have fuel damage in any of our other accidents.

The first three reactor loss of coolants, fuel-handling accidents and rod ejection accident, they talk about the radiation source being damaged fuel. That's for the acceptance criteria.

But if it's a NuScale reactor, we don't have any accidents of damaged fuel except the fuel-handling accident where it's assumed. And all of our other events we use coolant activity, RCS activity with iodine spiking as the source term.

DR. SCHULTZ: Paul, for the pre-incident spike, what do you assume for the coolant activity? What percentage of fuel failure do you have?

I've seen a couple of different numbers related to that.

MEMBER CORRADINI: You're looking at a fraction of fuel damage?

DR. SCHULTZ: It's the pre-incident spike, yes. I presume that it's related to your technical specification?

MR. INFANGER: We used the tech spec limit for the initial coolant but then you have an iodine spike on top of that, on top of the RCS allowable load.

DR. SCHULTZ: Okay, thank you.

MR. INFANGER: Okay, so for the radiological consequences analysis using the alternative source terms, we used essentially the same recommendations in Reg Guide 1.183 to evaluate the consequences of our design basis event, which is the iodine spiking event, and also, a beyond-design-basis core damage event.

And that's described in detail in the accident source term topical report. For feedwater line break, we reviewed it and found that the steam line break had more limiting consequences so we used that as a bounding event and didn't do a separate feedwater line break dose analysis.

We also looked at the reactor pool boiling, a very large reactor pool, and so Staff had requested us to look at what happens if you had an event where you had long-term boil off of that?

And we found that there's a small amount of tritium in that water from refueling and things like that, and we found that the dose was insignificant.

We looked at potential shine to the control room operators so if you're having an event in the reactor building, there is potential for some reactivity shine into the control room.

And that was found also to have no impact on total dose, or very small.

DR. SCHULTZ: Is that because of the shielding that you have around the control room?

(Simultaneous Speaking.)

MR. INFANGER: Yes, the wall's very thick, a lot of concrete between the reactor building and the control room.

DR. SCHULTZ: And the control room dose is the higher evaluation that you've got in the dose evaluation.

What did you assume for in-leakage to the control room, either from egress or from unfiltered in-leaking to the control room?

MR. INFANGER: I believe it was 10 cfm. I'll have to check on that. Does anyone back in Corvallis have the in-leakage used in the control room dose?

I think that's actually in Chapter 6.4 but Corvallis, are you on the line? I don't know if we have anybody. There's another meeting right now, unfortunately. Our rad protection guys are at another meeting --

MEMBER CORRADINI: You can come back to

us.

MR. INFANGER: We'll get back to you.

MEMBER CORRADINI: I would appreciate that.

PARTICIPANT: Paul, we can have that in a minute.

MR. INFANGER: Thank you. Okay, and of course, the conclusion is that the doses are acceptable for all the events. We used fairly standard industry computer codes to calculate the dose consequences.

We used SCALE, TRITON, and ORIGEN for the types and quantities of radioactive isotopes. We used NRELAP to define the thermal hydraulic conditions for the events and the steam line break.

For beyond-design-basis events, we used MELCOR to calculate a core damage event. The one thing they did a little different than in other sites is we used ARCON96 to calculate the dose for the EPZ and site boundary.

And we did that because the site is so much smaller than other sites, so the industry standard code is put on and that works well for longer distances.

But the ARCON is better utilized for short distances. And so since our site is so small, ARCON96 is used frequently for control room dose and we use it also for our safe boundary dose.

We used RADTRAD to calculate the radionuclide transports. STARNAUA was used for aerosol removal in containment.

We also used a NuScale-specific code, pHT, to calculate the containment pH, which is important for iodine, the evolution. And then we used MCNP for valuing the shine potential.

CHAIR MARCH-LEUBA: You said containment pH?

MR. INFANGER: Yes.

CHAIR MARCH-LEUBA: What is that a function of?

MR. INFANGER: Just the chemistry of the RCS, boron and --

CHAIR MARCH-LEUBA: I would assume that you have that -- were controlling the RCS before you start so how does it evolve? Unless you start to make a lot of hydrogen and things like that.

Never mind, I'm just curious.

MR. INFANGER: This is about NuScale pHT?

CHAIR MARCH-LEUBA: Yes, why is the pH varying during the event? I'm just curious.

MEMBER CORRADINI: He'll satisfy your curiosity later.

MR. INFANGER: Yes, we'll skip that and return to it shortly.

CHAIR MARCH-LEUBA: Don't change the slides, I still have another question. Another curiosity, in SCALE 6.1 TRITON, your design cross-sections are CASMO-5.

So, did you actually run a whole depletion calculation for your fuel with TRITON? I mean, it's a lot of work when you already have it done with CASMO.

Is that because ORIGEN is incompatible with CASMO cross-section? Are you going to convert them, or you don't know?

MR. INFANGER: No, I'm not aware.

CHAIR MARCH-LEUBA: I know people are not using TRITON anymore because it's a big amount of work. You need the whole depletion for so many fuel segments.

So, if you truly did that, you did a lot of work that you didn't need to do because you already have the CASMO cross-sections. So, you can find out?

MR. INFANGER: Yes.

CHAIR MARCH-LEUBA: It's not a minor

activity.

MR. INFANGER: Okay, the accident source term topical report, the number is listed there. It talks about the various sample calculations for each of the events.

The FSAR has the actual events that we analyze for DCA, with our sample calculations in the topical report. As far as using Reg Guide 1.183, we deviated from it and in most cases it was for Appendix C and D are related to BWRs and Appendix G is locked rotor.

Since we don't have reactor coolant pumps there is no such event. So, we didn't use those sections because they're not applicable to our design. We used Reg Guide 1.183's iodine spiking assumptions and decontamination factors for fuel-handling accidents.

We only credited it 23 feet of water even though the pool was much deeper than that. We did iodine removal in the secondary piping or the condenser.

Thermal hydraulic response due to the rod ejection accident shows that there was no fuel damage so we did not calculate a dose from the fuel-handling accident. It would just be handled just with the generic evaluation of RCS --

DR. SCHULTZ: Bounded by the other events.

MR. INFANGER: We used ARCON96 atmospheric dispersion methodology for the short distances to the EAB and LPZ. And we used RADTRAD and modeling techniques consistent with the Reg Guide.

So, again, the AST topical accident source term topical report has the methodology. There was an NEI position paper for small modular reactors, investigating some of the uniqueness of the small cores and information on that related to that.

I'll use B, spectrum of accidents from MELCOR, surrogate accident scenarios. So, we used MELCOR to simulate the core damage event and that was used a lot also for the PRA.

ARCON96 dispersion methodology was used, RADTRAD modeling techniques. And they're now used for the aerosols and pHT.

We showed this slide before and this is a summary of the doses. And if you look at the iodine spike design basis source term, the offsite dose is very, very low, less than 0.01.

The core damage event, again, is a beyond design basis event since none of our accidents involved core damage. But even those are well within the limits.

CHAIR MARCH-LEUBA: Paul, may I ask if the CR means control room?

MR. INFANGER: Yes.

CHAIR MARCH-LEUBA: So, this calculation, does it assume the compressed air system doesn't work?

MR. INFANGER: No, this assumes the compressed air system functions.

CHAIR MARCH-LEUBA: And how does the radioactivity get into the control room?

MR. INFANGER: In-leakage.

CHAIR MARCH-LEUBA: Even though it's high pressure?

MEMBER CORRADINI: That's what I assumed.

CHAIR MARCH-LEUBA: How does it physically happen?

MEMBER CORRADINI: I don't want to answer. I guessed the answer but I don't know.

MR. INFANGER: We'll get the number but I think there's a small amount of in-leakage from ingress and egress.

CHAIR MARCH-LEUBA: Every time you open the door?

MR. INFANGER: Yes.

CHAIR MARCH-LEUBA: So, it's an assumption of how many times you open the window?

MEMBER CORRADINI: It's an assumed number.

MR. INFANGER: It's an assumed number.

CHAIR MARCH-LEUBA: So, it has a large uncertainty, very large uncertainty?

DR. SCHULTZ: That's included in that. It also depends on whether you're filtering intake air.

MR. INFANGER: Initially, we use air bottles to pressurize the control room and after 72 hours we would go to filtered air.

DR. SCHULTZ: And then you have to take consideration of the filters?

MR. INFANGER: Right, and the dose analysis is done for 30 days for the control room.

CHAIR MARCH-LEUBA: So this is for 30 days?

MR. INFANGER: 30 days.

CHAIR MARCH-LEUBA: Okay, so the --

(Simultaneous Speaking.)

MR. INFANGER: The bottles run out after 72 hours.

CHAIR MARCH-LEUBA: Okay, so this is the effectiveness of your HEPA filters?

MR. INFANGER: Yes.

MR. PRESSON: And I would like to bring in

Kenny Anderson from Corvallis to answer a couple of these questions, specifically TRITON.

MEMBER CORRADINI: Is he going to be online or are you going to tell us?

MR. PRESSON: Online.

MR. ANDERSON: This is Kenny.

MEMBER CORRADINI: Go ahead.

MR. ANDERSON: This is Kenny Anderson with Corvallis.

In regards to the question what is TRITON used for, it calculates the microscopic cross-section, that's been said, it calculates macroscopic cross-sections similar to the ORIGEN tool and the SCALE package.

And then that sets the initial inventory for the core and then that inventory is used with the right processing to get the initial inventory into the RADTRAD models or the other models to eventually calculate the dose or the pH.

CHAIR MARCH-LEUBA: My comment was that you already had validated design cross-section sets generated by CASMO-5. And by running TRITON you're reproducing the same numbers that you already have.

And my experience, because I used to sit across from the guy that used to do the work, it's a lot of work to do a cycle depletion with TRITON. So, when did you do the work and who did it?

MR. ANDERSON: Each time we do an FSAR revision we redo it.

CHAIR MARCH-LEUBA: So you do it in house?

MR. ANDERSON: Yes, we commercially grade dedicated the components of the SCALE package that we use and I agree, you could use CASMO-5 to simulate the calculation but there are pros and cons associated with that.

We thought SCALE is the right tool for the right job.

CHAIR MARCH-LEUBA: Well, it gives you a direct plug-in to ORIGEN so I understand it. But my comment was I was surprised because it's a lot of work. But if you know what you're talking about, I'm happy.

Thank you very much.

MR. INFANGER: And Ken, could you talk a little bit about what we use the pHT program for?

MR. ANDERSON: Yes, that takes in the total acids and bases and shows generally a basic solution which as long as you're above a certain threshold, you can prove that there will be no iodine re-evolution.

The iodine that's in the water won't come from out of solution into the air space, which would then be eligible to be released. If there was iodine, we have to assume you'd have higher doses.

CHAIR MARCH-LEUBA: Yes, I understand that but what would change the pH inside the vessel after an accident? What are the input parameters you put to give to the code?

MR. ANDERSON: The initial acid that is released from the fuel and the fuel's damage, that's postulated and then as radiolysis occurs, that's going to be an acid contribution.

And then you take credit for the bases that are appropriate and you do the chemistry balance and get the final pH.

CHAIR MARCH-LEUBA: Okay, so mostly it releases from the fuel, which are chemical components, and radiolysis. Thank you very much.

DR. SCHULTZ: Kenny, we talked about this a little earlier, the shine dose, the way it's presented in a couple of documents is there just isn't any shine dose.

And I'm surprised with regards to the LOCA event and the fuel-handling accident that there's no shine dose to the control room. Can you describe why that would be?

Is there just sufficient shielding so that you don't calculate a shine dose?

MR. ANDERSON: We do have generic shine doses. We take the worst event and we calculate shine doses to the control room.

There's a few different mechanisms where the shine can reach the control room, and then we just apply that to each event. So, that's real conservative.

DR. SCHULTZ: Okay, that's more of what I would have expected. So, there is some but it doesn't increase the calculated dose from the other components of release to an extent that it really makes much difference.

Is that what you're saying? That's why you use a generic value or a maximum value?

MR. ANDERSON: Yes.

DR. SCHULTZ: Okay, thank you.

MR. PRESSON: And Table 15.0-15 contains some of the information on in-leakage. It looks like around 150 cfm.

CHAIR MARCH-LEUBA: I misunderstood. I thought that was not for 30 days but for 3 days.

MR. INFANGER: Okay, and with that, the

next section is on the containment response analysis. Megan's going to address that.

MS. McCLOSKEY: Okay, so as was discussed yesterday afternoon in the overview of Chapter 6, the containment is designed to withstand the full spectrum of primary and secondary system, mass and energy releases, accounting for the worst-case single failure and considering loss of power scenarios.

And the NuScale methodology for analyzing the containment response is based on NRELAP5 and it's described in our technical report. We are --

CHAIR MARCH-LEUBA: For those of us that memorize numbers, is that the LOCA report?

MS. McCLOSKEY: No, that's the containment response technical report.

CHAIR MARCH-LEUBA: It's a different, third report?

MS. McCLOSKEY: Yes.

CHAIR MARCH-LEUBA: Thank you. Have we seen that one?

MEMBER CORRADINI: It's with Chapter 6.

MS. McCLOSKEY: It's with Chapter 6, it's referenced in Chapter 6.

MEMBER CORRADINI: The results are in Chapter 6.

MS. McCLOSKEY: The results are summarized in a table in Chapter 6.

MEMBER CORRADINI: Right.

MS. McCLOSKEY: And then they're presented and discussed in more detail along with the methodology for biasing the initial in-boundary conditions in the technical report.

And so due to the design of the module with the small high-pressure containment and the ECCS valve opening, we are using NRELAP5 for calculating the mass and energy releases from the RPV and the containment pressurization response as an integrated model.

The limiting event scenarios that are addressed, on the primary side we are examining LOCA pipe-breaks in the CVCS discharge and injection lines and the pressurizer high-point vent line. So, two liquid space cases and then the vapor space break.

We also are considering the valve opening events for an inadvertent recirc valve opening or vent valve opening.

On the secondary side, we look at the main steam line breaks and the feedwater line breaks in site containment.

The qualification of the NRELAP5 code for these analyses is based on the qualification and the plant modeling approach that's described in the LOCA and the non-LOCA topical reports for the primary and the secondary-side events respectively.

And this report was used because the containment pressure response was considered as a figure of merit from the PIRT development in both of these Ems.

And in particular, since the primary-side release events are limiting, the LOCA EM identified high-ranked phenomena that are important for predicting the containment response.

So, again, the response is based on the models from those -- developed from the methodologies from those topical reports.

But we are biasing the initial and the boundary conditions used in the models in order to bias the mass and energy release and maximize the containment pressure and temperature response as opposed to addressing the critical heat flux or level acceptance criteria that are addressed in the LOCA topical report.

For the event analyses, we're applying the maximum break sizes or valve sizes in order to maximize the mass and energy release to the containment.

In the limiting-condition cases, the peak pressure occurs after ECCS valve opening and so, as was alluded to earlier, the peak pressure is a little bit of a balance between how much volume you have remaining in containment versus the energy still in the RCS at the time the ECCS valves open.

This slide summarizes the results of the containment analysis result. In the black or the orange font are the limiting case for each of the break or valve opening cases considered.

Grey are the base cases from the containment pressure analysis. So, the difference between the two considers effects of single failures, primarily the effects of single failures and power availability.

The initial conditions are biased consistently and then the containment acceptance criteria are shown at the bottom.

Our limiting event is the inadvertent RRV opening event with a maximum pressure of 986 psia. Compared to the base case, there's about a 45 PSI delta to the base case in that scenario.

That's our largest liquid space discharge event so you get an ECCS valve opening relatively early in the transient.

And that can be distinguished against the RCS discharge line break scenario that's up at the top here, where the base case assumes all power is available so ECCS actuates on high containment level.

And therefore, it's more than 1000 seconds into the event before the ECCS valves open. And that allows some cooling through the containment wall before the valves open and the peak pressure occurs.

CHAIR MARCH-LEUBA: Remind me, what's the limit for the CMV pressure?

MS. McCLOSKEY: It's 1050 PSI here.

CHAIR MARCH-LEUBA: That's what I thought. So this is close.

MEMBER CORRADINI: There was a change to that. You went from some value to 1050. Has that been reviewed and accepted by Staff?

MS. McCLOSKEY: We went from 1000 PSI to 1050 PSI and that was submitted to the Staff towards the end of last year.

MEMBER CORRADINI: So, that's under review and has been accepted?

MS. McCLOSKEY: Sorry?

MEMBER CORRADINI: It's been reviewed and accepted?

MS. McCLOSKEY: Yes, for the SER.

CHAIR MARCH-LEUBA: There are calculations that you satisfy the ASME code at the extra 50 PSI, correct?

MS. McCLOSKEY: Mm-hmm.

CHAIR MARCH-LEUBA: And you will have to do the hydraulic pressure and everything at this pressure? All the testing?

MS. McCLOSKEY: Yes.

CHAIR MARCH-LEUBA: You didn't have to make a change on this side? You're convinced that you have sufficient margin?

MS. McCLOSKEY: No, it was an analysis change.

CHAIR MARCH-LEUBA: I'm always worried when I see a result that is that close to the limit and every single plant has it on other pressure events. So, I will wait until two slides from now to ask you the question.

MS. McCLOSKEY: Okay.

MEMBER CORRADINI: Can I ask a different question? I know we're now creeping into another time window, but the peak pressure at 986 versus 941, are the conservatisms that were assumed to go to limiting case, is there one particular one or are there a series of --

MS. McCLOSKEY: The primary change there is the loss of power assumption, which affects the timing of the ECCS valve opening.

MEMBER CORRADINI: Okay.

MS. McCLOSKEY: And then the limiting wall temperature case is the injection line break case that is a break from the hot leg condition.

The next several slides have results of the limiting pressure case for the inadvertent opening of the RRV. We assume loss of AC and EDSS DC power at the event initiation.

We assume a single failure of the remaining RRV to open, which forces all of the -- when the ECCS valves open, it all has to vent through one RRV or the vent valves, which maximizes the energy release to containment with the ECCS valve opening.

The low bias IAB opening is assumed. We accounted for a fast release of non-condensable gas into containment to account for non-condensable gas that might be present as dissolved in the RCS fluid or in the pressurizer.

Per the methodology, there's no credit for DHRS operation so with the loss of DC power you get a relatively early opening of the ECCS valves about a minute after the event.

And then the peak containment pressure occurs shortly thereafter, but it's rapidly decreased to below 50 percent.

CHAIR MARCH-LEUBA: Here is where I want to ask the questions.

MS. McCLOSKEY: Okay.

CHAIR MARCH-LEUBA: If you look at the top left, which is the pressure, and you concentrate on the high times, you see an ISDK which, to me, that's the condensation of the steam on the vessel wall, containment wall.

Is that your understanding too?

MS. McCLOSKEY: Yes, and what you see in the plot on the lower right are several of the energy balance terms and it's a little hard to read in here.

But I want to clarify here that the containment heat removal shown on this plot is the heat removal to the reactor pool, and that's this red line here. That's when the energy gets all the way out to the reactor pool.

The total energy transfer from the break from the ECCS valves is this pink line that's coming down here. So, it does take 200 seconds before heat removal to the reactor pool is established, and it's well after the peak occurs.

That peak is turned around by the condensation on the inside of the wall and the absorption into the metal.

CHAIR MARCH-LEUBA: And there is on the top left a very nice inflection point at 100 seconds.

MS. McCLOSKEY: That's the ECCS valve opening.

CHAIR MARCH-LEUBA: And there is another inflection point around 400 PSI. My suspicion is the first decreased rate is one ECCS valve dumping steam, then right where you have your mouse, all the ECCS valves start dumping.

And at that point you've run out of inventory. You're dump everything you can dump and now you start cooling.

MS. McCLOSKEY: You dump everything you can dump and you can see the RCS pressure comes down accordingly.

CHAIR MARCH-LEUBA: There's only steam inside the vessel.

MS. McCLOSKEY: Yes. So, we don't lose all of the inventory from the RCS, there's a significant amount of inventory that remains --

CHAIR MARCH-LEUBA: You start boiling it

off.

MS. McCLOSKEY: It flashes, it becomes saturated, and you establish recirculation through the recirculation valves.

CHAIR MARCH-LEUBA: Roughly at 200 seconds, if you look at the bottom left, you have a net, not loss. And what comes in equals what comes out because this is the sum of the break plus the --

(Simultaneous Speaking.)

Now, if you had had higher inventory in the pressurizer, would that peak inflection point on the top one, would it be a little higher? If you had more inventory to lose?

MS. McCLOSKEY: We've biased the inventory high to the high-end of the normal operating --

CHAIR MARCH-LEUBA: This is already --

(Simultaneous Speaking.)

MS. McCLOSKEY: Yes.

CHAIR MARCH-LEUBA: This is fully closed. Okay, good enough.

MS. McCLOSKEY: I think that covers the points on that slide.

MEMBER CORRADINI: Maybe I don't remember. The one thing was the biased high pressurizer level. Are you starting with a vacuum?

MS. McCLOSKEY: Yes, at three psia --

(Simultaneous Speaking.)

MEMBER CORRADINI: There was a discussion between you and the Staff about 65 lost pounds of air. I want to understand where that was. There were non-condensables that were asked about, do you know what I'm talking about?

MS. McCLOSKEY: Yes, we account for the non-condensables both initially present in containment to give you that 3 psia and that are maybe present in the pressurizer vapor space or dissolved in the RCS.

MEMBER CORRADINI: So, is this a corrected calculation with the additional 65 pounds?

MS. McCLOSKEY: Yes.

MEMBER CORRADINI: And the source of the 65 pounds was internal to the RCS or external inside the containment? That's what I couldn't understand.

MS. McCLOSKEY: Internal to the RCS.

MEMBER CORRADINI: And what came out, a solution?

MS. McCLOSKEY: Yes.

PARTICIPANT: And what's in the pressurizer?

MS. McCLOSKEY: Yes.

MEMBER CORRADINI: Okay, thank you.

CHAIR MARCH-LEUBA: Because if it was in containment it wouldn't be operating. It would be --

MEMBER CORRADINI: 65 pounds is not a lot. I'm not sure, one PSI is how many pounds in your containment? I was going to calculate that but --

(Simultaneous speaking.)

MS. McCLOSKEY: The 3 PSI is on the order of 60 to 65 pounds. So, it happens to be about equal to what we get from the RCS as well.

CHAIR MARCH-LEUBA: Well, my brain is telling me is that somebody designed the size of the containment five years ago and we are really lucky because he was right in the know.

Or we've been losing margins since then. I'm sure you don't design your containment that close to limits. Probably you'll be losing margin.

MEMBER CORRADINI: I don't know, there's an awful lot of dries that are awful close.

MS. McCLOSKEY: And I've got two slides where we look at some of the other margins that aren't accounted for in that 986 number.

CHAIR MARCH-LEUBA: This is not unusual. You look at all the hours of pressure and every single BWR, they have less margin than you do.

Because if they hit the limit, they

sharpen the pencil and do a new calculation which is more accurate until he goes under.

MS. McCLOSKEY: The containment is very effective at removing energy after the initial blow-down.

So, in this case, by about half an hour into the event, we've reduced the pressure to below 50 percent of the design limit that's well within the 24 hours expected to support the radiological analyses.

And in terms of the long-term cooling, we'd refer to the long-term cooling technical report to demonstrate continued effective decay and residual heat removal.

And finally, in terms of the margin assessments, the maximum pressure has less than ten percent margin to the acceptance criteria, which is guidance from the DSRS.

Additional factors that are not accounted for in that 986 number that would provide additional margin are both accounting for the external pressures in the assessment and crediting some degree of heat removal from the DHRS.

This maximum pressure is taken at the bottom of the containment and it's conservatively evaluated assuming an external pressure of 0 psia that doesn't account for atmospheric pressure or the pool hydrostatic head.

It would provide about 22 PSI additional margin. The evaluation model also doesn't credit the single- failure-proof safety-related decay heat removal system that would be actuated.

We have some sensitivity calculations indicating about 37 PSI additional margin could be obtained but we've not pursued that due to the additional validations of NRELAP5 that would be required under these scenarios.

So, overall, our conclusion is that our analysis provides assurance that we provide sufficient margins to satisfy the requirements of GDC16 and 50.

MEMBER CORRADINI: Can I suggest a break?

MS. McCLOSKEY: Yes.

MEMBER CORRADINI: Please? Can we take a break until about 3:35 p.m.?

(Whereupon, the above-entitled matter went off the record at 3:23 p.m. and resumed at 3:35 p.m.)

MEMBER CORRADINI: So let's get back into session. So, Ben, we'll turn it back to you.

MR. BRISTOL: Okay, thank you. So the next topic we're going to cover here is long-term cooling. And just to sort of set the expectation, what we're trying to cover here is the scope that was defined in the long-term coolant technical report. So this isn't a full, comprehensive list of all of the issues related to the extended ECCS operation or even the HR operation, but we'll get into those a little bit later. I think we're prepared to address most of them.

So as I mentioned, specifically we're looking at demonstrating that post-LOCA ECCS kind of performance acceptance criteria maintaining coolable geometry and demonstrating ample cooling via ECCS. In addition, a couple other DSRS-specific considerations.

So in terms of acceptance criteria, there's primarily two that we're looking at in term s of this analysis scope, one being that core cooling is maintained and that is evaluated via acceptance criteria of collapsed liquid level, keeping the core covered.

In addition, cladding temperature is evaluated as part of the analysis. And then specifically, something a little different is that coolable geometry is maintained, and that's evaluated through demonstrating boron precipitation limits are not exceeded.

So in terms of the technical reports, RELAP is used to evaluate the integral response to decay heat removal paths from ECCS and containment heat transfer to the pool.

A point of, again, scope: recriticality is not analyzed as part of this analysis, so design basis decay heat or the design base is shut down condition is assumed, that overcooling analysis we'll actually get into in the next presentation. It's otherwise presented in the SR.

CHAIR MARCH-LEUBA: So you mean long cooling under decay heat conditions?

MR. BRISTOL: Under decay heat conditions, that's right.

So in terms of development of the evaluation approach, PIRT was used, addressing ECCS cooling conditions and within evaluation of long-term cooling PIRT phenomena. The LOCA EM is used as a sort of initial validation source. Much of the phenomena is addressed as part of that analysis.

In other places, bounding analytical techniques via inputs or methodology are assumed. We'll address a couple of those.

In addition, there were a couple of assessments that were specifically set up using the

NIST facility --

MEMBER CORRADINI: We'll talk about those in closed session. This kind of goes back to my question about something out of 15.02, and RELAP's ability to model these tests versus what I'd see at full scale.

So are we going to discuss these -- we're not going to discuss these now, I would assume.

MS. MCCLOSKEY: No, we're not going to discuss them now, and --

MEMBER CORRADINI: Otherwise I'll just wait and ask the staff tomorrow. I want to get to -- because staff had some comments about RELAP's ability to calculate what they saw in these relative to consistency in calculation for the full scale, and I want to address it. I just don't know where to do it.

MS. MCCLOSKEY: I think we'll hear the questions in closed session, and then --

MEMBER CORRADINI: Okay. Fine.

MS. MCCLOSKEY: -- try to get that one too.

MEMBER CORRADINI: All right.

MS. MCCLOSKEY: That's the way to respond to this.

MEMBER CORRADINI: Okay.

MR. BRISTOL: So just finally in terms of qualification, the qualification inclusions from LOCA EM are used for evaluating the LOCA-type initiating events, and then the non-LOCA EM is leverage for the non-LOCA DHRS events that then transition.

All of the event types are considered as part of the LTC technical report. All of the events that end up in ECCS mode in terms of what's presented in the technical report.

Okay. So just a couple of details on the precipitation analysis. It's a simple volume mixing approach. There's no time dependence; what we mean by that is that the boron redistributes as part of ECCS actuation.

That way, that redistribution and any of the boron that goes into containment provides would be non-conservative, so it's assumed that essentially the thermohydraulic transient is evaluated assuming that the core and lower riser region contain all of the initial boron mass from the start of the event.

In fact, the actual mass that's used is the bounding sort of startup condition. So for conservatism, the precipitation temperatures are evaluated against the maximum allowable boron mass with no credit for any of the redistribution mechanisms or potential loss mechanisms.

So in terms of the analysis scope, we look at a spectrum of the LOCA Break spectrum, including, in addition to the inadvertent opening of the RRV or RVV. Those are evaluated out through the longer term.

As part of the way the analyses are performed, there's a simplified model; I'm going to get into that in the next slide, but the calculations are set up from the start of the event, and then to ensure that the boundary conditions are effectively transferred.

So instead of transferring them, we actually do comparisons of the full EM models versus the simplified model to demonstrate that we're capturing the transient correctly such that the point that we distinguish the onset of long-term cooling is starting from the correct initial conditions.

As part of the transient scope, we look for -- it's about a 12-and-a-half-hour period that the transient calculations are performed. Beyond that, a state point method is used, and we'll discuss the basis for that.

Simply put, the transient phase or transient progression has recovered within the 12-hour window such that what we're looking for beyond the 12-

hour time frame is simply just limiting temperature conditions. That's where the state point analysis can give us a conservative result.

MEMBER CORRADINI: So maybe this is getting ahead, but do these include the results in terms of your response in RA8930, or is this prior to this?

MR. BRISTOL: No. We'll get into the 8930 topic in the closed session.

MEMBER CORRADINI: Okay. Fine.

MR. BRISTOL: So we essentially have three cases that we're kind of looking for in terms of the way the models are biased. The biases for minimum level are a little about different than minimum temperature. And then a more traditional sort of ECCS performance analysis, looking at maximum temperature is also evaluated.

It turns out maximum temperature cases are non-limiting or minimum cool-down cases are non-limiting for either the precipitation analysis or the minimum-level analysis, and that has to do with the vapor pressures that are reached and their effect on ECCS performance.

So just a summary of the results here: minimum level analysis confirms that the core remains covered, and the minimum temperature analysis confirms that precipitation limits are not reached.

The maximum temperature analysis just shows that we have a trend that's successful from a cladding perspective.

So what we've got in the figure here is 100 percent injection line break as compared to the more limiting LOCA SAL break. The five percent and sort of slower break. We see here that the definition of the onset of long-term cooling for the larger break is after the equilibrium conditions are reached, which is right about here for the larger break.

For the smaller break, it's actually this scope all the way up to the -- this dip is due to the ECCS actuation itself. In three of the recovery phase, this is what we consider in the LOCA methodology, and so the long-term cooling then picks up from there.

And so from kind of a phenomena perspective, we have two different things going on. There's depressurization-driven ECCS kind of capacity minimum level that we get through here. You see a little bit of it in this response here as well.

That's driven by the maximizing containment cooling, which minimizes the vapor pressure and containment and maximizing the stored energy and decay heat in the RCS, which maximizes the pressure in the RCS to maximize the differential pressure across the vent valve. And that's what really drives our long-term cooling minimum levels phenomenon.

MEMBER CORRADINI: Say that again slow.

MR. BRISTOL: So because it's a manometer problem --

MEMBER CORRADINI: Sure.

MR. BRISTOL: -- we want to maximize the deep heat in the vent space, and that's going to establish the actual equilibrium condition in the liquid space. Does that make sense?

So the deep heat across the vent valve, that's the flow-driven DP. The flow through the reset valve is very, very, very small, very low DP. So head is what's driving flow backward.

If my DP across my vent valve, my steam flow rate is too small, I'll continue to accumulate level in the containment, but now allow it to recirculate back into the RPB.

So what we want to maximize is the containment heat removal, which draws the vapor pressure down on the containment side, but the vapor generation rate on the RCS side. And that maximizes the flow capacity that's required for the vent valves.

And these are all evaluated with one train of ECCS assumed failed, which is really the only way we ever see this thermohydraulic transient response.

So in conclusion, I think I covered that in these couple of bullets here. The summary is that once decay heat begins to drop, and it's a combination of decay heat and the stored energy in the system, the level starts to recover the DP, the demand across the vent valves continues to drop, and we see a level recovery. That gives us the confidence that we can kind of switch analytical techniques; we don't need to continue running RELAP through these really kind of slow transient progressions or calculations.

So here's just a couple more figures describing the difference in the two events. Unless there's specific questions, I don't need to cover them in much more detail.

Okay. Minimum temperature case: so this would be a case where all of ECCS is assumed to function. I don't have the level figure in this particular case, but it's one the quickly reaches an equilibrium condition. So minimizing the temperature just by itself, including 100 percent actuation if

ECCS doesn't drive that same level of response, which is why we distinguished those two cases.

So in this case, we're primarily looking at ensuring that as temperature continues to drop, we don't reach the precipitation limits for boron.

MEMBER CORRADINI: Remind -- you're going to go to the next slide?

MR. BRISTOL: Yes.

MEMBER CORRADINI: So again, is the red line five percent break?

MR. BRISTOL: Yes.

MEMBER CORRADINI: So what is going on that I almost come up with the same behavior? I guess I'm a little confused. I'm sorry. I'm looking at the water level again.

MR. BRISTOL: Okay. So in this case, I believe it's an IAB blocked-out case, and so what we have is, without DHR cooling available, so we have a pressurized RCS with a very small --

MEMBER CORRADINI: So both trains of DHRS are blocked?

MR. BRISTOL: Yes, they're not credited. That's correct. So it's a very slow system depressurization, because they're mostly losing liquid inventory out the break, so that's a pretty low-energy release.

Eventually this level begins to accumulate in containment. I get more and more energy released through the RPV wall itself. Eventually, I depressurize to the IAB release pressure window, at which point ECCS actuates and then I continue the transient progression.

So this slide we're just looking at the maximum temperature case for both the two break scenarios. Again, this case would be where the pool temperature is biased to sort of the high condition, assuming we're practically at pool boiling conditions to ensure that even if the pool boiled and we initiated transient from there, we still would provide adequate ECCS performance.

So that's one of the cases where we sort of take an analytical conservatism that's outside the initial conditions of tech specs per se, in order to just sort of deterministically addressing pool heat up effects.

None of the long-term cooling specifically address transient pool temperature, but we do address it from a minimum condition in terms of the constant minimum or a constant maximum.

So long-term cooling conclusions: maximum temperature cases show adequate core cooling. Minimum temperature cases show margin to precipitation, and the minimum level cases show the core stays covered.

Of note, the limiting 100 percent injection line case had the minimum level of about 2.8 feet above the core. That minimum occurred about three and a half hours.

MEMBER SKILLMAN: Ben, are you assuming your minimum temperature for precipitation as 65 degrees Fahrenheit? Was that your marker for precipitation?

MR. BRISTOL: No. We're using, we're evaluating precipitation, so there's some calculations that are performed within the analysis that actually look at the volume, the mixing volume that's assumed, relative to the temperature in the core as a function of time.

MEMBER SKILLMAN: What was the maximum boron concentration you had on that track?

MR. BRISTOL: So I don't have --

MEMBER SKILLMAN: Yes, it isn't in any of your images here.

MR. BRISTOL: I believe it's in -- there's a table of results in the technical report that actually describe the precipitation conditions and the way that calculation's performed. I don't have a slide on that, no.

MS. MCCLOSKEY: And we look at it in terms of margin to the solubility temperature, since we calculate the solubility temperature as a function of the mixing volume, which is the liquid in the core and riser region.

And my recollection from the technical report results is that there's at least 30 degrees Fahrenheit margin to the solubility temperature in all cases.

MEMBER SKILLMAN: Thank you.

MEMBER CORRADINI: Is this the -- I guess maybe I should have asked this earlier. Is this the two-volume calculation where you've got the riser and the core in one volume, and all what's outside of in containment as the second volume?

MR. BRISTOL: No. This is an even more simplified conservative analysis. We'll get into that; I have a presentation --

MEMBER CORRADINI: I don't want to ask much more, but I thought it was -- I wanted to ask when it was -- we'll wait to closed session.

MR. BRISTOL: Yes. And we can circle back to how it relates to these results --

MEMBER CORRADINI: Okay.

MR. BRISTOL: -- when we get to that session.

MEMBER CORRADINI: Okay. Thank you.

MR. BRISTOL: Okay. Are we ready to move on?

MEMBER CORRADINI: Can't wait.

MR. BRISTOL: Loss of shutdown margin. This is the analysis of our postulated return to power event. A couple of elements of licensing vices, I suppose. The NuSCALE DCA includes an exemption request from GDC 27. I think the ACRS has been briefed on that before.

The reason for that: ECCS design does not include boron addition as part of the design, and therefore there's almost no way to meet the as-written definition of GDC 27.

As an alternative, NuSCALE has proposed -- and this is in review with the staff -- the principal design criteria 27. What we're trying to capture is the intent of the GDC as applied to past designs that wouldn't have necessarily active poison addition capabilities.

And so the demonstration of the acceptability of that variant from GDC is highly dependent on showing acceptable results from both an overcooling event with a stuck rod, and the one key, I think, for us in terms of the design criteria is the safety-related reactivity control system is designed and is required to always ensure the core stays sub-critical through cold-check conditions without boron addition.

So it's all rods in on a reload basis, we will demonstrate that the core can go Re critical and stay Re critical out cold conditions.

So compliance immediate shutdown margin is sufficient to turn the events around that we analyzed in Chapter 15 and ensure that the SAFDLs are met. Again, cold shutdown is achieved with all rods inserted, and the loss of shutdown margin consequences are sufficiently benign.

We've defined that to be a SAFDLs evaluation of a limiting return to power condition and that the overall heat removal capacities of ECCS and DHRS are not challenged by the event.

It turns out that heat removal capabilities above those systems are actually what drive the event.

And then on top of that, the overall probability of the combination of events that lead to a loss of shutdown margin is sufficiently small.

So just the reactivity kind of physics overview: there's two main drivers, or three, I guess, that one could postulate. Primarily one is moderator cooling that drives an insertion of reactivity as the systems decrease RCS temperature. Under the limiting kind of cold pool conditions with either DHRC or ECCS, the temperature decrease can happen fairly rapidly on the order of a few hours. We just looked at a couple of those figures already.

Another primary driver is the immediate or the time-dependent fission product decay. Core poisons, and so xenon being the primary one there. It actually inserts negative reactivity for the first few hours post-event, and then gradually decays over the course of 12 to 72 hours, about that time frame.

So in that sort of after six to 12 hours is really when it starts inserting a fair amount of reactivity.

MEMBER CORRADINI: In your analysis, you neglect this, so you conservatively come potentially under some conditions to return to power soon. Am I misunderstanding?

MR. BRISTOL: No, that's correct. That's correct.

MEMBER CORRADINI: And that was something that was required by staff, or is that something you chose to --

MR. BRISTOL: No, that was a conservatism that we discretionarily applied.

MEMBER CORRADINI: Okay. Have you done a calculation to see how much later you actually then delay return to power?

MR. BRISTOL: Yes. The next couple of slides, I'm going to get into that characterization and circle it back to what we've actually presented in 15.6 to try to give that a little more context.

One of the other open items, and again, we've got a presentation on this, but I just wanted to address from a return to power perspective is, boron redistribution.

Overall in ECCS mode, the core is boiling, so that tends to be a concentrator of boron. The containment is the condenser. The vapor leaves with a very low fraction of the boron, as opposed to the inlet conditions.

So we sort of know if the system starts with boron in it, then it's going to tend to accumulate in the core, leading to a lower boron condition probably being more limiting for the return to power analysis. That's one of the things we're working with the staff in reviewing of.

I think for us, the conclusion is, as long as the main loss mechanism of concern would be, could there be some solidification mechanism that then just leads to a loss of soluble boron from the system even after it starts to concentrate, because of the extended period of time that we're talking about.

And so that's a big part of the presentation that we'll get into. Our conclusion is that there's not.

CHAIR MARCH-LEUBA: The other concern is that you're accumulating lower rate in the containment, and 24 hours later when you open the ECCS, the low ratio goes into the lower plane and gets into the core.

MR. BRISTOL: Sure.

CHAIR MARCH-LEUBA: You can get the slag into the water. I just have difficult problem to monitor.

MR. BRISTOL: Right, and that's something I think we have addressed with the staff. Just the way that the ECCS actuation occurs and the way the circulation rates are, there's no mechanism to get a large slug actually through the system. So we've got some analysis of that.

Okay. So I think I covered this. Oh, so this is what I was alluding to. We're working through a better characterization, I think, of the transient nature of the event, using the physics a little more. Initially, the OCRP event that we presented as part of the SR was set up to be analyzed a little bit more like kind of a stylistic AOO analysis where we're using core design limits in trying to approximate these defects.

So the overall reactivity balance was mischaracterized a fair amount from a tiny perspective. The point really was to show that even through a pretty severe return to power kind of analysis, we were a long way from CHF limits and were well bounded by other event analyses.

So the purpose of this work is really a little bit more detailed characterization of the three things I talked about before: specifically that the cooldown rates of both ECCS and DHRS, relative to the xenon-driven reactivity insertion. So I kind of alluded to that before.

Over the 12- to 72-hour range, the reactivity insertion rate is really, really slow. So in sort of looking at this, worst rod stuck out Re criticality is achieved in RCS temperatures below 200 degrees, around 200 degrees, such that we used the state point analysis that we developed for the long-term cooling approach to then do a state point reactivity calculation --

CHAIR MARCH-LEUBA: Using what tool?

MR. BRISTOL: SIMULATE-5 for the --

CHAIR MARCH-LEUBA: Does the SIMULATE effectively calculate -- calculation?

MR. BRISTOL: That's right. So we actually iterated to critical power levels based on thermohydraulic conditions from equilibrium either ECCS or DHR cooling conditions.

CHAIR MARCH-LEUBA: That's good, because you're reaching a calculation with a constant MTC as you're going through the whole transient, which is crazy. SIMULATE is when the regulate operations, correct? You get a 200 degrees Fahrenheit if you have depressed rise, which you open to see if you are boiling.

MR. BRISTOL: That's right.

CHAIR MARCH-LEUBA: So with decay heat, does this calculation come from boiling, or is it not boiling?

MR. BRISTOL: It does not. We

conservatively assumed sub-cooled conditions for the boundary conditions to simulate. So that gives us a much more conservative critical power level.

The power levels are low enough; I'll get to them my next slide. But I think we've got overall confidence in the characterization of the event, even with the conservativisms that are applied.

We have looked at the influence of density with SIMULATE and demonstrated that very little voiding in the core would suppress the power, obviously. And so that's an important conclusion, but there is some concern if the low void fraction rates, where we would justify one, two, three percent boiling rates.

And as any ECCS mode, as we get to lower and lower vapor pressure conditions, the level head in the riser starts to drive a saturated kind of temperature curve, and we could postulate that the boiling could, or some of the two-phase exchange heat removal could be a current closer to the level of interface, and with convective mixing driving the cooling of the core region.

CHAIR MARCH-LEUBA: Considerable.

MR. BRISTOL: Yes. So like I said, we did a steady-state characterization of RCS as a function of the decay heat for both DHR and ECCS heat removal. There's a big spectrum calculation that was done; that was using a read-out model.

And then like I said, the worst rod stuck-out critical power level is iterated to using SIMULATE as a function of RCS temperature.

Conclusions that a loss of shutdown margin can be achieved on kind of the time frame of 40 hours or so with zero boron in the system after xenon's decay. Temperatures are below 200 degrees.

ECCS cooling is a little bit more effective, so it drives a little higher temperature conditions. Again, that's with the sub-cooling assumption is part of that.

Pool temperatures: so one of the things that I thought was important to characterize is that any pool heat of effect that could occur would actually mitigate the event as the pool gets above 140 degrees. It's going to shut -- the RCS temperature just climbs with it, and it's going to shut everything back off.

And then simple CHF analysis; I don't have any of the results, but the pool in the CHF analysis demonstrates large margin.

MEMBER CORRADINI: What do you use for the

CHF analysis?

MR. BRISTOL: That particular case is Zuber.

MEMBER CORRADINI: Foster-Zuber?

MR. BRISTOL: Zuber-Griffith.

MEMBER CORRADINI: Oh, okay.

MR. BRISTOL: So up here we've got the riser uncovered, so one of the things we'll also get to in the closed session, but we've addressed somewhat is riser uncovery nominally adds resistance to the cooling capability. So we looked at a comparison of, if inventory is maintained in the RCS, DHR is going to be more effective than if the riser uncovers, so that's kind of a quantification of the difference there.

Down here this is ECCS cooling or kind of the equilibrium temperature as a function of continual decreasing decay heat.

MEMBER CORRADINI: I don't understand this graph. I looked at it a few times, and I'm lost. Can you help us, please?

MR. BRISTOL: Okay. So each of these points represents a RELAP calculation that we did that set a constant heat input; constant decay heat, constant core power level, using our system model.

MEMBER CORRADINI: Oh, so you basically did a series of parametrics.

MR. BRISTOL: That's right.

MEMBER CORRADINI: Where you set some decay heat level.

MR. BRISTOL: Yes. And the equilibrium temperature condition in the core then is a function of the actual decay heat level.

MEMBER CORRADINI: Right.

MR. BRISTOL: So that's what we're doing there. If we were to assume that there's no residual heat removing, we're at a pure steady-state condition. This would be the temperature solution for that power level. This upper graph, like I said, is with the riser uncovered.

CHAIR MARCH-LEUBA: Only DHRS is working, right?

MR. BRISTOL: And only DHRS working.

CHAIR MARCH-LEUBA: Both of them or only one?

MR. BRISTOL: Two.

MEMBER CORRADINI: Okay. Which maximizes cooling.

MR. BRISTOL: That maximizes the heat removal. This is with the riser covered, so at any given heat level we're going to get quite a bit lower temperature condition, and this is the same calculation performed in ECCS mode.

MEMBER CORRADINI: And the black line?

MR. BRISTOL: This is the SIMULATE results. So --

MEMBER CORRADINI: So you're looking for the intersection of the little dots and the black line?

MR. BRISTOL: That's right. So the critical power level in ECCS mode would be evaluated to be about one percent reactor power --

CHAIR MARCH-LEUBA: Again, how do you do the SIMULATE?

MR. BRISTOL: So the SIMULATE calculation is performed taking the thermohydraulic boundary conditions --

CHAIR MARCH-LEUBA: From?

MR. BRISTOL: From -- well, it's a spectral analysis as well, so we just define -- iterate to a critical power level assuming a core inlet temperature of 100 degrees, 120 degrees, 130 degrees, et cetera. And so --

CHAIR MARCH-LEUBA: In SIMULATE you have to specify the power and the core inlet temperature and the flow?

MR. BRISTOL: In this case we're iterating to critical power level.

CHAIR MARCH-LEUBA: Critical power level?

MR. BRISTOL: That's right.

CHAIR MARCH-LEUBA: But even -- you have to specify a temperature and a flow?

MR. BRISTOL: A temperature and a flow, that's correct. So what we ended up doing is mapping the flow, the DHR flow, correlating to the riser cover condition as the flow condition. What the flow does is essentially just sets the delta T across the core to a given power level.

CHAIR MARCH-LEUBA: But what I'm going to is the black line is a function of core power and flow, and you've brought in the function of decay heat, which is power.

MEMBER CORRADINI: Yes, I don't think that's what he -- that isn't how I understood the curve. The black line is basically, for a set of thermohydraulic conditions of density and temperature, what's the power I achieve? And he's just simply plotting out the excess as if it were an equivalent decay heat.

MR. BRISTOL: That's right.

MEMBER CORRADINI: And so he's basically saying that at the worst set of conditions was the -- now I think I get it. The ECCS little dots, I have one percent plus 0.35 percent as the total power that I'd be producing. It's the red line plus the black line. Am I understanding --

MR. BRISTOL: No, the intersection, we're saying is that --

MEMBER CORRADINI: Oh, you're adding the two together?

MR. BRISTOL: Yes. That would be the equivalent critical power level for the --

MEMBER CORRADINI: But the conservatism that I'm seeing here, unless I misunderstood, is the black line is assuming zero void.

MR. BRISTOL: That's correct.

MEMBER CORRADINI: So I'm way to the right.

MR. BRISTOL: So it's pretty characteristic of the DHR conditions where we're sub-cooled. So that's about the real answer for DHR, and again, in ECCS conditions we do see under the extreme cooldown conditions that we can actually move the boiling up out of the core region itself and into the lower riser as a flashing type phenomena with an amount of convective --

MEMBER CORRADINI: Okay. Fine. But the black line right now is zero void?

MR. BRISTOL: Is zero void.

MEMBER CORRADINI: So I think it's like a reactor static calculation. The black line is a reactor static calculation overlaying and intersecting essentially a thermohydraulic parametric on power.

MR. BRISTOL: That's correct.

MEMBER CORRADINI: I got it, finally.

CHAIR MARCH-LEUBA: I ain't got it, but that's okay.

MR. BRISTOL: Okay.

MEMBER CORRADINI: Just so you guys have seen this before, the shine application for the construction permit when they were doing their annular thing, showed exactly the same sets of calculations about how they would essentially achieve criticality or sub-criticality under the shine medical technology.

Same sort of parametric calculation; that's why --

CHAIR MARCH-LEUBA: I just don't know where the decay heat power comes into the SIMULATE calculation if you specify what the temperature is.

MR. BRISTOL: So decay heat is -- maybe

decay heat is the confusing parameter here. RELAP is just given a point, basically, of a constant power level, and that generates four ECCS conditions at given temperature condition for DHR conditions at a different temperature condition.

And then we're recorrelating the SIMULATE results, which we specify flow and temperature, so we run a spectrum of temperatures. We assume just a constant flow for all of these analyses that's on the conservative high end. It correlates to the DHR covered conditions.

So for ECCS mode it's really pretty high. The internal flow isn't something we would necessarily --

CHAIR MARCH-LEUBA: In a realistic condition the flow would be essentially zero, but you will be uncovered.

MR. BRISTOL: Right.

CHAIR MARCH-LEUBA: And -- anyway --

MR. BRISTOL: Sure.

CHAIR MARCH-LEUBA: I think two reasonable people can come up with five different answers.

MR. BRISTOL: I suspect we'll see some more tomorrow. So on the topic of different answers, going back to what we've actually got in 15.06; so again, the purpose is, when we submitted the DCA was to do a pretty bounding analysis.

So the way that's performed is, a DHR cooldown is more productively performed. There's a couple of key conservatisms; on is that there's no xenon applied as a poison in terms of the overall reactivity balance. The minimum kind of core design limit shutdown margin is assumed. There's a constant MTC applied, and really all that's doing is setting the overall defect that gives us essentially the point at which we go Re critical.

When we do the CHS evaluation, we actually transition and use a density-based curve, so that's sort of a state point difference that's applied. And again, the calculation was actually updated to just apply the density curve through the entire transient.

So with DHR we actually use the temperature transient; generate a power overshoot, and that's used in core physics codes to calculate LOCA peaking factors. They are then applied in the CHF analysis, and then CHF techniques similar to what's used in the LOCA models is applied in terms of evaluating the CHF.

And so these are, I think, the updated figures in the SR. Again, we have the DHR-driven power overshoot. This particular case doesn't actuate ECCS at the peak; that's only performed through the CHF case, so this is the equilibrium DHR condition, and this is was the type of analysis that was used with the stability that Dr. Yarsky was talking about earlier today.

So we take these, apply them in a way in PIM and evaluate the decay ratio that results. Here's a figure of the temperature conditions.

So this is where we take the same scenario and actually open the valves at the point of the power peak. That drives the density-driven feedback. We reach sub-criticality. I don't have the extended figures in this analysis.

There's an RER response that shows if you were to extend this model, which really wasn't the purpose; the purpose was really to make the argument that actuating ECCS up here is more limiting than anything that would be kind of self-limited oscillations down in this regime.

But because the reactivity balance is so far from what's more characteristic of the core, it does show extended oscillations out this way.

And the CHS results: the conclusion in the SR, it's the conclusion still that this is --

either way you model it, it's non-limiting from a CHF or SAFDLs perspective in terms of transience. We're still under review on some of the finer points that I'm sure Jeff will cover more. And that's all I have on that topic.

MEMBER CORRADINI: Just, can we go back?

MR. BRISTOL: Yes.

MEMBER CORRADINI: But these, what you view as bounding analyses with a point kinetics in dynamic, in difference to the previous plot that we were asking about, which essentially is a what-if in terms of power, right?

MR. BRISTOL: That's right.

MEMBER CORRADINI: Okay.

MR. BRISTOL: That's right.

MEMBER CORRADINI: So these are steady-state calculations of a series of what-ifs, whereas the other two are considered to be your bounding calculations for dynamic?

MR. BRISTOL: Yes. So we don't actually believe there would be the dynamic response that we've got presented. And just to kind of contextualize that a little bit, again the purpose of that calculation was to do it kind of generically enough that it would be obvious that it was conservative.

The downside of that is that it paints it maybe a slightly variant picture as to the actual conditions that we would reach in long-term cooling phase.

MEMBER BROWN: Go back to 107. How far out does the line go?

MEMBER CORRADINI: That's 266 minutes.

MR. BRISTOL: That's line -- yes, that's where we truncated -- what we see extending this line is --

MEMBER BROWN: How long do you stay at power?

MR. BRISTOL: Pretty flat response.

MEMBER BROWN: What is it, three percent or two and a half? Something like that?

CHAIR MARCH-LEUBA: No, that's megawatts.

MEMBER BROWN: Oh, megawatts; that's fine. Whatever the power is, how long does it stay there?

MR. BRISTOL: So if we had no temperature increasing in the pool it would stay at that solution until an operator came and did something about it. Again, this is the --

MEMBER BROWN: What would he do?

MR. BRISTOL: Insert the rod, add boron: any number of things.

MEMBER BROWN: But the rod's stuck.

MR. BRISTOL: Shut off the DHR and stop the cooldown.

(Simultaneous speaking.)

MEMBER DIMITRIJEVIC: -- on-site power --

MR. BRISTOL: Actuate ECCS initiates the same --

CHAIR MARCH-LEUBA: I thought the obvious action in the ECCS would be to flood containment with borated water from the pool.

MR. BRISTOL: Yes, that would be --

CHAIR MARCH-LEUBA: And open ECCS.

MR. BRISTOL: Right, right. Again, this isn't really characteristic of the timing of the event. The purpose of this analysis is to show that we can conservatively evaluate.

CHAIR MARCH-LEUBA: Yes, I just feel that a 100 percent realistic analysis will show that this doesn't happen. I mean, it would have been nice that this didn't happen, and we didn't need GDC 27 exception, because look at Charlie. Every time he talks about it, he gets upset, and he is pro-nuclear.

(Laughter.)

CHAIR MARCH-LEUBA: All members of the public will hear this, they say, Oh, what are you building in my neighborhood? I think it's just an issue of the conservatisms that we had it. I don't think this happens.

MEMBER BROWN: It's not a matter of principle. Reactors should shut down under some particular design condition. The ones I'm familiar with that I've worked with over 35 years, when the rod stuck out, we shut down.

MR. BRISTOL: Sure.

MEMBER BROWN: We had to work at it. It wasn't easy, okay? We had to work at it, but that's what you did.

MR. BRISTOL: And I think, to that point, because of the unique aspects of our reactor design, that one stuck-out rod, which is traditionally applied to a big core, is a ton of margin, and --

CHAIR MARCH-LEUBA: His reactors are this size.

MR. BRISTOL: Fair enough. It's a lot of margin for our design, and so in demonstrating passive cooling for a coping period that outlives your poison, that's a lot of temperature defect to try to accommodate in deterministic analysis phase.

Certainly, maybe there's an answer. If one could address all the uncertainties appropriately, that would show, yes, we can squeak by without a critical configuration.

CHAIR MARCH-LEUBA: Can you go back to the dotted line? The one with -- that one. So if I read this correctly, all the red points have a K effective less than one.

MR. BRISTOL: Yes.

CHAIR MARCH-LEUBA: All the blue points that are above the line are K effective less than one, and same with the this. Anything that is under the line has a K effective greater than one and it boils to maintain criticality.

MR. BRISTOL: Or would respond back up to that temperature equilibrium condition.

MEMBER CORRADINI: They'd move up. It would move up.

MR. BRISTOL: So if the coolant were to drive you to here, you'd bump back up to here and --

CHAIR MARCH-LEUBA: You're grossly overcooling. Yes, okay. So if -- you could do it with sub-cooling or with voids.

MR. BRISTOL: Right.

CHAIR MARCH-LEUBA: Correct. So the nominal case for this is the ECCS, which is purple. And what this calculation is saying is, it happens, yes? Anything less than one percent decay heat.

MEMBER BROWN: You stay critical.

CHAIR MARCH-LEUBA: When do you get one percent decay heat for 12 hours?

MEMBER BROWN: No, it's below.

MR. BRISTOL: That's a good question. I would have to follow up on --

MEMBER CORRADINI: What do you mean?

CHAIR MARCH-LEUBA: You get one percent heat is 12 hours?

MEMBER CORRADINI: Three hours.

CHAIR MARCH-LEUBA: Three hours?

MR. BRISTOL: Three hours.

MEMBER CORRADINI: Based on the ANS standard. Rule of thumb is 10,000 seconds-ish.

DR. SCHULTZ: Did you put a team on this to do what Jose suggested? Look at the best estimate? Forget about the uncertainties for a moment and do the evaluation approach with the best code you've got and the best assumptions you can make and worry about applying uncertainty and evaluation later?

MEMBER BROWN: Steve's got a reasonable suggestion. The point is, this is a conservative analysis, and you do a best-estimate analysis under the particular circumstances. If you've done it right, it doesn't go critical.

(Simultaneous speaking.)

MEMBER CORRADINI: I think there's -- I guess maybe I'm different than the rest --

(Simultaneous speaking.)

MEMBER BROWN: -- get up critical and stay critical.

MEMBER CORRADINI: Yes, but I think there's a bounding calculation. That's the way I look at it. There's a set of bounding calculations under two initiators. If I start doing best estimates, I don't have two initiators; I've got a plethora of these, so I've got a plethora of best estimates. And as long as they stay under this, then --

CHAIR MARCH-LEUBA: No, because they all end up in the same place. They'll end up losing DC power after 24 hours or pull a timer and open the ECCS valve and going into the ECCS cooling through the vessel. It continues through; any initiator in transient will end up there, in that position.

MEMBER DIMITRIJEVIC: Well, you have loss of power and stack roll.

CHAIR MARCH-LEUBA: You have stack roll --

MEMBER DIMITRIJEVIC: And loss of power.

CHAIR MARCH-LEUBA: And you have lost

power for 24 hours, plus.

MEMBER DIMITRIJEVIC: Well, I mean, I think -- it's 24 hours because, I mean, if you can get that, you can get boron after 24 hours, you will not get --

CHAIR MARCH-LEUBA: Yes. And you don't take credit for any of the on-site AC power in these times.

MR. BRISTOL: So I think to answer that question, yes, there was a whole lot of consideration that went into any ways that we could, in Chapter 15 space with Chapter 15 rules, and it starts -- you know, you start pulling the thread, and it starts breaking down because again, when deterministically taking that stuck rod is driving the problem, and so if I start trying to pull in, Well, my pull is really going to be 100 degrees. Yes, that totally changes the dynamic response; it improves it quite a bit.

Am I going to operate a plant with a tech spec of 100 degrees? No. So that's not a thread that I can pull.

CHAIR MARCH-LEUBA: Yes, but that's still -- presume these results: say it is considerable that you can have these, but under nominal conditions, when my pool is going to be at 90 degrees or maybe 78 --

MR. BRISTOL: Certainly.

CHAIR MARCH-LEUBA: -- under nominal conditions, it will have it. Because if you could put 75 percent decay heat, you have enough heat up, it boils or heat up that pool will move up.

MR. BRISTOL: Yes.

CHAIR MARCH-LEUBA: So you're just killing yourself and ourselves with you by being too conservative. I would rather not have to give you an exception for anything, because you're satisfied. It's a marketing decision.

MEMBER BROWN: It's a principles decision.

MEMBER CORRADINI: We're shooting for about an hour late. Do we want to continue, or do you want to go into closed session?

MEMBER BROWN: I'm finished here.

MEMBER CORRADINI: I think we have to hear from the public.

MS. MCCLOSKEY: I have one correction to make. When we were discussing the margin to the boron precipitation and the solubility limits, in most of the long-term cooling cases that we have, there's 30 degrees margin of solubility limit or more. The limiting case for an injection line break at low power conditions, and therefore low decay heat has about 16 degrees margins of solubility temperature. I needed to get that correction in.

MEMBER CORRADINI: Thank you.

MEMBER DIMITRIJEVIC: I have a question. Because of the 24 hours loss, when you did this analysis, did you just assume the loss of power is not favorable? It doesn't matter how long, right?

MR. BRISTOL: That's correct.

MEMBER DIMITRIJEVIC: Because if you said that this occurred in three hours, and you analyzed this, if you get loss of power only before three hours, 24 hours is irrelevant. It's just that you lost on-site power?

CHAIR MARCH-LEUBA: You need to open the ECCS valve so you would have additional cooling to the vessel to get here.

MEMBER DIMITRIJEVIC: I know, but that's okay. I mean, 24 hours; why do you -- you don't have on-site power. That's what I'm --

CHAIR MARCH-LEUBA: You have batteries for 24 hours. That's what the assumption is. The batteries last for 24 hours, and then the ECCS open, and --

MEMBER DIMITRIJEVIC: So there is no on-site power, and you will have your batteries deplete.

CHAIR MARCH-LEUBA: You have a timer that you should log.

MS. MCCLOSKEY: There's a timer.

MEMBER DIMITRIJEVIC: Yes. How do you open the ECCS in this case? I mean, what happens?

MEMBER CORRADINI: It just fail --

MR. BRISTOL: So the transient -- if we go to this case, the event progression would be starting at hot zero power conditions in terms of RCS temperature and inventory, and then a simultaneous loss of AC and DC such that we go to IAB.

And right around the time that we're starting to deterministically applying an IAB release condition at this particular point.

MEMBER DIMITRIJEVIC: So you lost AC and DC simultaneously?

MR. BRISTOL: In this particular case, yes, because there would be no -- Dr. Yoo was saying that there wouldn't be an actuation of ECCS until the 24-hour point, except that ECCS was already actuated. So we're just riding the pressure down to the IAB release point and then sort of deterministically applying it as a release conservatism at the time of the power peak. So that's how we get the ECCS actuation.

CHAIR MARCH-LEUBA: This one you assume a DC failure of type zero?

MR. BRISTOL: Mm-hmm.

CHAIR MARCH-LEUBA: So as they cool down is the one that triggers the --

MEMBER CORRADINI: Right.

MR. BRISTOL: Right.

CHAIR MARCH-LEUBA: Yes. That's even less probable.

MR. BRISTOL: Right.

MEMBER CORRADINI: But that's why -- I think we're circling here. That's why I guess my personal view is, these are bounding. We have to make sure the range of potentials are bounded by them, but to try to hit a best estimate, I see that to be a very difficult assignment.

CHAIR MARCH-LEUBA: Oh, this is bounding, but that's two hours after shutdown, and the xenon is sufficient to keep you way down.

(Simultaneous speaking.)

CHAIR MARCH-LEUBA: In the 24-hour --

(Simultaneous speaking.)

CHAIR MARCH-LEUBA: -- because xenon is getting out. Unless you started from a xenon-free startup and you have this crammed right in -- well, anyway, we have to live with it.

MEMBER CORRADINI: Should we go to the public line? It's open? So does anybody in the room want to make a statement?

And on the public line, do we have anybody on the public line? Can you please acknowledge that you're out there?

Okay. Why don't we close the public line, please? And then can I ask NuSCALE and staff to make sure that -- we're going to close the line. We'll keep the NuSCALE staff line open, and we'll go into closed session. We'll just verify the public line is closed. Thank you, Mike. You guys going to stay where you are, and we'll get another set of slides? I can't wait.

CHAIR MARCH-LEUBA: He can misspell your name.

MEMBER CORRADINI: Since we're kind of quasi-open, quasi-closed, if I were to do a best estimate, I would rather just take your steady-state set of calculations and look and see the range of uncertainty on those. I wouldn't go through a dynamic. The dynamic to me just strikes me like a dead end, but at least your steady-state ones were looking for a cross point between a reactor static calculation and a thermohydraulic parametrics.

That, at least, would show you the range of what these things are as I make some assumptions of what's realistic and what's not realistic.

DR. SCHULTZ: You have to set that base case in some fashion and then do parametrics.

MEMBER CORRADINI: Because if you look at their steady state one on slide whatever, it's slightly more than the dynamic, because we're talking about 15 megawatts, where you're predicting in the dynamic of about 10 megawatts under those two bounded conditions.

So I'd rather, if I were an analyst, I'd rather try to noodle with this, which is a whole lot easier to do to try to see what the uncertainties might be, or what the range of uncertainties are, or the assumptions. Because you neglected a whole bunch of things to get to this already.

MR. BRISTOL: Is actually close to two.

MEMBER CORRADINI: Oh, I thought it was one percent of decay heat. Oh, one percent of decay heat.

MR. BRISTOL: One percent of reactor power.

(Simultaneous speaking.)

MEMBER CORRADINI: That's 15 megawatts.

CHAIR MARCH-LEUBA: No, 1.6.

MEMBER CORRADINI: Oh, one percent; 1.6? I'm sorry.

MR. BRISTOL: Yes, so the dynamic response is supposed to be at 10 percent.

MEMBER CORRADINI: Okay. Sorry. But this would be --- well, no, I guess I'm -- this would be -- I wouldn't look at that as the dynamic. I'd look at that at the long term -- the one that Charlie was asking about, where I set a limit.

MEMBER BROWN: The 104?

MEMBER CORRADINI: It's the one over long periods of time, yes? It's not the peak; it's --

MEMBER BROWN: It's 104, not 108 --

MEMBER CORRADINI: Yes. That's what I would --

(Simultaneous speaking.)

CHAIR MARCH-LEUBA: Yes. But that's a calculation based on a constant MTC using point kinetics. That's crazy. I mean, that has nothing to do with reality.

Since we're offering advice, what I would do is, I would take and pull up five. I would input the power one percent decay heat, 1.6 megawatts, and let it settle to whatever it wants to settle, on the flow, on the void, or the temperatures. Then take that void and temperature and put in SIMULATE.

It's going to factor into the one or not. That's a DC calculation. I think you complicated it too much by doing the critical search. And each of those purple points, you're -- it's -- but then on a closed-loop calculation, I recalculate what the flow is, I recalculate what the voids are, and put that into SIMULATE. That's what I would do.

MEMBER CORRADINI: Okay. So you had two pieces of advice, and they're worth about as much as the people that gave them to you, since there's no money involved. So we're closed.

(Whereupon, the above-entitled matter went off the record at 4:31 p.m.)

June 17, 2017 Docket No. PROJ0769

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Subcommittee Presentation: NuScale Topical Report – Evaluation Methodology for Stability Analysis of the NuScale Power Module," PM-0619-65962, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) NuScale Subcommittee Meeting on June 19, 2019. The materials support NuScale's presentation of the "Evaluation Methodology for Stability Analysis of the NuScale Power Module" topical report.

The enclosure to this letter contains the nonproprietary version of the presentation entitled "ACRS Subcommittee Presentation: NuScale Topical Report – Evaluation Methodology for Stability Analysis of the NuScale Power Module."

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 Bavol, NRC, OWFN-8H12

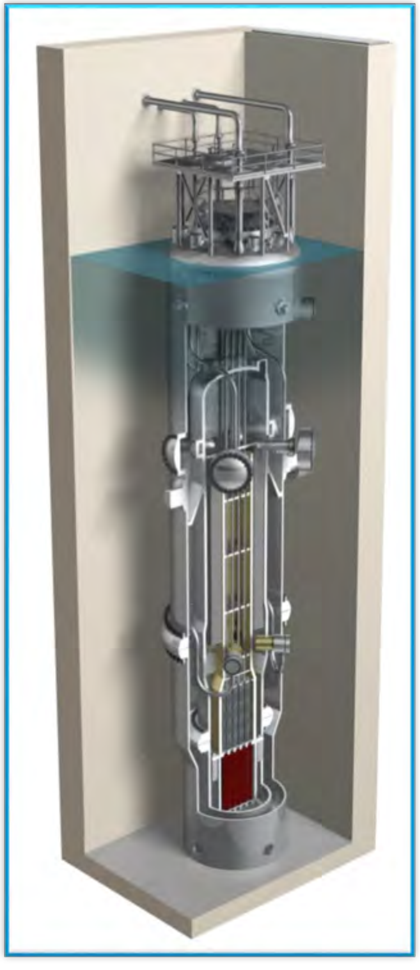
Enclosure: "ACRS Subcommittee Presentation: NuScale Topical Report – Evaluation Methodology for Stability Analysis of the NuScale Power Module," PM-0619-65962, Revision 0

Enclosure:

"ACRS Subcommittee Presentation: NuScale Topical Report – Evaluation Methodology for Stability Analysis of the NuScale Power Module," PM-0619-65962, Revision 0

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

ACRS Subcommittee Presentation



NuScale Topical Report

Evaluation Methodology for Stability Analysis of the NuScale Power Module

June 19, 2019

PM-0619-65962
Revision: 0

Template #: 0000-21727-F01 R5

Presenters

Dr. Yousef Farwila

System Thermal Hydraulics

Ben Bristol

Supervisor, System Thermal Hydraulics

Matthew Presson

Licensing Specialist

Agenda

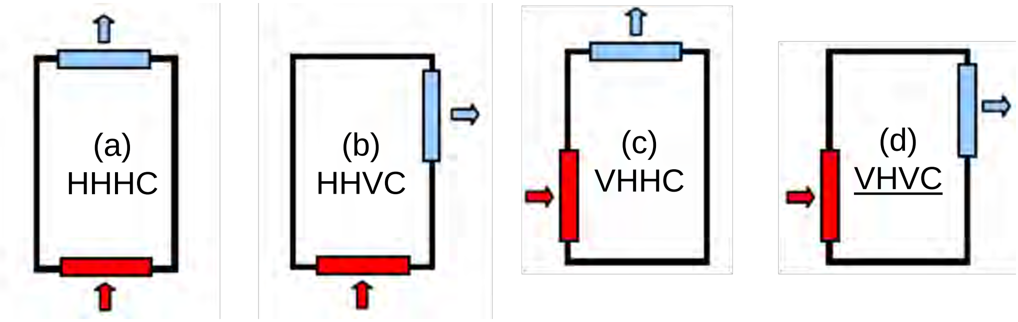
- Introduction
- Stability Solution Type
- Stability Investigation Description
  - Theoretical
  - Numerical Using New Code PIM
  - Experimental Benchmark
- Procedure and Methodology
- Summary
- Questions and Discussions

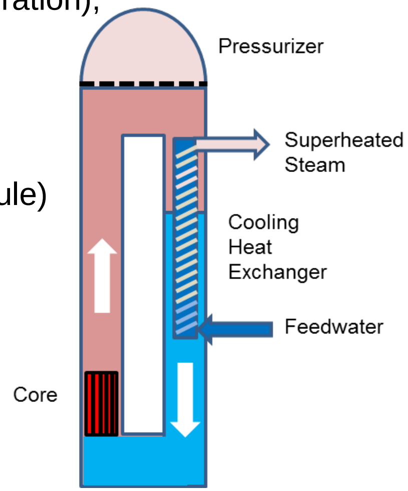
The Main Message

- The NuScale Power Module (NPM) 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.

Stability Evaluation

- Natural circulation instabilities were reported
  - 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

Module Design and Operational Domain
First Principles Theory -- Experience -- PIRT

Construct Models and Main Code (PIM)

Independent Models Reduced Order Model (RADYA)

Experimental Data and Benchmarking (NIST-1)

Comprehensive Analysis & Results
- Steady State Perturbations
- Stability during Transients

Conclusions:
- Stable within Operating Domain
- Threshold is Riser Voiding

Stability Solution:
- Protect riser subcooling with margin
- Hardware already part of MPS

Theoretical Investigation

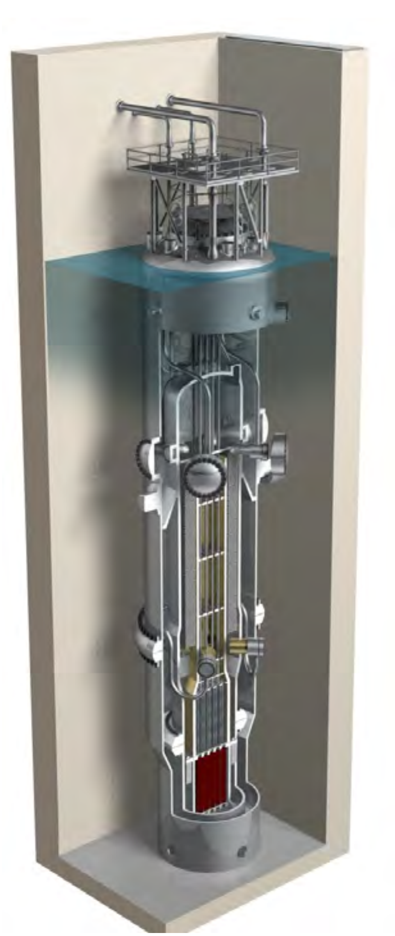
- Kick off with an expert committee to generate a first PIRT
- Scoping review of thermalhydraulic instability modes and contrasting with the NPM design features
- Identification of the possible instability mechanism
- Analysis from first principles
  - Riser-only mode (separate from cold leg)
  - Stability trend with power using a simple SG model
  - Inform design of stability experiments
- All medium ranked phenomena treated as highly ranked

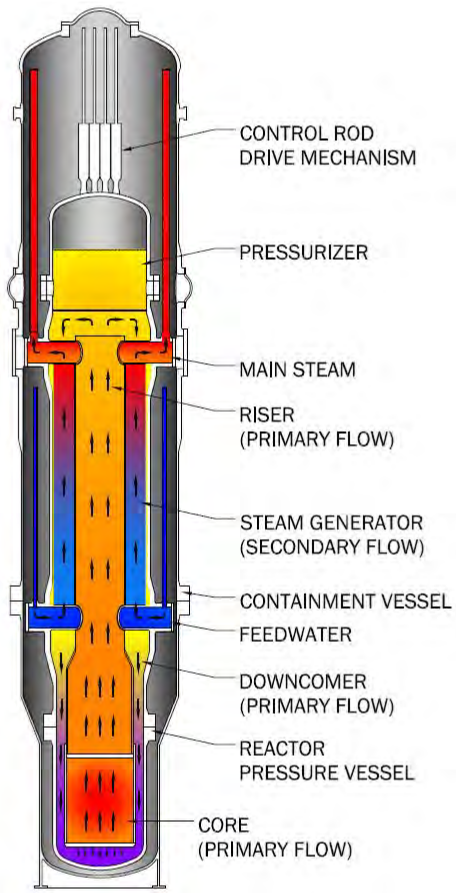
Theoretical and First Principles

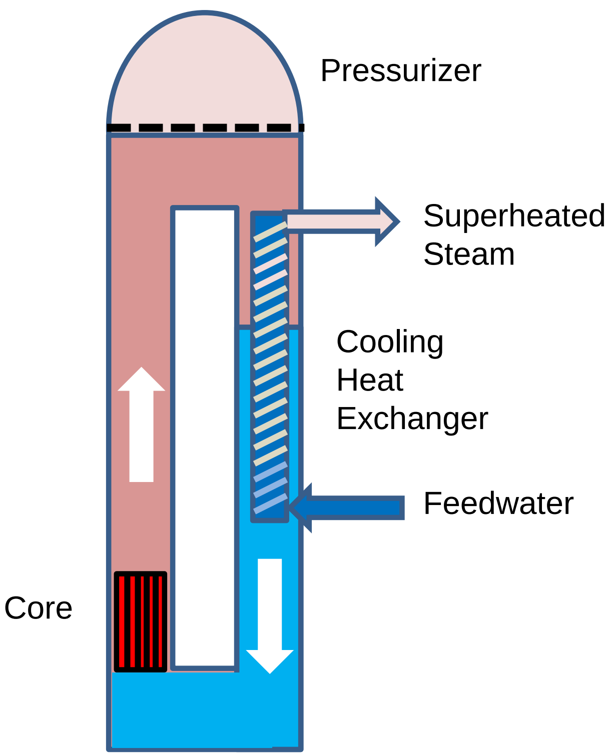
- A system with feedback processes may undergo oscillatory instability if the feedback is:
  - Negative (positive feedback is unconditionally unstable)
  - Delayed
  - Sufficiently strong
- NuScale natural circulation mode is examined
  - Feedback is negative. A perturbation increasing core flow decreases 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







PM-0619-65962
Revision: 0

Template #: 0000-21727-F01 R5

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)

$$\frac{d}{dt}\left(M_{l,n} h_{l,n}\right) = \dot{m}_{l,n-1} h_{l,n-1} - \dot{m}_{l,n} h_{l,n} - \Gamma_n h_{fg} + \dot{Q}_n$$

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

$$\Lambda \frac{d\Phi}{dt} = \beta\left(\rho - 1\right)\Phi + \lambda C$$

$$\frac{dC}{dt} = \beta \Phi - \lambda C$$

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

PIM Results of Perturbing SS

- 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 CHFR 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

London Office
1st Floor Portland House
Bressenden Place
London SW1E 5BH
United Kingdom
+44 (0) 2079 321700

[*http://www.nuscalepower.com*](http://www.nuscalepower.com)
Twitter: @NuScale_Power

NUSCALE™
Power for all humankind

Presentation to the ACRS Subcommittee

Staff Review of NuScale Topical Report

TR-0516-49417-P, REVISION 0
“EVALUATION METHODOLOGY FOR STABILITY ANALYSIS OF THE NUSCALE POWER MODULE”

Presenters:

Ray Skarda, Ph.D.- Reactor Systems Engineer, RES
Peter Yarsky, Ph.D.- Senior Reactor Systems Engineer, RES
Bruce Bavol - Project Manager, Office of New Reactors

June 19, 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)

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 brief advisory committee on reactor safeguards (ACRS) <u>full</u> committee on July 10, 2019
- 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.

Outline

Primary Review Areas

- Regulatory Criteria
- Long Term Stability Solution
- Instability Modes and Phenomenology
- PIM Evaluation Model
- Uncertainty and Acceptance Criteria
- Stability with Worst-Rod-Stuck-Out (WRSO)
- Stability Topical Report Conclusions
- Design Certification Document (DCD) 15.9 Stability

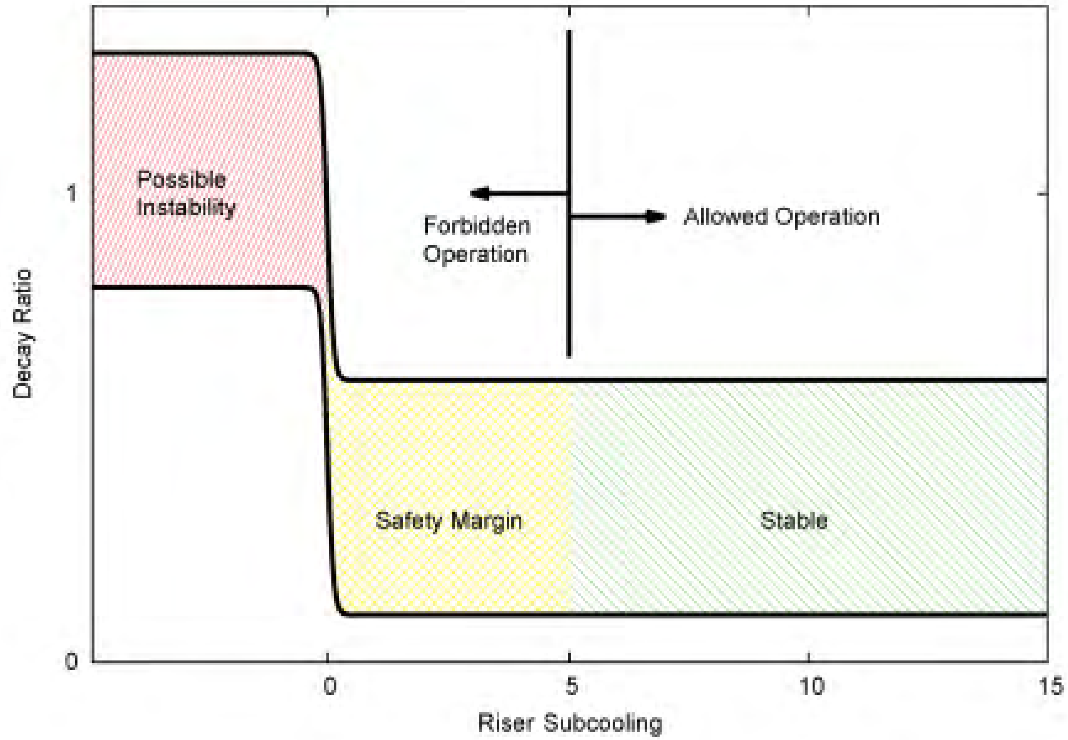
Regulatory Criteria

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.

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.



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.

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.

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.

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.

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.

Questions/comments before the closed session starts?

Backup Slides

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.

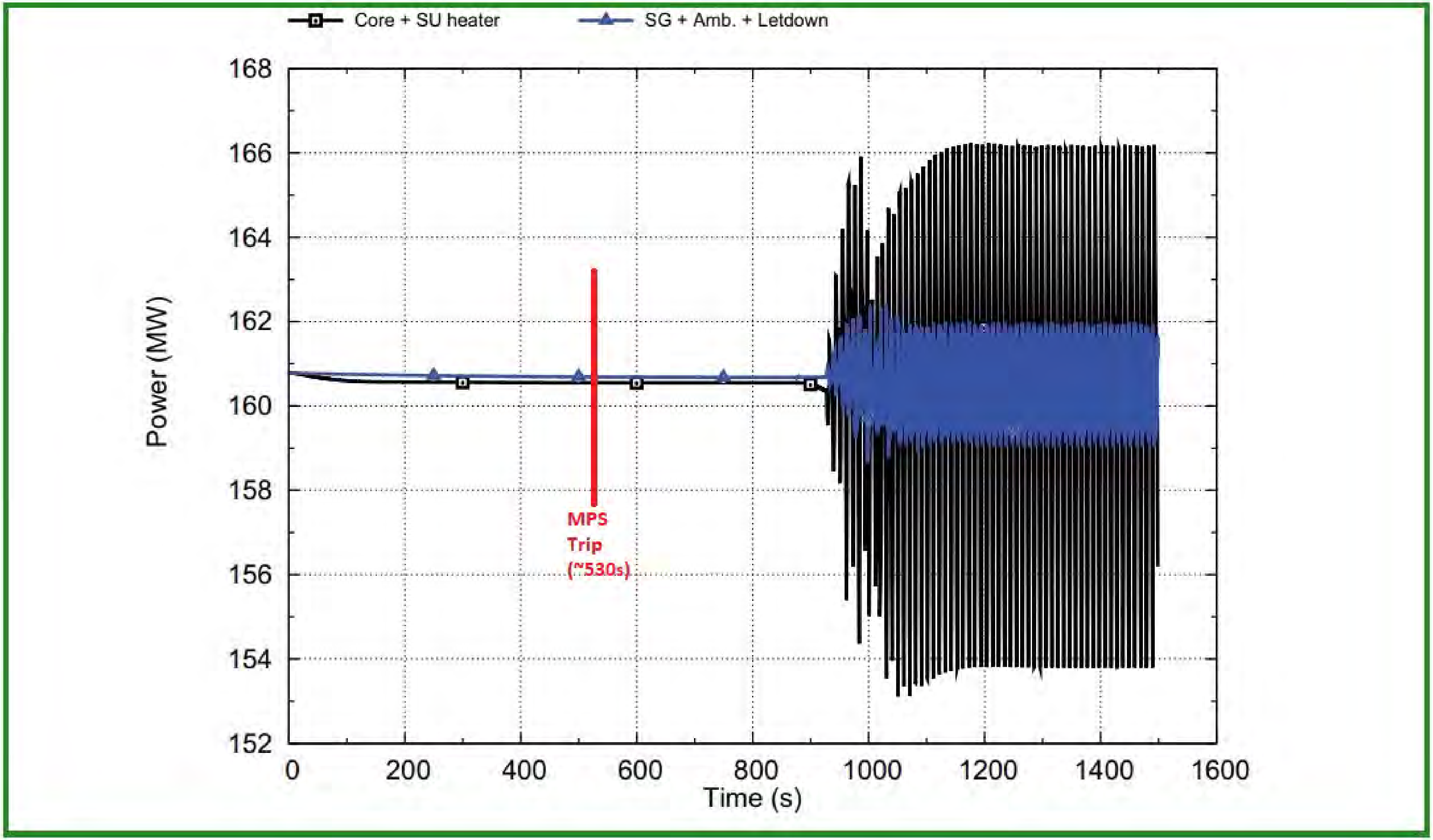
DCD 15.9 Stability Review
Reactivity and Power Distribution Anomalies

- 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



Non-Proprietary

June 17, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Subcommittee Presentation, NuScale FSAR Chapter 15, Transient and Accident Analyses," PM-0619-65981, Revision 0

The purpose of this submittal is to provide presentation materials for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) NuScale Subcommittee meeting on June 19, 2019. The materials support NuScale's presentation of Chapter 15, "Transient and Accident Analyses," of the NuScale Design Certification Application.

The enclosure to this letter is the nonproprietary presentation entitled "ACRS Subcommittee Presentation, NuScale FSAR Chapter 15, Transient and Accident Analyses," PM-0619-65981, Revision 0.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Matthew Presson at 541-452-7531 or at mpresson@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Robert Taylor, NRC, OWFN-7H4
Michael Snodderly, NRC, TWFN-2E26
Gregory Cranston, NRC, OWFN-8H12
Samuel Lee, NRC, OWFN-8H12
Rani Franovich, NRC, OWFN-8H12

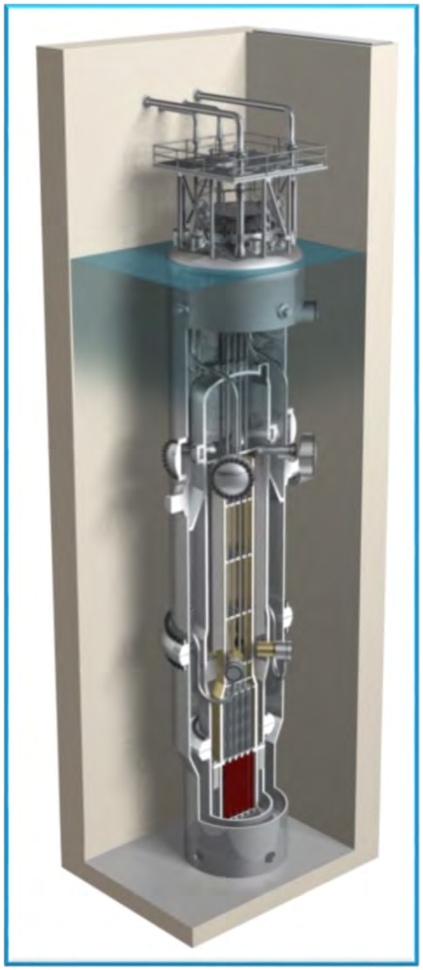
Enclosure: "ACRS Subcommittee Presentation, NuScale FSAR Chapter 15, Transient and Accident Analyses," PM-0619-65981, Revision 0

Enclosure:

“ACRS Subcommittee Presentation, NuScale FSAR Chapter 15, Transient and Accident Analyses,” PM-0619-65981, Revision 0

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

ACRS Subcommittee Presentation



NuScale FSAR

Chapter 15

Transient and Accident Analyses

June 19, 2019

PM-0619-65981
Revision: 0

Template #: 0000-21727-F01 R4

Presentation Team

Ben Bristol	- Supervisor, System Thermal Hydraulics
Dr. Pravin Sawant (*)	- Supervisor, Code Validation and Methods
Dr. Brian Wolf (*)	- Supervisor, Code Development
Dr. Selim Kuran (*)	- Thermal Hydraulic Software Validation
Meghan McCloskey	- Thermal Hydraulic Analyst
Mark Shaver (*)	- Supervisor, Radiological Engineering
Matthew Presson	- Licensing Project Manager
Paul Infanger	- Licensing Specialist
Greg Myers (*)	- Licensing Specialist

(*) 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 Radiological Dose Consequences

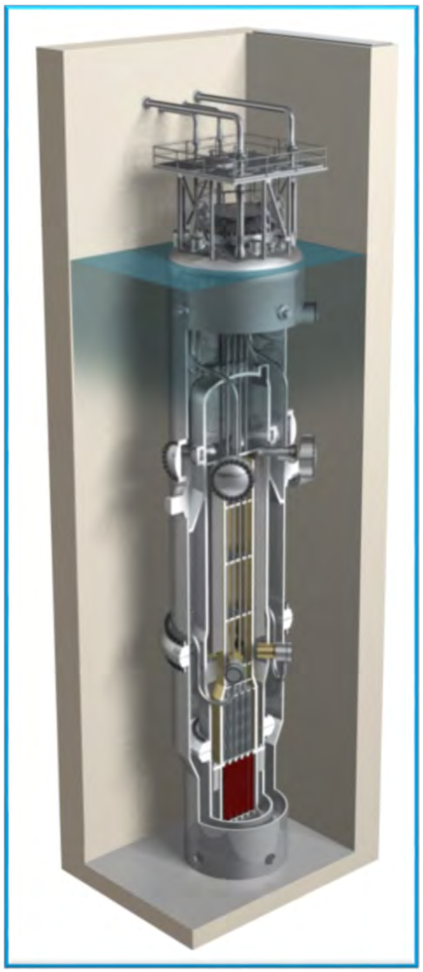
Event	Location	Acceptance Criteria (rem TEDE)	Dose (rem TEDE)
Iodine Spike Design Basis Source Term(1) (pre-incident iodine spike)	EAB	25.0	<0.01
	LPZ	25.0	<0.01
	CR	5.0	<0.01
Iodine Spike Design-Basis Source Term(1) (coincident iodine spike)	EAB	25.0	<0.01
	LPZ	25.0	<0.01
	CR	5.0	<0.01
Core Damage Event(2)	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.

Chapter 15 Overview Agenda – June 19th

- NuScale design overview
- Ch 15 overview
- Analytical assumptions for Ch 15 analysis
- System T/H analysis methodologies
- Selected transient results
- Radiological analysis
- Ch 6.2.1 Containment response analysis
- Long term cooling

ACRS Subcommittee Presentation



Chapter 15
NuScale Design Overview
Safety/Non Safety Systems

June 19, 2019

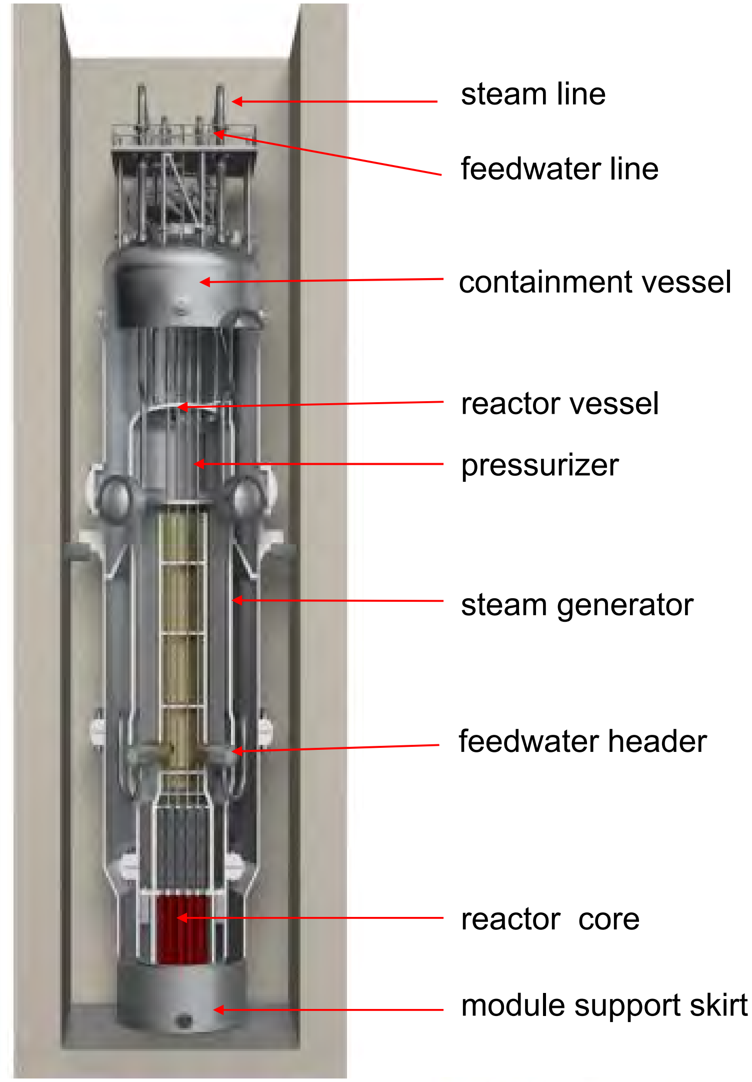
PM-0619-65981
Revision: 0

Template #: 0000-21727-F01 R5

Power Module Overview

Integral Pressurized Water Reactor

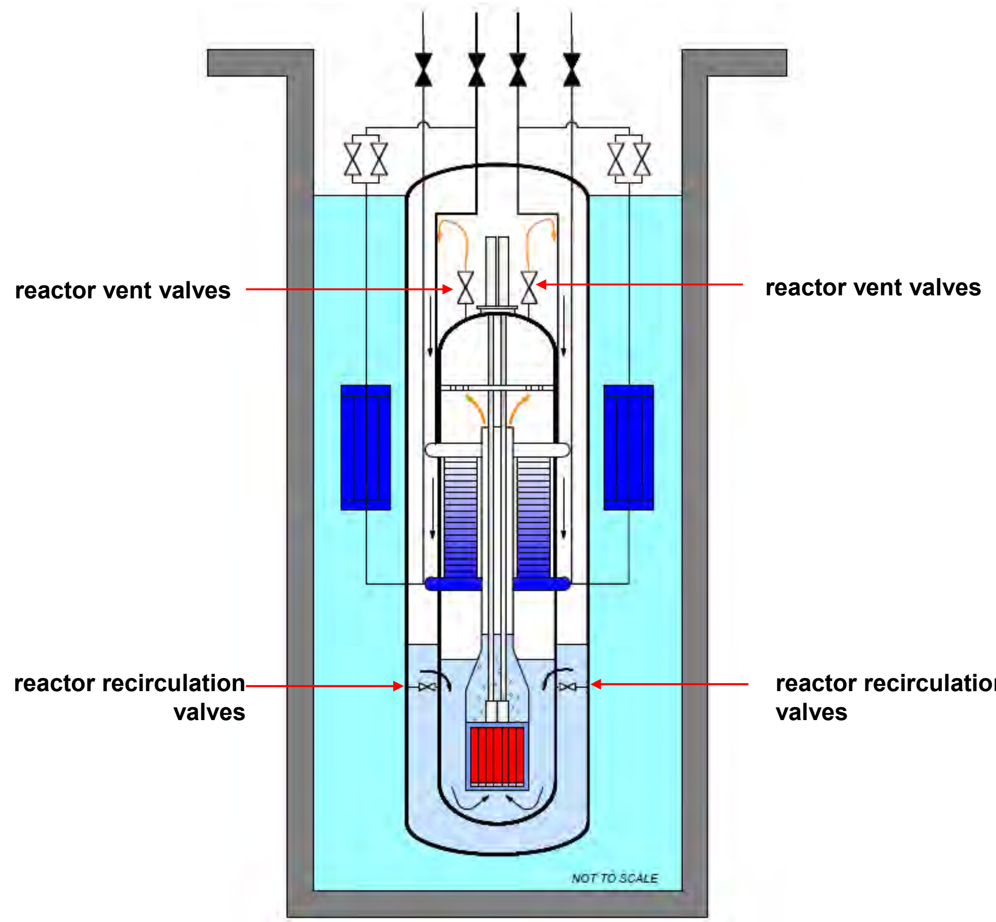
- Core, steam generator and pressurize in one vessel
- Integrated reactor design, no large-break loss-of-coolant accidents
- Reactor coolant system operated in single phase (liquid) density driven flow
- Safety decay heat removal systems are passive and fail safe
- Module protection system designed to automate event mitigation actuations (no operator actions)



ECCS

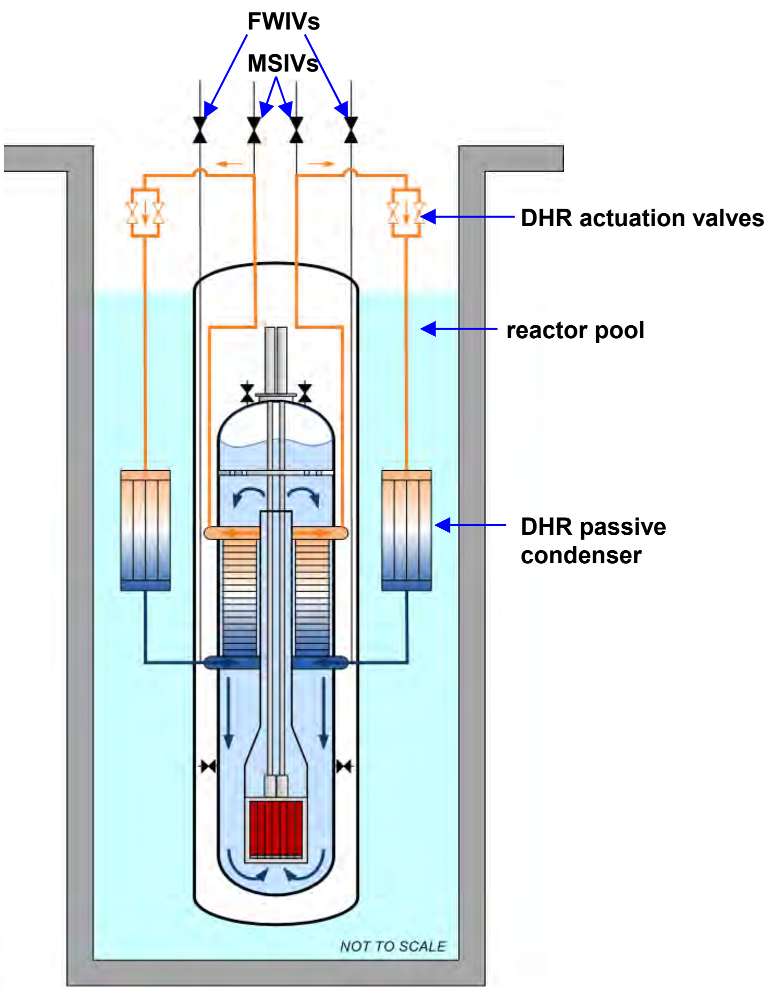
Emergency Core Cooling System

- ECCS valves open to a boiling/condensing circulation flow path to transfer decay and residual heat to reactor pool
  - Liquid from containment vessel enters RCS through reactor recirculation valves
  - Vapor vented from RCS to containment vessel through reactor vent valves
  - Steam condenses on inside surface of containment vessel
  - Heat transfer through vessel walls to the reactor pool
- Actuation Signals: High CNV level, 24hr loss of AC power
- Fail safe: ECCS valves open on loss of DC power

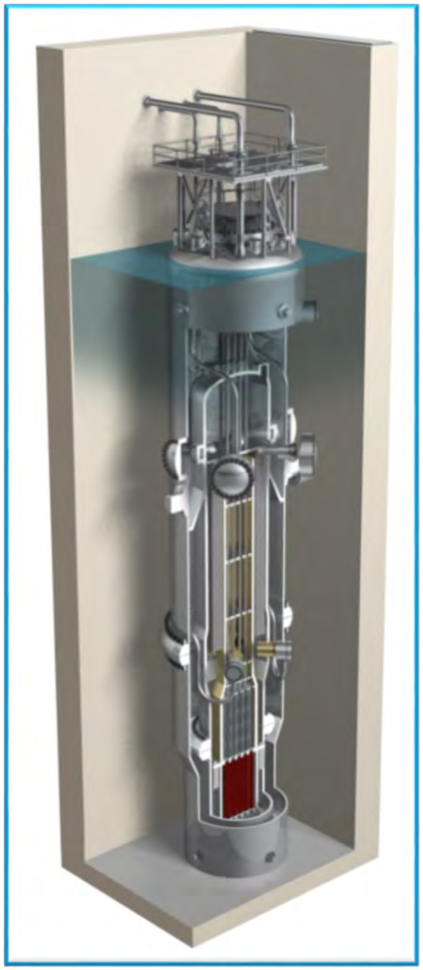


Decay Heat Removal System (DHRS)

- Removes heat after loss of normal cooling
- Boiling/condensing loop
- Two redundant trains
- Redundant actuation and isolation valves on each train
- Initiates on:
  - Loss of power
  - Loss of cooling indication (ESFAS Signal)



ACRS Subcommittee Presentation



Chapter 15.0
NuScale Design Overview
Event Classification, Acceptance Criteria, Methodology

June 19, 2019

PM-0619-65981
Revision: 0

Template #: 0000-21727-F01 R5

NPM Design Basis Event Identification

- Ch 15 initiating events considered for internal events in a single NPM while at power
  - Ch 15 long term cooling analysis considered effects from up to 12 modules rejecting heat into the shared reactor pool UHS
  - Ch 21 discusses the suitability of shared components and design measures taken to ensure these components do not introduce multi-module risk
- Use of NuScale PRA (DCA Section 19.1) as starting point to identify design basis initiating events
  - PRA initiating events summarize scope of internal events that cause a reactor trip/plant transient response, considering NPM systems and as well as industry references and advanced reactor PRAs
  - Examined systems identified as relevant to PRA initiating events for additional detail to identify specific impacts on module for design basis event identification

NPM Design Basis Event Classification

- **Event categories consistent with operating LWRs:**
  - Increase in heat removal by secondary system
  - Decrease in heat removal by secondary system
  - *Decrease in RCS flow rate (n/a to NPM design)*
  - Reactivity and power distribution anomalies
  - Increase in RCS inventory
  - Decrease in RCS inventory
  - Radioactive release from a subsystem or component

- **NuScale-specific phenomena/event progressions:**
  - PWR stability
  - Over-cooling return to power

NPM Design Basis Event Classification

- **Events categorized as AOOs, infrequent events, accidents**
  - Events that are expected to occur one or more times during an NPM lifetime (1E-2 per year or more) are classified as AOOs.
  - Events that are not expected to occur are classified as IEs or postulated accidents, or may be conservatively classified as AOOs.
  - Event classification is simplified by substituting a deterministic classification considering similarity to operating PWR event classification, where event consequences are small and calculation of a NuScale-specific event frequency is not warranted.

Design Basis Events

15.1 Increase in heat removal by secondary system

Sec.	Event	Classification	System T/H (code)	Subchannel (code)	Radiological
15.1.1	Decrease in feedwater temperature	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a
15.1.2	Increase in feedwater flow	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a
15.1.3	Increase in steam flow	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a
15.1.4	Inadvertent opening of steam generator relief or safety valve	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a
15.1.5	Steam piping failures	Postulated accident	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	Yes
15.1.6	Loss of containment vacuum/containment flooding (1)	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a

(1) NuScale unique event

Design Basis Events

15.2 Decrease in heat removal by secondary system

Sec.	Event	Classification	System T/H (code)	Subchannel (code)	Radiological
15.2.1	Loss of external load	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a
15.2.2	Turbine trip	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a
15.2.3	Loss of condenser vacuum	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a
15.2.4	Closure of main steam isolation valve	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a
15.2.5	Steam pressure regulator failure (closed)	n/a	n/a	n/a	n/a
15.2.6	Loss of non-emergency AC to station auxiliaries	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a
15.2.7	Loss of normal feedwater flow	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a
15.2.8	Feedwater system pipe breaks	Postulated accident	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	Yes
15.2.9	Inadvertent operation of the decay heat removal system (1)	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a

(1) NuScale unique event

Design Basis Events

15.4 Reactivity and Power Distribution Anomalies

Sec.	Event	Classification	System T/H (code)	Subchannel (code)	Radiological
15.4.1	Uncontrolled control rod assembly withdrawal from subcritical or low power	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a
15.4.2	Uncontrollled control rod assembly withdrawal at power	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a
15.4.3	Control rod misoperation	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a
15.4.4	Startup of an inactive loop or recirculation loop at incorrect temperature	n/a	n/a	n/a	n/a
15.4.5	Flow controller malfunction causing an increase in core flow rate (BWR)	n/a	n/a	n/a	n/a
15.4.6	Inadvertent decrease in boron concentration in RCS	AOO	Non-LOCA (n/a)	Yes (VIPRE-01)	n/a
15.4.7	Inadvertent loading and operation of a fuel assembly in improper position	IE	n/a	Yes (VIPRE-01)	n/a
15.4.8	Spectrum of rod ejection accidents	Postulated accident	Rod ejection accidents (SIMULATE-3K, NRELAP5)	Yes (VIPRE-01)	Yes

Design Basis Events

15.5 Increase in reactor coolant inventory

Sec.	Event	Classification	System T/H (code)	Subchannel (code)	Radiological
15.5.1	Chemical and volume control system malfunction	AOO	Non-LOCA (NRELAP5)	Yes (VIPRE-01)	n/a

Design Basis Events

15.6 Decrease in reactor coolant inventory

Sec.	Event	Classification	System T/H (code)	Subchannel (code)	Radiological
15.6.1	Inadvertent opening of reactor safety valve	AOO	Valve opening event (NRELAP5)	n/a	n/a
15.6.2	Failure of small lines carrying primary coolant outside containment	IE	Non-LOCA (NRELAP5)	n/a	Yes
15.6.3	Steam generator tube failure	Postulated accident	Non-LOCA (NRELAP5)	n/a	Yes
15.6.4	Main steam line failure outside containment (BWR)	n/a	n/a	n/a	n/a
15.6.5	Loss of coolant accidents resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary	Postulated accident	LOCA (NRELAP5)	n/a	n/a
15.6.6	Inadvertent operation of emergency core cooling system (1)	AOO	Valve opening event (NRELAP5)	n/a	Yes (2)

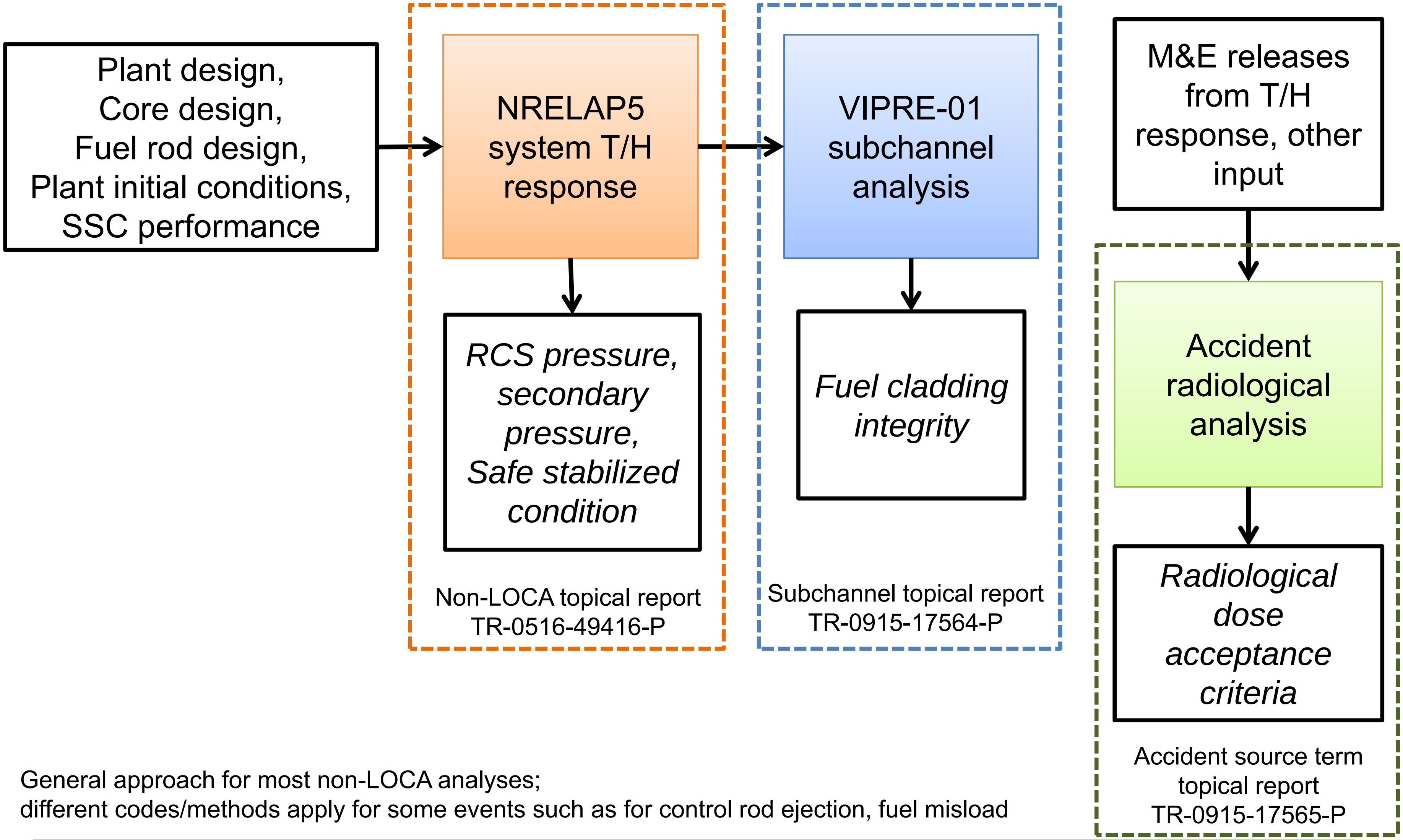
(1) NuScale unique event
(2) See FSAR Ch 12

Event Acceptance Criteria – Thermal/Hydraulic and Fuel

Event Classification (4)	Fuel Clad (1)	RCS Pressure/ Main Steam System Pressure (2)	Containment Pressure (3)	Event Progression
AOO	MCHFR ≥ limit	$\leq 110\% \, P_{design}$	$\leq P_{design}$	Does not develop into a more serious condition without other faults occurring independently
IE	MCHFR ≥ limit	$\leq 120\% \, P_{design}$	$\leq P_{design}$	Does not cause a consequential loss of function of systems needed to cope with the fault
Postulated Accident	MCHFR ≥ limit	$\leq 120\% \, P_{design}$	$\leq P_{design}$	Does not cause a consequential loss of function of systems needed to cope with the fault

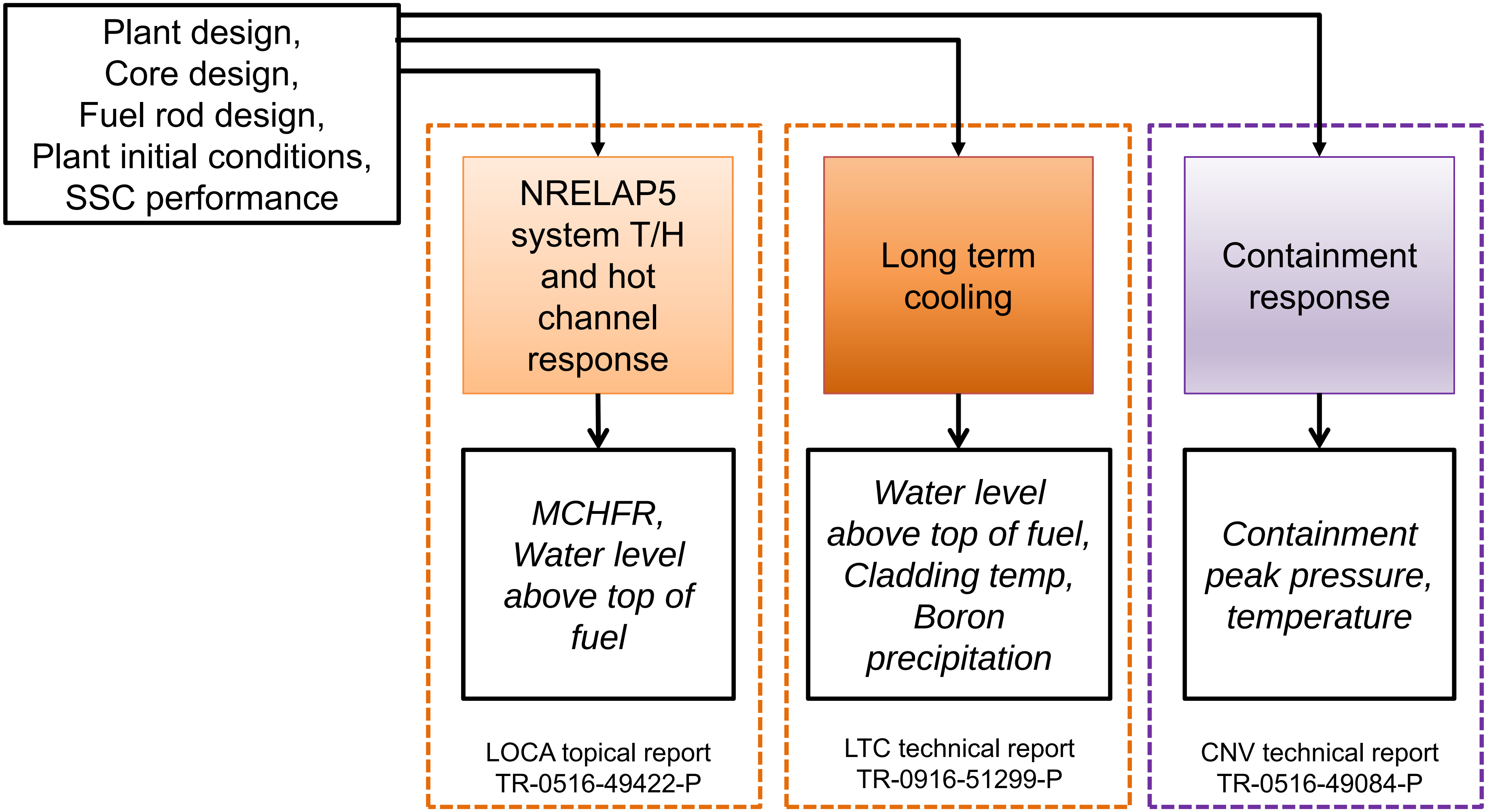
1. NuScale safety analysis methodologies developed to demonstrate fuel cladding integrity maintained. Fuel centerline temperature is examined but not challenged due to low linear heat rates.
2. RCS and secondary side design pressure are equal, 2100 psia
3. Containment design pressure 1050 psia; for peak pressure and temperature analysis see FSAR 6.2.1
4. Event-specific acceptance criteria for LOCA and rod ejection accidents are applied

Evaluation Models – General Non-LOCA Approach



General approach for most non-LOCA analyses;
different codes/methods apply for some events such as for control rod ejection, fuel misload

Evaluation Models –LOCA, Valve Opening, Containment Analysis Approach



General approach for LOCA or valve opening event analyses

Module Protection Functions

- Reactivity Control
  - Reactor trip
  - CVCS/Demineralized Water Isolation
- RCS and Secondary Inventory Control
  - Containment Isolation
  - Secondary Isolation
- Heat Removal
  - DHRS Actuation
  - ECCS Actuation
- Subcooling
  - Reactor trip

Event Mitigation

Increase in heat removal transients

- Reactor trip
- Secondary Isolation

Decrease in heat removal transients

- Reactor trip
- DHRS Actuation

Reactivity and power transients

- Reactor trip
- Demineralized water isolation

Increase in RCS inventory transients

- Reactor trip
- CVCS Isolation

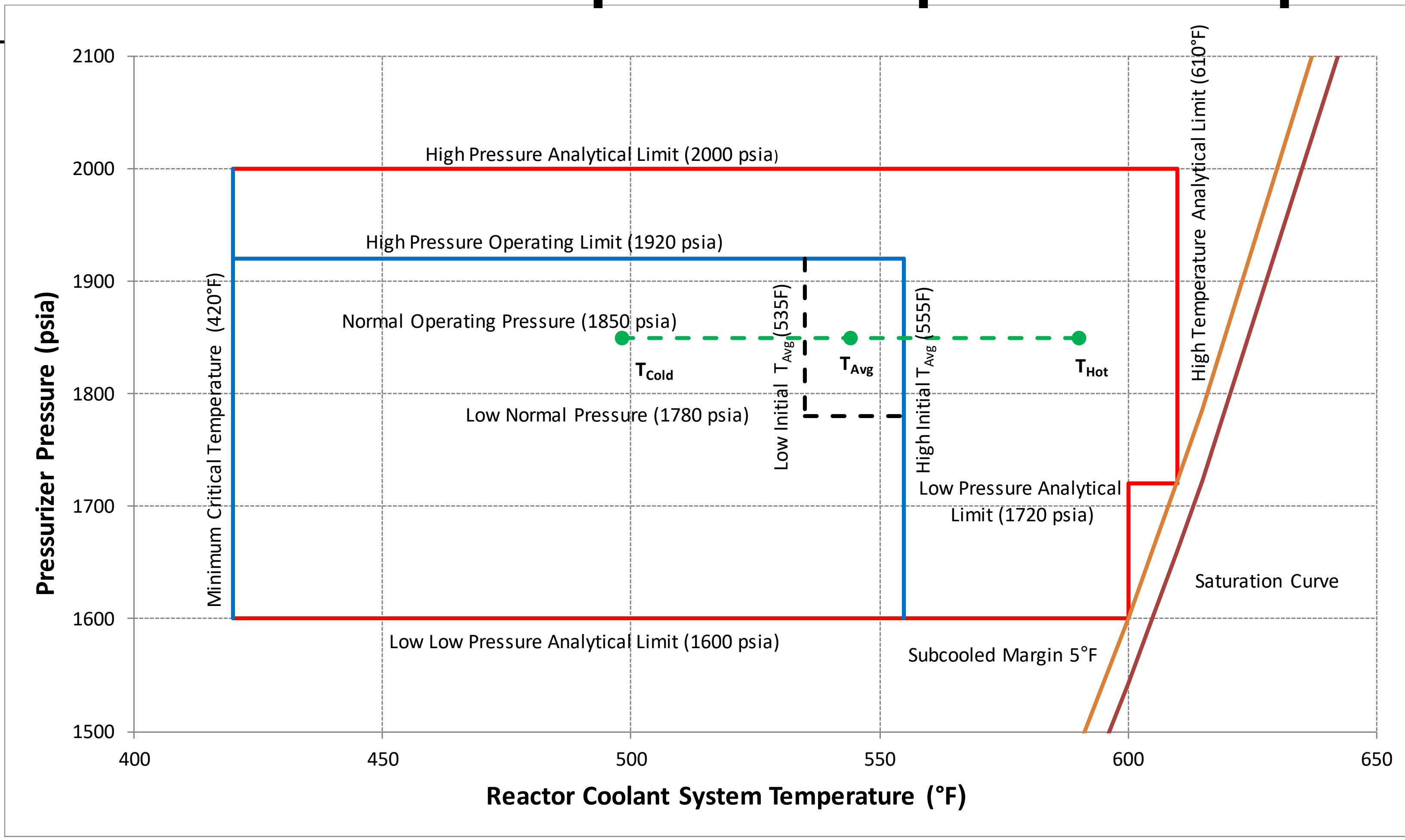
Decrease in RCS inventory transients

- Reactor trip
- CNV Isolation
- ECCS actuation

Stability

- Reactor trip

Pressure vs. Temperature Operation Map



- Module protection system (Ch. 7, red)
- Technical specification LCOs (Ch. 16, blue)

Analytical Assumptions for Ch 15 Analysis

- Operator action
- Single failure
- Loss of power
- Scope of event progression

Operator Actions

- **No operator actions required to achieve safety functions for 72 hours after an initiating event occurs**
  - Operator errors considered in identifying initiating events
  - Operator actions allowed by procedures will make consequences less severe and therefore are bounded by Ch 15 analysis
  - Multiple operator errors or errors that result in common mode failure are beyond design basis

Single Failures

Safety-related system considered (1) (2)	Relevant single failure(s) (3)	Comment
Module protection system	Single failure of instrument channel	Relevant for asymmetric reactivity events
Containment isolation valves: Main steam Feedwater	Failure to close	Nonsafety-related backup MSIV credited in safety analysis Nonsafety-related feedwater regulator valve credited in safety analysis
Feedwater line check valve	Failure to close	Nonsafety-related backup check valve credited in safety analysis
Emergency core cooling system	Failure of one RVV to open Failure of one RRV to open MPS failure to actuate one RVV and one RRV	IAB failing to close upon demand (due to assumed loss of DC power supply) is not treated as a single active failure

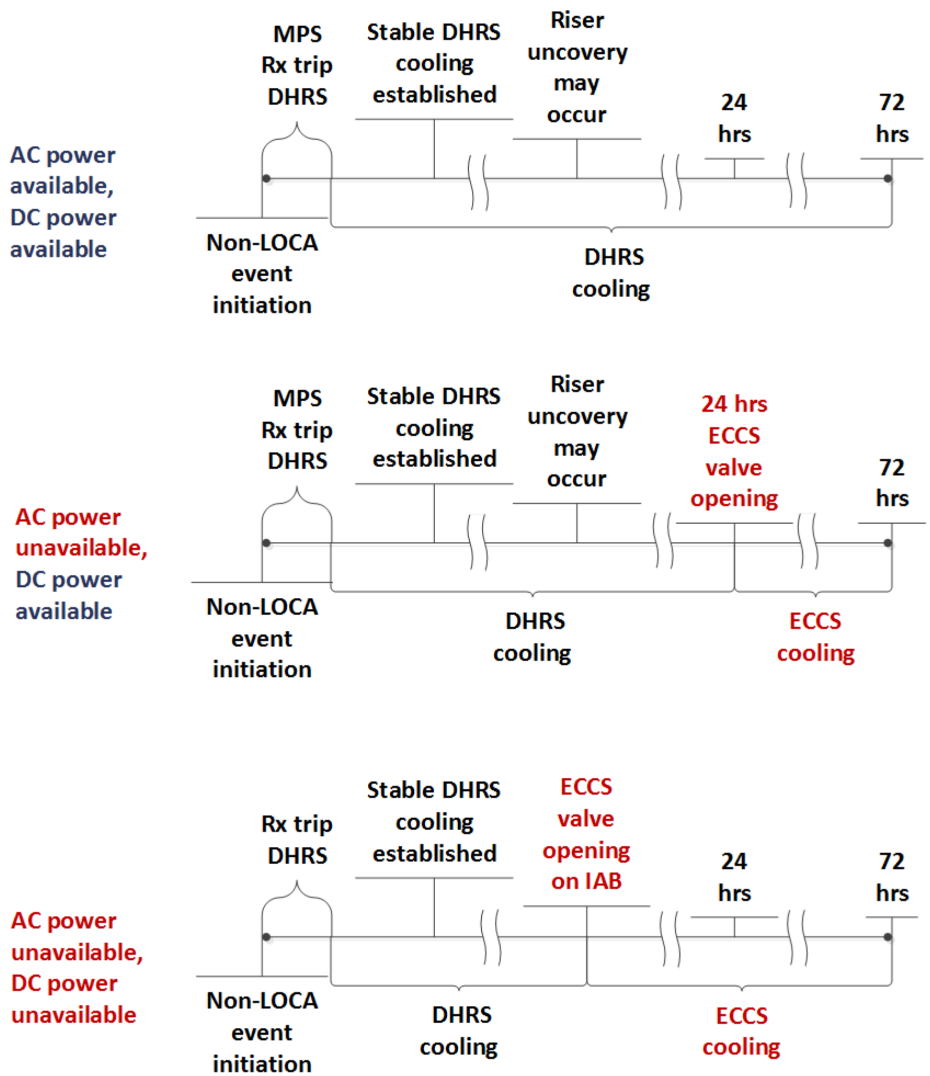
(1) The following systems were considered and no impact from a single failure was identified due to design redundancy: Containment isolation valves on CVCS, RCCWS, CFDS, CES piping; DHRS; reactor safety valves; demineralized water system isolation valves.
(2) The highest-worth control rod failing to insert is assumed in calculating scram worth

Loss of Power

Chapter 15 event analyses consider availability of AC power and DC power

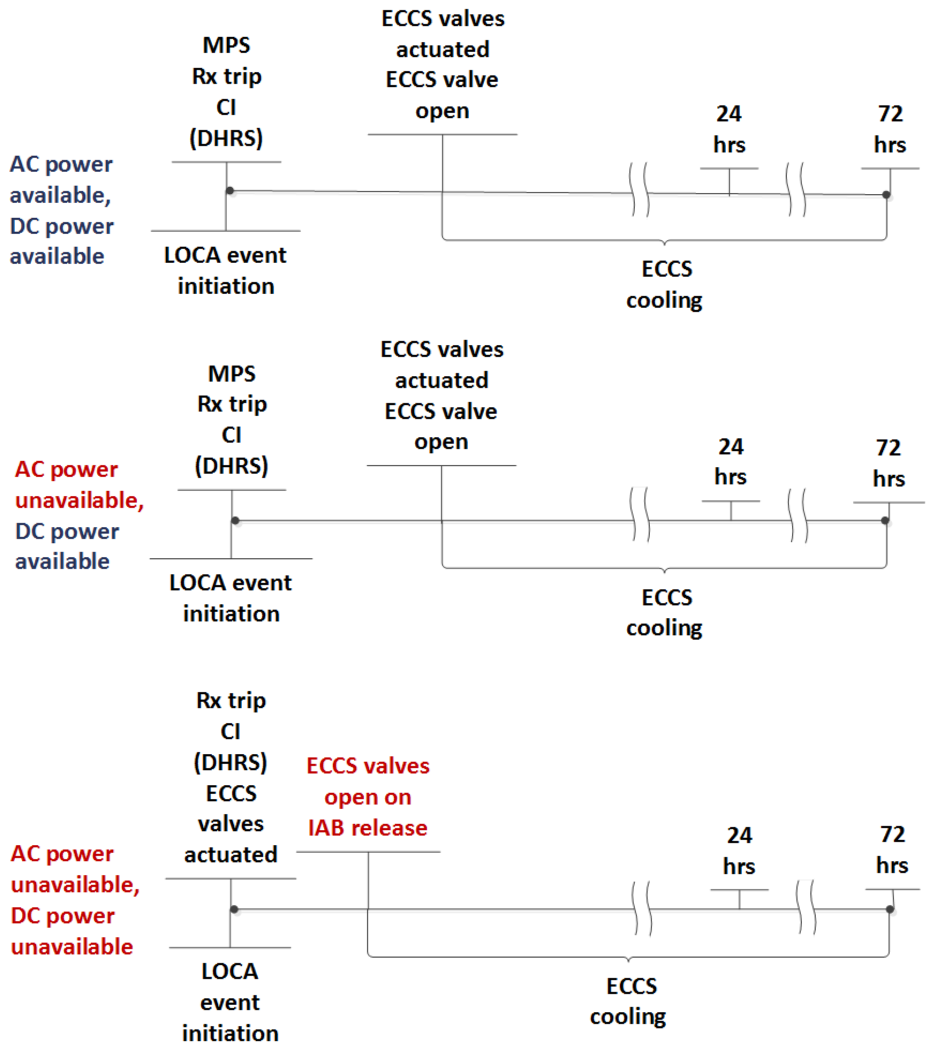
- Loss of AC power at time of event initiation or time of reactor trip
- If AC power is lost, highly reliable DC power system may be assumed available or unavailable
- For non-LOCA type events, power availability affects whether ECCS valves actuate and what time they open
- For LOCA-type events, power availability affects the time ECCS valves actuate and when they open

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

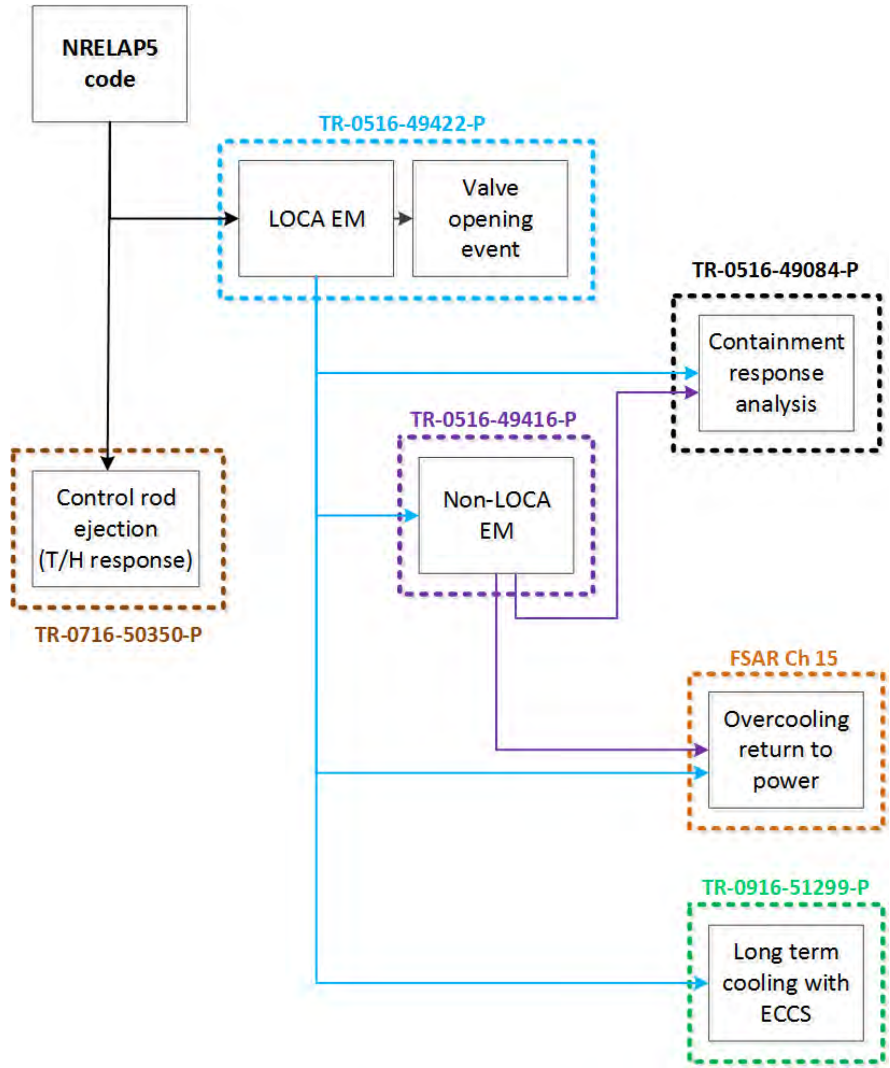
Design Basis Event Progression

15.0 Event Progressions

- Safety analyses of design basis events are performed from event initiation until a safe, stabilized condition is reached:
  - Initiating event is mitigated
  - Acceptance criteria are met
  - System parameters such as inventory levels, temperatures, and pressures are trending in the favorable direction
- After safe, stabilized condition is reached:
  - ECCS long term decay and residual heat removal
  - Overcooling return to power
  - Extended DHRS operation

System T/H Analysis Methodologies

- NRELAP5 code developed from RELAP5-3D
  - Modified to address NuScale-specific phenomena/systems
- LOCA EM developed following RG 1.203 EMDAP
- LOCA EM extended to derived EMs for other events by addressing unique aspects



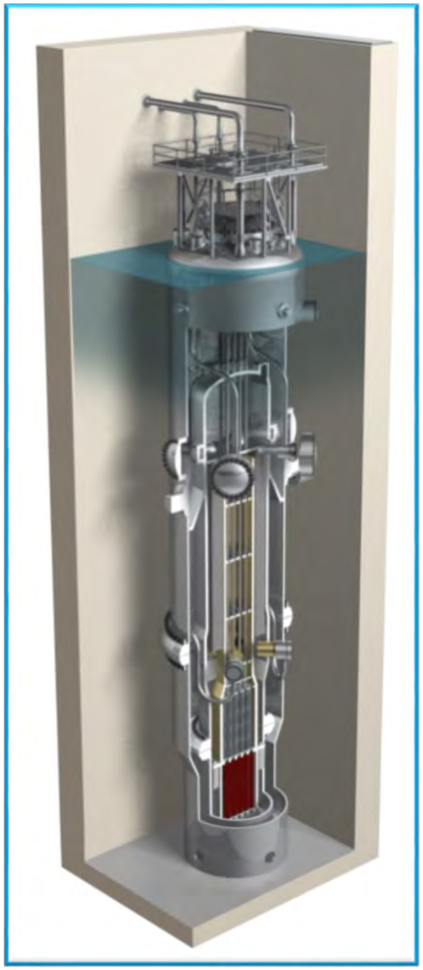
LOCA EM Development: TR-0516-49422-P

- Followed RG 1.203 EMDAP process
- PIRT developed to identify high ranking phenomena for LOCA and valve-opening event short-term response
- Assessment basis developed that includes SETs and IETs to address high-ranked phenomena
  - Unique phenomena addressed by NuScale-specific tests
- NRELAP5 code developed from RELAP5-3D to address NuScale-specific unique phenomena/systems
- Applicability evaluation performed including bottom-up and top-down analysis

Non-LOCA EM Development: TR-0716-49416-P

- Non-LOCA evaluation model developed to perform conservative analyses, following intent of the RG 1.203 EMDAP and applying a graded approach
- PIRT developed to identify high ranking phenomena considering different types of non-LOCA events
- Gap analysis performed to evaluate how high ranked phenomena are addressed
- Additional NRELAP5 code validation performed focused on DHRS and integral non-LOCA response

ACRS Subcommittee Presentation



Chapter 15.0
Transient Examples

June 19, 2019

PM-0619-65981
Revision: 0

Template #: 0000-21727-F01 R5

Analysis Results

15.1 Increase in heat removal by secondary system

Sec.	Event (1) (Acceptance criteria)	Peak RCS Pressure (< 110% P_{design}: 2310 psia) (< 120% P_{design}: 2520 psia)	Peak SG Pressure (< 110% P_{design}: 2310 psia) (< 120% P_{design}: 2520 psia)	MCHFR (> limit: 1.284)
15.1.1	Decrease in feedwater temperature	1959	1432	1.921
15.1.2	Increase in feedwater flow	1936	1424	1.944
15.1.3	Increase in steam flow	2018	1208	1.957
15.1.4	Inadvertent opening of steam generator relief or safety valve	2156	1346	1.861
15.1.5	Steam piping failures	1992	1342	2.761
15.1.6	Loss of containment vacuum/containment flooding (1)	1936	1424	1.944

(1) NuScale unique event

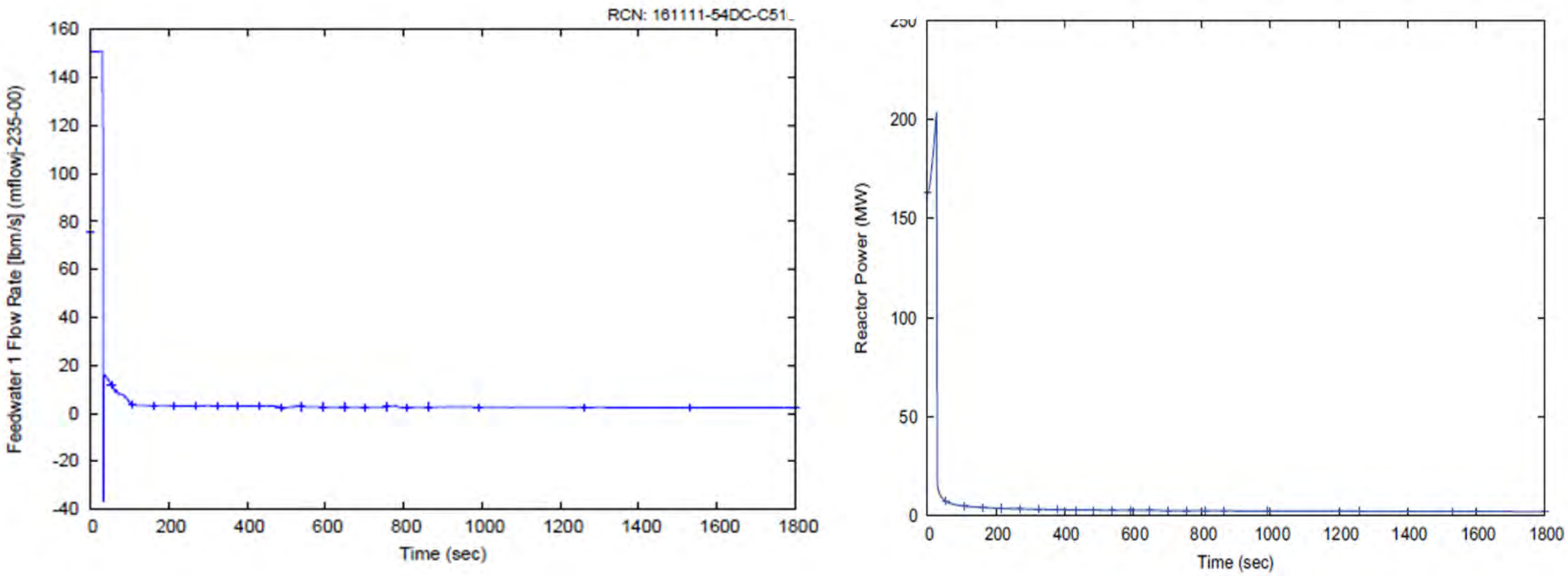
Significant margin to acceptance criteria for all events

Example: Increase in Feedwater Flow (IFF)

- Sequence of events

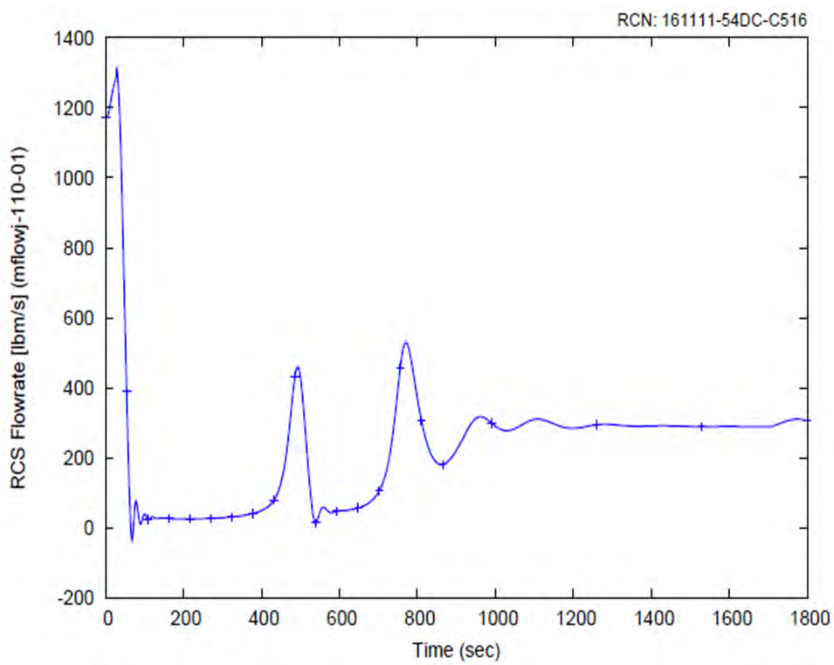
Event	Time [s]
Feedwater flow begins to increase	0
Regulating bank begins to withdraw in response to a decrease in average RCS temperature	~4
Low steam superheat limit is reached	24
High reactor power limit is reached	25
Reactor trips on high core power signal	27
Peak reactor power	28
DHRS actuation	32
Peak MSS pressure	84

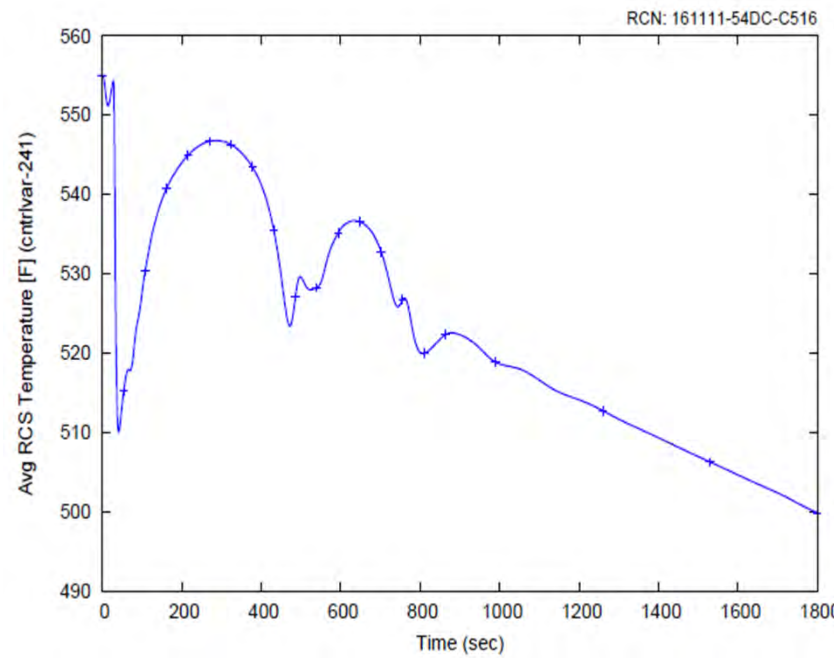
IFF: Feedwater Flow/Reactor Power



- 100% increase in feedwater flow
- Event detected and steam generators isolated before both steam generators overfill, to preserve adequate DHRS capacity

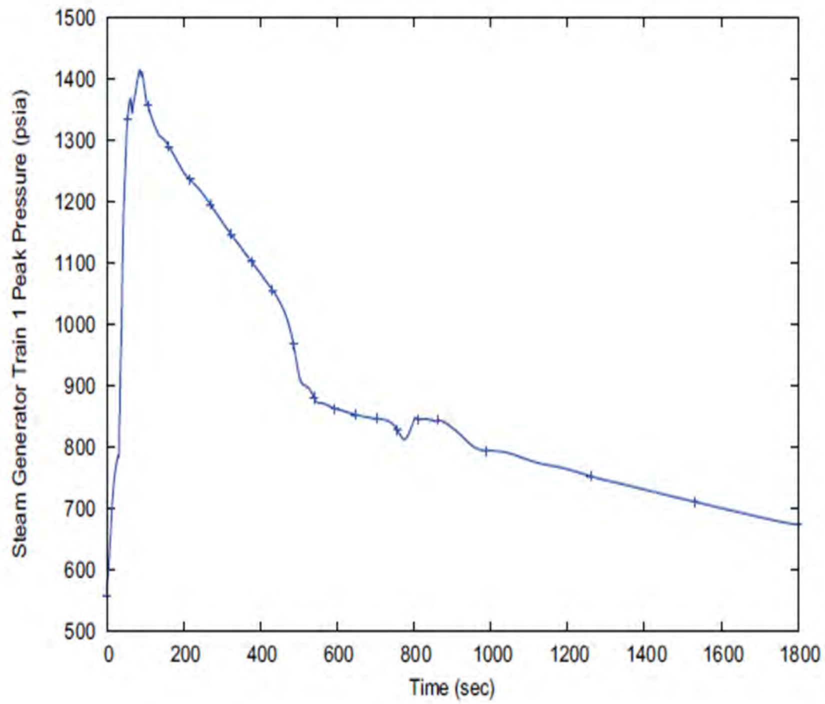
IFF: RCS Flowrate and Ave Temp

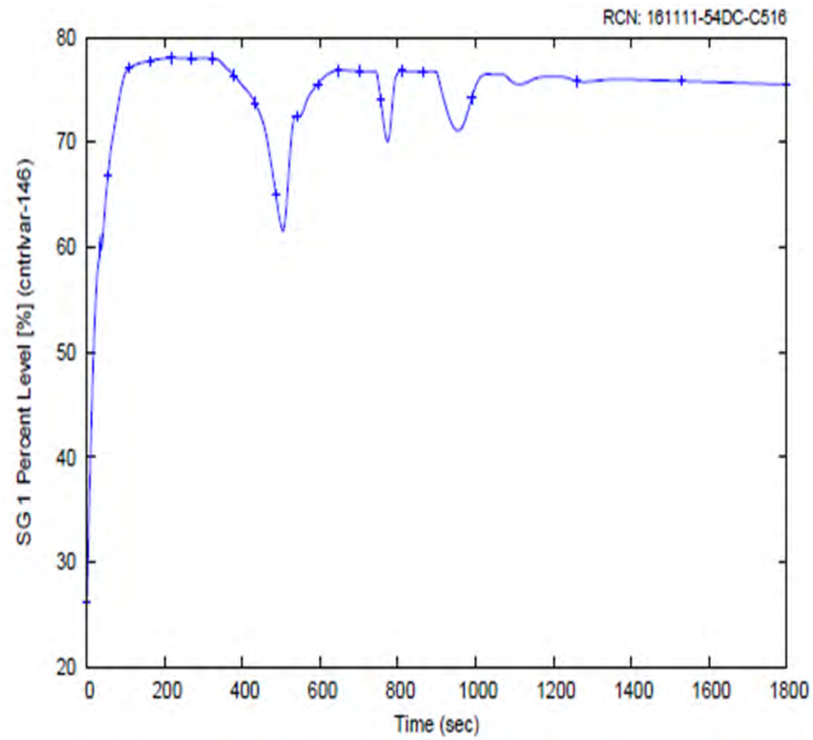




- RCS flow and average temperature response characteristic of NPM after reactor trip with DHRS actuation

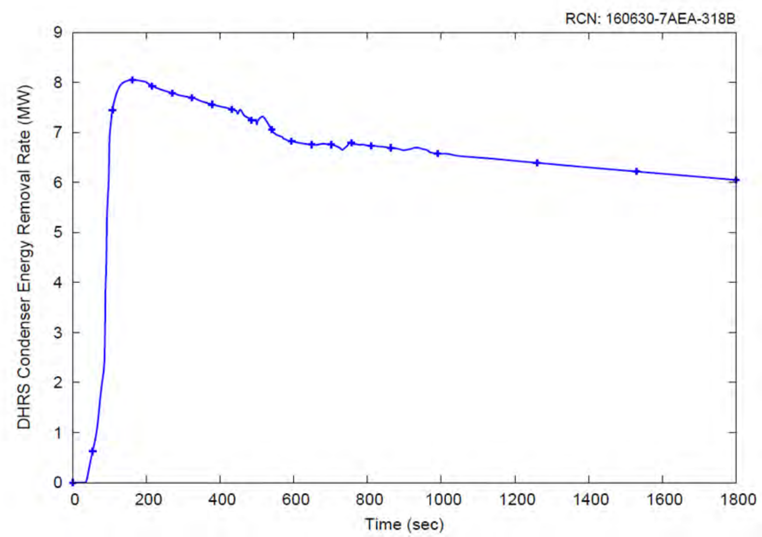
IFF: SG Pressure and Level

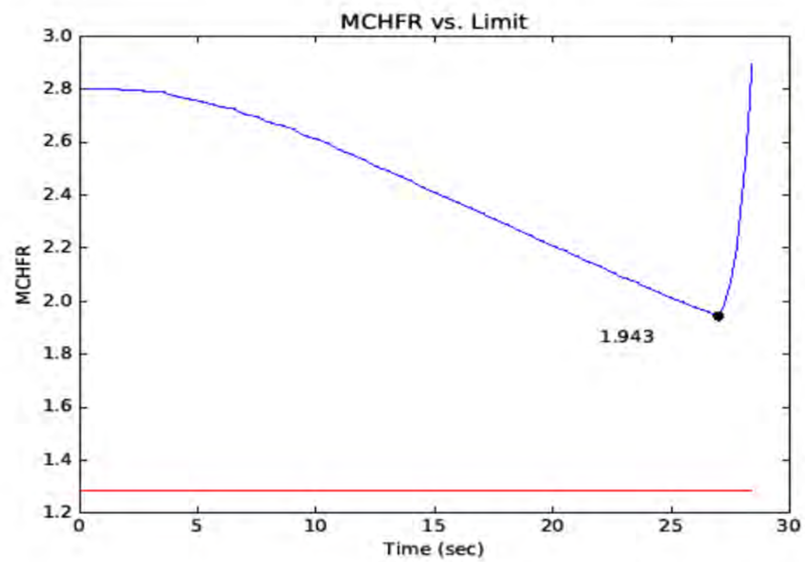




- Peak secondary side pressure occurs after DHRS actuation
- Event detected and steam generators isolated before both steam generators overfill, to preserve adequate DHRS

IFF: DHRS Heat Removal and MCHFR





- DHRS removes decay and residual energy early in transient response
- MCHFR margin increases after reactor trip

Analysis Results

15.2 Decrease in heat removal by secondary system

Sec.	Event [1] (Acceptance criteria)	Peak RCS Pressure (< 110% P_{design}: 2310 psia) (< 120% P_{design}: 2520 psia)	Peak SG Pressure (< 110% P_{design}: 2310 psia) (< 120% P_{design}: 2520 psia)	MCHFR (> limit: 1.284)
15.2.1	Loss of external load	2158	1474	2.579
15.2.2	Turbine trip	2158	1474	2.579
15.2.3	Loss of condenser vacuum	2158	1474	2.579
15.2.4	Closure of main steam isolation valve	2160	1481	2.567
15.2.6	Loss of non-emergency AC to station auxiliaries	2162	1361	2.569
15.2.7	Loss of normal feedwater flow	**2165**	1434	2.569
15.2.8	Feedwater system pipe breaks	**2164**	1328	2.607
15.2.9	Inadvertent operation of the decay heat removal system [1]	2163	**1582**	2.489

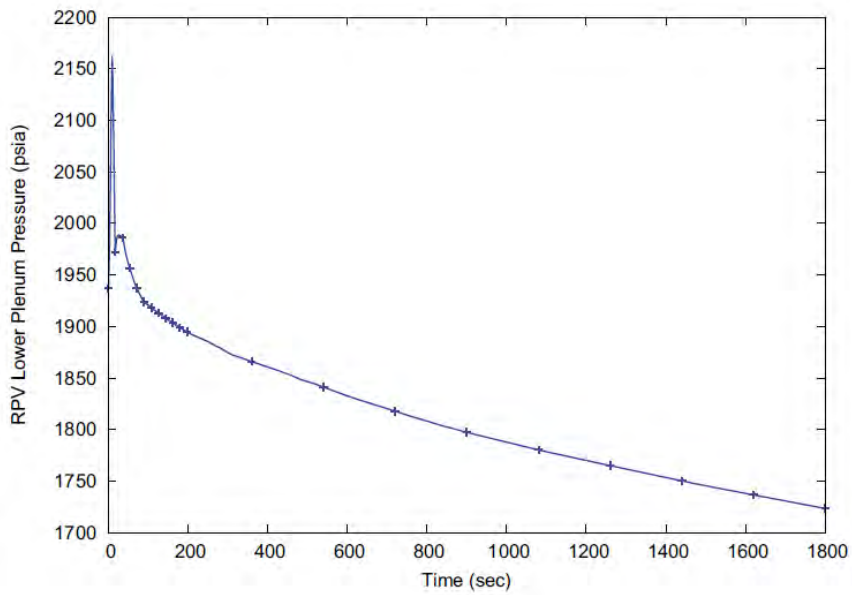
(1) NuScale unique event

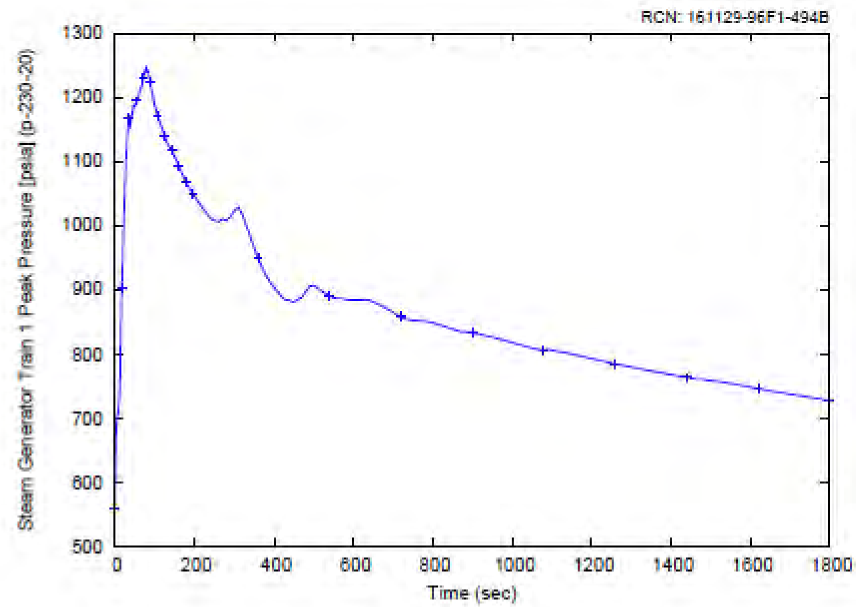
Significant margin to acceptance criteria for all events

Example: Loss of AC Power

Event	Time [s]
Loss of AC power occurs	0
Turbine trip occurs	0
Feedwater pump trips.	0
CVCS pump trips (approximated as CVCS isolation)	0
High pressurizer pressure is reached.	5
RTS actuation on high pressurizer pressure signal.	7
DHRS actuation on the high pressurizer pressure signal. DHRS actuation valves begin to open.	7
FWIVs and MSIVs begin to close.	9
RSV1 opens	10
Peak RPV pressure is reached.	10
MSIVs are fully closed.	14
FWIVs are fully closed.	14
DHRS actuation valves are fully open.	37
Peak steam generator pressure is reached.	79

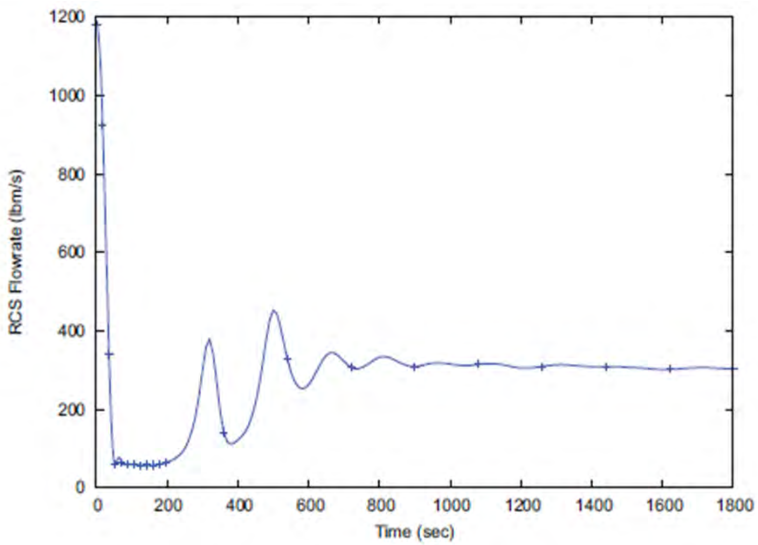
LOAC: RCS Pressure – SG Pressure

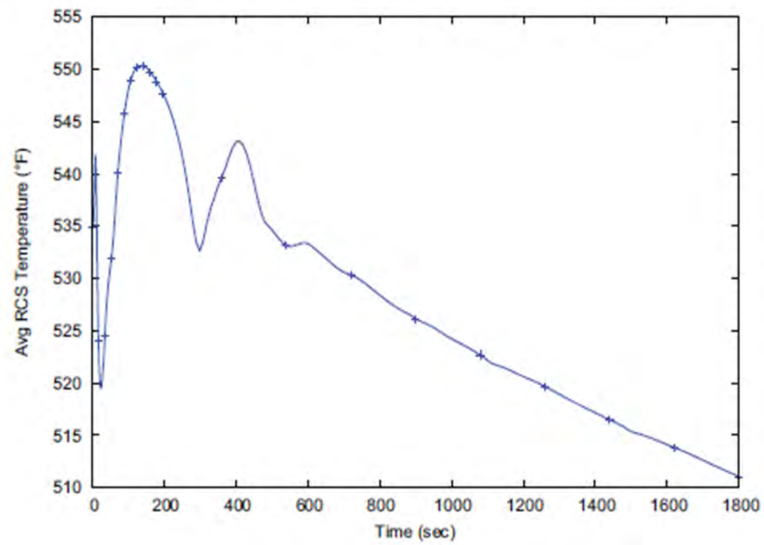




- RCS pressure decreases quickly after 1st reactor safety valve lift; Peak secondary side pressure occurs after DHRS actuation
- Maximum pressures remain well below acceptance criteria 2310 psia

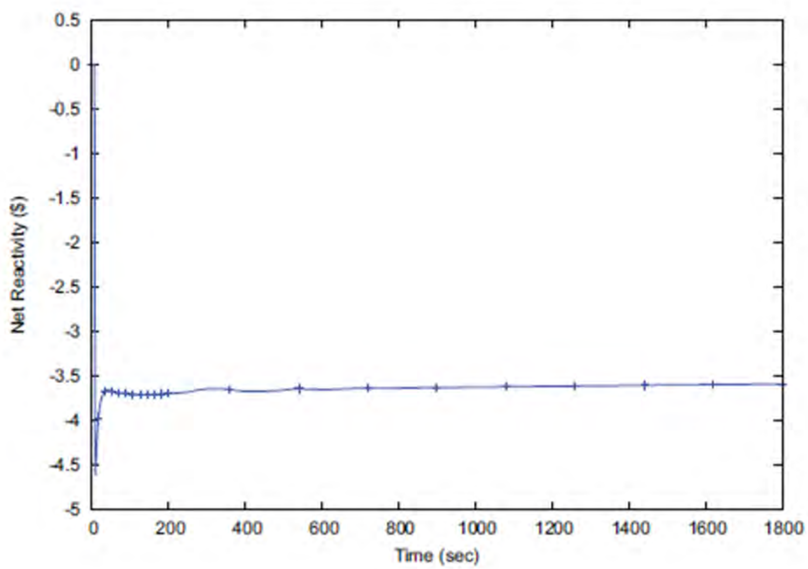
LOAC: RCS Flow – RCS Temperature

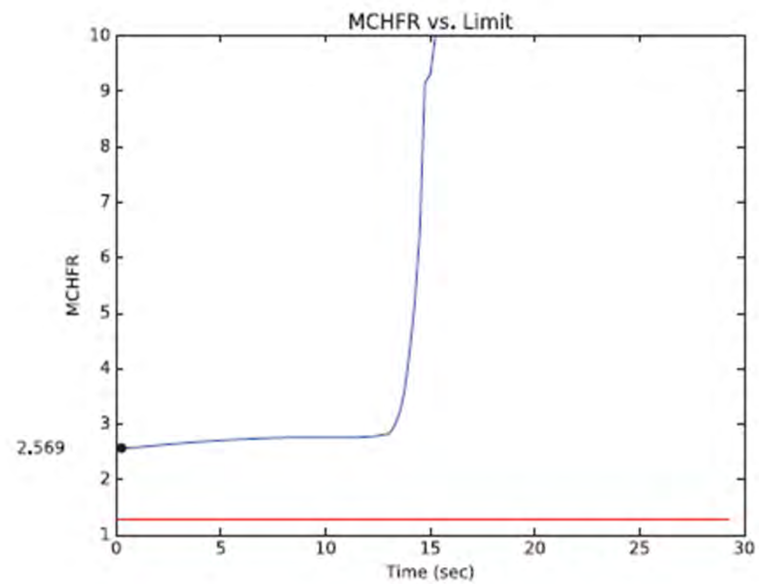




- RCS flow and average temperature response characteristic of NPM after reactor trip with DHRS actuation

LOAC: Net Reactivity - MCHFR





Results for MCHFR case

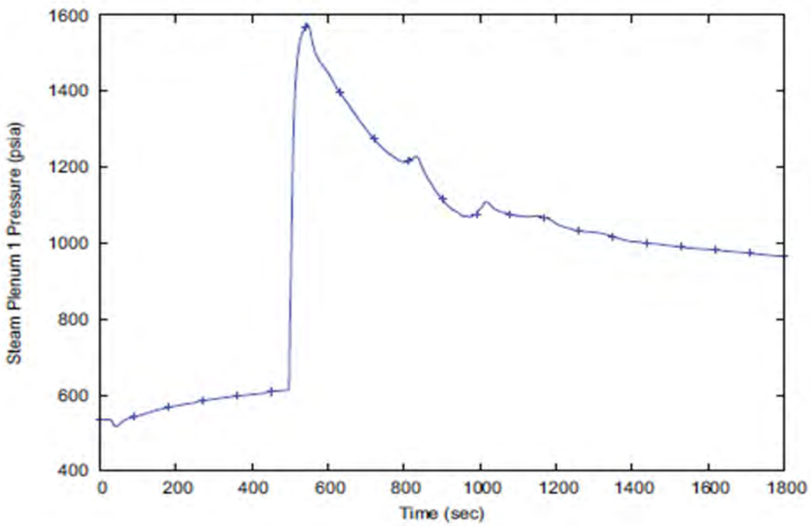
- Heatup event does not challenge MCHFR acceptance criteria

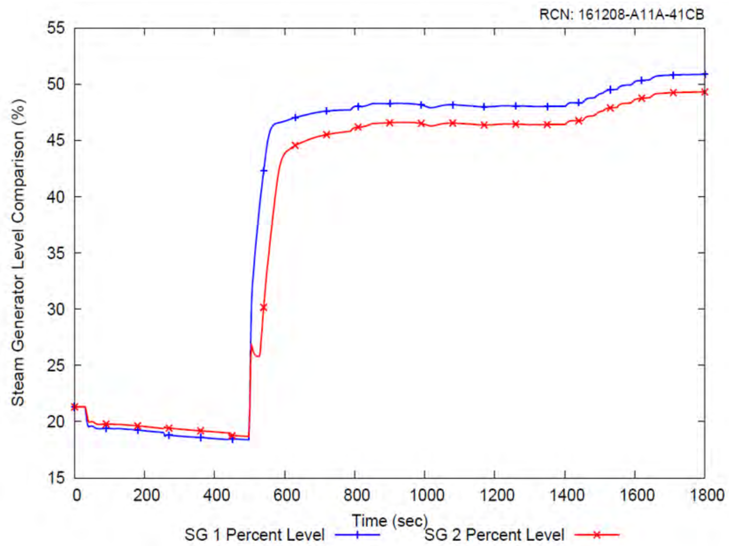
Example: Inadvertent Operation DHRS (IODHRS)

- Sequence of Events for spurious opening of 1 valve
  - Peak SG pressure case

Event	Time (s)
Transient initiation (spurious DHRS#1 valve actuation)	0.0
DHRS#1 actuation valve begins opening (Assumption)	29.7
DHRS#1 actuation valve fully open (Assumption)	30.0
High Hot Leg Temperature signal (610°F)	491.3
DHRS actuation (secondary isolation + DHRS #2)	499.4
RTS actuation	499.4
Peak SG Pressure (train #1)	545.0

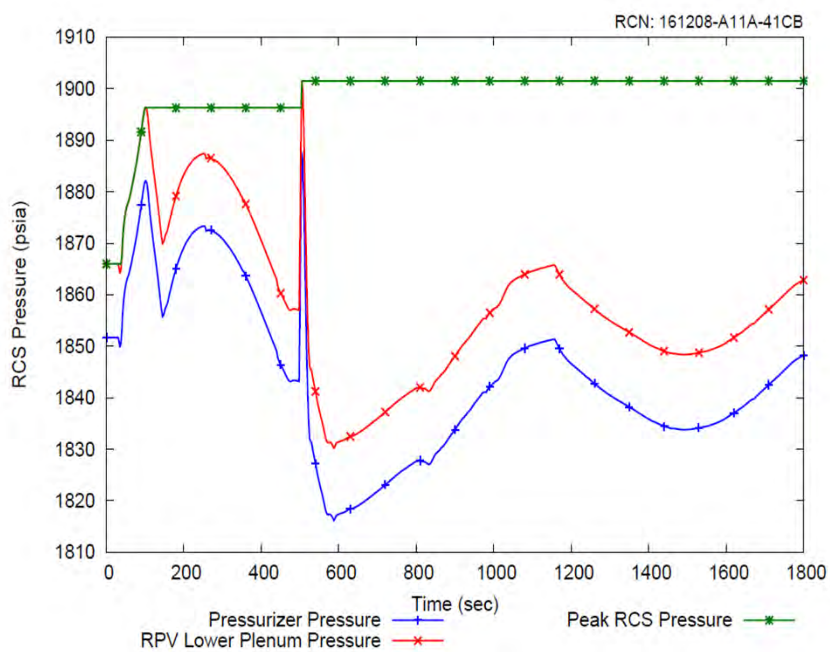
IODHRS: SG Pressure – SG Level

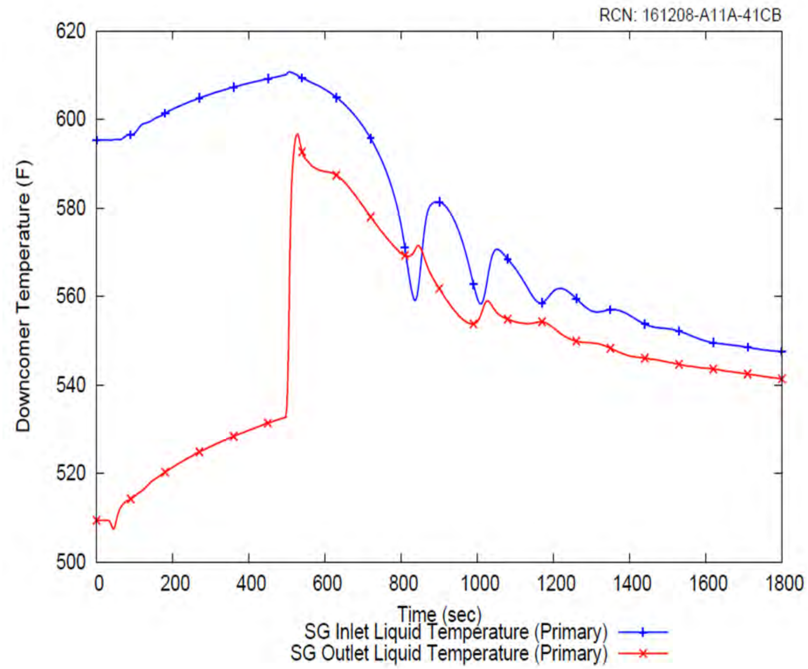




- DHRS valve opening at power diverts some feedwater flow from steam generator, resulting in a slow overheating event
- Maximum steam generator pressure well below acceptance criteria 2310 psia

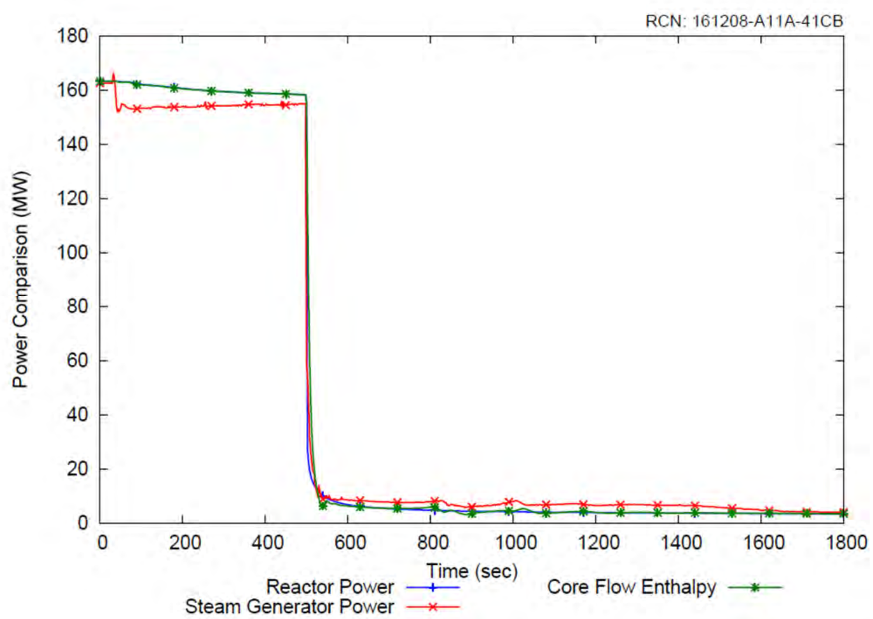
IODHRS: RCS Pressure – Temperatures

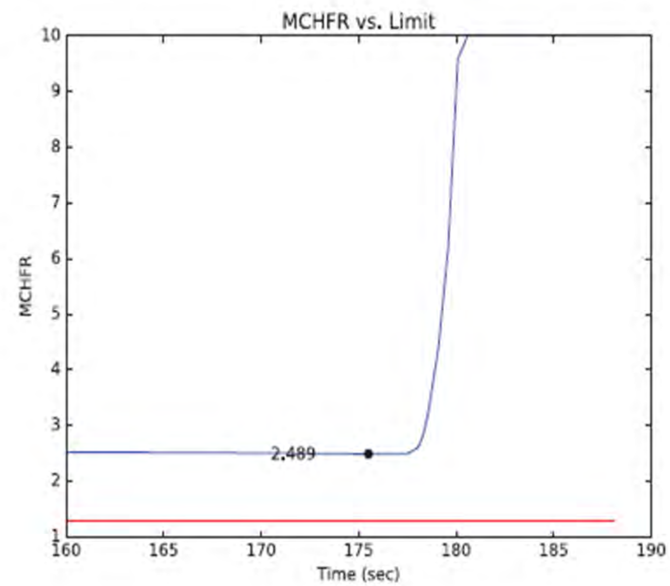




- DHRS valve opening at power diverts some feedwater flow from steam generator, resulting in a slow overheating event
- Maximum RCS pressure well below acceptance criteria 2310 psia

IODHRS: Power - MCHFR





- Heatup event does not challenge MCHFR acceptance criteria

Analysis Results

15.4 Reactivity and Power Distribution Anomalies

Sec.	Event (1) (Acceptance criteria)	Peak RCS Press. (< 110% P_{design}: 2310 psia) (< 120% P_{design}: 2520 psia)	Peak SG Press. (< 110% P_{design}: 2310 psia) (< 120% P_{design}: 2520 psia)	MCHFR (> limit: 1.284)	Fuel centerline (< Tmelt)	LHR (< 21.22 kW/ft)
15.4.1	Uncontrolled control rod assembly withdrawal from subcritical or low power	2038	685	>10	890.8 F	NA
15.4.2	Uncontrollled control rod assembly withdrawal at power	2160	1326	1.624	NA	8.97 kW/ft
15.4.3	Control rod misalignment	NA	NA	2.509	NA	7.10 kW/ft
15.4.3	Control rod withdrawal	NA	NA	**1.624**	NA	7.84 kW/ft
15.4.3	Control rod drop	NA	NA	1.641	NA	8.42 kW/ft
15.4.6	Inadvertent decrease in boron concentration in RCS	NA	NA	NA	NA	NA
15.4.7	Inadvertent loading and operation of a fuel assembly in improper position	NA	NA	1.916	NA	7.87 kW/ft
15.4.8	Spectrum of rod ejection accidents	2076	NA	2.477	2162 F	NA

Control rod withdrawal has limiting MCHFR for reactivity events

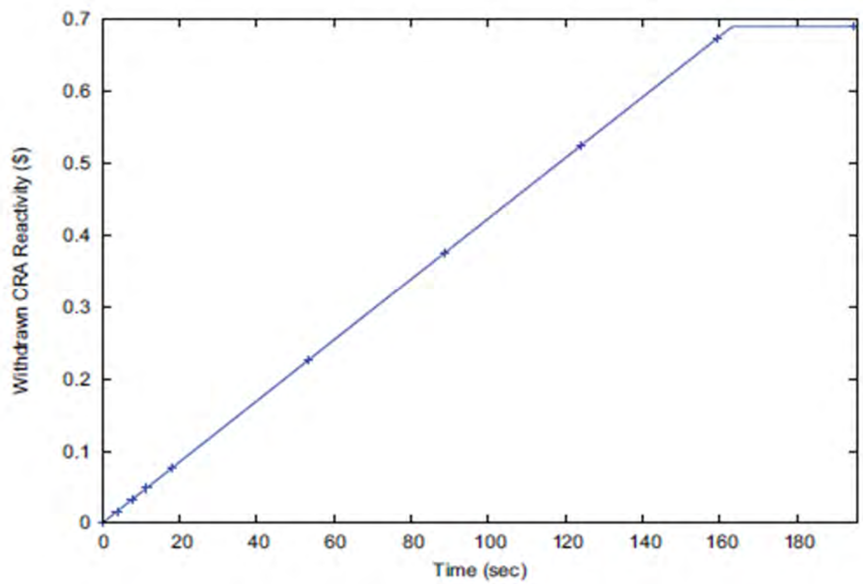
NUSCALE Power for all humankind
Template #: 0000-21727-F01 R5

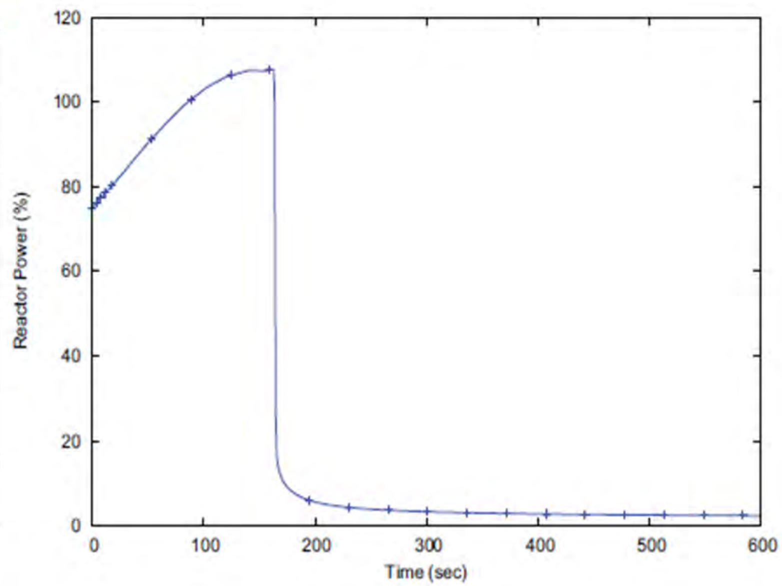
Example: Single Control Rod withdrawal (CRW)

- Sequence of Events

Event	Time [s]
Single CRA begins to withdraw	0
High hot leg temperature limit reached	155
High RCS pressure limit reached	162
Reactor trip actuation	163
Maximum RCS pressure occurs	170
DHRS valves fully open	193

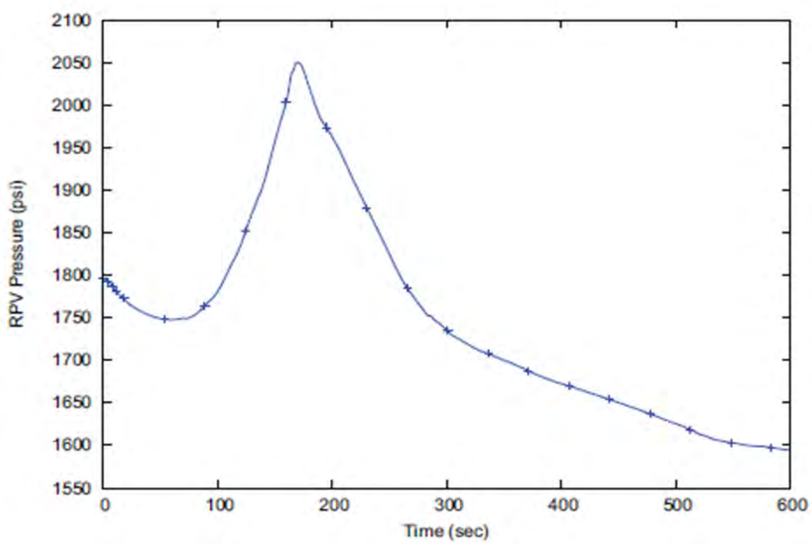
CRW: Withdrawn Reactivity and Power

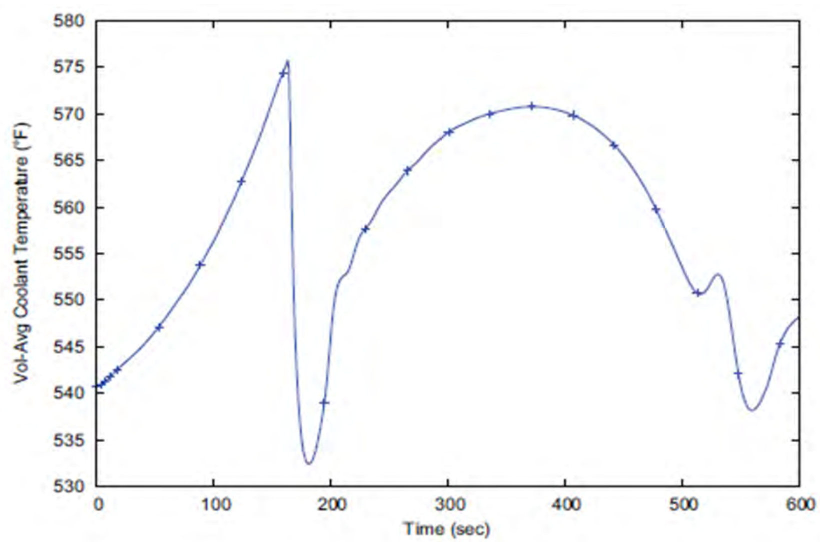




- Reactivity insertion results in corresponding increase in power

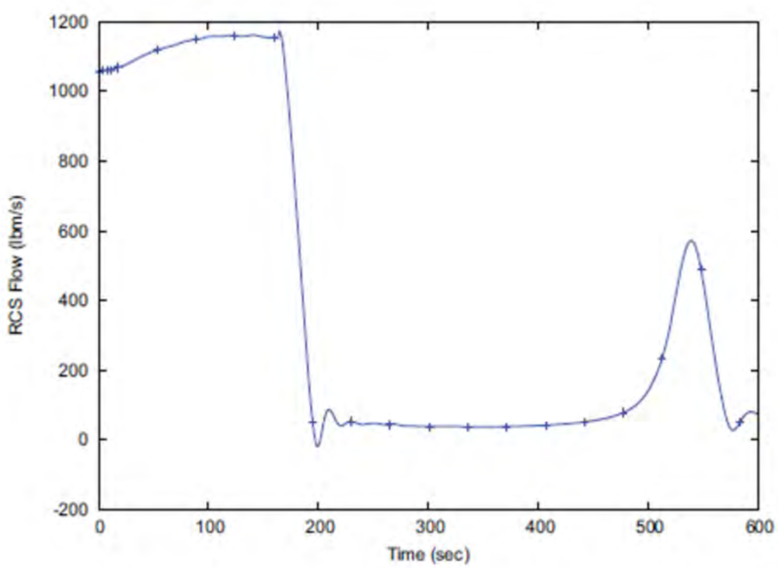
CRW: RCS Pressure and Temp

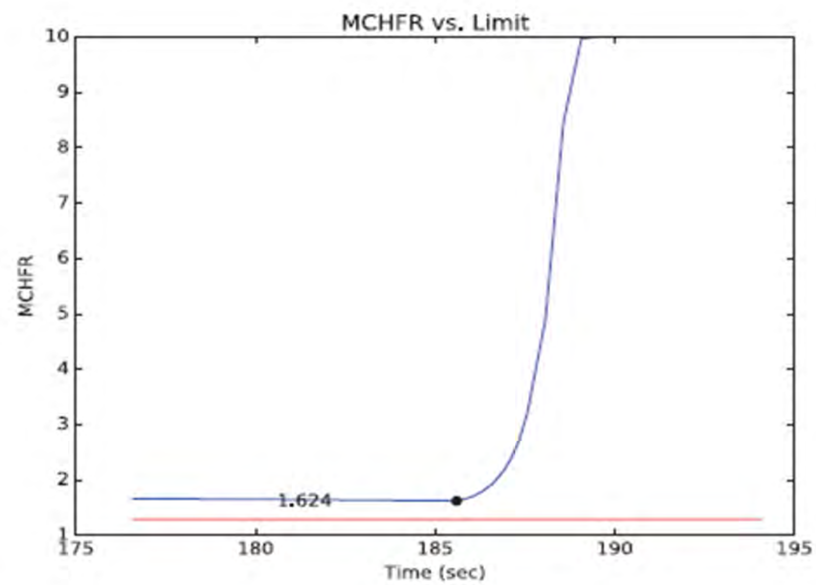




- RCS pressure and fluid temperatures increase as power increases
- Limiting MCHFR case occurs for case with high power, high temperature, high pressure conditions

CRW: RCS Flow and MCHFR





- RCS flow increases as power increases
- MCHFR margin increases after reactor trip

Analysis Results

15.5 Increase in reactor coolant inventory

Sec.	Event [(1)] (Acceptance criteria)	Peak RCS Pressure (≤ 110% P_{design}: 2310 psia)	Peak SG Pressure (≤ 110% P_{design}: 2310 psia)	MCHFR (≥ limit: 1.284)
15.5.1	Chemical and volume control system malfunction	2130	1418	2.379

Significant margin to acceptance criteria

Analysis Results

15.6 Decrease in reactor coolant inventory

Sec.	Event (1) (Acceptance criteria)	Peak RCS Pressure (< 110% P_{design}: 2310 psia) (< 120% P_{design}: 2520 psia)	Peak SG Pressure (< 110% P_{design}: 2310 psia) (< 120% P_{design}: 2520 psia)	MCHFR	Additional
15.6.1	Inadvertent opening of reactor safety valve	NA	NA	NA	NA
15.6.2	Failure of small lines carrying primary coolant outside containment	2047	1368	NA	Note 2
15.6.3	Steam generator tube failure	2073	**1806**	NA	Note 2
15.6.5	Loss of coolant accidents resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary	NA	NA	Result: 1.796 Acceptance criteria: > 1.29	Minimum level above top of core: 1.5 ft
15.6.6	Inadvertent operation of emergency core cooling system (1)	1936	588	**Result: 1.41** **Acceptance criteria:** > 1.13	

(1) NuScale unique event
(2) Mass release and iodine spiking time provided as input to radiological analyses

SG tube failure maximum secondary pressure remains below design pressure
Valve opening and LOCA events demonstrate margin to acceptance criteria

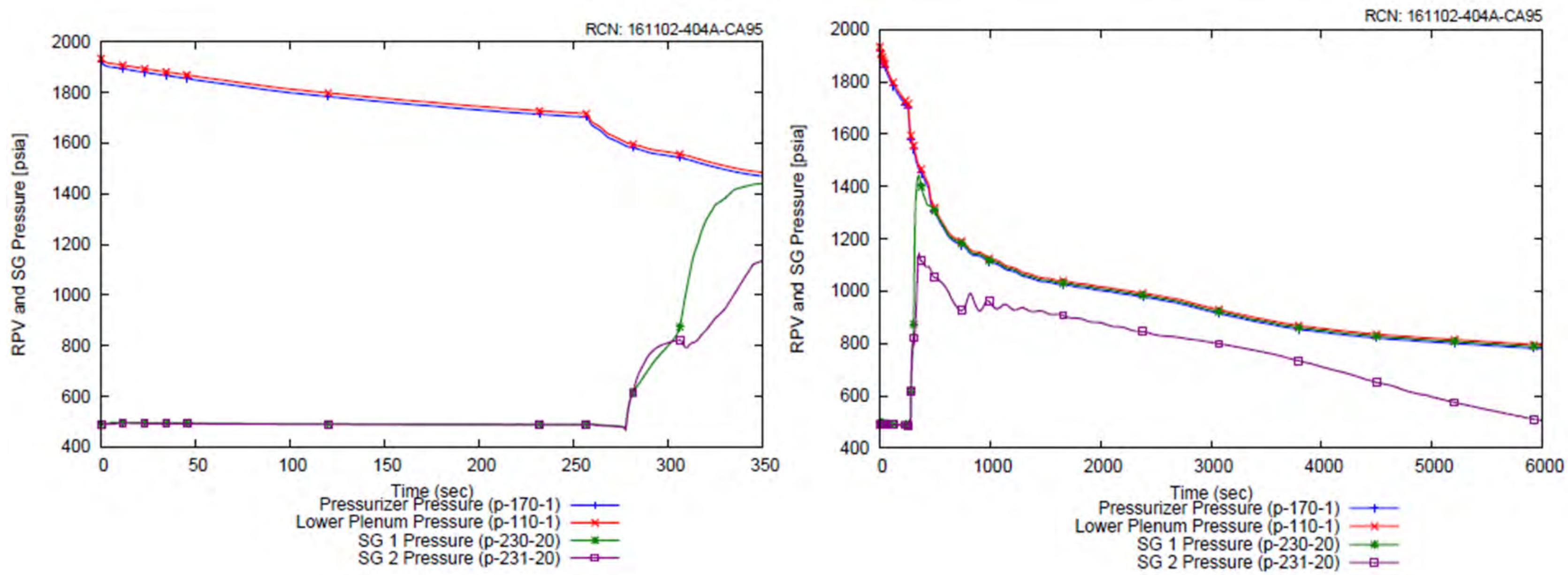
Example: SG Tube Failure (SGTF)

- Sequence of Events for limiting mass release case

Event	Time (s)[1]
SGTF at top of SG	0
Maximum RPV pressure occurs	1
Low Pressurizer Level (35%)	254
Pressurizer Heater Trip	255
Reactor trip	256
Low Pressurizer Pressure (1600 psia)	275
DHRS Actuation	277
MSIV Closure	277
Turbine Stop Valve Closure	278
MSIVs fully closed (Intact SG isolated)	283
Low Low Pressurizer Level (20%)	288
High Steam Line Pressure (800 psia) on intact SG	297
High Steam Line Pressure (800 psia) on faulted SG	301
Secondary MSIVs fully closed (Faulted SG isolated)	307
Maximum intact SG pressure reached	356
Maximum faulted SG pressure reached	356
Pressurizer Level reaches minimum value	860

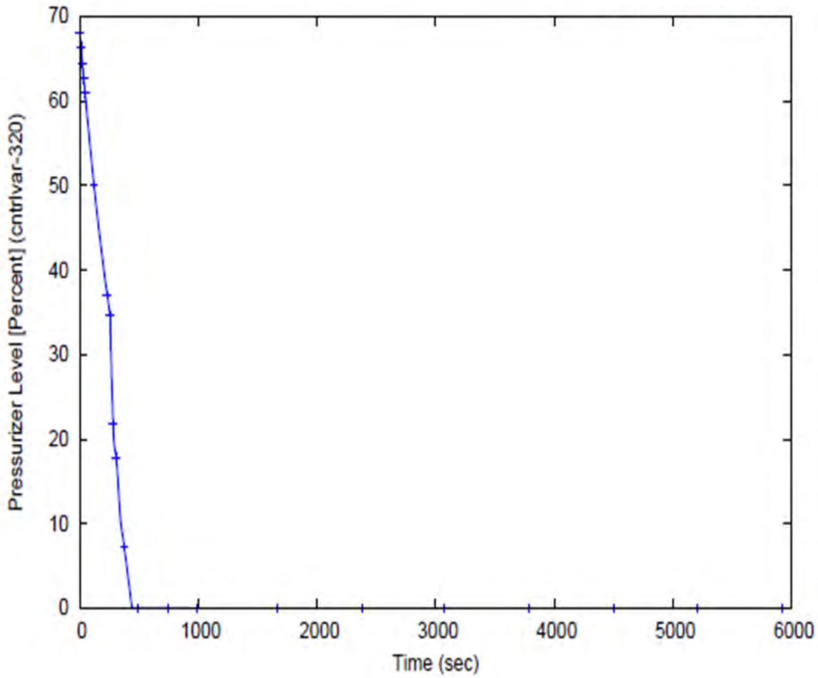
Notes: (1) Time rounded up to second.

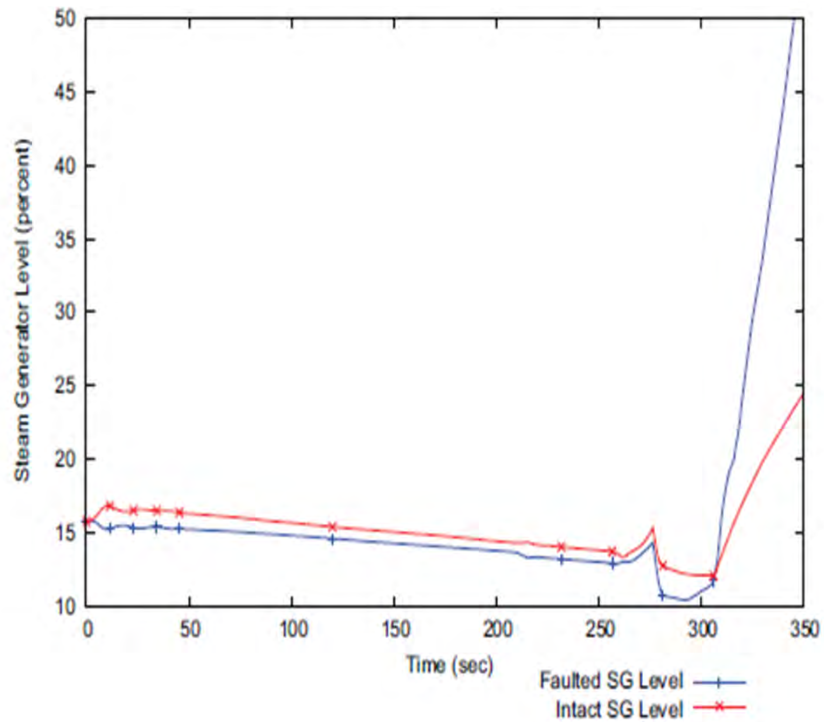
SGTF: Reactor Vessel and SG Pressure



- SG pressure increases after secondary side isolation
- DHRS on intact steam generator provides heat removal

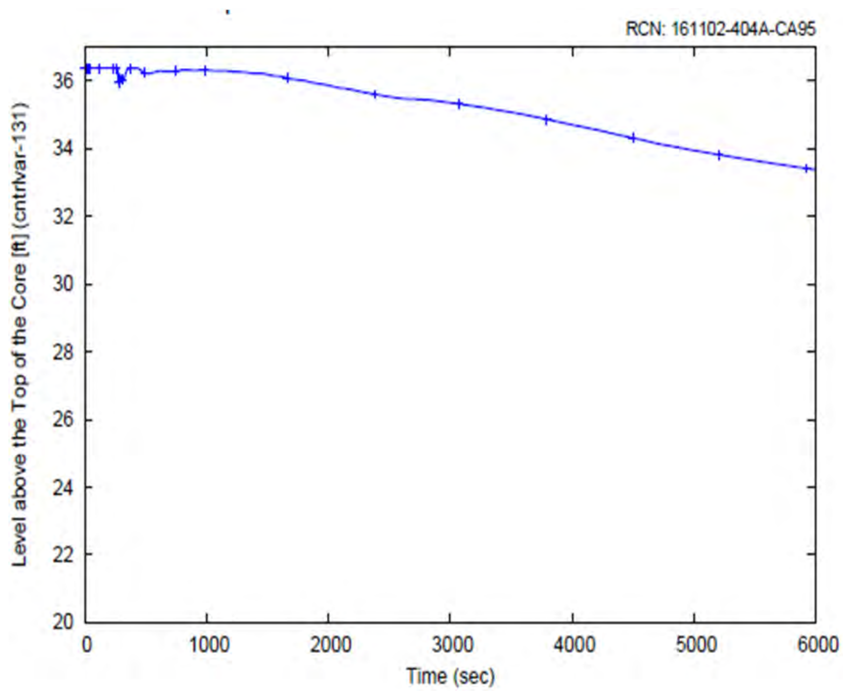
SGTF: Pressurizer and SG Level

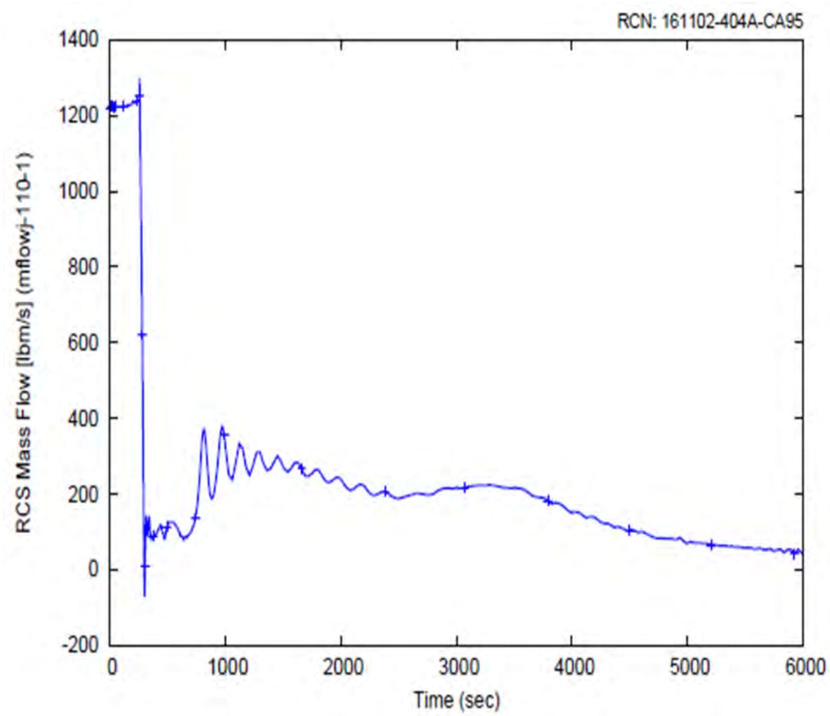




- Primary side inventory lost to environment through tube failure until secondary side isolation

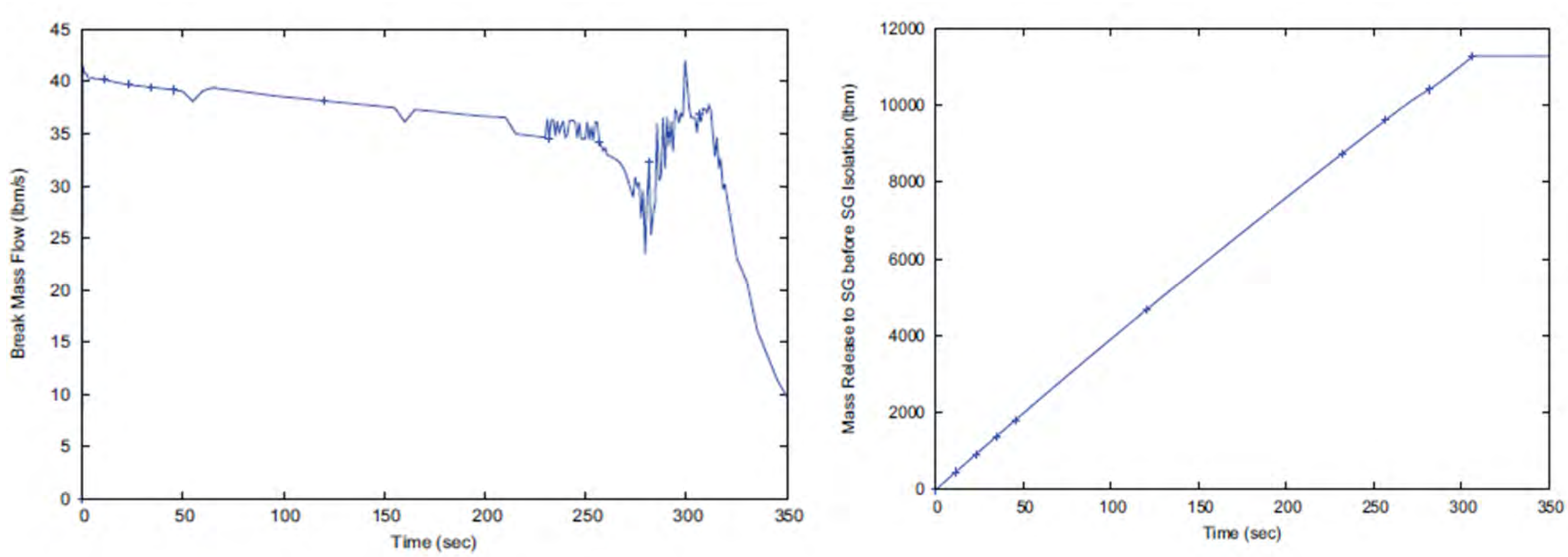
SGTF: Level Above Core – RCS Mass Flow





- Primary side inventory remains well above the top of the core
- DHRS on intact SG provides decay heat removal

SGTF: Break Mass Flow



- Total primary inventory released from the module limited by secondary isolation

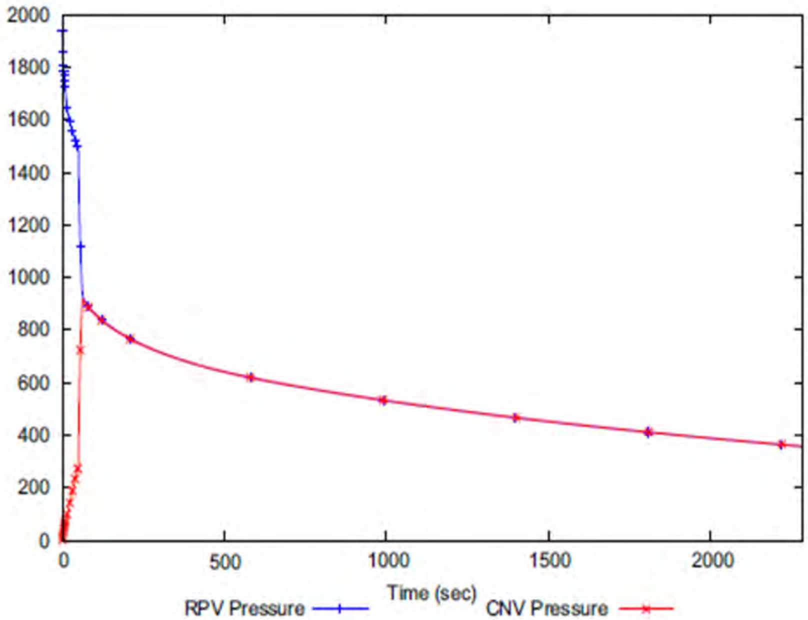
Example: Inadvertent RRV Opening (IORV)

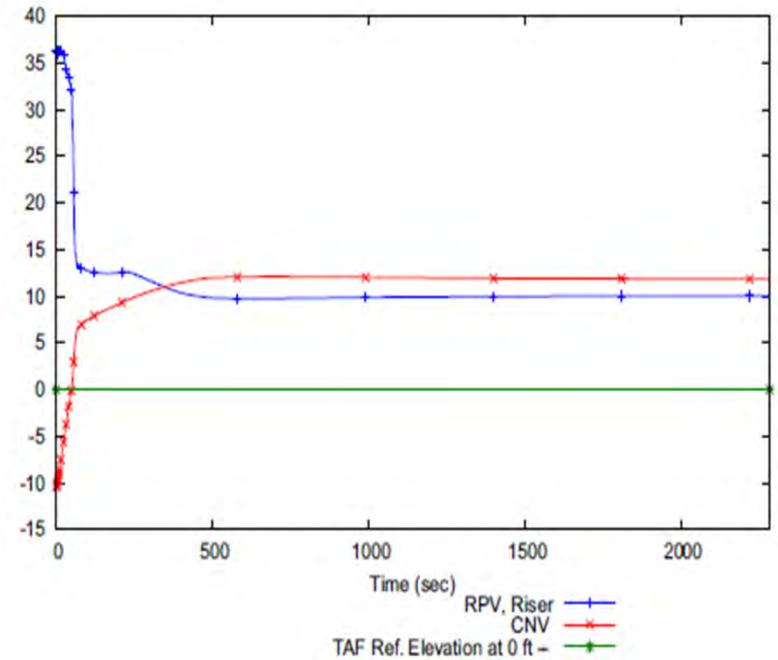
- Sequence of Events

Event	Time (s)*
RRV opens	0
Loss of normal AC and DC power	0
Control rods begin to fall	0
Minimum CHFR occurs	0.5
Control rods fully inserted into core	2.3
Remaining ECCS valves open	50
Natural circulation from containment to reactor pressure vessel is established	483
Peak steam generator pressure is reached	490
Minimum collapsed liquid level above the core	630

*Time rounded to the nearest tenth of a second.

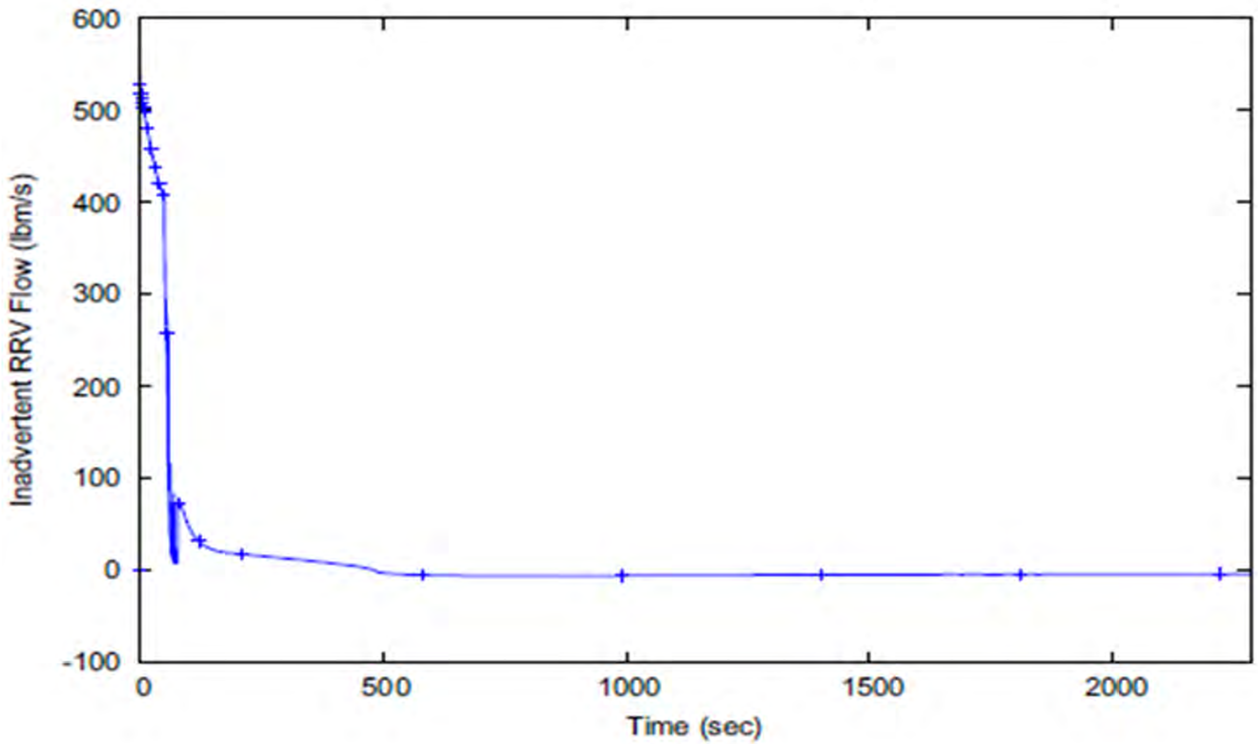
IORV: RPV and CNV Pressure – RPV Level





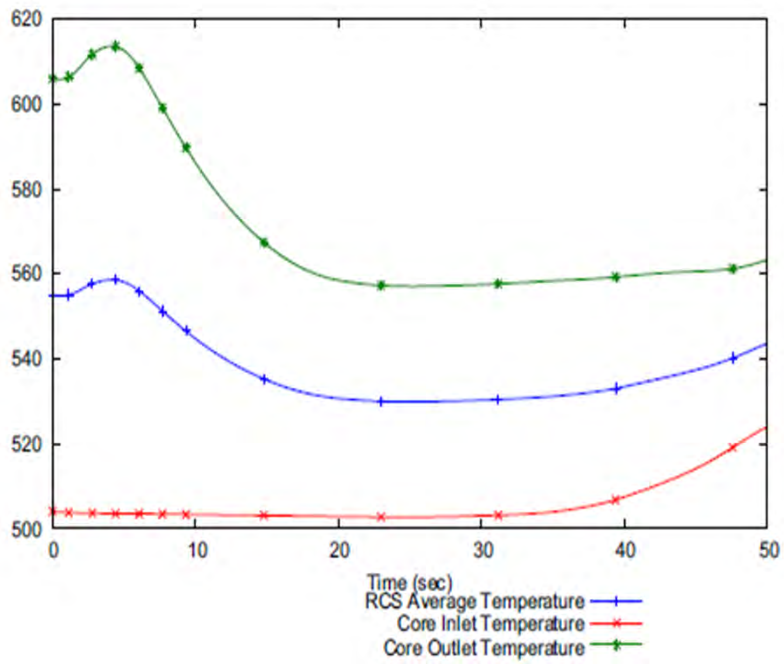
- RPV level remains well above top of active fuel

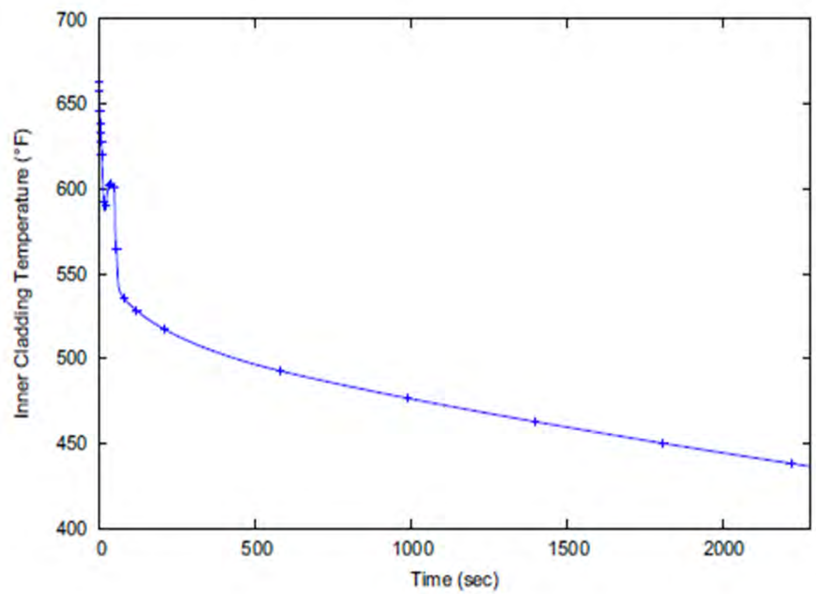
IORV: Reactor Recirculation Valve Flow



- After ECCS valves open, recirculation flow is established from containment into the RPV

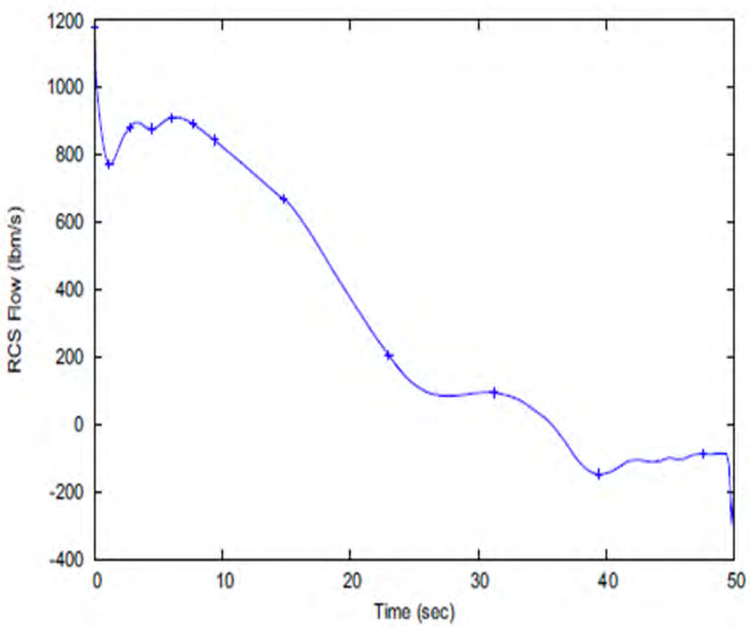
IORV: RCS and Cladding Temps

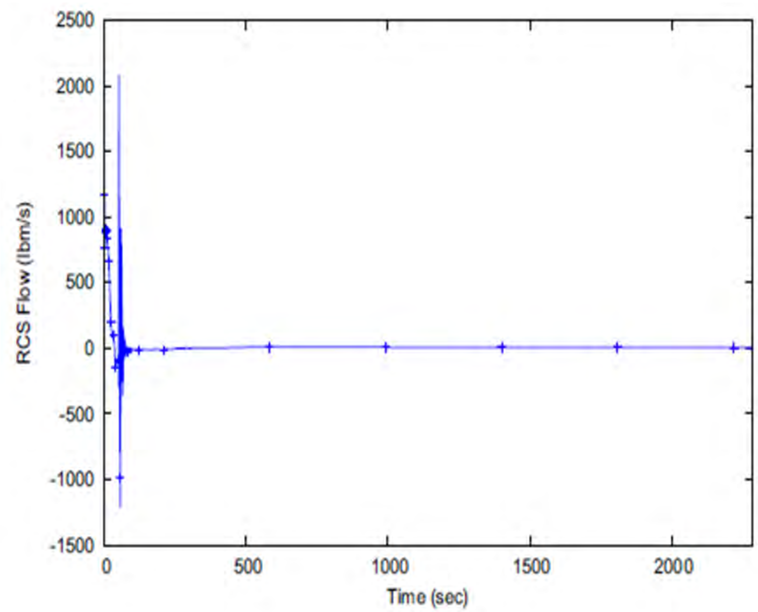




- Cladding temperatures decrease from steady-state condition

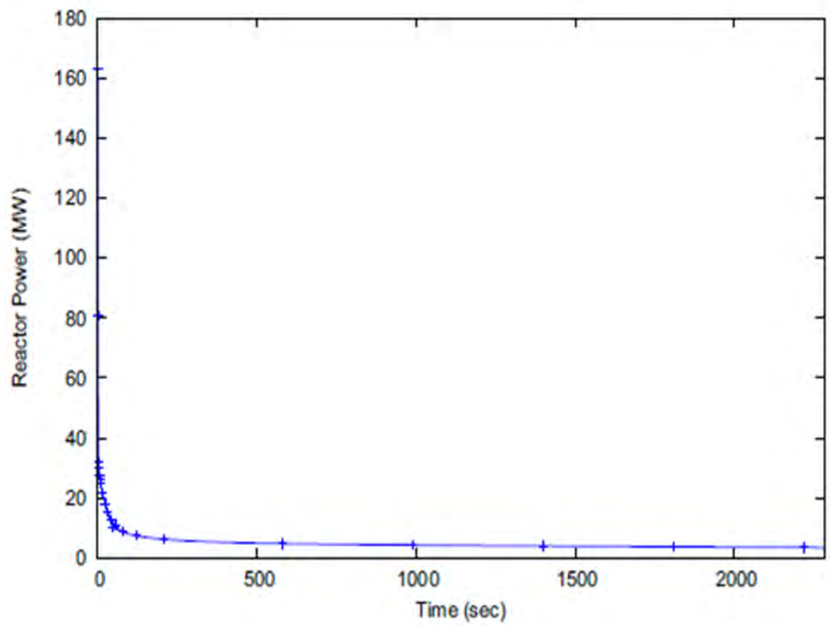
IORV: RCS Flow

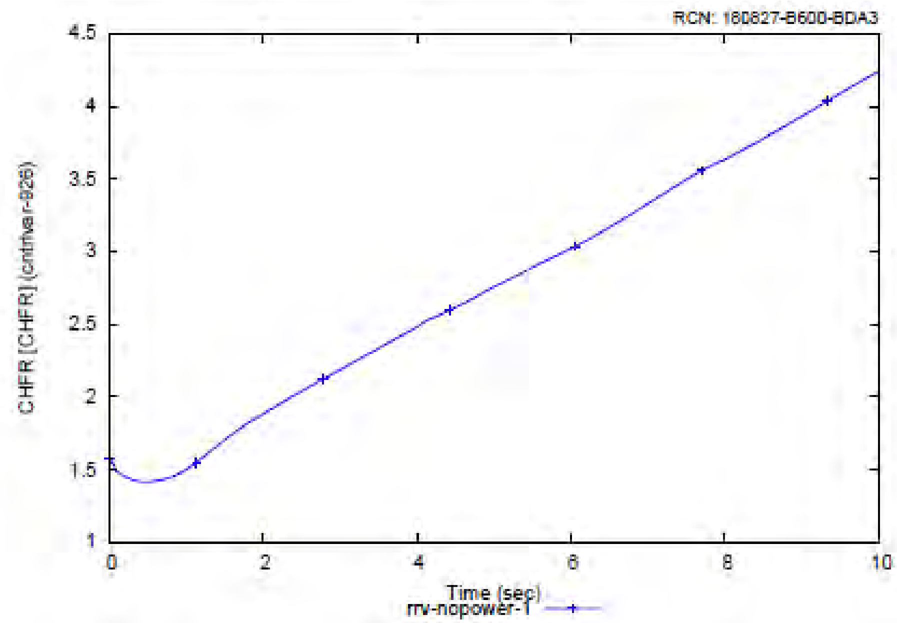




- Rapid RCS depressurization causes voiding in the core and momentary decrease in RCS flow leading to reduction in CHFR

IORV: Reactor Power - MCHFR





- MCHFR occurs in first few seconds of the transient, margin increase after reactor trip
- MCHFR result of 1.41 remains above acceptance criteria 1.13

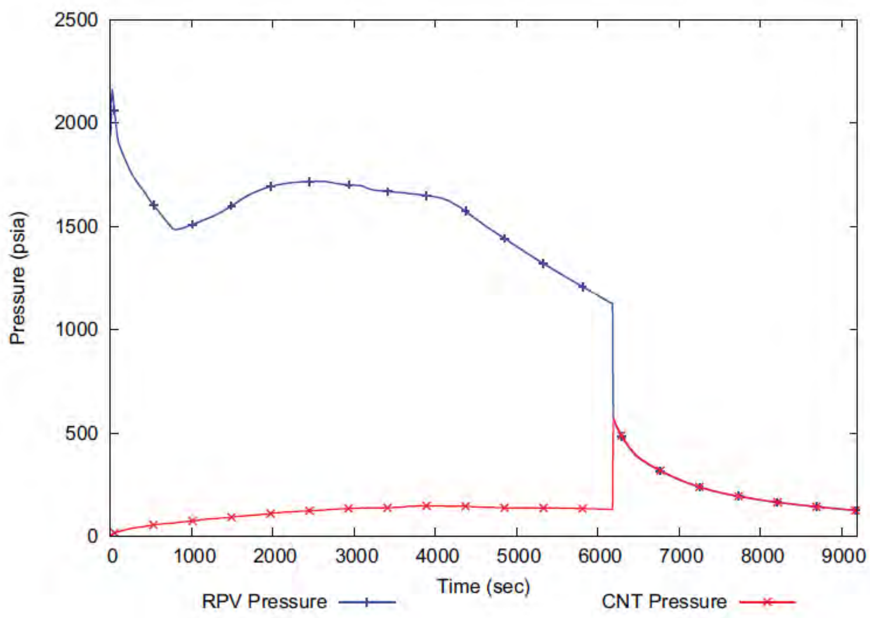
Example: LOCA

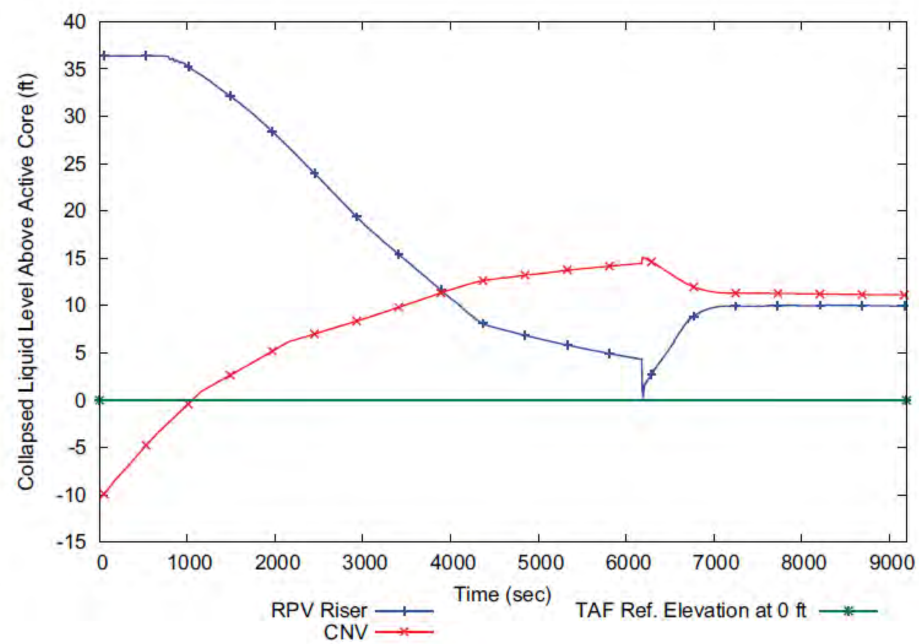
- Sequence of Events: 10% injection line break (0.53 inch equivalent diameter)

Event	Time (sec)*
Line break	0
Loss of normal AC	0
High pressurizer pressure (2000 psia)	6
Reactor Trip System actuation signal	8
Reactor trip	10
High containment pressure signal (9.5 psia)	16
Containment isolation	20
Low pressurizer level (35%)	323
Low pressurizer pressure (1600 psia)	504
Low Low pressurizer level (20%)	606
High containment water level (220 inches)	2238
Low RCS Level (370 inches)	3100
ECCS initiation begins	6181

*Time rounded to the nearest second.

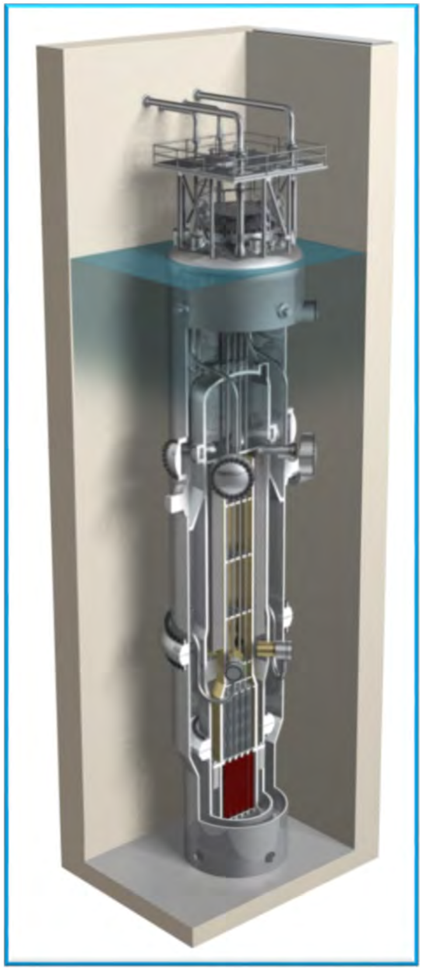
LOCA: RCS and CNV Pressure, Level





- DHRS heat removal conservatively not credited in LOCA EM:
  - ECCS valve opening on IAB release pressure is delayed until break flow sufficiently depressurizes RPV
  - RCS level decreases below equilibrium level before ECCS valves open.
- Minimum level at time of ECCS valve opening

ACRS Subcommittee Presentation



Chapter 15.0
Radiological Analysis

June 19, 2019

PM-0619-65981
Revision: 0

Template #: 0000-21727-F01 R5

Radiological Dose Consequences Event Summary and Acceptance Criteria

Event	Standard Review Plan Section	Regulatory Guide 1.183 Appendix[1]	Acceptance Criteria: Exclusion Area Boundary And Low Population Zone[2][3] (rem TEDE)	Acceptance Criteria: Control Room[4] (rem TEDE)	Radiation Source
Loss-of-coolant accident	15.0.3	A	25	5	Damaged fuel
Fuel handling accident	15.7.4	B	6.3	5	
Rod ejection accident	15.4.8.A	H	6.3	5	
Main steam line break	15.1.5.A	E	25 (Fuel damage or pre-incident spike) 2.5 (Coincident iodine spike)	5	Coolant activity (with iodine spiking)
Steam generator tube failure	15.6.3	F	25 (Fuel damage or pre-incident spike) 2.5 (Coincident iodine spike)	5	
Primary coolant line break	15.6.2	n/a	2.5	5	
Feedwater system pipe break	15.2.8	n/a	2.5[5]	5	

(1) Appendices C, D, and G were not included because they are not applicable to the NuScale design.
(2) Based on 10 CFR 52.47 (LOCA), RG 1.183, and 10 CFR 20.1301.
(3) Individual at the EAB shall not receive dose limit for any 2 hour period flowing the onset of release.
(4) Based on 10 CFR 52.47 and is for the duration of the event.
(5) Small fraction (10%) of regulatory dose reference value (25 rem TEDE).

15.0.1 - Radiological Consequence Analyses Using Alternative Source Terms

- A modified methodology based largely on RG 1.183 alternative source term is used to evaluate radiological consequences of design basis events (DBEs) and the beyond-design-basis core damage event (CDE).
- Radiological consequences of the feedwater line break event are bounded by the consequences of a steam line break; radiological analysis not required
- Reactor building pool boiling radiological consequences evaluated to show negligible dose consequence
- Potential radiation shine exposures to operators within the control room following a radiological release event are evaluated to have no impact to total dose
- Doses remain below applicable limits for all events

15.0.2.4 - Radiological Analyses Methodology

- SCALE 6.1, TRITON, and ORIGEN-SCALE – used to calculate the time-dependent isotopic source term for radiological transport models
- NRELAP5 – used to provide event-specific thermal-hydraulic conditions for radiological transport models for DBAs, which don't have core damage.
- MELCOR – used to model the progression of severe accidents for the CDE
- ARCON96 – used to calculate onsite and offsite atmospheric dispersion factors for DBEs, and the CDE
- RADTRAD – used to estimate radionuclide transport and removal for the various DBEs and the CDE
- STARNAUA – used to perform aerosol removal calculations for the CDE
- NuScale pH_T Code – used to calculate post-accident aqueous molar concentration of hydrogen ions for iodine re-evolution evaluation in the CDE
- MCNP6 – used for evaluating potential shine radiological exposures or doses to operators in the control room following a radiological release event.

15.0.3 - Design Basis Accident Radiological Consequences

- Included in topical report TR-0915-17565 for completeness
- RG 1.183 Appendices C and D (BWR), and G (locked rotor) were not addressed because they are not applicable to the NuScale design
- RG 1.183 iodine spiking assumptions and decontamination factors for fuel handling accident were utilized
- No credit for iodine removal in piping and condenser
- Thermal-hydraulic response to an REA shows no resultant fuel failure; radiological analysis not required per RG 1.183 Appendix H
- Iodine spike DBST evaluated to serve as a bounding surrogate for design-basis loss of primary coolant into containment events
- ARCON96 atmospheric dispersion methodology used due to short distance to EAB and LPZ
- RADTRAD modeling techniques utilized consistent with RG 1.183

15.10 - Core Damage Event Radiological Consequences

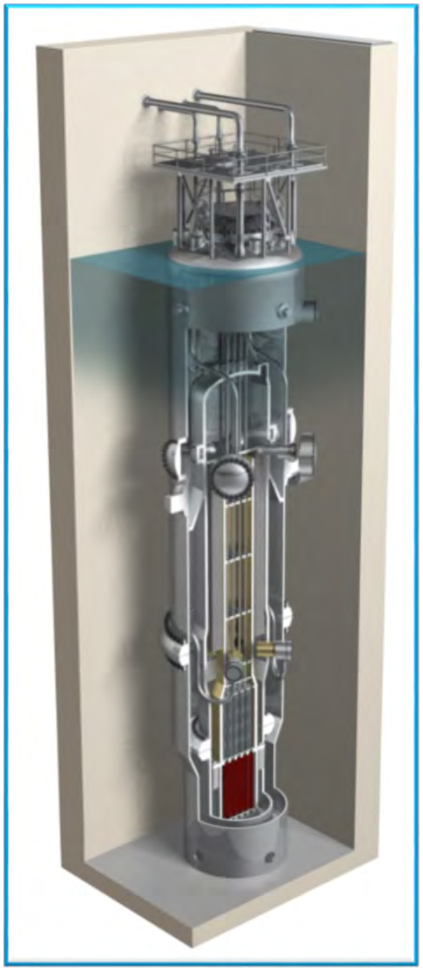
- Detailed methodology provided in AST topical report TR-0915-17565
- Utilized NEI position paper on Small Modular Reactor Source Terms
- Event characteristics derived from a spectrum of MELCOR surrogate accident scenarios
- ARCON96 atmospheric dispersion methodology used due to short distance to control room, EAB and LPZ
- RADTRAD modeling techniques utilized consistent with RG 1.183
- STARNAUA used for modeling natural removal of containment aerosols for the CDE
- pH_T evaluated to demonstrate negligible contribution of iodine re-evolution to dose consequences for the CDE

Chapter 15 Radiological Dose Consequences

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.

ACRS Subcommittee Presentation



Chapter 6.2.1
Containment Response Analysis

June 19, 2019

PM-0619-65981
Revision: 0

Template #: 0000-21727-F01 R5

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
- Limiting event scenarios addressed:
  - Primary side:
    - LOCAs: CVCS discharge line, CVCS injection line, pressurizer high point vent line
    - Valve opening events: Inadvertent RRV opening, Inadvertent RVV opening
  - Secondary side:
    - Main steam line break
    - Feedwater line break

6.2.1 CNV Response Analysis

- Qualification of NRELAP5 code based on code qualification and NPM plant modeling approach described in :
  - LOCA EM (for primary side pipe breaks and reactor valve opening events)
  - Non-LOCA EM (for main steam line and feedwater line break events)
- In LOCA EM development, PIRT identified high ranked phenomena important for prediction of the containment response.
- In non-LOCA EM development, containment pressure response was considered as a FoM in the PIRT

6.2.1 CNV Response Analysis

- 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
- In accordance with NuScale DSRS, initial RPV mass and energy maximized
- Maximum break sizes/valve sizes applied to maximize mass and energy release to the CNV
  - Choked flow discharge coefficient Cd=1.0 applied
  - Timing of ECCS valve opening evaluated based on level actuation setpoint and IAB release pressure to determine limiting condition

Containment Response Analysis Results

Event Description	CNV Pressure Limiting Case (psia)	[Base Case]	CNV Wall Temperature Limiting Case (°F)	[Base Case]
RCS discharge Break	943	[705]	510	[492]
RCS injection Line Break	959	[894]	526	[514]
RPV High point vent line break	901	[554]	489	[471]
Inadvertent RVV opening	911	[856]	486	[483]
Inadvertent RRV opening	986	[941]	512	[492]
Main steam line break	449	[< 449]	433	[< 433]
Feedwater line break	416	[< 416]	408	[< 408]

CNV design limits	1050 psia	550°F

NUSCALE
Power for all humankind

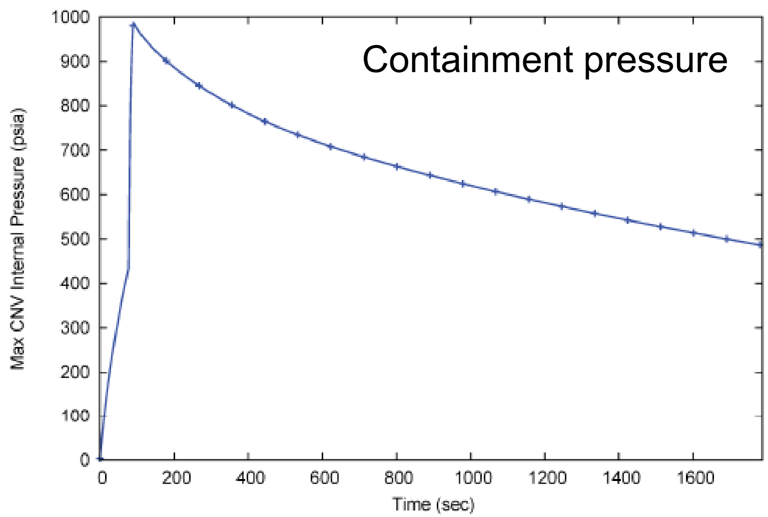
Containment Response Limiting Pressure Case

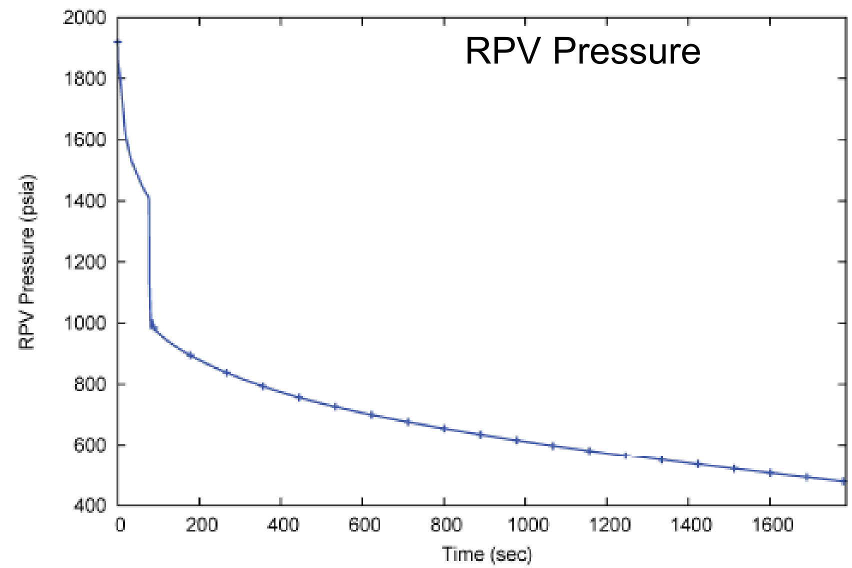
- **Initiated by inadvertent RRV opening**
- **Loss of normal AC and EDSS power at event initiation**
- **Single failure of remaining RRV to open**
- **Low bias IAB opening pressure**
- **Fast release of non-condensable gas into CNV assumed**
- **No DHRS operation credited**
- **Maximum CNV pressure: 986 psia < 1050 psia design pressure**

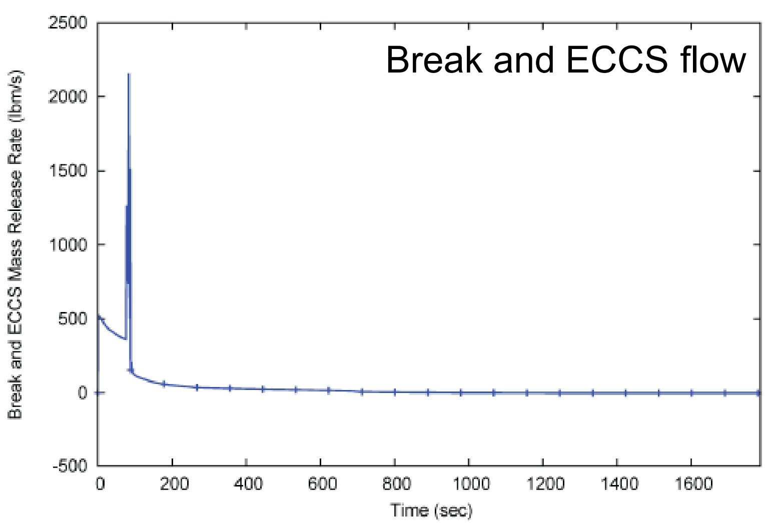
Time (sec)	Event
0	Inadvertent RRV opening Loss of normal AC and DC power FW/MS isolation Reactor trip
0.4	High CNV pressure – CNV isolation
74	ECCS valve opening on IAB release
91	Peak CNV pressure 986 psia
~1800	CNV pressure < 50% peak pressure

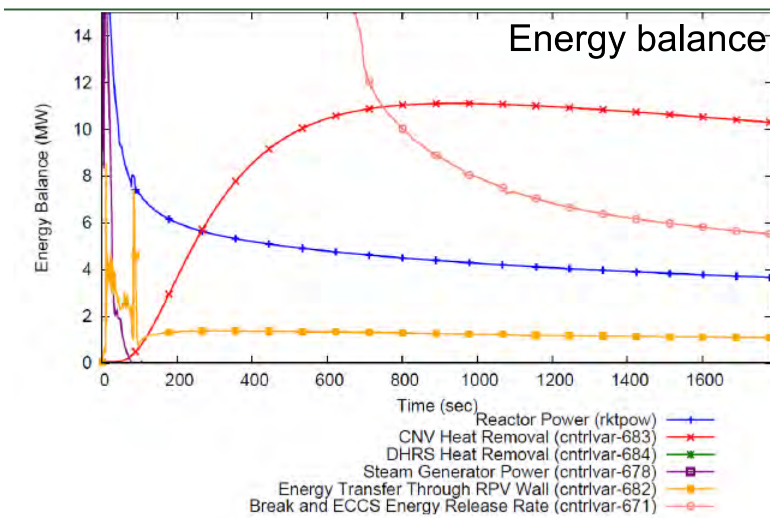
PM-0619-65981
Revision: 0

Containment Response Limiting Pressure Case

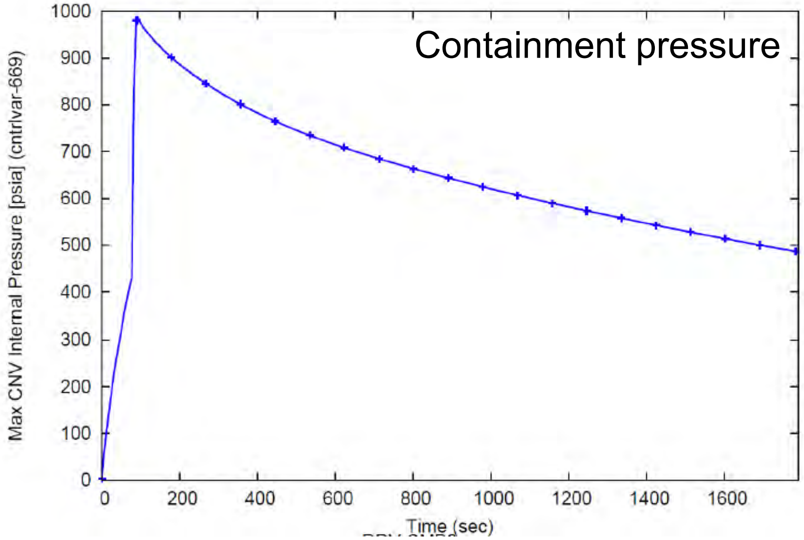


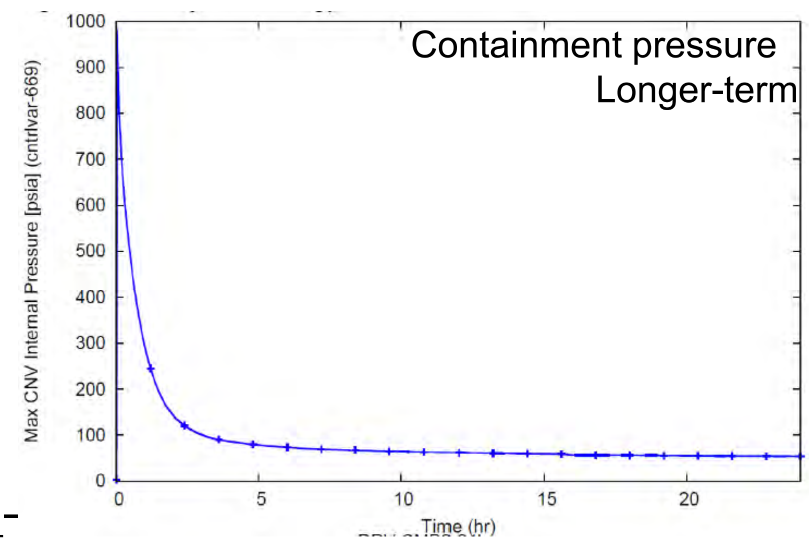






Containment Response Limiting Pressure Case





After peak CNV pressure, CNV continues to depressurize and pressure remains well below 50% of the design limit

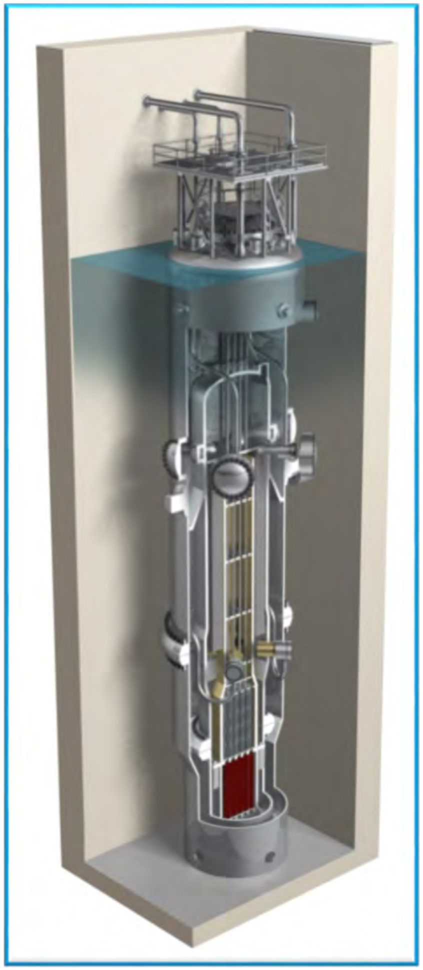
TR-0916-51299-P: Ch 15 long term cooling analyses demonstrate effective decay and residual heat removal

Peak pressure margin assessment

- **Maximum pressure 986 psia < 1050 psia design pressure**
- **Considering both external pressure and DHRS, ~59 psi additional margin not credited**
  - Internal pressures are conservatively evaluated with an assumed external pressure of 0 psia
    - Atmospheric pressure and pool hydrostatic head ~ 22 psi additional margin not credited
  - Decay heat removal system is a single-failure proof, safety-related system
    - Sensitivity calculations indicate ~ 37 psi additional margin could be gained

Containment response analysis provides assurance that the NPM design demonstrates sufficient margin to satisfy the requirements of GDC 16 and GDC 50

ACRS Subcommittee Presentation



Chapter 15
Long Term Cooling

June 19, 2019

PM-0619-65981
Revision: 0

Template #: 0000-21727-F01 R5

Long Term Cooling

Regulatory Requirements

- **10 CFR 50.46(b)**:
  - (4) Maintain coolable geometry
  - (5) Provide long term cooling after successful initial operation of the ECCS

- **PDC 35**: Provide abundant emergency core cooling to transfer heat from reactor core such that fuel and clad damage that could interfere with core cooling is prevented and clad metal-water reaction is limited to negligible amounts

- DSRS 6.3
  - Maintain safe shutdown conditions for 72 hours without operator actions and without nonsafety-related onsite or offsite power

- DSRS 15.6.5
  - Address boron precipitation
  - *'Steam generator tube rupture events shall also be reviewed as part of the LOCA break spectrum analysis. The reviewer shall review the potential coolant inventory loss from reactor vessel to the secondary side.'*

Long Term Cooling Acceptance Criteria

Core cooling is provided to remove decay and residual heat from the core

- Collapsed liquid level in the RPV remains above the top of fuel
- Cladding temperatures remain acceptably low

Coolable geometry is maintained

- Boron precipitation analysis ensures boron concentration in the core remains below solubility limit

NRELAP5 used to evaluate collapsed liquid level and core inlet temperature

Collapsed liquid level and core inlet temperature used to evaluate margin to boron solubility limits

NUSCALE™ Power for all humankind
Template #: 0000-21727-F01 R5

Long Term Cooling TR-0916-51299-P

- NRELAP5 used to model integral response of the decay heat removal path from ECCS and CNV heat transfer to the reactor pool
  - LTC analysis performed for design basis shutdown conditions
  - Confirmation of SAFDLs during overcooling return to power addressed by separate analysis
- PIRT high-ranked phenomena for ECCS cooling addressed by
  - LOCA EM validation
  - Bounded input/methodology
  - Validation against NIST-1 tests HP-19a, HP-19b
- Qualification of NRELAP5 code based on code qualification and NPM plant modeling approach described in :
  - LOCA EM (for primary side pipe breaks and reactor valve opening events)
  - Non-LOCA EM (for initial DHRS cooldown events)

Long Term Cooling Boron Precipitation Analysis

- Simplified, conservative mixing volume approach
- No time dependence
  - Assume all boron in RCS concentrated in liquid volume in core and riser region
  - Assume perfect mixing of boron in the core/riser region
- Maximum allowable boron concentration 1800 ppm (HZP conditions)
- For given liquid volume during transient, confirm core inlet temperature greater than boron precipitation temperature

Long Term Cooling Evaluation

- LOCA events transition to ECCS cooling
  - Full LOCA break spectrum evaluated, includes inadvertent opening of RRV or RVV
- Non-LOCA events transition from DHRS to ECCS cooling
  - Events considered are SGTF, loss of FW flow, and a general evaluation of DHRS cooldown
- Transient calculated for 12.5 hours after ECCS actuation to capture minimum RPV level and long term level recovery
  - LTC phase begins after ECCS cooling is established, however transient history can affect mid-term level response, and the initiating event phase is also modeled
- State point analysis calculates final conditions after 72 hours
  - Decay heat set to constant value at 72 hours and the model is run until temperatures converge
- Injection line break identified as overall limiting initiating event

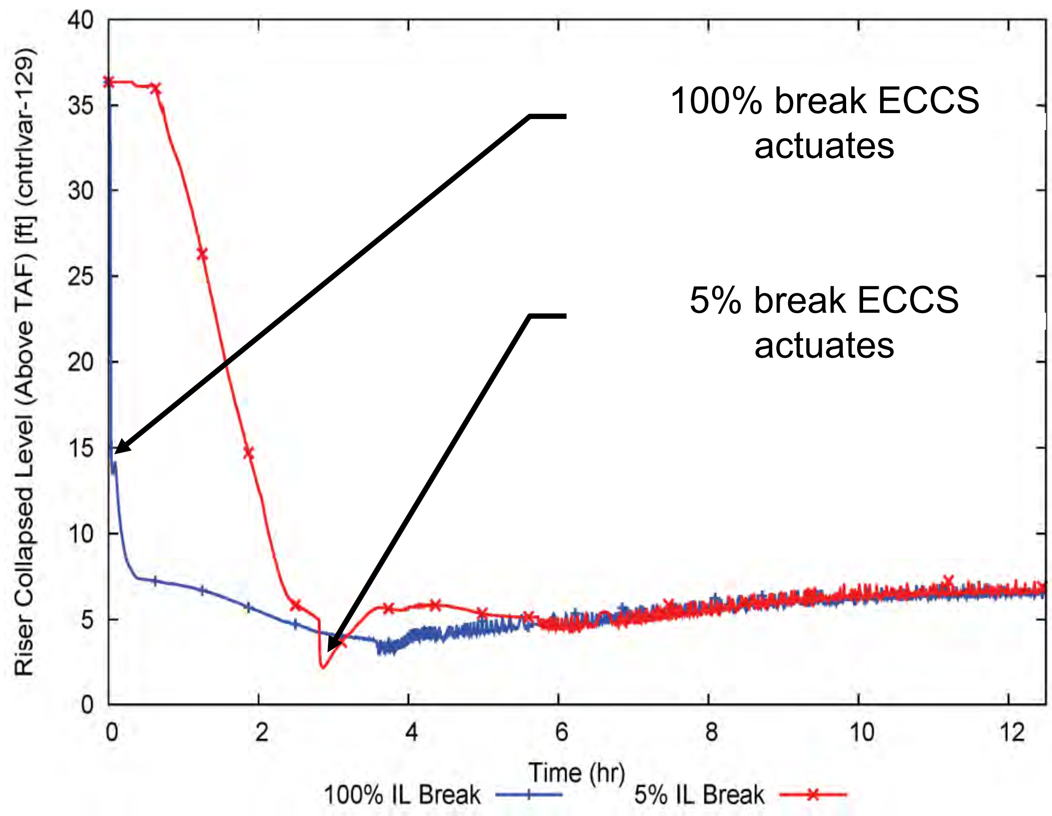
Long Term Cooling Evaluated Conditions

	Minimum Level		Minimum Temperature	Maximum Temperature	
Reactor Power (%)	102		13-102	102	
Decay Heat (multiplier)	1.2 (LOCA)	1.0 (nonLOCA)	0.8	1.2 (LOCA)	1.0 (nonLOCA)
RCS Avg. T. (°F)	555		535	555	
RCS P. (psia)	1780		1780	1920	
PZR Level (%)	52		68	52	
Pool T. (°F)	65		65	210	
Pool Level (ft)	69		69	55	
Non-condensable Gas (lbm)	0		0	~131	
ECCS Capacity (area and Cv)	minimum		maximum	minimum	
Expansion Factor (Y)	0.7		1	0.7	
Single Failures	RVV/RRV		none	RVV/RRV	
DHRS Enabled	No		Yes	No	

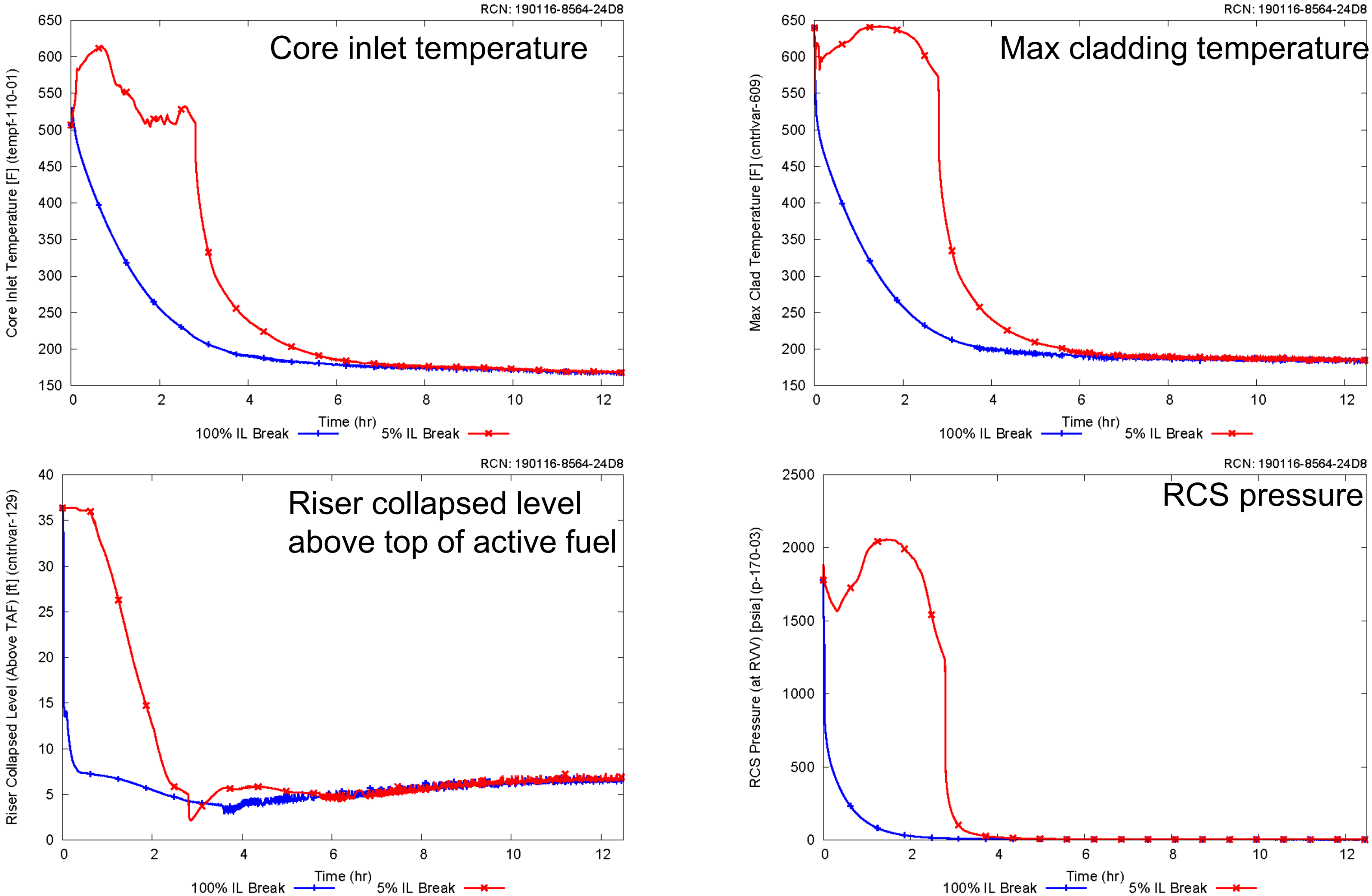
- Minimum level confirms core remains covered
- Minimum temperature confirms boron precipitation precluded
- Maximum temperature confirms acceptably low cladding temperature

LTC Results – Injection Line Minimum 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 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

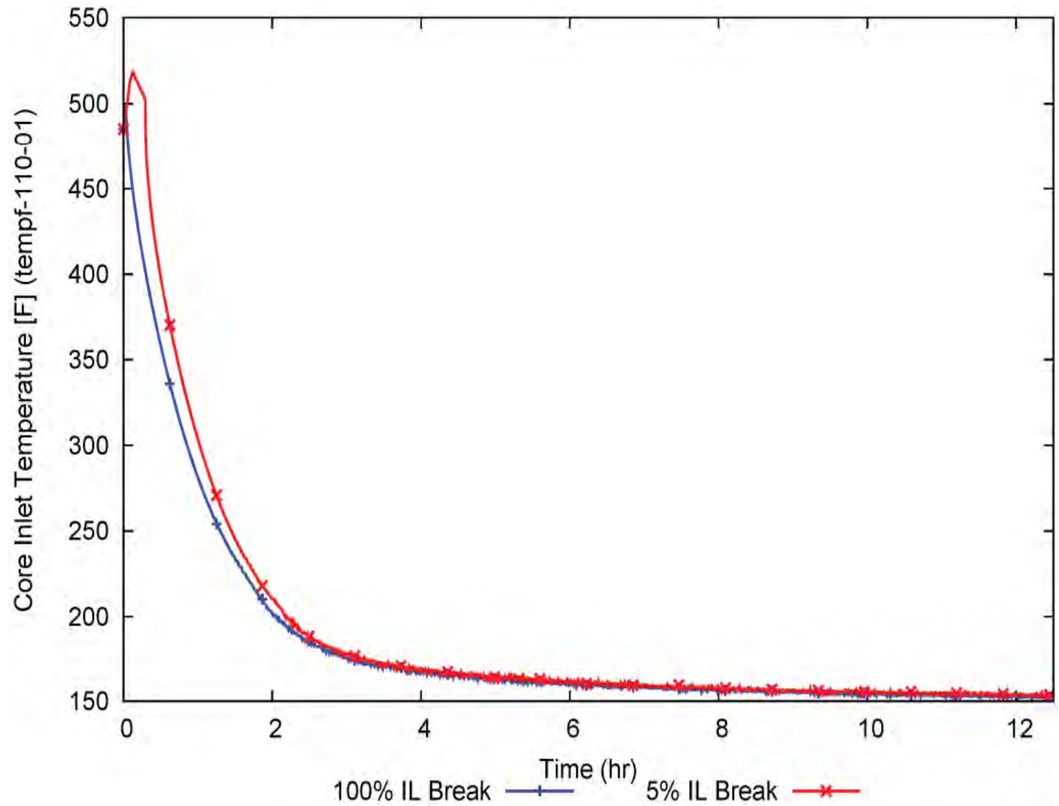


LTC Results – Injection Line Minimum Level



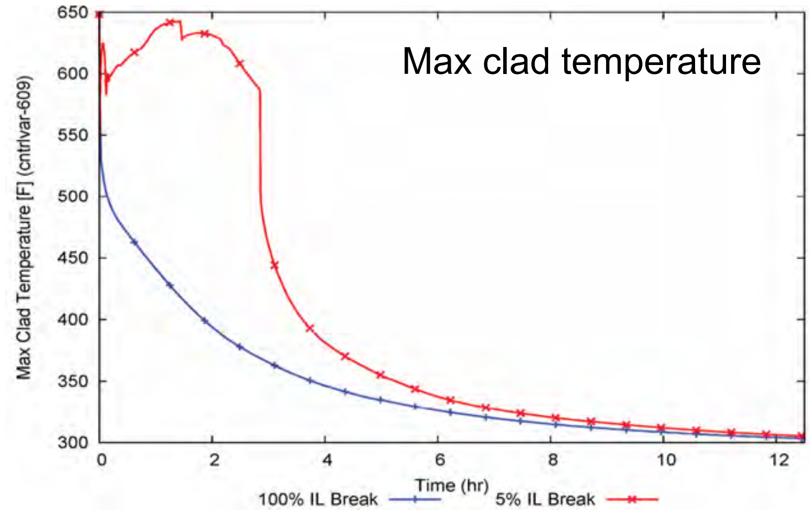
LTC Results – Injection Line Minimum Temperature

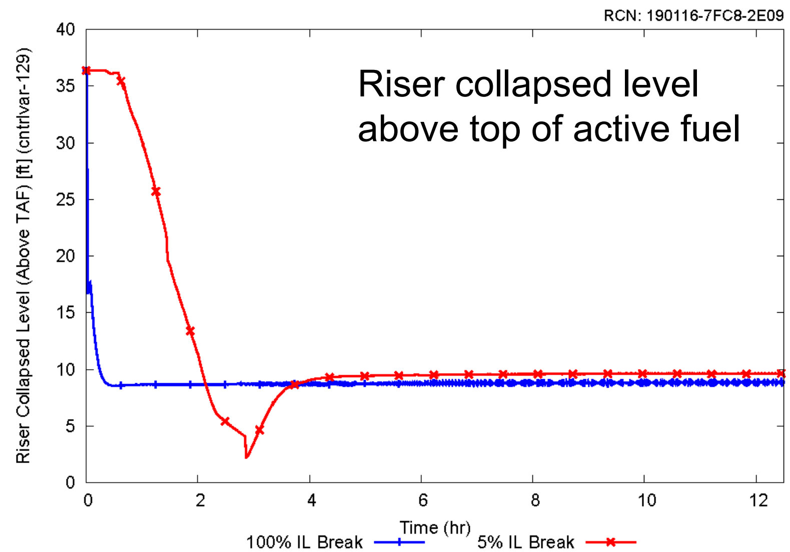
- Minimum temperature case minimizes initial RCS energy and decay heat while maximizing heat transfer to the reactor pool
- Core temperature used as input to boron precipitation analysis



LTC Results – Injection Line Maximum Temperature

- 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

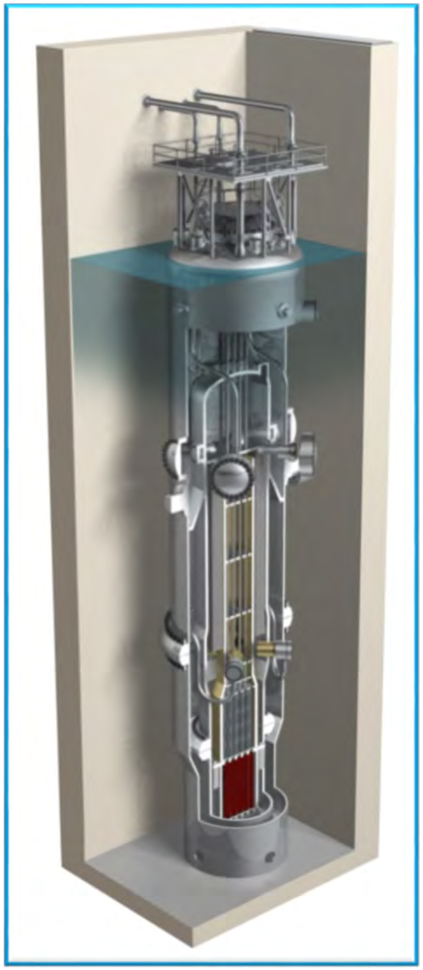




Long Term Cooling Results

- All maximum temperature cases showed decreasing cladding temperature, with final clad temperature remaining well below operating temperature at full power
- All minimum temperature cases showed margin to boron precipitation
- 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
  - The limiting level case is a 100% injection line break, with minimum level of 2.8 feet above the core occurring approximately 3.6 hours after ECCS actuation

ACRS Subcommittee Presentation



Loss of Shutdown Margin

June 19, 2019

PM-0619-65981
Revision: 0

Template #: 0000-21727-F01 R5

Principle Design Criteria 27

- NuScale DCA includes an exemption request from GDC-27
  - ECCS design does not include boron addition
  - NPM design does align with precedent based compliance with GDC-27
- Principle Design Criteria 27
  - Passive reactor GDC-27 equivalent
  - Ensures the safety related reactivity control system is designed to achieve and maintain subcritical core
  - Ensures fuel integrity for an extended overcooling in combination with a partial failure of reactivity system (stuck rod)

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 Benign
  - 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 Mechanisms

- Moderator overcooling
  - ECCS and DHRS designed to removed decay and residual heat
  - Under cold reactor pool conditions DHRS or ECCS can cause a fairly rapid temperature decrease and increased moderation (reactivity insertion)
- Fission product decay
  - Xenon decay causes a slow post shutdown reactivity insertion
- Boron redistribution
  - Boiling/condensing systems cause boron redistribution
    - Boron concentration in boiling region
    - Boron dilution in condensing region
    - Limited boron acid volatized and carried with vapor
  - Conclusion: Boron redistribution during extended ECCS operation increases boron and SDM in the core

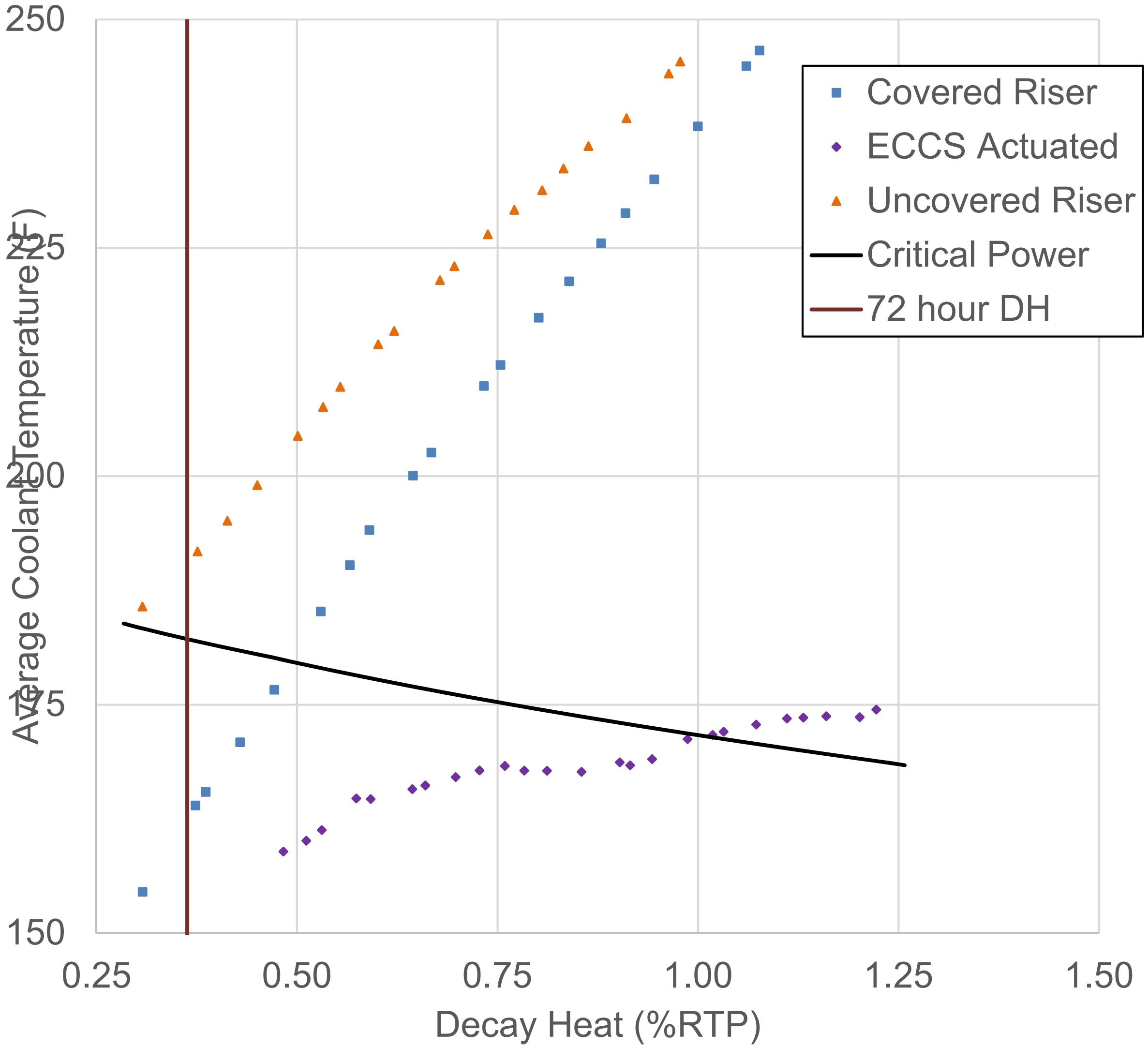
Evaluation of Return to Power

- DHRS or ECCS passive heat removal can be characterized as having a de-energizing phase and equilibrium phase
  - At higher RCS temperatures passive heat removal capabilities well exceed decay heat generation (de-energizing phase)
  - At lower RCS temperature conditions, passive heat removal nearly matches decay heat (equilibrium phase)
- Under maximum cooldown conditions, the equilibrium phase is reached in about 6 hours for either DHRS or ECCS
- Xenon decay inserts reactivity from 12-72hrs at an average rate of <0.02pcm/s
- WRSO criticality is achieved <200°F

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 WRSO critical power level as a function of RCS temperature.
- Conclusions:
  - Loss of SDM achieved >40hrs with zero Boron in RCS after Xenon decay
  - RCS temperature must be <200F
  - ECCS cooling is more limiting
  - Pool temperature >140F precludes WRSO criticality
  - Simple pool boiling CHF analysis demonstrates large margin

Return to Power Results



Bounding Dynamic Return to Power Analysis

- Purpose
  - Demonstrate no challenge to SAFDLs, RCPB or CNV
- Assumptions
  - Extended DHRS cooldown event without AC or DC power
  - Zero boron EOC core conditions are applied to HZP RCS operating conditions (MDC curve applied)
  - Pool temperature minimized to maximize cooldown rate
  - Xenon decay consider as a reactivity source without consideration of time dependent worth
- Evaluation Procedure
  - Evaluate maximum core peak power response (power overshoot)
  - Calculate local peaking with core physics codes
  - Repeat system calculation with IAB release (ECCS actuation) assumed at time of power peak
  - Evaluate transient MCHFR to confirm acceptability
  - Confirm acceptable RCS and CNV pressure response

Return to Power Analysis DHRS

Figure 40: Reactor Power

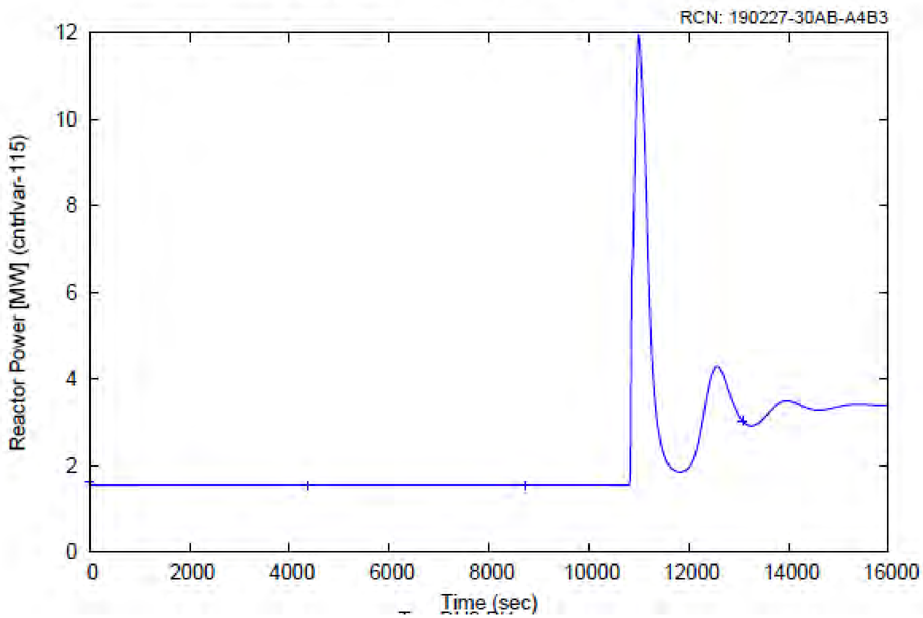
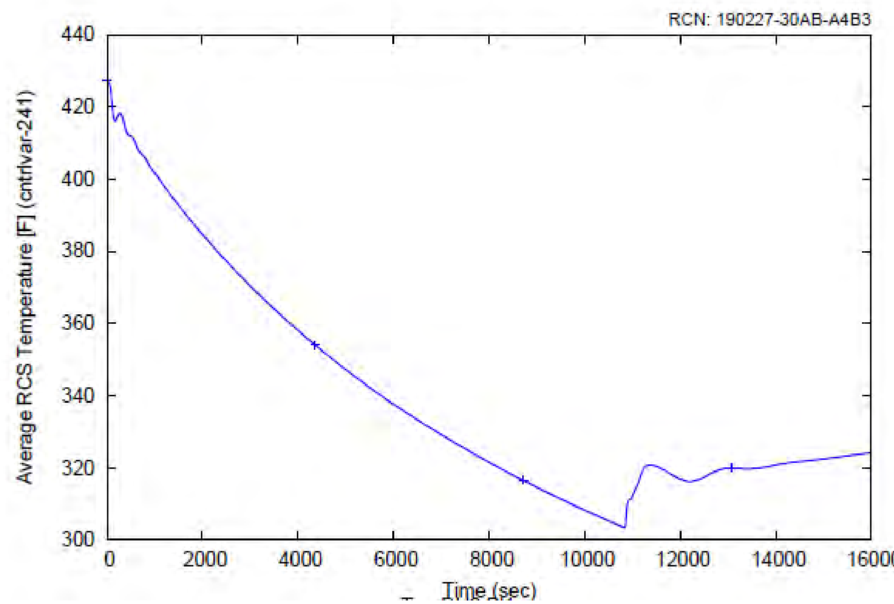


Figure 37: Average RCS Temperature



Return to Power Analysis ECCS

Figure 36: Reactor Power

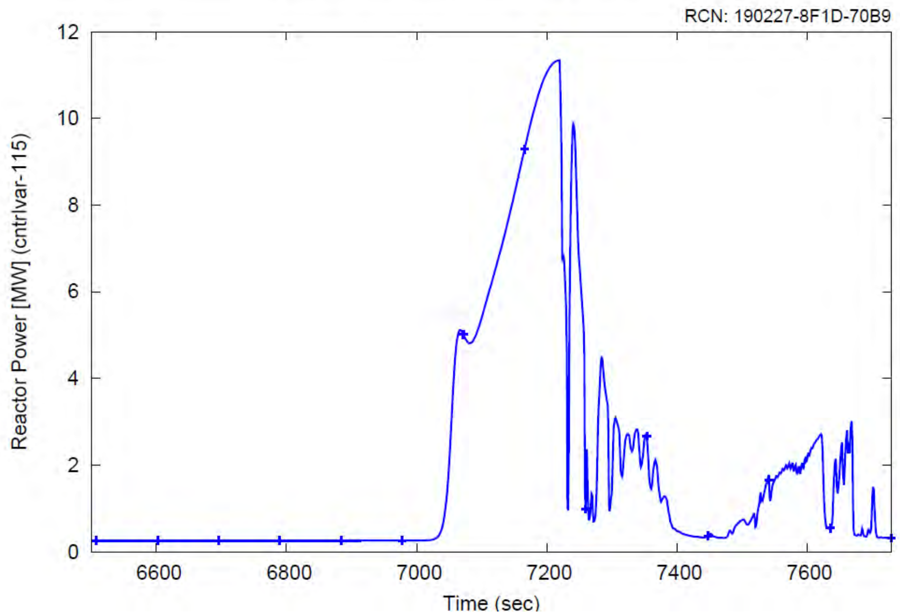
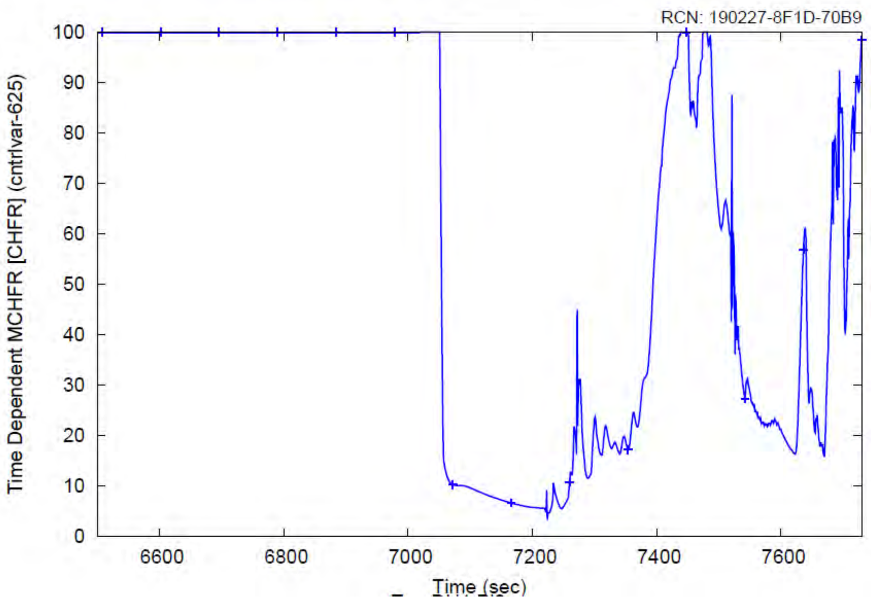


Figure 35: Time Dependent MCHFR



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
- CIV – Containment Isolation Valve
- CNV – Containment Vessel
- COL – Combined License
- CRDM – Control Rod Drive Mechanism
- CVCS – Chemical and Volume Control System
- DHRS – Decay Heat Removal System
- ECCS – Emergency Core Cooling System
- EFPY – Effective Full Power Years
- EPRI – Electric Power Research Institute
- ^{o}F – degrees Fahrenheit
- FIV – Flow Induced Vibration
- FSAR – Final Safety Analysis Report
- ft – feet
- FW – Feedwater
- FWIV – Feedwater Isolation Valve
- HZP – Hot Zero Power
- ISI – Inservice Inspection
- LOCA – Loss of Coolant Accident
- LTOP – Low Temperature Overpressure Protection
- MPS – Module Protection System
- MSIV – Main Steam Isolation Valve
- MSS – Main Steam System
- MWt – Megawatts thermal

Acronyms

- NDE – Non-destructive Examination
- NEI – Nuclear Energy Institute
- NPM – NuScale Power Module
- NPS – Nominal Pipe Size
- OD – Outside Diameter
- OE – Operations Experience
- 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
- RG – Regulatory Guide
- RPV – Reactor Pressure Vessel
- RSV – Reactor Safety Valve
- RT_{NDT} – Reference Temperature for Nil-ductility Transition
- RVV – Reactor Vent Valve
- SG – Steam Generator
- TRV – Thermal Relief Valve
- TS – Technical Specifications
- TT – Thermally Treated
- UHS – Ultimate Heat Sink
- USE – Upper Shelf Energy

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

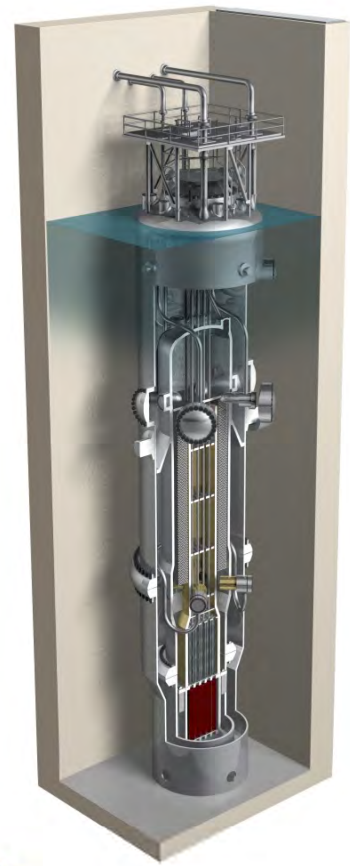
Rockville Office
11333 Woodglen Ave., Suite 205
Rockville, MD 20852
301.770.0472

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

NUSCALE™
Power for all humankind

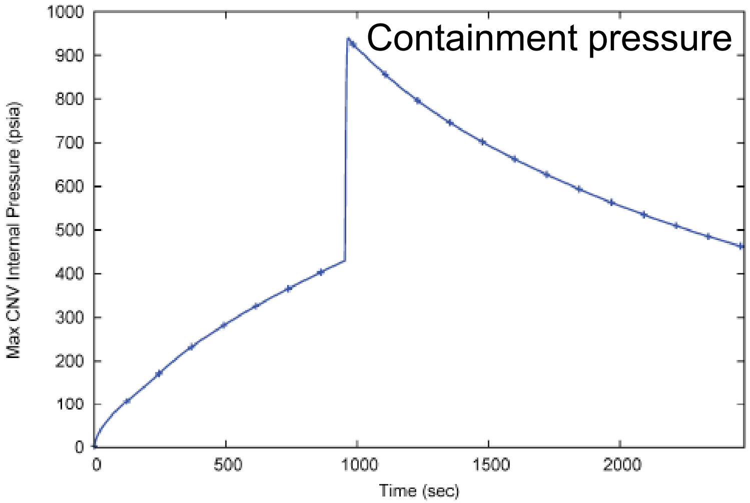
Backup Slides – Provide to Members

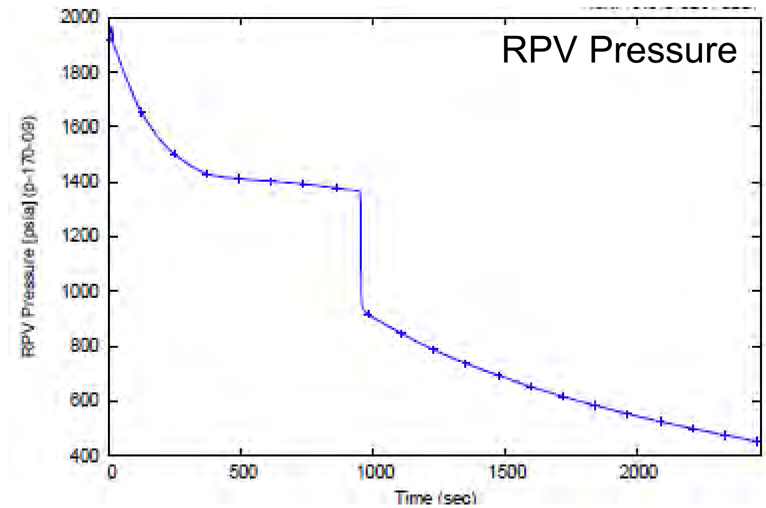
Containment Response Limiting Temperature Case

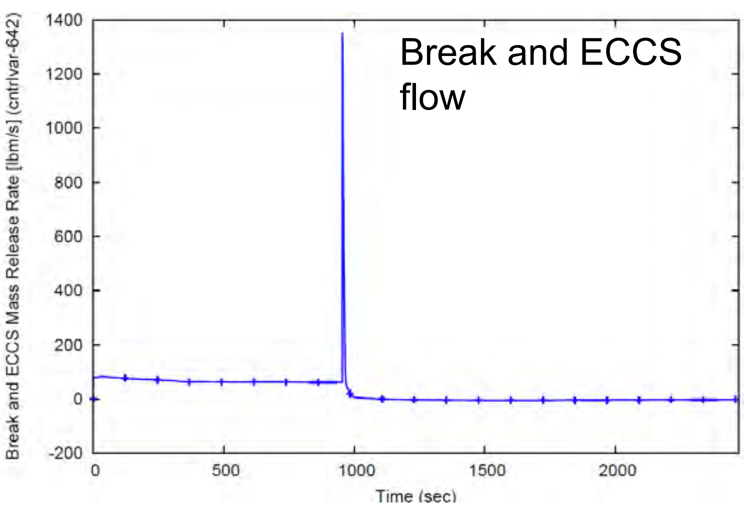
- Initiated by CVCS injection line break
- Loss of normal AC power at event initiation
- Single failure of one RRV to open
- Low bias High CNV Level setpoint
- No DHRS operation credited
- Maximum CNV wall temperature: 526°F < 550°F design temperature

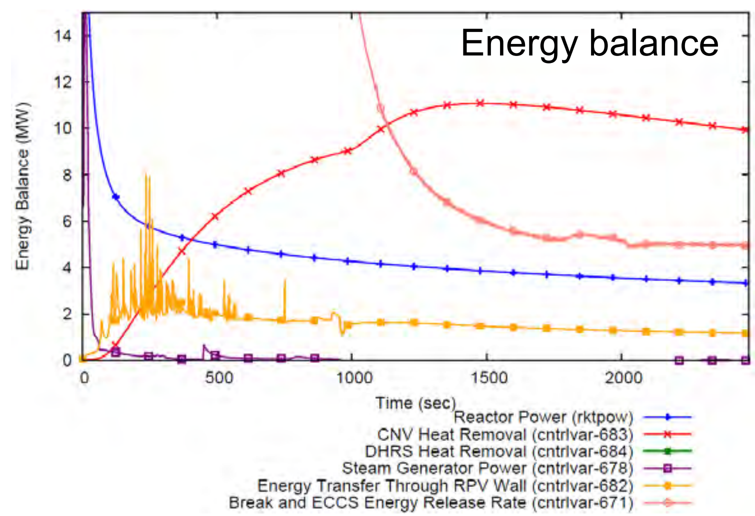
Time (sec)	Event
0	CVCS line break Loss of normal AC power FW/MS isolation
3	High CNV pressure: CNV isolation Reactor trip
952	ECCS actuation on high CNV level
955	ECCS valve opening
978	Peak CNV temperature 526°F
~2500	CNV pressure < 50% peak pressure

Containment Response Limiting Temperature Case

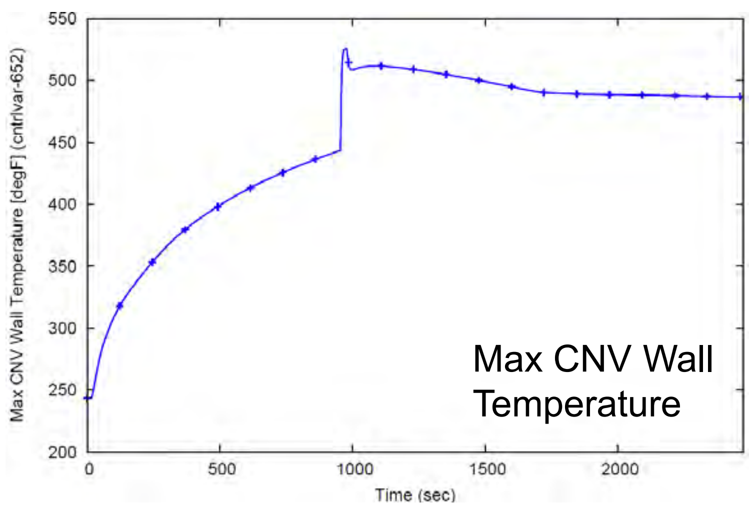


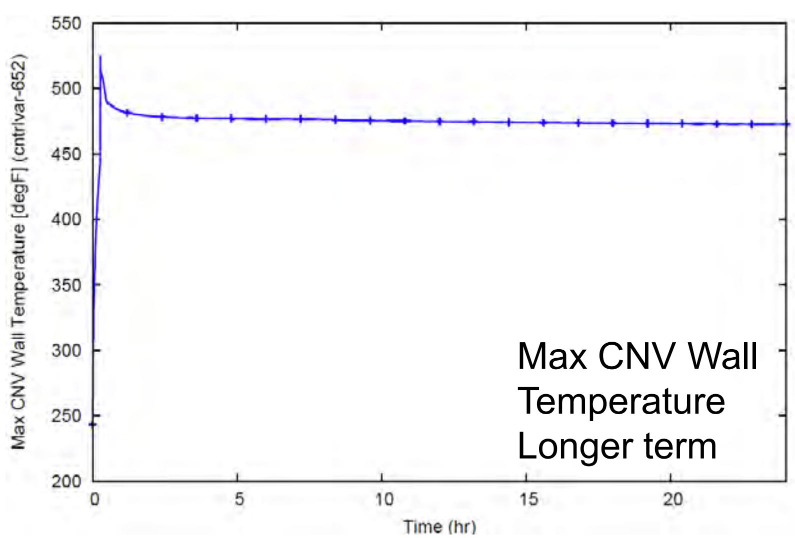






Containment Response Limiting Temperature Case





- **Maximum wall temperature results occur following ECCS actuation**
- **Long-term maximum temperature not reduced due to modeling conservatism of adiabatic wall boundary condition applied above pool level**

Long Term Cooling Results

- State-point results after 72 hours

Case Description	Core Inlet Temperature (°F)		Collapsed Riser Level (ft)		Boron Precipitation Margin (°F)	
	Transient at 12.5 Hours	State-point at 72 Hours	Transient at 12.5 Hours	State-point at 72 Hours	Transient at 12.5 Hours	State-point at 72 Hours
Maximum Temperature IL Break	292.8	270.4	8.9	9.1	208.9	187.8
Minimum Temperature IL Break	152.8	140.4	10.0	10.4	73.1	62.3
Minimum Level IL Break	165.3	154.5	7.3 (1)	8.0	76.2	69.0
Maximum Temperature IL Break, 45 ft reactor pool level (1)	-	280.3	-	9.2	-	197.6
Minimum Temperature IL Break, 13% initial power (1)	-	94.3	-	10.4	-	16.6
Minimum Temperature SGTF, 13% initial power (1)	-	112.1	-	10.1	-	33.3
Minimum Temperature DHRS cooldown, 13% initial power (1)	-	116.8	-	10.4	-	39.2

(1) A 12 hour transient simulation for these cases was not performed. Limiting conditions are only important at the end of the LTC phase at 72 hours.