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

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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671ST MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

OPEN SESSION

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THURSDAY

MARCH 5, 2020

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ROCKVILLE, MARYLAND

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The Advisory Committee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2D10, 11545 Rockville Pike, at 8:30 a.m., Matthew W. Sunseri, Chairman, presiding.

COMMITTEE MEMBERS:

MATTHEW W. SUNSERI, Chairman

JOY L. REMPE, Vice Chairman

WALTER L. KIRCHNER, Member-at-Large

RONALD G. BALLINGER, Member

DENNIS BLEY, Member

CHARLES H. BROWN, JR., Member

VESNA B. DIMITRIJEVIC, Member
JOSE MARCH-LEUBA, Member
DAVID PETTI, Member
PETER RICCARDELLA, Member

ACRS CONSULTANTS:
STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:
MIKE SNODDERLY

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

P R O C E E D I N G S

(8:30 a.m.)

CHAIRMAN SUNSERI: The meeting will now come to order. This is the first day of the 671st meeting of the Advisory Committee on Reactor Safeguards.

I am Matthew Sunseri, the Chair of the ACRS. Members in attendance today are Pete Riccardella, Ron Ballinger, Dave Petti, Joy Rempe, Walt Kirchner, Jose March-Leuba, Charlie Brown.

Dennis Bley is here. He'll be stepping in a minute and Vesna Dimitrijevic. We also have our consultant, Steve Schultz present as well. And I note that we have a quorum.

The ACRS was established by the Atomic Energy Act and it's governed by the Federal Advisory Committee Act.

The ACRS section of the U.S. NRC public website provides information about the history of the ACRS and provides documents such as our charter, bylaws, Federal Register notices for meetings, letter reports and transcripts of all full and subcommittee meetings, including slides presented at the meetings.

The Committee provides its advice on safety matters to the Commission through its publicly

available letter reports. The Federal Register notice announcing this meeting was published on February 21, 2020, and provides an agenda and instructions for interested parties to provide written documents or request opportunity to address the Committee.

The Designated Federal Official for this meeting is Mr. Mike Snodderly. During today's meeting the Committee will consider the following.

NuScale Area of Focus: Steam Generator Design, Containment Evacuation System and Hydrogen and Oxygen monitoring and number two, NuScale Topical Reports: Loss of Coolant Accident (LOCA), Non-LOCA and Rod Ejection Accident Methodology.

Following those presentations the ACRS will engage in preparation of reports. As reflected in our agenda, portions of the NuScale session may be closed in order to discuss and protect information designated as sensitive or proprietary. And I will say there will be closed sessions today.

A phone bridge line has been opened to allow members of the public to listen in on the presentations and Committee discussion. We have received no written comments or requests to make oral statements from members of the public regarding today's session.

There will be an opportunity for public comment and we have set aside time in the agenda for comments from members of the public attending or listening to our meetings. Written comments may be forwarded to Mr. Mike Snodderly, the Designated Federal Official.

A transcript of the open portion of the meeting is being kept and it is requested that speakers use one of the microphones, identify themselves and speak with sufficient clarity and volume so that they may readily be heard.

For the people that will be presenting today, I ask that you consider the following. We've seen a lot of the material. And in most of the subcommittee meetings on these topics we've had full committee membership participation.

So, please feel free to progress smartly through, you know, maybe the background material and stuff that we've seen before and focus your detail on the things that you've been briefed on as important to us because we know you know what topics are important to us.

If we need to slow you down we will slow you down. So, let us control the pace.

Just one thing before we get into the

presentations. I do have an item of interest that I want to make public. Today in the Federal Register notice a notice was published that we are seeking qualified candidates for membership on the ACRS.

The ACRS is seeking two members, one with nuclear power plant experience and a second one regarding, with risk analysis and the consideration of uncertainty in decision making. So, those positions fill out vacant and soon to be vacant with retirement the positions.

And any interested candidates should follow the instructions on the Federal Register notice. We will now begin the presentations with NuScale.

And I'll turn to staff to see if they have any remarks that you want to make before the NuScale presentation. Who is, Rebecca, are you over there?

MS. PATTON: No. We just thank the Committee for their time and hope for a productive dialogue.

CHAIRMAN SUNSERI: Okay, thank you. And now, Marty, the floor is yours for the NuScale.

MR. BRYAN: Okay, thanks, Matt. I'm Marty Bryan. I'm the licensing project manager for Chapter 3. I've got with me Bob Houser, Kevin Spencer,

Matthew Presson and also Brian Wolf will be joining us on the phone for part of the presentation.

So, today in open session it's fairly brief. We're going to get into more of the feedback we received in the closed session. But certainly ask questions if something comes up.

So, we're going to do just a brief overview of Steam Generator Design and then talk a little bit about the proposed DCA revisions that we intend to include in the errata for Rev 4. So, I'll turn it over to Kevin.

MR. SPENCER: So, I'm Kevin Spencer. I'll be doing a brief overview of the Steam Generator Design this morning. This was previously presented so I'll try to -- I'll make it fairly high level.

Each NuScale power module has two steam generators. On the shell side we have the primary fluid. On the tube side we have the secondary fluid.

We have about 1,380 tubes overall. They range in length from 74 to 86 feet. It is a helical coil design. Each tube is made out of Alloy 690, thermally treated material.

I have brought with me this morning a little, a plastic prototype of the steam generator tubes and how they interact with the steam generator

supports. I'll pass this around.

Feel free, it does come apart. If it falls apart you can put it back together easily. But it will allow you to take a look at how the helical coil tubes interact with the tube supports. So, I'll pass this around.

MEMBER MARCH-LEUBA: While you still have it in your hand, what's the length of the straight shot on the tube? When does it start curving because you're going to put the other thing, the metal thing inside it, right?

MR. SPENCER: Yes. So, the helical coil this is, the supports have the, work on the helical coil section of it.

But at the, where it intersects with the steam and feedwater plenum you can kind of see on the drawing on the left-hand side here there is a straight section, a straight leg section.

MEMBER MARCH-LEUBA: You need to look at the microphone or he can't hear you.

MR. SPENCER: Okay. There is a straight leg section down at the feedwater plenum and at the steam plenum. That's a transition from the helical coil to a straight tube.

That varies in length for each tube. But

it's typically on the order of 20 to 30 inches at least on the feedwater side.

MEMBER MARCH-LEUBA: So, you have like 20 inches of straight?

MR. SPENCER: Yes.

MEMBER MARCH-LEUBA: Good. That's good information to have.

MR. SPENCER: Yes. And I do want to note, we can actually just probably go to the next slide and I'll do the IFR.

I did bring a prototype inlet flow restrictor as well. Now this one is a prototype so it's a little bit longer than the one you'll see on the screen which is representative of the actual design.

Notably, this has eight sections and the actual design has five sections. This also doesn't have the threaded connection that will thread it onto the plate.

But it is kind of -- it's prototypical so you it would allow you to get a feel for it.

MEMBER MARCH-LEUBA: That's not proprietary, the design?

MR. SPENCER: No. Not in this form without dimensions and such.

MEMBER MARCH-LEUBA: The dimensions are proprietary. But the number of stages is not proprietary.

MR. SPENCER: Right, right.

MEMBER MARCH-LEUBA: Okay.

MEMBER RICCARDELLA: I note that from this model that tubes can slide axially. Is that true in the actual model?

MR. SPENCER: That won't be necessarily true in the actual model because the helical coil will be constrained on all sides.

But what I did want to mention here with the five, with the set of five expansions you'll notice that the IFR is contained within the actual tube sheet.

So, it doesn't extend out past the, it doesn't extend past the tube sheet into the heated area.

MEMBER MARCH-LEUBA: What is the tube sheet?

MR. SPENCER: Yes. So, it's not as long, the --

MEMBER MARCH-LEUBA: So, this is outside of the primary? It's not in contact with the primary fluid?

MR. SPENCER: That's correct. That's correct.

CHAIRMAN SUNSERI: So, when you said the tube straight piece is 30 inches or so is that including the length through the tube sheet or after it passes through the tube sheet?

MR. SPENCER: The straight section from the feedwater transition plenum is probably on the range from 20 to maybe 35 inches overall. And then that does include the length of tube which is, which passes through the tube sheet and is welded on the secondary face of the tube sheet.

I think I can say that it's probably not proprietary to say that. That's on the order of six inches is the thickness of the tube sheet.

MEMBER MARCH-LEUBA: So, just so I can visualize it. Where the IFR is inserted that is not a tube but is a stronger piece of material?

MR. SPENCER: It is, it's a tube that's passed through a hole. So, there's a six inch thick metal plate.

MEMBER MARCH-LEUBA: So, it's a thick metal plate with drills.

MR. SPENCER: Yes, with the appropriate -- the OD of the tube would be drilled through. The tube

is inserted into the tube sheet. It's hydraulically expanded.

So, it's pushed out with force up against those walls. And then it's, there's a fillet weld on the end of the tube on the secondary face.

MEMBER MARCH-LEUBA: So, it's welded at the bottom?

MR. SPENCER: So, in this drawing here it would be welded in between the IFR mounting plate. And you'll see there's clouding on that second side. That's to allow a similar metal weld.

MEMBER MARCH-LEUBA: The IFR is held in place from the back on the, with a screw?

MR. SPENCER: Yes. So, there's an IFR plate that all these, each IFR is inserted into the plate. It's mounted through a threaded section through the plate.

Ideally that's going to be a loose design when it's inserted into the tube so that it will allow each IFR to be seated into the tubes. That plate will be mounted through various mounting studs to the actual tube sheet.

That will prevent any sort of bowing or flexure of that plate. And then once all that is in position then those IFR, then the IFR threads

themselves will be tightened up and preloaded.

MEMBER MARCH-LEUBA: And you do this every refueling, to load it?

MR. SPENCER: Yes. This will be --

MEMBER MARCH-LEUBA: So, you loosen the screw in the back for every one of them and then put them in?

MR. SPENCER: Yes. Not for every refueling but for every inspection.

MEMBER MARCH-LEUBA: Yes, right. Every time you take it apart.

MR. SPENCER: Yes. And it may be during a steam generator inspection you may be doing 100 percent inspection of the tubes. You may also be inspecting some smaller number of the tubes based on the steam generator program that the utility sits on.

MEMBER MARCH-LEUBA: But all the IFRs are on the same plate?

MR. SPENCER: I'm sorry.

MEMBER MARCH-LEUBA: All of the IFRs are on the same plate --

MR. SPENCER: Yes.

MEMBER MARCH-LEUBA: -- for each entrance?

MR. SPENCER: Yes.

MEMBER MARCH-LEUBA: You have four of

them.

MEMBER BALLINGER: What is the orientation of the flow restrictor, is the left-hand end the furthest end of the tube sheet?

MR. SPENCER: The furthest end of the tube sheet is the tip, yes.

MEMBER BALLINGER: Yes. So, is there any concern about vibration there? It's a very short, it's a sharp V on the thing.

Is there any concern that you might have a wear problem on that point there because that's on the hydraulically expanded part?

MR. SPENCER: Yes.

MEMBER BALLINGER: So, is there possibility of this thing doing this?

MR. SPENCER: So, we've done a significant amount of testing with respect to forward flows, flows in the nominal direction from the feedwater into the tube at velocities, we've done prototypic testing where we're looking at Reynolds numbers that are much higher that we would expect and the turbulent buffing that we've looked at and any sort of vibration that we've looked at has not been a cause for concern for the IFR.

MEMBER MARCH-LEUBA: But that's assuming

no oscillations, no flow oscillations, correct?

MR. SPENCER: That's, so, yes. That explicitly has been forward flow on the IFR.

MEMBER MARCH-LEUBA: For 100, 120 percent nominal flow, not 300 percent nominal flow?

MR. SPENCER: I want to say that we've gone up to like maybe 800 percent flow in our testing.

MEMBER MARCH-LEUBA: On the --

MR. SPENCER: In the forward direction.

MEMBER MARCH-LEUBA: Vibration testing?

MR. SPENCER: Yes, prototypically. Not at temperature and pressure. But --

MEMBER MARCH-LEUBA: And this thing is screwed into a plate on the back, right?

MR. SPENCER: Yes.

MEMBER MARCH-LEUBA: Yes, a Phillips screwdriver. Hopefully you torque it the right position, you don't do it like I do?

MR. SPENCER: Yes. Well, it will be a hardware design that will prevent loose parts. So, we wouldn't want to have loose parts from this. But it will be, so it will be --

MEMBER MARCH-LEUBA: You have 1,200 of these. One of them after ten years is not going to get a little loose and go ping, ping?

MR. SPENCER: Well, so again these would be removed and, these would be considered to be a part of the Steam Generator Program. So, they will be inspected at the same frequency at which the tubes would be inspected as a part of that Steam Generator Program.

So, when the IFRs are removed they will, you know, any time that you return a threaded part to service part of your procedure in doing that is to look at the condition of the threads, at the condition of the mounting hardware to ensure that it can be put back into service safely.

MEMBER MARCH-LEUBA: And I assume you look inside the tube sheet to look for wear?

MR. SPENCER: Yes. So, there's 100 percent volumetric inspection of the tubes from the inside. So --

MEMBER MARCH-LEUBA: I'm told from the people that know about this that this particular alloy scratches easily. Is that correct?

MEMBER BALLINGER: I don't know about scratch easily. But its wear characteristics are much different than Alloy 600.

MEMBER MARCH-LEUBA: It creates oxide, you scratch the oxide, it creates oxide, you scratch the

oxide.

MEMBER BALLINGER: Now I have one more question. Is there any thought to having an ejection collar on one of those things?

What I'm saying is it would be a pretty bad hair day if the nut on the outside, if it were to fracture there and this thing ended up going into the tube.

But if it was designed so that there was a diameter change in the plate if the nut cracked it wouldn't be possible to send that thing into the tube.

MR. SPENCER: Yes, yes. So, we've done some preliminary test analysis. I guess, I mean the current design that we're here to present today is the current design for the DCA.

You know, we do -- as we change operation, if we change operationally in the future we're going to also be required to change this as a function of that to ensure that we have the same characteristics to prevent DWO that the inlet flow restrictor is designed to do.

So, if we change the operation that affects the design and that allows us to reexamine the design. But the current design that we're presenting today doesn't include that feature.

But we have done some preliminary stress results. We'll present those in the closed session a little bit to show that, you know, we think we have sufficient margin to, any sort of ASME, you know, any sort of ASME analysis on the thread or on the bolt or anything like that.

I think I've presented this slide kind of overall. If you have any questions about it otherwise I suggest we move on.

CHAIRMAN SUNSERI: You just move on and we'll stop you.

MR. BRYAN: One thing that is different from the last time we were here, we got a lot of feedback. We went back and evaluated it.

And we are now proposing a COL item to address the evaluation methodology. And so, I'll pause there just a minute and let you read the COL item.

But this is what we proposed to address developing a methodology that would evaluate the secondary side instabilities including reverse flow.

MEMBER MARCH-LEUBA: If I'm reading correctly you will ensure you have a validated tool that will be able to predict instabilities and what happens during them and how then to calculate the ASME

loads if they should happen. Is that what you're saying?

MR. BRYAN: Yes, correct.

MEMBER MARCH-LEUBA: And that will be a COL item?

MR. BRYAN: Correct.

MEMBER MARCH-LEUBA: Can we say carveout in the open session?

MEMBER KIRCHNER: That has a different meaning.

MEMBER MARCH-LEUBA: I know but, okay, maybe we'll wait for the -- yes, but can we talk about that?

CHAIRMAN SUNSERI: Why don't you wait until the staff --

MEMBER MARCH-LEUBA: All right. I wanted to see what the difference is. But we'll wait for the staff to tell us what the difference is.

CHAIRMAN SUNSERI: Would you envision that this is, this methodology be documented on a technical report or a topical report or something? I'm just trying to think of what, how that would get looked at.

MR. HOUSER: Yes, it would be. We would develop something that's very. Yes. It would be documented and available.

It would be much like the methodologies that were developed for the LOCA and non-LOCA topical reports in terms of content. We can get into that in a little bit more detail in the closed session.

MEMBER MARCH-LEUBA: You would issue a topical or a technical report?

MR. BRYAN: It would be technical, I think.

MEMBER MARCH-LEUBA: Yes, that would be more likely.

MEMBER BROWN: But you all developed the other reports. Now you're pushing this off to the COL who has no background in this design other than they have chosen you all as the design document, the design whatever you want to call it.

It's kind of hard to see this guy walks in cold and has to develop all this analysis technology and methodology for a design that they haven't even seen until they decided to go with you. Maybe I'm speaking out of turn.

This just seems to be kind of complicated when you all have spent several years developing your own evaluations and design analyses and topical reports, that's all.

MR. HOUSER: We are continuing to move

forward with development of that ASME scale, that methodology.

MEMBER BROWN: So, whey the COL if you're all doing it and you're all not going to provide it yourself?

MEMBER RICCARDELLA: The timing.

MEMBER BROWN: I understand that. But that's, time is nice. But I'm looking at it from the technical standpoint and the ability to get a, I guess a methodology that it's truly representative of what, you know, the design and what density wave oscillation.

I'm not a thermal hydraulic guy, okay. But I know that's not good.

MEMBER RICCARDELLA: But it's not realistic to assume that there's going to be a COL guy and NuScale is just going to walk away and this COL applicant is going to build the plant all by himself. Come on, Charlie.

MR. HOUSER: That will not happen.

MEMBER RICCARDELLA: That's absurd.

MEMBER BLEY: There's another thing here. Correct me if I'm wrong. If you issue it as a technical, I assume you'll just move into this and you're working on it.

If you finish it and it's a technical report it won't come to the staff or to us until there's a COL applicant. If you issued it as a topical it might come right away for approval.

Am I correct in that assumption of how things could progress?

MR. BRYAN: Yes. In terms of technicals and topicals that's correct.

MEMBER MARCH-LEUBA: But it's not necessary. I mean, you could send a technical ahead of time.

Given the visibility that it has already had you likely will or you will have a visit in Corvallis to go see it, I think.

MR. MELTON: I want to say, it's Mike Melton with NuScale. So, the COL items will be addressed with, you know, people that are technically qualified.

You know, all resources will be applied to make sure that methodologies or NuScale's involvement.

I don't think we need to be concerned about because the design expertise, analysis, you know consultants we'll have the right workforce to make sure that this gets done properly as with all our COL items.

I just want to assure the Committee. We'll make sure it gets done properly. And I think Marty is correct. This does sound like a technical report but I don't know if we've made that internal decision.

But because of the applicability it's probably that direction. If it does go in the form of a topical report it will follow the process.

MR. PRESSON: And in terms of process it's, you know, we have the ITAAC which is tagged to the COL. But this would ensure that methodology is reviewed prior to the ITAAC process. So, it would be captured in the FSAR portion of that.

MR. DUDEK: And just to add, this is Michael Dudek, the Branch Chief for Nuclear Reactors. The COL versus the carveout is really, as you said, a timing issue.

We have not seen or evaluated fully the proposed COL item. Previous to that we had identified a technical open item and that's where we proposed not giving them finality in the role which is AKA the carveout.

So, as we evaluate and go forward we may take that off the table. But as of now it's still an open item and we propose not giving finality through

the rulemaking.

MEMBER MARCH-LEUBA: So, that carveout would be a way to address this technical review, if it's a technical report.

MR. DUDEK: What they do could suffice and take that open off the table. But we have yet to reach that conclusion.

MR. BRYAN: Okay. So, just to wrap up. There is again, we got a lot of feedback. We heard the feedback. We went back for both the staff and the Committee and we revised both 3.9 to include the COL item.

And we also clarified the language in 5.4. There was a lot of discussion about the use of RELAP there. So, we took that discussion out and replaced it.

We thought you would have the errata by now. But that got held up, that you would have seen it before this meeting. But that will be forthcoming in the errata letter to clean up some of the 5.4 language.

So, that's really all we had planned to cover in the open session. We'll get into some more of the details in the closed session.

We know the staff is going to speak to the

carveout from our perspective. As Matthew said, by having successful completion of the ITAAC we have a COL item, we believe this constitutes the basis for NRC determination to allow operation of the facility certified under 10 CFR 52.

MEMBER MARCH-LEUBA: You say operation, you mean certification under 52, right?

MR. BRYAN: Yes.

CHAIRMAN SUNSERI: Okay, thank you. Members, any questions for the presenters?

MEMBER MARCH-LEUBA: I just wanted to put on the record that this is good. I'm happy that you're taking it seriously and we are going to follow through instead of trying to avoid it. So, today I'm happier than I was yesterday.

CHAIRMAN SUNSERI: Well, that's a milestone. Okay. All right, thank you. Let's bring up the staff now.

And as you all are taking the table I would remind you once again this is open. And if we ask any questions that drive us to proprietary information just refrain and we'll address those in the closed session later.

MS. JOHNSON: Good morning, everyone. My name is Marieliz Johnson. I'm the project, not yet.

(Off-microphone comments.)

MS. JOHNSON: Do you hear me better now? So, I'm Marieliz, sorry, Marieliz Johnson, project manager for NuScale the certification application.

Today we're going to present the NRC review of the NuScale steam generator. For the agenda we have the NRC staff review team. We have a brief summary of the review of the steam generator.

And we will go through a summary of the steam generator design issues that are not resolved by the, by certification, by the design certification application. Here's a list of the review team.

And then I'm going to turn it over to Greg Makar to continue.

MR. MAKAR: I'm Greg Makar from the Corrosion and Steam Generator Branch. And I want to briefly review our -- is that better?

I'm Greg Makar of Corrosion and Steam Generator Branch and I want to briefly review our findings on the topics for steam generator materials and Steam Generator Program. And then I'll turn our attention to the incomplete topic of secondary side flow stability.

We found in most cases, except for that one, we found the materials area acceptable. That

includes material selection and the associated requirements, things like the application of the ASME code and fabrication, cleaning, inspection requirements.

The design limits the crevices along the tubes and enables flow along the tubes and we found that important, degradation mechanisms associated with crevices.

The materials will be compatible with the planned primary and secondary environments. And the design provides for primary and secondary side access for inspection, cleaning, foreign object search and retrieval.

Next slide, please. Steam Generator Program we found to be acceptable. It is consistent with the standard tech specs and the industry guidelines.

We say appropriately acceptable because there are some differences in terminology and other aspects of the tech specs that are different for NuScale.

And the inspection program, it's a performance based framework that has some prescriptive elements and it defines tube integrity in terms of the structural and or describes the performance criteria

in terms of the structural and leakage integrity of the tubes.

They have provided a generic tube plug in criterion which is the amount of through wall loss of the tube that you can have before you have to take a tube out of service.

And the COL applicant will submit the and prepare the steam generator inspection program and implement that plan and provide any site specific information which includes their own degradation assessment, their own plug in criterion and timing and so forth. Next slide.

MEMBER BALLINGER: I just had something pop into my head. The standard tube integrity inspection technique is bobbin coil or something like that.

But usually the, it's on the primary side, goes up the primary side. In this case you're going to have to go up the secondary side.

And if the criteria is 40 percent through wall volumetric, right, that's one of the criteria for tube plugging, that volumetric will be on the inside not the outside of the tube. So, is there, that going to work out okay?

MR. MAKAR: Well, the inspection is

looking for any kind of degradation that you could expect according to your degradation assessment for that particular plant.

Some degradation has come from the inside of the tube, some secondary, some --

MEMBER BALLINGER: Cracking is not an issue. But I'm talking about removal of material, volumetric defect on the inside of the tube where the bobbin coil or pancake or whatever you're using goes up.

That's a little bit different, I think, then what you would find in a recirculating or once received generator like in a PWR.

MR. MAKAR: Well, the inspection will be able to detect volumetric on the inside or the outside.

MEMBER BALLINGER: Okay.

MR. MAKAR: As it does now.

MEMBER BALLINGER: Okay.

PARTICIPANT: Are you worried about the coil getting caught up?

MEMBER BALLINGER: You know, I'm not a coil expert. But if it's all of a sudden now you have a 40 percent volumetric defect once you have removed material.

MR. MAKAR: And that, still the most likely place for that is on the outside of the tubes at support structures. But it could be that this flow restrictor if that, you know, we talked about that.

MEMBER BALLINGER: Corrosion on the outside of that type doesn't concern me. You're not going to get any kind of thing because it's on the primary side.

MR. MAKAR: But the support structures are on the, are also on the outside.

MEMBER BALLINGER: Yes, okay.

MR. MAKAR: So, we still need to look for anything they expect on both the inside and outside.

MEMBER BLEY: I hadn't thought about it and it's not an issue here. But in the current designs where the primary is on the inside when you go in to work you've got a lot of streaming coming out of those tubes, radiation streaming.

I wonder if that's going to be different or better this way around. Go ahead, I'm just wondering.

MEMBER KIRCHNER: Proximity to the core of the tube sheets is going to make for a much different situation. In the current fleet the inspection of the PWRs is, like you said, it's

whatever, particulate corrosion, whatever inside the tubes.

This one you're much closer, the structure has been sitting much closer to the core. So, I wonder what activation --

The core is about what, ten feet lower than the start of the steam generator? But that's the difference I see in terms of personnel exposure. They take this, put it in the dry dock and then inspect it. It may be hotter, the material.

MEMBER BLEY: Kind of in general. But in the current ones you have it on the inside of the tubes and you really get a beam kind of coming out of it.

MEMBER MARCH-LEUBA: Activation is neutron flux and very few neutrons are going to make it through 20 feet of water. So, there will be a gamma flux.

But the gamma doesn't activate. In inspection the core will be in a different place.

MEMBER BALLINGER: That's about 20 tenth value layers.

MEMBER KIRCHNER: The other thing is that assuming they keep doing water chemistry, but these are low flows. So, if stuff is going to accumulate on

the primary side it's going to be around the tube sheet entrance on the primary side, if there's crud in other things.

MEMBER BALLINGER: The first inspection will be interesting.

MR. MAKAR: All right, next slide, please. We have determined that this, we have this issue of structural leakage integrity that has not been fully demonstrated.

And that's related to the effective density wave oscillations on tube integrity and also for the method of analysis for the secondary side, thermal hydraulic conditions and associated loads.

NuScale is working to address that topic. And if there are no, unless there are other questions about our Chapter 5 review I'm going to turn this over to Tom Scarbrough to talk more about the secondary flow instability topic.

MR. SCARBROUGH: Thank you, Greg. I'm Tom Scarbrough with the Mechanical Engineer Branch. We had quite a bit of discussions over the past few weeks regarding the steam generator tubes and their integrity.

And after quite a bit of significant interactions, you know, among all the technical

reviewers. There are a number of technical reviewers. You know, there are several chapters that are involved here of this.

And so, after a lot of deliberation we decided that at this point we're going to propose that we specify the structural integrity and leakage, the structural and leakage integrity of the steam generator tubes are not resolved and not receiving finality in the NRC draft proposed rule for design certification.

MEMBER BLEY: I would just interject here. We've had concerns about wear which could lead to two failures.

The PRA certainly has not reflected anything about this phenomena if it exists.

MR. SCARBROUGH: Yes. And you brought that point and that was one of our concerns that we've talked about quite a bit over the last few weeks.

And so, we're going to talk about the specific details of the technical reasons why in the next couple slides. But I'm just kind of telling you what the process is right now.

MEMBER RICCARDELLA: What you said, does that mean carveout?

MR. SCARBROUGH: It's a carveout, yes,

sir. I didn't use carveout --

MEMBER RICCARDELLA: That's different than what the licensee, the licensee was talking about a COL item and an ITAAC and you're talking about a carveout.

MEMBER BLEY: This is a carveout. They've been working independently on this.

MR. SCARBROUGH: Yes. They've been trying to resolve the issue themselves. And they proposed a COL item. We looked at the COL item.

We don't have a technical concern with the COL item. We actually think it's a good thing. But in terms of whether or not we could certify the specific aspects, and this is focused right, it's focused on the steam generator tube integrity.

And, you know, it's not the whole steam generators. And so, but in this focused area we do not feel we had confidence that we could decide on finality for this particular aspect.

MEMBER MARCH-LEUBA: So, from the way you envision the certificate is to have a carveout and a COL. Is that correct?

MR. SCARBROUGH: Yes, yes. In discussions when we had first seen their proposed COL item we said, you know, we have our own process for, you know,

going on carveout.

And the response we received, they felt like the COL item was a good thing, right. It was a benefit to their design in terms of what, how they presented their design certification application.

And we agreed. But it doesn't specifically affect what we're trying to do here with the finality.

MEMBER MARCH-LEUBA: I was going to make another momentous announcement in the fact that I'm happier with the applicant's proposal than with yours.

CHAIRMAN SUNSERI: So, let me ask a question. So, let me so, I guess this doesn't make a big difference.

But would you envision that when a COL applicant comes in and does what the applicant is saying in the COL item for this activity, would that information be sufficient to resolve the carveout?

They would have, I know they would have to license amendment or something like that to get it approved. But is the work that they plan to do for the COL the work that needs to be done to address your safety concerns?

MR. SCARBROUGH: Yes. They're very similar because they, if they plan to demonstrate that

they are not going to have issues with the potential DWO and the reverse flow the first step from this perspective is to develop a methodology that would predict that reliably.

And so, then they would use that. And then, you know, as through and we'll talk a little more about the sections that we have a concern with. But in the design certification they need to have a methodology listed right for all of the various aspects of the design.

And this methodology is not ready yet. And so, once they are ready they will use it to, in combination with probably the ANSYS model to show that the stress and the wear on the tubes are not significantly impacted by the DWO and reverse flow.

So, again that's the first step that the COL applicant would come in and say here is the methodology and this is how we're going to use it to show that we do not have significant wear on the tubes or damage the IFRs.

MEMBER BLEY: I'd like to try something because we haven't dealt with carveouts as such before this time around. It seems to me what we have the applicant has a COL item which will have to be met during the COL.

What you're saying is they're saying what they're going to do. You're just saying we haven't reviewed this yet. We have to review it at the COL time.

MR. SCARBROUGH: Yes, yes. And that's basically what a carveout is. It's, if we did not have a carveout and we didn't mention it at all officially we have finality on all the aspects of a steam generator.

We really don't have the authority to question the steam generator tubes anymore. And so, we're not ready there. We're not there yet, you know.

We still want to review the COL item and make sure the methodology is proper.

MS. PATTON: I think Mike has something to add.

MR. DUDEK: And, Mr. Chairman, just to dovetail into Tom's response is that the COL item is only one small piece of the carveout. I think you'll see that in the upcoming slides is that, yes, they can include the COL item.

And it may address one small piece of the carveout. But that doesn't resolve the larger picture of all of the open items that are included. And you'll see they are included in the carveout.

CHAIRMAN SUNSERI: Yes, I get it. But it does outline the methodology that would take them there, right?

MR. DUDEK: You'll see that's only one small piece.

CHAIRMAN SUNSERI: Yes, okay. All right.

MR. DUDEK: It's like a first step.

MS. PATTON: There is also a little difference in the legal definition between like a carveout versus a COL item. And a carveout makes it very clear that has to be done by the COL.

You know, you can rely on a carveout in making the findings and it's a little bit more limited how much reliance we can place on a COL item. And so, we're still working through that and some of the questions on COL item versus carveout.

So, I don't want to get ahead of that. But some of those differences are what's being considered in this as well.

CHAIRMAN SUNSERI: As Dennis said, we're still learning on this. But when it comes to carveout and I don't like using that vernacular.

But is there any timing issues regarding when a license then would be issued or when a licensee would be able to start operating the plant regarding

a carveout or anything?

MR. SCARBROUGH: Well, they would need to come in -- the COL applicant would come in and address this aspect of the design that did not reach finality as part of design certification.

MEMBER RICCARDELLA: So, doesn't that, I mean a carveout automatically implies a COL item, right? I mean you have to resolve the carveout because it hasn't been, that aspect of the design hasn't been approved.

MR. SCARBROUGH: In words or not, right, of course. And so, the COL item that NuScale is proposing is that first step to resolve this issue that's been carved out, exactly.

MEMBER MARCH-LEUBA: So, while you're making the presentation can you address my bias. I see opposite to what you said. I see that their COL proposal is broader than your very limited carveout.

MEMBER RICCARDELLA: Maybe we need to see the remaining, the additional slides.

MEMBER KIRCHNER: All they have proposed is a methodology. You still have to do all of the analysis and have to do the ASME code case, et cetera, et cetera. It's much more.

MEMBER BLEY: We've only seen their first

slide. Maybe we could look at some more.

MS. PATTON: There are two more slides on the carveout.

MEMBER MARCH-LEUBA: I want you to address my biases while you present it because what I hear here is as long as you satisfy the ASME code, the tube doesn't break, we're perfectly okay with it.

MR. SCARBROUGH: No. We're only on the first bullet on the first slide.

MEMBER MARCH-LEUBA: I have a bias of controllability and moisture in the steam line.

MR. SCARBROUGH: Exactly, yes. We're going to get there. So, this is just --

VICE CHAIRMAN REMPE: To follow up on Matt's question about how to fix things. I can remember with Vogtle that there was, they let them go ahead and pour concrete for some things but not some nuclear construction.

And that was a fuzzy line. When does the carveout have to be addressed? Does it affect what can be done in the construction for a COL applicant?

MR. SCARBROUGH: In this case and we are fortunate we had actually two OGC lawyers helping us with this, right, and so, because this is new ground for me too. This is carved out.

And this only affects the steam generator tube aspect of the design. Everything else goes forward the way it is supposed to go forward.

VICE CHAIRMAN REMPE: Okay. And that would be true for the other carveout too?

MR. SCARBROUGH: Yes.

VICE CHAIRMAN REMPE: It's just limited on that thing, thank you.

MEMBER BLEY: But at the COL stage an applicant could not get a license until these carveouts were fulfilled, reviewed and approved?

MR. SCARBROUGH: Yes. This aspect has to be completed, you know, for the COL applicant to receive the COL.

MS. PATTON: Right. It basically just identifies the portion of the design that wasn't granted finality through the rule, right.

So, it's basically takes a piece that would normally be in a design certification and says the COL when they apply has to provide this additional piece.

MEMBER BLEY: But since this is new to us, one last question. Assuming the Commission issues a design certification that rule would then say the following aspects have not yet been evaluated or

something.

MR. SCARBROUGH: Exactly. We are working with OGC on the exact words. And we're going to sort of show you the words that we're working with OGC to put into the rule itself that will indicate that this specific aspect of the steam generator tubes does not receive finality yet as to OGC license.

MS. PATTON: Right. There's a few lines that actually go directly into the rule and carve it out.

MEMBER BLEY: We have one or more other carveouts that are going on.

MR. SCARBROUGH: I believe there's two other carveouts on different topics.

MS. PATTON: That's why I said, there's a little difference in legal definition between like a carveout and a COL items and a carveout, you know, makes it very clear within the rule that needs to be provided.

CHAIRMAN SUNSERI: Thank you, thanks for taking us on this little detour of the regulatory practice here. Let's get back into the technical presentation. Go ahead, Mike.

MR. DUDEK: Just one more side note. Something that may help is that the legal definition

according to OGC has evolved for a COL item.

The COL items is being used now is more interpreted as just an information tracking item. It doesn't have any legal gumption or enforcement in the COL going forward. So, it's more of an information tracker versus an enforcement item.

MEMBER BLEY: I could get an operating license without fulfilling the COL item?

MS. PATTON: We would have to probably have an attorney answer that.

MEMBER BLEY: I think so. That really sounds bizarre.

MS. PATTON: My understanding is that, my little bit of understanding and, Mike, you can chime in is that there is, more like there could be a potential fight about that a little bit.

And this, a carveout makes it, gives it the force of law.

MEMBER BLEY: Is the authority here. It brings the strength.

MS. PATTON: It's stronger than a COL item.

MEMBER BLEY: We've supported a number of design certs under the assumption all COL items --

CHAIRMAN SUNSERI: Maybe we can take up

that topic at a different meeting. Okay, thanks. Tom, go ahead.

MR. SCARBROUGH: Okay. So, Appendix G is going to be the portion of Part 52 which is the NuScale design certification rule.

And so, there's a section, Section 6, I'll call it issue resolution which will talk about the steam generator tube integrity issue and indicate that it's not resolved within the meaning of 5263 Alpha 5.

And that, I went back and pulled that out. That has to do with all matters all resolved except for 10 CFR 2.335 which has to do with petitions.

So, that's what that has -- basically it's saying that this issue has not been resolved yet for finality for the design certification.

And then there is another section that will be in Appendix G, which is Section 4 which talks about what is the COL applicant responsible for. And it will talk about the fact that the COL applicant needs to provide the design information to address the steam generator tube integrity.

And so, those sort of two sections that we are working with OGC now to get the words just right from the legal perspective to make sure we carve it out to cover the issues but also, you know, it's only

the steam generator tube area aspect that's being carved out.

And so, the rule now, the proposed rule language it's with OGC right now and they're working on it to have it ready for Commission approval. So, that's where we are right now.

So, now Becky is going to walk us through -- there is two specific sort of parts to this carveout. And, but we talk about them separately just because it's easier to keep track of.

So, Becky is going to talk about the first part.

MS. PATTON: Okay. So, currently in the FSAR that NuScale submitted, Section 3912 there's a listing of the computer programs that are used by NuScale for the dynamic and static analyses and for the hydraulic transient load analyses.

So, you know, if you look in that currently it will list, you know, NRELAP, for example, as one of those codes. And then, you know, points you over to 1502 for the code description and the V&V.

And so, you know, my branch in Reactor Systems assisted, you know, with the review of NRELAP for those, you know, mechanical, those blow down loads.

Currently in the FSAR in Chapter 5 it also lists NRELAP as being used for determining the pressure drop in the IFR design to ensure acceptable mass flow fluctuations for power levels, et cetera, et cetera.

Our understanding is that, you know, NuScale has plans to, you know, modify that to clarify that. But basically, that's listing currently of NRELAP in 391 is intended for blow down loads currently.

That's what the staff had reviewed. We hadn't reviewed it for, you know, other loading conditions potentially for DWO.

So, this would be a portion of the carveout to say that 3912 with DWO loads being a potential loading condition you would need to list a method of analysis into 3912 for those loading conditions.

And those presently are not there. So, the carveout would specify that in demonstrating steam generator tube integrity a COL applicant would need to provide information to demonstrate that GDC 4 is met for the method of analysis to predict thermal hydraulic conditions of the steam generator fluid system and the resulting load stresses and

deformations from DWO.

So, our understanding is that NuScale is planning on, you know, adding some, you know, a COL item for one to this section to specify that would be done in the future. We would still, you know, the current plan is that we would still maintain this as part of the carveout.

But that's, basically the first portion would be that method, you know, hasn't been specified and it's integral to the finding in that section made by the Mechanical Branch that all those methods are listed.

MR. SCARBROUGH: Right, exactly. So, that's the first part. So, that would -- that's the COL item sort of section.

Now the other part is the actual steam generator tube integrity issue. And that sort of has been, I've been to the meetings in the past couple months of the ACRS and heard a lot about that.

But the bottom line is NuScale has not provided reasonable assurance that the flow oscillations that occur in the steam generator secondary fluid system will not cause damage to the steam generator tubes directly from DWO or reverse flow or indirectly by possible damage from the inlet

flow restrictors, IFRs where they might vibrate and such.

As you saw, they're kind of a cantilevered process. And NuScale talked about their forward flow testing.

But they really haven't done really much in the other direction to see if there was something that might cause these to have some issues in the opposite reverse flow direction. And so, that's what, the concern we have there.

So, and it sort of -- this issue sort of grew over time because, you know, if you go back to the original Rev 2 of the DCA it indicated in Section 5412 that the flow restriction devices would preclude DWO.

And then there was Rev 3 which came out that said well, there will be oscillations but they will be within acceptable limits. And as we've gotten more interaction with NuScale in terms of what that really meant and what the information was we determined that we weren't comfortable with the amount of degradation that might occur from reverse flow from DWO and such.

And so, based on that our concern is not like one tube failing. Our concern would be if there

was catastrophic failure of a number of tubes could it interfere with the natural circulation process because everything in this reactor relies on natural circulation for cooling.

And so, if you had a significant break of a number of tubes you could disrupt natural circulation cooling either from ECCS system which we talked a lot about this week and also the decay removal system.

You know, both of those are natural circulation processes. So, that was our concern. Until we are comfortable that there won't be this potential for catastrophic failure because there is GDC 4 which is dynamic effects and vibrations and such.

And then there's also GDC 31 which is the fracture prevention of the reactor coolant pressure boundary. And so, and that GDC talks about the fact that you need to have capability to ensure that you do not have a rapid, propagating failure of the reactor coolant pressure boundary.

And if you had a number of these IFRs come loose and go through these tubes you might have a number of tubes that fail at the same time. So, we did not feel comfortable that we had enough

information to be able to say that, yes, this issue can have finality.

And so, as part of this carveout is a specification, and this would be in the rule itself that a COL applicant will need to provide information demonstrating that 10 CFR Part 100, Appendix A, which is the seismic capability aspect and also Part 50 Appendix A, GDC 4 and 31 are met with respect to structural and leakage integrity for the steam generator tubes that might be compromised by these adverse effects from DWO and the secondary fluid system.

But we're going to be very clear in the carveout that these are the areas that we're carving out. You know, we're not carving out the entire steam generators and that sort of thing because we have to make sure that we focus it on what the concern was and what is not receiving finality.

And that's what is happening right now with the rule that OGC is helping us with. So, that is the two sort of technical issues.

So, there's no question, now I was going to have Yuken go through and kind of describe the DWO phenomenon and what's going on with that.

MR. WONG: My name is Yuken Wong. NuScale

had performed the TF-2 tests mainly for thermal hydraulic performance of the steam generators. These tests are also used for flow induced vibration purpose.

The TF-2 specimen had five columns of tubes with 250 tubes in total. And one column of tube with 52 tubes was used for the density wave oscillation tests.

Density wave oscillation was observed during the TF-2 testing with temperature and flow oscillations in the secondary cooling. The DWO frequency was low and will not excite the steam generator tube structural resonances. Based on the TF-2 strength gauge measurements, the staff estimate that the alternating stress intensities will be below the ASME fatigue endurance limits.

However, any differences such as geometry, material and operating conditions between the TF-2 and the actual as built steam generators have not been evaluated.

As discussed on the next slides the staff is concerned about the potential impact of the density wave oscillation on the steam generator tubes directly and indirectly by the inlet flow restrictors.

MEMBER RICCARDELLA: Excuse me. Could I

ask, those strain gauges that you show on the previous slide, were they on the inside or the outside of the tube?

MR. WONG: They are on the outside of the tubes.

MEMBER RICCARDELLA: Okay. So, they would pick up pressure oscillations. But if there were any thermal gradient effects that would only occur on the inside. It might not, you might not see it on the outside, right?

MR. WONG: They pick up the strains as well.

MEMBER RICCARDELLA: Not if there was a, if there was a thermal gradient and thus a strain gradient through the thickness of the tube it would not, you know, when you do a thermal shock on a component you get higher stresses on the inside than on the outside.

That's a fairly thin tube. But you still might have some through wall gradient.

MR. WONG: The tubes are very thin. And from the --

MEMBER RICCARDELLA: I understand.

MR. WONG: -- what the data indicates it does pick up the strain in this subset. They suspect

some of the strains at the thermal oscillations.

The steam generator in the flow restrictors are designed to provide the necessary pressure drop to limit density wave oscillation in the tubes.

As explained earlier, the flow restrictors are mounted on the mounting plate and inserted into the steam generator tubes. NuScale performed in the flow restrictor, excuse me, leakage flow instability tests for the conceptual design of the inlet flow restrictors.

The staff did not identify any concerns for the test for the normal flow or forward flow. However, these tests did not include density wave oscillation conditions as the forward flow.

NuScale has selected a final inlet flow restrictor design that is similar to one of the tested designs. And NuScale will perform validation testing for the final inlet flow restrictor design after design certification.

Next slide, please. Unstable density wave oscillation can cause reverse flow to the inlet flow restrictors including subcooled liquid from modest density wave oscillation or slug and two-phase flow for strong density wave oscillation.

NuScale has not yet evaluated potential impacts on steam generator tubes and inlet flow restrictors for reverse flow such as fatigue of bolted joints and loose inlet flow restrictors.

The concerns due to leakage flow instability cantilever the inlet flow restrictors unless stable under reverse flow conditions. Also, due to cyclic pressure drops and high speed turbulent two-phase flow through the inlet flow restrictors.

The concern also includes cavitation erosion of the steam generator tube walls and wear of inlet flow restrictors and the tube walls that can further worsen density wave oscillation.

MEMBER BLEY: Excuse me. Two related questions. When you say they're less, the flow restrictors are less stable under reverse flow conditions, what do you mean by that?

And my second question is I'm envisioning this thing maybe going back and forth a little bit. And can these screws back out? I've seen screws back out in vibrating situations.

And if they do I guess that flow restrictor is free to either flow out or go forward.

MR. WONG: Literature indicates when a cantilever structure, when the flow is going from the

support end to the free end it's more stable.

MEMBER BLEY: So, it's this kind of vibration that you're talking about?

MR. WONG: Correct.

MEMBER BLEY: Yes, that makes sense.

MEMBER MARCH-LEUBA: When the flow is going forward you're pulling. When you're pushing. The pushing is much more -- when you're pulling it straightens out.

When you're pushing it moves towards the wall, right.

MEMBER BLEY: That makes sense if that is what you're talking about.

MR. WONG: Yes, yes. And if the screws --

MEMBER BLEY: Let me, I've looked at these things and I kind of assume that you've got a lot of turns on that screw that hold it in place. But that screw is long enough to go through that plate.

I don't know how many turns you get. So, I'm -- the idea that a screw could back out might not be crazy.

MEMBER RICCARDELLA: It's preloaded, you know.

MEMBER BLEY: Yes. I know it's preloaded. But now you're jerking it back and forth.

MEMBER RICCARDELLA: Yes, but, you know, theoretically if the preload sustain you don't get oscillatory loads on a preloaded bolt. That's why you preload bolts.

MEMBER BLEY: But you preload them under assumptions and this assumption wasn't there.

MEMBER RICCARDELLA: Yes. You preload and there's also, typically there's something that keeps it from backing out like in LWR internals they use some sort of retainer device or something to keep it from unscrewing.

CHAIRMAN SUNSERI: NuScale said that there would be, you know, loose parts prevention measures applied, right. So, if that's what you're talking about.

MEMBER RICCARDELLA: Yes. But that doesn't, if you just contain it as a loose part like you would put a cap over it that doesn't keep it, that doesn't ensure that the preload is maintained. It could still lose preload.

It, you know, they're going to be doing a lot of work in this area obviously. That's detailed design work that has to be done.

MR. SCARBROUGH: Right. And that's, they're going to have to finish, you know, the design,

pick the final design and then qualify the design.

So, there's still quite a bit of work to do to address your issues that you're raising.

MEMBER RICCARDELLA: But theoretically, if it's properly preloaded you won't see oscillatory loads.

MEMBER BLEY: And one would think after this testing and analysis a consideration of reverse flow would be part of that preloads.

MEMBER RICCARDELLA: Yes, for sure.

MR. SCARBROUGH: Okay. Next slide, please. So, where do we go from here, okay?

Assuming that the design certification rule is issued, the COL applicant will be responsible to address the steam generator tube integrity in its COL application and it has these sort of two parts that we talked about.

One is the method of analysis that they have a COL item that's going to make sure the COL applicant knows they have to submit that. And then the second part will be demonstrating that the tubes will not be damaged by DWO directly or by, or indirectly by the IFRs vibrating and things of that nature causing some damage.

So, the COL applicants will be responsible

for demonstrating that in the process of receiving its COL. So, that's going to be a review that the staff will do.

And this will all come back to the ACRS for you all to take a look at as well. And then assuming that COL is issued, the next step will be a COL holder.

And there's a number of aspects that the COL holder is responsible for. There are ITAAC related to the ASME Boiler and Pressure Code, Section 3 requirements.

But there also, in addition to that there is the Comprehensive Vibration Assessment Program, the CVAP which Yuken reviews quite a bit in terms of the review for applicants.

And there's specific aspects. There is some additional testing. The TF-3 referred to as TF-3 testing that has to be done. There's also vibration testing that's specified in Tier 2 in Table 14.272 that had to do.

So, they have that to do. And plus they're going to have some instrumentation on, for the initial start of a steam generator.

So, the COL holder has quite a bit of work to do as well after that phase of receiving the COL.

So, that's the process after the design certification to make sure this issue is fully reviewed as part of the next step after design certification.

And then Becky is going to talk about next steps.

MS. PATTON: Sure. NuScale is currently preparing errata to the Revision 4 of the DCA. And you saw part of that with their proposed COL item that they presented earlier.

They are also, you know, preparing some other changes potentially to clarify some of the steam generator secondary fluid flow issues that could impact the tubes, the IFRs, some of the various statements, you know, made in the associated chapters.

So, we have prepared drafts for the proposed rule. And it discussed the steam generator tube integrity, the issue as a whole. It includes the method of analysis and as Tom mentioned, the portion of the carveout related to integrity of the IFR and the tubes.

So, the draft proposed rule would exclude both aspects of that issue from finality and will, basically what will happen is a COL applicant would have to provide those portions when they apply for the COL and then that's when the NRC staff would perform,

you know, that review.

Except, I think as noted they could, you know, put a topical report or something together on the method, you know, that could come in ahead of time. If it's a technical report it would typically come with the COL.

But either way the COL would, you know, fulfill that by either referencing like an approved topical report, you know, or providing the technical report.

So, other aspects of the steam generator design are considered acceptable to staff. Those would be granted finality but not the ones specifically identified in the carveout.

MEMBER MARCH-LEUBA: Okay. And that's where my earlier comment was. Apparently the staff is not concerned about controllability and operability of the steam generator?

MR. SCARBROUGH: That issue is, we consider, we separated. The design certification focuses on the reactor aspects. The COL applicant still will need to come in and talk about the secondary side, control and things of that nature.

But just from a design certification perspective we focused on is there a potential impact

on the reactor safety. And our concern was that if there was catastrophic failure of a number of tubes that could affect reactor safety.

And so, that's how we separate it. We haven't, it's not that we're not concerned about it. We just have put that over into the COL application review.

MS. PATTON: Right. The carveouts are linked to what the findings are that the staff has to make at the design certification stage specifically.

So, you know, you can as a finding right that he has to make on that IFR, for example, you know, show it doesn't fall apart and somehow impact the integrity of the tubes or fail to perform its function and therefore you could, you know, have oscillations impacting that.

The controllability of the plant, whether or not there are any issues with that, you know, I think if I remember correctly I believe the control system like gets, you know, that gets designed later.

I think there's a COL item on some aspects of the MPS control system. So, those are things that would be looked at, you know, at the COL stage.

You don't need a, you don't use a carveout for that.

MEMBER MARCH-LEUBA: Control and protection system will determine, you're still dumping moisture in the steam line that becomes an issue too.

MS. PATTON: Right. But, so some issues have to be, you know, looked at as part of the design, their design, the findings that need to be made, you know, under the regulations.

And so, those where you can't make them it's a carveout.

MEMBER MARCH-LEUBA: So, you have not made any finding about the controllability and operability of the secondary side?

MS. PATTON: No. The control system is part of --

MR. SCARBROUGH: That would be a COL item, COL application review not for design certification.

MEMBER DIMITRIJEVIC: Okay. Up to now I look at this as operability issue. I did not think it was a safety concern because of your putting, they don't call it steam generator tube rupture but steam generator tube failure. Now when you bring the safety concern isn't that too big to carveout because you cannot even make conclusion that this plant meets safety goal?

With this carveout you cannot make

conclusion in Chapter 19 that this plant is meeting safety goal.

MR. SCARBROUGH: Well, we're carving out just the aspect of the steam generator tube integrity aspect.

MEMBER DIMITRIJEVIC: Yes. But this is a risk steam generator to fail. That leads to loss. So, you are carving safety concern which can impact conclusions about safety of this plant. How can you do that?

So, by making it a, well by making it a carveout for one you're putting it directly in the rule. So, the COL applicant will have to demonstrate that IFR, you know, does remain intact, doesn't, you know, cause damage to the tubes, right, performs its function.

That is ensured to have to be demonstrated by the COL applicant by carving that out specifically. So, that's what we would expect.

MEMBER DIMITRIJEVIC: But then your Section 19 cannot make conclusions that this plant meets safety goal until that's proved. Just, I just want to say that.

Until this is proved by COL applicant we don't know that this plant meets safety goals.

MR. SCARBROUGH: The COL applicant will have to demonstrate this to be able to receive permission to load fuel. So, they're going to have to --

MEMBER DIMITRIJEVIC: No, no. I understand. But I just say the second sentence in Chapter 19 is this plant meets safety goals with badging, blah, blah, blah.

That's not true anymore. It won't be true until they prove that in the COL.

MR. SCARBROUGH: We have interacted with OGC on how this process works. And according to their legal opinion you sort of carve that, this very narrow focus out when you make that decision.

So, we're going through the process of OGC of what carveouts work. And so far they've indicated that this focused carveout is acceptable from the perspective of you can proceed with design certification with this carveout.

So, that's sort of where we are with the process.

MEMBER DIMITRIJEVIC: You know, if you think that this is safety concern, you know, it would be tough to agree with that, that you can proceed having such a big safety concern.

MR. SCARBROUGH: Okay, well thank you. I'll relay that back to OGC and make sure we're on good legal ground. Thank you.

CHAIRMAN SUNSERI: Okay. Any other Member comments?

VICE CHAIRMAN REMPE: Okay. Real quick, this has changed in the last few weeks. It's been changed again because we might have done a letter this week and how confident are we in the material that we've only seen in slides?

MR. SCARBROUGH: Well, in terms of the carveout I think we're pretty comfortable. We have OGC agreement on how the carveout works and how it's very focused on this specific aspect.

So, we're comfortable with this aspect. We don't plan to, this has to go to the Commission of course and they have to, you know, sign out the rule. But we do not plan to have any changes at this point in terms of how the carveout.

And it's very consistent with the slides you've seen in terms of the wording. The discussion in the rule is very short.

It's very similar to what is in the slides because OGC says you just have to focus it and make sure you that you carve out a very narrow, specific

concern that you have.

So, we don't anticipate any changes. But it does have to go to the Commission for their approval.

MS. PATTON: Right. I mean, the Commission, you know, review of the proposed rule always happens afterwards anyway.

VICE CHAIRMAN REMPE: So, just trusting and wanted to kick the tires and make sure. Thank you.

MS. PATTON: Right. I mean obviously feel free to weigh in one way or another because you're always before the Commission. Bob had --

MEMBER BROWN: Yes. You zipped right through something where you said changes in the MPS, Module Protection System. What --

MS. PATTON: No, I believe that's, I'm sorry I may have misspoke.

MEMBER BROWN: I was hoping you were, okay.

MS. PATTON: I believe it's the control system.

MEMBER BROWN: Okay. You're talking about the control system for like feedwater control or something like that.

MS. PATTON: But my understanding was the control system actually has like a COL item on it.

MEMBER MARCH-LEUBA: I said trip but I meant protection of equipment and protection of --

MEMBER BROWN: She used the words, the acronym MPS when she zipped right through a comment earlier.

MS. PATTON: Yes. I meant to say control, MCS.

MEMBER BROWN: Module Protection System is, has nothing to do with this.

MS. PATTON: No.

MEMBER BROWN: Thank you for the clarification.

MR. CALDWELL: This is Bob Caldwell. I'm the deputy director of DNRL. I just want to make sure. But we cannot make a safety finding based on a COL item.

We can't say the design is good or bad based on the COL item. It is a tracking item. However, COL items must be addressed during the COL application where we do a review, basically the same SRP type review of what's actually being built with all the final design details in it.

So, we actually look at it before a plant

will ever be built for that. So, a carveout is very specific. It's very focused. It's on one of the findings.

We have multiple findings during our DC review and the certification. So, they are findings by regulation. I'm not familiar that we ever make a finding that the plant is safe.

We say that the plant meets the regulations and that all the regulations are satisfied with the exception of an aspect of a regulation. So, we're very comfortable with the COL carveout, excuse me, the carveout process.

We're also very comfortable with the COL items. But we can't make a safety finding that the regulations are met based on a COL item.

MEMBER BROWN: So, you're confirming Member Dimitrijevic's comment that you can't give a firm basis that it meets the safety goal until, that's why you're saying later? That's what I heard you just say.

I'm sorry, I didn't talk to the mic. Vesna noted that how can you give a, say you meet the safety goals, I forgot what the words are, okay, as part of this rulemaking.

You have to, part of it's being deferred

because of this until the COL applicant completes whatever is necessary on the steam generator design issue. And you all will be reviewing it at that time.

You made a comment you can't make a firm commitment that it meets it until you finish this and that's going to be delayed. I'm just trying to confirm what Vesna said that I got it, that first of all they kind of waved their hands.

And you're saying well, she's really kind of right. That's the way I --

MEMBER RICCARDELLA: I'm not a policy person. But the rulemaking says hey, it meets the safety goals in everything except for these specific areas in which are carved out.

MR. CALDWELL: That's correct.

MEMBER RICCARDELLA: That's not a big deal.

MEMBER BROWN: I didn't say the rule was --

MEMBER DIMITRIJEVIC: There is three things core damage large release and conditional containment which this will impact significantly. So, those are three safety goals that come from the PRA perspective.

So, I mean that much we don't know. That

would be me. And, you know, this is not a carveout for the hydrogen, you know line. You are carving out a big part of the thing.

I mean, you know, it's not really small item like we were discussing yesterday the hydrogen and, you know, line. So, I mean, I really, you know, I am really, I am not comfortable with this.

MR. SCARBROUGH: Okay, well thank you. We'll go back and talk to OGC and make sure that we're on --

MR. CALDWELL: Let me just make it clear. Excuse me, this is Bob Caldwell again. For the items of which we determine finality they meet the safety goals.

For the items that we have not reached finality on we do not say one way or the other. But for everything that we have reached finality on we have, we believe we meet the Commission's safety goals.

MEMBER BROWN: But you won't have finality on this?

MR. CALDWELL: We won't have that on that before we actually get the review on the COL for that one aspect.

CHAIRMAN SUNSERI: And the plant won't

operate until they do.

MEMBER BROWN: So, that part I understand. But you need to know how to say, no.

CHAIRMAN SUNSERI: All right, Members, any other, I'm sorry, Tom, anything else?

MR. SCARBROUGH: No, we're good. Thank you.

CHAIRMAN SUNSERI: Members, any other comments or questions for staff while we're in the open session?

MEMBER MARCH-LEUBA: Let me put something on the open session. Certainly I like better the approach of the applicant than your approach in the sense that I believe, and this is a belief of religion if you want, that the output of that process will be ending up more validated so we will know for sure whether we are unstable or not.

And we will make the changes that will be necessary to the plant so that we won't be unstable at 100 percent flow. That's what I believe the output of the COL process will be.

And I love it. As I said before, I'm getting tired of winning. So, thank you very much.

CHAIRMAN SUNSERI: Okay. At this time I will ask any Members that are in the room that would

like to make a statement please come up to the mic and do so.

While we're doing this, Mike, can you get the public line open?

MR. DUDEK: I will. But just to clarify for members of the public that are on the line we are going to closed session in order to protect proprietary information to the NuScale design is the reason that we're going as announced earlier in the meeting that we can go to closed session to protect proprietary information.

We will reopen the line for public discussion or for the public to participate at 1:00 p.m. this afternoon when the open session will begin again. Thanks.

CHAIRMAN SUNSERI: Anybody in the room?

MR. DUDEK: This is Michael Dudek. I just have one additional comment to add on to what you said, Jose. It's not one or the other.

I think you're going to get both. So, I think you're going to get NuScale's proposed design fixes and you're going to get the carveout. So, that's just the extra regulatory assurance.

MEMBER MARCH-LEUBA: Well, let me reiterate, I'm happier today than I was yesterday.

CHAIRMAN SUNSERI: No comments from the room. So, I'll turn to the phone line. Any member of the public on the phone line that wishes to make a statement please state your name and your comment.

All right. We're going to close the phone line. And at this point we have reached the end of the open session. We're going to take a 15 minute break.

We're going to reconvene at 10 after ten in a closed session with NuScale presenting first. We are recessed until 10:10.

(Whereupon, the above-entitled matter went off the record at 9:53 a.m. and resumed at 1:03 p.m.)

CHAIRMAN SUNSERI: All right, we are reconvening the meeting now. We will start with NuScale in open discussion to begin with the -- lost my -- rod ejection accident.

MR. PRESSON: Matt Sunseri?

CHAIRMAN SUNSERI: Matthew, you all are ready to go?

MR. PRESSON: Yeah, thank you, and good afternoon. Appreciate you all taking the time to hear from us on these topical reports today. I'm Matthew Presson, Licensing Project Manager for NuScale Power. And we are going to be discussing the evaluation

methodologies for rod ejection accidents, loss of coolant accidents, and non-loss of coolant accidents.

The presentations provided today are identical to the presentations we gave to the Subcommittee on February 19, so we'll be moving through them at a pretty quick summary level today. But for interested members of the public, when the transcripts for that February 19 meeting come out, there will be a fair amount more detail there.

That being said, while we'll be giving a summary, if you have any questions, feel free to interrupt. And we have our engineers listening in on the phone or one of them here at Rockville, so let us know.

CHAIRMAN SUNSERI: All right, thank you.

MR. PRESSON: Next slide. All right, so slide 2. For our first presentation on the rod ejection method, it'll by myself up here, and Kenny Anderson is supporting from Corvallis as our Nuclear Fuels Analyst. Next slide.

For slide 3, I did want to spend a minute on this just to re-scope, given our week of discussing DCA and FSAR topics here. This slide provides us with a high level map of the technical and topical reports, which develop the methods needed for Chapter 15 and

other related thermohydraulic sections.

Today we will be looking specifically at rod ejection LOCA and non-LOCA. And while these do support the NuScale FSAR, the results of these as applied to the FSAR design are presented in Chapter 15. Our discussions today will be focused on the separate licensing submittals for these methods.

All right, for our agenda, our presentation will cover a quick summary of the event, our acceptance criteria, our expectations against future reg guides, especially DG-1327, a flow chart of the method, how we initialize and evaluate our events. And then a quick summary of that method again.

For slide 5, we discuss why we look at a separate method for rod ejection and for meeting our GDC-28 commitments. And it provides a couple of examples on why it's unique insofar as Chapter 15 events, such as its focus on nuclear physics instead of thermohydraulics, where that spatial focus is. Postulated causes, and definitely acceptance criteria, which we will also discuss on slide 6.

This slide 6 is another summary table providing information on which acceptance criteria are more unique to the rod ejection event than the rest of Chapter 15 events. For the NuScale method, most of

those acceptance criteria are covered by our method requirements to preclude fuel failure, there we go, and that's part of that footnote down at the bottom of the table.

Our next slide, while not applicable to the current method or FSAR DCA design, discusses why we feel pretty comfortable in meeting future proposed criteria for pellet clad interaction. As it is not current criteria, we do not have a full evaluation showing this. But as no exposure is credited in our rod ejection method and as M5 cladding is less susceptible to those interactions in general, we are confident that we won't be challenged when those criteria are revised.

So for slide 8, we are looking at a flow chart that shows an overview of our method, how we moved from SIMULATE5 to SIM-3K. And then eventually split it out to look at our peak RCS pressure, our MCHFR, and our fuel temperature and enthalpy requirements.

For slide 9, that's a very summary discussion, but it does provide some of the information for how we initiate and set up our steady state assumptions and evaluations. We use SIMULATE5 to set up the core response. SIMULATE5 is covered in

our nuclear analysis codes and methods qualification.

And our design does include the assumption bounding potential for an ejected assembly to damage adjacent assemblies, which has been discussed in terms of our FSAR design. I believe, if my notes are correct, that we and the NRC intend to follow up with that during DCA discussion in the April full committee insofar as the DCA design. For the scope of this method, it is simply an assumption that is built into those initial conditions.

Slide 10. Slide 10 shows how we build on from that steady state initialization and move into our dynamic response. SIM-3K is used to model the transient and what's benchmarked to demonstrate a combined neutronic, thermohydraulic and fuel time modeling capabilities. So the slide also lists some of the primary uncertainties that were applied for the simulations.

Slide 11 discusses how we move into our CHF evaluation, where we use VIPRE-01. This was originally demonstrated to be appropriate for our design in our subchannel analysis methodology.

There are some unique differences in this application versus that original topical report, such as smaller axial nodalization, case-specific radial

power distributions, couple of the other bullets seen there. And to that point, we evaluated additional sensitivities to holistically justify those changes.

For slide 12, to insure against our fuel heat-up criteria, we include a hand calc, which takes a adiabatic approach, including total energy generated by a SIM-3K, and runs that through either as a temperature or energy increase. Those values are compared against NRC-developed acceptance criteria. And some example values are included in the Subcommittee closed session slides from February 19.

Slide 13 looks at the first side of our dynamic system response. So we covered CHF in the previous slide. Our first type of dynamic response that we look at is our CHF evaluation. It takes a transient response and provides those system thermohydraulic conditions over to VIPRE for a subchannel evaluation.

Next slide, 14, discusses a quick summary of our second dynamic system response, which is looking for pressurization. For that we, it's a little bit different scenario. We are looking for something that raises the power quickly up to just below those high power and high power rate trip setpoints, and let it go for as long as it takes

before it trips the core.

So then from there we calculate the peak system pressure and compare that against our acceptance criteria.

So a very quick presentation, but in summary, we have a conservative analysis method for our unique rod ejection accident, at least in terms of Chapter 15 events.

And the topical report provides details and justification for software tools and acceptance criteria used, the applicability of the method and those tools, the appropriate treatment of uncertainties, and the results of this application of the method by input to our DCA FSAR Chapter 15. So.

MEMBER KIRCHNER: I do have one question. I didn't bring the slides from the previous Subcommittee meeting, but I thought on the slide for fuel that shows the figure fuel enthalpy rise versus oxide wall thickness, you drew a box in within the lefthand figure that you were using for your acceptance criteria.

You mention the next-to-the-last bullet, the upper limit that you were using, so I think I can say that. I was curious, I don't remember how you chose a point on the abscissa on oxide wall thickness.

Is that a proprietary number?

MR. PRESSON: I'll have to ask Kenny if that was a proprietary value.

MEMBER KIRCHNER: Yeah, I --

MR. PRESSON: But it was based on not needing to basically ever take credit or advantage of any of the space after you pass that point, so.

MEMBER KIRCHNER: So that was a box that you drew as your acceptance criteria.

MR. PRESSON: Yeah, that's correct.

MEMBER KIRCHNER: For the actual NPM, right?

MR. PRESSON: Yup.

MEMBER KIRCHNER: I'll go back and check on whether that was an open slide or a closed. But again, the basis for that was that that was the estimated maximum oxide oxidation you would see?

MR. PRESSON: Correct. And Kenny, if you're available --

MEMBER BALLINGER: That's number's a widely used number.

MEMBER KIRCHNER: Okay.

MEMBER BALLINGER: The one that they use, so.

VICE CHAIRMAN REMPE: So I forgot that I

needed to declare that I might have a conflict of interest in certain aspects of this discussion on this particular methodology and limit my participation in such discussions and deliberations.

CHAIRMAN SUNSERI: Noted.

MR. PRESSON: Kenny, are you available to chat? Because I do believe that value is open information, it just didn't show up on the slide.

MEMBER KIRCHNER: Okay.

MR. PRESSON: Yeah, you can talk right now.

MR. ANDERSON: Hi, this is Kenny in Corvallis. Yes, that number comes from our assumed or calculated maximum corrosion. And it, I think it is on the slide, but perhaps it's not showing up in the presentation.

MEMBER KIRCHNER: Yeah, okay. Thank you.

MR. PRESSON: Yeah, I'm 99% sure it's not, so. All right, that is the end of our presentation, so if there are any questions.

CHAIRMAN SUNSERI: Any members, comments, questions on rod ejection? All right, then we're done. Did that one.

MR. PRESSON: All right. Are we presenting this? Yeah. Good?

CHAIRMAN SUNSERI: Yes, when you're ready.

MR. PRESSON: All right, so next presentation will be a similar summary fashion. And again, same slides as before. This is the, our presentation on our NuScale topical report, loss of coolant accident evaluation model.

So here we have myself, Matthew Presson, on the line we have Dr. Pravin Sawant, a Supervisor of Code Validation and Methods. We also have Dr. Selim Kuran, who is our Thermohydraulic Analyst. And Ben Bristol, our Supervisor of System Thermohydraulics.

Slide 3 provides a quick overview of our agenda. We describe a very summary version of our methodology, provide a reference slide for our NPM safety systems. There were the four elements of our LOCA topical report and the PIRT, our assessment base. The evaluation model for NRELAP5, and our applicability evaluation. And we discuss how we extend the LOCA evaluation to an IORV event and end with conclusions.

So slide 4, little bit of background on the NPM and the LOCA. Some of the unique features involve our integrated design, which eliminates a lot of piping and limits potential breaks. Coolant is captured completely in containment, cooled and

returned to the reactor pressure vessel using a large pool.

Our regulatory requirements that we use to build our method are, well, that we used to make sure our method met, was the 10 CFR 50.46 acceptance criteria. And we looked to maintain maximum PCT at steady state with no clad heat-up. To meet those for our evaluation method, we used conservative LOCA acceptance criteria. These are figures of merit that the core remains covered, and therefore it collapsed liquid level over the top of active fuel.

Our MCHFR is greater than our CHFR limit of 1.29, and our containment pressure and temperature are below the design limit.

For slide 6, this provides kind of a roadmap for how we take those acceptance criteria and develop them out into a method. So we start with our 10 CFR 50.46 requirements. We then process that using Reg Guide 1.203. And we develop that into our LOCA PIRT Element 1. Use that to develop our assessment base for separate effects testing and integral effects testing.

Move on to Element 3, where we developed the evaluation model. And finally, with Element 4, we use all the prior elements to assess that adequacy.

Slide 7 is just a quick picture for reference in case any information is needed, but it provides a lot of information about our safety systems, kind of how they're oriented. All right.

Slides 8 and 9 get us into Element 1 of our PIRT process. So there we assessed our relative importance of phenomena. We would recognize experts and NuScale subject matter experts in our PIRT panel. And we targeted those figures of merit, CHF, collapsed level above top of active fuel, and containment pressure and temperature. That when we used rankings in importance and knowledge to see where we needed to focus our, any evaluations on.

It was a result of that for slide 10, we developed this understanding of phases, Phase 1a blowdown, Phase 1b ECCS actuation, and Phase 2 flow reversal at RRVs. For LOCA, we focus on Phase 1a and 1b. We move onto long-term cooling for Phase 2.

All right, yeah, for slide 12, it goes into how we develop our NRELAP5 code. We use RELAP5 3D, version 4.1.3, as the baseline code. We maintain a code configuration control and development consistent with NuScale's NQA-1 2008 and 2009 NQA program.

And some of the specific modifications we

made for NRELAP5 were to consider NuScale specific components such as our helical coil steam generator. Make sure that we met those regulatory requirements from earlier and apply error corrections as they're determined.

Slide 13 is a very high level, but we did want to point out that we have a fair number of tests spanning our integral effects testings and separate effects testing. And for slide 14, we present our NIST-1 facility, where a large portion of those tests took place. It's the primary source of our NuScale-specific test data, and it includes a good number of design features that look to scale and provide information for our LOCA and non-LOCA events.

All right, so for our NuScale LOCA model overview, we look into the analysis and justifications of why we use NRELAP5, what we need for time-step controls, how we set up those boundary conditions, and how we maintain and treat setpoints and trips. We also take a look at the LOCA break spectrum and dig into the methodology of sensitivity calculations.

Those are required by Appendix K, they are phenomena-specific, and we use them to establish a conservative bias.

For Element 4, our applicability

evaluation, we took both the bottom-up and top-down approach. For the bottom-up approach, we identified the dominant models and correlations for the hydraulic phenomena, it's in table 8-1 of the topical report. Identified a lot of key parameters and reviewed those models and correlations. Again, a lot of that is in Chapter 8.

For the integral performance, the top-down portion of it, we reviewed the codes and evaluated the integral performance of those codes using those integral effects test data. And we compared that test data to NRELAP5 scalability via scaling and distortion analysis. And we note those differences and distortions between the NPM and NIST and look to see how we can account for them using NRELAP5.

So our conclusions for the LOCA method is that there are a number of conservatisms built into it. We have both as much from 10 CFR 50, Appendix K, as is applicable to the NuScale design. And we look to make sure that those other unique considerations are considered by other methodology conservatisms.

We developed this using the cycle independent bounding LOCA analysis. It is supported by an extensive experimental database. A lot of those new to NuScale using this one, as well as several

others. Applicability evaluation is consistent with Reg Guide 1.203, and we maintain -- we look to maintain those figures of merit.

So CHF is not challenged, our collapse level in the reactor remains above the top of active fuel. There is no clad or fuel heat-up, and our pressure and temperature remain below design limits.

And the next slide, slide 20, really slide 21, go into how we kind of extend our LOCA into IORV space. So we're looking to kind of evaluate liquid space, RRV and steam space, RVV and RSV discharge. And these are fairly similar transients to the LOCA.

From that, we followed a very similar process as our LOCA, developing the method. And yeah, next slide. On slide 22, we account for a couple of the differences. The main difference is our key acceptance criteria, our MCHFR limit moves to 1.13 and 1.37.

And our conservatisms are the same as LOCA, but with the following exceptions. That we remove an additional 15% bias in fuel. We have our limiting axial power shapes and radial peaking based on subchannel analysis. The Moody choked flow model for two phase is applied to the initiating valve, and the initial conditions are biased to minimize MCHFR.

So on slide 23 we come to similar conclusions. IORV as an extension of the LOCA method. Maintains its own PIRT assessment and applicability within the LOCA. The minor method differences mainly account for the AOO classification of that.

And MCHFR occurs early within that transient, and then rapidly rises, given the flow-to-power ratio. So our primary concern there, that the collapsed liquid in the RPV does remain above the top of active fuel.

MEMBER MARCH-LEUBA: The MCHFR occurs early but does not violate the limit, right?

MR. PRESSON: Correct.

MEMBER MARCH-LEUBA: Because the way you have it written, I said, wait a moment.

MR. PRESSON: Yeah, well, and it is still the minimum, or maximum but it does not violate --

MEMBER MARCH-LEUBA: I know exactly what you mean, it can be misinterpreted.

MR. PRESSON: Yup. And that is our LOCA presentation.

CHAIRMAN SUNSERI: Members, any comments or questions for NuScale? No? All right. So you may proceed with the non-LOCA,

MEMBER BLEY: It just strikes me that if

I were listening in, I would think we have no interest. But we had a Subcommittee meeting on this where we delved into the associated issues in great detail.

CHAIRMAN SUNSERI: That's a good point, and we had good, full Committee participation at those subcommittees as well.

MEMBER MARCH-LEUBA: And it was two days ago.

CHAIRMAN SUNSERI: Yes.

MEMBER MARCH-LEUBA: So that's why we're so quiet, because this is just a pro forma presentation.

MR. PRESSON: Yeah, two days ago for Chapter 15 and two weeks ago for the original Subcommittee for this. But those transcripts aren't up yet.

CHAIRMAN SUNSERI: But it's important to get it on the record for public --

MR. PRESSON: Yeah. There was a good full day of conversation on this.

VICE CHAIRMAN REMPE: Good, huh?

MR. PRESSON: I would say so, yeah. Hey, it's nuclear industry, we value a questioning attitude.

All right, and our final presentation for this afternoon is on the Non-Loss of Coolant Accident Topical Report. Again, presenters are myself up here, as well as Megan McCloskey, who is our Thermohydraulic Analyst. We have Ben Bristol on the line, who is our Supervisor of System Thermohydraulics, and Paul Infanger, our Licensing Specialist is in the audience as needed.

So for slide 3, we go over our outline where we, just the outline of the presentation. We give a scope of the non-LOCA LTR as compared to other Chapter 15 events, as well as other FSAR events. We discuss those non-LOCA events that are covered in the method. We discuss the development of our non-LOCA method and give a general overview of how we perform those analyses and look at a couple of specific events.

So slide 4, discussing scope. Our non-LOCA method does look at NRELAP5 system transient analysis of non-LOCA events. It looks at that interface to subchannel and accident radiological analysis. And goes over the short-term transient progression with DHRS cooling. So what is out of scope for the non-LOCA method is the SAFDLs, which are evaluated and downstream subchannel analysis, with its

own topical report.

All of these out-of-scope items are either captured in topical reports or technical reports. Also includes accident radiological dose analysis, control rod ejection, which we already covered, as well as LOCA, and those IORV events. Peak containment pressure has its own technical report. And the long-term transient is covered in the long-term cooling technical report.

So our non-LOCA evaluation method is applicable to the following events. We covered cooldown events, heat-up events, reactivity events, inventory increase and inventory decrease. Most of these are fairly standard events for Chapter 15, but a couple of unique ones for NuScale giving our design our loss of containment vacuum and containment flooding. As well as the heat-up event of an inadvertent operation of DHRS.

A quick overview of non-LOCA event acceptance criteria. This table presents those criteria in general, so for the minimum critical heat flux ratio and the maximum fuel center line temperature, you'll note that both of those point to the Footnote 1, where we, that was pretty much as collapsed down to the same AOO acceptance criteria.

And here we have a zoom-in on how our non-LOCA method interacts with those other topical reports. We have our, you developed a design, you look at the events. Our non-LOCA methods covers the system thermohydraulic response. That then passes that information on to VIPRE for subchannel analysis, looking at CHF.

And then mass and energy releases from the thermohydraulic response and other inputs are looked at in our accident radiological analysis, which is bounded by our accident source term topical report.

MEMBER MARCH-LEUBA: And this might be relevant for some other topic, but not every single transient evaluated within RELAP gets evaluated with VIPRE.

MR. PRESSON: Correct.

PARTICIPANT: You use screening criteria.

MR. PRESSON: Yup.

MEMBER MARCH-LEUBA: Say two words about it?

MR. PRESSON: Yeah, we'll actually cover that on a later slide, but that is correct, yeah. For slide 8, we look at our margin to acceptance criteria. For non-LOCA, we are looking at MCHFR. Primary pressure, secondary side pressure, radiological

release, and establishing those safe, stable conditions to pass on down.

For slide 9, the evaluation method development follows a fairly similar path as our LOCA. Followed the same Reg Guide 1.203 process in developing the graded approach. Element 1 is looking at establishing the applicable transients and acceptance criteria and to create that non-LOCA PIRT.

Elements 2, 3, and 4 leverage a fair amount of information from LOCA, but it definitely does focus on the differences between high ranked phenomenon, well, the differences between the LOCA and non-LOCA high ranked phenomena, make sure that we have additional NRELAP5 code validation performed to focus on, for example, DHRS and the integral non-LOCA response.

Slide 10 covers the results and what was considered in our non-LOCA PIRT, including the general categories of event types, the SSCs that were considered, as well as the phases that are part of our non-LOCA, our pre-trip transient, our post-trip transition, and finally Phase 3 of stable natural circulation.

Slide 11 gives a quick summary of NRELAP5's applicability for non-LOCA. As mentioned

before, there was a KATHY analysis performed to determine how to address those high ranked phenomena, looking to see what validation was still applicable, as taken from the LOCA evaluation model and adding additional validation and benchmarks for non-LOCA.

That also looked to our conservative inputs and make sure that we had suitable subchannel analysis established. Sorry, yeah.

Overall conclusion is that the NRELAP5 code with the NPM system model is applicable for calculation of the NPM non-LOCA system response, so.

Slide 12 goes over that analysis process. Topical report section 4, where we develop that plant base model. We adapt it as needed for the specific events. You perform you steady state and transient calculations within RELAP5, and you evaluate those. You confirm your margins to RCS pressure acceptance, steam generator pressure acceptance criteria.

And you, this kind of goes to your point earlier, you identify the cases that you look to examine further with subchannel analysis and extract the boundary conditions as applicable. So we're looking conservative bias directions of maximal reactor power, core exit pressure, core inlet pressure, minimum RCS flow rate.

And the NRELAP5 CHF calculations for non-LOCA may be used as a screening tool to assist analysts in determining limiting cases to be evaluated in that downstream subchannel analysis of that CHF. It's not itself used for those non-LOCA events.

So, and 6, you look to identify if any applicable radiological analysis needs to be performed.

MEMBER MARCH-LEUBA: How do you identify the step 6, what do you use as criteria?

MS. McCLOSKEY: For the events with downstream radiological analysis, we look at the system transient response and which cases have the maximum mass release, which would carry the radioactivity and increase the dose. And the maximum iodine spiking time between reactor trip and isolation of the break.

MEMBER MARCH-LEUBA: But if all your analyses show no clad damage, what do you do?

MS. McCLOSKEY: Is the question why do we do it, or what do we do?

MEMBER MARCH-LEUBA: What do you do if you run all of your transients and none of them results in clad damage? So your core is intact.

MS. McCLOSKEY: We still pass the boundary

conditions to the radiological analysis, and they use an appropriate source term based on, I think, and I am not radiological analysis analyst, you've got tech spec limits on fuel failure rates and normal operating coolant that can be --

MEMBER MARCH-LEUBA: So you assume normal operation failure rates, and that is what is gives you the source term.

MS. McCLOSKEY: Again, I'm not an expert on the radiological analysis of what they used for the source term.

MEMBER MARCH-LEUBA: Okay, I don't remember, but that sounds familiar.

MS. McCLOSKEY: But there are source terms that are evaluated.

MEMBER MARCH-LEUBA: It was like one -- yeah.

MR. PRESSON: From tech spec concentration, just got a note, so.

Slide 13 looks at our general methodology and event-specific methodology. In general we're looking at steady state conditions, our treatment of plant controls, loss of power, single failure, making sure we have bounding reactivity parameter input. And then bias the other parameters as needed. And we also

look at operator action as needed.

For the event-specific methodology, we then dive a little deeper into the description of the event initiation and progression. And we make sure we appropriately scope for the acceptance criteria of interest and target limiting single failures, the loss of power scenarios, and whether or not we need additional sensitivity calculations. The initial condition biases and conservatisms that already existent, or if we need, again, to perform more sensitivities.

And then tabulated representative results of those sensitivity calculations. So, and those sample analysis results are provided in Section 8 of the non-LOCA method.

So, for conclusions, slide 14. Our non-LOCA system transient evaluation model is developed following that graded approach we discussed in accordance with guidance provided in Reg Guide 1.203. It applies to NPM-type plant design, natural circulation water reactors with helical coil steam generators and an integral pressurizer.

NRELAP5 is used to simulate those systems thermohydraulic responses to demonstrate primary and secondary pressure acceptance criteria are met, and

that safe and stable conditions are achieved. And system transient results provide the boundary conditions that are then passed down to our subchannel methods and radiological analyses.

And that concludes our non-LOCA.

CHAIRMAN SUNSERI: Members, any questions or comments for NuScale?

MEMBER MARCH-LEUBA: Not today.

CHAIRMAN SUNSERI: Okay, well, good, we appreciate the recap and the presentation. So at this time we can transition over the staff for their comments.

So as the presenters are taking their seats, I'll turn to Rebecca and ask if you have any overarching remarks that you want to make at this point.

MS. PATTON: No, just thank you.

CHAIRMAN SUNSERI: Because I skipped you earlier today. Okay, so Bruce, are you the lead here? All right, well, whenever you're ready.

MR. BAVOL: All right, good afternoon, everybody, my name is Bruce Bavol, I'm the Project Manager on the NuScale project. This afternoon from the NRC staff we're going to be talking several topical reports, the first being rod ejection, the

second being loss of coolant accident analysis, and the third non-loss of coolant analysis.

To my right, Chris Van Wert will be leading the rod ejection. We're, since we've talked a lot about these topics, I'm just going to move right into the staff review and turn it over to Chris.

CHAIRMAN SUNSERI: Yeah, and I think similar kind of comments. I mean, there's not, you don't have to read every bullet on the slide, we're well versed in the topic to hit the high points and the important message that you want to leave us with.

MR. VAN WERT: All right, good afternoon, this is Chris Van Wert. And since we're jumping here into the review, just want to point out that what is included and not included within the review, we did look at the criteria and the methodology as a whole, as well as the assumptions that went into it.

And it's worth noting that the analysis itself for the DCA is not part of this review, that is handled separately under the Chapter 5 staff evaluation report. It's also worth noting that the staff did audit calculations and other supporting information during its review.

As far as the analysis criteria itself, we did look at the RCS pressure, fuel cladding failure,

core coolability, and fission product inventory. And we did determine that they either followed the guidance provided in SRP 4.2's Appendix B, or were conservative compared to it.

And as we discussed during the Subcommittee, it was also not part of the staff's review, but we were cognizant of the draft guidance that's out there in terms of revised guidance for rod ejection accidents.

And we did compare the two to see where NuScale fell within it. But again, since that's draft guidance, that wasn't a criteria that they had to follow. But they were conservative in regards to either criteria.

So next was the evaluation of the code suite. In terms of rod ejection, they used CASMO5 to SIMULATE5, you know, SIMULATE-3K and RELAP5 and VIPRE. Most of those, with the exclusion of SIMULATE-3K, were already reviewed and approved as part of another topical report, the nuclear analysis codes and methods, so that was not part of this review.

However, SIMULATE-3K was unique to this and the validation was contained within it, so the staff's review did cover it. And we did determine that they successfully demonstrated that they could

use it properly and get accurate results.

MEMBER MARCH-LEUBA: Has SIMULATE-3K been licensed by any other vendor or facility?

MR. VAN WERT: So SIMULATE-3K has been used in licensing actions and has been reviewed by the staff. It has not been submitted by Studsvik as the standalone methodology topic report. So there's no generic, yeah.

MEMBER MARCH-LEUBA: But any licensee or vendor?

MR. VAN WERT: Licensees have submitted.

MEMBER MARCH-LEUBA: Some licensees use it?

MR. VAN WERT: Yeah.

MEMBER MARCH-LEUBA: Okay, good.

MR. VAN WERT: For plant cycle, this attribute did include plant cycle assumptions used by NuScale. And in general, they included ranges and power and cycle time and range of operating conditions and show that they used limiting conditions.

The staff also agreed that the assumptions in terms of the automatic systems response of non-safety systems were conservative, and that the methodology regarding timing of loss of AC power conservatively biases the RCS pressure evaluation.

The staff reviewed the methodology itself, including how information is passed between the different codes, the uncertainties, the modeling assumptions, and the handling of reactor trips. And in conclusion, the staff determined that they were conservative and that the methods were acceptable for demonstrating compliance with the acceptance with acceptance criteria.

And in conclusion overall, the staff concludes that the criteria used for evaluating REA either follows or is more conservative than the staff guidance, and that the methodology accounts for various potential operating conditions in time in life and conservatively addresses uncertainties in plant conditions.

The staff therefore finds the use of this topical report acceptable for evaluating reactivity-initiated accidents from the NuScale plant design.

And if there are any questions? And if not, pass it on to Shanlai Liu.

CHAIRMAN SUNSERI: Members? No, all right. Continue on.

DR. LU: Okay, Shanlai Lu from the staff, NRR.

Okay, right away jumping to the -- okay,

the review team might have skipped that, so you already talked about that one.

So the design features of course that you guys have already gone through one. Very simple design, there are three reactor vent valves on top of the reactor vessel, two reactor, you know, return valves. And then containment functions as a part of ECCS.

So the scope of this topical report of course is number one, it's to underline the LOCA. And then as a part doing part of the review process, they extended this topical report to cover the IORV methodology. And as part of it, it also supports the peak containment pressure and non-LOCA topical report and non-term cooling analysis models.

Applicable regulation for LOCA of course 10 CFR 50.46. They decided to use Appendix K, which does give them some flexibility to reduce the number of runoffs that don't have to do the best estimate a whole bunch of statistical sampling. Okay, next slide.

The review approach, and we did take an early engagement and, so that we can -- we conducted extensive audits, all the way to, you know, a couple months before this some presentation. And because of

that effort, and then we only identified a total number of 13 RAIs, which is 45 RAI -- but through the process we resolved 210 other issues.

And those were really resolved based on extensive staff sensitivity studies and based on NRELAP5 confirmatory analysis with TRACE, thanks for our research support.

And the primary and, you know, scope of this review is a focus on LOCA and a non-LOCA too. And related to IORV. So the review area number one is PIRT. And based on the staff's review, we conclude that the PIRT process they had followed the CSAU methodology.

And we used NRELAP5 code, which is a derivative of NRELAP-3D, which has been used extensively before. But they did add additional NPM special features. We went through all each features before.

And in order to confirm and then benchmark the code, they conduct the extensive testing which lasted a very long time, actually, more than ten years. And then they also performed the scaling distortion analysis. We reviewed that one, identified the issues, and they did additional testing. And, which resolved the issues there too.

And as part of IORV analysis methodology, and then we dived very deep into the actual CHF correlations by our staff. And what is used for low and low flow and high flow conditions, including the STERN and the KATHY facility specific fuel databases they used for AOOs. So those are review areas we covered. Next slide.

And as I mentioned that we did extensive staff confirmatory analysis, which covers the separate event test and the integral effect test, extensively on the NIST models itself. And we used both TRACE and a RELAP5 code, and more than 55 sets of calculation were performed.

And because of all the effort, we were be able to resolve a lot of the, you know, audit issues. So we can zoom in to the RAIs, like total questions are only 45. Those are the confirmatory analysis.

Based on the review, we concluded at the end NuScale LOCA EM model. And RELAP5 version 1.4 approved for determining critical heat flux and collapsed liquid level for NuScale NPM in compliance with 10 CFR 50.46 key requirements.

And the code is, can be used to determine the peak containment pressure, but with the limitation that they have to apply certain specific peak

containment pressure analysis criteria there. And the CHF model is approved, subject to limitations and the conditions for low flow and the high flow conditions.

So with that, that's the conclusion of staff's presentation on LOCA topical report. All right.

CHAIRMAN SUNSERI: Comments from members? All right, Alex, your turn.

MS. SIWY: Is this the one that doesn't work?

CHAIRMAN SUNSERI: Yeah, I'm sorry, use the one to your right.

MS. SIWY: Okay, all right. My name is Alex Siwy and I'm a Technical Review in the Reactor Systems Branch in NRR. To provide a basic summary of the staff's review process, we conducted our review of the non-LOCA topical report in accordance with the applicable NRC regulations and guidance. Our SER is based on Revision 2 of the topical report.

The staff conducted audits similar to what was done for LOCA, two audits in four different phases that covered different topics. We examined about 140 different issues as part of the audits, and overall, the audits really helped to confirm the staff's understanding of the docketed information and to

inform RAIs.

In total, we issued 33 RAI questions, and to date all of these have been resolved and responses have been incorporated into the topical report as appropriate.

So this slide covers the scope of the non-LOCA methodology, which NuScale covered well in their presentation. I think the thing that I would highlight here is that some of the items that are discussed in the topical report the staff is not making conclusions on as part of the topical report review, because we feel that those items are more appropriate for a design-specific application of the methodology. These include items like the limiting loss of power assumptions and single failures.

One of the major areas of staff review were the key design features and models that would be particularly relevant for non-LOCA event analysis. The staff reviewed things like the natural circulation design, the helical coil steam generator models, the DHRS modeling, and the fact that the evacuated containment vessel produces the potential for a new type of event.

The staff also extensively reviewed the applicability of NRELAP5 to performing non-LOCA

transient analyses. As the applicant discussed, they developed the non-LOCA EM based on the LOCA EM using a grade approach. The staff reviewed the applicant's non-LOCA PIRT to ensure that the important phenomena were identified and appropriately captured in the non-LOCA topical report.

And the staff reviewed how the applicant addressed each of the highly ranked non-LOCA phenomena, which included methods such as separate and integral effects tests, code-to-code benchmark, use of bounding input values, as well as other analysis methodologies.

Related to this topic was one significant issue that we encountered as part of our review. In particular, the staff requested additional justification for how multidimensional flow effects in the RCS and thermal stratification in the reactor pool are addressed as part of the non-LOCA EM. The staff's major concerns on this topic were the potential for reduced RCS flow rates, as well as degradation in DHRS performance.

To summarize, the applicant's RAI response resolved the issue, as was confirmed by the staff audit of underlying calculation notes, as well as audit discussions with the applicant.

The staff reviewed each of the NRELAP5 assessments against test data presented in the non-LOCA topical report, as well as a couple that were presented as part of the LOCA topical report. And overall, the staff finds that the KAIST, the NIST HP-03 and HP-04 tests served to validate the NRELAP5 DHRS models.

The SIET TF-1 test validated the steam generator secondary side phenomena, but the staff had some concerns about the ability of the SIET TF-2 test to fully validate primary to secondary heat transfer.

The NLT2A, 2B, and 15P2 integral effects test together demonstrate the applicability of NRELAP5 to evaluate non-LOCA transients. And the benchmark against RETRAN-3D provides confidence that the NRELAP5 point kinetics model with the thermohydraulic feedback produces results that are consistent with those of an NRC-approved code.

There were a couple of significant review issues related to the assessment against NRELAP5, or assessments of NRELAP5 against test data.

In particular, the applicant removed steam generator and DHRS heat transfer biases from the methodology in response to staff questions about the steam generator heat transfer uncertainty based on the

SIET TF-2 concerns that I mentioned on the previous slide. And this was associated with the DCA Chapter 15 UOI, as well as concerns about DHRS nodalization.

To address these concerns, the applicant provided justification that non-LOCA figures of merit are not sensitive to these biases. And based on its review of the justification, as well as audits of the underlying calculations, the staff finds that the removal of the DHRS and steam generator heat transfer biases is supported for NPM model Revision 2.

But we did impose a related limitation and condition because some of the sensitivities were specific to the particular design at hand.

The staff also reviewed the general and event-specific non-LOCA methodology. Overall, the process for analyzing non-LOCA events, including the interfaces with other methodologies, provides an acceptable analysis framework. The staff also finds that the deterministic approach using conservative or bounding inputs, initial conditions, and assumptions is acceptable for conservative calculations of non-LOCA events.

In addition, the staff reviewed each of the event-specific methodologies and concluded that the application of those methodologies will ensure

conservative results.

And finally, the staff reviewed the representative non-LOCA event calculations in Section 8 of the topical report and concludes that they adequately illustrate how the non-LOCA methodology can be applied to conservative transient analyses.

This slide just summarizes the limitations and conditions found in the staff SER. I won't go through them line by line, but there are six different limitations and conditions.

And in conclusion, the staff finds that all technical issues from the course of the review have been resolved and that the use of NRELAP5 with the non-LOCA methodology described in the topical report is acceptable for the non-LOCA safety analyses of the NuScale NPM design, subject to the specified limitations and conditions.

CHAIRMAN SUNSERI: Very good, thank you. Members, any questions or comments?

MEMBER KIRCHNER: I just would like to thank NuScale and the staff for their very good presentations during our February Subcommittee meetings and their excellent short summaries today. Thank you.

CHAIRMAN SUNSERI: Any other comments?

All right, so we'll ask if there are any members in the room that would like to make a comment. And while we're doing that, if we can open up the phone lines for public comment.

MR. PRESSON: Hey, Matthew Presson with NuScale. I wanted to confirm for you that the 100 mil corrosion limit is indeed non-proprietary. So good to use.

MEMBER BALLINGER: It's only a 100 -- it's a 100, not 80?

MR. PRESSON: That is what was emailed to me, yes.

MEMBER BALLINGER: Okay. All right. Eighty has been around for the last 15 years or 20 years.

CHAIRMAN SUNSERI: All right, there's no comments from the room, so we'll turn to the phone line. If there is a member of the public that is on the phone line that wishes to make a comment, now is your opportunity. Please state your name and provide your comment.

MR. LEWIS: Marvin Lewis.

CHAIRMAN SUNSERI: Okay, Marvin, we'll take yours first.

MR. LEWIS: Wonderful, thank you. Look,

it sounds reasonable -- saying that it's going to mean that the reactor will operate without problems. But at least the verbiage sounds good. I do have a question, mainly about density waves fluctuate -- density wave oscillations.

When you get the water hammer and everybody runs out of the nuclear power plant, how do you know there's going to be enough people left to handle the -- resume without emergencies, thank you.

CHAIRMAN SUNSERI: Thank you for your comment. Ms. Fields, I think you're next.

MS. FIELDS: Yes, this is Sarah Fields. I brought this up at the NuScale Subcommittee meeting a few days ago. I do not understand how the NRC will be finalizing the draft rule and submitting it to the Commission on March 19, which is two weeks from now. And then the NRC intends to publish rulemaking effort on June 1.

There's still a few things to iron out between the ACRS, NuScale and the NRC that have been discussed over the past few days. The ACRS won't finalize their -- or submit their final letter until June 23, I believe. And then the NRC staff won't finalize the SER until November. And yet the NRC appears to be going ahead with this rulemaking as if

all the T's have been crossed and the I's have been dotted, which they haven't.

So I think the NRC's schedule for this rulemaking is rather premature. Also, there really is no rush. The prospective COL applicant, the only prospective applicant, is the Utah Associated Municipal Power Systems, or UAMPS.

The type of reactor that UAMPS intends to construct and operate would have 25 more percent power than the current NuScale design. Therefore, UAMPS must wait until the NuScale -- after NuScale submits its standard design approval application, which would include that 25% power increase, before they could submit their COL application to the NRC. And the NuScale SDA application's not expected until the latter part of 2021.

So basically, there really is no COL applicant out there who will be submitting an application specifically referencing this design certification. So I just wanted to put that out there. I think that the public should be able to wait until all ACRS and NRC staff documents related to this design certification are complete before the rulemaking. Thank you.

CHAIRMAN SUNSERI: Thank you. Any other

members of the public on the phone line that wish to make a statement? Okay, we will close the phone line at this point, thank you. And we're at a transition point here. Let me poll the Committee here. Do we see the need for a closed session to talk to staff or NuScale about any proprietary information?

MEMBER MARCH-LEUBA: I recommend that we go into closed session to read the letters for proprietary content, so NuScale can tell us they're not proprietary. And then we go back to open session to discuss them.

VICE CHAIRMAN REMPE: But we should be all done with the transcriber.

CHAIRMAN SUNSERI: Yeah, we can do that --

MEMBER MARCH-LEUBA: Off the transcript.

CHAIRMAN SUNSERI: Yeah, off the. Well, we're going off the record anyway at this point in time. So I think we'll proceed along that, those lines. Walt, is that okay with you?

All right, so we are going to go off the record at this point in time. The next time we will be on is at 10:45 tomorrow morning when we'll look at the biannual review of the Nuclear Safety Research Program.

We are going into closed session now for

report writing --

VICE CHAIRMAN REMPE: Matt, say again what you said. We're not going to have any more transcribers, right, for the rest of this session or this meeting? Because we're not going to need a transcriber for that or for P&P. P&P's public, but --

CHAIRMAN SUNSERI: Well, I don't know about transcribers, I'm just talking about open session.

MEMBER MARCH-LEUBA: We'll stay have a transcriber. You need to put your microphone on.

VICE CHAIRMAN REMPE: P&P is open.

CHAIRMAN SUNSERI: Okay, we are going closed.

(Whereupon, the above-entitled matter went off the record at 2:07 p.m.)

February 28, 2020 Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Titled "ACRS Full Committee Presentation: NuScale – Steam Generator Design," PM-0220-69051, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) NuScale Full Committee Meeting on March 5, 2020. The materials support NuScale's presentation of the NuScale steam generator design.

The enclosure to this letter is the nonproprietary presentation titled "ACRS Full Committee Presentation: NuScale – Steam Generator Design," PM-0220-69051, Revision 0.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Marty Bryan at 541-452-7172 or at mbryan@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Robert Taylor, NRC, OWFN-8H12
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Bruce Bavol, NRC, OWFN-8H12

Enclosure: "ACRS Full Committee Presentation: NuScale – Steam Generator Design," PM-0220-69051, Revision 0

Enclosure:

"ACRS Full Committee Presentation: NuScale – Steam Generator Design," PM-0220-69051, Revision 0

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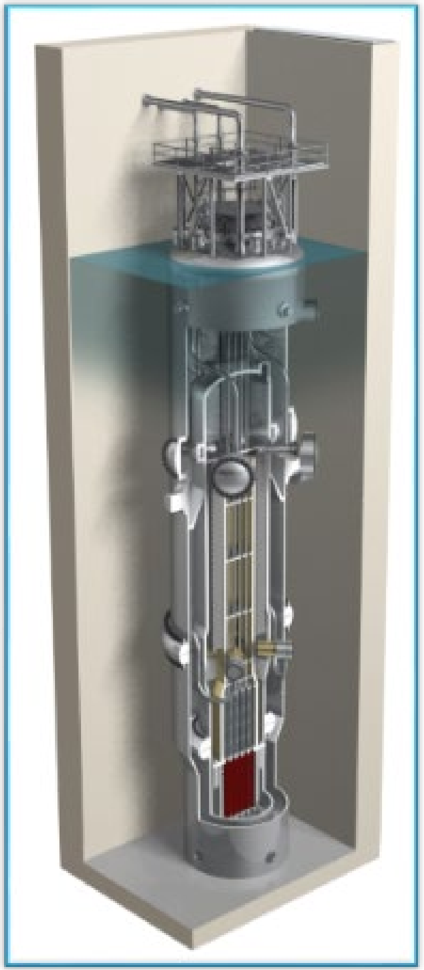
ACRS Full Committee Presentation



NuScale

Steam Generator Design

March 5, 2020

Presenters

Kevin Spencer
Engineer, NSSS Engineering

Bob Houser
Manager, Testing and Code Development

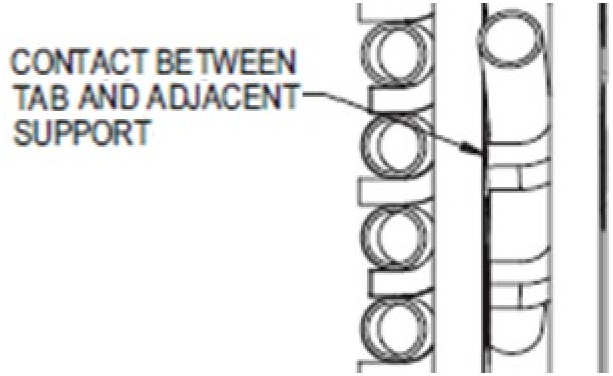
Brian Wolf
Supervisor, Code Development

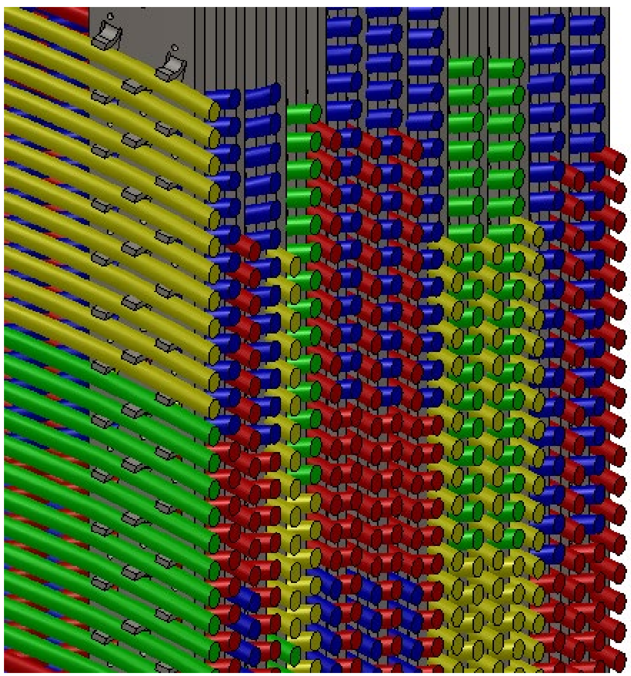
Marty Bryan
Licensing Project Manager

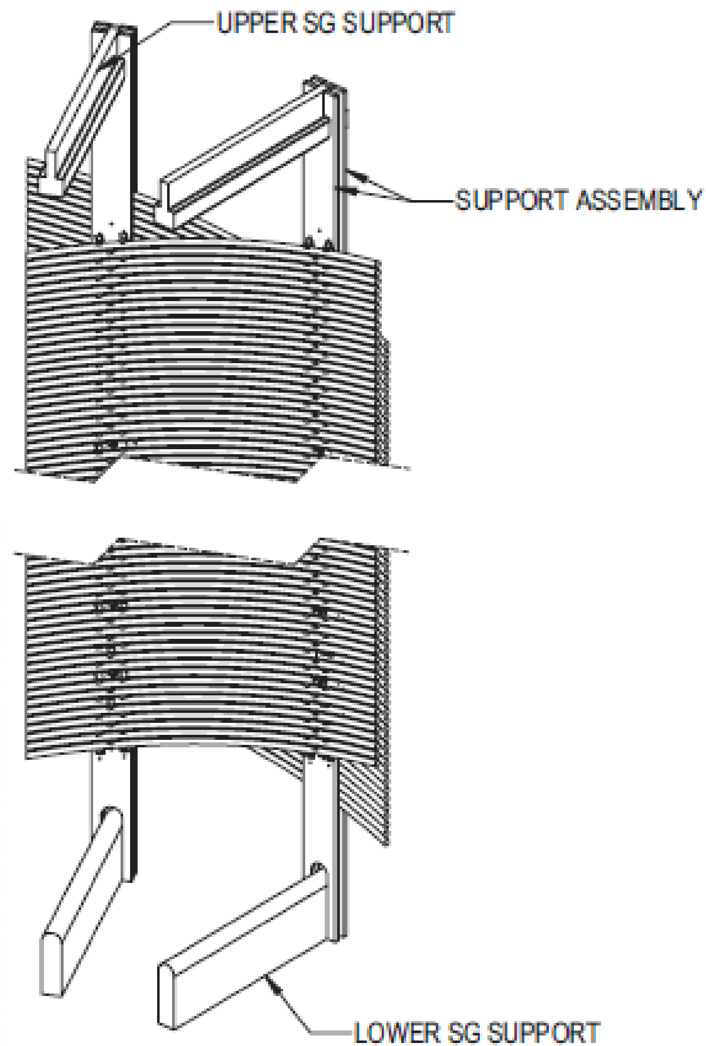
Agenda

- Steam Generator Design
- DCA Revisions

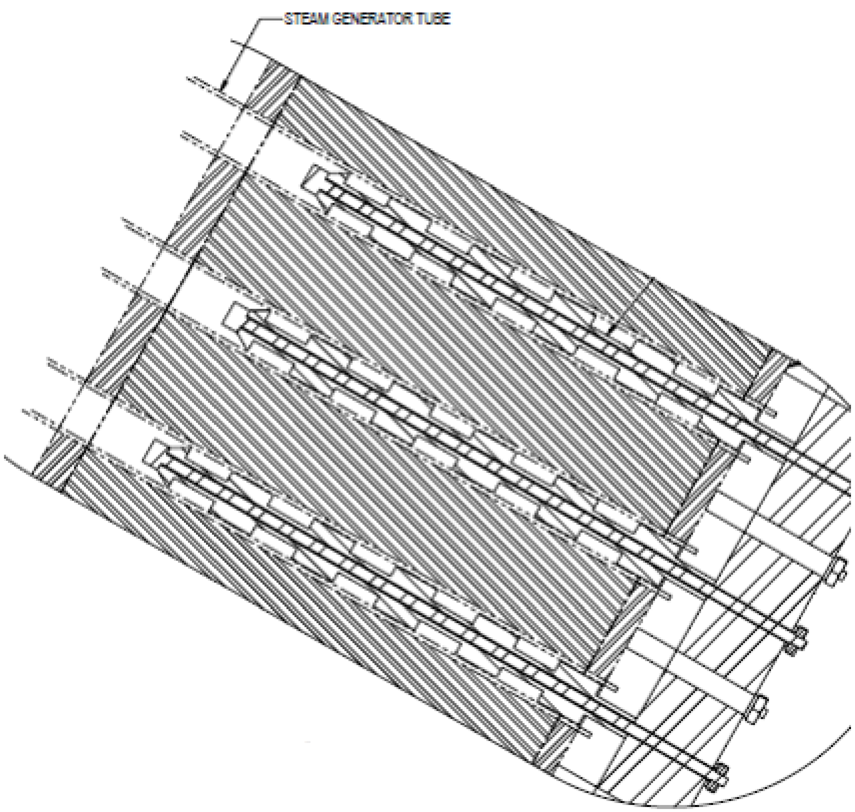
Steam Generator Design



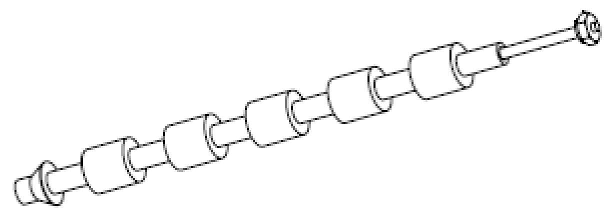




Steam Generator Inlet Flow Restrictor



IFR in Tubesheet



Inlet Flow Restrictor (IFR)

PM-0220-0220-69051
Revision: 0

Template #: 0000-21727-F01 R5

Steam Generator Design

- **Integral Helical Coil SG Design features**
  - Shell side is primary side - Tube side is secondary side
  - Alloy 690 TT (1380 tubes, 74 – 86 ft long, 5/8” OD)
  - Low flow in primary (~1ft/sec)
  - Tube wall degradation allowance (0.010” > ASME min wall)
  - Support 100% volumetric inspection
  - Normal access to shell side of tubes from below during refueling
- **Steam Generator Program and In-service Inspections**
  - Follow guidance of NEI 97-06 & EPRI (COL Item 5.4-1: Develop and implement a SG Program)
- **SG is designed with a flow restrictor at tube inlet to reduce the potential for density wave oscillations (DWO)**

DCA Revisions

- An Action Item has been established for the Combined License applicant (COL Item 3.9-14)

> A COL applicant that references the NuScale Power Plant design certification will develop an evaluation methodology for the analysis of secondary-side instabilities in the steam generator design. This methodology will address the identification of potential density wave oscillations in the steam generator tubes, and qualification of the applicable portions of the reactor coolant system integral reactor pressure vessel and steam generator given the occurrence of density wave oscillations, including the effects of reverse fluid flows within the tubes.

DCA Revisions (cont'd)

- FSAR Section 3.9 has been revised and establishes a COL Item for development of an evaluation methodology for analysis of secondary side instabilities.
- FSAR Section 5.4 clarifies language related to secondary side instabilities.

NuScale Conclusion

- The successful completion of ITAAC and the COL Item constitutes the basis for the NRC determination to allow operation of a facility certified under 10 CFR 52

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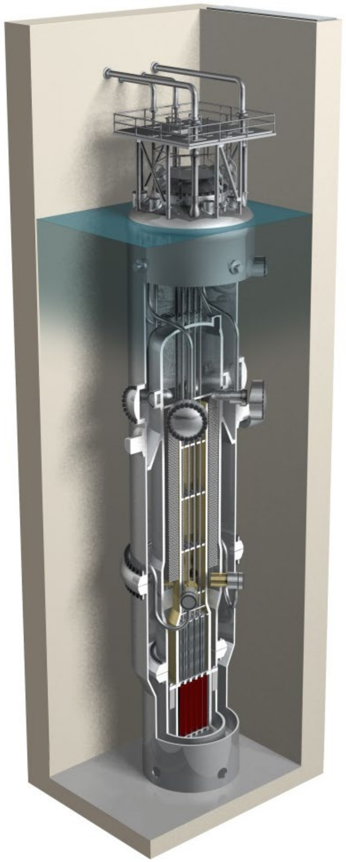

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_Twitter: @NuScale_Power_

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Backup Material

ITAAC Closure Path for DWO

- Resolution of DWO is to be achieved through ITAAC activities related to the steam generator
- Tier 1 Table 2.1-2 defines the NuScale Power Module (NPM) ASME Code Class 1, 2, 3, and CS components that comply with ASME Code Section III requirements including:

Equipment Name	ASME Code Section III
RCS Integral RPV/SG/Pressurizer	1

- Number 02.01.01 specifies that “each ASME Code Class 1, 2, and 3 component (including piping systems) of a nuclear power plant requires a Design Report in accordance with NCA-3550”

ITAAC Closure Path for DWO (continued)

- An ITAAC inspection is performed of the NuScale Power Module “ASME Code Class 1, 2, 3, and CS as-built component Design Reports to verify that the requirements of ASME Code Section III are met”
- From Subsection NCA of the 2013 Edition of the ASME Code –
  - NCA-2142.2 requires that Design Specifications identify all loadings (e.g. pressure, temperature, mechanical loads, cycles, and/or transients) and the service limits a component will experience
    - Loading combinations for the RPV (including SG tubes) defined in Table 3.9-3 of DCA
    - Transient (TH) loads are based on time history of design basis transients, described in DCA Section 3.9.1.
  - NCA-3254 and 3255 provide additional information about design specifications
  - NCA-3260 requires that the Design Report evaluate the loads as defined in the design specification

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

NRC Review of NuScale Steam Generator

NuScale Design Certification Application

ACRS Full Committee Meeting
March 5, 2020

(Open Session)

Agenda

- NRC Staff Review Team
- Summary of Review of Steam Generator (SG) Materials, Design, and Inspection
- Summary of SG Design Issue Not Resolved by Design Certification Application (DCA)
  - Safety Significance
  - Method of Analysis
  - Appendix G to 10 CFR Part 52

NRC Staff Review Team

- Technical Reviewers:
  - Gregory Makar, materials engineering
  - Leslie Terry, materials engineering
  - Yuken Wong, mechanical engineering
  - Peter Yarsky, Office of Research
  - Raymond Skarda, Office of Research
  - Carl Thurston, reactor systems
  - Kaihwa Hsu, mechanical engineering
  - Steven Hambric (consultant)

- Project Management:
  - Marieliz Johnson
  - Bruce Bavol

- Technical Management:
  - Thomas Scarbrough, mechanical engineering
  - Rebecca Patton, reactor systems
  - Steven Bloom, materials engineering

NuScale Steam Generator
SER Sections 5.4.1 and 5.4.2
SG Materials, Design, and Inspection

FINDING: SG Materials and SG Program meet applicable requirements for most review areas:

- Materials acceptable with respect to selection, fabrication, testing, and inspection
- Design limits crevice areas along tubes
- Primary and secondary water chemistry acceptable (based on industry guidelines)
- Design provides primary and secondary access for inspection and for removal of corrosion products and foreign objects

SER Sections 5.4.1 and 5.4.2 SG Materials, Design, and Inspection

FINDING: SG Materials and SG Program meet applicable requirements for most review areas:

(Continued)

- SG Program based on applicable industry guidelines and consistent with the Standard Technical Specifications
- Generic tube plugging criterion determined in accordance with applicable guidance
- Combined License (COL) applicant will develop and implement an SG Program and provide corresponding plant-specific information

SER Sections 5.4.1 and 5.4.2 SG Materials, Design, and Inspection

SG DESIGN – Secondary Flow Oscillations

- NRC staff considers design demonstration of structural and leakage integrity for SG tubes to be incomplete for DCA including:
  - Ability of SG tubes to maintain structural and leakage integrity during density wave oscillation (DWO) in SG secondary fluid system
  - Method of analysis to predict thermal-hydraulic conditions and loads of SG secondary fluid system
- NuScale is working to demonstrate SG tube integrity subsequent to design certification

Regulatory Process for Incomplete SG Tube Integrity

- NRC staff is proposing to specify structural and leakage integrity of SG tubes as not resolved and not receiving finality in NRC draft proposed rule for NuScale design certification.
- Appendix G to 10 CFR Part 52, Section VI, “Issue Resolution,” is being proposed to clarify that SG tube integrity is not resolved within the meaning of §52.63(a)(5)
- Section IV, “Additional Requirements and Restrictions,” is being proposed to state that COL applicant is responsible for providing design information to address SG tube integrity.
- Draft proposed rule currently in concurrence process prior to being provided to the Commission for approval.

SG Secondary Fluid System Method of Analysis

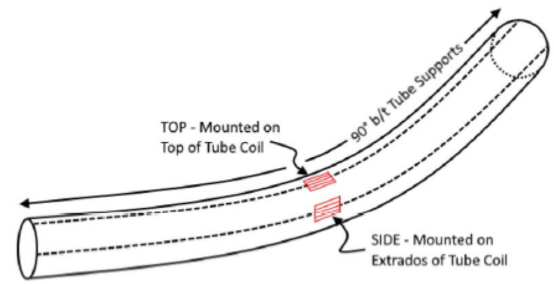
- DCA Part 2, Tier 2, Section 3.9.1.2 states that it lists computer programs used by NuScale for dynamic and static analyses and hydraulic transient load analyses.
- Section 3.9.1.2 does not include the method of analysis to appropriately predict thermal-hydraulic conditions and loads of SG secondary fluid system.
- In demonstrating SG tube integrity, COL applicant will need to provide information demonstrating that 10 CFR Part 50, Appendix A, GDC 4, is met for the method of analysis to predict thermal-hydraulic conditions of SG secondary fluid system and resulting loads, stresses, and deformations from DWO.

Demonstration of SG Tube Integrity

- NuScale has not provided reasonable assurance that flow oscillations that occur in SG secondary fluid system will not cause damage to SG tubes directly from DWO or indirectly by inlet flow restrictors (IFRs).
- COL applicant will need to provide information demonstrating that 10 CFR Part 100 and Part 50, Appendix A, GDC 4 and 31, are met with respect to structural and leakage integrity of SG tubes that might be compromised by adverse effects from DWO in SG secondary fluid system.

DWO Phenomenon

- TF-2 testing involved a full scale mock-up of 252 tubes.
- DWO was observed during TF-2 testing with temperature and flow oscillations in the secondary coolant.
- DWO frequency during TF-2 testing did not excite SG tube structural resonances.

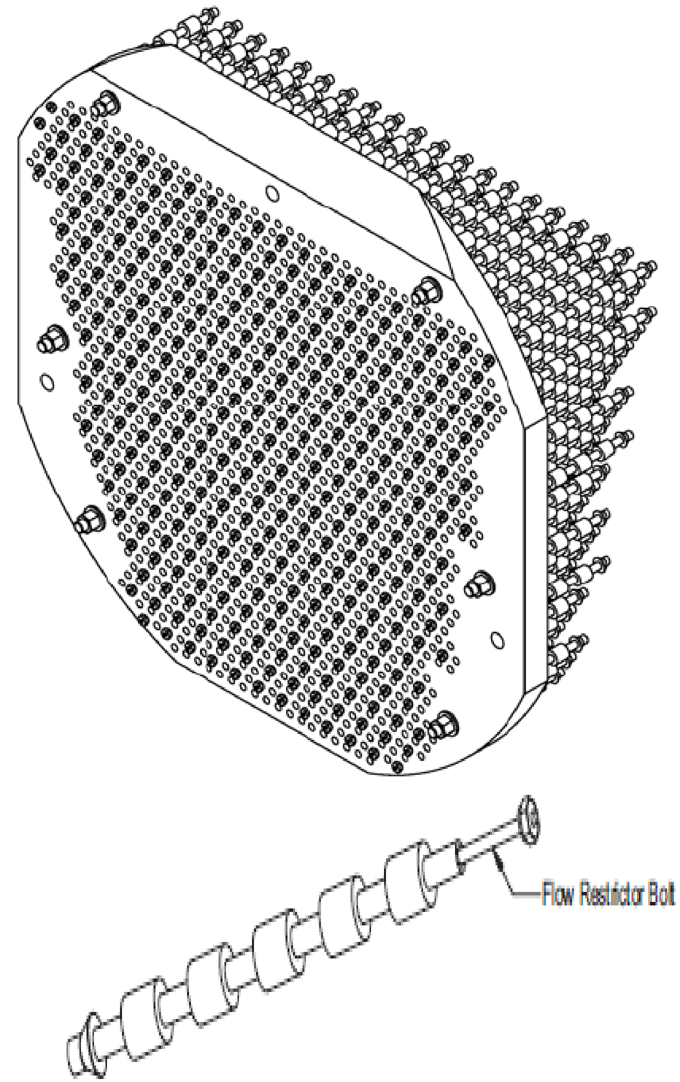


Alloy 690 (Ni-Cr-Fe)

- TF-2 alternating stress intensities for instrumented TF-2 tubes were below fatigue endurance limit, although TF-2 geometry, materials, and operating conditions might not be conservative compared to as-built SG.
- As discussed on the next slides, the staff is concerned about the potential impact of DWO on the SG tubes directly and indirectly by the IFRs.

SG Inlet Flow Restrictor

- SG Inlet Flow Restrictor (IFR) designed to provide necessary pressure drop to limit DWO in the SG tubes.
- Staff evaluated leakage flow instability (LFI) between IFRs and SG tubes during forward flow test (separate from TF-2) and did not identify any concerns.
- However, testing did include DWO conditions.
- NuScale has not validated the final IFR design.



SG Inlet Flow Restrictor – DWO Concerns

- Unstable DWO could cause reverse flow through IFRs
  - Subcooled liquid for modest DWO
  - Slug and two-phase flow for strong DWO
- NuScale has not yet evaluated the potential impacts on SG tubes and IFRs for reverse flow such as:
  - Fatigue of bolted joints, and loose IFR parts
    - LFI in that cantilevered IFRs are less stable under reverse flow
    - Cyclic pressure drops
    - High speed turbulent two-phase flow
  - Cavitation erosion of SG tube walls
  - Wear of IFRs and/or tube walls that could further worsen stability

Post-Design Certification

- COL Applicant will address SG tube integrity in the COL application as follows:
  - Provide validated SG secondary fluid system flow thermal-hydraulic method of analysis
  - Demonstrate that SG tubes will not be damaged by DWO directly or indirectly by IFRs
- COL Holder will verify SG construction including:
  - Complete ITAAC on Tier 1 Table 2.1-4 (#1) to confirm that ASME BPV Code Class components designed to ASME BPV Code Section III
  - Implement Comprehensive Vibration Assessment Program (CVAP) – COL Item 3.9-1
    - Satisfy Tier 1, TF-3 flow testing requirement, and Tier 2, Table 14.2-72 SG flow-induced vibration testing
    - Instrument one tube in initial startup SG testing with strain gages at top, middle, and bottom, for FIV evaluation

Next Steps

- NuScale is preparing errata for Revision 4 to DCA to clarify SG secondary fluid flow issues that could impact SG tubes and IFRs.
- NRC staff discusses SG tube integrity, including SG secondary flow method of analysis, in the draft proposed rule for NuScale design certification to be provided for Commission approval.
  - Draft proposed rule excludes SG tube integrity from finality.
  - NRC staff will address SG tube integrity as part of a NuScale COL application review.
- Other aspects of the NuScale SG design are acceptable to the NRC staff and would be granted finality.

Questions?

LO-0320-69151

March 4, 2020

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Full Committee Presentation: NuScale Topical Report – Rod Ejection Accident Methodology," PM-0320-69146, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) NuScale Full Committee Meeting on March 5, 2020. The materials support NuScale's presentation of the "Rod Ejection Accident Methodology" topical report.

The enclosure to this letter is the nonproprietary presentation entitled "ACRS Full Committee Presentation: NuScale Topical Report – Rod Ejection Accident Methodology," PM-0320-69146, Revision 0.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Matthew Presson at 541-452-7531 or at mpresson@nuscalepower.com.

Sincerely,

Zackary W. Rad
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Enclosure:

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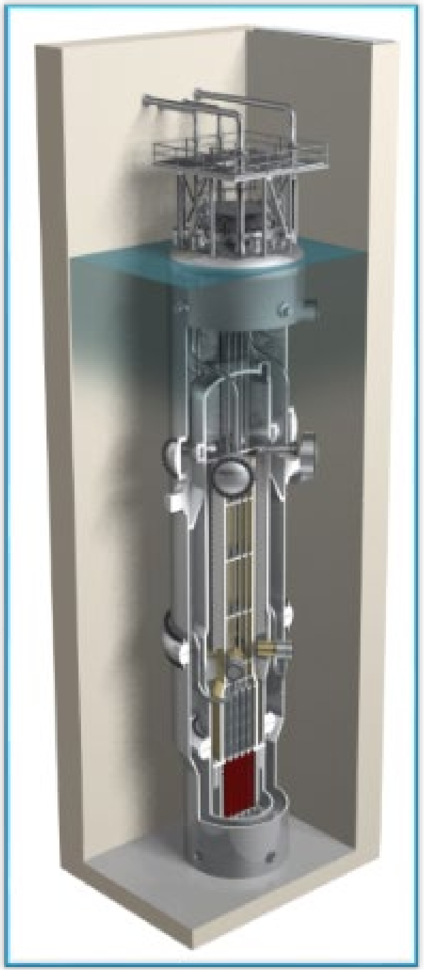
ACRS Full Committee Presentation



NuScale Topical Report

Rod Ejection Accident Methodology

March 5, 2020

PM-0320-69146
Revision: 0

Presenters

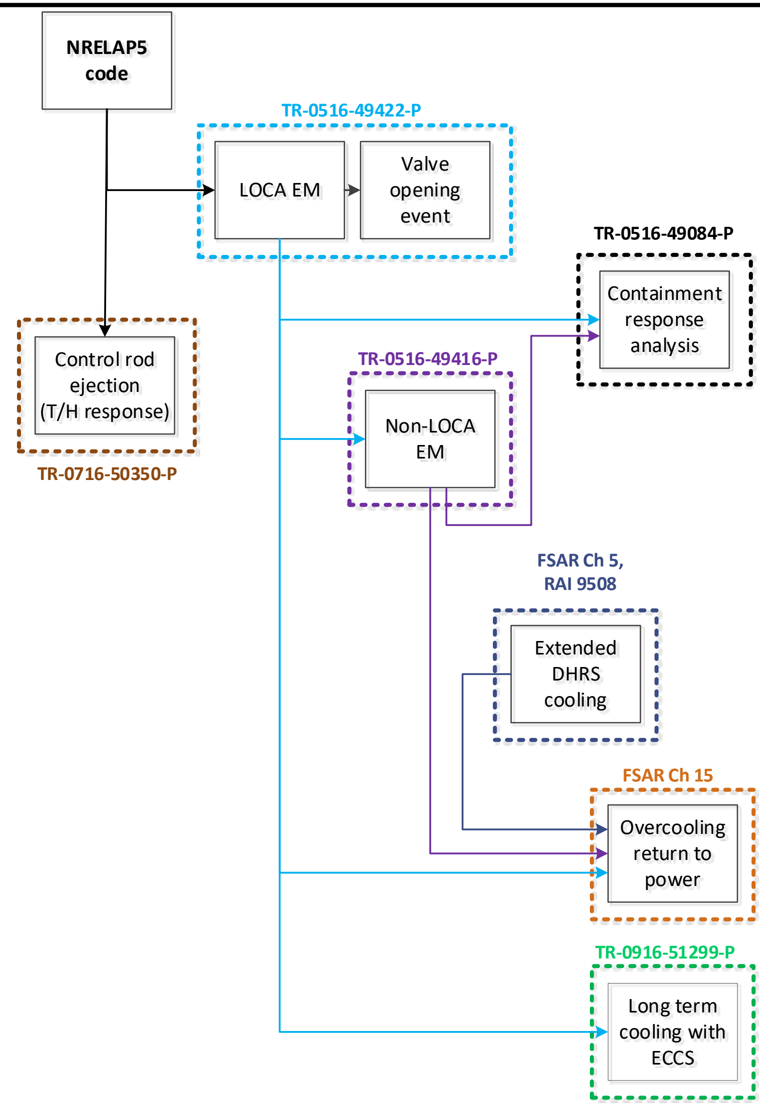
Kenny Anderson
Nuclear Fuels Analyst

Matthew Presson
Licensing Project Manager

Opening Remarks – NuScale T/H Methods

System T/H Analysis Basis

- NRELAP5 code developed from RELAP5-3D
  - Modified to address NuScale-specific phenomena/systems
- LOCA Evaluation Model (EM) developed following RG 1.203 EMDAP
  - LOCA EM extended to derive EMs for other events as shown in this figure.
  - LOCA EM assessment basis leveraged for non-LOCA.
- Additional supporting EMs include
  - Nuclear Analysis Codes – TR-0716-50350-P-A
  - Critical Heat Flux – TR-0116-21012-P-A
  - Subchannel Analysis – TR-0915-17564-P-A



Agenda

- Event Overview
- Acceptance Criteria
- PCMI Criteria – DG-1327
- Method Flowchart
- Steady State Initialization
- Event Evaluations
- Summary

Overview

- NuScale seeks approval of methodology for modeling rod ejection accident (REA) events
- Bounding reactivity initiated accident (RIA) from General Design Criteria (GDC) 28
- REA is unique in comparison to other Ch. 15 events

Description	Rod Ejection	Other Events
Dominant Physics	Nuclear	Thermal-Hydraulics
Timing	milli-sec	sec to hr
Spatially	Local	Global
Peak power	~5x Full Power	~1.2x Full Power
Integrated Energy	Low	Low to High
Postulated Cause	Failure of ASME Class 1 Pressure Boundary	Single Equipment Failure
Acceptance Criteria	Specialized	Generic

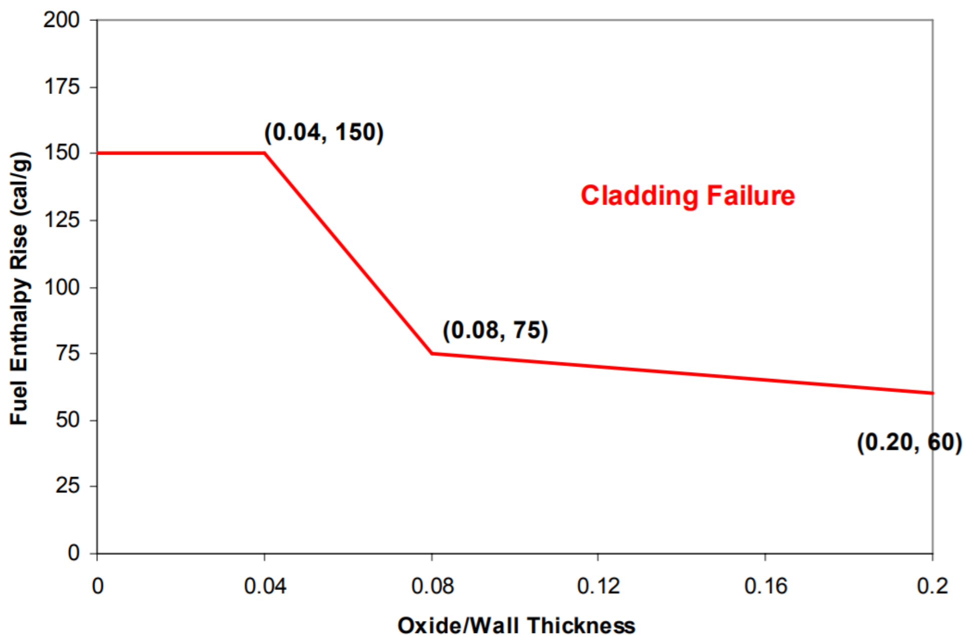
Unique Event Acceptance Criteria

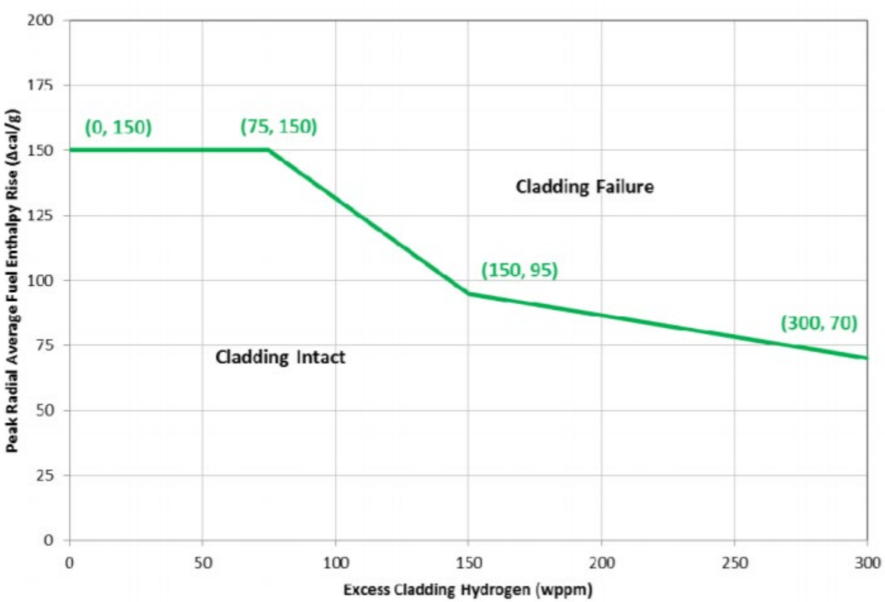
Criteria Description	Topical Section	Unique?
Maximum reactor coolant system pressure	5.3	No
Hot zero power (HZP) fuel cladding failure	5.5.2	Yes
FGR effect on cladding differential pressure	N/A	Yes
Critical heat flux (CHF) fuel cladding failure	5.4.1	No
Cladding oxidation-based PCMI failure	5.5.3	Yes
Cladding excess hydrogen-based PCMI failure	N/A	Yes
Incipient fuel melting cladding failure	5.5.1	No
Peak radial average fuel enthalpy for core cooling	5.5.2	Yes
Fuel melting for core cooling	5.5.1	No
Fission product inventory (failed fuel census)	5.6	Yes

- Submitted NuScale design and method inherently precludes fuel failure, thus no accident radiological consequences are evaluated.
- PCMI: Pellet-Clad Mechanical Interaction

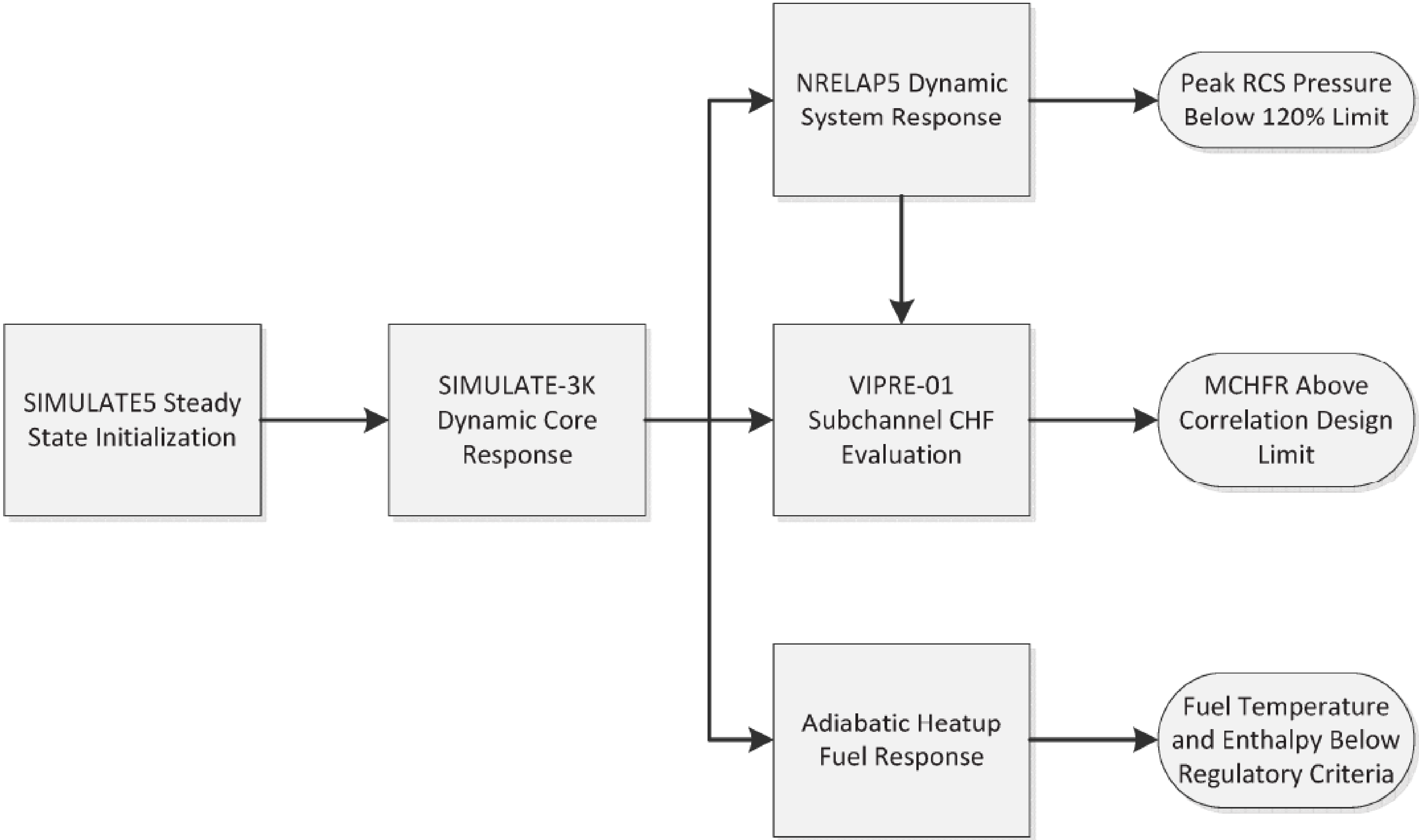
Revised PCMI Criteria

- In general, the NuScale REA methodology has adopted the limiting criteria of the ‘Clifford Letter’ (ML14188C423), now included in draft guide DG-1327 (ML16124A200). In spirit, NuScale is prepared for this regulatory change:
  - Closed session presents example results, showing large margins for enthalpy rise
  - A technical ‘formality’ inhibits complete adoption at this time. NuScale does not currently have a validated cladding H_2 model to convert local exposure to excess cladding hydrogen
  - Oxidation criteria from NUREG-0800 Section 4.2, Appendix B (ML07074000) is used
  - To simplify method, no exposure is credited (Limit: 75 Δcal/gm)
  - NuScale M5 cladding less susceptible than other zirc alloy-type clad used in the industry





Unique Event Method (Flowchart)



Steady-State Initialization

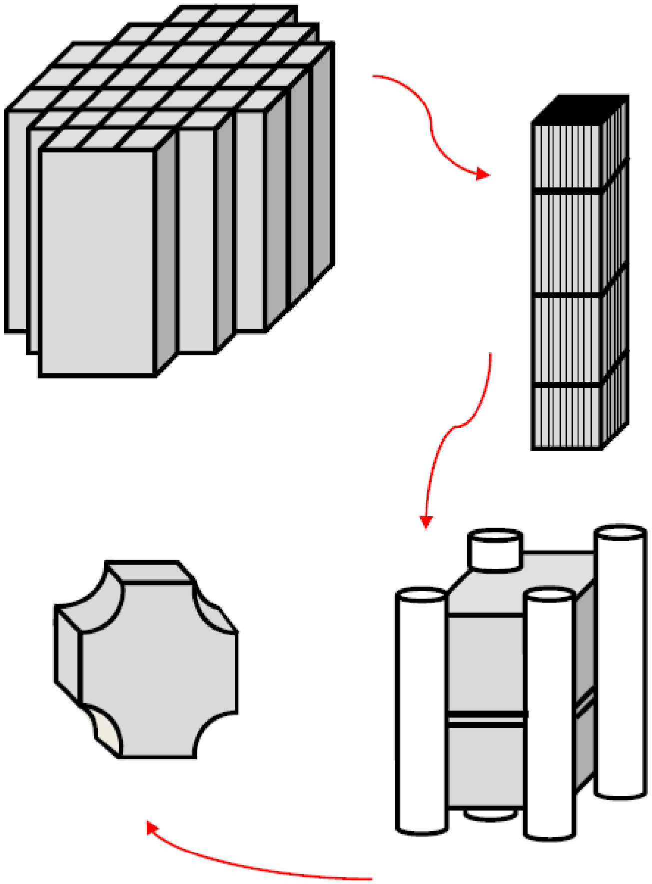
- SIMULATE5: Setup the core response analysis
- Code shown to be appropriate in TR-0616-48793-A (Nuclear Analysis Codes and Methods Qualification)
- Determination of the worst rod stuck out (WRSO)
  - Assumption bounds potential for ejected assembly to damage adjacent control rod assembly
  - Due to rapid nature of the event, location does not significantly affect the results in NuScale application

Dynamic Core Response

- SIMULATE-3K: Model transient core response
- Benchmarked to SPERT-III experiment and NEACRP computational benchmark
  - Benchmarks demonstrate the combined transient neutronic, thermal-hydraulic, and fuel pin modeling capabilities
  - SIMULATE-3K results generally in excellent agreement with the results from the two benchmark problems
- Uncertainties applied for each simulation:
  - Delayed Neutron Fraction
  - Ejected Rod Worth
  - Doppler Temperature Coefficient
  - Moderator Temperature Coefficient

CHF Evaluation

- VIPRE-01: Model detailed thermal-hydraulics
- Evaluate critical heat flux (CHF) acceptance c
- Code shown to be appropriate in TR-0915-17
  Analysis Methodology)
- Unique event differences in method:
  - Smaller axial nodalization (smaller time steps)
  - Radial power distribution (case-specific)
  - Axial power distribution (peak assembly)
  - Convergence parameters
- Additional parametric sensitivity cases pe
  application to holistically justify difference

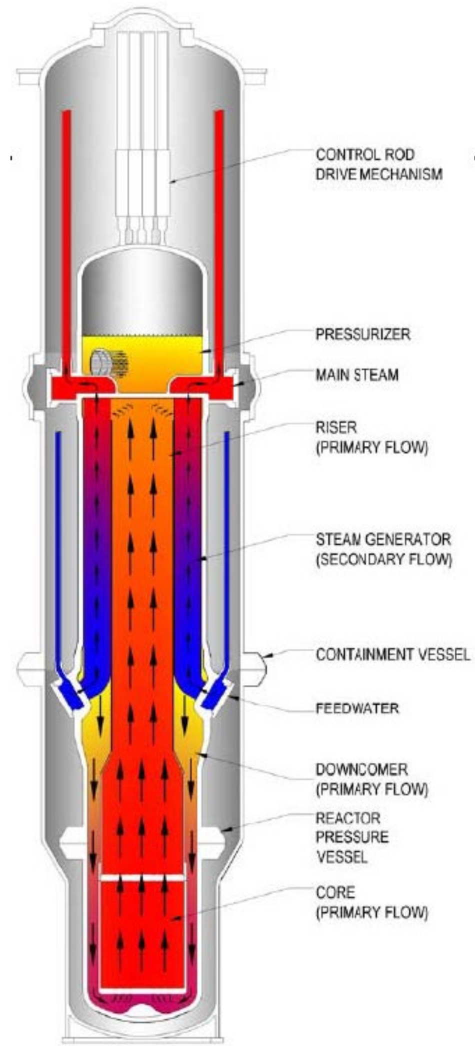


Adiabatic Fuel Heatup

- Hand-Calculation: Model fuel response
- Total energy (from SIMULATE-3K) during the transient is integrated
- Conservative as no energy is allowed to leave the fuel rod
- Energy is then converted into either a temperature or enthalpy increase
- Fuel rod geometry, heat capacity, and power peaking factors taken into account
- Calculated values compared to NRC developed acceptance criteria
  – Example values provided in closed session

Dynamic System Response I

- NRELAP5: Evaluate system response for input to CHF Evaluation
- Code shown to be appropriate in TR-0516-49416 (Non-LOCA Methodologies)
- Transient power from SIMULATE-3K utilized as input
  - No reactivity calculation performed in NRELAP5
- Provides system thermal-hydraulic conditions to subchannel (CHF) evaluation
  - System flow, pressure, and inlet temperature
  - ‘Screens’ cases for potential to be limiting
  - Family of limiting cases evaluated with VIPRE-01



Dynamic System Response II

- NRELAP5: Evaluate system response for pressurization
- Limiting scenario: Low ejected worth that raises the power quickly to just below both the high power and high power rate trip 'setpoints'
- Point-kinetics model used based on bounding static worth
- Peak system pressure calculated compared to acceptance criteria
- Example results to be presented in closed session

Summary

- A conservative analysis method for the unique rod ejection accident
- Topical report provides details and justification for:
  - Software tools and acceptance criteria used
  - Applicability of the method and tools
  - Appropriate treatment of uncertainties
- Results from application of the method provide input to FSAR Chapter 15

Acronyms

- CHF – Critical Heat Flux
- GDC – General Design Criteria
- HZP – Hot Zero Power
- MCHFR – Minimum Critical Heat Flux Ratio
- NEACRP – Nuclear Energy Agency Committee on Reactor Physics
- PCMI – Pellet Clad Mechanical Interaction
- REA – Rod Ejection Accident
- RIA – Reactivity Initiated Accident
- WRSO – Worst Rod Stuck Out

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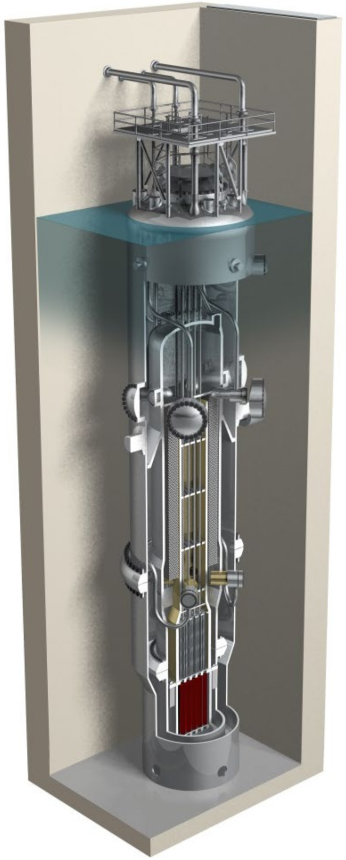

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Power for all humankind

March 4, 2020 Docket No. 52-048

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SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Full Committee Presentation: NuScale Topical Report, Loss-of-Coolant Accident Evaluation Model," PM-0320-69138, Revision 0

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ACRS Full Committee Presentation

NuScale Topical Report

Loss-of-Coolant Accident Evaluation Model

March 5, 2020

PM-0320-69138
Revision: 0

Presenters

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Thermal Hydraulic Analyst

Ben Bristol
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Agenda

- Methodology Overview
  - Background
  - Regulatory Requirements
  - Methodology Roadmap
- NPM Safety Systems Overview
- Element 1: PIRT
- Element 2: Assessment Base
- Element 3: NRELAP5 Evaluation Model
- Element 4: Applicability Evaluation
- Extension of LOCA EM to IORV
- Conclusions

Background

- Unique NPM Design Features
  - Integrated design eliminates piping and limits potential breaks
  - Coolant captured completely in containment, cooled and returned to RPV using a large pool as ultimate heat sink
- Simple LOCA Progression with Well-Known Phenomena
  - Choked/un-choked flow through break and ECCS valves
  - Core decay heat and RCS stored energy release
  - CNV heat transfer to pool (condensation, conduction, convection)
- EM Development Approach
  - Follows Regulatory Guide 1.203 EMDAP (Table 2-1)
  - Compliance with 10 CFR 50.46 and Appendix K requirements (Table 2-2)

Regulatory Requirements

- 10 CFR 50.46 Acceptance Criteria
  - Max. clad temperature < 2200 ºF
  - Cladding oxidation > 0.17 times thickness
  - Hydrogen generation < 0.01 times total hydrogen from oxidation of all cladding
  - Core remains amenable to cooling
  - Long-term cooling maintained
- Maximum PCT at steady state, no clad heat up
- Conservative LOCA EM Acceptance Criteria (<u>FOMs</u>)
  - Core remains covered: <u>collapsed level</u> > TAF
  - M<u>CHFR</u> > CHFR Limit (1.29)
  - <u>Containment pressure and temperature</u> below design limit

Methodology Roadmap

- 10 CFR 50.46 Appendix K Compliance (Section 2.2.3 of LTR)
- RG 1.203 EMDAP (Section 2.1 of LTR)

Element 1
Establish Requirements for EM Capabilities
LOCA PIRT (*LTR Ch. 4*)

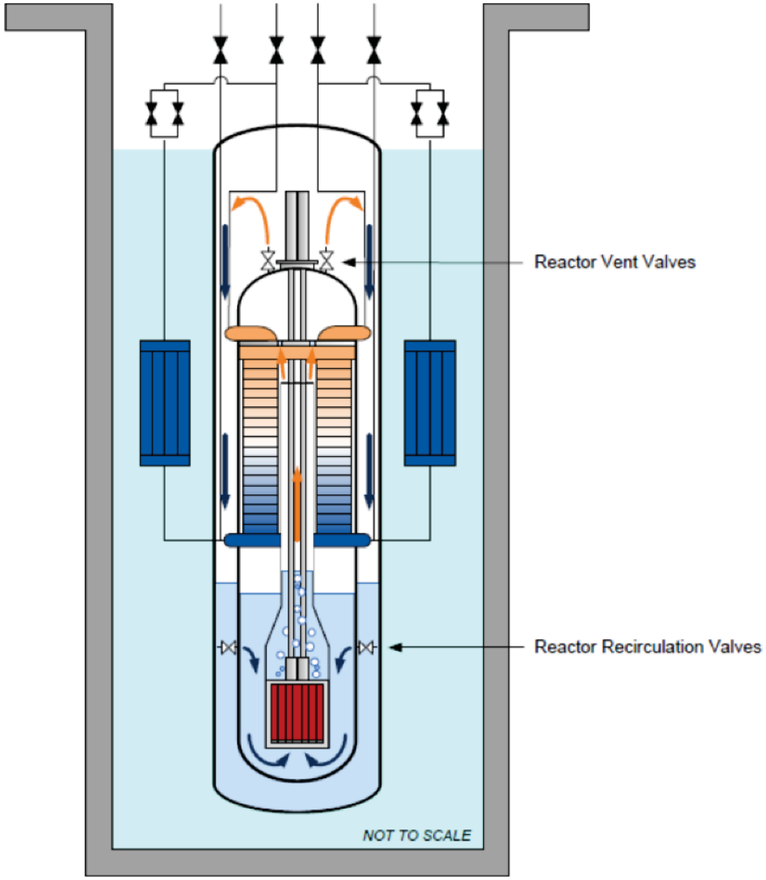
Element 2
Develop Assessment Base
SET and IET Assessment (*LTR Ch. 7*)

Element 3
Develop Evaluation Model
NRELAP5 Code (*LTR Ch. 6*)
Evaluation model (*LTR Ch. 5*)
Break spectrum and sensitivity calcs. (*LTR Ch. 9*)

Element 4
Assess EM Adequacy
Top-down and Bottom-up evaluation (*LTR Ch. 8*)

NPM Safety Systems

- ECCS
  - Opens a boiling/condensing circulation flow path to transfer decay and residual heat to reactor pool
  - Reactor Recirculation Valves (RRV): 2 valves
  - Reactor Vent Valves (RVV): 3 valves
  - Actuation Signals: High CNV level, 24-hour loss of AC power
  - Fail safe: ECCS trip valves open on loss of DC power
- Inadvertent Actuation Block (IAB)
  - Prevents inadvertent opening of ECCS valves at high RCS pressure
  - Actuation based on differential pressure between RPV and CNV
- Module Protection System (MPS)
  - Reactor scram
  - Steam Generator (SG) and Containment (CNV) Isolation
  - Passive safety system activation (ECCS and DHRS)
- Decay Heat Removal System (DHRS)
  - Passive, boiling-condensation system
  - Removes heat from RCS through SG via two trains
  - Each trains capable of removing 100% decay heat
  - Not credited in LOCA EM



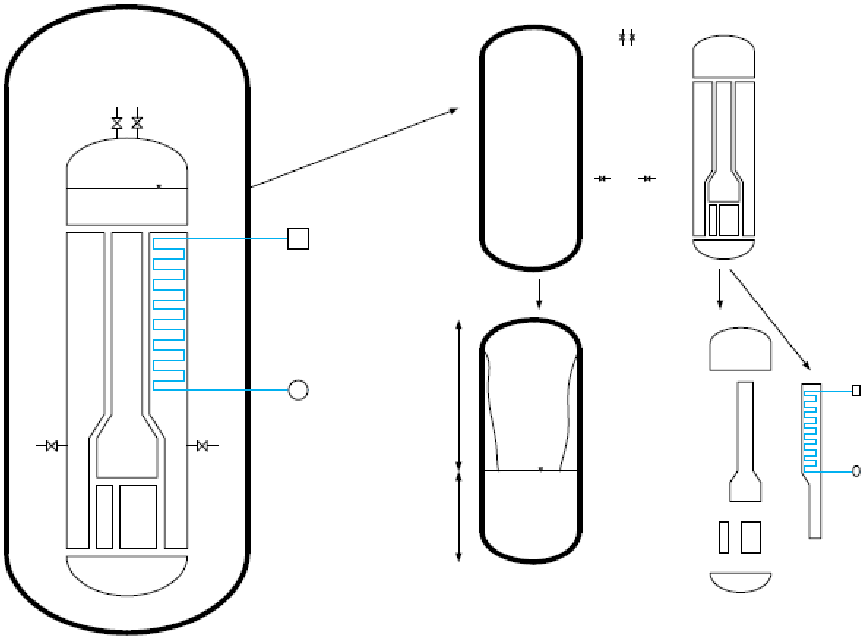
Element 1
PIRT

PIRT Process

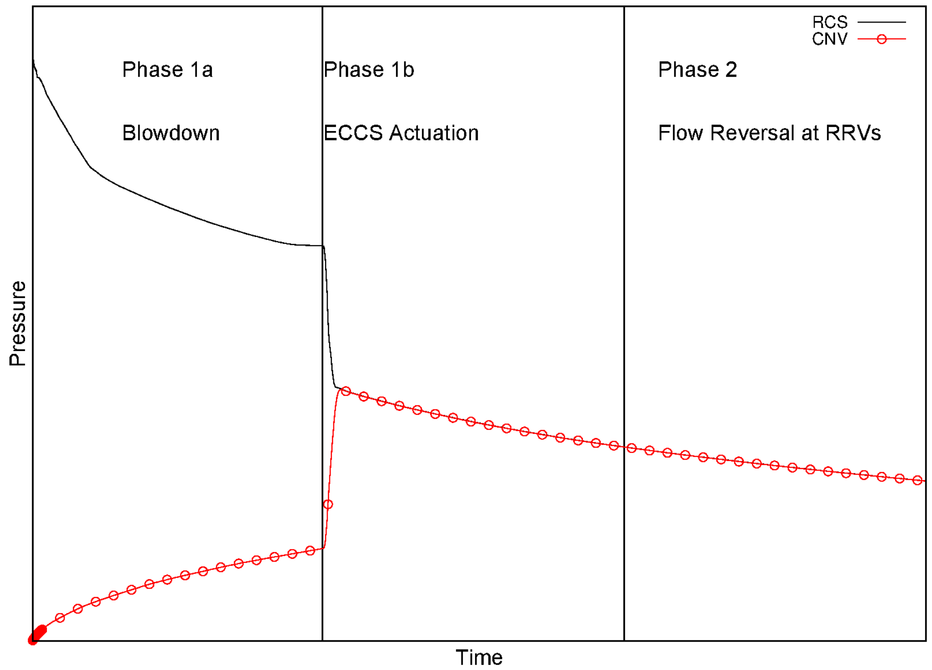
- Assessment of relative importance of phenomena
  - Unique phases
  - Key components
- PIRT panel included recognized experts and NuScale subject matter experts
  - State-of-knowledge, design description, LOCA description, NRELAP5 calculations
- Figures-of-Merit
  - CHF, Collapsed level above top of the active fuel, CNV P & T
- Rankings
  - Importance: High, Low, Medium, Inactive
  - Knowledge: Well known (small uncertainty), Known (moderate uncertainty, partially known (large uncertainty), very limited

Spatial and Temporal Decomposition

- Phenomena identified for Systems, Structure, Components (SSCs) and LOCA phases
  - Phase 1a: Blowdown
  - Phase 1b: ECCS activation (opening)



System/Subsystem/Module decomposition



Distinct phases of a typical NPM LOCA

Element 2
Assessment Base

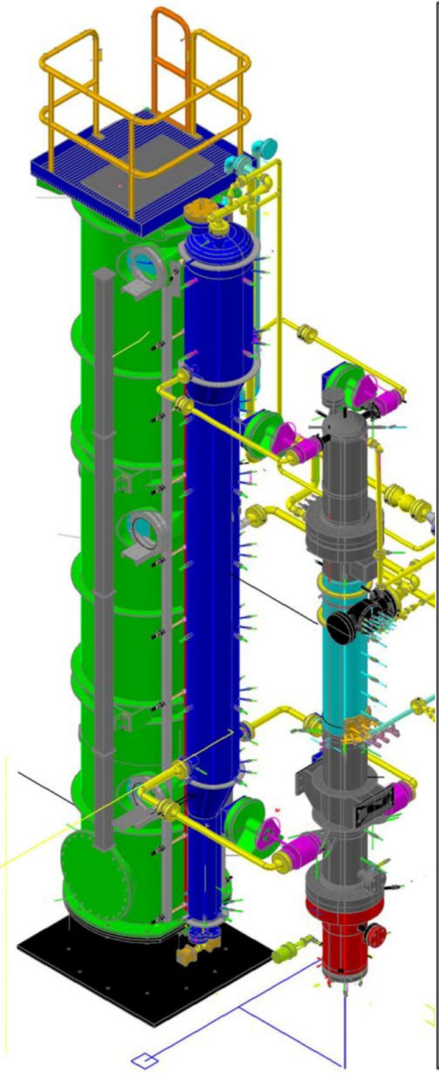
NRELAP5 Code

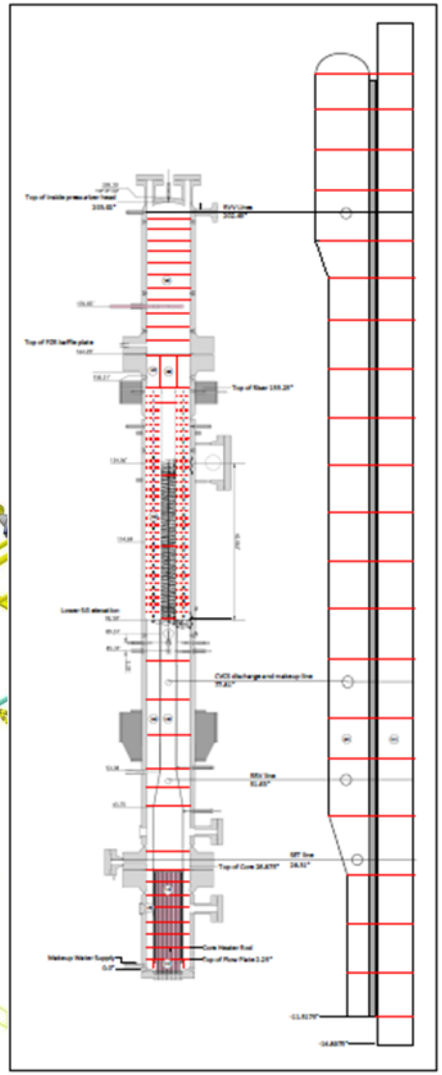
- RELAP5-3D© v4.1.3 used as a baseline code
  - Two-fluid model (thermal and mechanical non-equilibrium) for hydrodynamics with
    - Non-condensable gases with gas phase
    - Semi-implicit scheme for time integration
  - Heat conduction across 1D geometries (slab, cylinder, sphere)
  - Neutron Kinetics with thermal hydraulic feedback
  - Special Process Models
  - Comprehensive control/trip system modeling
- Code configuration control and development consistent with NuScale's NQA-1 2008 / 2009a QA program
- Modifications for NRELAP5:
  – NuScale specific components (e.g., helical coil SG)
  – Regulatory requirements (i.e., Appendix K)
  – Error correction

IET and SET Data

- Extensive database with adequate coverage of all high-ranked phenomena
- Integral effects tests (IET)
  - Six (6) NIST-1 tests
- Separate effects tests (SET)
  - Two (2) NIST-1 SETs
  - Four (4) other NuScale SETs
  - Nine (9) Legacy SETs

NIST-1 Facility





- Primary source of NuScale-Specific IET and SET data
- Design Features
  - Integral Reactor Vessel with electrically heated rod bundle core, helical coil steam generator, and pressurizer
  - Containment with HTP and Cooling Pool
  - DHRS, ECCS, CVCS lines represented
  - ~700 instruments
- Scaling Basis
  - Power/Volume Scaling
  - Reduced height and reduced volume scale
  - Full Pressure and Temperature
  - Same Time Scale (isochronicity)

Element 3
NRELAP5 NPM LOCA

NPM LOCA Model Overview

- Analysis and Justifications
  - NRELAP5 model nodalization and input options
  - Time-step control
  - Initial and boundary condition biases
  - Treatment of setpoints and trips
- LOCA break spectrum
  - Break location and sizes
  - Single failures
  - Power availability
- Methodology sensitivity calculations
  - Required by Appendix K
  - Phenomena-specific
  - To establish conservative biases

Element 4
Applicability Evaluation

Applicability Evaluation

- Evaluated models and correlations (bottom-up)
  - Identified dominant models/correlations for ‘H’ phenomena (Table 8-1 of LTR)
  - Identified key model/correlation parameters and phenomenological domain where models/correlations are used (Tables 8-2 and 8-4)
  - Reviewed models/correlations (Table 8-18 of LTR)
    - Pedigree, Applicability range, Fidelity to SET data, Scalability
- Evaluated integral performance of EM (top-down)
  - Reviewed code governing equations and numerics
  - Evaluated integral performance of code using IET data (Table 8-19 of LTR)
  - Evaluated IET data applicability and NRELAP5 scalability
    - Scaling and distortion analysis
    - Differences and distortions between NPM and NIST can be accounted using NRELAP5

Conclusions

- Number of conservatisms built into the NuScale LOCA EM
  - 10 CFR 50 Appendix K
  - Other methodology conservatisms
- Cycle independent bounding LOCA analysis
- Supported by extensive experiment database, well qualified code, and several sensitivity calculations
- Applicability evaluation consistent with RG 1.203
- CHF not challenged
- Collapsed level in RPV remains above TAF
- No clad or fuel heat-up
- CNV P&T remain below design limits

Appendix B to LOCA LTR

Extension to IORV Event

IORV Background

- LOCA EM Extended to IORV
  - Liquid space (RRV) and steam space (RVV, RSV) discharge
  - Similar transient phenomena and progression
- EM Development Approach
  - Compliance with DSRS for NuScale SMR Design 15.6.6
  - Follows RG 1.203 EMDAP
  - Element 1 (PIRT), Element 2 (Assessment), and Element 4 (Applicability) remains same as LOCA EM
    - Initial LOCA PIRT addressed IORV
  - Element 3 (NRELAP5 Model) unique due to event classification

Differences from LOCA EM

- Minor methodology differences given AOO classification
- Key Acceptance Criteria
  - MCHFR ≥ Limit (≥ 1.13 high flow range, ≥ 1.37 low flow range)
- Conservatisms same as LOCA with exceptions:
  - Fuel properties still biased to maximize stored energy, but additional 15% bias removed
  - Limiting axial power shapes and radial peaking based on subchannel analysis
  - Moody choked flow model for 2-phase flow choking applied to initiating valve
  - Initial conditions biased to minimize MCHFR

Conclusions

- IORV is an extension of LOCA EM given similar transient phenomena and progression
  - PIRT, Assessment, and Applicability same as LOCA
- Minor methodology differences for AOO classification
  - Focused on conservative CHFR evaluation
- MCHFR occurs early in transient, then rapidly rises given increasing flow to power ratio
- Collapsed level in RPV remains above TAF

Acronyms

1-D	one-dimensional	HP	high pressure
3D	three-dimensional	HS	heat sink
AC	alternating current	HTP	heat transfer plate
ANS	American Nuclear Society	H2TS	hierarchical two-tiered scaling
CCFL	counter current flow limitation	IAB	inadvertent actuation block
CHF	critical heat flux	IET	integrated effects test
CNV	containment vessel	INL	Idaho National Laboratory
CVCS	chemical and volume control system	KATHY	Karlstein thermal-hydraulic test facility
DC	direct current	kW	kilowatt
DCA	Design Certification Application	LOCA	loss-of-coolant accident
DHRS	decay heat removal system	LTR	Licensing Topical Report
ECCS	emergency core cooling system	Max	maximum
EM	evaluation model	MCHFR	minimum critical heat flux ratio
EMDAP	evaluation model development and assessment process	Min	minimum
		$Mlb/ft^2 \cdot hr$	pounds mass per square foot per hour
FW	feedwater	MPS	module protection system
FSAR	Final Safety Analysis Report	MSIV	main steam isolation valve
FOM	figure of merit	NIST-1	NuScale Integral System Test Facility
HL	hot leg	NPM	NuScale Power Module

Acronyms

P&T	pressure and temperature
PCT	peak cladding temperature
PIRT	phenomena identification and ranking table
psi	pounds per square inch
psia	pounds per square inch absolute
PZR	pressurizer
QA	Quality Assurance
RCS	reactor coolant system
RG	Regulatory Guide
RRV	reactor recirculation valve
RPV	reactor pressure vessel
RVV	reactor vent valve
SG	steam generator
SET	separate effects test
SIET	Società Informazioni Esperienze Termoidrauliche
StDev	standard deviation
TAF	top of active fuel

March 4, 2020 Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Full Committee Presentation: NuScale Topical Report – Non-Loss-of-Coolant Accident," PM-0320-69141, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) NuScale Full Committee Meeting on March 5, 2020. The materials support NuScale's presentation of the "Non-Loss-of-Coolant Accident" topical report.

The enclosure to this letter is the nonproprietary presentation entitled "ACRS Full Committee Presentation: NuScale Topical Report – Non-Loss-of-Coolant Accident," PM-0320-69141, 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 mpresson@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
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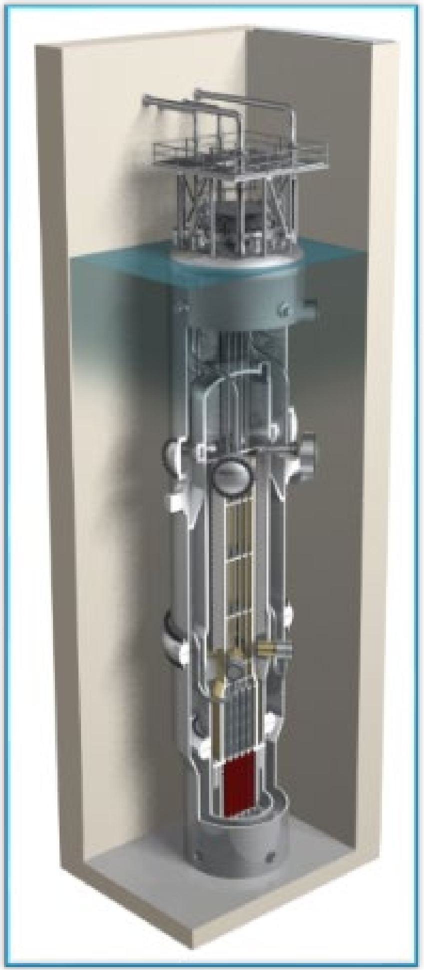
Enclosure: "ACRS Full Committee Presentation: NuScale Topical Report – Non-Loss-of-Coolant Accident," PM-0320-69141, Revision 0

Enclosure:

"ACRS Full Committee Presentation: NuScale Topical Report – Non-Loss-of-Coolant Accident," PM-0320-69141, Revision 0

ACRS Full Committee Presentation



NuScale Topical Report

Non-Loss-of-Coolant Accident

March 5, 2020

PM-0320-69141
Revision: 0

Presenters

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Outline

- Scope of non-LOCA LTR
- Non-LOCA events
  - Events and acceptance criteria
  - Interface to other methodologies
  - Factors controlling margin to acceptance criteria
- Development of non-LOCA EM
  - PIRT and gap analysis
  - Focus of NRELAP5 validation for non-LOCA
- General event analysis methodology
- Specific event analysis

Scope of Non-LOCA Topical Report

In Scope

- NRELAP5 system transient analysis of non-LOCA events
- Interface to subchannel and accident radiological analysis
- Short-term transient progression with DHRS cooling

Out of Scope

- SAFDLs evaluated in downstream subchannel analysis
- Accident radiological dose analysis
- Control rod ejection
- LOCA and valve opening events
- Peak containment pressure/temperature analysis
- Long term transient progression with DHRS
  - Riser uncovery
  - Return to power

Non-LOCA EM

EM applicable to NuScale Power Module plant design
Applicable initiating events:

- **Cooldown events**
  - Decrease in FW temperature
  - Increase in FW flow
  - Increase in steam flow
    Inadvertent opening of SG relief or safety valve
  - Steam piping failures (postulated accident)
  - *Loss of containment vacuum*
    Containment flooding

- **Heatup events**
  - Loss of external load
    Turbine trip
  - Loss of condenser vacuum
  - Closure of MSIV
  - Loss of non-emergency AC power
  - Loss of normal FW flow
  - Feedwater system pipe breaks (postulated accident)
  - *Inadvertent operation of DHRS*

- **Reactivity events**
  - Uncontrolled bank withdrawal from subcritical
  - Uncontrolled bank withdrawal at power
  - Control rod misoperation
    - Single rod withdrawal
    - Control rod drop
  - Inadvertent decrease in RCS boron concentration

- **Inventory increase event**
  - CVCS malfunction

- **Inventory decrease events**
  - Small line break outside containment (infrequent event)
  - Steam generator tube failure (postulated accident)

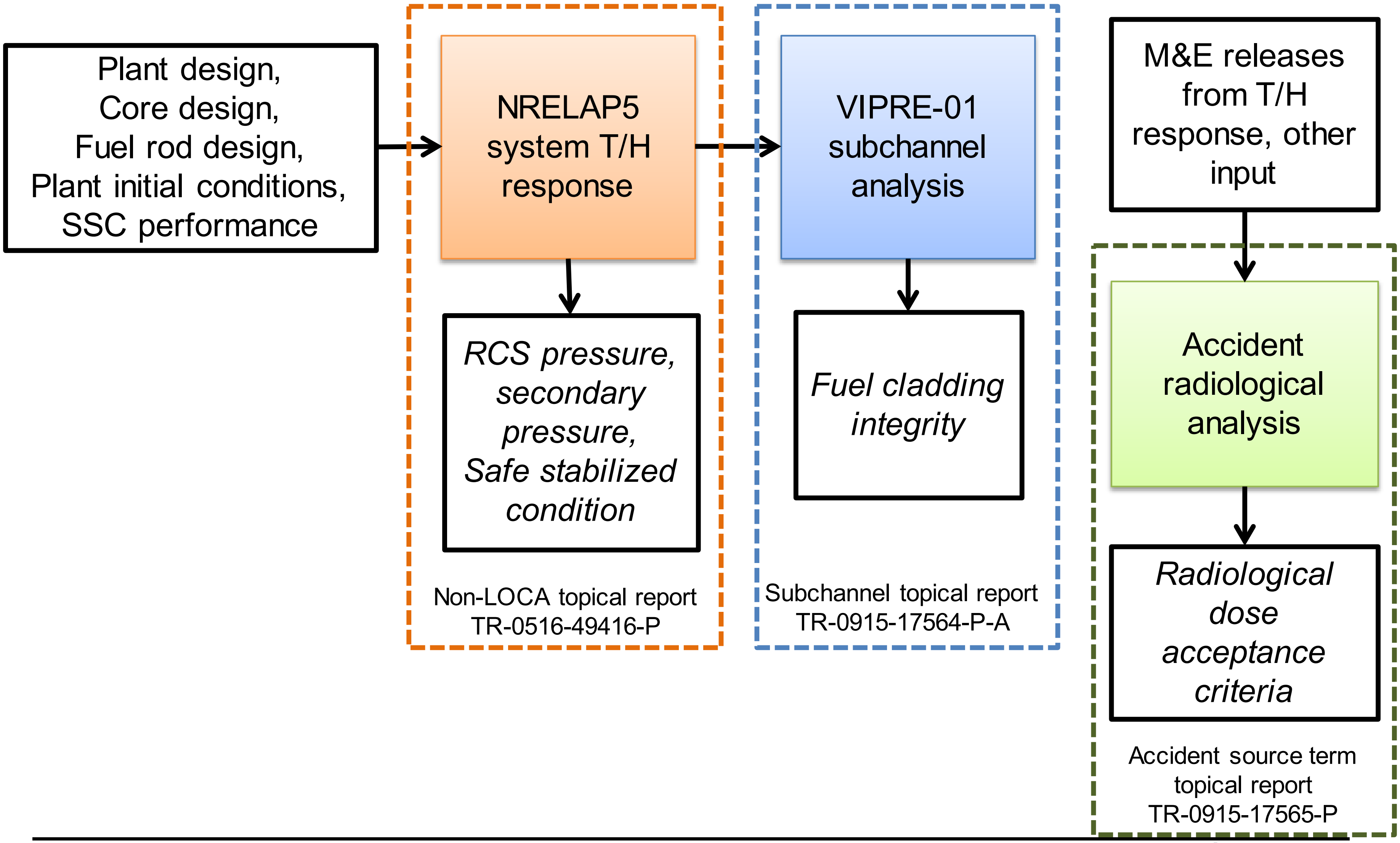
NuScale unique event

Non-LOCA Event Acceptance Criteria

Description	AOO Acceptance Criteria	Infrequent Event Acceptance Criteria	Accident Acceptance Criteria	Analysis
Reactor Coolant System Pressure (P_{design}= 2100 psia)	**≤ 110% of Design**	**≤ 120% of Design**	**≤ 120% of Design**	**Non-LOCA NRELAP5**
Steam Generator Pressure (P_{design}= 2100 psia)	**≤ 110% of Design**	**≤ 120% of Design**	**≤ 120% of Design**	**Non-LOCA NRELAP5**
Minimum Critical Heat Flux Ratio	> Limit	If limit exceed, fuel assumed failed (1)	If limit exceed, fuel assumed failed (1)	Subchannel
Maximum Fuel Centerline Temperature	< Limit	If limit exceed, fuel assumed failed (1)	If limit exceed, fuel assumed failed (1)	Subchannel
Containment Integrity	< Limits (pressure, temperature)	< Limits (pressure, temperature)	< Limits (pressure, temperature)	Containment P/T analysis
Escalation of an AOO to an accident (AOO) or Consequential loss of system functionality (IE or accident)?	**No**	**No**	**No**	**If other acceptance criteria are met**
Radiological Dose	Normal Operations	< Limit	< Limit	Normal or Accident radiological

(1) NuScale safety analysis methodologies developed to demonstrate fuel cladding integrity maintained.

Evaluation Models – General Non-LOCA Approach



Non-LOCA Events - Margin to Acceptance Criteria

Design characteristics governing non-LOCA event transient response and margin to acceptance criteria

– <u>MCHFR</u>: Limited by combination of high power, high pressure, high temperature conditions occurring around time of reactor trip, for reactivity insertion events

– <u>Primary pressure</u>: Protected by RSV lift

– <u>Secondary side pressure</u>: Limited by primary side temperature conditions

– <u>Radiological release</u>: MPS designed to rapidly detect and isolate based on measured conditions

– <u>Establishing a safe, stable condition</u>: MPS designed to trip, actuate DHRS to protect adequate inventory in at least 1 steam generator

Non-LOCA EM Development

- Non-LOCA evaluation model developed to perform conservative analyses, following intent of the RG 1.203 EMDAP and applying a graded approach
- Element 1 – Establish applicable transients and acceptance criteria, develop non-LOCA PIRT
- Element 2, 3, 4
  - Leverage NRELAP5 development, NRELAP5 assessments performed during LOCA evaluation model development.
    - Gap analysis performed to evaluate how high ranked phenomena are addressed
    - Focused on differences in high ranked PIRT phenomena between LOCA and non-LOCA
    - Additional NRELAP5 code validation performed focused on DHRS and integral non-LOCA response
  - Suitably conservative initial and boundary conditions applied for non-LOCA analyses
  - Sensitivity calculations used to demonstrate factors controlling margin to acceptance criteria

Non-LOCA PIRT Development

Event Types
Increased heat removal
Decreased heat removal
Reactivity anomaly
Increase in RCS inventory
Steam generator tube failure

SSCs Considered in PIRT	
Reactor coolant system	Main feedwater system
Containment vessel	Main steam system
Decay heat removal system	Chemical volume control system
Reactor pool	Containment evacuation system

Phase	Identification	RCS Response	DHRS Operation *	PIRT Figures of merit
1	pre-trip transient	higher flow levels at full power levels	inactive	CHFR RCS pressure
2	post-trip transition	transitional flow levels at transitioned power levels	startup	CHFR RCS, secondary, containment pressures
3	stable natural circulation	lower flow levels at decay power levels	fully effective	CHFR RCS mixture level Subcriticality

* If DHRS actuated by protection system

- Different non-LOCA events involve different plant systems and responses
- PIRT developed considering all non-LOCA event types and important SSCs
- Short-term response divided into 3 generic phases with associated FoM

NRELAP5 Applicability for Non-LOCA

After non-LOCA PIRT developed, gap analysis performed to determine how to address high-ranked phenomena:

- Validation performed as part of NRELAP5 assessment for LOCA evaluation model
- Additional validation or benchmark for non-LOCA
- Conservative input
- Subchannel analysis

Key areas identified from gap analysis for short-term non-LOCA analysis:

- DHRS modeling and heat transfer
  - NRELAP5 validation against KAIST tests; NIST-1 SETs HP-03, HP-04
  - NPM sensitivity calculations
- Steam generator modeling and heat transfer
  - NRELAP5 validation against SIET-TF1, SIET-TF2 tests
  - NPM sensitivity calculations
- Reactivity event response
  - NRELAP5 benchmark against RETRAN-3D
- NPM non-LOCA integral response
  - NRELAP5 validation against NIST-1 IETs NLT-2a, NLT-2b, NLT-15p2

Overall conclusion: NRELAP5 code, with NPM system model, is applicable for calculation of the NPM non-LOCA system response

Non-LOCA Analysis Process

Topical report Section 4

1. Develop plant base model NRELAP5 input (geometry, control and protection systems, etc)
2. Adapt NRELAP5 base model as necessary for specific event analysis and desired initial conditions
3. Perform steady state and transient analysis calculations with NRELAP5
4. Evaluate results of transient analysis calculations:
   - Confirm margin to maximum RCS pressure acceptance criterion
   - Confirm margin to maximum SG pressure acceptance criterion
   - Confirm appropriate transient run time execution to demonstrate safe, stabilized condition achieved
5. Identify cases for subchannel analysis and extract boundary conditions (if applicable)
   - Conservative bias directions:
     - Maximum reactor power
     - Maximum core exit pressure
     - Maximum core inlet temperature
     - Minimum RCS flow rate
   - NRELAP5 CHF calculations for dummy hot rod may be used as a screening tool to assist analysts in determining limiting cases to be evaluated in downstream subchannel analysis
6. Identify cases for radiological analysis (if applicable)
   - Maximum mass release case
   - Maixmum iodine spiking case

Non-LOCA Methodology

General Methodology (Section 7.1):

- Steady-state conditions
- Treatment of plant controls
- Loss of power
- Single failure
- Bounding reactivity parameter input
- Biasing of other parameters: initial conditions, valve characteristics, analytical limits and response times
- Operator action

Event-specific Methodology (Section 7.2)

- Description of event initiation and progression
- Acceptance criteria ‘of interest’
- Limiting single failure, loss of power scenarios, or need for sensitivity calculations
- Initial condition biases and conservatisms, or need for sensitivity calculations
- Tabulated representative results of sensitivity calculations

Example analysis results provided in Section 8

Conclusions

- Non-LOCA system transient evaluation model developed following a graded approach in accordance with guidance provided in RG 1.203
- Applies to NPM-type plant design natural circulation water reactor with helical coil SG and integral pressurizer
- NRELAP5 used to simulate the system thermal-hydraulic response
  - Demonstrate primary and secondary pressure acceptance criteria are met
  - Demonstrate safe, stabilized condition achieved
- System transient results provide boundary conditions to downstream subchannel and radiological analyses

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Presentation to the ACRS Full Committee

Staff Review of NuScale Topical Report

TR-0716-50350, Revision 1, "Rod Ejection Accident Methodology"

TR-0516-49422, "Loss-of-Coolant Accident Analysis Methodology"

TR-0516-49416, "Non-Loss-of-Coolant Accident Analysis Methodology"

Presenters:

Chris Van Wert – Senior Reactor Systems Engineer, Office Nuclear Reactor Regulation
Shanlai Lu – Senior Nuclear Engineer, Office Nuclear Reactor Regulation
Alex Siwy – Reactor Systems Engineer, Office Nuclear Reactor Regulation

March 5, 2020

(Open Session)

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Presentation to the ACRS Full Committee

Staff Review of NuScale Topical Report

TR-0716-50350, REVISION 1

"Rod Ejection Accident Methodology"

Presenters:

Chris Van Wert – Senior Reactor Systems Engineer, Office of Nuclear Reactor Regulation

March 5, 2020

(Open Session)

NRC Technical Review Areas/Contributors

- NUCLEAR METHODS, SYSTEMS & NEW REACTORS BRANCH / NRR:
  Rebecca Patton (BC)

- ADVANCED REACTOR TECHNICAL BRANCH / NRR:
  Jeff Schmidt
  Chris Van Wert

Non-Proprietary

Staff Review Timeline

TR-0716-50350, "ROD EJECTION ACCIDENT METHODOLOGY"

- NuScale submitted Topical Report (TR)-0716-50350, "Rod Ejection Accident Methodology," Revision 1, on November 15, 2019, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19319C684).
- Staff briefed advisory committee on reactor safeguards (ACRS) subcommittee on February 19, 2020.
- Staff plans to issue its final SER in March 2020.
- Staff plans to publish the "-A" (approved) version of the TR prior to finishing Phase 6 of the NuScale DCA.

Staff Review

- The staff's review included:
  - Evaluation of the analysis criteria
  - Evaluation of the code suite used within the analysis methodology
  - Evaluation of the plant and cycle assumptions used in the analysis methodology
  - Evaluation of the rod ejection accident analysis methodology

- The staff's review does *not* include the licensing basis Reactivity Initiated Accident (RIA) analysis for the NuScale Design Certification Application (DCA)
  - Contained in Section 15.4.8 of the Safety Evaluation Report (SER) for the NuScale Design Certification

- During its review, staff audited calculations and other supporting information

Analysis Criteria

- The staff reviewed the proposed analysis criteria
  - Reactor Coolant System Pressure
  - Fuel Cladding Failure
  - Core Coolability
  - Fission Product
- The staff concluded that the proposed criteria either followed or were conservative to the guidance provided in Standard Review Plan (SRP) Section 4.2 Appendix B
- Staff also notes that DG-1327, "Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents" is currently being developed.
  - Draft guidance is not staff requirements, but the staff notes that the more stringent internal limits imposed by NuScale would not exceed the draft guidance limits as they currently stand

Evaluation of Code Suite

- The NuScale REA analysis is based on the following codes and packages:
  - CASMO5/SIMULATE5: provides reactor core physics parameters
  - SIMULATE-3K: 3-dimensional nodal reactor kinetics code which supplies power input to downstream analyses
  - NRELAP5: transient system response
  - VIPRE-01: subchannel analysis
- Applicability of CASMO5, SIMULATE5, NRELAP5, and VIPRE-01 has been reviewed and approved for NuScale in TR-0616-48793-P-A, Revision 1, "Nuclear Analysis Codes and Methods Qualification".
- The validation of SIMULATE-3K is included as part of TR-0716-50350 and is therefore included in the staff's review.
  - Staff concluded that NuScale successfully validated S3K against experimental data and the NEACRP control rod ejection problem computational benchmark

Plant and Cycle Assumptions

- The staff reviewed the plant and cycle assumptions used in the NuScale rod ejection analysis methodology
  – The staff determined that the methodology included ranges in power, time in cycle, and core power that covered a wide range of operating conditions and would capture the most limiting condition
  – The staff agreed that the assumptions associated with the automatic system response of non-safety systems were conservative
  – The staff determined that the methodology regarding the timing of loss of AC power conservatively biases the reactor coolant system (RCS) pressure evaluation

Rod Ejection Accident Analysis Methodology

- The staff reviewed the analysis methodology including steady-state initialization, dynamic core response, dynamic system response, subchannel critical heat flux evaluation, and the adiabatic heatup fuel response
- The staff's review included the methodology by which information is passed between codes, application of uncertainties, modelling assumptions used for inputs, and handling of reactor trips.
- The staff concluded that the methodology for calculating the system response, subchannel, and fuel response analyses was conservative and acceptable for demonstrating compliance with the acceptance criteria

Staff SER Conclusions

- The staff concludes that the NuScale criteria used for evaluating REA either follows or is more conservative than staff guidance

- The staff concludes that the methodology accounts for the various potential operating conditions and time in life, and conservatively addresses uncertainties and plant conditions

- The staff finds the use of TR-0716-50350-P acceptable for evaluating reactivity initiated accidents for the NuScale plant design.

Questions?

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Presentation to the ACRS Full Committee

Staff Review of NuScale Topical Report

TR-0516-49422

"Loss-of-Coolant Accident Analysis Methodology"

Presenters:

Dr. Shanlai Lu – Senior Nuclear Engineer, Office of Nuclear Reactor Regulation

March 5, 2020

(Open Session)

Review Team

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Dr. Weidong Wang

Mr. Ron Harrington

Dr. Shanlai Lu

Mr. Antonio Barrett

Dr. Tim Drzewiecki

Dr. Syed Haider

- **NuMark Associates**

Mr. Marvin Smith

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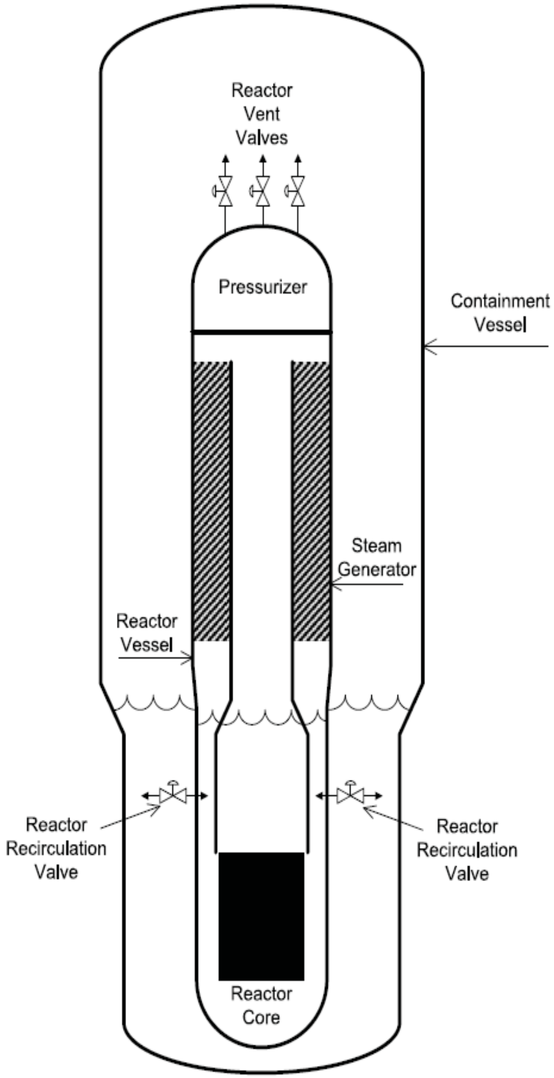
Mr. Bert Dunn

- **Brook Haven National Lab**

Dr. Upendra Rohatgi



Design Features And Scope

3 RVVs 2 RRVs each its own IAB, trip valve and trip reset valve

Containment functions as part of ECCS

- A methodology to analyze LOCA
- A methodology to analyze IORV
- Support Peak Containment Pressure, Non-LOCA TR and Long Term Cooling Analysis

Applicable Regulation:

10CFR50.46 Appendix K

Non-Proprietary

Review Approaches

- Early Engagement And Extensive Audits Through Electronic Reading Room
    - Pre-application engagement
    - Initial on-site visits and audit meetings
    - Two phases of continuing audits throughout review period
- Issues Raised:
    - 45 RAI Questions
    - 210 Audit Issues
- Staff performed sensitivity analysis with NRELAP5 and confirmatory analysis with TRACE

Review Areas

- Phenomena Identification and Ranking Table
  Following CSAU method, NuScale identified twenty-one phenomena as important to capture in the LOCA model

- NRELAP5 code is used to model NPM
  Steam Generator Model, Containment Wall Condensation Model, Critical Flow Model, CHF Correlations. NPM Model and Nodalization

- NIST Tests, Scaling and Distortion Analysis
  A new scaling analysis approach was used with distortion analysis to justify the applicability of NIST IETs

- IORV Analysis Methodology
  Two different sets of CHF correlations are used for low flow and high flow conditions. STERN and KATHY facilities provide specific fuel CHF databases

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NRC Sensitivity & Confirmatory Analyses

- Separate Effect Tests (SETs):
  - KAIST model: DHRS tube condensation experiment, non-LOCA
  - SIET model: helical coil steam generator tube/shell side heat transfer, non-LOCA
  - NIST-1 model: high pressure condensation test (HP-02)
- Integral Effect Tests (IETs):
  - NIST-1 models: loss of coolant accident (LOCA) and inadvertent emergency core cooling system (ECCS) operations

  NPM models: licensing calculation confirmation and sensitivity studies, LOCA, non-LOCA
- Both TRACE and NRELAP5 codes were used. More than fifty five sets of calculations were performed. RAIs were issued and NRELAP5 code was updated from V1.3 to V1.4. Good agreements were obtained with NuScale analysis results

Conclusions

- NuScale LOCA EM and NRELAP5 V1.4 are approved for determining critical heat flux and collapsed liquid level for NuScale reactor in compliance with 10CFR 50.46 Appendix K requirements
- NRELAP5 computer code V1.4 is also determined applicable to predict containment pressure and temperature subject to specific modeling requirements
- The CHF modeling is approved subject to limitations and conditions

Questions?

Presentation to the ACRS Full Committee

Staff Review of NuScale Topical Report

TR-0516-49416

"Non-Loss-of-Coolant Accident Analysis Methodology"

Presenters:

March 5, 2020

(Open Session)

NRC Staff Review Team

- NRC Technical Reviewers:
  - Antonio Barrett, NRR
  - Jeff Schmidt, NRR
  - Alex Siwy, NRR
  - Ray Skarda, RES
  - Peter Lien, RES
  - Ron Harrington, RES
  - Jason Thompson, RES
- Consultants (Energy Research, Inc.):
  - Mohsen Khatib-Rahbar
  - Walter Tauche (subcontractor)
  - Morgan Libby
  - Michael Zavisca

Review Process Overview

- Staff conducted its review in accordance with applicable NRC regulations and guidance
- Safety evaluation report (SER) is based on TR-0516-49416, Revision 2
- Two audits conducted in four phases
  - About 140 audit issues
  - Helped to confirm staff's understanding and inform requests for additional information (RAIs)
- 33 RAI questions issued
  - All resolved and responses incorporated into TR-0516-49416, Revision 2, as appropriate

Non-LOCA Methodology Scope

- Provides a methodology for performing system transient analysis of specified non-LOCA design-basis events for the NuScale Power Module (NPM)
- Evaluates primary and secondary pressure figures of merit
- Includes interfaces with other methodologies, both upstream and downstream
- Covers time frame during which mixture level is above top of riser and natural circulation is maintained
- Includes certain event-specific assumptions and conservative bias directions for initial conditions
- The staff is evaluating some items discussed in the TR as part of a design-specific application of the methodology

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Key Design Features and Models for Non-LOCA

- Staff focused its review on several key features of the NuScale design and their representation in the NRELAP5 model:
  - Natural circulation design
  - Helical coil steam generators (SGs)
    - Transfer heat from reactor coolant system (RCS) to feedwater
  - Passive decay heat removal system (DHRS) condensers
    - Transfer decay heat to reactor pool using the SGs
  - Evacuated containment vessel

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Applicability of NRELAP5 to Non-LOCA Analysis

- The applicant developed the non-LOCA evaluation model (EM) from the LOCA EM using graded approach described in RG 1.203
- The staff reviewed the applicant's non-LOCA phenomena identification and ranking table (PIRT) to ensure that important phenomena were identified and captured in the non-LOCA TR
- The staff reviewed how the applicant addressed highly ranked non-LOCA phenomena:
  - Separate effects tests: NIST HP-03, HP-04, KAIST, and SIET
  - Integral effects tests: NIST NLT-02a, NLT-02b, NLT-15p2
  - Code-to-code benchmark against RETRAN-3D
  - Use of bounding input values
  - Other analysis methodologies (e.g., subchannel)

Significant Review Issue – Multi-Dimensional Flow Effects

- Staff requested additional justification for how multi-dimensional flow effects in the RCS and thermal stratification in the reactor pool are addressed (RAI 9351, Question 15.00.02-31)

- Staff's major concerns were the potential for reduced RCS flow rates and degradation in DHRS performance

- The applicant's RAI response resolved the issue, as supported by the staff audit of underlying calculation notes and audit discussions with the applicant

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NRELAP5 Assessments Against Test Data

The staff finds that:

- The KAIST, NIST-1 HP-03, and NIST-1 HP-04 tests validate the NRELAP5 DHRS models
- The SIET TF-1 tests validated steam generator secondary side phenomena, but the staff had concerns about the ability of the SIET TF-2 tests to fully validate primary-to-secondary heat transfer
- The NLT-02a, NLT-02b, and NLT-15p2 integral effects tests together demonstrate applicability of NRELAP5 to evaluate non-LOCA transients
- The benchmark against RETRAN-3D provides confidence that the NRELAP5 point kinetics model produces results similar to those from an NRC-approved code

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Significant Review Issues – NRELAP5 Assessments

- The applicant removed steam generator and DHRS heat transfer biases from the methodology in response to staff questions about:
  - Steam generator heat transfer uncertainty based on the SIET TF-2 tests, associated with DCA Chapter 15 Unclear Open Item 15.0.2-4 (RAI 9466, Question 15.00.02-6)
  - DHRS nodalization (RAI 9374, Question 15.00.02-22)
- The applicant provided justification that non-LOCA figures of merit are not sensitive to these biases
- Based on its review of the justification and audits of underlying calculations, the staff finds that removal of the heat transfer biases is supported for NPM model Revision 2
- The staff imposed the associated Limitation/Condition 3

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General and Event-Specific Non-LOCA Methodology

- The staff reviewed the overall non-LOCA analysis process and finds that it provides an acceptable analysis framework
- The staff finds that the deterministic approach using conservative or bounding inputs, initial conditions, and assumptions is acceptable for conservative calculations of non-LOCA events
- The staff reviewed each event-specific methodology and ensured that they will ensure conservative results when implemented
- The staff reviewed the representative non-LOCA event calculations in the TR and concludes that they illustrate how the non-LOCA methodology can be used for conservative transient analyses

Staff SER Limitation and Condition Summary

I. Future changes to LOCA TR must be assessed for impacts to Non-LOCA EM

II. Non-LOCA EM scope limited to non-LOCA events defined in the TR prior to the time of riser uncovery for evaluation of primary and secondary pressures and potential for loss of system functionality

III. Additional justification must be provided for elimination of SG and DHRS heat transfer biases if applying methodology to a design other than NPM model Revision 2 or a model update made pursuant to a change process specifically approved by NRC for changes to the NPM model

IV. Any credit for secondary MSIVs (not safety-related) must be approved through design review

V. Event-specific electrical power assumptions, single failures, and operator actions must be approved through design review

VI. Non-LOCA EM use limited to NRELAP5 v1.4 and NPM model Revision 2, unless changes are made pursuant to a change process specifically approved by the NRC staff for changes to NRELAP5 and the NPM model

Conclusions

- All technical issues from the course of the review have been resolved
- Use of NRELAP5 with the non-LOCA methodology described in the TR is acceptable for the non-LOCA safety analyses of the NuScale NPM design subject to the specified limitations and conditions

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Acronyms

- ACRS Advisory Committee on Reactor Safeguards
- DCA design certification application
- DHRS decay heat removal system
- EM evaluation model
- LOCA loss-of-coolant accident
- NPM NuScale Power Module
- PIRT phenomena identification and ranking table
- RAI request for additional information
- RCS reactor coolant system
- RIA Reactivity Initiated Accident
- SER safety evaluation report
- SG steam generator
- TR topical report