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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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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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NuSCALE SUBCOMMITTEE

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THURSDAY

MAY 16, 2019

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ROCKVILLE, MARYLAND

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The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2D10, 11545 Rockville Pike, at 8:30 a.m., Gordon R. Skillman and Michael L. Corradini, Co-Chairs, presiding.

COMMITTEE MEMBERS:

MICHAEL L. CORRADINI, Co-Chair

GORDON R. SKILLMAN, Co-Chair

RONALD G. BALLINGER, Member

DENNIS BLEY, Member

CHARLES H. BROWN, JR. Member

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JOY L. REMPE, Member

MATTHEW W. SUNSERI, Member

ACRS CONSULTANT:

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

MIKE SNODDERLY

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

P-R-O-C-E-E-D-I-N-G-S

8:29 a.m.

CO-CHAIR SKILLMAN: Ladies and gentlemen, good morning. This meeting will come to order.

This is a meeting of the Advisory Committee on Reactor Safeguards, NuScale Subcommittee. I'm Gordon Skillman, Co-Chairman for today's subcommittee meeting along with Mike Corradini. Members in attendance today are Dr. Michael Corradini, Dr. Joy Rempe, Dr. Vesna Dimitrijevic, our esteemed Mr. Charlie Brown, Matt Sunseri, our Vice Chairman, and Dr. Ron Ballinger. We have our consultant Dr. Stephen Schultz with us, too. Mike Snodderly is the designated federal official for this meeting.

The Subcommittee will review the staff's evaluation of Chapter 14, Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria, and Chapter 3.9.2, Dynamic Testing and Analysis of Systems, Components, and Equipment of the NuScale Design Certification Application.

Today we have members of the NRC staff and NuScale to brief the Subcommittee.

I will say in opening that if we are swift in moving through Chapter 14 ITAAC, we will move directly into 3.92 to conserve time and resources.

The ACRS was established by statute and is governed by the Federal Advisory Committee Act, FACA. This means that the Committee can only speak through its published letter reports. We hold meetings to gather information to support our deliberations. Interested parties who wish to provide comments can register -- can contact our office requesting time after the meeting announcement is published in the *Federal Register*. That said, we set aside 10 minutes for comments from members of the public or those listening our meetings. Written comments are also welcome.

Important at this point is the comments today represent the comments of individual members, not the ACRS. And the point of this paragraph is that we speak only through our letter reports that come from our Full Committee meetings. So comments today are from individual members.

The ACRS section of the U.S. NRC public web site provides our charter, bylaws, letter reports and full transcripts of all Full and Subcommittee meetings including slides presented there.

The rules for participation in today's meeting were announced in the *Federal Register* on May 6th, 2019. The meeting was announced as an open/closed meeting. We may close the meeting after the open portion to discuss proprietary material. Presenters can defer questions that should not be answered in the public session at that time.

No written statement or request for making an oral statement to the Subcommittee has been received from the public concerning this meeting.

A transcript of the meeting is being kept and will be made available as stated in the *Federal Register* notice. Therefore, we request that participants in this meeting use the microphones located throughout the meeting room when addressing the Subcommittee. We ask that the participants first identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

For those in the meeting room, we ask that you please silence your electronic devices.

We have a bridge line established for the public to listen to the meeting. To minimize disturbance the public line will be kept in a listen-in mode. To avoid disturbance again I request that all attendees put their electronic devices in the noise-free mode.

We will now proceed with the meeting and I'll call on Zach Rad, Director of Licensing of

NuScale, to begin today's presentation.

Sir?

MR. RAD: Thank you. Carrie's going to -- Carrie Fosaaen is going to provide introductions to this presentation.

MS. FOSAAEN: Thanks.

Good morning. Carrie Fosaaen, NuScale Licensing Supervisor, for Chapter 14. Today we've got Chris Maxwell and Edan Engstrom who will be presenting an overview of our Chapter 14, which is Initial Test Program and ITAAC.

Chris?

MR. MAXWELL: Good morning. Chapter 14 consists of three sections: Section 14.1 contains --

CO-CHAIR SKILLMAN: Chris, would you make sure your green light is on?

MR. MAXWELL: Both?

CO-CHAIR SKILLMAN: Okay. You got stereo. There you go. Thank you, sir.

MR. MAXWELL: Yes sir.

CO-CHAIR SKILLMAN: Thank you.

MR. MAXWELL: So Chapter 14 has three sections: Section 14.1 contains specific information that's to be addressed in the Initial Plant Test Program, Section 14.2 is the Initial Plant Test

Program, and Section 14.3 is the certified design material inspections test analysis and acceptance criteria.

CO-CHAIR SKILLMAN: May I make a comment here?

MR. MAXWELL: Yes sir.

CO-CHAIR SKILLMAN: For 14.3 what I'm going to request is not specifically identified in your slides, but I would like to communicate what we as a committee are interested in.

We're interested in the organization of ITAAC. There is Tier 1 ITAAC and Tier 2 ITAAC. And we understand that Tier 1 ITAAC would be material that a future COLA applicant would require use of a departure to change. In other words, Tier 1 is locked in a license changed only by departure and Tier 2 is the governing detail for that particular ITAAC. So there are really two tiers that we're talking about in your application. So we'd like to through this discussion perhaps hear the distinction between those two if that's relevant to what you may be presenting.

We're also interested in the different types of ITAAC. There are apparently seven different types of ITAAC. There's as-built analysis, as-built inspection, design acceptance criteria, design analysis, equipment qualification ITAAC, pre-operational test ITAAC, and vendor test ITAAC. We know you can't address all of those now, but to the extent that that discussion may be relevant we'd like to hear about that.

MR. MAXWELL: Yes sir.

CO-CHAIR SKILLMAN: We also would like to hear about overall accountability. This is a massive application; multiple systems, only a few of very high safety significance. How is the entire application combed to ensure that all ITAAC are accounted for? When one reviews the ITAAC Tier 1 and Tier 2, the ITAAC are in table after table after table leading to the question how do you know that's thorough and what controls are on those data bases to ensure everything that needs to be identified has been identified? And then further, how closeout is going to occur.

MR. MAXWELL: Yes sir.

CO-CHAIR SKILLMAN: Finally, how is ITAAC accounted for as the design matures? As we saw in the CDF curve yesterday, the CDF has been consistently decreasing by the improvements that have been made in this application. Our sense is the drivers for that reduction in CDF are probably accounted for somewhere down in the design and those items probably rise to the ITAAC level. How are changes in ITAAC accounted for as the design continues to mature?

MR. MAXWELL: Understood.

CO-CHAIR SKILLMAN: I's a lot of words, but it really comes down to recordkeeping, accountability, your QA Program oversight on ITAAC and ensuring that when the application is finally ready for approval that the deck is complete.

MR. MAXWELL: Yes sir.

CO-CHAIR SKILLMAN: With that, proceed. Thank you.

MR. MAXWELL: And I'll do my best to address each of those as I go through this where there's an appropriate location to discuss them, but please feel to remind me of any that I fail to discuss as we go through it.

CO-CHAIR SKILLMAN: Yes sir. Thank you.

MR. MAXWELL: Because Section 14.1 is essentially a list of documents, we'll jump right into Section 14.2, the Initial Test Program.

The Initial Test Program consists of three major categories of testing: The first is pre-operational testing, startup testing and then first-of-a-kind testing. Startup testing can be further broken down into initial fuel loading and pre-critical testing, initial criticality testing, low-power testing and power-ascension testing.

Guidance for the contents of the Initial Test Program come from Regulatory Guide 1.68, and Reg Guide -- regarding pre-operational testing specifically, Reg. Guide 1.68 states that pre-operational testing consists of those tests conducted following completion of construction, inspections and tests, but before fuel loading to demonstrate to the extent practical the capability of SSCs to perform the performance requirements to satisfy the design certification.

Also in Reg Guide 1.68, Appendix Alpha, there's a list of systems and attributes for each of those systems to be verified or demonstrated during conduct of the pre-operational testing.

So NuScale evaluated each of those systems and their attributes for applicability to the NuScale design and incorporated them into our pre-operational testing where appropriate, however, we recognize that the time that Reg Guide 1.68 was written they couldn't anticipate all of the design features and system attributes that would be included in the NuScale design. So we wanted to -- we sought after a method to identify all the functionality to be tested to ensure that we adequately demonstrated the design.

The answer came to us in the Design Reliability Assurance Program, or D-RAP. D-RAP is described in Chapter 17.4, but what D-RAP does for us is it provides a list of all the inter-system functions for each system and then describes those functions in a support-system-to-supported-system relationship, and then finally classifies the risk and safety significance for each of those functions.

So what we did was then we took each of those functions and we asked the question is this function testable? And if the answer was yes, we first tried to align it with ITAAC testing requirements. So one of the aspects of ITAAC that you mentioned as there's pre-operational testing.

I'm going to step out just a second and kind of go into ITAAC for a second to say that it's not prescribed whether it be an analysis or a pre-operational test or equipment qualification test to satisfy the ITAAC, but rather you first identify the design commitment and then identify the appropriate inspection test or analysis to be conducted to demonstrate that you do meet that design commitment. When it's testing sometimes it will line up with pre-operational testing.

If I need to as part of a design commitment verify that my containment isolation valves stroke closed within a specific time, well, I'm also going to do that as a part of pre-operational testing. So we'll record it in both programs, in both the Initial Test Program and as part of ITAAC, but perform the test once.

CO-CHAIR SKILLMAN: Would you go back a slide, please?

MR. MAXWELL: Yes sir.

CO-CHAIR SKILLMAN: At your first -- the second carat, Reg Guide 168, Pre-operational Testing, for the record as you receive a module -- or let me say it differently. You've got a module that's, if you will, ready for sign off at your manufacturing facility, wherever that might be. My presumption is at that point, that vendor has conducted a series of tests particularly related to ASME and pressure testing and that type of thing. That vendor might have done some additional testing that is the cousin of, the brother or sister of, or the same as what you would insist be completed as part of your pre-operational testing.

So my question is, where in the supply chain would a vendor conduct tests that satisfy ITAAC?

And when that module is in its operating location for its pre-operational test, does the -- or conducted at the manufacturer or waived because they have been conducted at a prior time?

MR. MAXWELL: We identify some tests that can be satisfied by the vendor to -- and those tests will demonstrate ITAAC. When it comes to pre-operational testing, it's a very limited scope because the purpose of pre-operational testing is to demonstrate really the adequacy of the construction. So there's some factory acceptance testing in the module protection system for instance that will credit for both ITAAC and as a prerequisite to be completed to commencing the pre-operational testing. But that is identified in the pre-operational -- in the initial test program.

So it's not a matter of waiving pre-operational testing. It's still -- you still conduct all the pre-operational testing. But again, that testing --

CO-CHAIR SKILLMAN: Do those records accompany that module?

MR. MAXWELL: Yes sir.

CO-CHAIR SKILLMAN: Are those records protected under your QA program or your vendor's QA program?

MR. MAXWELL: They'll be the licensee's QA program.

CO-CHAIR SKILLMAN: So there will be a chain of custody for those records for that module?

MR. MAXWELL: Yes, there will be.

CO-CHAIR SKILLMAN: Yes sir. Okay. Thank you.

MR. MAXWELL: And kind of going into another question you mentioned was about the closure of that ITAAC. So if I have an ITAAC again for the module protection system or we'll go back to the module itself. If I've got an ASME inspection that's required and it's an ITAAC for the valve, that could be conducted at the manufacturer's location. All the closure paperwork for the ITAAC could be identified and an ITAAC closure notification could be submitted to the NRC to close that ITAAC prior to the module arriving on site.

CO-CHAIR SKILLMAN: Okay. Thank you.

MR. MAXWELL: Yes sir. So again, when possible, we align pre-operational testing with ITAAC. Then we add those functions. Those testable functions are added to the initial test program and we develop component-level or system-level tests for those functions as appropriate, again, to demonstrate the ability of the SSC -- the capability of the SSC to meet the performance requirements to satisfy the design criteria.

So I'd like to use an example here. This is a D-RAP function that's identified in Chapter 17 associated with a reactor building crane.

CO-CHAIR SKILLMAN: Just a minute. Would you go back a slide, please. What you have here on the red, orange, yellow, and green, was that 50.69, the four boxes? Is that what that is?

MR. MAXWELL: I can't speak to that. It's the D-RAP designation.

CO-CHAIR SKILLMAN: I was just curious. I believe that that's what that is. Red is supposed to be, hey, people, this one is really, really important. Orange is kind of important but not as important as red. Yellow is, eh, and green is, don't worry about it. Is that what we're looking at here?

MR. MAXWELL: I would not assign that to those colors necessarily, that they have that amount of weight. But rather just to differentiate between safety significant -- I'm sorry, safety related and risk significance.

CO-CHAIR CORRADINI: I guess what he's

getting at is, is the level of testing change with the -- that's what I thought you were going with it.

CO-CHAIR SKILLMAN: Go ahead.

CO-CHAIR CORRADINI: But my impression is all of these are just categorized, and then you have to decide what you want to test to demonstrate function below.

MR. MAXWELL: Right. What I would say is that with the safety related and risk significant functions, those functions have ITAAC associated with them. The non-safety related, non-risk significant functions may or may not have ITAAC associated with them. However, if the function is testable, it's tested as part of the initial test program. There's no distinction made as far as pre-operational testing but there is for ITAAC.

CO-CHAIR SKILLMAN: Are all the reds Tier 1?

MR. MAXWELL: Every safety related testable -- safety related will be found as an ITAAC.

CO-CHAIR CORRADINI: In Tier 1?

MR. MAXWELL: That's correct, in Tier 1 -- ITAAC for Tier 1.

CO-CHAIR SKILLMAN: Thank you.

MR. MAXWELL: Yes sir. So again with this example here, we're actually looking at a non-safety related risk significant function associated with reactor building crane. And as described a moment ago, the D-RAP functions are provided to us in a support system to support the system format.

So this function for the reactor building crane is that the reactor building crane supports the NuScale Power Module by providing structural support and mobility while moving from refueling, inspection, and operating bay.

Now if we go to the initial test program itself in Section 14.2, we find Table 14.2-52 titled, Reactor Building Cranes Test No. 52. This is the test abstract for the reactor building crane.

The D-RAP function, we just discussed. It was identified to be a testable function, and so it is inventoried in this test abstract. Each of the testable functions will be inventoried on that test abstract.

In addition to the function, we have the categorization of safety and risk significance as well as a column that lists the test or tests that demonstrate that functionality. And this is where NuScale's test abstracts are significantly different than previous applicants and that we provide this roadmap so that for each function you can go and look at the test that specifically verify that functionality.

Also want to point out to -- there was a question yesterday about pre-operational testing about whether it's performed once or for each module and how that occurs. I'll point out the first line after the title of this test abstract that states that pre-operational test is required to be performed once unless otherwise noted in the test. The cue to say, depending on the test will tell you in the details of the test whether or not you need to do it once or more than once. And I'll elaborate on that here in a moment.

The next section of the --

DR. SCHULTZ: Excuse me, Chris. How do you differentiate -- and the figure wasn't clear. But as you go through the item by item description as an example, how is the differentiation done between what's required for the component test and what's required for the system test?

MR. MAXWELL: I'll get to a description of the component systems test here in just a moment.

DR. SCHULTZ: Excellent. Thank you.

MR. MAXWELL: The next section of our test abstract is the prerequisites. The prerequisites are required to be completed prior to commencing the pre-operational testing. So we don't find them listed. Here we see for the reactor building crane there's some site acceptance testing that's required to be completed. And then we have some various rated-load tests are required to be completed and approved. And last, a prerequisite that all the instrument calibration for the reactor building crane must be completed prior to commencing pre-operational testing.

The next section is component-level testing. Component-level testing is standardized to provide an equal level of detail across the systems. So what you'll find depending on which system you're looking at, we have a standard set of component-level test that we apply to the systems.

So you'll find items like verification of remote operation of equipment, manual control of variable speed pumps and fans, verification that pump operation doesn't result in water hammer, equipment response to -- automatic equipment response to signals for plant equipment protection, and verification of instrumentation signals designed to be monitored in the control room.

So with that concept of component-level

testing, you look at the reactor building crane. And the component-level test that we end up with are the verification of the reactor building crane controls, verification of the response of the system to abnormal condition signals, and a verification of all the reactor building crane instrumentation signals designed to be monitored in the control room.

So that's the component-level piece. The next section is --

CO-CHAIR SKILLMAN: Chris, what is your estimate of the number of component-level tests that are required for this design?

MR. MAXWELL: It'd be hazardous to guess a number. But what I can do is say that for each pump we test the ability to remotely operate that pump if it has remote operation. So in the feed and condensate system, there's six pumps. For each valve that could be remotely operated, we verify that we can stroke that valve remotely. So every valve in each of the systems that's true. The instrumentation signals we verify for every system. The NuScale design has significantly fewer components in it compared to a traditional design. But each of those undergo component-level testing.

CO-CHAIR SKILLMAN: So it your estimate

100, 1,000, 10,000?

MR. MAXWELL: You have a number for me, Edan? I'd measure it in thousands.

CO-CHAIR SKILLMAN: I'm thinking four or five thousand number.

MR. MAXWELL: I think that's a reasonable number.

CO-CHAIR SKILLMAN: That's a guess.

MR. MAXWELL: I have a database that has that information, but I don't have access to it.

CO-CHAIR SKILLMAN: Well, I think that information is important in the context that this is a new design, a very conservative and robust new design. And at least from my experience, you're doing some things that nobody has ever done before. And the devil is in the details.

And I think of startups that I've been involved in and the pre-operational testing I've been involved in and how many times two years later we've gone back and said, where's the record for that? We can't find it. And you're in a unique position to be able to have those records for what will be your future customers.

So as unpopular as recordkeeping is in this business, it is an essential part of the data set that the new owner is going to inherit. And if that new owner has the right paper for each module and for each subsystem, for each CVCS, for each evacuation system, that that customer has a legacy that enables that customer to be successful.

And I'm preaching to the choir. But all I'm saying is that number is an important number, and the quality of the information in that pile of paper, the quality of the information is really important.

And I should've mentioned Dr. Dennis Bley has joined the team. Dennis, thank you. Please proceed.

MEMBER BLEY: Thanks. I mean, you said all the valves, they can be remotely operated or stroked. I haven't seen detailed P&ID. So I don't know where you have manual isolation valves. But there must be some and there must be some on systems you really want to be able to put water through. Are they not part of the test program?

MR. MAXWELL: They're part of construction testing. Manual valves were verified as part of construction testing.

MEMBER BLEY: But not with flow at that point?

MR. MAXWELL: Well, there will be some.

There'll be flushing of systems and hydros required through construction testing. But we also have system-level testing where we flow water through the systems --

MEMBER BLEY: Okay.

MR. MAXWELL: -- or air through the systems.

MEMBER BLEY: Okay. So checking on those.

MR. MAXWELL: Yes sir.

MEMBER BLEY: I've seen -- well, you all know anecdotes. But with all the QA and everything else, we had a valve that wouldn't open and fought it and fought it and fought it. It wasn't right when it was put in. It was a little time later. Finally took it apart and here somebody during construction had stuffed the bonnet full of things that prevented it from moving more than a tiny bit.

MR. MAXWELL: Understood. And we're certainly listening to the OE, that we're hearing from AP1000 and incorporating that in our test program. The prerequisite you saw were the prerequisites required. This is the test abstract phase. There's the detailed procedure -- test procedure phase that occurs later from these test abstracts to verify items like you're suggesting to make sure that these valves operate.

MEMBER BLEY: As long as you have full flow. A full system test, I think that covers it.

MR. MAXWELL: That's correct.

MEMBER SUNSERI: So Chris, let me add. I think you're describing this, but I just want to hear it confirmed, right? So you have the Reg Guide for pre-operational and initial to start up this thing and all that stuff. You guys are following it without exception, right? The NuScale design may be unique, but it's not unique that it has pumps. It's not unique that it has valves. It's not unique that it has breakers. It's not unique that it has control systems. And that's what this, the Reg Guide, describes how to test, right?

And so you are following -- so the question is, are you following without exception unless as applicable the Reg Guide for a pre-operational test?

MR. MAXWELL: Yes sir, we are. What we recognize was the need to go beyond the Reg Guide because again of those unique design features and attributes. You're right, spot on with the pumps and valves. The guidance is clear, and we meet that guidance. We follow it. It's just what we've done in our test abstracts is look ahead.

And because we've had the benefit of having a P&ID, detailed P&IDs and procedures ahead of time, we're able to be a little more -- again that roadmap. We're able to provide that roadmap, a detailed look at how exactly we intend to demonstrate the performance of the system.

CO-CHAIR SKILLMAN: Maybe this would be a good place -- no, let's wait till 14.3 and I'll ask my question. Go ahead.

MR. MAXWELL: The next section after component-level testing are the system-level tests. And they are used to demonstrate and verify integrated functionality or functionality at the system level rather than just the component level.

So we're looking at the first system-level test now for the reactor building cranes. It's System-Level Test 52-1. And to kind of elaborate a little bit the difference between the component testing and the system-level testing. Again, the component-level testing, I'm verifying this pump works, this valve works. Here at the system-level test, now I'm demonstrating the overall capability of the system.

CO-CHAIR SKILLMAN: When you say, works,

I think you mean it is functionally successful for the task intended.

MR. MAXWELL: Yes sir, that's much better put. We have a set of parameters, for instance, pump curves that -- or valve stroke times and we verify the components meet those requirements, that design criteria.

CO-CHAIR SKILLMAN: Thank you.

MR. MAXWELL: For Test 52-1, what you see is a -- it's a test of the ability to install and remove a module from its operating bay. To the discussing whether a test needs to be performed once or multiple times, reactor building crane is a good example of this in that reactor building crane itself has attributes that need only be tested once. In other words, that attribute will be used identically for all 12 modules.

But there are other attributes which is installing the module in the bays that are module specific. So I need to be able to demonstrate the capability of the crane to not only correctly lift Module 1 in its operating bay and move it but also Module 6.

So what we do in the system-level test, if you see in the test method at the bottom, it says, repeat this sequence for teach NPM installation. So I'd have to repeat this test 12 times. But in the acceptance criteria, the first four acceptance criteria are specific to the crane. The involve the ability to move the bridge and trolley, verify the speeds of movement and the ability to operate the main hoist and the limitations associated on the hoist.

So Acceptance Criteria 1 through 4, there's a note at the bottom. Acceptance Criteria 1 through 4 only need to be satisfied for the first performance of this test. Now by satisfied, we mean documented for a completion of this test. You'll still observe those characteristics each time you perform the test. But to document completion for the initial test program, you'll do it for the first test.

The Acceptance Criteria 5 and 6 again are that module-specific piece verifying that the module is positioned at the correct location, module-specific. So Acceptance Criteria 5 and 6 need to be satisfied for each module.

If we go back and look at the reactor building crane function that we were looking at, we see that one of the elements of that function was satisfied by 52-1 -- System-Level Test 52-1 and that's the operating bay portion.

The remaining elements of the function are demonstrated in System-Level 52-2 which demonstrates the ability to assemble and disassemble a module. Because this process, once the module is moved to the stands, whether it be the containment or the reactor stand, the functionality of the crane at that point is identical module to module. So the system-level test is performed once.

CO-CHAIR CORRADINI: Maybe this you're going to get to, but I'm kind of curious. So you have -- you build out the reactor pool, install all the pieces. You're going to now install Module 1. Is there an estimate for the time it takes to perform both component and system-level testing before -- because the time I bring in the module and then actually take it to fuel load? Or I'm asking that wrong because of the way this thing words. Before I take it to bring it to critical?

MR. MAXWELL: We're working on those estimates at this time.

CO-CHAIR CORRADINI: Given, I guess, the way you describe it, it sounds like it would be potentially longer than what we'd have in a large light water reactor. That's not the case because of differences in number of testing. I'm just trying to understand. This seems fairly comprehensive to me, but I'm trying to understand.

MR. MAXWELL: Understood. I believe significantly shorter than what you would have in a traditional design. The number of components is significantly lower and --

CO-CHAIR CORRADINI: Is part of it, to what Dick was asking, relative to factory acceptance testing that is appropriately documented and is in lieu of or is done prior to actually the module arrival?

MR. MAXWELL: A portion, very small portion.

CO-CHAIR CORRADINI: Okay.

MR. MAXWELL: I'd say the module protection system is really one of the larger factory acceptance testing systems.

CO-CHAIR CORRADINI: Okay.

MR. MAXWELL: But the reason I would say it's a shorter amount of time relative to the other --

CO-CHAIR CORRADINI: Is just number?

MR. MAXWELL: Number and then complexity of systems and also scale. It's something I have to continuously remind myself of is the scale of these systems is much smaller.

CO-CHAIR CORRADINI: But you're going to do it in time?

MR. MAXWELL: That's correct. But I would say the design was made with the benefit of operating experience. I'll take, for example, we create testing that the valves -- in a traditional design performed that we create testing, you may have a very large volume that you need to pressurize, allow that volume to stabilize.

Where ours, if we're talking about a two-inch line with the containment isolation valves integral to the same body with a predesigned connection to accomplish the testing. So the system was designed anticipating the need to perform that testing and get very small volumes to press up and stabilize, significant reductions in test time.

CO-CHAIR CORRADINI: Okay. Thank you.

MR. MAXWELL: So the -- again, just to close that loop, that the function that we were looking at for the reactor building crane. The remaining elements of that function are verified through the second system-level testing group and the reactor building crane test.

Next section after pre-operational testing is the startup testing. Regulatory Guide 1.68

describes startup testing as equipment performance tests completed during and after fuel load and outlines in detail the testing to be performed for -- again, for initial fuel load and pre-critical testing, initial criticality, low power testing, and power ascension testing.

CO-CHAIR CORRADINI: Is this where the ECCS valves are tested, or is it also in the prior component testing where they're tested?

MR. MAXWELL: The answer to your question is yes in both areas. We have pre-operational testing. We do it cold to verify the stroke times of the ECCS valves, then hot functional testing.

CO-CHAIR CORRADINI: Which is?

MR. MAXWELL: Hot functional testing is without nuclear heat. We use a module heating system and a chemical and volume control system to heat up our module to normal operating pressure and as high a temperature as we can achieve under the module heating system.

CO-CHAIR CORRADINI: Okay.

MR. MAXWELL: And then in those conditions, we will stroke the ECCS valves again.

CO-CHAIR CORRADINI: Okay.

MR. MAXWELL: And then do we performance

for -- I don't know if we stroke ECCS for startup testing.

CO-CHAIR CORRADINI: I doubt it.

MR. MAXWELL: I do too, not at the normal operating pressure and temperature. We do lower --

(Simultaneous speaking.)

CO-CHAIR CORRADINI: I see. And then when it's appropriate, I'm kind of curious. So the DHRS is primarily startup testing?

MR. MAXWELL: DHRS, we also do the same hot functional testing.

CO-CHAIR CORRADINI: Because they're valves?

MR. MAXWELL: That's correct. And also to demonstrate to a degree to ability to cool us down below 420 degrees, again in pre-operational space. And then in DHRS, we do in startup testing as well.

CO-CHAIR CORRADINI: Okay. Thank you.

MR. MAXWELL: So with the very detailed requirements or suggestions, recommendations for testing these various stages of startup testing, again, NuScale evaluated each of those for applicability and added those to our initial test program where applicable. And once again, looked for any differences, something that was not anticipated for our design. And a good example of that would be an island mode operation which we do for performance as a part of startup testing.

CO-CHAIR SKILLMAN: Chris?

MR. MAXWELL: Yes sir.

CO-CHAIR SKILLMAN: Chris, what is the path forward when a licensee is conducting this testing and the testing does not go as planned? And the root cause is determined to be either the test was not as it should've been. It wasn't accurate or wasn't, if you will, crafted properly or there is an unexpected phenomenon in the NuScale design that led to the failure of the test. What is the path forward from there?

MR. MAXWELL: As part of the initial test program, we have a startup administration manual that outlines what procedures and policies are required for the initial test program. One of those is that there's essentially a board that oversees the testing. If you have a failed test and the issue was the test itself, if there needs to be a revision to the test, there's a process for revising that test and approval all the up through the board prior to recommencing or re-conducting that test.

As far as it being a design issue, again that's addressed as part of the startup administration manual. There will be a process for backing out of that test and addressing it with -- I assume with the vendor, with NuScale.

CO-CHAIR SKILLMAN: I'm glad you said that because it may be that the only intellectual reservoir of risks in Corvallis for the resolution of that problem, I bring to your attention the condensate return issue that was discovered on the AP1000. When that is finally sorted out, it really does come back to the NSSS vendor. And the NSSS vendor, for better or for worse, has a unique accountability for technical detail that's well beyond the applicant's ability to examine.

So I appreciate your saying that may come back to the NSSS vendor. If it does, might that result in a departure because it could be Tier 1 information?

MR. MAXWELL: Potentially. I can't say with a certainty. But given the detail that is in Tier 1, it absolutely could result in departure.

CO-CHAIR SKILLMAN: Okay. Thank you.

MEMBER SUNSERI: I presume you would run those kind of issues, test failures, whatever, through your corrective action program for a root cause, extent of condition, corrective action, all that kind of stuff?

MR. MAXWELL: It would definitely meet the definition of entering the corrective action program.

MEMBER SUNSERI: Okay.

CO-CHAIR SKILLMAN: One more. If you're in that position -- and this is really theoretical. So excuse me for being -- trying to be clairvoyant. I'm not. I'm just thinking it through.

So here you are eight years from now. An applicant is working his or her way through this, and we've discovered a failed test. And it really does call into question, if you will, the function performance requirements or the architecture of the test and it results in some serious questions. Does this get handled through a 50.59-like process where an applicant would say, well, this doesn't really -- it really doesn't cross the line into the license, or, it does and I better go and file the paperwork for 50.59 with the NRC?

MR. RAD: So the short answer -- this is with the licensing answer -- is it depends. It depends on what changes. So to the extent that the test impacts Tier 2, it may be a departure. And for clarification, changes to the information in Tier 1 would be an exemption because that is the information certified in the design certification rule.

CO-CHAIR SKILLMAN: Thank you. Exactly right. Thank you.

MR. RAD: So it may be a license amendment request. It may be an exemption to the Tier 1 information. Or it may simply be a departure from the Tier 2 information that's handled without prior approval from the NRC if it doesn't meet that threshold.

CO-CHAIR SKILLMAN: Great, thank you. And thank you for that clarification.

MR. MAXWELL: The test abstracts associated with the startup testing are similar in content but slightly different in layout because of the different nature because there isn't a component-level feature to them. Essentially, the startup tests become an individual system-level test if you will. That's what parallels, but although the startup tests involve multiple systems.

Example of that we see here is this is Test 81 of Control Rod Assembly Full-Height Drop Time Test contains those elements of the objectives, the prerequisites, test method, and the acceptance criteria for that test.

Again, I want to point out that as with each of our test abstracts, the first line, it states that this startup test is required to be performed for each NPM. Not surprisingly, but the control rod assemblies are tested to be module specific.

CO-CHAIR SKILLMAN: Why doesn't that communicate for each NPM, comma, for each core?

MR. MAXWELL: The initial plant test program is completed prior to initial -- sorry, for your initial startup. And then after that, you're in your in-service testing and you'll have testing associated with control rods and that program.

CO-CHAIR SKILLMAN: Okay. Thank you. All right.

MR. MAXWELL: Another, also addressed by Reg Guide 1.68 are first-of-a-kind test which Reg Guide 1.68 defines as tests that are new, unique, or special tests used to verify design features being reviewed for the first time by the NRC.

We did a comb of our systems, their features, and the components and attributes to identify any first-of-a-kind testing and constructed Table 14.2-110 to list those new design features. Included in those first-of-a-kind type test is like our ECCS valve design first-of-a-kind, a containment evacuation system, and our island mode operation.

CO-CHAIR SKILLMAN: Before you jump in here on 14.3, you've introduced us to, I think, an excellent example with a crane and sufficient reference, if you will, of plumbing systems. You haven't said much about instrumentation and control system ITAAC or electrical system ITAAC. May I ask you to comment on those two categories, please.

MR. MAXWELL: Sure. First of all, I want to make a distinction between the initial test program and ITAAC. They really are separate items occasionally that overlap through pre-operational testing. So Reg Guide 1.68 describes the pre-operational testing requirements and then we have ITAAC which is a 10 CFR requirement.

As far as elements of electrical distribution, all of the functions in our electrical distribution system, AC and DC, are all part of -- there's a test abstract for each of them. We test every testable function of those systems. The same is true for our I&C systems, the module control system, plant control system, plant protection system, and the module protection system.

One of the things I mentioned earlier was the factory acceptance testing and site acceptance testing. Being the digital systems that they -- or being the control systems that they are, that there's an ITAAC also for module protection system in addition to the pre-operational testing. So there's overlap there. But again, we have a test abstract for each of them and the functionality is verified through either factory and site acceptance testing or through a demonstration.

When you get to the -- the demonstration comes at the device. The module protection system may have the -- it has a function to close the containment isolation valves. So the factory acceptance testing and the site acceptance testing will demonstrate all the logic, all the input and outputs for the logic. But we still have to verify that given a signal from the module protection system that the actuated component actuates.

So the records will be verified to open the containment isolation valves, to close ECCS valves, to get heat removal system actuation valves, all part of our pre-operational testing program also happen to be verified as part of ITAAC.

CO-CHAIR SKILLMAN: Thank you.

MR. MAXWELL: The final section of Chapter 14 is 14.3, Certified Design Material.

MEMBER BLEY: Is that just a requirement, or how does it happen that we have these both under pre-op testing and ITAAC?

MR. MAXWELL: It's --

MEMBER BLEY: It seems like a bit of a burden to track it. Or do you build that right into your pre-op test, the ITAAC verification?

MR. MAXWELL: They are two separate requirements, and so they're tracked separately. But like I said, where you can pre-operational test to satisfied an ITAAC, we'll do that. We do that.

MEMBER BLEY: I was just thinking how you'll provide this to the COL people. And I would hope that at least somehow the ITAAC -- appropriate ITAAC are tied to the test -- the initial test procedures.

MR. MAXWELL: So --

MEMBER BLEY: Otherwise, that seems a nightmare of keeping track.

MR. MAXWELL: Absolutely. We recognize that. And we have two things that we've done to help with that. The first is that where I say where we can credit a pre-operational test for ITAAC, we do. If you go to that test abstract, the acceptance criteria has the ITAAC number --

MEMBER BLEY: Right there?

MR. MAXWELL: -- listed right in the acceptance criteria. That's the first flag. We also have, again, a roadmap from ITAAC and Tier 1 over to Tier 2, Chapter 14.3. For each ITAAC, if there's a pre-operational test associated with it, the description of that ITAAC says, this ITAAC will be verified by pre-operational --

MEMBER BLEY: Okay.

MR. MAXWELL: -- tests, and points to it. And the second thing that we're doing is ITAAC travelers. So ITAAC can be a many-to-one, one-to-many, or many-to-many relationship.

I may have -- I'll use containment isolation valves as an example where I have a single ITAAC to verify that I've got a ASME design report for each of the containment isolation valves. So many valves have to have a report to close that ITAAC. We have -- using database, have linked each of the components from our MEL (phonetic) to the individual ITAAC to track closure for that ITAAC.

But also for each of the components, a containment isolation valve will have multiple ITAAC it's associated with. So you also can see for a given component all the ITAAC that have to occur for that component.

We also recognize that ITAAC will occur at different phases. Some of them will happen in vendor, like equipment qualification testing. Some of them will happen during the module manufacture. Some will occur during pre-operational testing. And again, the containment isolation valve, that's true, all three of those phases. So we recognize which phase that ITAAC is to occur and our ITAAC travelers it identifies at each phase, which ITAAC --

MEMBER BLEY: Somehow, you've --

MR. MAXWELL: -- will be completed.

MEMBER BLEY: -- built this into a computer-based system, I assume, to help people along. I've heard a lot of complaints from people trying to deal with the massive amount of ITAAC information. I just wondered if the guidance that's out there has gone further than the requirements and if you couldn't do this in a simpler way.

MR. MAXWELL: I'm sure. The database helps us, that we use the operating experience from the current plants to inform us and to develop tracking mechanisms to ensure that we don't miss ITAAC.

MEMBER BLEY: Okay, sounds good.

CO-CHAIR SKILLMAN: Chris, you've touched on several of the key issues that I identified in my opening comments. You just mentioned several different types of ITAAC. You just mentioned how it is captured in your database. Going forward, I will be interested in hearing how ITAAC accountability, how the roundup is certified thorough for the various -- appear to be about seven different types. How all of them are called and accounted for so that at the end of the process the license shows completion. So as you proceed, I just would ask you to keep that in mind.

MR. MAXWELL: Sure. I think I have a slide that'll help with that. What I want to walk us through is how the detailed design information, that Tier 2 information gets selected to be included in our Tier 1 Design Descriptions, including the design commitments and how those design commitments form our ITAAC.

So I'll start with the purpose of Section 14.3. One of the purposes is to provide the guidance for selecting that detail design information in Tier 2 to be included in the Tier 1 certified design material, including the ITAAC required by 10 CFR 25.47(b)(1).

As a reminder of what ITAAC is, ITAAC are those inspections, tests, analysis, and acceptance criteria identified in the combined license that are met by the licensee, are necessary and sufficient to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license, the provisions of the Atomic Energy Act as amended, and the Commission's rules and regulations.

MEMBER BLEY: So that sounds as if ITAAC should be a subset of the results of the pre-op testing. Is that always true, or are there some ITAAC that aren't covered in pre-op testing?

MR. MAXWELL: The latter. There's a lot of ITAAC that are not covered as a part of pre-operational testing that occur long before pre-operational testing before the components are installed.

MEMBER BLEY: Yeah, that's right. Okay.

MR. MAXWELL: So other detailed design information that's contained in Tier 2, only the most safety significant aspects that each of the systems are included in the Tier 1 design descriptions. These aspects -- these safety significant aspects are referred to as top level design features and top level performance characteristics. To identify what information qualifies for inclusion in Tier 1, we applied a first principle screening process.

Section 14.3 provides guidance that the Tier 1 design descriptions are limited to the top level design features of the following categories: safety related SSC, non-safety related SSC that provide protection to safety related components, security system physical SSC, and risk significant non-safety related SSC determined by results of PRA.

In those categories, top level design features that fall into these categories, not surprisingly include categories like safety related pressure boundaries and component performance, fuel storage and seismically qualified structures, and physical security.

So this is the -- to answer a little bit of the question that's being asked is we've taken the Tier -- all the detailed Tier 2 design information. We apply our first principles to select the information to be included in the design description Tier 1 information. If the information in Tier 2 changes, we reevaluate it to see if Tier 1 needs to be updated as we go through the design certification.

CO-CHAIR SKILLMAN: Has that occurred?

MR. MAXWELL: Yes sir.

CO-CHAIR SKILLMAN: What's an example, please.

MR. MAXWELL: Well, one of them was the module lift adapter originally was not part of the standardized ITAAC. But through PRA of Chapter 17 said it's a risk significant component, and ITAAC was created. It was described in Tier 1, and ITAAC were created for it.

CO-CHAIR SKILLMAN: Thank you.

MR. MAXWELL: So now with the information selected to be included in Tier 1, we divide that information in two categories. The detailed design information selected for Tier 1 that describes system function, safety classification, and general location make up the system description portion of the design description while the design features such as seismic and ASME classifications and environmental qualification requirements make up the design commitments portion of the system description. And this is important because it's the information in the design description that is verified by ITAAC. This is the basis for our ITAAC.

So for each system with a design commitment, a table -- an associated table of ITAAC entries are provided. ITAAC themselves are broken into three columns. The first column is the design commitment, and the design commitment contains the text that's to be verified and is extracted directly from the design commitment in the design description in Tier 1.

The second column is the inspections test and analysis which includes the method in which the design commitment will be verified. And before I move to the acceptance criteria, so you mentioned before that there was seven categories throughout vendor testing. This is where I think the important distinction here is that we have a design commitment to meet. That's what we have to demonstrate that we meet this design commitment.

Now we can go to that pool of possible. Is it a test that will demonstrate an analysis? What is the most appropriate method to demonstrate we meet that design commitment? So it's we don't go to the -- we don't start with a list of tests and then build it. So you could have one of those areas where you actually didn't have an ITAAC to utilize that method. I would say that a test is always my first choice.

CO-CHAIR CORRADINI: You're starting with a function.

MR. MAXWELL: You're right.

CO-CHAIR SKILLMAN: Either the credited function or whatever the design feature that must be fulfilled.

MR. MAXWELL: A function --

CO-CHAIR SKILLMAN: Learning rom that to how do you confirm that.

MR. MAXWELL: That's correct. And I'll add, or feature --

CO-CHAIR SKILLMAN: Or a feature, yeah.

MR. MAXWELL: -- or a requirement. So in the third column is the acceptance criteria. It identifies conditions that you must demonstrate or verify again to demonstrate that you meet -- that the licensee meets the design commitment.

So any areas that you wanted me to discuss that I haven't met yet?

MEMBER BLEY: I have something I'd like to ask you about. It doesn't really fit here. But if you've been around the last few days, you'd know we've asked a lot of questions about the ECCS valves because that's the one thing. If they somehow don't work the way the reliability you think, it could really change the risk results and put whoever is operating the plant in a more troublesome spot.

It strikes me, and I wonder if you guys have thought about this some. It's not a startup issue. It's a later one. And perhaps a more thorough tracking of all the data on all of the ECCS valves over the years as they're stroked and tested to see if there are patterns among the valves and trends over time.

And I know you look for that generally. But here we're looking for things that might indicate there's some kind of degradation going on in all of those valves. A real emphasis on that kind of program for after startup, have you talked about that all? Because you can't do all this and manufacturer's test. You can do some accelerated aging, but they don't mimic exactly the conditions in the plant.

MR. MAXWELL: We have Gary McGee on the line in Corvallis.

MEMBER BLEY: Who is free to speak up. We can't quite hear you.

MR. McGEE: Okay. I just need some clarification. What exactly are we --

MR. MAXWELL: In service testing.

MR. McGEE: What's the question?

MR. MAXWELL: In service -- if you could describe in service testing for ECCS valves and how we plan to track the conditions of the valve to identify potential degradation of the valve over time once the valve is in service.

MEMBER BLEY: And not just individual valves. But if there's a plan to see if there's a common aging effect among all of them.

MR. McGEE: Well, in service testing is -- the whole thing about the OM code is in service testing is a trending program where you just don't look at stroke time. We also -- we're committed to the -- currently committed to OM-2012.

But we also take portions of OM-2017 where we take Mandatory Appendix 4 which is AOV testing. But we apply it to both the CIVs and ECCS for performance -- what we call performance of assessment testing. And we take various characteristics of the valve.

And so it's not only stroke time but we take the block valve and other characteristics of those valves and do comprehensive testing to -- the great thing about having 12 modules is you do test the same valves. It's like having 12 different plants and you look at those. You trend all those valves and see what each valve is doing.

So if you have an anomaly in one valve,

you take a look at all your other valves and see what you got going. And that's what you've done in regular traditional plants as well with your valves. You had your -- you grouped your valves and you look at your different air op valves. And when you have something come up, a bad stroke time in an air op valve, you look at your other ones to see what they're doing.

We do the same thing with our ECCS valves. If a valve doesn't stroke correctly or it starts leaking, you look at the other valves and see what you've got going.

MEMBER BLEY: Okay. Well, that's kind of the standard approach, and I think that's essential. What I was suggesting is because of the really significant impact of these valves in this particular design, something that perhaps digs a little deeper at trends among the various sets might be useful. Just something to think about. You don't have to answer that at this point.

MR. MAXWELL: Thank you, Gary.

MR. McGEE: Sure.

CO-CHAIR SKILLMAN: Chris, you asked whether there were any other issues.

MR. MAXWELL: Yes sir.

CO-CHAIR SKILLMAN: Please speak to two

things, the robustness of NuScale's configuration management program to keep all of this information organized and the NuScale QA program to ensure that the information for your safety equipment is logged and maintained and in a manner that is retrievable for a future user. Yes, please.

MR. RAD: Okay. This is Zach Rad. Well, you know that. So just to talk about a couple of things. And so as we mentioned, as Chris did a great job of explaining, Tier 1 is built off of Tier 2.

It's important to recognize that both of those documents are built off of a much larger volume of source documentation. And in fact, each statement of fact is required by our process to have a source document that supports it. Those source documents are the controlled documents. And when I say controlled, I mean controlled under our document control which is one provision of the NQA-1 program.

So to address the quality assurance requirements, I won't go into the details because I simply can't recite them. But I'll tell you that it meets the requirements of NQA-1 document control and retention. So that is the required standard.

CO-CHAIR SKILLMAN: That's sufficient. Thank you.

MR. RAD: When it comes to maintaining and controlling what's frequently called configuration, there are a couple ways this happens. And so the design control process requires that when a design change is made, that impacts to other design documents. And other programmatic documents is reviewed. So those are the source documents, the underlying source documents. Any changes that are necessary, conforming changes need to be addressed as part of that design change.

Secondarily, that same process, the design change process, drives us to review the licensing basis documents for impacts. There's a configuration management database that links every XY section, every topical report, technical report to its underlying source documents. So there's an immediate reference available.

Those impacts are reviewed. If there's changes to Tier 1, those are made. Those are provided currently now during the review to the NRC as updates. If it's post-licensing and it's in the COL, that'll be a departure, an exemption, or a license amendment. It'll be a departure and exemption during the COL licensing process. It'll be a departure, an exemption, or an amendment post-COL license.

Let's see. What else is there? Oh, yeah. And then, of course lastly, anything site specific is going to get its own version of inspection tests, analysis, startup testing, et cetera. So if they demand changes in the design that are site specific or there are site specific characteristics that are yet to be fleshed out like EP and security, they'll get their own testing, et cetera, that'll have to be defined in the COL application.

CO-CHAIR SKILLMAN: Thank you. Chris, go ahead.

MR. MAXWELL: Okay.

MR. RAD: One other item. There's one question you raised. You asked if our classifications for SSCs were consistent with 50.69. They're similar to but different. So 50.69 calls out Risk Informed Safety Class 1 through 4, and it calls them safety related that perform safety significant functions and non-safety and safety significant functions and then similar for the 3 and 4, safety related non-safety significant and then non-safety with low safety significant functions. So different wording but similar concept.

CO-CHAIR SKILLMAN: Yeah, the color coding on slide 2 or 3 appeared to be almost the same at

50.69 is why I asked the question. Thank you.

MR. MAXWELL: So describe how we take the design commitments from the system from the design descriptions in Tier 1 and then those become our ITAAC. We received a letter dated April 8th, 2016 from the NRC that provide a set of standardized DCA ITAAC for the site application.

So what we did was take all of our design commitments. And where they aligned with the standardized ITAAC, we used the standardized ITAAC. Where there wasn't a representative standardized ITAAC, we created a design specific ITAAC. And that effort resulted in the complete set of ITAAC provided in Tier 1.

CO-CHAIR SKILLMAN: I do have that letter, and the letter said, this is a suggestion. Clearly, you used the portion that was a suggestion and added to it or modified it to create the present very thorough listing that you've created. So thank you. I understand.

MEMBER REMPE: So I'm a little slow, but you're about -- you're at the end of your presentation. But I was interested in the description of the documentation and traceability that Zach provided. And it sounds good, but people are people.

And so what assurances you've got your staff doing it correctly? I mean, do you internally audit as well as the staff been by to check this? And it'd be spot checking but found a few errors and done some corrections and process corrections and decided that it's working well?

MR. RAD: Yes. So the programs under the quality assurance program have mandatory audits and surveillance is under the NQA-1 program. And then that same group as well as our internal self-assessment program have oversight over the configuration control between design and the licensing basis.

MEMBER REMPE: Thank you.

DR. SCHULTZ: Chris, one general question. You mentioned -- well, over the last few days, we've been talking about the PRA and its impact on the overall evaluation that has been done. And you mentioned one example of identifying some risk significant items through the PRA recently that caused changes to your testing program.

Other examples -- and I have one question. Is there other examples that you may have that have been affected by the work that has been done in the risk and reliability areas? And secondly, have there also been -- what type of oversight reviews have been done to assure that there's connections between what's been done and the risk evaluation area and the testing program?

MR. MAXWELL: The design frankly doesn't include a lot of risk significant components or functions that are also safety related. So the first effort captured the bulk of those. I mentioned the module lift adapter being added later. So I don't have another example from a risk informed position that created additional testing. And then the second part of your questions was?

DR. SCHULTZ: Any oversight evaluations that have been done to assure that the overall program -- just to review and assure the overall program has captured what needs to be and to help prioritize the testing functions, the functionality of the test to assure that those things that are most important in the testing program -- the detailed testing program are captured appropriately.

MR. MAXWELL: The best method is that we have all of the functions that are risk significant listed in Section 17.4. And we've done a one-for-one. Again, a benefit of having done our pre-operational test abstracts the way we did where we identified the actual function to be tested that we're able to do a one-to-one verification of each of those risk significant functions to verify that they are, in fact, inventoried in an abstract and verified.

The staff review of our test program is an example of the oversight to ensure that we do have that alignment between Section 17.4 and Section 14.2, Pre-Operational Testing.

DR. SCHULTZ: Good. Thank you.

MR. MAXWELL: Yes sir.

DR. SCHULTZ: It also was mentioned that you got the benefit of -- with the 12 modules, you got the benefit of gathering substantial data over what one would hope a very quick or very short time frame to validate that the information that you get from the testing program and use that information. And you spoke -- Zach, you spoke to the capability to collect that information and utilize that moving forward.

Is that plan pretty much in place, envisioned? Where does that stand at this point?

MR. MAXWELL: Well, we have the initial -- the in service test program requirements that detail the information that'll be recorded. And this will be gone into a little more detail in the Chapter 3 discussion which I'm not sure when that occurs but later. So I don't have a lot of detail about the in service testing piece.

But what I will say is that from a pre-operational testing standpoint that absolutely the benefit piece of the 12 where I do the pre-operational test on the first module. I gain instant experience on how to conduct those tests.

The issue was or the question was asked earlier how we address if there's a problem with the testing. Well, part of that now I fixed that test for the next 11 modules. And we assume that we'll -- well, we know that we'll educate ourselves as we go on to both be more efficient and accurate in our testing.

DR. SCHULTZ: The staff is going to address open items that they've identified in the safety evaluation. Any comments that you're prepared to address the status of open item --

(Simultaneous speaking.)

MR. MAXWELL: I have no open items at this time.

DR. SCHULTZ: Okay. Thank you.

MEMBER DIMITRIJEVIC: Well, I would like to add something in this connection with PRA and these functions. And I was watching for this crane because there is a different function a component has to perform what it's designed for and then there is some functions which we are concerning the risk analysis.

For the crane function is to move the module from A to B. PRA is concerned that during that movement accidents don't happen. There is normal speed, normal height. And so when I check your test, you do check for those limitations in control.

For some functions like these ECCS valves, there is a couple of different functions credited in the PRA. One is they open for the signal and one is, for example, opens without signal in the low delta P conditions or something like that.

So when we have multiple considered in the risk analysis, are those multiple functions separated? Like, for example, a good example will be on the ECCS valves opening with the signal and opening without?

MR. MAXWELL: Yes, that functionality -- that specific functionality is separated. The initial test program will test the function of the valve to open with a signal to open once below the differential pressure threshold of the inadvertent actuation blocking feature but not the condition where there's a malfunction. That's not a piece of the initial test program.

MEMBER DIMITRIJEVIC: Okay.

MR. MAXWELL: That concludes the presentation for Chapter 14 if there's any questions.

CO-CHAIR SKILLMAN: NuScale team, thank you very much. Colleagues around the table, do you have any additional questions for the NuScale team before we move on here? Hearing none, NuScale, thank you.

We are in recess for 15 minutes. We're going to take the break early so that we give the NRC staff unbroken time for their portion. Please return at ten minutes after on that clock.

(Whereupon, the above-entitled matter went off the record at 9:46 a.m. and resumed at 10:07 a.m.)

CO-CHAIR SKILLMAN: We reconvene this meeting, and we welcome Tanny and the NRC staff to the presentation on ITAAC for the NuScale design.

Tanny?

MR. SANTOS: So, thank you, Dick.

First off, I just want to extend my thanks to the Committee members for their accommodating the staff's request to receive the Chapter 14 SER in pieces. It was very helpful to the staff. So, we appreciate your flexibility on that.

Again, my name is Tanny Santos. I am the Chapter 14 Project Manager. Sam Lee is the Branch

Chief. He's back there.

Listed on this slide are all the technical reviewers who provided input to Chapter 14. As you can see, it is quite a list because of the scope of the information reviewed in this chapter. So, it is a lot. It's not just NRO staff; it is also NRR and NSIR. So, it is really an agency effort for this review.

The outline of the staff's presentations is in two parts, similar to what NuScale had presented. SER Section 14.2 is on the initial test program, and that will be led by Taylor Lamb. The second half of the staff's presentation is on 14.3, focused on ITAAC. That will be discussed by myself, Nick Hansing, and BP Jain, to my right.

And so, with that, I'll just turn it over to Taylor to begin the discussion on 14.2.

MS. LAMB: Hi. My name is Taylor Lamb. I'm in the Quality Assurance Vendor Inspection Branch. I've been working at the NRC for nine years, and I've been the lead technical reviewer for the initial test program. But I would like to note that there was a significantly larger group associated with this review, aside from just me. So, I might point to other people in the room, if you ask any technicallyspecific questions.

So, with the review, consistent with past submittals, it contained multiple sections, organization and staffing, test procedures, individual test descriptions. This is pretty consistent with what you've seen in the past.

Next slide, please.

So, the review objectives for this were to review Tier 2, Section 14.2, for completeness and suitability of the development of an ITP by a COL applicant against the guidance in the DSRS Section 14.2 and Reg Guide 1.68 by using a risk-informed approach. So, I'd like to focus on the note about the COL applicant developing an ITP.

The regulations in 52.47 for Design Certification Application do not stipulate requirements for pre-operational and startup testing. However, 52.79(a)(28) for a COL application does specify requirements for pre-operational and startup testing. The guidance in Reg Guide 1.68 goes over some more specific requirements, as was discussed earlier.

So, in order to accomplish the review, especially the risk-informed approach of the review, we utilized SECY-11-0024, the use of risk insights to enhance the safety focus of small modular reactor reviews in development of our guidance. So, the revised ITP review, it focuses on providing reasonable assurance that the risk-significant structure system and component functions are tested, and a test abstract adequately addressed the design functionality. So, we're just looking at the design functionality of the test abstracts that we did review. So, from that, the DSRS, Section 14.2 was developed, and it provided the general guidance for the NRC staff to review the proposed ITP.

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With that being said, going back to what I stated earlier, the DSRS even notes in the introduction that there is no requirement for a design certification applicant to provide an ITP submitted under 10 CFR Part 52, Subpart B. But the staff did review this, specific test abstracts, under DSRS Section 14.2 and Reg Guide 1.68.

So, in order to determine which test abstracts the staff would review in this modified approach, we utilized Table 17.4-1, the D-RAP SSC functions, categorizations, and categorization basis, which stipulated the risk-significant system functions that, I suppose, are -- I'll leave it at that.

So, with that, we proposed this to NuScale. NuScale staff, they came back, specifically requested a larger scope of review that included additional test abstracts that might not have been identified via the D-RAP. They discussed a larger scope, and they mentioned the reactor-building crane system, for instance.

The NRC approved only those test abstracts listed in 14.2-1 of the SER. so, we separated out the ones that we did review and would approve, and the other test abstracts that are not going to be approved are in Table 14.2-2 of the SER. So, those test abstracts should be addressed by, must be addressed by the COL applicant, since 52.79(a)(28) requires that they provide plans for pre-operational and startup testing.

MEMBER BLEY: I got a little confused in your description. When the Applicant requested that you review additional tests, did you review those?

MS. LAMB: Yes.

MEMBER BLEY: Okay. And they're among the ones you've approved?

MS. LAMB: Yes.

MEMBER BLEY: Okay.

MS. LAMB: And those are in Table 14.2-1

of the SER.

CO-CHAIR SKILLMAN: Taylor, does the absence of direction to create an ITP in Part 52, Sub B, constitute a deficiency in the regulation? It sounds like you've gone through an awful lot of effort to create a path forward where one is not prescribed. I'm not an advocate of more regulation, but I'm just wondering if there's a hole in Part 52.

MS. LAMB: I do not think that I would be the appropriate individual to answer that question. But I will say that, as the DSRS states, the initial test program is typically reviewed in the design certification stage in order to better prepare for a COL application. I'm not sure if that clarifies the question, but if Kerri Kavanagh --

MS. KAVANAGH: Hi.

CO-CHAIR SKILLMAN: Hi, Kerri.

MS. KAVANAGH: How are you?

CO-CHAIR SKILLMAN: Well, thank you.

MS. KAVANAGH: This is Kerri Kavanagh. I'm the Chief of the Quality Assurance Vendor Inspection Branch, and now in NRR.

You've got to remember that this is for design certification, not the testing of a plant. But most designers in the past, the AP1000s, the ESBWRs, they wanted as much to be approved in the design certification phase, so that their COL applicants would not have to create that kind of information, because they're the design authority. NuScale followed suit.

We do not believe that the regulations are deficient. It's just it's not a requirement in a design cert. It's just a practice that has been done by most vendors for a design certification.

CO-CHAIR SKILLMAN: Thank you. Okay.

MEMBER BLEY: And the review would have been done in the COL stage under --

MS. KAVANAGH: It will still be done in the COL stage.

CO-CHAIR SKILLMAN: Go ahead, Taylor. Continue, please.

MS. LAMB: Okay.

CO-CHAIR SKILLMAN: Thank you.

MS. LAMB: So, if the design certification is approved, the staff would recommend that the certification rule include clarifying language that those other test abstracts in 14.2-2 of the SER, that they're outside the scope of the certified design.

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So, with our conclusion, we do have one

open item linked to Test Abstract 14.2-47, "Emergency Core Cooling System Test No. 47". It's Open Item 03.09.06-1. So, that is in Chapter 3. Until that item is resolved, we will leave open Test Abstract 14.2-47.

CO-CHAIR CORRADINI: Can you tell us a bit more?

MS. LAMB: Since that is a Chapter 3 item, Tom Scarbrough can speak to that.

MR. SCARBROUGH: Good morning. I'm Tom Scarbrough.

Yes, in Section 3.9.6, we have an open item for the ECCS valves. Under 50.43(e), NuScale is required to have a design demonstration of this new ECCS valve system. They plan to do demonstration testing in June for that, and we'll be there to monitor that. And we will write a report about that. And once that's all complete, then we'll be able to determine if we can close that open item. Until then, we have kept it open, and they're tracking it in Chapter 14 as well.

CO-CHAIR CORRADINI: Thank you.

MR. SCARBROUGH: Okay. Thank you.

MS. LAMB: Okay. In performing the review, the staff ended up reviewing a number of proposed markups to DCA Part 2, Tier 2. Therefore, we have confirmatory item 14.2-1 that will be tracking the incorporation of those proposed changes in future revision of the DCA.

With that, the staff concludes, using the information presented in the DCA, and pending the confirmatory and open items, that the Applicant has demonstrated compliance with the NRC regulations and guidance.

MR. SANTOS: So, with that, I'd like to move on to SER Section 14.3, ITAAC. This CR section reviews Tier 1 information, all the Tier 1 information, including definitions, site parameters, interface requirements, as well as the ITAAC tables.

So, the regulatory finding in 14.3 that the staff is making is with regard to 52.47(b)(1) that I think was mentioned earlier in the presentation. This is the requirement that the ITAAC be necessary and sufficient and provide reasonable assurance that, if the inspection test analyses are performed and the acceptance criteria are met, that the plant will be designed and operated in accordance with the design in the NRC's regulations.

So, to assist the staff in making that regulatory finding, there are several guidance documents the staff had used. One, of course, is the Standard Review Plan, Section 14.3. The second is the set of standardized ITAAC that the NRC provided NuScale. I think it was referred to in NuScale's presentation back in 2016, provided them a set of draft standardized ITAAC that could be submitted as part of a Design Certification Application. And as I said earlier, NuScale did employ many of those ITAAC in their application.

And the third document I'd like to discuss is relatively new. So, I want to spend some time on it. It is SECY-19-0034, which was just issued in April, April 8th I believe. This title is approving design certification content and describes some new general principles for how Tier 1 information should be reviewed as part of a design certification.

Now many of the principles described in the SECY are similar to what is in SRP 14.3 now, but there are three new principles that this SECY describes, and I'd like to point these out to you on these slides, these three bullets here.

One is that Tier 1 information should be typically at a qualitative or functional level of detail. Tier 1 should also not include any detailed information that would require NRC approval for a departure from the certified design that would have minimal safety significance. And lastly, the use of numeric information in Tier 1 to try to be minimized.

So, the general theme for these three new principles are try to emphasize the importance of avoiding any unnecessary detail in Tier 1, with "unnecessary" implying that an applicant would have to come to the NRC to request a departure or an exemption from something that has a minimal safety significance.

CO-CHAIR CORRADINI: Is this a nice way of saying a lesson learned?

MR. SANTOS: Yes, sir, this is, I think, a lot from Vogtle --

CO-CHAIR CORRADINI: Yes. Fine.

MR. SANTOS: -- information from Vogtle, yes.

CO-CHAIR CORRADINI: Thank you.

MR. SANTOS: You're welcome.

As I said earlier, this SECY was just issued in April. And given the timeframe of that, the staff has not really had an opportunity to apply all of these new principles to all of the Tier 1 information in the NuScale application, but we plan to do that as part of our phase 4 review.

CO-CHAIR CORRADINI: So, let me ask you a different question.

MR. SANTOS: Sure.

CO-CHAIR CORRADINI: You have a number of open items.

MR. SANTOS: Yes.

CO-CHAIR CORRADINI: Would some of these go away, given this?

MR. SANTOS: I'm not sure because the number of open items has to do with exemption requests --

CO-CHAIR CORRADINI: Okay.

MR. SANTOS: -- but they may not impact the --

CO-CHAIR CORRADINI: Okay.

MR. SANTOS: But it, theoretically, could reduce the amount of information in Tier 1. If it was agreed to that there is some detail in there that could be removed based on this, there could be a reduction in Tier 1 information and/or ITAAC.

CO-CHAIR CORRADINI: Okay. Thank you.

MR. SANTOS: Yes. So, like I said, we haven't applied this to all of NuScale's application, but there is an attachment to this SECY that describes, as an example, how these new principles could be applied or how these new principles would be applied to the NuScale review for the structural integrity review only. So, that's an attachment to the SECY paper to come. So, have it done in that area.

CO-CHAIR CORRADINI: We received this a week ago, yes. No, two weeks ago. I didn't see the attachment. I have the 10-page, it was the 10-page SECY. Was there an attachment to that? I only had the SECY.

MR. SANTOS: Yes, I think there should be an enclosure or an attachment to the --

CO-CHAIR CORRADINI: You got it? Okay. then, that's my fault then. Never mind.

MR. SANTOS: Okay.

CO-CHAIR SKILLMAN: Keep going, Tanny.

MR. SANTOS: Okay. So, the next slide.

The rest of the staff's presentation is really going to focus on the rest of the SER section in 14.3, but only those that have open items. So, listed here are the six sections of the staff's SER that do not have any open items. So, we don't plan to have any additional discussion or go into any detail about these sections. I will go on with the other sections.

CO-CHAIR SKILLMAN: So, Tanny --

MR. SANTOS: Yes, sir?

CO-CHAIR SKILLMAN: What does that mean? Does that mean that your review of the proposed ITAAC by NuScale is without comment --

MR. SANTOS: No.

CO-CHAIR SKILLMAN: -- or without suggestion for change?

MR. SANTOS: No. There may be, for example, confirmatory items that need to be closed in some of these sections. And the other sections may not have open items, but no conclusion can be met because it may rely on information on another section that has an open item. So, it's just the way that the staff's SER is organized, the ITAAC that falls under these systems has no specific open items in them. There may be confirmatory items associated with them, though.

CO-CHAIR SKILLMAN: Fair enough. Okay. Thank you.

MR. SANTOS: Okay? So, the first section I'd like to discuss is 14.3.1. This section discusses the selection criteria for Tier 1. I think NuScale described the first principles approach, but the staff has specifically excluded from its review the first principles approach that NuScale is using for identifying Tier 1 in ITAAC. Their approach is very similar to an approach defined by NEI in 1502 in an NEI white paper. The staff has not endorsed these approaches. We provided comments and discussions and meetings with NEI, but we have not formally endorsed them. So, the staff is not taking a position on this first principles approach in their application.

It's described in Section 14.3.2 of their Tier 2 application. So, that would mean that, if and when this were to go to design certification rulemaking, it would not, this section would not be incorporated by reference into the rule because the staff is not taking a position on this particular methodology. Okay.

MEMBER BLEY: Ignoring the methodology, you do take positions on the results of the applying the methodology?

MR. SANTOS: Yes, sir, we do make a finding on the ITAAC themselves, but not the method for identifying what's in the ITAAC or Tier 1.

CO-CHAIR CORRADINI: I mean, this leads to a bunch of ancillary questions. So, in the past design certifications did the past applicants use this methodology or their own individual methodology?

MR. SANTOS: I don't believe this approach has been used in other applications.

CO-CHAIR CORRADINI: Okay. Fine.

MR. SANTOS: But I could be wrong. But I don't believe this is --

MR. WELCH: This is Chris Welch.

It hasn't, Tanny.

MR. SANTOS: Okay. Thank you, Chris.

CO-CHAIR SKILLMAN: So, what does this, if you will, exclusion from incorporation by IBR, do to an applicant? What is the consequence of this position?

MR. SANTOS: I don't believe it -- I think the ITAAC would still have to be closed because the staff is making a finding on the ITAAC and the information in Tier 1, not the methodology used to identify what Tier 2 information rises to the level of being Tier 1. There would be an impact. The applicants would still have to follow the Tier 1/Tier 2 information that's been approved. If they want to make a change to Tier 1 or Tier 2, follow existing processes.

CO-CHAIR CORRADINI: So, what you're saying is how they came to a conclusion you're not going to say positive or negative?

MR. SANTOS: Right.

CO-CHAIR CORRADINI: You're just going to look at the conclusion and see if it's reasonable?

MR. SANTOS: Yes.

CO-CHAIR CORRADINI: Fine.

MR. SANTOS: I think there are plans by NEI to later submit an update to this guidance document for NRC to maybe eventually approve later. I think that is in the works, but I don't have the schedule for that. But there are plans for that, I believe.

CO-CHAIR SKILLMAN: Let me see if I understand it. You're saying NuScale's approach as a process is not endorsed by the NRC, and therefore, it may not be IBRed. But the ITAAC that it identifies must be executed?

MR. SANTOS: Yes, the ITAAC itself would be certified.

CO-CHAIR SKILLMAN: Got it. All right. Now I understand the logic. Thank you.

MR. SANTOS: Okay. The last bullet just states that the remaining Sections 14.3.2 through 14.3.13 document the staff's review of the ITAAC or point to another SER section where they're evaluated and a conclusion is found.

So, there are two open items in 14.3.1.

The first is an open item initially identified from Chapter 17. It has to do with the Design Reliability Assurance Program. And in SECY-18-0093, the staff actually sent a paper to the Commission recommending that the ITAAC to verify the effectiveness of the D-RAP be discontinued. And so, NuScale did not provide an ITAAC for the D-RAP. But the staff is still waiting for the Commission to make a decision on this paper. So, once that is provided to us, if the decision agrees with the staff, then this open item can be closed. But if the Commission decides that ITAAC for D-RAP should still be implemented, staff would be requesting that NuScale provide an ITAAC for this, similar to what was done for other applications.

CO-CHAIR CORRADINI: Well, just for everybody, we discuss this when we talk about Chapter 17.

MR. SANTOS: Thank you.

The second open item has to do with a recently-issued RAI that the staff provided just last week to NuScale in a draft form. Similar to what the staff did for the APR1400, the staff reviewed the Tier 1 information, focusing on the ITAAC themselves, and review it for clarity and format to make sure that there's no ambiguous acceptance criteria, to make sure that the design commitment, inspection test analysis, and acceptance criteria are similar in scope, to minimize any possible misunderstand or misinterpretation of the language.

Based on that review, the staff issued an RAI requesting NuScale make some language changes to the Tier 1 information and ITAAC. And so, again, a similar request and review was done for the APR1400. Staff still might well engage with NuScale on that because it was just issued last week. So, it's still an open item for the staff's review.

CO-CHAIR SKILLMAN: So, is this an administrative item or is this a technical item?

MR. SANTOS: It could be technical/legal, to later down the line avoid any misunderstandings or discrepancies on what does this new language mean. If there is language in an ITAAC that could be ambiguous or interpreted multiple ways, this is to try to prevent that from happening by making sure the language is clear, unambiguous, and avoid any problems down the line.

CO-CHAIR CORRADINI: And the lawyers are going to solve it?

MR. SANTOS: It is a lesson learned from --

CO-CHAIR CORRADINI: And the lawyers are going to solve this?

MR. SANTOS: They are providing us advice.

CO-CHAIR CORRADINI: Okay.

MR. SANTOS: I will say that.

Okay. So, the next aspect of 14.3.1 is the discussion of interface requirements. The Tier 1 information has an interface requirement regarding the failure of a structure that's not within the scope of the certified design, not causing any of the seismic Category I structures that are within the scope of the design to fail.

Now this specific interface requirement will be evaluated by the staff in Chapter 3. But there is another requirement in 52.47(a)(26) that states that this interface requirement must be verifiable through ITAAC, right? So, NuScale did provide two ITAAC to verify that, as built, Non-Seismic Category I SSCs will not impair the ability of Seismic Category I SSCs. But the staff can't make a finding that this particular requirement has been met yet because of an open item with one of these two ITAAC that BP will discuss in his part of the presentation.

MEMBER BLEY: Without having an as-built

plant or a 3-dimensional model, how can you satisfy this ITAAC? I mean, usually, we have done this in the past by a walk-down of the plant and saying that's Seismic II. It's over I. It can fall and hit.

MR. SANTOS: Right.

MEMBER BLEY: Then, we would have to fix that. Do you have 3D models that you can look at?

MR. SANTOS: Not 3D models, but there is a standardized ITAAC that staff provided --

MEMBER BLEY: So, you can approve the ITAAC?

MR. SANTOS: There's a standardized ITAAC that addresses, I believe, this.

MEMBER BLEY: Okay. But the ITAAC cannot be fulfilled until the plant is in place?

MR. SANTOS: Yes, until it is built, yes.

MR. WELCH: That's correct. This is Chris Welch. It's an as-built ITAAC.

MR. SANTOS: Thank you, Chris.

And with that, I'll turn it over to BP to talk about this and Section 14.3.2.

MR. JAIN: All right. So, we reviewed the structural integrity of the reactor, radioactive waste, and control building, Section 14.3.2 in the SER. And the purpose, the scope of a review, was to ensure, to make sure that the ITAAC will ensure the final as-built plant structure is built in accordance with the certified design. That was the high-level focus.

In reviewing it, we find that the acceptance criteria for the ITAAC for these three buildings is sort of incomplete in the sense that, in order to meet the goal, it lacks certain details. For example, it does not talk about reconciliation of the deviation between the as-built loads and the assumed constructed load. So, there is a deviation in what you are assume in the design and what you actually put in. So, how to reconcile is not talked about.

CO-CHAIR CORRADINI: Is a deviation another way of saying tolerances? I don't understand what a deviation is.

MR. JAIN: It doesn't actually have to be a tolerance. It has to be bigger than that. Let's say you're designing the equipment and you assume 100 pounds weight in your design. And then, you actually put the equipment back in there and find it's 500 pounds.

CO-CHAIR CORRADINI: So, it has to be outside of the specified tolerances --

MR. JAIN: Yes, yes.

CO-CHAIR CORRADINI: -- is considered a deviation?

MR. JAIN: Sure.

CO-CHAIR CORRADINI: Okay. Fine. Okay. Got it.

MR. JAIN: So, if you take the cumulative effect of the change in the loads and our configuration of the plant, that, in fact, is not being addressed in the acceptance criteria. Because you can look at it individually, like we talked about, 100 pounds versus 500 pounds. But if you do thousands of those components, it would change those loads. The cumulative effect could be, does need to be addressed. And that has not been in the acceptance criteria.

The same thing when you do the demand analysis with this new configuration and the as-built load. You need to address it to make sure that the seismic response of the building is not adversely affected and is bounded still by the certified design because that feeds into the system analysis, the seismic response. So, that aspect is not being addressed in the acceptance criteria.

So, those are the basic steps/issues, and we have discussed this with NuScale. And my understanding is that they'll incorporate or they were in agreement with the staff's findings.

CO-CHAIR CORRADINI: So, it's being fixed?

MR. JAIN: That is what I heard last time when we spoke about it two weeks ago or so.

CO-CHAIR SKILLMAN: Thank you. Keep on going.

MR. JAIN: The second open item has to do with the seismic interactions, the Non-Seismic Category Is with the Seismic Category I structures. And this one, the views on this open item, it's really twofold. One, the reactor building ITAAC, that matches with the standardized ITAAC, what the staff proposed in their April 6th letter. But, for the control room building, it does not. For whatever reason, NuScale knows better. So, we discussed that anomaly, if you will, and they said they will fix it; they will make it consistent with the reactor building and the standardized ITAAC. But staff still, when we saw it, you know, that's what our review finding was.

CO-CHAIR SKILLMAN: So, are these going to be construction-stage ITAAC for closeout?

MR. JAIN: Well, these are not -- this is you do after you have as-built. You have to do it as you go along.

CO-CHAIR SKILLMAN: As the construction --

okay.

MR. JAIN: And then, at the end of the construction, you have to reconcile the deviations --

CO-CHAIR SKILLMAN: Okay.

MR. JAIN: -- in a global sense.

CO-CHAIR SKILLMAN: I think the answer to my question is, yes, these are construction-stage ITAAC that get closed out when there's confirmation that the open item has been satisfactorily addressed?

MR. JAIN: Right, on a global basis, though.

CO-CHAIR SKILLMAN: Yes. Okay. And are you comfortable that NuScale understands this?

MR. JAIN: Yes, from the discussion we had on the May 8th telecon, that's our understanding.

CO-CHAIR SKILLMAN: Okay. Thank you.

MR. JAIN: Okay.

MR. SANTOS: Next is 14.3.3, and that will be presented by Nick Hansing.

MR. HANSING: Good morning. My name is Nick Hansing. I was the lead reviewer for Section 14.3.3, "ITAAC for Piping Systems and Components".

I wanted to highlight one open item on the following two slides. It's entitled, "NPM Valve Installation Verification ITAAC". The ITAAC, as mentioned earlier, you need to satisfy 10 CFR 52.47(b)(1), to provide reasonable assurance that, in this case the NuScale power module, NPM, safety-related valves are constructed and will operate in conformity with the design certification.

The NPM valve installation verification ITAAC will require a walk-down inspection of the emergency core cooling system, ECCS, valves; containment isolation valves, and decay heat removal system actuation valves, to ensure that the valves will not be prevented from performing their safety functions.

Next slide, please.

CO-CHAIR SKILLMAN: Nick, before you move on --

MR. HANSING: Yes?

CO-CHAIR SKILLMAN: I've got the Safety Evaluation, and I see your Confirmatory Item 14.3.3-1, but also 14.3.3-2 and 3.3-3, having to do with the safety valves.

MR. HANSING: Yes.

CO-CHAIR SKILLMAN: So, you're addressing just Open Item 3-3-1. How about the other two?

MR. HANSING: The other items, we've received proposed markups that the staff have found acceptable. So, we're just simply tracking those for incorporation into the next version of the DCD.

CO-CHAIR SKILLMAN: Understand. Thank you. All right.

MEMBER SUNSERI: Is this one of these examples, like NuScale talked about, where they had their initial test program requirement, and then, an ITAAC requirement? I mean, this is an initial startup program, right, or re-operational startup program?

MR. HANSING: I believe --

MEMBER SUNSERI: So, one test satisfies two requirements?

CO-CHAIR CORRADINI: Yes, I don't understand. Is it that it's physically where it's supposed to be or is it that it's been tested? That's what I -- I was reading your words. Is it both?

MR. HANSING: So, this ITAAC specifically is for a walk-down inspection of the as-built components to the --

CO-CHAIR CORRADINI: But it's physically where it's supposed to be?

MR. HANSING: Physically where it's supposed to be, its router where it needs to be. There's accessibility, which I'll get to in the next slide --

CO-CHAIR CORRADINI: Okay.

MR. HANSING: -- about some of those specific aspects.

So, the walk-down inspection will verify installation of the valves previously mentioned and their hydraulic lines, consistent with the specifications for geometric configuration, orientation, accessibility, and line routing, such that each valve can perform its safety functions.

So, together with the current ITAAC, the NPM valve installation verification ITAAC will provide reasonable assurance that the ECCS valves, CIVs, and DHRS actuation valves will operate properly to allow core cooling and provide containment isolation under design basis conditions.

And the NRC staff held a public teleconference with NuScale on May 8th, 2019, to discuss the path forward to a resolution of this open item.

CO-CHAIR SKILLMAN: And what was the result of that conversation?

MR. HANSING: NuScale intends to submit ITAAC to resolve this open item.

CO-CHAIR SKILLMAN: Thank you.

MR. HANSING: You're welcome.

Any additional questions?

(No audible response.)

MR. SANTOS: Okay. So, 14.3.6 reviews the Tier 1 and ITAAC related to electrical systems. This includes equipment qualification for seismic and harsh environments, containment electrical penetrations and lighting.

This section has two open items. The first was originally identified as a Chapter 8 open item. It has to do with NuScale requesting exemptions from GDC 17 and 18. So, GDC 17 requires that, well, let's see, systems -- onsite and offsite electrical power systems be provided to permit functioning of SSCs that are important to safety. GDC 18 requires that electrical power systems important to safety be designed to permit appropriate inspection and testing.

So, as I said, NuScale requested exemptions from these two GDC. The staff is still reviewing these exemptions. And so, if the exemptions are approved, the ITAAC proposed by NuScale would be sufficient and this open item could be closed. But if these exemptions are not approved, any equipment used to verify the GDC would need to have an appropriate ITAAC to verify the functionality.

MEMBER BLEY: There's something of a rat's nest of chapters that are all related to this.

MR. SANTOS: Yes, I agree.

MEMBER BLEY: Who's actually watching it -- watching it? -- driving the show on this? Is it Chapter 8 people, electricals?

MR. SANTOS: Yes. Well, the individual exemptions that are reviewed in other chapters would that, and 14 would track the --

MEMBER BLEY: It needs a coordinator of this?

MR. SANTOS: Yes, right.

MEMBER BLEY: Who is the coordinator, is what I was asking.

MR. SANTOS: For? For the exemptions, all the specific exemptions?

MEMBER BLEY: Yes.

MR. SANTOS: I guess that would be my Branch, us.

MEMBER BLEY: Oh, okay. Not the electricals?

MR. SANTOS: Well, they're doing the review, but tracking the actual exemptions and ensuring they're closed out is a licensing -- we'll take the lead for that.

MEMBER BLEY: Okay.

MR. SANTOS: But the actual review for the exemption approvals or not would be the technical staff.

CO-CHAIR CORRADINI: So, this would be handled in the next phase of the review? That is, once the Open Item 8.3.1 is closed --

MR. SANTOS: Then, it would be closed, 14, yes.

CO-CHAIR CORRADINI: Okay. Then, all the other things would cascade down?

MR. SANTOS: Yes. There's similar, other similar examples like that.

CO-CHAIR CORRADINI: Okay. Fine. Yes.

MR. SANTOS: Yes, right.

CO-CHAIR CORRADINI: No, Dennis as well -- okay, thank you.

MR. SANTOS: Right. So, the second open item in this section has to do with just an editorial error the staff found in one of the Tier 2 tables. Table 14.3-1 has some additional information about how an ITAAC is performed or closed out. And there's an incorrect reference to a Chapter 8 section, a Tier 2, Chapter 8, section there that the staff identified, and NuScale agreed should be corrected. So, once that's done, that open item could be closed.

The next section is 14.3.8, and that has to do with radiation protection ITAAC and Tier 1. So, again, there's two open items here, but they're both related to the same issue. It's just two different RAIs that led to these two open item numbers. That has to do with borated polyethylene shielding.

And now, Tier 1, Table 3.11-1 described reactor building shield wall geometry. It's not the ITAAC table, right? But the ITAAC table has -- there's an ITAAC with an acceptance criteria that references this table by saying, no, the thickness of the reactor building shielding barriers is greater to or equal than that value found in 3.11-1.

So, what's happened is the bio-shield design has changed. Originally, there was borated polyethylene in the bio-shield. And so, there was an item in Table 3.11-1 for it. When the borated polyethylene shielding was removed from the design, it was removed from the table. But, then, subsequently, it was reincorporated back into the bio-shield design, but not put into the table. So, the staff would be looking for an entry into the 3.11-1 table for that ITAAC acceptance criteria.

CO-CHAIR SKILLMAN: Tanny, this is the thing that I was speaking about in my opening comments, whereas the design is maturing, it is essential to keep the ITAAC --

MR. SANTOS: Right.

CO-CHAIR SKILLMAN: -- current with inappropriate change.

MR. SANTOS: Right.

CO-CHAIR SKILLMAN: I mean, let's give them credit.

MR. SANTOS: Yes.

CO-CHAIR SKILLMAN: What they're trying to do is to protect actually the line-of-sight module people on the other side with this curtain, and also protect against hydrogen.

So, here's an exact case where we see the design evolving and the essential nature of keeping the critical licensing documentation consistent.

MR. SANTOS: Correct.

CO-CHAIR SKILLMAN: To me, that's exactly what this is, and just ensuring that it is accounted for is the important piece.

MR. SANTOS: Right. Thank you for that.

Okay. The next section is 14.3.9 on "Human Factors Engineering". So, the focus of the staff's review here is to ensure that the as-built human system interface for the MCR is consistent with that resulting from the HFE design process.

And so, that led to an open item that was originally identified in Chapter 18. The staff's concern here is ensuring that the insights from the entire human factors engineering design process are appropriately applied to the as-built human system interface in the main control room. So, the ITAAC provided in NuScale's application had a design commitment for the main control that did not include changes to the HSI design that could occur after the integrated system validation phase. Staff has had subsequent discussions with NuScale since that was identified, and I think we've agreed to some revised ITAAC language that would address the staff's concern. So, I anticipate this open item being closed in the next phase.

The second open item has to do with the staff's review of ITAAC looking at the system-level displays, alarms, and controls. There is an ITAAC to verify this for the main control room, but there is not an ITAAC for these displays, alarms, and controls for the remote shutdown station. So, NuScale has submitted a request for an exemption from GDC 19 that requires equipment outside the control room be capable to bring the reactor to cold shutdown if the main control room is evacuated. If this exemption request is approved, the staff agrees that no ITAAC would be needed because the remote shutdown station is not being credited for meeting GDC 19.

CO-CHAIR SKILLMAN: How can it not be credited? I mean, it's identified in their documentation on 7.1.1.2.3. They credit the shutdown station as the go-to when you have to evacuate the control room.

MR. SANTOS: But I think they had a subsequent -- in March of this year, they submitted another docketed letter requesting an exemption from GDC 19 that provides markup to Tier 1 and Tier 2, requesting exemption for GDC 19.

CO-CHAIR SKILLMAN: I guess I'm -- what's the right word? -- I'm confused. I mean, they've taken credit for the RSS. It seems to be what we would have called your backup, your remote shutdown station.

MR. SANTOS: Yes.

CO-CHAIR SKILLMAN: It's protected on purpose. Why would they seek an exemption?

MR. GREEN: This is Brian Green, Human Factors.

The intent, as I understand it, is that

there will still be an RSS that will provide information to the operators in the case that they need to go there. However, there will not be control functions at that station; rather, manual actions at the MPS cabinets would be how the operators would perform that function. And this is under review in the exemption request.

CO-CHAIR SKILLMAN: And then, would the exemption request include changes in the documentation in their Chapter 7? I would guess so.

MR. SANTOS: There are changes to 7. I don't remember -- I think there are changes to 7 in there.

MEMBER BLEY: I'm also a little confused because, even if you don't control with switches at the RSS, you're using those instruments to coordinate the manual operations. And why wouldn't you need an ITAAC for the instruments, whether or not you have controls there?

MR. GREEN: Yes, I'm not the reviewer conducting that review. So, I can't get into too many of the details. But these are the things that staff is considering as part of this exemption request.

CO-CHAIR CORRADINI: Can we get the Applicant to say something?

MEMBER BLEY: He's standing up.

CO-CHAIR CORRADINI: Oh, good. I recognize him.

MR. MAXWELL: So, there are no manual operations --

CO-CHAIR CORRADINI: And who are you first?

MR. MAXWELL: Sorry. Chris Maxwell, NuScale Power.

MEMBER BLEY: Chris, that's for the record when you're not up here.

MR. MAXWELL: The equivalent of GDC 19 credit is taken for actuations in the MPS rooms themselves that will trip the reactor, isolate containment, initiate DHRS.

First, let me back up one step and say that we have switches dedicated to that in the control room. So, the expectation is the operators perform that action prior to evacuating the control room to begin with, but, then, we still provide those actuation switches at the MPS cabinets. That establishes safe shutdown of the modules with no further operator action. So, when the operators take over the remote shutdown station, they do have indications there, but they don't use those indications to direct manual actions. It's really balance of plan beyond that point.

CO-CHAIR SKILLMAN: Well, let me take this on because the text in your application, at least in my mind, is crystal clear. "The RSS provides an alternate location to monitor and to operate." And then, later on three or four paragraphs, "The MPS manual isolation switches are mounted in a Seismic I enclosure. The MCS equipment in the RSS provides an independent alternative shutdown capability that is physically and electrically separate."

So, I don't know why you would be asking for an exemption from 19 when you're actually fulfilling it.

MR. MAXWELL: That wording has been revised.

CO-CHAIR SKILLMAN: 10-4. Okay. This is Revision 2.

MR. MAXWELL: Yes, sir.

MEMBER BLEY: Well, maybe not 10-4 all the way around the table.

CO-CHAIR SKILLMAN: Well, okay, Skillman, 10-4.

(Laughter.)

Go ahead, Dennis.

MEMBER BLEY: I agree changes there would make things consistent, but what Chris described is how they expect things to happen.

CO-CHAIR SKILLMAN: Yes. I'm just saying --

MEMBER BLEY: In a real-world event that drives you out of the control room, maybe they don't happen the way you expect them to happen, and you really need to be able to observe there. The other side, I'm not sure about because it isn't talked about here.

At some point in degraded operations, at least at other facilities, control can actually move to the technical support center, where you would, likewise, want instrumentation that's reliable and informative. And you haven't talked about that at all.

MR. MAXWELL: Let me elaborate.

MEMBER BLEY: Feel free to respond, yes.

MR. MAXWELL: The controls, the switches in the control room that we expect the operators to operate, those are safety-related switches with separate divisions controls. The switches at the MPS, that is the location remote from --

MEMBER BLEY: No, no.

MR. MAXWELL: -- the control room, again, safety-related switches operate to begin to trip the reactor, isolate containment, initiate DHRS. So, there's the redundancy outside the control room. The indications still are provided at the remote shutdown station and at the technical support center.

MEMBER BLEY: Yes, but it's the indications at the remote shutdown station that you don't want to have an ITAAC on. And I'm wondering why you don't want an ITAAC on those instruments. They seem essential if you get run out of the control room. Even if it's hands-off, you still want to be able to monitor and make sure everything is working as expected.

MR. MAXWELL: I agree, as an operator, I want to monitor, but it is not safety-significant. It's not related to safety.

MEMBER SUNSERI: Well, let me ask you, though, you said that you're fulfilling Reg Guide 1.68, Pre-operational Test Program. You're going to test all these things then, aren't you?

MR. MAXWELL: Yes, the indications will all be tested as part of the pre-operational --

MEMBER SUNSERI: And calibrated and verified functional and accurate, and all that stuff?

MR. MAXWELL: Yes, all the indications at the remote shutdown station are a subset of the indications provided in the plant control system and the module control system, all verified through pre-operational testing. It just doesn't rise to the level of the ITAAC, that significance to safety.

MEMBER DIMITRIJEVIC: Well, there is a couple of human actions which are significant for safety not from the internal events, but from shutdown in Level 2, as we learned a couple of days ago. So, there is a couple, like starting the charging system and --

MEMBER SUNSERI: They're not safety-related.

MR. MAXWELL: The important human actions --

MEMBER DIMITRIJEVIC: Yes.

MR. MAXWELL: -- we don't assume a control room evacuation coincident with another accident.

MEMBER DIMITRIJEVIC: No, no, it's not -- I see what you're saying. But, I mean, you know how they defined this, the significance of that is the modeling accidents in many --

MR. MAXWELL: And we have ITAAC for those important human actions to verify that the back jumpers can perform them from the control room. All the indications controls required to inform those actions are available. And actually, again, in pre-operational testing, we demonstrate capability to do that. That is a case where the ITAAC and the pre-operational testing overlap.

MEMBER BLEY: I'm kind of arguing against myself. As I said earlier, there are areas where the pre-op testing and ITAAC overlap in ways that seem inefficient. So, I ought to be happy with this, but the reasoning that gets us there leaves me a little uncomfortable.

Thank you.

CO-CHAIR SKILLMAN: Let me just close it in my own mind. It sounds like there will be a Revision 3 of Chapter 7 that gives new information on the RSS?

MR. MAXWELL: That's correct.

CO-CHAIR SKILLMAN: Understand. Thank you.

Let's march. Keep on going.

MR. SANTOS: Okay. So, the last section we would like to discuss is 14.3.11 on "Containment Systems". There is one open item here regarding integrated leak rate testing. NuScale has requested exemption from the integrated leak rate test requirement in 10 CFR 50, Appendix J, for the Type A test. So, no ITAAC was provided for this testing.

MEMBER BLEY: Can you remind us what the Type A is?

MR. SANTOS: Oh, I was afraid you would ask that.

CO-CHAIR CORRADINI: That's when you blow up the thing.

(Laughter.)

MR. SANTOS: "Measure the primary coolant overall integrated leak rate after the containment has been completed and is ready for operation and at periodic intervals thereafter."

CO-CHAIR CORRADINI: It's the 100-and-something percent of --

MR. SANTOS: Overall integrated leak rate as opposed to local leak rate testing or containment isolation valve leakage rates.

MEMBER BLEY: I'm still not sure I understand. You have 12 containment vessels --

MR. SANTOS: Anne-Marie?

MEMBER BLEY: -- and you've got to do the testing on each of those, I assume. And are they interpreting the overall is somehow going beyond the

12 individual containments?

MS. GRADY: This is Anne-Marie Grady with the NRC.

The integrated leak rate test is --

CO-CHAIR SKILLMAN: You've got to talk into the microphone.

MEMBER BLEY: It's got to be on the record, Anne-Marie. Sorry.

MS. GRADY: This is Anne-Marie Grady with the NRC.

The integrated leak rate test, the Type A test is to pressurize the containment vessel and show that it doesn't leak beyond its allowable leakage, and that's the exemption request, on that test only. Type B, which are the mechanical penetrations, and Type C, which are the containment isolation valves, still will be tested, and they are not part of the exemption request.

MEMBER BLEY: I kind of get it.

CO-CHAIR CORRADINI: Well, normally, when you do the containment leak rate test, you test the electrical penetrations, the mechanical penetrations, the isolation valves. Then, you close it all up, pressure --

MEMBER BLEY: And see if there's any leaks somewhere.

CO-CHAIR CORRADINI: -- it at low temperature, at a given pressure --

MEMBER BLEY: I know how you do it.

CO-CHAIR CORRADINI: Okay. And that's the one they're stopping.

MEMBER SUNSERI: For this plant, they're going to be maintaining it under a vacuum all the time.

CO-CHAIR CORRADINI: Thank you.

MEMBER SUNSERI: So, they'll know if it will hold the vacuum or not, right?

CO-CHAIR CORRADINI: Right.

MEMBER BLEY: That's the argument.

CO-CHAIR CORRADINI: That would be my argument.

MEMBER BLEY: Yes.

MS. GRADY: We're going to discuss this in Chapter 6 on June 18th --

MEMBER BLEY: Oh, we'll look forward to that.

MS. GRADY: -- in great detail.

(Laughter.)

CO-CHAIR SKILLMAN: Can you give us a preview?

(Laughter.)

MEMBER BLEY: Tom now will want to keep us in suspense, right?

(Laughter.)

CO-CHAIR SKILLMAN: It works for me.

All right. Thank you. Anne-Marie, thank you.

MR. SANTOS: So, basically, the open item in 14 was just -- since Chapter 14 was preceding the Chapter 8, Chapter 6, we had an open item in 14 to account for this acceptability of the no ITAAC for this while that review is ongoing. But, like Anne-Marie said, we'll discuss in June the acceptability of granting that exemption request. And so, if that exemption request is granted, then this would address this open item and close it for 14.

CO-CHAIR SKILLMAN: Okay. Thank you.

MR. SANTOS: And so, the last slide is just the conclusion slide, basically. For those sections that have an open item, staff is not able to finalize any conclusions at this point. But, for those six sections that have open items, the staff would conclude that, pending the resolution of any confirmatory items in there, that the 10 CFR 52.47(b)(1) requirement has been met.

And that concludes the staff's presentation.

MEMBER BROWN: Okay. I have a comment, a question.

MR. SANTOS: Yes?

MEMBER BROWN: You had some backup slides which I finally looked at, since I just got them today. And I noticed that you identify the test abstracts reviewed and it included the Module Protection System Test Abstract, 14.2-63. However, you did not review Test Abstracts 61 and 62, which cover the module control system and the plant control system operations and controls.

So, I went off and looked at those two, since I just got it today and saw you didn't see them, and found that -- I've got to get to the right thing here. If I go back to Chapter 14, the two abstracts satisfactorily identify that they will verify communications to the components that they're supposed to control, both the plant control system and module control system. And they also identify, both 61 and 62 identify that they will verify their bidirectional interface between the PCS and the MCS, which is, according to the drawing, satisfactory.

The one thing they don't talk about is the communication verification from the MCS and PCS to the planet network, which is a deterministic one-way communication device. There's no mention of that at all, which is a critical communication device for verification, which we just went through a final resolution where that would be specified as not just deterministic, but a hardware-based device, not software-configured.

So, I guess my question is, they look like they are incomplete relative to the -- and no answer you can give me is going to satisfy me.

MR. SANTOS: Well, I'll let Kerri try.

MEMBER BROWN: Well, I had to throw that in just to make sure you knew I was serious; that's all.

(Laughter.)

MS. KAVANAGH: Oh, goodness. This is Kerri Kavanagh again from the Quality Assurance Vendor Inspection Branch.

For those test abstracts that we did not review, no finality was provided.

MEMBER BROWN: What do you mean by "finality"?

MS. KAVANAGH: They will be reviewed in the COL space for their application. They are open for the staff review to have a complete review when they come in under the COL. So, the staff, because it did not meet the criteria for review, for the risk-informed review, for this particular application, the staff will review those that did not provide finality, that we did not provide finality on during the COL application.

MEMBER BROWN: Why does the risk-informed criteria not include control of every component in the plan?

MS. KAVANAGH: I cannot answer that for D-RAP. You will have to ask NuScale as to how they did their D-RAP. And we provided that. The staff did that presentation in 17.4 back in March, I believe we were here. But that's not my group's review.

What I'm trying to say is that my group took the results of 17.4, we looked at the safety-related, risk-informed components that came out of that analysis, and we did the review of those test abstracts based on that input.

MEMBER BROWN: Okay. I thank you for your answer.

MS. KAVANAGH: You're welcome.

MEMBER BROWN: My conclusion doesn't change though.

(Laughter.)

CO-CHAIR SKILLMAN: Okay. Understand that. Okay.

MEMBER BROWN: I just thought I'd pass that on.

CO-CHAIR SKILLMAN: Colleagues, any further questions for the staff? We've got the gentlemen and lady in front of us. Any other questions?

(No audible response.)

Okay. Before we go to public, Colleagues, any comments, any questions?

Yes?

CO-CHAIR CORRADINI: Do we need to go to the public comments?

CO-CHAIR SKILLMAN: Well, we are at a point where --

CO-CHAIR CORRADINI: If we're not going to go to closed session, we can do public comments at the end of the day.

CO-CHAIR SKILLMAN: Well, we're going to break into a completely different --

CO-CHAIR CORRADINI: Oh.

CO-CHAIR SKILLMAN: -- regime of discussion when we go to 3.9.2. What I would like to do is to clear public comments in 14. If there are none, then we're a clean shot into 3.9. I do not believe we need a closed session.

Colleagues, closed session? I'd say no.

Before we go to the phone lines, are there any members of the public in the audience that would like to make a comment? If so, please come to the microphone and make your comment.

(No audible response.)

Seeing none and hearing none, on the phone line, if there is anyone on the phone line, would you just simply say "hello"?

(No audible response.)

Hearing none, let's close the phone line.

Tanny and staff, thank you. I want to thank the NuScale staff for the Chapter 14.

I want to change out and go into NuScale 3.9.2. I want to get as much done as we can before 1200.

(Pause.)

Ladies and gentlemen, we will continue this meeting, and this meeting now is focused on NuScale's Chapter 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment".

And with this, Marty Bryan will take the lead from NuScale.

Marty?

MR. BRYAN: Thank you.

CO-CHAIR SKILLMAN: Yes, sir.

MR. BRYAN: Good morning.

Yes, I'm Marty Brown. I'm the Licensing Project Manager for NuScale for 3.9.2. And with me I have Dylan Addison, Olivia Hand, and J.J. Arthur. And J.J. is going to take us through the areas we're going to cover today.

MR. ARTHUR: Good morning.

This morning we will be discussing Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment" with a focus on four Technical Reports that are incorporated by reference in this section. These include the NuScale Power Module Seismic Analysis Technical report, two Technical Reports for the Comprehensive Vibration Assessment Program. The first is our CVAP Analysis Technical Report; the second, the CVAP Measurement and Inspection Program Technical Report. And then, finally, the NuScale Power Module Short-Term Transient Analysis Technical Report.

So, my colleague Dylan Addison will start with the seismic analysis.

MR. ADDISON: All right. Good morning.

My name is Dylan Addison. I've been with NuScale for about two years, and most of my work is focused on the seismic methodology and on answering the staff's questions about the seismic methodology.

And let's walk right through the way that we qualify the NPM components. This graphic depicts a high-level overview of the seismic methodology. And the purpose of this portion of the presentation is to reacquaint the Committee members with how NuScale gets from a three-field seismic input acceleration. It goes through the soil, through the building, through this very unique feature of the approximately 8-million-gallon reactor pool, and then, finally gets to the loads that are applied to the NPM components for ASME stress analysis, which our ITAAC design commitments ensure that we'll complete by the time we put modules into service.

So, on the far left of the graphic, the analysis begins with a detailed 3D model of the NuScale power module in a single operating bay. And that ANSYS Model is used to tune the response of an NPM Beam Model, so that the two are dynamically equivalent. And that's performed using a modal harmonic and transient analyses. And why do we need a 3D model that's dynamically equivalent to a 2D Beam Model? Because we use SASSI for our soil structure interaction analysis, and SASSI requires a simple 5 Beam Model representation in order to run in a reasonable amount of time; and also, because SASSI is not capable of explicitly modeling the fluid in the pool.

So, what do we do? We take the NPM Beam Model representation, and in this second image, we're showing the SASSI Model, the reactor building with the light blue soil elements on the outside, the dark blue reactor building, and the red Beam Model representations of the NPMs. Twelve copies are inserted into the model, and a soil structure interaction analysis is performed in the frequency domain. And the results are, then, transformed back into the time domain and used as inputs in the next analysis.

And what are those results? They are time history accelerations at the NPM supports, which are the containment vessel seismic lugs, the three lugs about halfway up the module and at the containment vessel skirt. And there are inputs to the entire reactor pool -- that is the pool walls and the pool floor -- that are mapped onto to this third model, this yellow model, which has created in ANSYS of the entire pool. And we do it this way because you have to work from the outside in. You have to perform the SSI analysis first, and we also need to account for how the pool reacts to seismic excitation.

CO-CHAIR CORRADINI: So, does ANSYS model the water?

MR. ADDISON: That's right. These yellow elements are fluid 30 elements in ANSYS.

CO-CHAIR CORRADINI: So, it will try to describe what's happening to the pool as you shake it?

MR. ADDISON: That's right.

CO-CHAIR CORRADINI: Okay.

MR. ADDISON: And the input accelerations from the SSI analysis are mapped onto the whole surface of the pool. And in the corner there, you see in position 6 of 12 is the detailed 3D model of the reactor. And that's the same 3D model that's used initially to tune the Beam Model.

MEMBER BLEY: Can you -- well, you'll have to tell us something about the range of earthquakes you'll consider. And when there is a COL, you'll have to redo, I guess, for the specific seismicity at the site. But can any of your calculations show water sloshing out of the pool? And if it does slosh out, is there an easy path for it to get back or can we lose water?

MR. ADDISON: We have analyzed for sloshing, and it's, I think, on the order of about 2.5 feet to 3 feet, and it's not, past that analysis, it's not necessary to look at.

MEMBER BLEY: Okay.

CO-CHAIR CORRADINI: So, since we're onto sloshing, it, then, transmits a load to the vessel, or the containment -- excuse me -- and that's accounted for in terms of it pushing on the vessel as I have a seismic event?

MR. ADDISON: So, sloshing is analyzed separately.

CO-CHAIR CORRADINI: But it creates a flow past the vessel, the containment vessel, which, then, creates an asymmetric load, yes?

MR. ADDISON: It would and --

CO-CHAIR CORRADINI: And that analyzes part of the seismic event?

MR. ADDISON: And it's analyzed separately. In this model, in this 3D ANSYS Model, we're not looking at sloshing because a separate analysis shows that it's not necessary to look at.

MEMBER SUNSERI: Let me --

MEMBER BLEY: That's interesting, and we haven't talked about that. But I assume the excitation to the model is much stronger through the mechanical connections?

MR. ADDISON: That's right.

MEMBER BLEY: I'm sorry, I cut you off.

MEMBER SUNSERI: No, I think that was the same question I had. Normally, I mean, the module is still supported to err through the structure, right? But the difference is, instead of being surrounded by air, it's surrounded by water in this case. So, how does that affect the stresses on the support? That's what Dennis just asked.

MEMBER BLEY: Right.

MR. ADDISON: How does the fact that it's surrounded by water affect it? Well, by virtue of the fact that we model it, we know what the loads end up being. And the acoustic resonances in the pool are accounted for in this analysis.

CO-CHAIR CORRADINI: I guess my way of asking my question is, if the water weren't there and it was air, would I see a significant difference in how it shakes?

MR. ADDISON: Yes.

CO-CHAIR CORRADINI: Okay.

MR. BRYAN: So, I think he's just wanting to confirm that in a seismic event the effect of the water is considered.

MR. ADDISON: Absolutely.

CO-CHAIR CORRADINI: Okay. Fine.

CO-CHAIR SKILLMAN: Let me ask this, Dylan: that pool is a fairly long pool. The good news is that each reactor is in its own bay. So, it's in a cell. So, whatever the water conditions might be, it's relatively isolated, each module from the other module.

Is the compressibility of the water a factor in this analysis? Does ANSYS assign some, if you will, some compressive characteristic to the fluid as a mechanical transition, a --

MR. ADDISON: I believe the fluid is assumed incompressible.

CO-CHAIR SKILLMAN: Thank you. All right.

MR. ADDISON: Yes.

CO-CHAIR SKILLMAN: Thanks.

MR. ADDISON: So, the outputs of this entire pool analysis in ANSYS are enveloped and broadened in structure response spectra throughout the module, bounding forces and moments, bounding relative displacements at different locations through use as seismic anchor motions in piping analysis, for instance, and also, time histories throughout the module. And all those results are used there again as inputs to downstream stress analysis, which is what actually qualifies the components.

So, this rather light graphic we included to make clear another unique aspect of the design, which is the refueling area. And because we move the whole module in order to refuel it, an analysis is performed for the scenario in which the lower reactor pressure vessel and the lower reactor pressure vessel internals and the fuel are situated inside the reactor flange tool, the RFT. And that time history analysis ensures that the whole assembly remains upright and the integrity of the fuel is maintained during a seismic event.

CO-CHAIR CORRADINI: What is the clearance between the containment and the wall? One individual NPM has got a concrete wall on either side of it. What's the clearance?

MR. ADDISON: I don't have that number in front of me. But, just from the scale, on the order of 10 feet or --

CO-CHAIR CORRADINI: Oh, 10 feet?

MR. ADDISON: -- 8 feet maybe, yes.

CO-CHAIR CORRADINI: Oh, I just was looking at it. It looked awful thinner than that. So, that's why I --

CO-CHAIR SKILLMAN: Oh, no, they're actually trunnions.

MR. ADDISON: Oh, maybe I misunderstand the question, yes.

CO-CHAIR SKILLMAN: And it appears as though, based on the drawings that I saw yesterday, that the insert of the module is actually into a slip that ensures there is neither east-west nor north-south movement. So, it's actually locked in concrete buttresses that are highly reinforced.

CO-CHAIR CORRADINI: That's what it looked like.

MR. ADDISON: Yes, I think I misunderstood the question.

CO-CHAIR SKILLMAN: And I would say maybe it's pretty close. I mean, it's --

MR. ADDISON: The lugs interface directly with corbels on the --

CO-CHAIR SKILLMAN: Bingo.

MR. ADDISON: -- reactor bay wall.

CO-CHAIR CORRADINI: So, there is a structural support about midway up?

MR. ADDISON: That's right.

CO-CHAIR CORRADINI: Is that what I'm seeing?

MEMBER BLEY: That's that thing, yes.

CO-CHAIR CORRADINI: Okay. Thank you.

MR. ADDISON: Thank you.

MEMBER BLEY: Dylan, when you do the analysis for the refueling area, did you look at cases with everything intact or cases where some of the fuel is actually in the process of being moved when the earthquake hits?

MR. ADDISON: The only scenario analyzed is where the lower reactor vessel internals are still in place, still bolted on.

MEMBER BLEY: So, you would get something different, no doubt, if some rods were moving?

MR. ADDISON: Yes, and that would be a highly non-linear analysis that --

MEMBER BLEY: Not fair to you, yes.

MR. ADDISON: -- is not practical to perform and doesn't make sense to perform in this --

MEMBER BLEY: So, when we look at the -- it would have been good if we had had this before we had the PRA presentation. When we look at now the seismic margins analysis and later the complete seismic PRA, this issue of what happens to fuel that might be in the process of being moved is one that has not been addressed as yet, I think. We didn't ask --

CO-CHAIR CORRADINI: Can you have a seismic event in transit, during transit?

MEMBER BLEY: You don't know when the seismic event is going to hit.

CO-CHAIR CORRADINI: I know, but that's what you're asking.

MEMBER BLEY: That's what I'm asking. Well, yes, you have it in transit, but also after you're in the refueling bay when you're halfway through the refueling process. See, not everything is where it is assumed to be here, which provides -- okay. We'll have to look at that later when we get those people back again.

MEMBER REMPE: But now, when we know that, I guess, each module is at some location, almost or in contact with something from the bay for each module, the distance does look -- the gap. Can you give us some location? I mean, down low it looks like there's quite a bit of distance, well, a bit more distance, but other places it looks like, well, the cylindrical part of the containment vessel looks fairly close to the concrete wall. How big is that distance? Is it like 10 inches or something less than a foot? Like right where your hand is right there, how big is that gap?

MR. ADDISON: Well, if the whole module is 70 feet high or so, then, yes, we are looking on the order of a foot or two.

MEMBER REMPE: Okay.

MR. ADDISON: Yes.

MEMBER BLEY: Is the module anchored on the bottom or just supported by those things that were called trunnions before?

MR. ADDISON: It's held at the bottom by a passive support. It's a big ring that the CNV skirt, the containment vessel skirt, sits within.

MEMBER BLEY: Okay.

MR. ADDISON: All right.

MEMBER BLEY: That's part of the analysis, the stresses down there?

MR. ADDISON: Right, the CNV skirt is analyzed.

So, here we just have an overview of all the Service Level D stress analyses that NuScale has performed and which are under revision, which demonstrate that the loads produced by this methodology are acceptable. And the bottom line here is that our ITAAC design commitments ensure that NPM components conform to the rules of ASME Section 3.

And that brings us to your point from earlier, which is the one COL item relevant to Section 3.9. It essentially states that site-specific seismic analysis will be performed and the results will be compared to the results generated for the design certification, and where the results are not bounded, action will be taken to demonstrate how to put margin in or redesign the components.

With that, I will turn it over to Olivia Hand.

DR. SCHULTZ: Just to follow up on your comment, Dennis. And that is, in this configuration with the 12 modules and refueling, and so forth, even though it's a very difficult analysis, we ought to have some consideration of what the impact would be, given that for about 10 percent of the time during the year you're going to be moving a module, refueling a module. A lot of activity is associated with that, with a module not in its bay.

MEMBER BLEY: Yes, it's not and it's a substantial fraction of the time.

DR. SCHULTZ: That's right.

MEMBER BLEY: Yes.

DR. SCHULTZ: It's different than what we're used to thinking about.

MS. HAND: Okay?

CO-CHAIR SKILLMAN: Yes, please proceed, Olivia.

MS. HAND: My name is Olivia Hand, and today I'll be talking about the Comprehensive Vibration Analysis Program and, also, the Short-Term Analysis Methodology.

So, the Comprehensive Vibration Analysis Program is designed to look for mechanisms of flow-induced vibration in the plant, and to either preclude them by design for the strongly coupled mechanisms or just show that over the design life it's not going to provide any vibration energies that are detrimental to the components.

And so, we look at both components in our primary coolant flow path, which, of course, you guys know is natural circulation, and we also look at the secondary side. So, we're analyzing everything inboard of the disconnect flanges, so all the containment components, all of the reactor vessel components.

And the analysis program consists of three parts, in accordance with Reg Guide 120, which is to perform analysis, then perform validation via measurement, and also to inspect. So, the first step of the analysis program, of course, is to screen all of the components. So, we go through every component in the module and we see if it meets a screening criteria for one of the six FIV phenomena that we consider, which are vortex shedding, fluid-elastic instability, acoustic resonance, turbulence, flutter, and gallop.

So, an example of this, for example, vortex shedding, we're looking at a component that's bluff body, exposed to crossflow. So, we could use some operational experience to, say, screen out things that we know have been operating in PWRs for many years without issues; for example, maybe thermowells or steam generator tubes.

But we take a conservative approach and we just look at straight screening criteria. And if the component meets the screening criteria, we include it in the analysis program.

CO-CHAIR SKILLMAN: Does that mean you basically excluded OE as a basis to not consider a component?

MS. HAND: We excluded OE, yes, in the case that it would not allow us to analyze it.

CO-CHAIR SKILLMAN: Yes. Okay. Thank you.

MS. HAND: Because we wanted to be conservative and analyze everything, you know, to come up with a safety margin and prove to ourselves whether or not it is something that needs to be considered in the measurement program for validation. And then, ultimately, everything that gets screened into our analysis program is inspected following initial startup tests. So, we want that level of assurance as well.

CO-CHAIR SKILLMAN: I think you're to be commended for that because this unique design would simply invite, "Why didn't you look at that when you had the chance?", if there were to be an incident many years down the road. By taking the high ground and analyzing everything, you've pretty much provided a basis to respond to that challenge.

MS. HAND: Yes. Being a first-of-a-kind design, you know, it's not just a design that we're scaling up where we said, okay, well, this is what we did last time, so we'll do the same this time.

CO-CHAIR SKILLMAN: It's different.

MS. HAND: We needed to start from scratch.

CO-CHAIR SKILLMAN: Thank you.

MS. HAND: So, that's the approach that was taken.

So, as mentioned, we move into the analysis program where we use industry standard methods for coming up with a safety margin, which is defined as either the margin that you have to the onset of a phenomena that's strongly coupled, which is those are the phenomena that we need to preclude by design because, if we're having strongly coupled vibrations, we can't show that we would have acceptable performance over the 60-year design life. Or for turbulence, it's with respect to the fatigue usage factor for the component, based on the alternating stresses.

So, we perform the analysis program, and then, any components with a safety margin less than 100 percent, we commit to validation via the measurement program. And then, as I mentioned before, all components that are a part of the analysis program will be inspected at the completion of initial startup testing.

And then, the figure on the right is just a view of our natural circulation primary flow path for the reactor module. I believe you probably saw this in Chapter 5. And one thing to point out, this image does not show the primary and secondary coolant piping that's part of the RPV and CNV connection and up to the disconnect flange. So, that's included in our program, but not shown in this figure.

We do have some differences compared to recent applicants. So, we're analyzing a lot of components that have not been traditionally the focus of past programs; for example, the steam generator tubes. We also are considering mechanisms that apply to these components like vortex shedding and fluid loss instability; whereas, in the past, applicants have mostly focused on turbulence and associated degradation mechanisms.

We have very much lower primary coolant flow rates. The table on this slide provides a comparison of some of the average velocities that we see throughout the module compared to past designs. And if you think about this more in terms of flow rate, that's also provided, and primary coolant loop transit time, which is just the amount of time it would take for a control volume to move from the core all the way through and back. It takes about a minute for us. So, that kind of gives you a feel for the velocities that we're seeing inside of the module.

CO-CHAIR CORRADINI: But don't you have higher steam flow rates inside the tubes? Steam mass flux in inside the tubes because you've got this kind of reversed situation?

MS. HAND: Yes. So, our secondary coolant is inside of the tubes. I'm not actually sure whether our velocities are higher than primary coolant flow inside of a PWR tube. I've never actually done that comparison.

CO-CHAIR CORRADINI: Okay. Well, I'm trying to decide where to worry about things. I'm not worried about the outside of the tube because your Reynolds number has to got to be super-small, but I'm thinking about inside the tubes. So, you're also doing flow-induced vibrations because of inside flow?

MS. HAND: We are. We account for that in our turbulent analysis. And we'll have a slide coming up where we show some of our margins where we actually do have the need to validate some of our analysis for flow outside the tubes.

CO-CHAIR CORRADINI: Okay.

MEMBER BALLINGER: What does "no proprietary scale testing" mean?

MS. HAND: It just means that, since this is a first-of-a-kind design, we don't have previous testing of other reactor designs that we're using to inform our analysis approach. We do have some limited benchmark testing that has been used to help inform our analysis approach, but --

MEMBER BALLINGER: So, there would be no proprietary scale testing available? I'm just trying to --

MS. HAND: Yes.

CO-CHAIR CORRADINI: They don't have anything available.

(Laughter.)

MEMBER BALLINGER: Yes.

MS. HAND: Not available.

CO-CHAIR CORRADINI: Lt me ask about those tests because we were told about -- and again, if we're going into something that's got to go to closed session, you stop us. But my understanding is you did -- I can't remember exactly -- but you did what I'll call a single row of tubes for testing in a test facility outside of NuScale or done for NuScale, is that correct?

MS. HAND: It's actually two separate test facilities.

CO-CHAIR CORRADINI: Right.

MS. HAND: So, tubes, yes.

CO-CHAIR CORRADINI: But the helical geometry of the testing was done in that test at full flow for the steam, steam production?

MS. HAND: Yes.

CO-CHAIR CORRADINI: Okay. And it wasn't a complete generator, but it was one column of tubes? Am I remembering correctly?

MS. HAND: So, TF-1 test was three single tubes.

CO-CHAIR CORRADINI: Right.

MS. HAND: Yes. We have some pictures coming up.

CO-CHAIR CORRADINI: Okay.

MS. HAND: And then, TF-2 is a five-column assembly.

CO-CHAIR CORRADINI: Okay. You have pictures?

MS. HAND: I do, yes.

CO-CHAIR CORRADINI: Okay. Fine. We'll wait then.

MS. HAND: Okay. So, just to close out this slide, we're going to be performing the majority of our validation testing prior to startup testing. This is unique and it's something that we're choosing to do just to make sure that we get high-quality, high-quantity data to close out any of our validation concerns prior to actually building the first module and getting it in place, which would be a more precarious time to find an issue.

And we have a larger inspection scope. So, we're inspecting everything that's screened into our analysis program. We're going to be looking at over 50 locations in the module after startup testing.

CO-CHAIR SKILLMAN: Before you change, what are you communicating on steam generator gap?

MS. HAND: So, the gap velocity is defined as the velocity as it moves through the steam generator tube bundle. So, it's the velocity through the tubes.

CO-CHAIR CORRADINI: It's the flow velocity when it's between the two tubes?

MS. HAND: In the gap of the tubes.

CO-CHAIR CORRADINI: In the gap?

MS. HAND: Uh-hum.

CO-CHAIR CORRADINI: Think of it as ratio. It's the flow inside the narrow portion, right?

MS. HAND: Correct.

MEMBER SUNSERI: So, that kind of ratio, it's a ratio that, then, I would think the answer to your question earlier, what's the flow rate inside the tube, it's going to be, you know, a factor higher than --

CO-CHAIR CORRADINI: Well, but it's steam. So, the mass flux is going to be much --

MEMBER SUNSERI: I mean, it's similar, though, on the -- I mean, it's boiling on the SONGS plant, right? And it's not boiling on the NuScale.

CO-CHAIR CORRADINI: Oh, I see what you're saying.

MS. HAND: Yes, so this is a comparison for outside the tubes. I don't have a comparison for inside the tubes.

CO-CHAIR CORRADINI: But I know what he's asking.

MS. HAND: Okay.

CO-CHAIR CORRADINI: I know what he's pointing out. I'm with you.

CO-CHAIR SKILLMAN: What triggered my question is, I'm pretty familiar with steam generators. That appears to me to be gap tube center line to tube center line minus the halves of the tubes. So, it's the gap between the OD of one tube to the adjacent OD of the next tube. I think that that's what that is. But the 18 on SONGS is the difference between a flat plane of tubes, I think.

MS. HAND: So, this is a velocity reported. So, it's in feet per second.

CO-CHAIR SKILLMAN: Uh-hum.

MS. HAND: So, it's not referring to the --

CO-CHAIR SKILLMAN: Geometry?

MS. HAND: -- pitch or the diameter, although those feed into this, how it's calculated.

CO-CHAIR SKILLMAN: It is the mass flow rate through that gap, the velocity of the steam going through that gap?

MS. HAND: And also, the density.

CO-CHAIR SKILLMAN: Yes.

MS. HAND: And then, the pitch diameter.

CO-CHAIR SKILLMAN: Thank you. All right.

MS. HAND: Okay. So, this slide has a summary of some of our analysis results. First, just high-level, you know, how we perform our design analysis. Most of these require modal response inputs. So, we determine our frequencies and mode shape. We determine our flow velocities past and through the components of interest. And then, we generally use Appendix N guidelines to help inform our analysis methodologies.

All 50 margins reported in this table are positive, which means that FIV is not predicted to occur for the strongly coupled mechanisms or for the case of turbulence. You know, we're not running up against a fatigue factor limit.

And our limiting regions, reported in this table, are the helical coil steam generator, in-core instrument guide tubes, the control rod drive shaft, and the decay heat removal system steam piping.

CO-CHAIR CORRADINI: Wait. So, the analysis category, what is "TB"?

MS. HAND: Turbulent buffeting.

CO-CHAIR CORRADINI: Ah, okay. So, vortex shedding behind the tube?

MS. HAND: Vortex shedding at the very bottom row of the steam generator tubes.

CO-CHAIR CORRADINI: At the very bottom?

MS. HAND: Yes. So, vortex shedding won't occur if you have a downstream flow obstruction. So, it's only the very bottom of the tubes that are susceptible.

CO-CHAIR CORRADINI: This is the primary flow?

MS. HAND: Primary coolant flow, correct.

CO-CHAIR CORRADINI: Okay. Thank you.

MS. HAND: And then, FEI is fluid-elastic instability. And then, down for the DHRS steam piping, AH is acoustic resonance. So, that's happening in --

CO-CHAIR CORRADINI: The what? I'm sorry.

MS. HAND: Acoustic resonance which occurs in the flow-occluded cavity. So, when you have high-velocity flow, it generates a vortex and it could lock in with an acoustic frequency in the cavity.

DR. SCHULTZ: The "most limiting results" refers to what? Is that location? Or can you help describe what that means?

MS. HAND: Yes. So, we think about our limiting results in terms of the safety margin, which is the combination of velocities and frequencies and how much margin we have to the onset of the phenomena occurring. So, this table shows that our most limiting location is the helical coil tubes for fluid-elastic instability.

DR. SCHULTZ: Okay. Thank you. So, location by location.

MEMBER BALLINGER: So, is 10 percent considered okay?

MS. HAND: We believe so, because we believe we have aspects of our design analysis that are potentially overly conservative.

MEMBER BALLINGER: Because Section 3, there's a piece of Section C that says, if you think you're designing within a certain window, you're okay. But if you have reason to believe that you could be outside of the normal sort of database that exists, that you need to do some additional testing, which is one of the things that got SONGS. So, is 10 percent okay?

MS. HAND: Well, so we're doing additional testing to demonstrate that 10 percent is okay --

MEMBER BALLINGER: Okay.

MS. HAND: -- and to validate that safety margin.

MEMBER BALLINGER: And that includes all wear issues, and stuff like that, that could exist?

MS. HAND: So, when I think of wear at least, I think of that as more of a turbulence lifecycle issue, which is something that we track over the 60-year life of the component through inspections. If fluid-elastic instability occurs, that is degradation that will cause tube damage and leaking.

MEMBER BALLINGER: So, is there any likelihood of threshold behavior? Meaning that you've got this 10 percent margin, the fluid-elastic instability, is there anything that's non-linear in the analysis where, if you get a certain amount of wear, where you increase a gap here and there, you go from a good hair day to a bad hair day?

MS. HAND: The analysis is highly non-linear, yes.

MEMBER BALLINGER: That's what I suspected. So, that's what I keep wondering about this 10 percent business. My wife wouldn't like that.

MEMBER SUNSERI: So, let me ask you this, though --

MEMBER BLEY: You're worried that it might be kind of cliff edgy?

MEMBER BALLINGER: Yes. Well, she said it's cliff edgy.

(Laughter.)

MS. HAND: But what I will say is NuScale's position is that we've actually biased --

MEMBER BALLINGER: Okay.

MS. HAND: -- the majority of our inputs to get us to that cliff edge.

MEMBER BALLINGER: So, if you do what would amount to --

MS. HAND: So, a less conservative analysis would show that we have --

MEMBER BALLINGER: If you do what amounts to a best estimate analysis, as opposed to all these conservatisms, what do you get? In other words, how big a number does it -- how does it change?

MS. HAND: So, we have not provided that yet to the NRC. That's actually part of our pretest prediction work that we do for these two mechanisms for our upcoming TF-3 test. Preliminary results are showing approximately about 80 percent.

MEMBER BALLINGER: Eighty percent?

MS. HAND: Yes. Those will be submitted --

MEMBER BALLINGER: That makes me feel better?

(Laughter.)

MEMBER SUNSERI: So, the application of the helical coil steam generator to NuScale is unique, but they're not unique in the industry, right?

CO-CHAIR CORRADINI: No.

MEMBER SUNSERI: So, how does this data compare to industry-available data on the ones that are in service someplace?

MS. HAND: We have not actually been able to find much data that would give a parameter like a safety margin. We've used industry data related to helical coil, steam generator Connors' constant coefficients, which are our design analysis input for fluid-elastic instability. And those compare very favorably with straight tube.

CO-CHAIR CORRADINI: What are you comparing again? Could you repeat that, please?

MS. HAND: The Connors' constants which are used in the fluid-elastic instability analysis.

CO-CHAIR CORRADINI: Oh, okay.

MS. HAND: So, they're like an empirical-fed coefficient and exponent that you use to come up with the safety margin calculated value. And the helical coil ones provide a much higher safety margin than the straight tube ones, but we have conservatively used the straight tube ones in our --

CO-CHAIR CORRADINI: Straight tube meaning this or straight tube meaning crossflow? When you say "straight tube," I'm trying to understand. Is it flow this way over the tubes or is it flow straight in terms of perpendicular crossflow?

CO-CHAIR SKILLMAN: Transverse.

CO-CHAIR CORRADINI: This way?

CO-CHAIR SKILLMAN: Oh, it's linear or transverse? That's the question he's asking.

MS. HAND: I think that's correct.

MEMBER SUNSERI: No, she's talking about

a tube that goes like this versus one that goes back.

CO-CHAIR CORRADINI: Right, but it's still external to the tube? Am I understanding correctly?

MS. HAND: Yes, fluid-elastic instability is only applicable when the flow is on the outside of the tube bundle.

CO-CHAIR CORRADINI: Yes. But I'm with you. It's this versus that.

MEMBER REMPE: Just to make sure I understand, on the separate effects tests for the steam generator tubes, are you going to be considering all of the different, like different mode shapes, different frequencies, even though it's a little bit of integral test that considers all of the inputs that you want to verify, right?

MS. HAND: Yes.

MEMBER REMPE: Or are you doing separate effects for each of these things?

MS. HAND: No, separate effects in that it's just looking at the steam generator. And then, in that testing, we're doing modal testing and flow testing. So, as a part of the modal testing, we'll be doing both hammer and shaker testing to determine mode shapes, frequencies, and damping. And we'll be performing that both in air and in water to determine the effect of hydrodynamic.

MEMBER REMPE: Okay. Thank you.

MS. HAND: If there is nothing else on this slide, I'll move on to the next.

CO-CHAIR SKILLMAN: Yes, please do, yes.

MS. HAND: So, this actually shows pictures of the test facilities that we've mentioned a bit.

So, we have as NuScale these two dedicated benchmark tests, TF-1 and TF-2, that we use to help inform our design analysis. TF-1 is shown on the top left. And as mentioned before, it's actually a test facility with three electrically-heated tubes where we're mainly just focused on the thermal-hydraulic phenomena happening inside of the tube. And tubes 2 and 3 were instrumented with acoustic pressure sensors to look at the effect of the turbulence due to the flow and boiling inside the tube. And they were located at five different vertical heights along the tubes.

CO-CHAIR CORRADINI: So, I don't understand 3. I mean, we're looking down the throat of it, but is it one column of tubes?

MS. HAND: So, yes, in this orientation, actually, both the TF-2 and TF-3 images are shown on their side. During flow testing, they're lifted to be vertical.

CO-CHAIR CORRADINI: And it's essentially, as we see with TF-2, it's a single column? It's a single stacking versus multiple stackings?

MS. HAND: So, in the TF-3 image, you're actually looking at two columns installed. So, TF-3 will, when it's completely built, consist of the five middle columns.

CO-CHAIR CORRADINI: Oh, five?

MS. HAND: And it will be full prototypic in height, and those are fully prototypic in the dimensions of the columns and, also, the tube supports. But we modal test. We're modal testing columns 12, 11, and 9. So, that's showing like the extent that is installed to allow modal testing of our first column.

CO-CHAIR CORRADINI: Are you allowed to say where is this testing being done?

MS. HAND: It's in Piacenza, Italy.

CO-CHAIR CORRADINI: Oh, it's still with Ansaldo? Or whatever is the company now?

MS. HAND: Yes.

CO-CHAIR CORRADINI: Okay.

MS. HAND: Yes.

MEMBER BALLINGER: Is that 690 tubing?

MS. HAND: It's actually not. As an engineering simplification for the testing, we just use stainless steel because the material properties are similar enough for our modal testing and flow testing needs.

Okay. So, to go back to TF-2, I guess, TF-2 kind of was the next iteration of testing after TF-1. So, it's a fluid-heated test facility where we had flow on the inside and the outside. It was also five columns and they were toward the outer third of the steam generator, I think between half and two-thirds. That kind of corresponds to the bend radius of the tubes.

CO-CHAIR CORRADINI: Are we allowed to -- how many columns, are you allowed to say, are the full-scale steam generator?

MS. HAND: Twenty-one.

CO-CHAIR CORRADINI: Twenty-one?

MS. HAND: Uh-hum. And in TF-2, we instrumented that facility with strain gauges which were located about a third of the way and two-thirds of the way up on the tube pipes.

And one thing to note about the TF-2 test facility, we are still in the design process of finalizing our steam generator tube support designs. So, the TF-2 tube supports are only located or they support at four radial locations throughout the bend; whereas, the final design support is at eight locations. So, based on that, the TF-2 testing, we expect that those tubes are kind of, you know, have lower frequencies, less stiffness than our final design.

We tested up to 200 percent of our licensing basis flow rates for the primary coolant in that test facility, and we did not experience any fluid loss against stability. So, that gives us some good feeling about the results that we expect to see in our final validation tests with our final design tube supports installed.

MEMBER BALLINGER: This is just kind of a personal observation. Six ninety, that is going to be the first use of 690 for this kind of configuration, I believe. Has there been any thought to actually using 690 for some of these tests, so that you gain experience from the actual fabrication using 690 in this configuration?

CO-CHAIR CORRADINI: You're talking more a fabrication issue.

MEMBER BALLINGER: Well, I mean, it's a

freebie. I mean, if you're building a test facility already anyway --

CO-CHAIR CORRADINI: Nothing is free.

MEMBER BALLINGER: Okay.

MS. HAND: Yes, so I will say that I wasn't directly involved with it, but I know we gained a lot of experience through just the process of making TF-3, regardless of the material that was used.

CO-CHAIR CORRADINI: Was TF-1 690 or was it also stainless steel?

MS. HAND: I am not sure --

CO-CHAIR CORRADINI: Okay.

MS. HAND: -- but at that time I can say that we probably had not even selected our material --

CO-CHAIR CORRADINI: Okay. Fine.

MS. HAND: -- for the steam generator tubes. That testing was many years ago.

CO-CHAIR SKILLMAN: Olivia, I've been wanting to ask this question since I saw this design. When the individuals are creating this tube bundle, are the individuals actually in the cylinder on their hands and knees with pads, actually placing the tubes into the locking devices? I say that because I watched the B&W steam generators being assembled. Those tubes are around 80-feet long. And we recruited capable, but smaller people to be in there to feed those tubes --

(Laughter.)

-- all 15,100 of them. I mean, that was a learning experience to fabricate the OTSGs, which are handcrafted just like this. I'm just wondering, is this a handcraft operation or?

MR. BRYAN: Let me turn it over to Tamas Liszkai to address the 690 and fabrication part of it.

Tamas, are you there?

CO-CHAIR SKILLMAN: Okay.

MR. LISZKAI: Good afternoon. This is Tamas Liszkai.

Yes, just wanting you to know that we kept the mechanical properties. Alloy 690 and stainless steel, for practical purposes, are identical. And it's cheaper to procure that.

However, we did do fabrication testing with alloy 690 tubing for our prototypical design. That was a separate project from this one. It's not included in the slides. But we used TCP as our Center for Advanced Manufacturing in Philadelphia, to apply the tubes in full prototypical length, alloy 690 tubes, and they used commercially-available banding apparatus to actually do the 3-dimensional banding of our tubes. The results of that were that everything stays within tolerances.

CO-CHAIR SKILLMAN: Thank you.

So, how is the tube inserted into the bundle?

MR. LISZKAI: This is Tamas Liszkai again.

I can answer that. It's not inserted into the bundle. The current fabrication methodology that we employ is column by column, going from the outside in. And so, you would be placing the tubes individually or in a group, slide it up in the annulus. You would sort of uncoil them and insert them down into the feed plenum. And then, you would use these individual support bars to slide them over the T-bar on top of the baffle plate, and just proceed column by column, basically.

We have worked with two fabricators to develop the fabrication process of that. And they identified a very similar process of installation.

MEMBER BALLINGER: It's a 20-ton piece of jewelry.

CO-CHAIR SKILLMAN: Thank you. I think I can go from here. Thank you very much. That was very helpful.

MEMBER SUNSERI: I have one question, back again on this fluid-elastic instability. That's, I'll call it, the new known/unknown, made famous here recently by some operating experience. My question is, have you convened any kind of expert panel or anything to think about what other phenomena might occur in this kind of style of steam generator that you should be highlighting as your analysis point?

MS. HAND: Well, I guess my comment would be that my personal opinion is that FEI is not an unknown phenomena. It's been documented in literature for years.

MEMBER SUNSERI: Yes, I know, but nobody talked about it until more recently, right? I'm overstating that. Okay, I understand that. But my point is still, what about other phenomena that might affect this particular design?

MS. HAND: So, I mean, the NuScale approach with this is that in power applications we've had components that looked like this in the past and, essentially, even more limiting operating conditions. I don't think we're going to discover new FIE phenomena of interest. I think a lot of research has been done on that in the past and they are pretty well documented. The important part is to follow the code and to use appropriate design analysis practices to identify a problem before you get to operation and discover it.

MEMBER SUNSERI: Yes, and I don't disagree with that, but I will maybe push on just following a code that's based on current or previous experience may not be sufficient when you're going into first-of-a-kind new realms.

MS. HAND: Yes, and in that situation, you know, specifically from the steam generator design, that's why we couple our design analysis with validation testing, measurement during startup testing, and inspection following the startup testing, so that we could never get to a similar situation like SONGS of detecting this through primary-to-secondary-side leakage.

CO-CHAIR CORRADINI: That's the plan.

MEMBER SUNSERI: And just a closing thought on that. I mean, if -- "if," and I know it may be remote -- but if you discover a major problem and it's way down the road late, to recover from that.

MS. HAND: Yes.

CO-CHAIR CORRADINI: You replace the upper part of the NPM because it's integral to the NPM module, upper portion. Yes?

MS. HAND: Yes, and that's, I guess,

something important to point out with this TF-3 testing. Part of the reason, other than wanting to collect higher quality and higher quantity of data by having a dedicated test facility -- we could have just said, hey, we're going to perform the string startup testing. But that becomes a very commercially-risky thing for us to do because, if there is a problem, we want to find out years in advance of fabricating the first module.

So, actually, commissioning of the TF-3 test facility, pictures shown here, modal testing starts this summer. And then, it will be probably about a two-year program to wrap up the flow testing. That will provide us with the technical confidence going into manufacturing phase that we don't have FIV concerns for this part of the design.

MEMBER BALLINGER: But you say there wasn't much data out there. I don't recall the exact dimensions and everything, but the Fort St. Vrain steam generators were helical and they had steam being generated on the inside, I think, and gas on the outside. But were the velocity, the parameters and stuff, were they anywhere near what you're dealing with?

MS. HAND: So, I actually did look into

that because I happen to live in Denver. So, I could visit the plant. But it was just --

MEMBER BALLINGER: So, it was colder then.

MS. HAND: The design was performed such a long time ago that the documentation doesn't exist. Since it's not still an operating plant, everything has been kind of --

CO-CHAIR SKILLMAN: Lost?

MS. HAND: -- lost, lost in the shuffle. But if we could get data from Fort St. Vrain, it would be very interesting to compare against, but not important to safety, since we are doing our own testing. We did pursue that, but we couldn't find anything.

MEMBER BALLINGER: That was designed using a Commodore 64 computer.

(Laughter.)

CO-CHAIR SKILLMAN: I don't think there was a Commodore.

(Laughter.)

MEMBER BALLINGER: Oh, you didn't even have one of those, right? A slide rule.

(Laughter.)

MEMBER REMPE: Just out of curiosity, did you try General Atomics and some of the people from

GA?

MS. HAND: No, we did not.

MEMBER REMPE: It might be worth exploring.

CO-CHAIR CORRADINI: But your point is you're going to do prototypic testing with a five-column generator to document what you are predicting?

MS. HAND: Yes. And then, we're going to follow it up with instrumenting the steam generator during the startup testing and inspecting it after startup testing.

CO-CHAIR CORRADINI: Okay. Thank you.

MS. HAND: So, to move on to the inspection program a bit, any component that's screened for a mechanism will be inspected following, well, both before and after startup testing. First, just to get its initial view of what it looks like, and then, after to see if any damage has occurred.

Obviously, turbulence is a lifecycle issue. So, if we are seeing wear at the conclusion of the startup test program, that means that we have a strongly coupled mechanism that we need to look into. And we're going to provide enough flow time at initial startup test conditions to make sure that we provide at least a million cycles of vibration for all of these components. That should take about two-and-a-half days at full flow conditions for our plant. And we'll be performing inspections in accordance with ASME Section 11 guidelines.

CO-CHAIR CORRADINI: Is the thinking to run a series of tests, stop, inspect, and then, go back and run a series of tests?

MS. HAND: No. We're going to do all of the testing, because this is one aspect that's different for the NuScale design. This will be performed during the initial startup testing after core load.

CO-CHAIR CORRADINI: Oh, I'm sorry. Excuse me. So, I thinking -- this is not TF-3?

MS. HAND: This is not TF-3, no. This is --

CO-CHAIR CORRADINI: Okay. I'm sorry. I'm sorry.

MS. HAND: Yes.

CO-CHAIR CORRADINI: Okay. Excuse me.

CO-CHAIR SKILLMAN: This was the CVAP. This is the real deal.

MEMBER BALLINGER: But, presumably, you could tinker with the flow restrictors and stuff like that, if you detect an issue?

MS. HAND: Could you explain that a bit more?

MEMBER BALLINGER: I think if I'm thinking --

CO-CHAIR CORRADINI: No, you got it right.

MEMBER BALLINGER: I got it right?

CO-CHAIR CORRADINI: Uh-hum.

MEMBER BALLINGER: There's individual, well --

CO-CHAIR CORRADINI: Tube by tube.

MEMBER BALLINGER: Tube by tube almost flow restrictors or maybe there's a different word for it --

CO-CHAIR CORRADINI: It's a good word.

CO-CHAIR SKILLMAN: Flow restrictor.

MEMBER BALLINGER: -- which could be tinkered with, meaning change the geometry a little bit to alter the vibrational frequency that you're having a problem with.

MS. HAND: So, yes. It sounds like you're referring to the steam generator in that flow restrictor --

CO-CHAIR CORRADINI: That's it.

MEMBER BALLINGER: Yes.

MS. HAND: -- which provides a pressure

drop on the secondary side as you're entering the steam generator tube.

MEMBER BALLINGER: Yes.

CO-CHAIR CORRADINI: Right.

MS. HAND: So, that component is susceptible, per our screening rules, to leakage flow instability, and we have a separate effects dedicated test facility where we'll be validating that before initial startup testing --

MEMBER BALLINGER: But what I think I'm meaning is, if you detect a vibrational problem in the tubes, you could, in principle, modify the flow restrictor to change that vibrational --

MS. HAND: Unless the vibration is coming from the primary flow.

MEMBER BALLINGER: Okay.

MS. HAND: But, yes, if it's a concern with secondary flow, that is a component that could be modified.

Okay. There is one COL item in this area, and that is that, prior to any of these validation tests, we will provide test procedures and submit results, both results of the flow testing and the inspection results testing, in accordance with the Reg Guide. The purpose of this COL item is just to continue to have engagement with the regulator as we move through this last phase of the CVAP program.

If there are no other questions about CVAP, I can move on to the short-term transient analysis. The short-term transient analysis is modeling of the dynamic effects associated with a pressure wave that's generated when you have a breach in a high-energy pressure boundary.

CO-CHAIR CORRADINI: To the vessel or to the containment?

MS. HAND: Both.

CO-CHAIR CORRADINI: Okay.

MS. HAND: Yes, both. Although the one going into the vessel is stronger since it's moving through a subcooled fluid. Whereas, the containment is at vacuum conditions or maybe a low-pressure steam condition.

CO-CHAIR CORRADINI: Okay.

MS. HAND: For this analysis, we use our thermal-hydraulic code, RELAP5, to generate the boundary conditions at the break. So, we take the pressure, the mass flow rate, the density, and we convert that into a fluid acceleration and, also, a thrust force. The fluid acceleration gets applied to acoustic elements in the ANSYS structural model, and the thrust force gets applied to the structural, pretty much like the diameter of the piping in that model.

And then, we use ANSYS to simulate the fluid structure interaction, using a time history analysis, and the outputs are the forces and moments on different structures inside of the NuScale power module. And then, those eventually get used in the component stress analysis, similar to what Dylan was talking about for the seismic results.

CO-CHAIR CORRADINI: Is this to help determine restraints or is this to look at --

MS. HAND: It's just to generate a design basis load that needs to be considered when you --

CO-CHAIR CORRADINI: For restraint, for placing of restraints for the piping, I assume?

MS. HAND: No, not really. Just a load that needs to be evaluated for a component. If the load was unacceptable, we would have to come up with a design solution for it.

CO-CHAIR CORRADINI: So, let me ask you a question. What if the IAV doesn't work and I start discharging ECCS above the prescribed pressure? Is that analyzed?

MS. HAND: Yes.

CO-CHAIR CORRADINI: It is?

MS. HAND: It is.

CO-CHAIR CORRADINI: Okay. So, at the full pressure, you actually look at a discharge of an ECCS valve, both RVV and RRV?

MS. HAND: That's correct.

CO-CHAIR CORRADINI: Okay.

MS. HAND: And that, actually, hadn't -- well, it has two notes on that. One, the IAV didn't exist at the time that that analysis was first performed, and then, it was conservative to carry that forward in the future.

And the other note, too, is these loads are not very limiting compared to, you know, when you're talking about your 14-inch, double-ended guillotine break in a traditional PWR.

So, the first point of this slide, you know, we're analyzing primary coolant lines that are only NPS 2 and we have no double-ended guillotine breaks. So, when the break happens, the wave is traveling into the containment and into the vessel. It can't go into the vessel two ways. Like if you have a break in a traditional PWR, the valves, specifically the vent valve, represents the largest breach --

CO-CHAIR CORRADINI: Can you remind me? I apologize. Excuse me. But "NPS" is a designation for a pipe size?

MS. HAND: Nominal pipe size, correct.

CO-CHAIR CORRADINI: Okay.

MS. HAND: Yes. So, about a 2-inch pipe.

We also operate at lower pressures than traditional PWRs. We have less sub-cooling, which is important for the change in enthalpy that happens and the power that you see in the rarefication wave.

And then, the CNV is a single sub-compartment. So, we don't have to do multiple sub-compartment analyses for the different rooms in containment and loads that walls are seeing, since we don't have any walls.

Legacy codes that you might be familiar with are how applicants have analyzed this problem in the past, but with the more state-of-the-art technology now, we just used our thermal-hydraulic code and ANSYS. So, codes that we were using in the company for other purposes. It allows us to simulate this phenomena a lot more accurately.

CO-CHAIR CORRADINI: Let me just make sure I understand what you just said.

MS. HAND: Yes.

CO-CHAIR CORRADINI: So, with NRELAP, what are you using as the critical flow model that you say is accurate?

MS. HAND: We investigated sensitivities with both Henry-Fauske and Ransom-Trapp. And Henry-Fauske showed a slightly better comparison to literature.

CO-CHAIR CORRADINI: Good.

MS. HAND: So, we went with that.

CO-CHAIR CORRADINI: Okay. I'm happy.

MS. HAND: Okay. So, to move on to the benchmarking and this analysis a bit, we used three different open-source literature test cases to show that our predictions are accurate using this method. The main one we used was the Heissdampf reactor, which was a test facility in Germany in the 1980s. And we were able to obtain a lot of pressure and displacement data.

What we did is we used that to kind of determine what parameters we needed to set at what levels to match the experimental data the best way we could. So, what is shown in this image on the upper right is the Heissdampf reactor is experiencing a break in the hot leg and this break is very, very nearly located to the core barrel. And there is a sensor on the core barrel that measures the differential pressure as a function of time. And again, for these events, we are just looking at a few milliseconds. That's when the important phenomena happens in terms of the forces and moments.

So, in this sensitivity case, we looked at different discharge coefficients for our choking model, and really for all of them, they just slightly over-predicted the differential pressure that would be seen across the core barrel, which is appropriate for performing this type of dynamic analysis.

So, these parameters that we, you could call it, tuned in terms of our benchmarking problems, were the optimal settings that were used in the NuScale power breach locations. And then, we analyzed all of our breach locations and took the maximum forces and moments determined over the whole spectrum of breaks that we looked at, and we put them in a loading spec. And then, our component stress analysts used them for the ASME code stress evaluations.

CO-CHAIR CORRADINI: So, I don't want to take too much time, but I want to make sure I understand. This is not the steady-state blowdown? You're looking at the initial transient response from a break?

MS. HAND: Correct.

CO-CHAIR CORRADINI: And all your breaks are either near-saturated conditions or steam conditions, is that correct?

MS. HAND: We also had some subcooled breaks, for example, when the reactor recirculation valve opens, because that sees cold leg temperature.

CO-CHAIR CORRADINI: Okay. So, did you look at the Marviken experiments?

MS. HAND: We did.

CO-CHAIR CORRADINI: You did?

MS. HAND: We did, yes.

CO-CHAIR CORRADINI: Okay. Okay, but these look more limiting in terms of the geometry? That's what I --

MS. HAND: So, the Marviken we used just for showing that our boundary conditions were accurate. We did not find helpful force or displacement information as far as impact.

CO-CHAIR CORRADINI: They just didn't have the data?

MS. HAND: Yes.

CO-CHAIR CORRADINI: I see. Okay. Okay.

MS. HAND: The Heissdampf reactor was kind of the whole deal. It validated our thermal-hydraulics and our structural. Marviken we used mostly for thermal-hydraulics and the Bettis hydraulic pressure pulse was mostly for structural.

CO-CHAIR CORRADINI: Okay.

MS. HAND: So, we kind of looked at a spectrum of things to make sure we were covered for our operating conditions that we were going to be analyzing and the forces and moments that we would be seeing.

CO-CHAIR CORRADINI: Thank you. Thank you.

MS. HAND: And that's it, and there are no COL items in that area.

CO-CHAIR SKILLMAN: How are those results -- your next-to-the-last bullet, "Bounding breach locations analyzed, maximum forces determined," how is that captured so that your documentation demonstrates NuScale components thoroughly fit for duty? How is that captured?

MS. HAND: Ultimately, it will be captured in the ASME designer port for each component.

CO-CHAIR SKILLMAN: Okay.

MS. HAND: And then, in the interim, since we've only really gotten to the final point of generating the loads, those are in the loading spec which is referenced in all of the design specifications. So, those tables that provide all of the loads you need to consider direct you to go to the loading spec to get this load for a certain component.

CO-CHAIR SKILLMAN: Thank you. Okay. Thank you very much.

Colleagues, do you, do any of you, have additional questions for our NuScale colleagues?

(No audible response.)

NuScale, thank you very much.

MS. HAND: Thank you.

CO-CHAIR SKILLMAN: You are able to leave.

A question of protocol to the NRC staff: would you like to begin now or begin at 12:30? Up to you, whichever you choose. We were going to be on break until -- we were going to drive until 12:30, and then, lunch from 12:30 to 1:15. My question is, NRC, would you like to begin now or take an hour for lunch now and come back at 1300? Either, it's up to you, whichever you choose.

Lunch? You good?

We are in recess for 60 minutes. We will come back at 5 minutes at 1:00 on that clock.

(Laughter.)

Thank you. NuScale, thank you very much.

(Whereupon, the above-entitled matter went off the record at 12:06 p.m. and resumed at 1:02 p.m.)

MS. VERA: Good afternoon. Oh, sorry.

CO-CHAIR SKILLMAN: Please, go ahead.

MS. VERA: Okay. Good afternoon everyone. I'm Marieliz Vera, the Operating Manager for the Chapter 3 of the NuScale DC Application with you.

Today we're going to present Section 3.9.2, Dynamic Testing and Analysis of System Structures and Components. The rest of Chapter 3 will be presented on June 8 -- in the June 18 subcommittee meeting.

For the agenda, we're going to -- I'm going to introduce the staff. We'll have an overview. And then we're going to talk about the NuScale Power Module Level D analysis and the Reactor Internal Comprehensive Vibration Assessment Program, or CVAP.

So, the review team is Dr. David Ma, Yuken Wong, and Dr. Steve Hambric, Marieliz, myself, and the lead project manager is Greg Cranston. Now I'm going to turn it over to Yuken Wong so he can say.

MR. WONG: My name is Yuken Wong. I'm in the Mechanical Engineering Branch. I'm sorry. My name is Yuken Wong in the Mechanical Engineering Branch in the Division of Engineering Safety Systems and Risk Assessment in NRO.

I'm going to discuss the review of the NuScale DC Application 3.9.2. The review is conducted according to the standard rebuild plan, and the relevant -- and the relevant Reg Guide.

Reg Guide 120 is the reactor internals' comprehensive vibration assessment program. Reg Guide 1.61 is the damping values for seismic design. Reg Guide 1.122 is the Development of Floor Response Spectra.

I'm going to present a review of the following two areas in Section 3.9.2. The first one is a dynamic system analysis of the reactor internals under Level D conditions.

Level D is the folder condition involving the simultaneous safe earth down -- safe shut down earthquake and pipe breaking events.

And the other is the reactor internals' Comprehensive Vibration Assessment Program, or CVAP.

CO-CHAIR SKILLMAN: Dr. Wong, before you proceed, may I please ask our colleague from NuScale for a follow up? Would you permit me to do that, please?

MR. WONG: Yes. Thanks.

CO-CHAIR SKILLMAN: Dylan, please proceed.

Identify your name and your issue.

MR. ADDISON: This is Dylan Addison from NuScale. I need to issue a correction to a question you asked, Gordon, about whether the fluid elements in the ANSYS model are designated as compressible or incompressible. They are compressible.

That's one of the underlying assumptions of the element formulation.

CO-CHAIR SKILLMAN: Thank you, sir. Dr. Wong, please proceed. Excuse me. Thank you.

MR. WONG: All right. Thanks. The NuScale power module dynamic analysis under Level D conditions, the DCA application 3.9.2 and Appendix A contain a summary of the dynamic analysis and the NuScale power module seismic analysis and technical report.

It contains the detailed analysis methods, including motion, structure, or modeling of major NPM components and analysis results. Including displacement, in structural response spectra, seismic forces, and movements and component interfaces.

The staff focused a review on the reactor vessel internals and steam generator components. The review includes structural modeling, analysis methods, low combination, input motion, fluid modeling, fluid distribution, damping value, depth conditions, stress classification and linearization, ASME Section Three stress acceptance criteria, and stress phase out.

The short term transient report provides the pipe break structural loads for reactor vessel internal stress analysis. And the Section 3.9.2 reveal is focused on the ANSYS modeling.

The -- NuScale's 3D ANSYS NPM model consists of five --

MEMBER REMPE: Just a second.

CO-CHAIR SKILLMAN: Excuse me. For those who are on the phone line, would you please mute your phone? Please proceed.

MR. WONG: Okay. The NuScale 3D ANSYS NPM model consists of five set sub-models. The containment of the model with the reactor pool, the reactor pressure vessel, lower reactor vessel internals, upper reactor vessel internals, and the control rod drive mechanism.

These sub-models are connected by coupling nodes, constrained equations, contact elements, and fluid coupling. The connection methods are revealed along with other in fluent aspects such as the adequate number of elements.

The 3D ANSYS NPM model is converted to the equivalent beam ANSYS model, and then the equivalent beam SAP2000 model. The SAP2000 model is used for the reactor building SASSI analysis.

The dynamic responses of the models are matched through tuning the mass distribution and stiffness. The staff reviewed this tuning process.

The 3D ANSYS NPM model is modified to include the entire volume of the reactor pool. The NPM reactor pool model contains the Bay 1 NPM module or the Bay 6 NPM module.

Nonlinear time history analysis is performed on the NPM entire pool model by applying the acceleration time history from the SASSI analysis as boundary conditions at the pool floor, walls, and NPM supports.

CO-CHAIR SKILLMAN: Why were Bay 1 and Bay 6 chosen?

MR. WONG: The analysis show the location in Bay 1 and Bay 6 provides the bounding values --

CO-CHAIR SKILLMAN: Thank you.

MR. WONG: For all other Bays.

CO-CHAIR SKILLMAN: Okay. Thank you.

MR. WONG: Next slide. I'm going to discuss in the following slides, some of the result issues, as well as the four remaining open items.

Next slide, please.

In the original analysis, NuScale assumed that 7 percent damping for safe shutdown earthquake in the NPM system and component seismic analysis. This 7 percent damping is higher than the damping specified in Reg Guide 1.61, Table Six, for pressure vessels and major pressure boundary components. Which is specified as 3 percent.

Reg Guide Table One specifies 4 percent for welded steel or bolted steel with friction connections. Which means no sliding. And specifies 7 percent for bolted steel with bearing connections, which allows sliding.

NuScale's justification for using 7 percent damping are the reactor vessel internal joints allow sliding. Which are similar to the bolted scale structures with bearing connections in Reg Guide 1.61. The recommended values is 7 percent.

There are many sliding steam generator to support interfaces which generate large friction of the space at forces. Next slide, please.

The staff noted that there are many valve structures in the reactor vessel internals. Such as the weld between the lower core plate and the core barrel. The Reg Guide specifies 4 percent.

The steel to steel coefficient of friction is smaller underwater then in dry condition. Because the fluid -- the water film acts as a lubricant. So, the 7 percent damping will not be achieved in the steam generators.

In response to the NRC concern, NuScale will use 4 percent instead of 7 percent for the NPM seismic analysis. The staff finds that 4 percent damping acceptable, because the integrated NPM with many connections and internal structures is unlike traditional shell type pressure vessels.

And there is additional energy dissipation provided by the connections and internal structures. This issue has been resolved and closed.

CO-CHAIR CORRADINI: Sorry. So, the first open item has been resolved?

MR. WONG: This --

CO-CHAIR CORRADINI: I'm sorry, excuse me. So the first open item that you were discussing with us is resolved?

MR. WONG: In the Phase Two of the SE, this is not an open item. It's just that we --

CO-CHAIR CORRADINI: Oh.

MR. WONG: Yeah. We want to bring it up because this is one of the issues that we raised and resolved.

CO-CHAIR CORRADINI: Okay. Okay. Thank you.

MR. WONG: Okay. In the figure it shows the core barrel. And inside the core barrel we factor blocks. And there's a thin fluid gap between that.

In the original analysis, this fluid gap was not considered. So the frequency of the core barrel and the reflectors are lower due to the added mass effects of the fluid gap.

So in response to the NRC concern, NuScale did a sensitivity study. And found the loads at the upper and lower core barrel are higher if the gap is considered, if the fluid gap is considered.

So, they revised the analysis to include the modeling for the fluid gap using ANSYS four year nodes. And the staff finds the approach acceptable, because the dynamic response of two concentric cylinders with a fluid gap in between can be simulated by four year nodes. This issue is resolved.

The NuScale NPM modules are submerged in a pool of water. Analyzing the NPM in a pool of water for seismic responses are new and unique to NuScale.

Initially NuScale assumed the reflection of acoustic waste energy 100 percent -- excuse me, 100 percent reflection of the acoustic wave energy at the bottom of the pool.

So in a seismic analysis when applying frequency shift, the acoustic mode of the pool aligns with the structural mode of the NPM module. And this led to unreasonable and excessive amplification of the structure response.

So, in reality, the acoustic wave energy is constantly reflected and passively absorbed by the concrete floor and the surrounding soil. Next slide, please.

CO-CHAIR SKILLMAN: No. Not so fast. Back up one. What is the transmitter for the acoustical energy?

CO-CHAIR CORRADINI: What creates the initial pulse? The assumption?

MR. WONG: The seismic event.

CO-CHAIR CORRADINI: Okay.

CO-CHAIR SKILLMAN: Oh, okay. Very good. Thank you.

MR. WONG: So, in order to estimate the acoustic absorption coefficient at the bottom of the pool, NuScale created two interesting models.

The first model is the integrated model including the NPM reactor pool water, reactor building, and the backfill soil. And a damping was applied to the concrete and backfill soil to dissipate the pool acoustic energy.

And Model two is the standard NPM model within NPM and the reactor pool water. Various acoustic absorption coefficients are applied at the bottom of the reactor pool to dissipate the pool energy, the pool via the acoustic energy.

One G vertical acceleration was applied at the bottom of the pool of the two models. And then compared the responses at the key NPM locations between the two models.

And absorption coefficient of .75 produced the best match between the two models. To verify the acoustic absorption analysis and the -- the NRC staff with the assistance from Dr. Ma and Dr. Hambric performed an audit in December 2018. Next slide, please.

During the audit NuScale modified the absorption coefficient from .75 percent to .4 percent. The staff concluded the absorption coefficient of .4 percent reasonable, because the damping in the building structure and the surrounding soil dissipates acoustic energy.

The staff also compared the impedance

ratio between concrete and water based on the absorption coefficient of .4. And the impedance rate, the actual impedance ratio between concrete and water based on the density and speed of sound.

The impedance ratio, based on the absorption coefficient of .4 is higher. Higher impedance ratio leads to more acoustic wave reflection and less obstruction, and is more conservative.

Therefore, this issue is resolved and closed. Next slide, please.

In Division Zero of the seismic --

CO-CHAIR CORRADINI: Can I ask a question? I'm sorry that --

MR. WONG: Yes.

CO-CHAIR CORRADINI: I was reading it, and I didn't think fast enough. But your initial source is within the ground, right?

Or are we thinking of it as horizontal shaking and then a reflection? I'm still trying to figure out where the source is relative to the reflection and transmission.

But you're looking at this as essentially at the base of the reactor pool, right? The absorption coefficient?

MR. WONG: Yes.

CO-CHAIR CORRADINI: Okay.

MR. WONG: It's at the base of the pool, yes. Between the --

CO-CHAIR CORRADINI: But the seismic wave is coming up from below? Is that -- am I understanding this correctly?

MR. WONG: We're looking for the -- at the bouncing of the --

CO-CHAIR CORRADINI: Right.

MR. WONG: Of the acoustic energy. Which is --

CO-CHAIR CORRADINI: Yeah, how much passes through the interface and how much essentially is reflected, right?

MR. WONG: Yes.

CO-CHAIR CORRADINI: Right, okay.

DR. HAMBRIC: Remember, they're using a two-part modeling approach. They've got the big model that generates the loads that are being applied to the reactor.

Then you go to the small model with just the water in the pool, and the approximate absorption coefficient, and shake the reactor around.

CO-CHAIR CORRADINI: Oh. That's --

DR. HAMBRIC: And they needed one of those to respond appropriately.

CO-CHAIR CORRADINI: So now we're shaking the reactor in the water. And this is what it's applied to.

DR. HAMBRIC: Yeah. And if you don't put that absorption coefficient on it, you get this crazy amplification.

CO-CHAIR CORRADINI: Yeah. Sure.

DR. HAMBRIC: Which is, you know, close too real.

CO-CHAIR CORRADINI: And .4 is close enough?

DR. HAMBRIC: It's --

CO-CHAIR CORRADINI: I mean, I would just take the ratio of Row C over Row C and use that. But that's two best estimates?

I -- you see my question?

DR. HAMBRIC: They're aware of the actual impedance ratio. They have chosen to be conservative and use .4.

CO-CHAIR CORRADINI: Okay. Fine.

CO-CHAIR SKILLMAN: Which is acceptable. And .4 is more conserv -- is conservative compared to .75?

DR. HAMBRIC: Um-hum.

CO-CHAIR SKILLMAN: Okay.

CO-CHAIR CORRADINI: Okay. Go ahead. Thank you. Sorry.

MR. WONG: Okay. In Revision Zero of the seismic chemical report, it states that the in-structure response spectra at the location of the equipment supports within the NPM envelope and point according to ASCE 4-13 for component design.

ASCE 4-13 is the standard for seismic analysis of safety related nuclear structures. ASCE 4-13 permits a 15 percent reduction of the narrow frequency peak amplitude of the in-structure response spectra if certain conditions are met.

The staff finds the use of ASCE 4-13 in the generation of ISRS not consistent with Reg Guide 1.122. The Reg Guide does not permit the reduction of frequency peak amplitude.

In response to the NRC concern, NuScale stated that frequency amplitude reduction was not performed in the generation of ISRS. And removed ASCE 4-13 from the description of the generation ISRS and made the revision to the scientific report. So, this issue is resolved and closed.

I'm going to discuss the open items in the Phase 2 SE. In the original analysis, NuScale considered six cases for the seismic analysis. Some of the cases are in nominal or 77 percent nominal NPM stiffness.

The staff concern is that it 130 percent of nominal NPM stiffness should be considered to account for the uncertainty in the NPM model input and assumptions. The plus/minus 30 percent stiffness is equivalent to plus/minus 15 percent of frequencies.

So, it's a standard practice to perform frequency shift in the seismic analysis. So, if not considering the 130 percent of the NPM stiffness, we will -- it's possible to have a conservative seismic analysis results.

So, in response, NuScale performed 12 seismic analysis runs, including the 130 percent nominal NPM stiffness. And they have included the result in Revision Two of the Seismic Report.

And it's currently under review. Next slide, please.

CO-CHAIR CORRADINI: So, can you go back? So, I'm not familiar. So, the 77 percent and the 130 percent are equivalent?

MR. WONG: Equivalent like plus/minus.

CO-CHAIR CORRADINI: That's what I guessed. Okay.

MR. WONG: Yeah. Um-hum.

CO-CHAIR CORRADINI: All right. Thank you. That's what I didn't catch. Thank you very much.

DR. SCHULTZ: And then the -- what is the resolution here? They've performed --

CO-CHAIR CORRADINI: They're in the middle of that.

DR. SCHULTZ: The evaluation.

MR. WONG: They performed additional cases.

DR. SCHULTZ: Yes.

MR. WONG: Including the 130 percent NPM stiffness, that is equivalent to plus 30 percent, you know, NPM stiffness.

DR. SCHULTZ: And now they've included a broader evaluation, including the 130 percent stiffness.

MR. WONG: Yeah. They updated seismic analysis. And instead of having six cases originally, now there are 12 cases to consider. Or to finding the boundary values.

DR. SCHULTZ: Okay.

CO-CHAIR CORRADINI: To find -- thank you, that will help me. To find the boundary line for?

MR. WONG: The component response for the stress analysis.

CO-CHAIR CORRADINI: Okay. Thank you.

DR. SCHULTZ: And that's currently under review? That --

MR. WONG: Correct.

DR. SCHULTZ: By the staff. Thank you.

DR. WONG: Okay. Here's the picture of the reflector blocks on the upper right. The reflector blocks are sitting on the lower core plate.

And they are restrained in the horizontal direction, with a long piece. But they are not restrained vertically.

And these, as you can see on the lower right, the lower core plate in-structural response spectra shows at the high frequency peak the acceleration exceeds the gravity acceleration.

So, the concern is that during a seismic event, the reflector blocks may uplift. And this uplift was not considered in the original seismic analysis.

So, in response to the NRC concern, NuScale updated the seismic model, the seismic modeling to include contact elements between the lower core plate and the reflector blocks, to simulate uplift.

And the analysis shows that the uplift distance of the reflector blocks is small. And it will not close the gap between the reflector blocks and the upper core plate.

The staff revealed the analysis modeling results, and found the response acceptable.

CO-CHAIR SKILLMAN: What are the contact elements that you refer to here?

MR. WONG: The contact element is the ANSYS modeling to -- between the reflector block and the lower core plate.

CO-CHAIR SKILLMAN: So it is an analytical model?

MR. WONG: Yes. It's a modeling technique that can capture the position uplift of the reflector blocks.

CO-CHAIR CORRADINI: So they model the interface by using a contact finite element model. Is that -- am I understanding this correctly?

MR. WONG: Yes. Using the contact elements between the two --

CO-CHAIR CORRADINI: You've got two blocks. And then you have a contact element between them.

MR. WONG: Right.

CO-CHAIR CORRADINI: But it's -- to get to Member Skillman's question, it's not real. It's a modeling -- it's a modeling approach to try to model the contact between the two blocks?

MR. WONG: Correct.

CO-CHAIR CORRADINI: Okay. So is the open item closed?

MR. WONG: Yes. So this is a Phase 2 SE open item. And it has sense closed.

CO-CHAIR CORRADINI: Okay. Thank you.

DR. SCHULTZ: And so from what you show acceleration (off mic)

CO-CHAIR SKILLMAN: Mike Steve.

DR. SCHULTZ: Oh, excuse me. You're showing acceleration. You did mention that the uplift was small.

What's small?

MR. WONG: Another concern is that the reflector blocks will uplift and impact the upper core plate. So, the uplift distance is small.

So, it will not impact the upper core plate.

CO-CHAIR CORRADINI: I think he's asking for how small. A millimeter? A centimeter? I think that's what you're asking.

DR. SCHULTZ: Yes. Because --

MR. WONG: Then NuScale can speak to it. And I'm not sure. They can discuss the value.

MR. ADDISON: This is Dylan Addison from NuScale. The uplift of the reflective blocks from the lower core plate is on the order of a sixteenth of an inch more or less.

DR. SCHULTZ: Thank you.

CO-CHAIR SKILLMAN: Does that sixteenth of an inch, even though it's small, impose a momentum load when the gap closes?

MR. ADDISON: So, when -- Dylan Addison again. When the reflector block impacts with the core plate, it does change the loads on the lower core plate.

And those transmit into the fuel. And we are analyzing the fuel for those loads.

CO-CHAIR SKILLMAN: Is that load significant or not significant?

MR. ADDISON: Well, we think that we have a clear path forward for demonstrating the fuel integrity with those loads. So, it's different then it would be modeled linearly.

CO-CHAIR SKILLMAN: So it sounds like it's not yet resolved that you believe you have a success path ahead.

MR. ADDISON: And that will be addressed in Chapter 4.

CO-CHAIR SKILLMAN: Thank you. All right. Thank you.

CO-CHAIR CORRADINI: Hold on. Addressed in Chapter 4 by a subsequent open item because of the fuel -- because the fuels -- the fuel response to a seismic event, is that where we're going?

MR. ADDISON: That's correct.

CO-CHAIR CORRADINI: Got it. Okay. Thank you very much.

MR. WONG: Okay. In this figure it shows in normal operation the NPM modules are in the operating bay. And during refueling the NPM will be in the flange tools for disassembly.

NuScale introduced a nonlinear contact elements and the interface between the lower reactor pressure vessel and the refueling flange tube to simulate the uplift of the NPM during a seismic event.

And the results are documented in Revision Two of the Seismic Report. And the Seismic Report is currently under review. Next slide, please.

The staff issued an RAI requesting NuScale to provide a seismic analysis details and Level D stress results for the major NPM components and a steam generator major NPM components.

NuScale provided the stress evaluation results in their REI response. However, the results are based on the original six cases without a 130 percent NPM stiffness.

NuScale will provide the new results using the 12 cases, including the 130 percent NPM stiffness case in a supplemental response. And the staff will review this supplement and the results.

CO-CHAIR CORRADINI: But this one, if I understand what you're saying, this is connected to the original one for seismic analysis, where they're going from six to 12 cases.

It's the same 12 simulations. And you're using the results to answer these questions. Have I understood this correctly?

MR. WONG: The 12 cases generate a new instructive response factor.

CO-CHAIR CORRADINI: Right. Okay. Got it.

MR. WONG: It's just --

CO-CHAIR CORRADINI: I'm with you. Thank you very much.

MR. WONG: Okay. Then the next topic is the reactor internals' comprehensive vibration assessment program. Next slide, please.

The staff basically reviewed primarily on two technical reports. One is the comprehensive vibration assessment technical report. And the second one is the measurement and inspection plan technical report.

The staff also conducted three audits to review the un-docketed information. The staff attended face to face meetings with NuScale and conducted periodic conference calls with NuScale to discuss the issues.

And now I'm going to turn over to Dr. Hambric to discuss the CVAP review.

DR. HAMBRIC: Okay. Thanks Yuken. This is Steve Hambric from Penn State University. And I've broken this next section up into a couple of parts.

The first one is a high level overview with a bottom line up front assessment of the risk areas as we see them. And they're the same as what NuScale showed you.

But I'll go into following that in more detail on each of those components. What areas in particular are a concern about, and how we hope to move forward and resolve them.

So here are the big FIV mechanisms that were evaluated. The first one turns out quite benign, turbulent buffeting.

That's not benign in traditional reactors because the flow is a lot faster. As NuScale has pointed out to you, their natural circulation reactor, the flow is very low.

And turbulent buffeting load scale on an exponent of flow speed. So, it's really not a major concern for them.

The other ones are. Things like vortex shedding and lock in of that vortex shedding with the structural residence is of great concern to us. There's a couple of structures we're looking at there.

Fluid elastic instability is even worse the vortex shedding. And in particular we're looking at one structure there.

Acoustic residence, those are the types of loads that have been causing us trouble with BWRs in the fleet now. Fluids over standpipes generating big high amplitude pressure waves that in the case of BWRs were damaging steam dryers with fatigue failure.

And the next two are leakage flow instability, and then flutter and galloping.

CO-CHAIR CORRADINI: Which are or aren't of concern?

DR. HAMBRIC: Leakage flow instability is. So, there's a couple of components we'll talk about there.

Flutter and galloping, we were able to retire that risk. There should be no issue in the NuScale plans for that.

As NuScale pointed out, they did screen all their components. And we did check. There was a thorough screening.

And the ones in the box are the ones that have less than 100 percent margin. And we'll focus the rest of our time on here today.

The helical coil steam generator is not typically reviewed in this chapter. But since it's integral to the reactor here, it is. And so we'll focus here on that quite a bit.

The steam generator inlet flow restrictors, as we'll see in a little while, are being looked at for leakage flow instability. Make sure they're not subject to that.

Control rod drive shaft and in-core instrument guide tubes, I'll show you some pictures of that in a moment. But they're in cross flow. So we're looking at them for that.

And then really any piping at all that has a side branch, we have been looking at along with NuScale to look for the possibility of acoustic residence.

You may have seen in the SE that that box extended down a bit to include the CRAGT. That is no longer a concern of ours. And so we won't talk about it today. But, details of that assessment are in the Phase Two SE.

So, risk areas. You've seen that diagram on the right before. It is a natural circulation plant. So the flow heads upward through the center.

And then when it reaches the top, it makes a U-turn and heads back down past all the steam generator tubes. Flow is nice and slow. And depending on where you look, the flow rates are say, five to 25 times lower then you might see in a typical PWR.

And if you think back to previous design applications, a lot of the reviews that were done, a lot of effort, a significant effort was spent on benchmark and turbulent loading, but because the flows are so fast. And they had to do that because those loads were important and could lead to high alternating stresses and potentially fatigue.

And that's just not the case here. So you're not going to see much of that at all in our review. So, that's a good thing.

However, low flow or not, you still have to evaluate flow over other components to look for the things we talked about earlier, vortex shedding, fluid elastic instability, and leakage flow, things like that.

Particular in the NuScale plant, you have some very long thin rods and tubes. The ICIGT and CRDS in particular. That's non-typical.

And --

CO-CHAIR CORRADINI: What's the S --

DR. HAMBRIC: Hum?

CO-CHAIR CORRADINI: You're talk -- oh. Helium coil steam generator. Okay. Fine.

DR. HAMBRIC: Yeah. Right. And an extra high --

CO-CHAIR CORRADINI: And what's the ICIGT?

DR. HAMBRIC: Inlet core instrument guide tube. Control rod drive shaft. And I should have mentioned, there's a list of all the acronyms towards the back if you want a cheat sheet.

CO-CHAIR CORRADINI: But one more time,

ICIGT?

DR. HAMBRIC: In core instrument guide tube.

CO-CHAIR CORRADINI: Ah. From a --

DR. HAMBRIC: You'll see a picture of it in a minute.

CO-CHAIR CORRADINI: Okay. Okay.

DR. HAMBRIC: Those are long thin structures, and it doesn't take a lot of cross flow to potentially lock into them.

Also, it's really impossible to come up with a piping system that has no side branches. And in this case the decay heat removal system piping does have some.

And they are potentially subject to acoustic resonance. But NuScale's done everything they can do to mitigate their amplitude.

But nevertheless, it is a risk area that we'll talk about. Most regions where leakage flow instability could occur generally are in low pressure difference regions.

However, there are a couple of locations we're paying more attention to, the steam generator and the flow restrictors, and maybe a few other locations that we'll talk about later. So, those are the big ones that you'll be hearing about through the rest of the brief.

Now NuScale screening studies used well-established methodologies. A lot of them come right out of the ASME Boiler and Pressure Vessel Code, Appendix N, which spends a lot of material on flow induced vibration of tubes.

But also two other well recognized names in our area, Blevins Flow and Use Vibration, Second Edition is really a primer on this area. And then M.K. Au-Yang has a nice textbook.

It's really a cookbook set of formulas you can use to assess flow and use vibration of power and process line components. So those are all good references to use.

But they do require calculations for running those methodologies. Particularly you need flow velocities, you need structural modes, mode shapes, frequencies, mobile masses, and you need damping.

So, for two of those, NuScale has used calculation methodologies for the flow speeds. They're using CFD methods.

But it is for the simplified model. And by simplified I mean that the core that dumps heat into the system and the steam generator which takes heat out, are highly simplified. They're just sort of modeled as bulk regions.

So even they don't get local flow velocities out of those simulations. So, you get a mean flow velocity and then you have to sort of make some assumptions to get to the peak velocities after that.

The FE models are generally of individual components, structural models. And simplified boundary conditions are assumed.

And these are intended to give you the lowest resonance frequencies. So when we walk through some of these mechanisms that potentially lock into modes, you're really looking for the lowest resonance frequency and the highest excitation frequency. So those are what you're after.

Damping, that's very hard to calculate. And in Reg Guide 1.20 we allow 1 percent without any proof. And in general, NuScale is at or below 1 percent except in one case, where they're assuming one and a half percent.

And the Reg Guide calls for pretty rigorous substantiation of that, because if you increase damping, you're reducing margin -- or you're increasing your margin. You're reducing the vibration of your structure. So we'll talk about that one instance more coming up.

Margins of safety, even with NuScale's conservatisms, are low by FIV standards. Being within 10 percent of a lock in is not a great place to be.

Here they are for the steam generator. They got 20 percent against vortex shedding and lock in. Ten percent against fluid elastic instability.

CO-CHAIR CORRADINI: Would their test program help in removing what is apparent closeness to a margin?

DR. HAMBRIC: Absolutely. Yeah, we'll talk about that.

CO-CHAIR CORRADINI: Okay.

DR. HAMBRIC: And reactor vessel internals, the only two of concern are the control rod drive shaft and in core instrumentation guide tubes. Both coincidentally with about 25 percent margin against vortex shedding.

The side branches and the decay heat removal system piping, you have about a 20 percent margin against what I'm going to call primary acoustic resonance. But no margin against secondary. And hold that thought, and we'll get into the details of what primary and secondary mean in a few slides here.

The steam generated in the flow restrictor has low risk. And that's based on some testing that NuScale did of several design concepts.

But, the final design so it's unquantified risk. And there's testing pending to hopefully retire that risk.

Now, during our review, we did uncover some non-conservatisms that may outweigh the conservatisms that NuScale cited.

And all of those are currently being addressed by NuScale via the RAIs. And you'll find them in the open items. And I should have added, there is a list of all the open items at the end of all this that you can refer to.

And as I go through the individual components in detail, I've got the open item on the cover sheet. So you can see which goes with which.

Now, the non-conservatisms really span most of these components. I didn't link them to a specific component.

Instead we just talked about the non-conservatisms in general. And so here they are broken down into a flow modeling, structural modeling, and then one extra concern at the end there.

So the flow modeling, it is using a course simplified COD model. And there's a couple of things to worry about there.

Number one, even though turbulent buffeting, we don't think is a big concern, the empirical forcing functioning models that they employed, call for peak free stream velocities to be used.

For example, if you had annular flow in the flow rising up through the core. It's a big pipe flow essentially. And the boundary layers would meet in the center of the pipe.

And to estimate wall forcing functions in that sort of flow, you would go to the center peak velocity, plug that into your empirical model along with some other parameters, and estimate a forcing function.

NuScale took their CFD solution, averaged it over that region, and that lowered the velocity. Therefore, lowering the forcing functions.

So we've been having kind of long debates about that. And we're waiting for an RAI response from that to resolve it.

CO-CHAIR CORRADINI: So this is -- this is the way you describe it is taking the peak velocity is a standard approach?

DR. HAMBRIC: That is the approach that the empirical models assume you have done. It's the edge of your boundary layer, the edge of your annual flow.

The other one that's a little more of concern, because again, turbulent buffeting, we don't think there's going to be a problem, is the spatial variability of the flow. Particularly around the steam generator.

Modeling that is just a bulk region in getting one velocity. We know from past experience, and I'll show an example of that, that the flow can be quite inhomogeneous through steam generators. It goes where it wants to go.

And you're going to have regions of high speed flow and regions of low speed flow. And that has not been accounted for yet.

In the structural modeling, the meshes we have seen to date are pretty course. And course find it on the structural meshes biases resonance frequency is high.

And as we'll see momentarily, what we're after is the lowest resonance frequency and the highest flow excitation frequency. That's a conservative comparison. And so we've asked them a question about that for analyzing each of their components.

There are also some boundary conditions that gave us pause. I'll show you examples shortly. But, some of the structures go through supports. But the supports aren't really supports.

And they've assumed idealized somewhat pinned boundary conditions. That gives you shorter structures. That gives you higher resonance frequencies. Again, non-conservative.

And also we've had to verify that the fluid loading, that all of these components are sitting in water. Then we have to make sure that the added mass is conservative and drives the frequency as low as is possible.

Now the final concern is that if these margins hold up, if you're looking at 10 to 20 percent margin, and by margin we mean, how close is the resonance frequency of the structure to the excitation frequency of the flow.

If they're within 10 percent, you really need to do a force response calculation. So for all NuScale has done, is just compare the frequencies and said, if they don't match, we've got margin.

CO-CHAIR CORRADINI: So Steve, what do they have to do? I didn't understand.

DR. HAMBRIC: So, so far they're just looking for coincidence of excitation frequency and structural response frequency.

CO-CHAIR CORRADINI: Right.

DR. HAMBRIC: And saying that if they're not coincident, they're good.

But if those frequencies start coalescing within 10 percent say, that structure is going to respond to that vortex shedding. It won't lock into it, but it's certainly going to respond.

And so far there's been no forced response calculation done. If --

CO-CHAIR CORRADINI: So you would have to assume some sort of distribution function to look at how the two things interact. That's your point?

DR. HAMBRIC: You'd have to drive the structure with a vortex shedding source, with whatever your frequency offset is. And examine how much further that is.

CO-CHAIR CORRADINI: And rule of thumb is, it's got to be how far apart before you can ignore that?

DR. HAMBRIC: There's not a great rule of thumb. It depends. But 10 percent is well within that rule of thumb I'd say.

CO-CHAIR CORRADINI: But if it doesn't have --

DR. HAMBRIC: I'd do an analysis if it was within 50 just to be sure.

CO-CHAIR CORRADINI: Oh. Within 50 percent? So then you'd have a --

DR. HAMBRIC: That's just Steve Hambric's rule of thumb. There may be other rules of thumb they can cite in the literature as well.

CO-CHAIR CORRADINI: But your point is -- your point is that you -- the practice that you would do is if you were within a factor of two, you'd want to look at how the two tales interact --

DR. HAMBRIC: A factor of 50 percent.

CO-CHAIR CORRADINI: Oh. It depends on whether it's in the numerator or denominator. But, okay.

DR. HAMBRIC: Right.

CO-CHAIR CORRADINI: I'm with you.

DR. HAMBRIC: But that's me. And again, everybody has their own rule of thumb. But, I'm not aware of one that's in the literature.

MR. WONG: This is Yuken Wong. In other

design certification applications, such as AP1000, they performed the forced vortex shedding response analysis for the reactor internal components.

CO-CHAIR CORRADINI: Say that again, please. I'm sorry. So AP1000, they did what was suggested?

MR. WONG: Correct, correct.

CO-CHAIR CORRADINI: Okay.

DR. HAMBRIC: And I'd like to go on the record as saying that that 50 percent is not a direction to NuScale or anybody else.

CO-CHAIR CORRADINI: No.

DR. HAMBRIC: That's a response to an ACRS --

CO-CHAIR CORRADINI: Just an opinion.

DR. HAMBRIC: -- question and opinion.

CO-CHAIR CORRADINI: Yeah, okay.

DR. HAMBRIC: There has not been a lot of testing done to date. Part of that is because the turbulent buffeting is not a concern for the NuScale plant. So there's less than usual, and the focus is different. In previous plants, there's been a lot of focus on benchmarking, what those forces are, and whether they're being appropriately estimated.

Here, we're looking at slightly different issues. Now, there is a concern. This is a unique new design. There are some interesting innovative new features. And there's no operating history to lean on.

The other thing we've asked them early on is the possibility of some pre-operational FIB testing. But that's just not feasible, because there're no pumps to pump the flow through. It's a natural circulation design. However, I'll say it again. The turbulent buffeting benchmark just really isn't necessary.

So we're taking an alternate approach, but we believe it's a valid alternate approach. And that is looking at some preliminary validation testing which is focusing on the key FIB mechanisms we're talking about here that have low margins of safety and high uncertainty. And I'll go into the details shortly.

And additional startup testing, instead of being a benchmarking test, is really focused on disaster insurance, making sure nothing bad is going on in there. We've seen past best intentions lead to unexpected flow induced vibration at nuclear plants before. We don't want it to happen here.

And so NuScale has committed, verbally

anyway, to instrumenting the inside of the plant to look for unexpected surprises and, if those surprises occur, to be able to localize them so they know what component to go back and examine more closely and hopefully fix. We've not seen any of that yet, that's pending. But that's one of our open items.

CO-CHAIR CORRADINI: So, Steve, could you please -- I'm sorry, go on.

MEMBER SUNSERI: I thought the plant has a module heating system that they use to get the natural circulation flow going initially when the core is cold. And you can't use that to get some flow data?

DR. HAMBRIC: I'll let NuScale address their reasons for why they don't think hot functional testing is reasonable.

MS. HAND: So this is Olivia Hand. The module heat up system is really just to heat the plant up and to get enough flow so we can mix boron. Flow rates that we're talking about here are up against our maximum mechanical design allowable velocities.

So we're, you know, like, operating this regime maybe we could get to about, like, 20 percent of those velocities. But it wouldn't give us any spill vibration information.

CO-CHAIR CORRADINI: And if you did that, extrapolating that extra factor of five is inappropriate? In other words, I seem to remember by some other curve, which I think is in the open, somewhere around 15 or 20 percent you can switch over, go critical, and away you go.

So at 20 percent of that, that's still not enough to give you some indication that what you --- I'm looking more of a benchmark of what you calculate, I assume that's where Matt was going, what you calculate versus what you see.

MS. HAND: So similar to what Steve was saying, you know, having the module instrumented during initial startup testing, we kind of have been referring to it as go/no go testing.

We're not going to be able to collect, you know, vibration information that's going to necessarily validate these safety margins. Because vortex shedding is either going to occur or it's not going to occur. So we're hoping to just measure that's not occurring.

You know, you need to be able to increase velocities to the point that it does occur in order to collect meaningful vibration data which would be per our design analyses above our licensing basis maximum flow rates.

CO-CHAIR CORRADINI: Okay.

MS. HAND: So as you extrapolate down to the lower power conditions, you're going to just move further away from that.

CO-CHAIR CORRADINI: Okay, thank you.

DR. HAMBRIC: And just to preview, the testing they are proposing outside of the plant but in prototypic conditions, we like it because they're going to go well above their design flow speeds. They'll find where these mechanisms occur and show us how much margin they've actually got.

CO-CHAIR CORRADINI: So TF-3 helps?

DR. HAMBRIC: TF-3 helps, and what they're going to do with the steam generator in the flow restrictor also helps.

CO-CHAIR CORRADINI: Okay.

DR. HAMBRIC: Being able to run at super high speeds to tell us what margin they've got will definitely be an improvement over trying to do hot functional testing.

CO-CHAIR CORRADINI: But I wanted to make sure. Does that, at least for the steam --- I'm looking back to your list of all of your areas of worry. This essentially deals with the steam generator tubes, but it doesn't deal with the reactor vessel internals.

DR. HAMBRIC: It will.

CO-CHAIR CORRADINI: How so?

DR. HAMBRIC: Oh, being able to run at a higher speed. Yeah, we couldn't run it at higher speed. But the reactor vessel internals, other than the steam generator, steam generator and the flow restrictor, CRDS, ICIGT, we don't have significant concerns about.

CO-CHAIR CORRADINI: Oh, I see. Okay. All right, so it more actually is the margin from the steam generators?

DR. HAMBRIC: Yeah, that would be determined with outside testing. And really, the initial startup testing is meant for disaster insurance. Nothing else is going on that we might have missed, in spite of our best intentions.

DR. SCHULTZ: Just so it's clear, the staff proposed the following for reasonable assurance. Where, in fact, do things stand in terms of the program that NuScale is going to perform?

DR. HAMBRIC: We'll go through all those details coming up.

CO-CHAIR CORRADINI: That's coming up.

DR. HAMBRIC: Why we think they're valid.

DR. SCHULTZ: Okay. Thank you.

DR. HAMBRIC: Okay, so that's the overview. And now I've got short sections on each of the key areas with a header sheet on each one and the open items associated with it.

So the first one is looking at vortex shedding and lock-in but of two components, the reactor vessel internals, it's actually two in that, and the helical coil steam generator tubing.

So this is vortex shedding. I pinched this off the Internet. So the picture you're looking at is --

CO-CHAIR CORRADINI: It's not your experiment? I'm shocked.

DR. HAMBRIC: It is not my experiment. It is actually a simulation. I used to have the citation for it, but I left it out of here. I probably should have added it. But that's flow over a cylinder. And this is problem that's been studied to death for many, many, decades now.

But what you're looking at is, at a certain flow speed, the vortices that are building up behind it, remember just how they're really kind of moving up and down. And it's probably a little bit harder to see, but they're also pushing back and forth. And so that's a magic flow condition where those vortices are out of phase with each other and drive that cylinder both up and down, and backwards and forwards.

By itself, that's nothing to worry about. If that cylinder starts moving, and I don't have an animation of that, but I'll use an example in a minute, if the cylinder starts moving, either shaking up and down or going back and forth, it's going to reinforce the strength of those vortices, make them even stronger.

That in turn makes the structure vibrate more which in turn strikes as a vortices, and you end up with this feedback. Eventually, there's a limit, but that limit is not someplace you want to be.

CO-CHAIR CORRADINI: That's when the two ---

DR. HAMBRIC: This is when they coincide.

CO-CHAIR CORRADINI: Okay, that's what I thought.

DR. HAMBRIC: If they're a little bit off in frequency, that cylinder's still going to move and vibrate, but not nearly to the point where it will if they coincide.

CO-CHAIR CORRADINI: And if I'm outside of the, I'm trying to remember all your acronyms, if I'm talking about the control rod drives and the in ---

(Simultaneous speaking.)

DR. HAMBRIC: Yeah.

CO-CHAIR CORRADINI: -- this is calculable once I do a screening.

DR. HAMBRIC: Exactly.

CO-CHAIR CORRADINI: That's your point.

DR. HAMBRIC: Yep.

CO-CHAIR CORRADINI: Okay.

DR. HAMBRIC: We'll get to all of that. So the classic example of this is not really a cylinder. It's like Tacoma Narrows Bridge which collapsed in 1940, classic vortex shedding, lock-in problem that nobody paid attention to going in.

CO-CHAIR CORRADINI: And everybody's seen the movie.

DR. HAMBRIC: Everybody's seen the movie. Look it up online, it's engineering disasters 101. So we don't want that to happen.

Now, how can that happen? It takes more than just a coincidence of the frequencies. It also takes a low impedant structural node. That means low mass, or low damping, or both. So we'll talk about

ASME criteria for that in a minute.

Now, the curious thing about this vortex shedding phenomena, I've got this little plot over here on the right. The transverse forcing, what I'm going to call the lift, and this is over time, it's at a higher frequency, or excuse me, a lower frequency than the unsteady drag force.

So the drag force is loading this thing at twice the frequency. Why does that matter? Because you're getting closer to the structural nodes. So it's actually the drag frequency you're mostly worried about here, intentionally lock in to a structural frequency.

Now, the good news is this has been studied so much we've got a nice formula that everybody uses that you can back out the velocity at which you're going to have vortex shedding or the frequency. And it's a Strouhal number. That's what the ST is. It happens to be a quarter.

F is frequency, D is the diameter, which you know, and U is your velocity. So if you know your flow velocity, you know your diameter, you can back out the frequency of that excitation and compare it to the structural resonance frequency and see how close you are. And that's your initial screening, really.

And so here are the reactor vessel internals we're concerned about. And we'll talk about helical coil steam generator in a minute. But on the left is the array of ICIGTs, and these are just half of them. There's a symmetry plane about the middle.

And up at the top, in the boxes, is where the rods are in cross flow. So over on the left, CFD analyses and the DUN (phonetic) are the flow speeds up at that top region, And those four velocities are compared to the diameters. Look at the Strouhal number, you've got a frequency of vortex shedding, both in the lift and in the drag direction.

CO-CHAIR CORRADINI: But I'm not putting the whole rod -- the whole rod is not seen. I'm seeing 10 or 15 percent of the length. That is not considered in the Strouhal number rule of thumb.

DR. HAMBRIC: Correct. So it'd just be over that top section which is a good thing. It means ---

CO-CHAIR CORRADINI: Right, I was going to say ---

DR. HAMBRIC: -- it'll be less likely to lock in, we're hoping, is the case.

CO-CHAIR CORRADINI: So the rest of this acts essentially as a long anchor relative to these things wiggling up at the top when the flow turns on you.

DR. HAMBRIC: Well, let's talk about that.

CO-CHAIR CORRADINI: Okay.

DR. HAMBRIC: Over on the right you've got the control rod drive shaft, bigger tubes. All of these rods are going through little holes in the support drills. So if you look, you can see these little kind of grills showing up.

CO-CHAIR CORRADINI: Yeah.

DR. HAMBRIC: And they've got to periodically space throughout the riser. And they've got different holes for the ICIGTs and different holes for the control rod drive shaft.

NuScale is assuming that those gaps are small enough that they can represent them as pinned boundaries.

CO-CHAIR CORRADINI: Pinned?

DR. HAMBRIC: Pinned.

CO-CHAIR CORRADINI: P-I-N-N-E-D?

DR. HAMBRIC: Yes.

CO-CHAIR CORRADINI: So is this ---

DR. HAMBRIC: Not exactly pinned, but kind of close. They're restraining in plain motion at those boundaries. They're allowing vertical motion.

They're restraining in flow (phonetic) motion. What it effectively does is limit your mode shapes to between the supports which shortens your length, which increases your resonance frequency, which gives you potentially biased margin.

The ICIGT gaps are pretty small. So that may be true, but we've seen nothing to really confirm that. You get a squeeze film effect that could cause a pinned condition.

Control rod drive shaft, the gaps are much bigger. These things have to be able to drop. So we really can't argue that point for the CRD shaft holes. They're much larger gaps. So we view the worst case, the most conservative resonances as due to the longer sections, lower frequencies, more of a chance of locking in. So that's a open item that NuScale is busy addressing.

The helical coil steam generator tubing, the only ones we have to worry about are down near the bottom. Here, the boundaries are those blue sections. Those are the supports, and so looking at the unsupported lengths of tubing and the flow over those lengths of tubing.

The supports are these little clips. And There's little schematics of them over on the right.

And there are clearances about maybe 10,000, a little bit less. You have to have clearance, otherwise you can't put it together. So they're supposed to kind of snap in there.

The argument NuScale is making, and we're waiting for calculations to substantiate this is, when you turn the heat on, the thermal expansion of everything should lock it all together. If that's the case, some good things happen.

Number one, there really shouldn't be any cause to worry about an inactive support, if any of those clips are just --- the boundary clearance is too big, then you've got a tube that's just got a floating space which means that you're unsupported length is longer, your resonance frequency is lower, you have more risk for locking into the vortex shedding. So if all the supports are fully active, we're in excellent shape.

NuScale is also assuming their supports are pinned, but if you look on the upper right, they've set the clips up cleverly so that each tube really has two supports on one side and one on the other. And it sort of alternates as you go through the tubes.

What that means is you're not only

restricting translational motion but moments as well. And that's a stiffer boundary condition which hopefully drives the resonance frequencies higher. But we have not seen proof of that yet, and we're hoping to get that proof in this new testing they're doing that we're going out to have a look at in about a month.

CO-CHAIR CORRADINI: You'll go and actually see the test rig?

DR. HAMBRIC: Yes.

CO-CHAIR CORRADINI: Okay.

DR. HAMBRIC: So let's talk about lock-in avoidance criteria. ASME provides us with some very concrete criteria in the pressure vessel and piping, Appendix M. And here it is. I'm going to walk through this and then lead up to a pretty important parameter that we also need confirmation of, and that's damping.

So the hatched region is the bad region. This is where flow over a cylindrical cross section could lock in and lead to synchronized response, and really high vibration amplitudes, and potential impacting of the nearby supports which we don't want. That would mean a lot of wear over time and potential failure.

So the vertical axis is speed. So down at the bottom is no flow. And then when you crank the flow up, you're going faster, faster, faster, faster. And eventually, you hit the bottom of this cross-hatched region. And that's where the structural resonances start locking in.

This is a non-dimensional velocity, it's maybe called a critical velocity. It's actually the inverse of the Strouhal number. But they're dividing by the resonance frequency restructure and the diameter of your tube.

CO-CHAIR CORRADINI: Say that again please, slower.

DR. HAMBRIC: Yes. That is the inverse of the Strouhal number.

CO-CHAIR CORRADINI: Where you're not computing the frequency, you're actually --

DR. HAMBRIC: Right.

CO-CHAIR CORRADINI: So using the first mode resonance frequency of the structure.

DR. HAMBRIC: Plug that in, plug your diameter in, and you'll get a critical velocity which is, in fact, what they did.

CO-CHAIR CORRADINI: Okay, got it.

DR. HAMBRIC: So they want to see if

they're in that cross-hatched region. Now, here's this structural impedance parameter. It's actually called a mass damping parameter. And this has been around for a long time, suggested by people that have done vortex shedding, lock-in over cylinders.

And there's two terms that they're after. One is the non-dimensionalized mass, so that's the modal mass of your structure. So if you have a big, heavy tube, you have a big heavy mass. But it's non-dimensionalized by essentially the mass of your forcing function which is this density of the fluid times the diameter squared. So that's the effective mass of the water displaced by the structure. So if you're in a heavy fluid, that gives you a smaller mass parameter.

The delta is damping. That's why they call it mass damping, mass times damping. Now, good things happen when mass goes up, when we slide over to the right. And good things happen when damping goes up, and we slide over to the right. We escape that synchronized range, completely.

If you look in the ASME guidance, there are criteria A through D. And I've got them in the appendix if you want to look them up. Three of those criteria pretty much say the same thing. It's make sure your frequencies don't align. You don't want the excitation frequency to align with the structural frequency. And that's these guys.

And I'm not going to go over them at all. Just try to make sure your flow velocity is as low as possible and your resonance frequencies are as high as possible. But for the helical coil steam generator, NuScale assumed the damping of one and a half percent.

Okay, so there's a couple of things on Criteria B. And that just says make sure your mass damping is above 32. Now, NuScale went and computed their critical velocities, their reduced velocities for the modes in the helical coil steam generator. And that's that up and down error.

This is the range they're in. Sometimes they're well below this range, but there are a few modes that are above it. However, if you see one and a half percent damping, and you come to the end of that synchronization region, all is well. There's no way that structure can possibly lock in to the vortex shedding. It's just got too much damping to do it.

If you go back and use one percent damping, as we do in Reg Guide 1.20, that slides over to the left, right, your mass damping parameter is down by 50 percent. And then some of your modes are falling in that synchronization range. That doesn't mean they'll lock in, but it means they could.

CO-CHAIR CORRADINI: This is where you're saying that past applicants essentially did an analysis of excitation versus structural resonance routes?

DR. HAMBRIC: No. This is just looking at whether you have a shot at locking in.

CO-CHAIR CORRADINI: No, I understand. But ---

DR. HAMBRIC: Oh, oh, if that happens, yes --

CO-CHAIR CORRADINI: If that happens the next recommendation would be to do a more detailed analysis.

DR. HAMBRIC: Right.

CO-CHAIR CORRADINI: Okay, all right. I got it.

DR. HAMBRIC: So I'll jump forward a bit to what they're trying to do with this next testing. If they can confirm that the resonance frequencies that they've been assuming so far are, in fact, conservatively low, and that their bounding conditions are, in fact, stronger, and the resonance frequencies go up, I come over here, and I put in a bigger resonance frequency, the F of N gets bigger. That slides down. My reduced velocity comes down and goes out of the synchronization range. So there's two paths to success.

CO-CHAIR CORRADINI: This way or down?

DR. HAMBRIC: One, get the resonance frequency as high as you can and prove to us that that's where they're going to be in the real plant. Two, if you've really got one and a half percent damping, prove it. And if they can prove it, we'll accept it. And we can retire that risk.

Okay, so that's vortex shedding. Next up is just the helical coil steam generator but now looking at something called fluid elastic instability which is worse than vortex shedding.

So this involves not only a lock-in between a flow excitation mechanism and a structural resonance, but the lock-in with multiple structural resonances which all influence the flow.

So I don't know if anybody's seen an animation, I should have probably tried to bring one, of an array of tubes experiencing FEI. But it's pretty dramatic. Probably ought to find them on the Internet sometime.

But this is just an example of many

measurements that have been done in this phenomena. It's an array of four tubes, up here in the upper right. And these little paths you're seeing here, kind of clockwise and counter-clockwise motion, that's what happens when you get full on, fluid elastic instability of that array.

All of those cylinders start vibrating out of phase with the others. So if you look at one, it might be moving clockwise, the one next to it will be counterclockwise, and the next two are clockwise. And as they do that, they're all grabbing hold of the fluid and shaking it around and reinforcing the flow excitation.

It's the same sort of mechanism just different. If this starts happening, then it's serious trouble. So this is an example of a measurement over several different tube types. This time, I've got flow speed on the bottom. But here's increasing flow. And this time I've got to vibration amplitude in the Y Axis.

So all the measurements start out with low flow, and they measure the vibration of the cylinders. And it starts going up, and in this particular case, this is a vortex shedding.

You've got a bit of a peak. Thankfully,

it does not lock in. The amplitude comes back down again. They keep increasing the flow, and then this coupled phenomenon, where all the tubes are moving and combining when the flow excitation kicks in.

And it doesn't take much to go up to --- this is a factor of six up here, just monstrous vibration amplitude. It becomes unbounded until these cylinders start hitting each other. And then everything goes non-linear, and who knows what happens after that. But that's not a position you want to find yourself in.

So the good news is, people have done a lot of measurements on these tubes. NuScale mentioned Connors. He really did the most famous ones. And his constants are used to this day in all of the cookbooks that you'll find on avoiding this mechanism.

And these were tube arrays in cross flow, different amounts of mass, different amounts of flow speed, different diameters, different damping. And the reason they did all of these ranges of tests was to try to come up with a universal design criteria that you could use.

S.S. Chen went through and did the same thing, but with helical coil steam generators many years later. And I cite that as well in a review. And it comes down to that same mass damping parameter all over again. And so here is the magic formula. This is the critical velocities. So it's velocity normalized by the resonance frequency. So you go off --

CO-CHAIR CORRADINI: And the critical velocity is the highest velocity as it goes through the gap?

DR. HAMBRIC: Yes, in this case, yeah. That's what you're looking for. What is your gap velocity?

CO-CHAIR CORRADINI: Okay.

DR. HAMBRIC: And the critical velocity, different investigators kind of pick different threshold points. It's just when things start going bad. So it's not when you've reached horribly bad, it's just the beginning of it. So that buys you a little bit of margin. It's not much, but eventually some. So I've got a little arrow pointing down here. They marked this as the beginning of critical loss.

So again, you get that by your known diameter and your structural resonance frequency. You can back out a critical velocity.

Okay, this is mass damping again, same parameter we had before. And what people do, and this is in the backup, is they'll make all these measurements and plot critical velocity versus that parameter. And they'll just eyeball a curve through it following this functional form.

And there's two empirical constants if you back out. One is just a constant that is an amplitude, that's the C. And over here, this Alpha is an exponent. And that's it.

Subsequent to that, more people have tried to come up with more involved terms, but this is conservative, it's fine. And it's what we're using here.

Now, there are two terms, one from Connors, which is straight tubes, and another from Chen, which is more prototypic to that, which is the helical coil steam generator.

So NuScale went through and looked at their tubes, assuming the conservative simply supportive bounding conditions, or pinned bounding conditions. I shouldn't call it pinned, because they do a lot of sliding through their support.

CO-CHAIR CORRADINI: We're talking the tubes, where there is one tab and two below?

DR. HAMBRIC: Yes.

CO-CHAIR CORRADINI: And they assume that they were just held like this?

DR. HAMBRIC: Which is conservative.

CO-CHAIR CORRADINI: Okay.

DR. HAMBRIC: Gives you a lower resonance frequency.

They also had to estimate the mass damping of each mode. And to do that you need the modal mass of each mode shape, the water mass that it displaces, it's that ratio. And for damping, usually they put it at one and half percent which needs to substantiated.

Over on the right is a schematic of the results. This red box is the range of their critical velocities. And this is for frequencies up to 30 Hz, because above 30 Hz there's really no risk of FEI occurring.

If you use Connors' constants, the line to meet is a critical velocity of 2.15. They're underneath that. If you use S.S. Chen's constants, and in particular Chen's constants, which span 95 percent of his data, so it gives you some certainty there, 95 percent certainty, I get a critical velocity of one and a half, putting you in the danger zone.

CO-CHAIR CORRADINI: I've got to ask, because I thought when this was presented by NuScale I got the inference that, by using the straight tube data, it was better. Am I misremembering?

DR. HAMBRIC: Yeah, and that's the blue on the top. Straight tube data is 2.15 on the top. That's Connors.

CO-CHAIR CORRADINI: And it's the angle of attack that's causing it, or the fact that cross flow versus parallel flow --

DR. HAMBRIC: We really don't all ---

CO-CHAIR CORRADINI: -- that causes the 2.15 versus 1.5?

DR. HAMBRIC: We really don't know. All we know is that Chen measured it, and that's what he got.

CO-CHAIR CORRADINI: Okay.

DR. HAMBRIC: There's been a lot of speculation as to what's going on there.

CO-CHAIR CORRADINI: But once again, the TF, whatever the name of the experiment is, we should be able to verify --

DR. HAMBRIC: Yep.

CO-CHAIR CORRADINI: Okay.

DR. HAMBRIC: That's why it's so critical to put this to rest. So we still view this a risk area. And that is motivating the TF3 testing.

So if any of the supports are inactive, if the thermal expansion is not sufficient to lock all of those tubes into all those supports, you might have some tubes that are just kind of floating through those tabs which means a much longer unsupported region, which means a lower resonance frequency, which means more chance of locking in.

We mentioned earlier there hasn't been any mesh conversion studies presented to date. So that means the resonance frequencies that they're predicting could be biased high.

Also flow velocities, they did an analysis assuming all active supports and came up with one set of critical velocities. And then they went and did an analysis assuming some inactive supports and said we're going to use average velocity instead of gap velocity for that, which we cannot understand why. So that's an RAI to them. That doesn't make sense to us.

And the final thing, I mentioned this earlier, is that assuming that the flow velocities are uniform through the entire section does not bear up. If you look at past measurements, this comes out of the Chen paper, this is a cross section of measurements that he made of the gap velocities at various regions through his steam generator array.

In this case, the mean flow is 0.72 meters per second. That's up on the top. And then if you squint, you can look at all the other velocities. But they vary quite a bit, some lower, some higher, but certainly not uniform.

And so we believe that, to be safe, some sort of an upper bound ought to be considered for this sort of assessment. But again, the proof will be in the flow testing that they do.

So testing to date, NuScale showed you earlier that the first sets of tests were for single tube internal flows. And they did various conditions. They had just water. They had water that was starting to boil, and then they had water that was boiling and converting to steam halfway up, so a lot of conditions.

The ones that stood out to us are the ones that showed this unexpected high spectral peak in the pressure specter that they measured. We don't know what this is. But it so far has not been included in the NuScale assessments of the forced response of their tubing. So it's the secondary flow, boiling, or transitioning to boiling, inside the tubing.

CO-CHAIR CORRADINI: Can you say that again. I don't think I understand.

DR. HAMBRIC: Okay. That's a pressure

spectrum measured inside the tubing of the secondary flow.

CO-CHAIR CORRADINI: Pressure spectrum inside the tubing.

DR. HAMBRIC: Yeah, so it's the wall pressures.

CO-CHAIR CORRADINI: Yeah.

DR. HAMBRIC: Those are the pressure pulsations inside the secondary flow.

CO-CHAIR CORRADINI: Oh, pulsations.

DR. HAMBRIC: As measured in the TF1 testing.

CO-CHAIR CORRADINI: So what is the Y axis again, one more time?

DR. HAMBRIC: Amplitude.

CO-CHAIR CORRADINI: Amplitude.

DR. HAMBRIC: That's the spectral peak that you're looking at there. It's high pressures.

CO-CHAIR CORRADINI: Oh, this is the frequency spectrum.

DR. HAMBRIC: Yep.

MEMBER BALLINGER: It's like a formation and collapse of the hose itself?

DR. HAMBRIC: We don't know. Where I found this before is in the oil and gas industry.

They spent a lot of time looking at upwardly moving boiling flow. And you go through these different regimes of flow where it's all just sort of aggregate. And then you get these bubbles forming and slugs. And the slugs can kind of generate something like this.

And eventually, you wind up with two-phase flow where you've got water in the walls and steam in the middle. And then things are okay again. But in that intermediate region, you can see stuff like this in the literature, these big peaks.

CO-CHAIR CORRADINI: And has NuScale, I'm sorry, I was trying to find your plot, NuScale has analyzed this or has not?

DR. HAMBRIC: They are in the process of analyzing it, and we're awaiting their forced response results.

CO-CHAIR CORRADINI: Ah, okay. Fine.

DR. HAMBRIC: But there's an RAI out to them, hey, what is this doing? Is this something to worry about or not?

CO-CHAIR CORRADINI: I guess my only guess is it's bubbly to annular flow transition.

DR. HAMBRIC: Yeah, could be.

CO-CHAIR CORRADINI: Yeah, okay.

DR. HAMBRIC: The tricky part is it's a

pretty high frequency for that. But I have no idea what the state of the fluid is, what the actual flow velocities are. It all depends on that, right.

CO-CHAIR CORRADINI: They could analyze it with anything, even RELAP is a tube, it's one dimensional for all intents and purposes.

MEMBER SUNSERI: And is the -- so on the left side it looks like it's coming down real --- is that just measurement, or is that something real?

DR. HAMBRIC: Oh, the zero Hz, yeah, that's just, like, static behavior.

MEMBER SUNSERI: Okay.

DR. HAMBRIC: They also had something called density wave oscillation in there. That'll cause that as well. And that's the whole point of their steam generator and the flow restrictors. That gets rid of that.

CO-CHAIR CORRADINI: To get rid of that.

DR. HAMBRIC: Yeah. But in their testing here, it showed up sometimes. There's nothing to really worry about. So that's an outstanding item.

CO-CHAIR CORRADINI: And your red wasn't shown in TF2. TF2 did not see what they saw in --

DR. HAMBRIC: No.

CO-CHAIR CORRADINI: But how is the TF1

supported? I don't remember now. It's just one wiggly tube?

DR. HAMBRIC: Yeah. I don't think there was a significant structural support. It was just, sort of, they held it in place so it wouldn't go anywhere.

Yeah, neither of these were meant to be FIV tested.

CO-CHAIR CORRADINI: Understood.

DR. HAMBRIC: They just took the opportunity to piggyback and get some data to help them out.

CO-CHAIR CORRADINI: I see.

DR. HAMBRIC: Now TF2, as Olivia showed you, was a non-prototypic early version of all of this, again, not intended for FIV, limited instrumentation and really short acquisition time. So it was hard to see much in it.

But I've got that in green, because we did spend a lot of time going through this with them. We see some resonant peaks, but nothing that's all that concerning. There's no evidence of, with increasing flow speed, some non-linear increase that would indicate lock-in.

So that's encouraging, but it's still not sufficient. And due to our concerns about the non-conservativisms potentially in their analyses, and the low margins, that's really putting a lot of importance on this TF3 test.

CO-CHAIR SKILLMAN: Steve, what is the image, that lower image?

DR. HAMBRIC: Oh, that's a zoom of the tubing and the supports in the TF2 test, right. So those are the supports they had there, kind of these long chunks of metal with grooves cut out of them which is, again, not what they're using now.

CO-CHAIR SKILLMAN: So is that prototypical for production? Or is that ---

DR. HAMBRIC: No, no. That's just a test they did, again, primarily for thermal hydraulics.

CO-CHAIR SKILLMAN: Thank you. All right, thanks.

DR. HAMBRIC: Yeah, what's more prototypic is what I showed you earlier and what's in this picture here.

So they're putting together five columns which should be enough to deal with this. They're heavily instrumented. We have no qualms whatsoever about what they're doing with the instrumentation, lots of sensors.

There's two sets of tests, one, that are going to help us in the near term and support our final safety evaluation report. And that is the modal dynamic test where they're going to go around and excite these tubes in various ways and try to back out resonance frequencies and damping factors.

The resonance frequencies they can use to go back and fine tune their modeling procedures. Once those procedures are fine-tuned, then they can apply them to their actual design and show us, here's what the real resonance frequencies are. And the hope is they're much higher than what they have in their current design application which gives us margin.

The tough one is going to be structural damping. Because they can't really heat this thing up to emulate thermal expansion effects. So they've come up with a methodology to kind of press all these tubes into the supports.

But this exercise they're busy going through now is trying to estimate how much force do we need to emulate what's really going to happen in thermal expansion. So we're waiting for that from them to convince us that what they're doing is prototypic.

Flow tests, they've committed to them

verbally, but we've got nothing in writing to that effect. We don't know when they're going to happen. Given the deadlines for the final SE, it would be highly unlikely that you're going to have flow data to support it.

So the best we can do is work rigorously with NuScale to ensure their flow test procedures are robust, that they're going to go across a wide enough range of flow speeds, be careful enough for their measurements, commit to resolving issues if they show up unexpectedly, things like that, try to make it as solid as we can so that we have reasonable assurance that nothing bad will happen in the actual plant when they go to prototype testing.

And once again, we'll be out there in a month looking at the first component of this which is the modal testing.

CO-CHAIR SKILLMAN: Steve, you mentioned several times the importance of fit of the tube into the clip. Is the fabrication process intended to spring load the tube so that when the tube is released it actually finds its way into its, if you'll clamp, into its support notch?

DR. HAMBRIC: It's supposed to snap into those little triads of clips, right.

CO-CHAIR SKILLMAN: Now, does that occur because the bending process, if you will, compresses the hoop so that when the hoop is released, under manufacturing, the tube snaps into the clip?

DR. HAMBRIC: That I don't know. See, in spite of NuScale's best efforts, I still don't have a great grasp on this whole thing hangs together. And that's a reason for going out and looking at it. I think once we see it, and they explain it to us, it'll make more sense. But, you know, we've got all these diagrams of clips, and supports, and it sounds great when they discuss it with us, but I'm still not making the connection mentally with how it's being put together.

CO-CHAIR CORRADINI: But as I heat up, I would expect things to essentially lock in better.

DR. HAMBRIC: That is their argument, right.

CO-CHAIR CORRADINI: Yeah.

CO-CHAIR SKILLMAN: Unless you get enough of them and heat them all, you may get an outboard compressive force that is greater than you anticipated for the support members that are placed vertically through that steam generator.

So my concern is the balance between the force necessary to find the clip -- to find the tube seated the way you want it seated, and having so many tubes expanding, when you use the modular heating system, that you actually injure a portion of the structure, because you've got so many tubes.

DR. HAMBRIC: That would be a question for NuScale in a different chapter, I'm afraid.

CO-CHAIR SKILLMAN: And it may be even being proprietary, so I don't fully understand that.

DR. HAMBRIC: It probably is, yeah.

MEMBER BALLINGER: You need to remember that the difference between San Onofre Unit 2 and Unit 3 was simply the fit.

DR. HAMBRIC: It's an important question. And we're still waiting for them to get back to us in a thermal expansion analysis which presumably should include all of that, right.

CO-CHAIR CORRADINI: But the fact, well, okay.

CO-CHAIR SKILLMAN: Maybe we'll discover this when we go out there in July.

CO-CHAIR CORRADINI: Well, keep in mind their testing is in Italy.

MEMBER BALLINGER: Well, maybe we should go to Italy.

(Laughter.)

CO-CHAIR CORRADINI: I don't think the invited us there.

CO-CHAIR SKILLMAN: I just know that there's some magic in steam generator fabrication. I worked for B&W for a long time, and there's a lot that's written and a lot that's craft. But there is magic in these steam generators. I know that for a fact.

DR. HAMBRIC: I cannot comprehend how they're putting this thing together. I guess that is beyond my mental ability to geometrically visualize it. But I'm taking them at their word.

MR. LISZKAI: I apologize, this is Tamas. Maybe I can clarify some of these questions that ---

DR. HAMBRIC: Please.

MR. LISZKAI: So the result of the thermal expansion, and the thermal expansion of the tubes, and components that are interacting in the steam generator, currently SME codes, which is the federal regulation, we will have to evaluate them. We are evaluating the thermal point pressures associated with thermal expansion and relative displacement of these components.

And those rules are your assurances that we're not exceeding any design stresses that will be damaging to these tubes as a result of thermal stresses. But that's really not on the review of the CVAP program. And it belongs in another chapter in our ACRS. I believe it's Chapter 5 ---

CO-CHAIR CORRADINI: Yeah, we discussed this under five.

MR. LISZKAI: And that has been addressed under that.

CO-CHAIR CORRADINI: So let me ask NuScale, is the firm that is doing the testing and manufacturing of your prototypic testing the same firm that's going to build your steam generators?

MR. LISZKAI: No, they are not.

CO-CHAIR CORRADINI: Is there a transfer of -- well, okay. That answers the question I wanted. That's fine. Thank you.

DR. HAMBRIC: Okay. So those are actually the more complicated structures. And we've got a couple more to go, but they're a little simpler. The next one is looking at the acoustic resonances and particular in the decay heat removal system piping. There are some side branches to be careful of there.

And here is what happens when you've got a flow instability locking into an acoustic resonance.

So it's the exact same phenomenon, just different equations. Here the flow instability is flow over the opening, and this is a side branch up here. And the certain magical speed you wind up with an instability exciting the opening there.

There are screw hole numbers that go with this which I'll show you in a minute. And by itself, there's nothing to worry about. If the frequency of that instability aligns with an acoustic resonance frequency in that fluid column, then you've got the potential for feedback. The acoustic mode reinforces the instability which reinforces the acoustic mode and the infinite loop to some sort of a limit cycle which you don't want to be at.

This is an example down at the bottom here. This is what we call a spectrogram, so it's not a frequency spectrum. There's frequency on this axis. Here's amplitude over here. This collection of little peaks here keeps growing, because on the axis on this side I've got increasing flow velocity.

So we increase the flow, and you get this huge non-linear increase in amplitude until you eventually hit some sort of limit usually associated with the damping in your system.

This is what caused the Quad Cities dryers to crack and fatigue fail. Believe it or not, the acoustic pulsations were strong enough to break apart a building. And that's how big they were. So we've got to watch out for them.

I mentioned earlier we have primary instabilities. Those are the ones that are generally the strongest. They're in phase across this opening. It's a half wavelength across the opening. So it's like a big piston source driving the acoustic --

They're also at twice, excuse me, at half the flow sped of the primary, so I slow the flow speed down, and I get a full wavelength across that opening. So it's like a dipole. I'm getting this pulsation on the opening. It's still trying to drive the acoustic mode, but it just doesn't do as good a job of it.

CO-CHAIR CORRADINI: You're most worried about the steam side, not the water side?

DR. HAMBRIC: I'm worried about anything.

CO-CHAIR CORRADINI: No, but I meant --- what I was trying to get at though was your analogy with BWR concerns is essentially the steam side of the DHRS.

DR. HAMBRIC: Right.

CO-CHAIR CORRADINI: Because the frequencies there would be lower ---

DR. HAMBRIC: Correct.

CO-CHAIR CORRADINI: -- relative to the water side.

DR. HAMBRIC: Yeah. And they evaluated everything.

CO-CHAIR CORRADINI: Okay.

DR. HAMBRIC: And there's plenty of margin in general on the water side. There was only two that we're still worried about that they'll be instrumenting to look for.

So even though the secondary excitation is weaker, it still has a potential to lock in. We've seen excitations on dryers before at the secondary flow instability, believe it or not. So it is something to be concerned about. Anything higher than that, like third order, we don't care about but first and second order, we do.

So the range of screw hole numbers, that's that frequency times diameter of the opening divided by flow speed, is wider here. And generally, we have to pick the most conservative one. And NuScale goes through and estimates these for all their openings.

And after their big screening, they really found only two locations that they didn't have great margin for. One had about a 20 percent margin against the primary number, so here their calculated Strouhal number is 0.75 compared to the upper bound here where they got about a 20 percent margin.

A better location, I'll show you both of these in a second, has a lot higher margin, like a factor of two. But for the secondary instability, cut that in half, and now you're getting close again.

CO-CHAIR CORRADINI: Why would you worry about the secondary instability?

DR. HAMBRIC: Again, we've seen it in plants before. It'll excite modes and ---

CO-CHAIR CORRADINI: So a higher mode.

DR. HAMBRIC: Yeah. So they have been there.

So they're instrumenting, in particular this location, and these are the DHRS actuation valves. They're down at the bottom there. And if you take a look, you can see side branches and Ts, and all of that other stuff. So flow over those regions are what we're concerned about.

And they put together a nice list of instrumentation. It really doesn't take much. If this thing locks in, you'll hear anything, strain gage, accelerometer, pressure transducer, it'll be everywhere.

The other concern is, if it does lock in, you want to look at your valves. The other thing that happened in Quad City is the valves failed just because of the amplitude of the excitation. It's obviously not something you want either.

But their approach makes sense. They're going to do this testing during initial startup. So when they turn the actual plant on, so we'll know whether there's a problem or not. And if it's significant, they've committed to resolving it.

CO-CHAIR CORRADINI: So how would we resolve it? Change the inlet flow condition or the location of the --- because the dead end portion of the valve is pretty well fixed just by design. So it has something to do with the inlet to the location?

DR. HAMBRIC: Change the diameter of the opening, you can change the ---

CO-CHAIR CORRADINI: So in other words --

DR. HAMBRIC: -- size of the side branch, the length of it? Yeah. Just change anything, really.

CO-CHAIR CORRADINI: But some of these things are kind of tough to ---

DR. HAMBRIC: Yeah, once they lock in, they can be annoying.

MEMBER REMPE: So when you talk about proposed instrumentation for initial startup, is that on the first module or ---

DR. HAMBRIC: Yes.

MEMBER REMPE: -- all 12 every time? Or just ---

DR. HAMBRIC: No, no. They're prototype modules.

MEMBER REMPE: First one, okay.

DR. HAMBRIC: And we'll talk more about the other instrumentation in a second. But his is a big part of what they're going to be measuring in initial startup. And if nothing happens, great. We've retired that risk for good.

All right, last one, leakage flow instability, there's a couple of open items here. And leakage flow instability, differed flow phenomenon, but now either locking in with a structural resonance, which is more typical, or in some cases an acoustic one. But here we're mostly interested in structural resonances.

But this happens in gap flow. So these are images I pinched from Tom Mulcahy's report from ANL about, boy, 30-something years ago. The reason he wrote that report is this stuff happens in reactors.

You don't want it to, but it happens.

A classic example is control rods and guide tubes. So these are all examples of structures inside tubes and the flow trying to squeeze by the gap. And there's other examples of this in turbo machinery with centrifugal pumps. You've got a suction adaptor, we get back flow and that leakage flow can increase and shake the pump around. It happens quite a bit.

And the issue is this. If I've got the flow, kind of sneak around here, actually that looks like it's the middle figure here. All right, the flow's coming in, and it's trying to squeeze around. And what happens is you wind up with an equal and opposite pressure loading on each side of the structure that starts shaking it up and down.

So the structure starts vibrating, and if things go wrong, the pressure on either side happens to be in phase with the structural motion. So the structure moves up, the pressure goes down. On the other side, as the gap opens, the pressure goes up which is the opposite of what you want.

So it starts shaking the structure even more which, in turn, reinforces the flow excitation, in turn reinforces the structural vibration, and off you go. It's banging against the wall, and squealing, and doing all sort of bad things. This can occur even with low flow rates. Is just depends on the geometry, and the gap size, and everything else.

The one great qualitative guideline that came out of Mulcahy's report is aim all of your obstructions downstream. And the good news is, for the steam generator and the flow restrictors, that's exactly what NuScale did. It's oriented in the right direction. So we're happy with that.

The other guidance from Mulcahy is every situation is unique. Measure it to assess the risk. So trying to come up with analytical approaches for this particular design is just not really worth it.

So here is the structure we're talking about. This is a big array of inlet flow restrictors. So you get one of these going into every steam generator tube. And the whole point of this is to try to mitigate this density of density wave oscillation, the super low frequency sloshing that can occur all the way up and down your tubes if you don't do something like this. This essentially damps all this out and keeps it from happening. You lose some head flow, but that's okay. It's factored into the design.

Again, there's really no clean way to

analyze this, at least not anything that would take you less than multiple years and a lot of scrutiny on our part.

So what they did instead is came up with a bunch of possible designs and stuck them in a bunch of holes, and drove flow through those holes at way higher than prototypic flow rates, measured everything, discarded the ones that shook around a bit, and kept the ones that had no sign of leakage flow instability, and then picked their favorite one, tweaked the design, and are moving forward with a follow-up test to prove to us that this final design is safe.

It's not going to be in initial startup testing, it's going to be in its own little side test. It's a validation test. But the benefit of that is, as we mentioned before, it can really crank up the flow to well above prototypic rates, find where leakage flow actually happens, if it does at all, and say here's how much margin we've got. We're safe, nothing to worry about. So it's a design validation test.

They provided us a test plan and measurement inspection program. The test plan looks good to us. They have taken their lessons learned from the initial tests, which they used to evaluate their designs, and applied that to their proposed validation testing.

Sadly, we will not get the results until after design certification. I hope we get to see them at some point, because it's a good final proof that all will be well. But thus far, we're just going with reasonable assurance that things should be safe.

CO-CHAIR SKILLMAN: Those who are on the public line, or on the Corvallis line, would you put your phone on mute. Excuse me.

DR. HAMBRIC: That's fine. Okay, so that's our assessment of leak in the flow restrictor. We have one more leakage flow instability topic to get to, and that is of general reactor vessel internals.

The few locations we kind of looked at and wondered about, some of them have been resolved, and others we have ongoing assessments at NuScale which we're auditing.

The one on the lower left, remember that NuScale puts their upper and lower risers together with sort of a press fit. And that's that tapered region. You've got kind of a bellows up above all of that which gives you some flexibility. But you've got high pressure inside, you've got lower pressure in the steam generator region so we wondered, can leakage flow kind of squeeze through that gap?

And they went through an assessment of what they're calling the hold down force just due to weight, and pressure, and just prove to us that the pressure difference is way smaller than the hold down force. And they just do not expect any flow sneaking through that gap at all. So we're happy with that.

The other components are the ones we talked about before, the ICIGTs, CRDSs, they're all being kind of snaked through these holes and these support structures. As the flow rises, it's going to try to squeeze through those holes and potentially shake those structures against the holes causing a leakage flow instability.

They've gone through and found some nice open literature references that assess that situation. And they're performing calculations using the gap widths, the flow speeds, and the pressure drops and try to show us if they've got margin as leakage flow for both of those cases.

There's also cases for the CRAGT over on the right. There're some flow gaps there as well. And they're assessing that also. So we expect a report from them in the next couple of months, hopefully before we submit our final SER.

Okay, plant measurements and inspections, we mentioned a couple of these already. But just to finalize it, these are all described or will be described in the measurement inspection program. The version we have now is incomplete, but the new version should cover everything.

We're still awaiting the final TF3 flow testing plan. And our path forward is to try to be as rigorous as we can to give us reasonable assurance that, when they're done, they will have found and mitigated any issues if they show up at all.

They will provide follow-up instrumentation and pre-test predictions. I think the pre-test predictions are pretty simple. They don't expect anything to happen. For initial startup testing in the prototype, these are limited in scope from what you may have seen in previous applications. But again, that's fine. We're really not trying to validate anything here other than nothing bad is going to happen.

But the instrumentation should be sufficient in breadth and location to capture and localize any unexpectedly high vibrations, so they can find it and mitigate it if that does occur.

CO-CHAIR SKILLMAN: Steve, would you go back a slide, please?

In this image, you show the control rod drive lead screws or extensions. You show the in-core instrument guide tubes and the control rod assembly guide tubes. And your focus has been on the fluid hydraulic interaction at what are basically the sleeve supports. What attention is given to the horizontal members?

DR. HAMBRIC: For the ICIGT?

CO-CHAIR SKILLMAN: Yeah. I mean, the horizontal members, at least, appear to be fragile to what is transverse flow.

DR. HAMBRIC: Yep?

CO-CHAIR SKILLMAN: What attention is given to them?

DR. HAMBRIC: Yeah, we argued about that quite a bit, actually, just because the flow through there is so poorly understood. There's been really no calculation of it. But we were able to resolve that a couple of different ways.

Number one, those are structures they actually have operating history for. The design is based on a previous design which has been subjected to much higher flow rates than what they're going to get in their plant.

They also went through an analysis using very conservative assumptions and showed that they expected a very minimal amount of wear over the life of the plant. And all of that is written up in the SE. So you can find their evaluation and our justification for why we think they're okay there.

CO-CHAIR SKILLMAN: Thank you.

DR. HAMBRIC: Okay. And I think you can ---

MR. WONG: Okay, thanks, Steve. The NuScale inspection plan is provided in the measurement and, sorry, NuScale provided the measurement and inspection plan in the measurement inspection technical report.

The components evaluated in the analysis program are inspected before and after the initial start test for any evidence of loose parts or wear as a result of vibration.

Components most susceptible to FIV are examined in limiting and representative locations such as load bearing elements, restraints, locking and building components, and contact surfaces.

Visual inspections are performed using VT-1 and VT-3 per ASME Boiler and Pressure Vessel Code

Section 11. VT-1 is for examining cracks and wear. And VT-3 is for determining the general mechanical and structural condition of components. The staff finds the inspection method and areas consistent with other design certification applications in Reg Guide 1.20.

These are just a summary of the issues for each open item. And Steve has addressed those issues in detail. And I'm not going to go over these again.

Next one, please. Again, these are the open items.

Next one, please. So next up, Steve already mentioned we're going to audit the SIET test facility in Italy in the summer of 2019. We're going to reveal the RAI responses, as well as the updated CVAP report, and the measurement and inspection report. And we're going to make a finding on the component design against the FIV.

And also, we need to make a decision regarding deferring the steam generator to a TF-3 test to after design certification. And this is the end of the presentation.

CO-CHAIR SKILLMAN: Gentlemen, thank you very much for a very thorough presentation.

Members, do you have any questions for the staff and the staff consultant?

(No audible response.)

CO-CHAIR SKILLMAN: No? Thank you very much.

MR. SNODDERLY: Dick, this is Mike Snodderly. I had one thing I just wanted to clarify. So for the full committee meeting, Marieliz, what it your intent that we wouldn't cover 392 in June, we would do 392 with all of Chapter 3 in July? Would you remind me again, what did you want?

MS. VERA: We're doing full committee June 5th.

MR. SNODDERLY: Okay. All right.

MS. VERA: So it's with the same group that --

(Simultaneous speaking.)

MS. VERA: -- presented this.

MR. SNODDERLY: Okay. I just wanted to confirm the availability of, okay. So then, okay, we'll do this June. Great.

CO-CHAIR SKILLMAN: Thank you. Let's first --- would you make sure the phone line is open, if it is, please?

PARTICIPANT: Yes.

CO-CHAIR SKILLMAN: Is there any individual in the room that would like to make a comment, please?

(No audible response.)

CO-CHAIR SKILLMAN: Seeing none, we're checking on the phone line.

MR. SNODDERLY: Is there anybody on the bridge line that would like to make a comment from the public?

CO-CHAIR SKILLMAN: Hearing none, thank you. I would like to thank the staff for their work and for the presentation. I want to thank NuScale for your travel, for your presentation, for your follow-up regarding compressibility of the fluid. Members, any final comments before we adjourn?

Dr. Schultz, thank you for coming down. To the members, thank you.

CO-CHAIR CORRADINI: So you're asking, I do have something.

CO-CHAIR SKILLMAN: You do? Oh.

CO-CHAIR CORRADINI: Yes, I do. I'm sorry.

CO-CHAIR SKILLMAN: Please.

CO-CHAIR CORRADINI: My recommendation for the full committee, if you're going to have this as part of the June meeting, is that there is -- the particular size that you guys went through, I'm going to talk --- I think in NuScale's case, they did a nice job of compressing it into a frame. So I'm not going to make a suggestion.

With all due respect, yours was a tad lengthier than I expected. My suggestion is there were three or four slides, I think they were slides 20 through 25, where you summarized the physical phenomena issues and the structures where you were concerned about the physical phenomena issues. I think that would be, for the members that aren't here, we lost a few, about half of us will want to hear that, in particular, Dr. Riccardella.

So my suggestion is, at the very least, hone in on those half dozen slides where you've actually said what's the phenomena and where are the structures that you're worried about. And then I leave it to you to, kind of, how you want to weave in the open items. I think, Professor, I think it started with 20, reactor internals comprehensive vibration.

DR. HAMBRIC: Right here?

CO-CHAIR CORRADINI: Yeah,

DR. HAMBRIC: Yeah, so this bottom line up front was what we're --

CO-CHAIR CORRADINI: Where you actually

talked about the physical phenomena --

DR. HAMBRIC: Yeah.

CO-CHAIR CORRADINI: -- where they occur, and what your worries were.

DR. HAMBRIC: Yeah.

CO-CHAIR CORRADINI: That would be my recommendation to help the rest of the members.

DR. SCHULTZ: But then some summary associated with the tie-in to the testing program --

CO-CHAIR CORRADINI: Correct.

DR. SCHULTZ: -- and how there is a path for resolution. The timing may be different than what one might like, but there is a path, a program that's planned.

CO-CHAIR CORRADINI: Correct, thank you very much. That sounds perfect. But that would be my recommendation. Because we have, for all chapters, 392, 14, 19, and 21, we have only a half a day. So we have to be somewhat Spartan as to what we can present to the rest of the committee.

DR. HAMBRIC: Sure.

CO-CHAIR CORRADINI: That would be my recommendation.

CO-CHAIR SKILLMAN: Yep.

CO-CHAIR CORRADINI: but I thought you

guys did a great job, both NuScale and the staff, and consultants.

CO-CHAIR SKILLMAN: Yep, me too. Colleagues, anything else?

(No audible response.)

CO-CHAIR SKILLMAN: Safe travels, everybody. We're adjourned.

(Whereupon, the above-entitled matter went off the record at 2:51 p.m.)

May 08, 2019 Docket No. 52-048

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

SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Subcommittee Presentation: NuScale FSAR Chapter 14, Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria," PM-0519-65457, Revision 0

The purpose of this submittal is to provide presentation materials for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) NuScale Subcommittee meeting on May 16, 2019. The materials support NuScale's presentation of Chapter 14, "Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria," of the NuScale Design Certification Application.

Enclosure 1 is the nonproprietary presentation entitled "ACRS Subcommittee Presentation: NuScale FSAR Chapter 14, Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria," PM-0519-65457, Revision 0.

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

If you have any questions, please contact Carrie Fosaaen at 541-452-7126 or at cfosaaen@nuscalepower.com.

Sincerely,

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

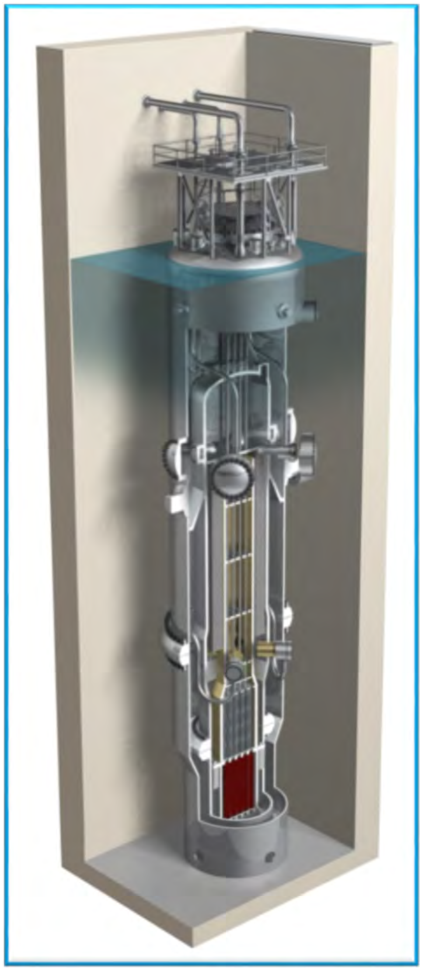
Distribution: Robert Taylor, NRC, OWFN-7H4
Michael Snodderly, NRC, TWFN-2E26
Gregory Cranston, NRC, OWFN-8H12
Samuel Lee, NRC, OWFN-8H12
Cayetano Santos, NRC, OWFN-8H12

Enclosure 1: "ACRS Subcommittee Presentation: NuScale FSAR Chapter 14, Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria," PM-0519-65457, Revision 0.

Enclosure 1:

"ACRS Subcommittee Presentation: NuScale FSAR Chapter 14, Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria," PM-0519-65457, Revision 0.

ACRS Subcommittee Presentation:



NuScale FSAR

Chapter 14

Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria

May 16, 2019

PM-0519-65457
Revision: 0

NUSCALE
Power for all humankind

Template #: 0000-21727-F01 R5

Presenters

Chris Maxwell
Senior Reactor Operator 4

Edan Engstrom
Senior Reactor Operator 4

Zack Rad
Director, Regulatory Affairs

Carrie Fosaaen
Supervisor, Licensing

14 Verification Programs

Section	Title
14.1	Specific Information to be Addressed for the Initial Plant Test Program
14.2	Initial Plant Test Program
14.3	Certified Design Material and Inspections, Tests, Analyses, and Acceptance Criteria

14.2 Initial Plant Test Program

14.2 Initial Plant Test Program

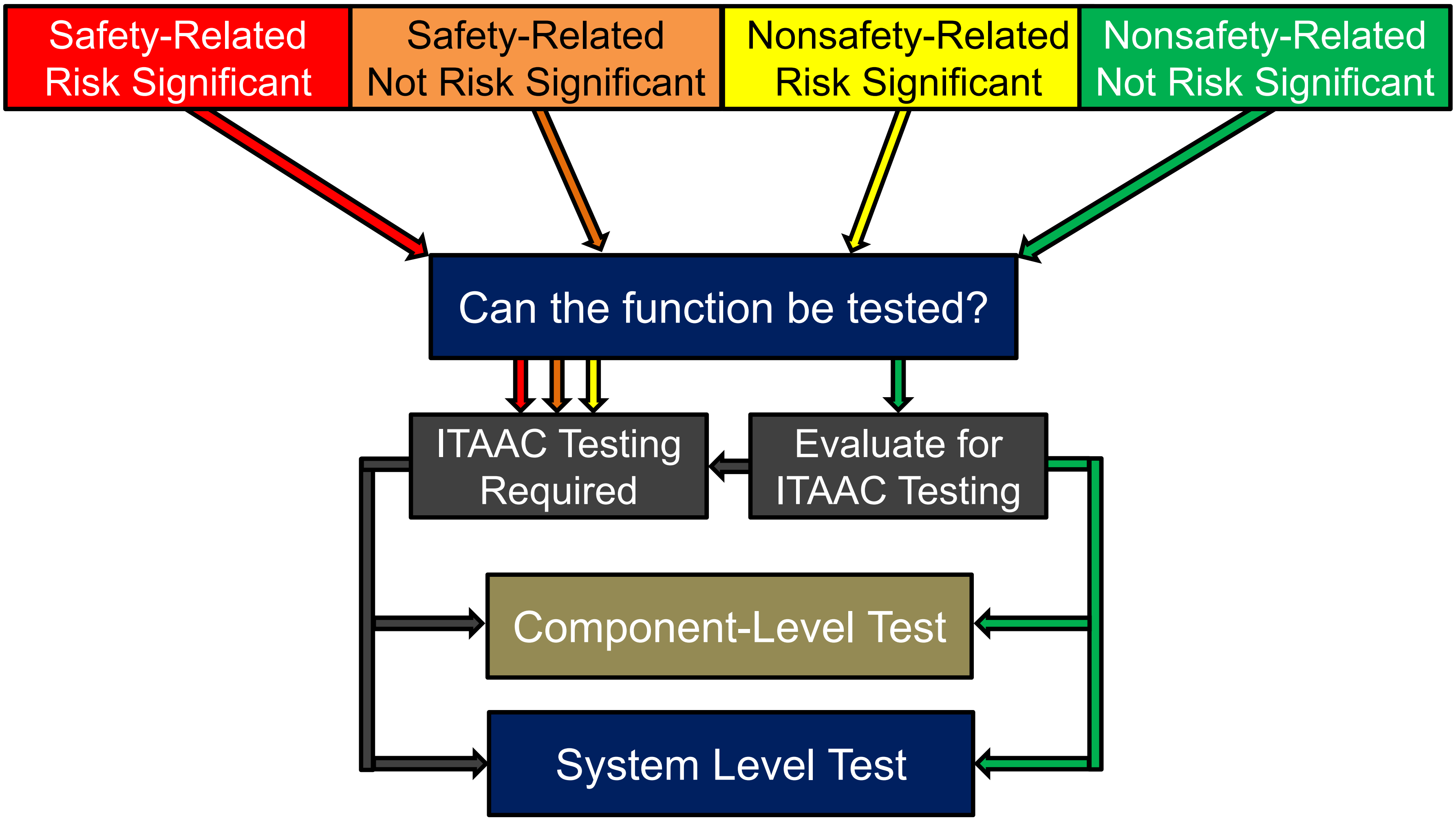
- Preoperational testing
- Startup testing
  - Initial fuel loading and pre-critical testing
  - Initial criticality testing
  - Low-power testing
  - Power-ascension testing
- First-of-a-kind testing

14.2 Initial Plant Test Program

- Preoperational Testing
  - From Regulatory Guide 1.68:

    “Preoperational testing,” as used in this regulatory guide, consists of those tests conducted following completion of construction inspections and tests, but before fuel loading, to demonstrate, to the extent practical, the capability of SSCs to meet the performance requirements to satisfy the design criteria.
  - RG 1.68, Appendix A, A-1 Preoperational Testing
  - Design Reliability Assurance Program (D-RAP)
    - Described in Chapter 17.4 of the NuScale FSAR
    - Functions were developed to describe each system
      - » Functions described in a “support system to supported system format”
      - » Functions were classified by safety and risk significance

14.2 Initial Plant Test Program



14.2 Initial Plant Test Program

- Testable D-RAP functions

Reactor Building Crane (RBC)			
• Supports NuScale Power Module by providing structural support and mobility while moving from refueling, inspection and operating bay	B1	• Reactor building crane • Module lifting adapter	Determination by PRA and concurrence by the expert panel as being needed for containment integrity

14.2 Initial Plant Test Program

- Testable D-RAP functions

Reactor Building Crane (RBC)			
• Supports NuScale Power Module by providing structural support and mobility while moving from refueling, inspection and operating bay		• Reactor building crane ...le lifting adapter	Determination by PRA and concurrence by the expert panel as being needed for containment integrity

• Supports NuScale Power Module by providing structural support and mobility while moving from refueling, inspection and operating bay

14.2 Initial Plant Test Program

- Table 14.2-52: Reactor Building Cranes Test #52

Table 14.2-52: Reactor Building Cranes Test # 52

Preoperational test is required to be performed once unless otherwise noted in the test.		
The RBC system is described in Section 9.1.5 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The RBC supports the NPM by providing structural support and mobility while moving from refueling, inspection and operating bay.	nonsafety-related, risk-significant	Test #52-1 Test #52-2
2. MAE bolting supports the CNT by providing material handling to allow for disassembly and reassembly of the CNV lower flange.	nonsafety-related	Test #52-2
3. MAE bolting supports the RPV actively by providing material handling to allow for disassembly and reassembly of the RPV lower flange.	nonsafety-related	Test #52-2
4. The CNTS supports the RBC by providing lifting attachment points that the RBC can connect to so that the module can be lifted.	nonsafety-related, risk-significant	Test #52-1 Test #52-2
Prerequisites i. An RBC site acceptance test has been completed and approved. ii. A rated-load test has been completed and approved on the RBC on the following equipment in accordance with ASME NOG-1 paragraph 7423. a. RBC main hoist b. RBC auxiliary hoists c. RBC wet hoist iii. A rated-load test has been completed and approved on the following equipment in accordance with ANSI N14.6. a. Module lifting adapter b. NPM lifting fixture iv. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify RBC controls that limit RBC motion and speed.	Actuate or simulate actuation of the RBC interlocks contained in Table 14.2-52a.	Local visual observation indicates that the interlocks limit RBC motion and speed.

14.2 Initial Plant Test Program

- Reactor Building Crane Test Abstract – Functions
  - Each testable function is listed on the test abstract, and the tests used to verify the functions are specifically identified.

Table 14.2-52: Reactor Building Cranes Test # 52		
Preoperational test is required to be performed once unless otherwise noted in the test.		
The RBC system is described in Section 9.1.5 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The RBC supports the NPM by providing structural support and mobility while moving from refueling, inspection and operating bay.	nonsafety-related, risk-significant	Test #52-1 Test #52-2

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14.2 Initial Plant Test Program

- Reactor Building Crane Test Abstract – Prerequisites
  - The prerequisites required to be completed prior to commencing preoperational testing are listed.

Prerequisites

i. An RBC site acceptance test has been completed and approved.

ii. A rated-load test has been completed and approved on the RBC on the following equipment in accordance with ASME NOG-1 paragraph 7423.

   a. RBC main hoist

   b. RBC auxiliary hoists

   c. RBC wet hoist

iii. A rated-load test has been completed and approved on the following equipment in accordance with ANSI N14.6.

   a. Module lifting adapter

   b. NPM lifting fixture

iv. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.

14.2 Initial Plant Test Program

- Test Abstract – Component Level Tests
  - Used to demonstrate and verify system functionality at the component level.

Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify RBC controls that limit RBC motion and speed.	Actuate or simulate actuation of the RBC interlocks contained in Table 14.2-52a.	Local visual observation indicates that the interlocks limit RBC motion and speed.
ii. Verify RBC remains in current position on loss of control or power or seismic event.	Initiate the following real or simulated signals: i. Loss of control. ii. Loss of power. iii. Seismic switch actuation.	Local visual observation indicates that the bridge, trolley, main hoist, wet hoist, auxiliary hoist trolley and auxiliary hoist brakes are set.
iii. Verify each RBC instrument is available on an MCS or PCS display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each RBC system transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.

14.2 Initial Plant Test Program

- Test Abstract – System Level Tests
  - Used to demonstrate and verify integrated functionality at the system level.

14.2 Initial Plant Test Program

System Level Test #52-1		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify RBC load path and removal of an NPM from a reactor bay. ii. Verify RBC load path and installation of an NPM in a reactor bay.	Place the module lifting adaptor on the RBC. Lift an NPM and move the RBC with the attached NPM to its design home location. i. Use the RBC semi-automatic programmed controls to install the NPM in the lead NPM bay location and return the RBC to the design home location ii. Use the RBC semi-automatic programmed controls to retrieve the NPM from the lead NPM bay location and return the RBC with attached module to the design home location. Repeat this sequence for each NPM installation.	i. The bridge and trolley speeds do not exceed maximum design speeds. ii. The bridge and trolley does not move at the same time. iii. The bridge and trolley maximum allowable speed is toggled from full-speed to microspeed when the RBC hook gets within the design distance of a predefined reference location. iv. The main hoist only moves within the predefined elevation zones. v. The NPM is positioned at the design rotation at predefined reference locations. vi. The NPM is fully seated in the reactor bay receiver. (Acceptance Criteria i through iv only need to be satisfied for the first performance of the test. Acceptance Criteria v and vi need to be satisfied for each NPM)

14.2 Initial Plant Test Program

- Test Abstract – System Level Tests
  - One element of the system function is demonstrated and verified by Test #52-1.

Table 14.2-52: Reactor Building Cranes Test # 52		
Preoperational test is required to be performed once unless otherwise noted in the test.		
The RBC system is described in Section 9.1.5 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The RBC supports the NPM by providing structural support and mobility while moving from refueling, inspection and operating bay.	nonsafety-related, risk-significant	Test #52-1 Test #52-2

14.2 Initial Plant Test Program

– The remaining elements of the system function are demonstrated and verified by Test #52-2.

System Level Test #52-2		
Test Objective	**Test Method**	**Acceptance Criteria**
i. a. Verify the NPM can be disassembled using the CNV support stand and the RPV support stand and associated tooling. b. Verify the RBC semi-automatic controls can be used to transport the NPM through the disassembly process. ii. a. Verify the NPM can be assembled using the CNV support stand and the RPV support stand and associated tooling. b. Verify the RBC semi-automatic controls can be used to transport the NPM through the assembly process.	The RBC is at the design home location with an NPM attached to the module lifting adaptor (MLA). i. Use the RBC semi-automatic programmed controls to move the NPM from the design home location to the CNV support stand and seat the NPM lower CNV in the CNV support stand. De-tension and remove the lower CNV closure bolts. Use the RBC semi-automatic programmed controls to move the NPM from the CNV support stand to the RPV support stand and seat the NPM in the RPV support stand. De-tension and remove the lower RPV closure bolts.	i. a. The NPM is disassembled using the CNV support stand and the RPV support stand and associated tooling. b. The RBC semi-automatic controls are used to transport the NPM through the disassembly process. ii. a. The NPM is assembled using the CNV support stand and the RPV support stand and associated tooling. b. The RBC semi-automatic controls are used to transport the NPM through the assembly process.

14.2 Initial Plant Test Program

- Startup Testing
  - From Regulatory Guide 1.68:

    Initial startup testing, as used in this regulatory guide, consists of equipment performance tests completed during and after fuel loading. These performance tests are normally completed during fuel loading, pre-critical, initial criticality, low power and power ascension phases to confirm the design bases and demonstrate, to the extent practical, that the plant will operate in accordance with design and that it is capable of responding to anticipated transients and postulated accidents as specified in the FSAR.

  - RG 1.68, Appendix A
    - A-2 Initial Fuel Loading and Pre-Critical Tests
    - A-3 Initial Criticality
    - A-4 Low-Power Testing
    - A-5 Power Ascension Testing

14.2 Initial Plant Test Program

- Test Abstract – Startup Testing
  - Different layout than the preoperational testing test abstracts, but include the same elements:
    - Test Objectives
    - Prerequisites
    - Test Method
    - Acceptance Criteria

14.2 Initial Plant Test Program

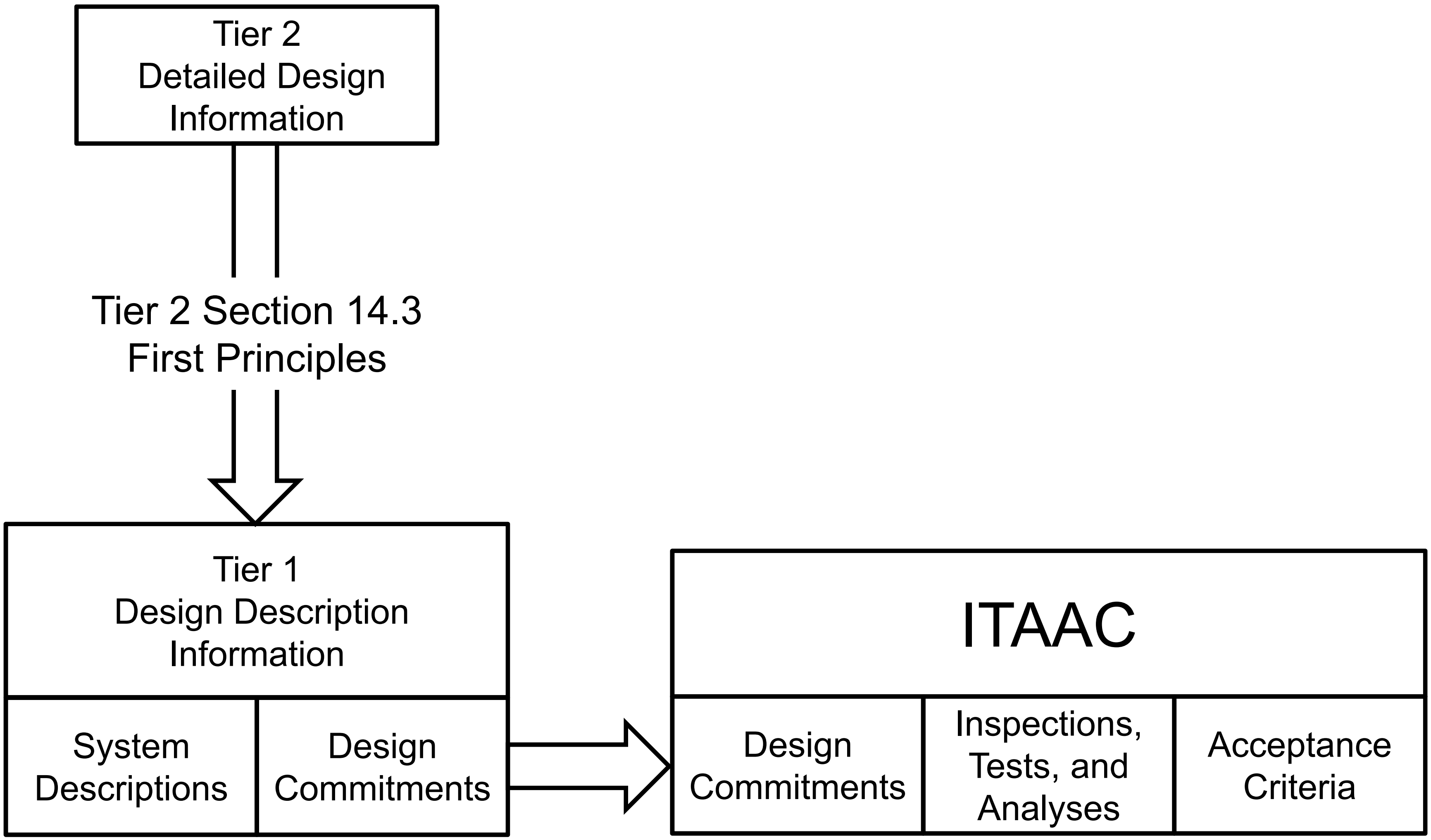
Table 14.2-81: Control Rod Assembly Full-Height Drop Time Test # 81
Startup test is required to be performed for each NPM.
This test is performed after initial fuel loading but prior to initial criticality.
Test Objective
Verify each CRA satisfies the CRA drop time acceptance criteria for RCS flow at 0 percent reactor thermal power.
Prerequisites
i. The core is installed. ii. The NPM is fully assembled. iii. The RCS is at hot zero power (RCS at normal operating pressure and RCS temperature at the maximum temperature obtainable when heated only by the MHS). iv. All RCS temperatures satisfy the minimum technical specification temperature for criticality. v. The nuclear instrumentation system is calibrated and operable. vi. The SDM is within the limits specified in the core operating limits report.
Test Method
i. Fully withdraw each individual CRA. ii. Interrupt the electrical power to the associated CRDM. iii. Measure the CRA drop time.
Acceptance Criteria
i. Each CRA drop time is within Technical Specification limits. ii. Each CRA drop time is within two sigma of the drop time data for all control rods, or has been verified within Technical Specification limits by a minimum of three additional performances of this test.

14.2 Initial Plant Test Program

- Test Abstract – First-of-a-Kind (FOAK) Tests
  - Regulatory Guide 1.68, Appendix A, A-6
  - FOAK tests are new, unique, or special tests used to verify design features that are being reviewed for the first time by the NRC.
  - Listed in Table 14.2-110, ITP Testing of New Design Features, and includes features such as:
    - ECCS valve design
    - Containment evacuation system
    - Island mode operation

14.3 Certified Design Material and ITAAC

14.3 to ITAAC Flowpath



14.3 Certified Design Material and ITAAC

- Provides guidance regarding the certified design material (CDM) in Tier 1, including Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) required under 10 CFR 52.47(b)(1).

- ITAAC – Those inspections, tests, analyses, and acceptance criteria identified in the combined license that if met by the licensee are necessary and sufficient to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license, the provisions of the Atomic Energy Act, as amended, and the Commission's rules and regulations.

14.3 Certified Design Material and ITAAC

- Detailed design information is contained in Tier 2.
- Tier 1 Design Descriptions include only the most safety-significant aspects of each of the systems described in the Tier 2 information.
  - Top-level design features
  - Top-level performance characteristics
- A “first principles” approach was used to select Tier 2 information for inclusion in the Tier 1 design descriptions.

14.3 Design Description First Principles

- Tier 1 Design Descriptions are limited to the top-level design features of the following:
  - Safety-related SSC
  - Nonsafety-related SSC that protect safety-related components
  - Security system physical SSC
  - Risk-significant, nonsafety-related SSC determined by results of probabilistic risk assessment

14.3 Design Description First Principles

- The top-level design features contained in Tier 1 design descriptions are:

- Reactor coolant pressure boundary
- Containment pressure boundary
- Seismic Category I Reactor and Control Buildings
- Radwaste Category RW-IIa Radioactive Waste Building
- Control room envelope
- Safety-related equipment qualification
- Safety-related component performance
- SSC providing protection of safety-related components
- Safety-related protection systems
- Components providing radiation protection for personnel and safety-related equipment
- New and spent fuel storage
- Security system physical components

Tier 1 Design Descriptions

- With the information selected to be included in the Tier 1 Design Description, the information is further divided into categories which include:
  - System Descriptions: a concise description of system functions, safety classification, and general location.
  - Design Commitments: a list of design features, such as seismic and ASME Code classifications, Class 1E equipment designation, and environmental qualification requirements.
- Only the design features described in the Design Commitments are verified by ITAAC.

Tier 1 ITAAC

- For each system with Design Commitments, a table of ITAAC entries is provided.
- ITAAC consists of three columns:

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
The NuScale Power Module ASME Code Class 1, 2 and 3 piping systems listed in Table 2.1-1 comply with ASME Code Section III requirements.	An inspection will be performed of the NuScale Power Module ASME Code Class 1, 2 and 3 as-built piping system Design Reports required by ASME Code Section III.	The ASME Code Section III Design Reports (NCA-3550) exist and conclude that the NuScale Power Module ASME Code Class 1, 2 and 3 as-built piping systems listed in Table 2.1-1 meet the requirements of ASME Code Section III.

Standardized ITAAC

- In a letter dated April 8, 2016, the NRC sent NuScale a set of standardized DCA ITAAC for use in a design certification application.
- Standardized ITAAC were incorporated along with design-specific ITAAC.

Portland Office
6650 SW Redwood Lane,
Suite 210
Portland, OR 97224
971.371.1592

Corvallis Office
1100 NE Circle Blvd., Suite 200
Corvallis, OR 97330
541.360.0500

Rockville Office
11333 Woodglen Ave., Suite 205
Rockville, MD 20852
301.770.0472

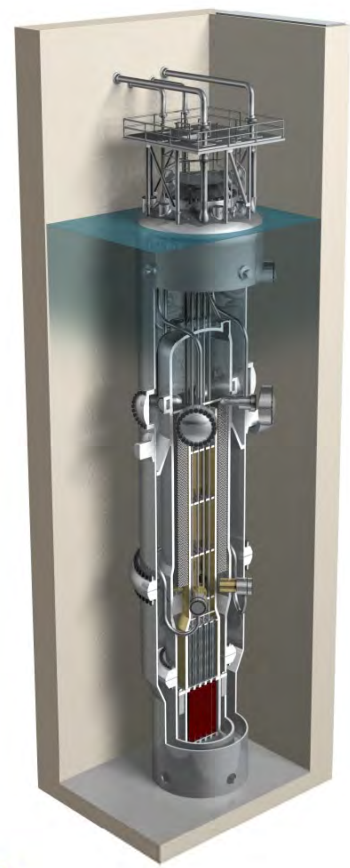
Charlotte Office
2815 Coliseum Centre Drive,
Suite 230
Charlotte, NC 28217
980.349.4804

Richland Office
1933 Jadwin Ave., Suite 130
Richland, WA 99354
541.360.0500

Arlington Office
2300 Clarendon Blvd., Suite 1110
Arlington, VA 22201

London Office
1st Floor Portland House
Bressenden Place
London SW1E 5BH
United Kingdom
+44 (0) 2079 321700

http://www.nuscalepower.com
Twitter: @NuScale_Power



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Presentation to the ACRS

NuScale Design Certification Application Review

Safety Evaluation with Open Items: Chapter 14

INITIAL TEST PROGRAM AND ITAAC

May 16, 2019

Staff Review Team

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- **Technical Reviewers**

Aaron Armstrong
Joe Ashcraft
Clinton Ashley
John Budzynski
Alexander Chereskin
Nan Chien
Jorge Cintron-Rivera
Timothy Drzewiecki
Greg Galletti
Joe Giacinto
Anne-Marie Grady
Zach Gran
Nick Hansing
Raul Hernandez
BP Jain
Nadim Khan
Taylor Lamb
Ronald LaVera
Pete Lee
Chang Li
Amanda Marshall
Ryan Nolan
Jonathan Ortega-Luciano
Paul Prescott
Sheila Ray
Robert Roche-Rivera
Fanta Sacko
Tom Scarbrough
Maurin Scheetz
Alex Siwy
Angelo Stubbs
Edward Stutzcage
Dinesh Taneja
Boyce Travis
Alexander Tsirigotis
Bob Vettori
Hanry Wagage
Yuken Wong

- **Project Managers**

Cayetano Santos– Chapter PM
Greg Cranston – Lead PM

Outline of Presentation

- **SER Section 14.2 – Initial Test Program (ITP)**

- **SER Section 14.3 – Inspections, Tests, Analyses and Acceptance Criteria (ITAAC)**

Technical Topics of Interest:
Section 14.2

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Tier 2, Chapter 14, Subsections:

14.2.1 - Summary of Initial Test Program and Objectives

14.2.2 - Organization and Staffing

14.2.3 - Test Procedures

14.2.4 - Conduct of the Test Program

14.2.5 - Review, Evaluation, and Approval of Test Results

14.2.6 - Test Records

14.2.7 - Test Programs Conformance with Regulatory Guides

14.2.8 - Utilization of Reactor Operating and Testing Experience in Test Program Development

14.2.9 - Trial Use of Plant Operating Procedures, Emergency Procedures, and Surveillance Procedures

14.2.10 - Initial Fuel Loading, and Initial Criticality

14.2.11 - Test Program Schedule and Sequence

14.2.12 - Individual Test Descriptions

Technical Topics of Interest:
Section 14.2

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Review Objectives

- Reviewed Tier 2, Section 14.2 for completeness and suitability for development of an ITP by a COL applicant against the guidance in the DSRS Section 14.2 and RG 1.68 by using a risk-informed approach.

Staff Review

- SECY-11-0024, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews," dated February 18, 2011, requested Commission approval of the staff's recommendation to develop a risk-informed and integrated framework for the review of the iPWR designs. On May 11, 2011, the Commission approved staff's approach.
  - Revised ITP review focuses on providing reasonable assurance that risk significant SSC functions are tested and a test abstract adequately addresses design functionality.
- NuScale DSRS Section 14.2, "Initial Plant Test Program – Design Certification and New License Applicants," dated July 11, 2016, provides guidance to the NRC staff for review of the proposed NuScale ITP in accordance with the approved approach.

Technical Topics of Interest:
Section 14.2

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Staff Review Continued

- The DSRS noted that there is no requirement for a DC applicant to provide an ITP submittal under 10 CFR Part 52, Subpart B, “Standard Design Certifications,” but the staff has reviewed the test abstracts provided by previous DC applicants for completeness and suitability for development of an ITP by a COL applicant against the guidance in the Standard Review Plan Section 14.2 and RG 1.68.
- The staff utilized Table 17.4-1, “D-RAP SSC Functions, Categorizations, and Categorization Basis,” in the DCA to determine the set of test abstracts to review using the risk-informed approach and for efficiency.
  - NuScale staff requested a larger scope of review.
- NRC approved only those test abstracts listed in Table 14.2-1 of the SER
- Test abstracts not approved are listed in Table 14.2-2 of the SER
  - Must be addressed by a COL applicant
  - If design certification is approved, staff would recommend that the certification rule include clarifying language that these test abstracts are outside the scope of the certified design.

Technical Topics of Interest: Section 14.2

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Staff Conclusion

- Open Item 03.09.06-1: The staff will keep open test abstract 14.2-47, "Emergency Core Cooling System Test #47" while the Chapter 3 open item is being resolved.

- Confirmatory Item 14.2-1: NuScale's response to the staff's review of the test abstracts in Table 14.2-1 of the SER included proposed markups to DCA Part 2, Tier 2. Therefore, the staff is tracking the incorporation of the proposed changes in these letters into a future revision of the DCA

- The staff concludes, using the information presented in the DCA, and pending the confirmation of the confirmatory item and closure of the open item, that the applicant has demonstrated compliance with NRC regulations and guidance.

Technical Topics of Interest:
Section 14.3 (ITAAC) Overview

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- Review of Tier 1 information including definitions, significant site parameters, interface requirements, and ITAAC tables

- Regulatory Bases
  - 10 CFR 52.47(b)(1)
  - SRP Section 14.3
  - Standardized ITAAC in letters dated April 8, 2016 and June 21, 2016
  - SECY-19-0034, "Improving Design Certification Content," describes revised general principles for the review of Tier 1
    - Tier 1 should typically be at a qualitative and functional level of detail.
    - Tier 1 should not include detail that could necessitate NRC approval for departures from the certified design that have minimal safety significance.
    - Numeric values in Tier 1 should be minimized.

Technical Topics of Interest:
Section 14.3 (ITAAC) Overview

- 14.3 subsections with no open items
  - 14.3.4 – Reactor Systems
  - 14.3.5 – I&C
  - 14.3.7 – Plant Systems
  - 14.3.10 – Emergency Planning
  - 14.3.12 – Physical Security
  - 14.3.13 – External Flooding

Technical Topics of Interest:
Section 14.3.1 - Selection Criteria for Tier 1

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- Staff excluded from its review NuScale's First Principles approach for determining the scope of Tier 1 and ITAAC
  - NuScale's approach similar to NEI 15-02, "Industry Guideline for the Development of Tier 1 and ITAAC under 10 CFR Part 52" and NEI white paper, "First Principles for Use in Developing Design Certification Tier 1 ITAAC," which NRC has not endorsed
  - DCA Part 2, Tier 2, Section 14.3.2 would not be incorporated by reference into a design certification rule

- Sections 14.3.2 through 14.3.13 document the staff's review of the ITAAC or reference other SER Chapters containing ITAAC evaluation

Technical Topics of Interest:
Section 14.3.1 - Selection Criteria for Tier 1

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- Open Item 17.4-1
  - In SECY-18-0093 staff recommended the use of ITAAC to verify effectiveness of D-RAP be discontinued and is awaiting Commission decision
  - No ITAAC provided for the D-RAP
- Open Item 14.3.1-1
  - Staff conducted review of Tier 1 for form and clarity and requested that NuScale make changes (RAI 9681)

Technical Topics of Interest:
Section 14.3.1 - Selection Criteria for Tier 1

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- Tier 1 Interface requirement evaluated in SER Section 3.7
  - Failure of any structures not within the scope of the certified design will not cause any of the Seismic Category I structures within the scope of the certified design to fail
  - 10 CFR 52.47(a)(26) requires that interface requirements be verifiable through ITAAC
  - Two ITAAC verify that as-built non Seismic Category I SSC will not impair the ability of Seismic Category I SSCs
  - Staff cannot make a finding that the 10 CFR 52.47(a)(26) requirement has been met because of Open Item 14.3.2-2

- **Open Item 14.3.2-1**

The ITAAC for the structural integrity of the reactor, radioactive waste, and control buildings are incomplete.

The application did not (i) address the deviations between assumed design loads and as-constructed loads, nor did it address the changes in demand resulting from these deviations, (ii) state that the design report will document the demand analysis using the same methodology used for the certification

- **Open Item 14.3.2-2**

The ITAAC for the seismic interaction of seismic category I SSCs with non-seismic category SSCs of the control building are not consistent with the ITAAC for RXB and are not in conformance with the Standardized DCA ITAAC acceptance criteria

Technical Topics of Interest:
Section 14.3.3 – Piping Systems and Components

- Open Item 14.3.3-1: NPM Valve Installation Verification ITAAC
  - ITAAC need to satisfy 10 CFR 52.47(b)(1) to provide reasonable assurance the NuScale Power Module (NPM) safety-related valves are constructed and will operate in conformity with the design certification.
  - NPM Valve Installation Verification ITAAC will require a walkdown inspection of the emergency core cooling system (ECCS) valves, containment isolation valves (CIVs), and decay heat removal system (DHRS) actuation valves to ensure the valves will not be prevented from performing their safety functions.

Technical Topics of Interest:
Section 14.3.3 – Piping Systems and Components

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- Walkdown inspection will verify installation of the ECCS valves, CIVs, and DHRS actuation valves and their hydraulic lines consistent with the specifications for geometric configuration, orientation, accessibility, and line routing such that each valve can perform its safety functions.
- Together with the current ITAAC, NPM Valve Installation Verification ITAAC will provide reasonable assurance that the ECCS valves, CIVs, and DHRS actuation valves will operate properly to allow core cooling and provide containment isolation under design-basis conditions.
- NRC staff held a public telecon with NuScale on May 8, 2019, to discuss the path forward for resolution of this open item.

Technical Topics of Interest:
Section 14.3.6 – Electrical Systems

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- The staff reviewed the NuScale design to determine whether the applicant established appropriate Tier 1 design commitments for the electrical systems and that they are verified by ITAAC.
  - Equipment Qualification for Seismic and Harsh Environment
  - Containment Electrical Penetrations
  - Lighting

- Open Item 8.3-1
  - The completion of the staff's review is awaiting for the completion of a Chapter 8 open item related to the GDC 17 and 18 exemptions.
- Open Item 14.3.6-1
  - The staff identified editorial errors in Tier 2, Table 14.3-1.

Technical Topics of Interest:
Section 14.3.8 – Radiation Protection

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- Open Items 14.3.8-1 and 14.3.8-2
  - Borated polyethylene shielding in Tier 1, Table 3.11-1.
  - The applicant revised the bioshield design several times. Borated polyethylene was originally provided on the top of the bioshield to shield neutrons. This shielding was also identified in Tier 1, Table 3.11-1. The applicant removed borated polyethylene from the top of the bioshield and from Tier 1, Table 3.11-1. Later, the applicant incorporated borated polyethylene into the front of the bioshield faceplate but did not add the faceplate borated polyethylene into Tier 1, Table 3.11-1.

Technical Topics of Interest:
Section 14.3.9 – Human Factors Engineering

- **Focus:** The as-built Human System Interfaces (HSI) in the Main Control Room (MCR) will be consistent with the HSI resulting from the applicant's Human Factors Engineering design process. ITAAC exist for the verification of system level displays, alarms and controls in the as-built MCR and Remote Shutdown Station (RSS).
- Open Item 18-22
  - The Design Commitment for the MCR does not include changes to the HSI design that could occur after Integrated System Validation.
- Open Item 14.3.9-1
  - The applicant did not include ITAAC for RSS displays, controls and alarms because there is no manual control of safety-related equipment from the RSS. Acceptability of this approach depends on the staff's approval of a partial exemption from the portion of GDC 19 requiring equipment outside the control room with a potential capability for subsequent cold shutdown of the reactor when the control room is evacuated.

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- Open Item 14.3.11-1
  - NuScale requested an exemption from the integrated leak-rate test requirement for the containment vessel (10 CFR 50 Appendix J, Type A) so no ITAAC was provided for Type A testing
  - SER Section 6.2.6 evaluation recommends granting this exemption so Open Item 14.3.11-1 is closed
    - SER Chapter 6 will be presented during a future meeting

Section 14.3 Conclusions

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- For those sections with open items, staff is unable to finalize its conclusions
- For those sections without open items, pending the resolution of any confirmatory items, the staff finds that the NuScale DCA contains the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria are met, a facility that incorporates the certified design has been constructed and will be operated in conformity with the applicable portions of the design certification, the AEA, and the NRC's rules and regulations.

Backup Slides (14.2)
Test Abstracts Reviewed

United States Nuclear Regulatory Commission
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Abstract	Test Title
Table 14.2-4	Pool Surge Control System Test #4
Table 14.2-5	Ultimate Heat Sink #5
Table 14.2-9	Auxiliary Boiler System Test #9
Table 14.2-18	Control Room Habitability System Test #18
Table 14.2-19	Normal Control Room HVAC [Heating, Ventilation, and Air Conditioning] System Test #19
Table 14.2-20	Reactor Building HVAC System Test #20
Table 14.2-24	Balance-of-Plant Drains Test #24
Table 14.2-25	Fire Protection Systems Test #25
Table 14.2-33	Turbine Generator Test #33
Table 14.2-35	Liquid Radioactive Waste System Test #35
Table 14.2-36	Gaseous Radioactive Waste System Test #36
Table 14.2-38	Chemical and Volume Control System Test #38
Table 14.2-41	Containment Evacuation System Test #41
Table 14.2-42	Containment Flooding and Drain System Test #42
Table 14.2-43	Containment System Test #43
Table 14.2-44	Control Rod Drive System Flow-Induced Vibration Test #44
Table 14.2-45	Reactor Vessel Internals Flow-Induced Vibration Test #45
Table 14.2-46	Reactor Coolant System Test #46
Table 14.2-47	Emergency Core Cooling System Test #47
Table 14.2-48	Decay Heat Removal System Test #48
Table 14.2-51	Fuel Handling Equipment System Test #51
Table 14.2-52	Reactor Building Cranes Test #52
Table 14.2-60	Plant Lighting System Test #60
Table 14.2-63	Module Protection System Test #63
Table 14.2-66	Safety Display and Indication Test #66
Table 14.2-68	Communication System Test #68
Table 14.2-70	Hot Functional Testing Test #70
Table 14.2-72	Steam Generator Flow-Induced Vibration Test #72
Table 14.2-73	Security Access Control Test #73
Table 14.2-74	Security Detection and Alarm Test #74
Table 14.2-76	Initial Fuel Load Test (Test #76)

Backup Slides (14.2)
Test Abstracts Reviewed

United States Nuclear Regulatory Commission
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Abstract	Test Title
Table 14.2-77	Reactor Coolant System Flow Measurement Test (Test #77)
Table 14.2-78	NuScale Power Module Temperature Test (Test #78)
Table 14.2-79	Primary and Secondary System Chemistry Test (Test #79)
Table 14.2-80	Control Rod Drive System – Manual Operation, Rod Speed, and Rod Position Indication Test (Test #80)
Table 14.2-81	Control Rod Assembly Drop Time Test (Test #81)
Table 14.2-81a	Control Rod Assembly Ambient Temperature Full-Height Drop Time Test #81A
Table 14.2-82	Pressurizer Spray Bypass Flow Test (Test #82)
Table 14.2-83	Initial Criticality Test (Test #83)
Table 14.2-84	Post-Critical Reactivity Computer Checkout Test (Test #84)
Table 14.2-86	Determination of Zero-Power Physics Testing Range Test (Test #86)
Table 14.2-87	All Rods Out Boron Endpoint Determination Test (Test #87)
Table 14.2-88	Isothermal Temperature Coefficient Measurement Test (Test #88)
Table 14.2-89	Bank Worth Measurement Test (Test #89)
Table 14.2-91	Core Power Distribution Map Test (Test #91)
Table 14.2-92	Neutron Monitoring System Power Range Flux Calibration Test (Test #92)
Table 14.2-93	Reactor Coolant System Temperature Instrument Calibration Test (Test #93)
Table 14.2-94	Reactor Coolant System Flow Calibration Test (Test #94)
Table 14.2-95	Radiation Shield Survey Test (Test #95)
Table 14.2-96	Reactor Building Ventilation System Capability (Test #96)
Table 14.2-97	Thermal Expansion Test (Test #97)
Table 14.2-98	Control Rod Assembly Misalignment (Test #98)
Table 14.2-99	Steam Generator Level Control Test (Test #99)
Table 14.2-100	Ramp Change in Load Demand (Test #100)
Table 14.2-101	Step Change in Load Demand Test (Test #101)
Table 14.2-102	Loss of Feedwater Heater Test (Test #102)
Table 14.2-103	100 Percent Load Rejection Test (Test #103)
Table 14.2-104	Reactor Trip from 100 Percent Power Test (Test #104)
Table 14.2-105	Island Mode Test for NuScale Power Module #1 (Test #105)
Table 14.2-106	Island Mode Test for Multiple NuScale Power Modules (Test #106)
Table 14.2-108	NuScale Power Module Vibration Test (Test #108)

Backup Slides (14.2)
Test Abstracts Not Reviewed

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Abstract	Test Title
Table 14.2-1	Spent Fuel Pool Cooling System Test #1
Table 14.2-2	Pool Cleanup System Test #2
Table 14.2-3	Reactor Pool Cooling System Test #3
Table 14.2-6	Pool Leak Detection System Test #6
Table 14.2-7	Reactor Component Cooling Water System Test #7
Table 14.2-8	Chilled Water System Test #8
Table 14.2-10	Circulating Water System Test #10
Table 14.2-11	Site Cooling Water System Test #11
Table 14.2-12	Potable Water System Test #12
Table 14.2-13	Utility Water System Test #13
Table 14.2-14	Demineralized Water System Test #14
Table 14.2-15	Nitrogen Distribution System Test #15
Table 14.2-16	Service Air System Test #16
Table 14.2-17	Instrument Air System Test #17
Table 14.2-21	Radioactive Waste Building HVAC System Test #21
Table 14.2-22	Turbine Building HVAC System Test #22
Table 14.2-23	Radioactive Waste Drain System Test #23
Table 14.2-26	Fire Detection System Test #26
Table 14.2-27	Main Steam System Test #27
Table 14.2-28	Feedwater System Test #28
Table 14.2-29	Feedwater Treatment System Test #29
Table 14.2-30	Condensate Polishing System Test #30
Table 14.2-31	Feedwater Heater Vents and Drains System Test #31
Table 14.2-32	Condenser Air Removal System Test #32

Backup Slides (14.2)
Test Abstracts Not Reviewed

Abstract	Test Title
Table 14.2-34	Turbine Oil Storage System Test #34
Table 14.2-37	Solid Radioactive Waste System Test #37
Table 14.2-39	Boron Addition System Test #39
Table 14.2-40	Module Heatup System Test #40
Table 14.2-49	In-core Instrumentation System Test #49
Table 14.2-50	Module Assembly Equipment Test #50
Table 14.2-53	Process Sampling System Test #53
Table 14.2-54	13.8kV [kilovolt] and Switchyard System Test #54
Table 14.2-55	Medium Voltage AC [alternating current] Electrical Distribution System Test #55
Table 14.2-56	Low Voltage AC Electrical Distribution System Test #56
Table 14.2-57	Highly Reliable DC [direct current] Power System Test #57
Table 14.2-58	Normal DC Power System Test #58
Table 14.2-59	Backup Power Supply System Test #59
Table 14.2-61	Module Control System Test #61
Table 14.2-62	Plant Control System Test #62
Table 14.2-64	Plant Protection System Test #64
Table 14.2-65	Neutron Monitoring System Test #65
Table 14.2-67	Fixed-Area Radiation Monitoring System Test #67
Table 14.2-69	Seismic Monitoring System Test #69
Table 14.2-71	Module Assembly Equipment Bolting Test #71
Table 14.2-75	Initial Fuel Loading Precritical Test #75
Table 14.2-85	Low-Power Test Sequence Test #85
Table 14.2-90	Power Ascension Test #90
Table 14.2-107	Remote Shutdown Workstation Test #107

May 7, 2019 Docket No. 52-048

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

SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "FSAR Section 3.9.2: Dynamic Testing and Analysis of Systems, Components, and Equipment," PM-0419-65367, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) NuScale Subcommittee Meeting on May 16, 2019. The materials support NuScale's presentation of Section 3.9.2 "Dynamic Testing and Analysis of Systems, Components, and Equipment," of the NuScale Design Certification Application.

The enclosure to this letter is the nonproprietary version of the presentation titled "FSAR Section 3.9.2: Dynamic Testing and Analysis of Systems, Components, and Equipment," PM-0419-65367, Revision 0.

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

Sincerely,

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

Distribution: Robert Taylor, NRC, OWFN-8H12
Michael Snodderly, NRC, TWFN-2E26
Samuel Lee, NRC, OWFN-8H12
Gregory Cranston, NRC, OWFN-8H12
Marieliz Vera, NRC, OWFN-8H12

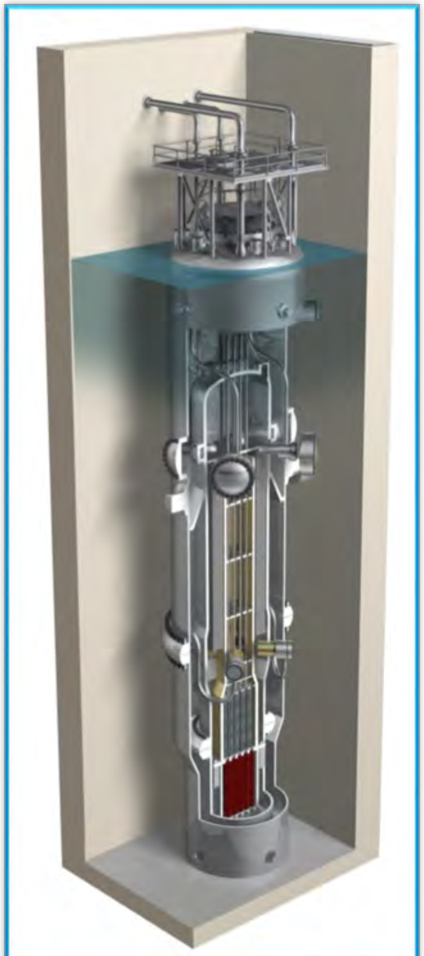
Enclosure: "FSAR Section 3.9.2: Dynamic Testing and Analysis of Systems, Components, and Equipment," PM-0419-65367, Revision 0

Enclosure:

“FSAR Section 3.9.2: Dynamic Testing and Analysis of Systems, Components, and Equipment,” PM-0419-65367, Revision 0

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

ACRS Subcommittee Presentation:



NuScale FSAR Chapter 3.9.2,

Dynamic Testing and Analysis of Systems, Components, and Equipment

May 16, 2019

Presenters

J.J. Arthur, P.E.

Manager, Structures and Design Analysis

Dylan Addison

NuScale Mechanical Engineer

Olivia Hand, P.E.

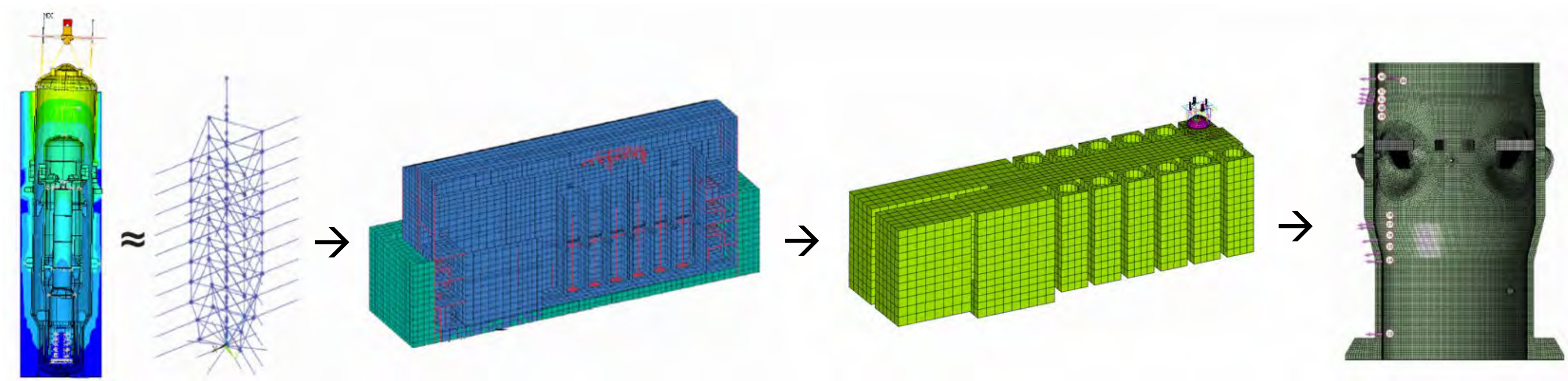
NuScale Mechanical Engineer

FSAR 3.9.2 Subject Areas

- The NuScale Power Module Seismic Technical Report, TR-0916-51502 addresses the requirement to perform dynamic analysis of the systems, components, and equipment
- The NuScale Comprehensive Vibration Assessment Program technical report, TR-0716-50439, and the NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan technical report address the requirement for flow induced vibration assessment.
- The NuScale Power Module Short-Term Transient Analysis technical report, TR-1016-51669 addresses transients caused by failure/actuation of valves and HELB

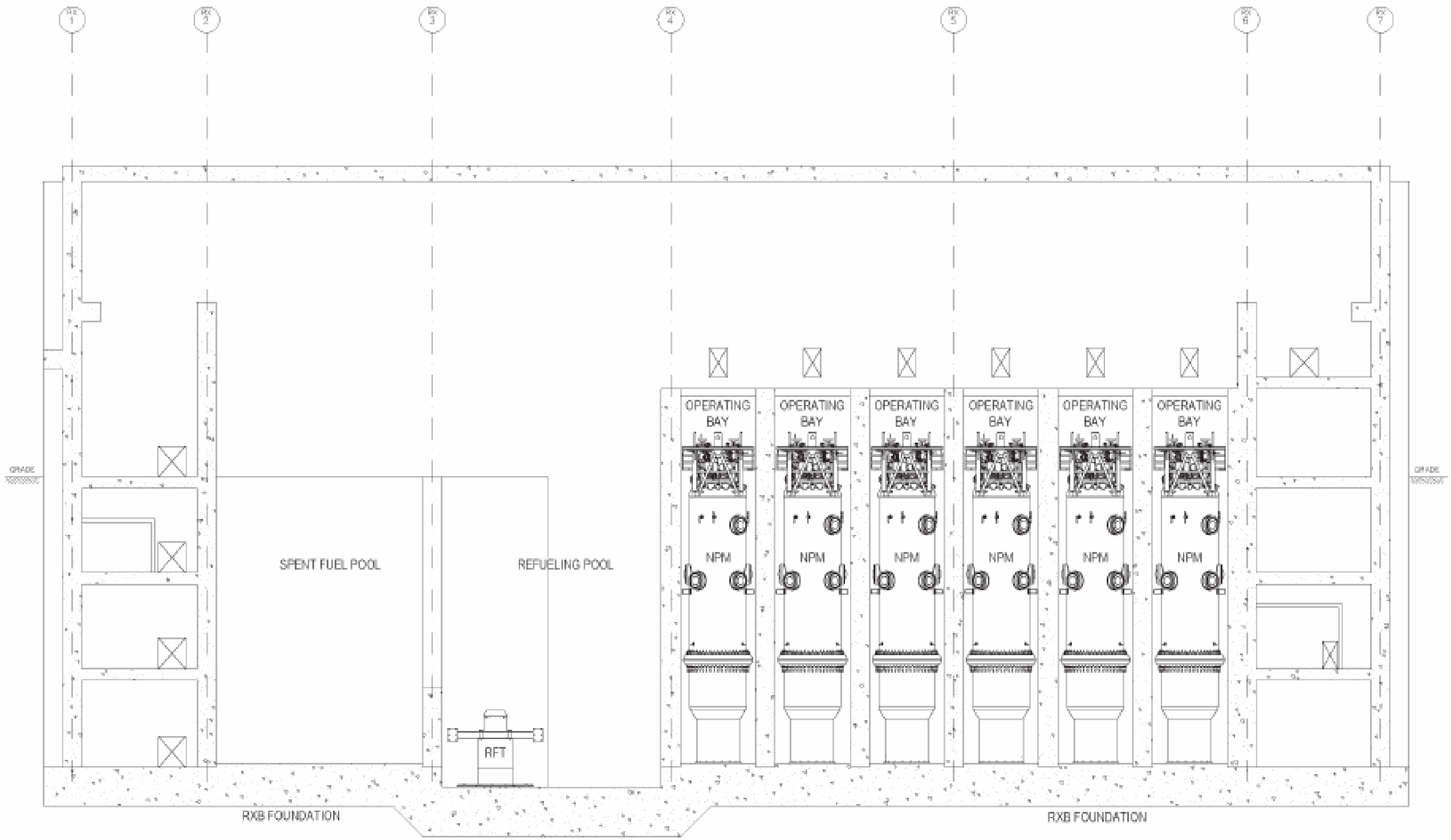
NPM Seismic Qualification

- Confirms functional integrity and operability of SC-1 mechanical equipment after a seismic event
- Overview of methods addressed in FSAR Sections 3.7, 3.10, 3.12, and App 3A:



NPM Beam Model (dynamically equivalent to detailed 3D ANSYS model) → SSI Analysis (Reactor Building SASSI model) → Detailed 3D Analysis (full pool ANSYS model) → Stress Analysis (SSE loads applied)

NPM Seismic Qualification



NPM Seismic Qualification

- The methodology is utilized for generating seismic loads for use in stress analyses
- Section 3.8.2 addresses stress analysis of the CNV, including the SSE
- Service Level D stress analysis for the RPV has been assessed and found acceptable
- Service Level D stress analyses for the RVIs are currently being updated (tracked as SER open item 03.09.02-11)
- ITAAC design commitments ensure that NPM components conform to the rules of construction of ASME BPVC Section III

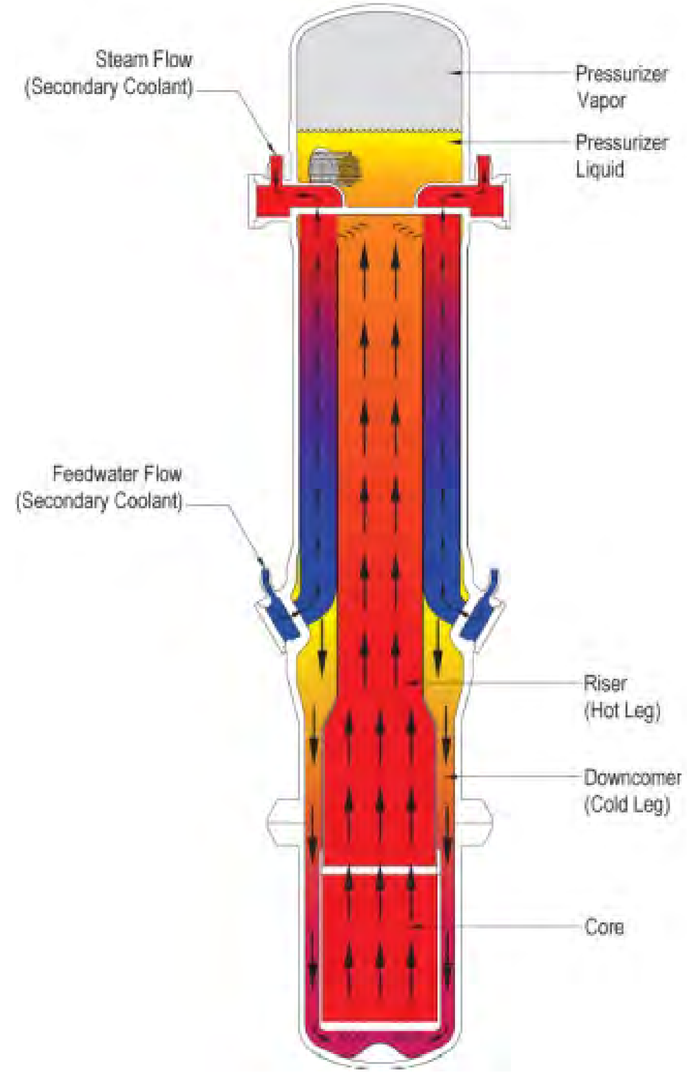
NPM Seismic Qualification COL Item

COL Item 3.9-12:

A COL applicant that references the NuScale Power Plant design certification will perform a site-specific seismic analysis in accordance with Section 3.7.2.16. In addition to the requirements of Section 3.7, for sites where the high frequency portion of the site-specific spectrum is not bounded by the CSDRS, the standard design of NPM components will be shown to have appropriate margin or should be appropriately modified to accommodate the site specific demand.

NPM CVAP Overview

- Addresses components subject to natural circulation primary coolant or secondary coolant flow in the NPM
- Consists of analysis, measurement, and inspection programs:
  - Screens all NPM components for six FIV phenomena
  - Evaluations based on industry standard analytical approaches and benchmark testing
  - Validation testing is performed for components with safety margin less than 100%
  - Inspection of all components regardless of analytically-predicted safety margin



Program Differences Compared to Recent CVAPs

- More components evaluated (not limited to RVI)
- More FIV mechanisms considered in the analysis program
- Significantly lower primary coolant flow rates:

Design (Note 1)	Average Velocity (ft/s)				Maximum Design Flow Rate (lb_m/s)	Primary Coolant Loop Transit Time (seconds)
	Steam Generator Gap	Downcomer	Core	Upper Internals Cross Flow		
NuScale	**1.2**	**1.7**	**3.6**	**1.5**	**1,456**	**60.8**
EPR		24	16	30	55,000	9.9
AP1000		19	16	40	34,800	10.3
US-APWR		23	14	30	54,092	12.6
SONGS	18	-	-	-	-	-

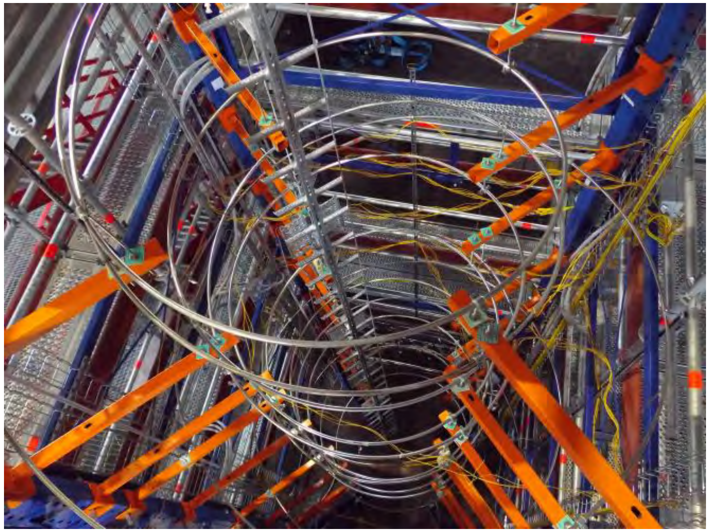
- First of a kind design, no proprietary scale testing
- Performing majority of validation testing prior to the start-up testing program (SG assembly and inlet flow restrictor)
- Larger inspection scope

NPM CVAP Design Analysis Summary

- Determine calculation inputs
  - Component modal response
  - ASME Appendix N guidelines
  - Flow velocities
- All safety margins are positive (i.e., FIV is not predicted to occur). The most limiting results are shown in table below:

Component	Analysis Category	Safety Margin	Major Inputs to be Verified	Testing Phase
Helical SG tube	FEI	~10%	Frequencies mode shapes vibration amplitude	Separate Effects
	TB	~80%		
	VS	~20%		
ICIGT	VS	~25%	fundamental frequency	Factory
	TB	100%	N/A	N/A
CRD shaft	VS	~25%	fundamental frequency	Factory
	TB	100%	N/A	N/A
DHRS steam piping	AR	~20%	vibration amplitude	Initial Startup

CVAP Measurement Program Summary



TF-1

TF-2 (above)
TF-3 (right)

- Benchmarking using TF-1, TF-2 and TF-3 "build-out" test results for the SG assembly, and SG IFR testing
- Post-DCA validation testing using TF-3, SG IFR, and initial startup testing (steam piping)
- Initial startup testing sensors provided to confirm lack of vibration, supplementing the CVAP inspections following initial startup testing

CVAP Inspection Program Summary

- Components that screen for an FIV mechanism will be inspected before and after initial start-up testing to look for mechanical wear or signs of vibration-induced damage

- Purpose of inspection is to confirm results of analysis and validation programs

- Initial start-up testing provides for 1 million cycles of the most limiting (lowest fundamental frequency) component, helical SG tubes

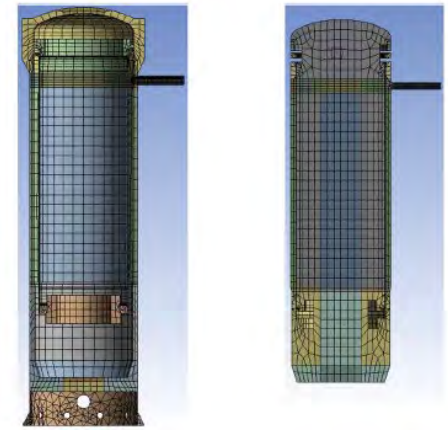
- Inspections are performed using VT-1, VT-3 and general visual inspections defined in ASME Section XI, Subsection IWB-2500

CVAP COL Item

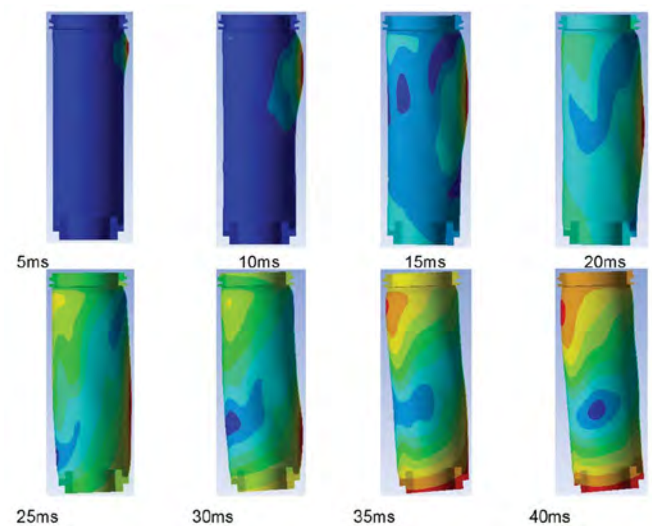
- COL applicant to provide test procedures prior to start of testing and will submit test and inspection results in accordance with RG 1.20

NPM Short Term Transient Analysis

- Purpose is to determine time-history structural response of NPM to the pressure wave resulting from a breach in the pressure boundary
- NRELAP5 is used to generate thermal hydraulic boundary conditions (thrust force, fluid acceleration)
- ANSYS is used to simulate the fluid structure interaction and resulting forces and moments on structures



Heissdampf reactor (HDR) structural and acoustic model



HDR time history core barrel deformations

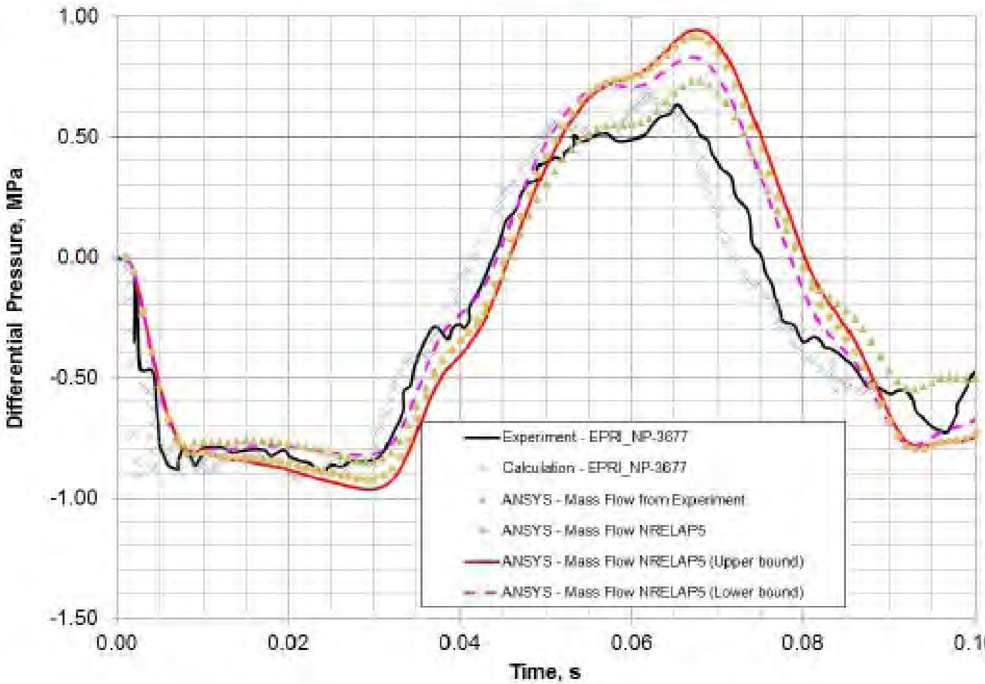
Program Differences Compared to Previous PWR Designs and Legacy Analysis Methods

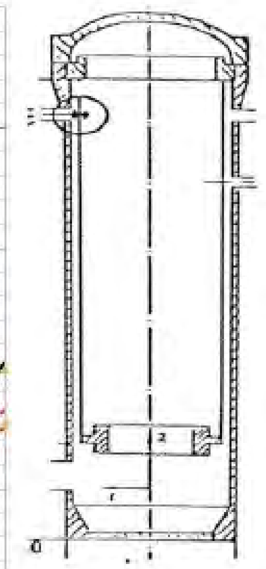
- No large diameter primary coolant piping (NPS 2).
- Valves represent the largest possible breaches in the reactor coolant pressure boundary
- Lower operating pressures, less sub-cooling, CNV is a single sub-compartment
- Legacy codes like MULTIFLEX and CRAFT2 generally use homogeneous equilibrium,1D fixed mesh, simplified or no FSI to simulate blowdown loading
- With availability of more modern codes and computing power, use of non-equilibrium, 3D models, and simulation of FSI via acoustic elements provides for enhanced prediction of pressure wave phenomena and resulting NPM loading

NPM Short Term Transient Analysis

<u>Benchmarking</u>

- Analysis method is benchmarked against Heissdampf reactor, Marviken and Bettis Hydraulic pressure pulse experiments to demonstrate ability to accurately simulate the thermal hydraulic and structural time histories.





- Sensitivity studies were performed to determine optimal analysis settings for NPM breach locations.

- Bounding breach locations analyzed and maximum forces at moments and pressure vessel and interface locations are determined for use in NPM component stress analysis.

- No COL Items for this analysis area.

Acronyms

AR	acoustic resonance
ASME	American Society of Mechanical Engineers
BPVC	Boiler and Pressure Vessel Code
CFR	Code of Federal Regulations
CNV	containment vessel
COL	combined license
CRD	Control Rod Drive
CSDRS	Certified Seismic Design Response Spectra
CVAP	Comprehensive Vibration Assessment Program
DCA	Design Certification Application
DHRS	Decay Heat Removal System
EPRI	Electric Power Research Institute
FEI	fluid elastic instability

Acronyms (continued)

FIV	flow-induced vibration
FSAR	Final Safety Analysis Report
FSI	fluid-structure interaction
GDC	General Design Criteria
HELB	high energy line break
ICIGT	In-Core Instrument Guide Tube
NPM	NuScale Power Module
NPS	nominal pipe size
RAI	request for additional information
RVI	reactor vessel internals
RXB	reactor building
SC-I	Seismic Category I
SER	Safety Evaluation Report

Acronyms (continued)

SG	steam generator
SSI	soil-structure interaction
TB	turbulent buffeting
TF	SIET test fixture

Portland Office
6650 SW Redwood Lane,
Suite 210
Portland, OR 97224
971.371.1592

Corvallis Office
1100 NE Circle Blvd., Suite 200
Corvallis, OR 97330
541.360.0500

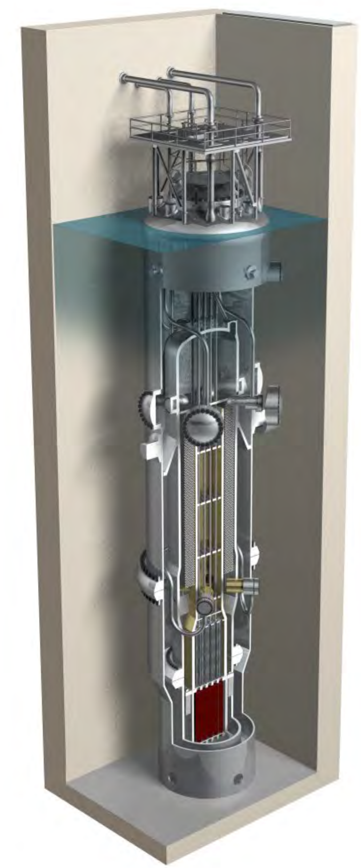
Rockville Office
11333 Woodglen Ave., Suite 205
Rockville, MD 20852
301.770.0472

Charlotte Office
2815 Coliseum Centre Drive,
Suite 230
Charlotte, NC 28217
980.349.4804

Richland Office
1933 Jadwin Ave., Suite 130
Richland, WA 99354
541.360.0500

Arlington Office
2300 Clarendon Blvd., Suite 1110
Arlington, VA 22201

London Office
1st Floor Portland House
Bressenden Place
London SW1E 5BH
United Kingdom
+44 (0) 2079 321700



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Safety Evaluation with Open Items: Section 3.9.2, “Dynamic Testing and Analysis of Systems, Structures, and Components”

NuScale Design Certification Application

ACRS Subcommittee Meeting
May 16, 2019

Agenda

- NRC Staff Review Team
- Overview
- NuScale Power Module (NPM) Level D Analysis
- Reactor Internals Comprehensive Vibration Assessment Program (CVAP)
- Abbreviations

NRC Staff Review Team

- Technical Staff
  - Yuken Wong, NRO
  - Dr. Stephen Hambric, (Consultant)
  - Dr. David Ma (Consultant)
- Project Management
  - Marieliz Vera, Project Manager
  - Greg Cranston, Lead Project Manager

Overview

- Staff reviewed Section 3.9.2 in accordance with the SRP and RGs 1.20, 1.61, and 1.122
  - RG 1.20, Rev. 3, “Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing”
  - RG 1.61, Rev. 1, “Damping Values for Seismic Design of Nuclear Power Plants”
  - RG 1.122, Rev. 1, “Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components”
- Reviewed the following areas in Section 3.9.2:
  - Dynamic system analysis of the reactor internals under service level D conditions
  - Reactor internals comprehensive vibration assessment program (CVAP)

NuScale Power Module Dynamic Analysis Under Service Level D Conditions

NuScale Power Module Dynamic Analysis and Level D Stress Evaluation

Staff's review Scope:

- DCA Part 2, Tier 2, Section 3.9.2.5, "Dynamic System Analysis of the Reactor Internals Under Service Level D Conditions"
- DCA Part 2, Tier 2, Appendix A, "Dynamic Structural Analysis of the NuScale Power Module"
- TR-0916-51502, Rev. 1, "NuScale Power Module Seismic Analysis"
  - NuScale power module (NPM) seismic analysis
  - NPM component Level D stress evaluation
    - Reactor vessel internals (RVI)
    - Steam generators (SG)
- TR-1016-51669, Rev. 0, "NuScale Power Module Short-Term Transient Analysis" – ANSYS modeling only

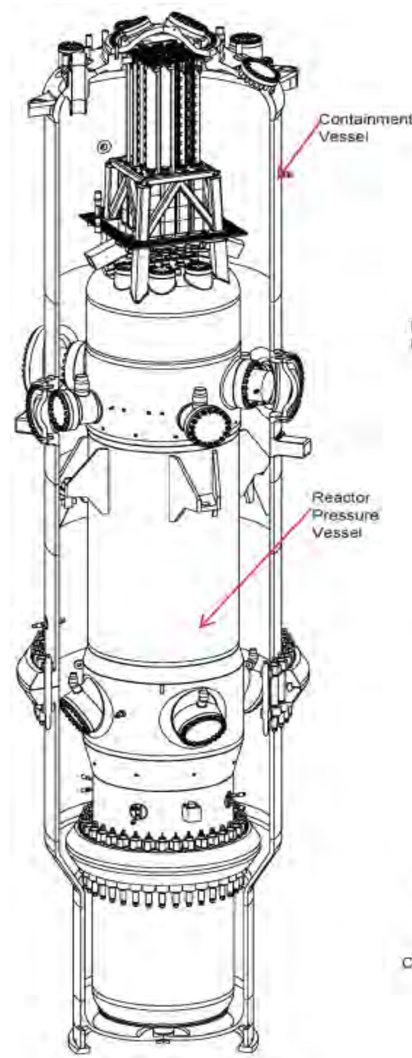
NPM Seismic Models Reviewed

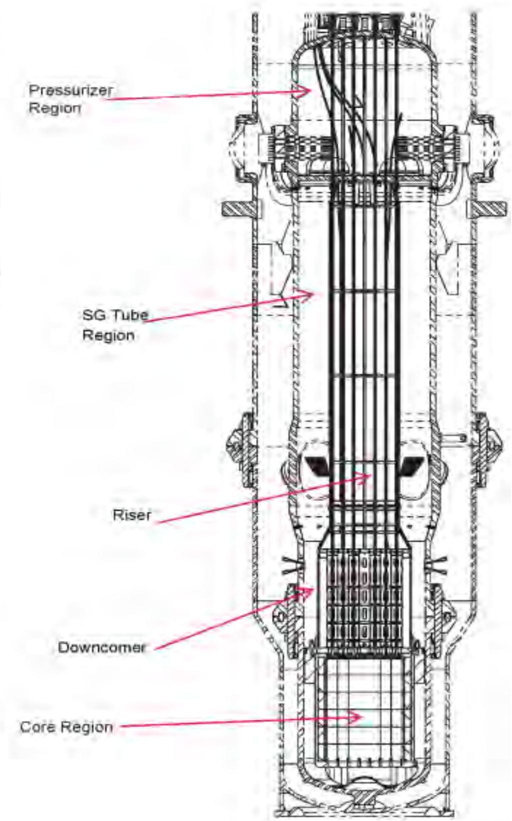
- 3D ANSYS NPM model
  - CNV submodel
  - RPV submodel
  - Lower RVI submodel
  - Upper RVI submodel
  - CRDM submodel

- Equivalent beam ANSYS model

- Equivalent beam SAP2000 model

- 3D NPM Entire Pool ANSYS Model





Review of NPM Seismic Analysis

- Resolved major issues:
  - System damping
  - Fluid gap
  - Acoustic absorbing coefficient
  - Generation of instructure response spectra (ISRS)
- Four open items:
  - NPM seismic analysis cases
  - Uplift of reflector blocks
  - Reactor flange tool (RFT)
  - RVI and SG stress evaluation

System Damping

RAI 202-8911, Question 03.09.02-46

- **Concern:** Applicant assigned 7% damping in the NPM system and component seismic analysis
  - RG 1.61, Table 6 (mechanical and electrical components) specifies 3% for pressure vessels.
  - Table 1 (Structural material) specifies 4% for welded steel or bolted steel with friction connections
  - Table 1 specifies 7% for bolted steel with bearing connections
- Applicant's justifications of using 7% damping include:
  - RVI joints are analogous to bolted steel structures with bearing connections (7% - RG 1.61, Table 1).
  - Many sliding SG tube-to-support interfaces generate large frictional dissipative forces.

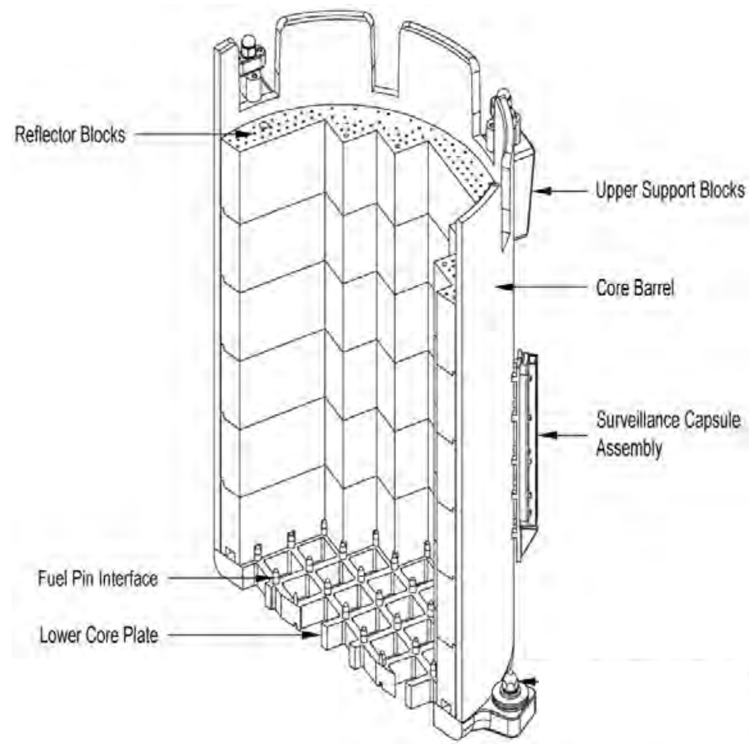
System Damping (Cont'd)

- The staff noted that there are welded structures in the RVI.
  - RG 1.61, Table 1 specifies 4%
- 7% damping may not be achievable in the SGs.
  - The steel/steel coefficient of friction is smaller underwater than that in dry condition.
- **Applicant response:** Instead of 7% damping, 4% is used for the NPM system and component seismic analysis.
- Staff finds that using 4% damping in the NPM systems and component seismic analysis is reasonable.
  - The integrated NPM with many connections and internal structures is unlike traditional shell type pressure vessels.
  - There is additional energy dissipation provided by the connections and internal structures.
- RAI 202-8911, Question 03.09.02-46 is resolved and closed.

Fluid Gap

RAI 410-9310, Question 03.09.02-70

- **Concern:** The core barrel and reflector are separated by a thin fluid gap. The fluid gap was not considered in the analysis.
- With presence of fluid gap, frequencies of the core barrel and reflector are lower due to added mass effect of the fluid gap.
- **Applicant response:** The lower RVI submodel was updated to capture the added mass effect of the fluid gap using the ANSYS Fourier node method.
- The staff finds the response acceptable and will review the updated lower RVI submodel in TR-0916-51502 Revision 2
- RAI 9310 Question 03.09.02-70 is resolved.



Acoustic Absorption Coefficient

- In the NPM seismic analysis, NuScale initially assumed 100% reflection of the acoustic wave energy at the bottom of the pool.
- Results in excessive and unreasonable amplification of NPM response.
- In reality, the acoustic wave energy is partially reflected and partially absorbed by the concrete floor and surrounding soil.

Acoustic Absorption Coefficient (cont'd)

- NuScale built two ANSYS models to estimate the acoustic absorption coefficient.
  - Model 1: Integrated model including an NPM, reactor pool water, RXB and backfill soil.
    - Applied damping to concrete and backfill to dissipates pool acoustic energy.
  - Model 2: Standard model with an NPM and reactor pool water (without the RXB and backfill). Various acoustic absorption coefficients are applied at bottom of the reactor pool to attenuate the pool acoustic energy.
- Applied a 1g vertical excitation at bottom of the pool water of the two models.
- Compared the responses at key NPM locations between the two models.
- An absorption coefficient of 0.75 produces the best match between the two models.
- The staff performed an audit on the acoustic absorption analysis on December 19, 2018.

Acoustic Absorption Coefficient (cont'd)

- During the audit, NuScale proposed to use an absorption coefficient of 0.4 in the NPM seismic analysis.
- Staff concluded that absorption coefficient of 0.4 is reasonable and conservative based on:
  - The attenuation of the acoustic energy by the building structure and surrounding soil
  - Relative acoustic impedances of concrete and water (Higher impedance ratio leads to more acoustic wave reflection and less absorption).
    - Actual impedance ratio between concrete and water is about 5.
    - Absorption coefficient of 0.4 yields impedance ratio of 7.9.
- This issue is resolved and closed.

Generation of Instructure Response Spectra

RAI 202-8911, Question 03.09.02-38

- **Concern:** TR-0916-51502-P, Rev. 0, states that ISRS at locations of equipment supports within the NPM are enveloped and broadened according to ASCE 4-13 for component design.
  - ASCE 4-13, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary"
  - ASCE 4-13 permits a 15 percent reduction of the narrow frequency peak amplitude of the ISRS if certain conditions are met.
- The staff finds that the use of ASCE 4-13 in ISRS generation is inconsistent with RG 1.122, Rev. 1 (no reduction of frequency peak amplitude).
- **Applicant response:** TR-0916-51502, Rev. 1 was updated to remove the reference to ASCE 4-13.
- The staff finds the applicant's response acceptable.
- RAI 202-8911, Question 03.09.02-38 is resolved and closed.

NPM Seismic Analysis Cases

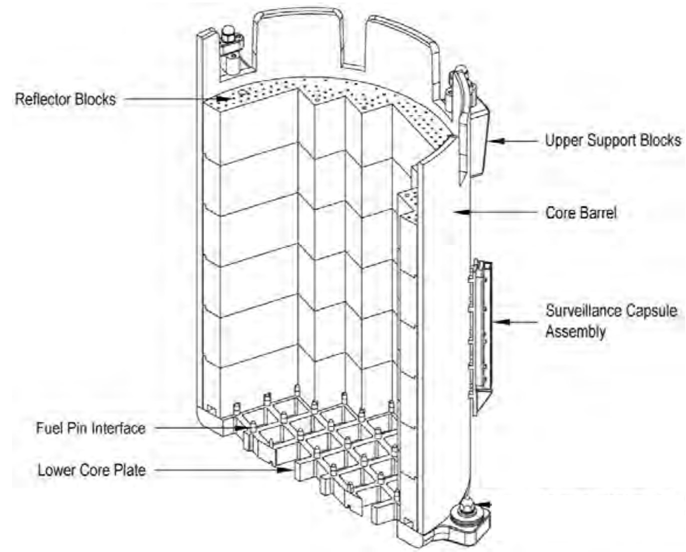
Open Item 03.09.02-8 (RAI 202-8911, Question 03.09.02-43)

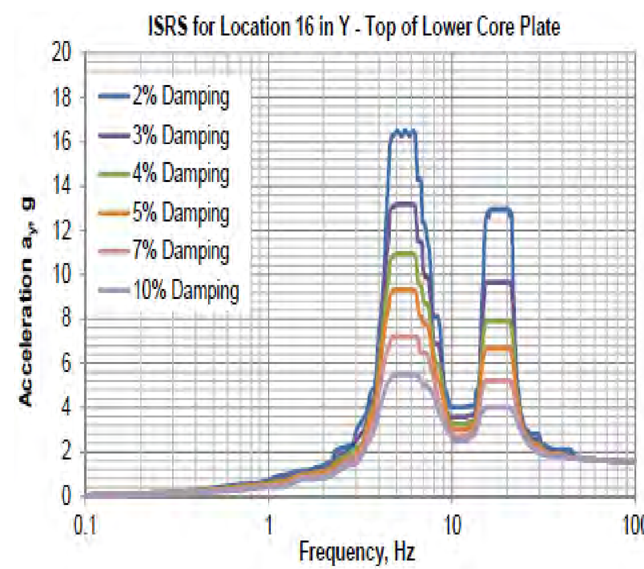
- The NPM entire pool model was analyzed for six cases with the following scenarios:
  - NPM 1 or 6
  - Uncracked or cracked concrete condition
  - Nominal or 77% nominal NPM stiffness
- **Concern:** 130% nominal NPM stiffness should also be considered to account for uncertainty in the NPM model input and assumptions.
- **Applicant response:** Performed 12 NPM runs, including 130% nominal NPM stiffness.
- Details of the updated NPM seismic analysis are documented in TR-0916-51502-P, Rev. 2.

Uplift of Reflector Blocks

Open Item 03.09.02-9 (RAI 202-8911, Question 03.09.02-45)

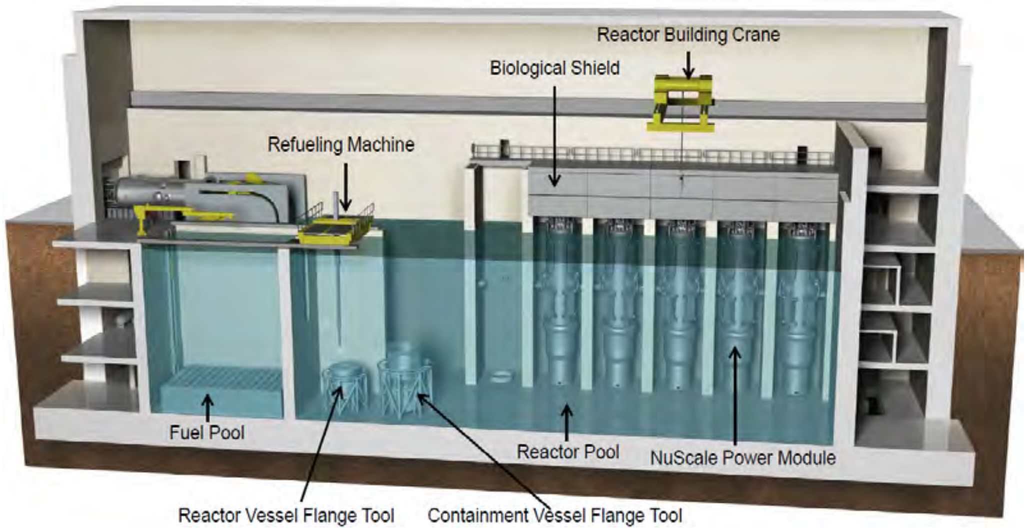
- **Concern:** The reflector blocks are stacked and not restrained in the vertical direction.
- Lower core plate vertical ISRS acceleration at high frequency end exceeds gravity acceleration.
- Uplift of the reflector blocks from the lower core plate during an SSE was not considered in the original NPM analysis.
- **Applicant response:** The 3D ANSYS NPM model was modified to include ANSYS contact elements between reflector block and lower core plate to simulate uplift of the reflector blocks.
- Staff finds response acceptable.





Reactor Flange Tool

Reactor building houses NuScale power modules, fuel pool, and reactor pool



Open Item 03.09.02-10

- The applicant introduced nonlinear contact elements at the interface between lower RPV and RFT to simulate uplift of the NPM
- Documented the results in the NPM Seismic Report TR-0916-51502, Revision 2.
- The staff is reviewing the RFT modelling in NPM Seismic Report, Rev. 2.

RVI and SG Stress Evaluation

Open Item 03.09.02-11 (RAI 202-8911 Question 03.09.02-18)

- The applicant was requested to provide the seismic analysis details and Level D stress evaluation results of major RVI and SG components.
- Applicant response:
  - Provided seismic analyses and Level D stress evaluation of the RVI and SG components in the RAI response.
  - The analyses used the ISRSs generated from the original 6 seismic runs (i.e., not considering 130% NPM stiffness cases).
  - Updated seismic analyses and Level D stress evaluation of RVI and SG components using ISRS generated from the 12 seismic runs (i.e., 130% NPM stiffness case) will be submitted as an RAI supplement.
- The staff will review the updated RVI and SG stress evaluation in the supplemental response.

Reactor Internals Comprehensive Vibration Assessment Program

Documents Evaluated

- Docked documents
  - TR-0716-50439, Rev. 1, “NuScale Comprehensive Vibration Assessment Program Technical Report” (ML18022A221)
  - TR-0918-60894, Rev. 0, “NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report” (ML18341A337)
- Undocketed documents
  - Audit 1: May, 16 – November 2, 2017 (ML18023A091)
  - Audit 2: September 5 – October 4, 2018 (ML18333A221)
  - Audit 3: 1 March 2019 – TBD

FIV Mechanisms Evaluated

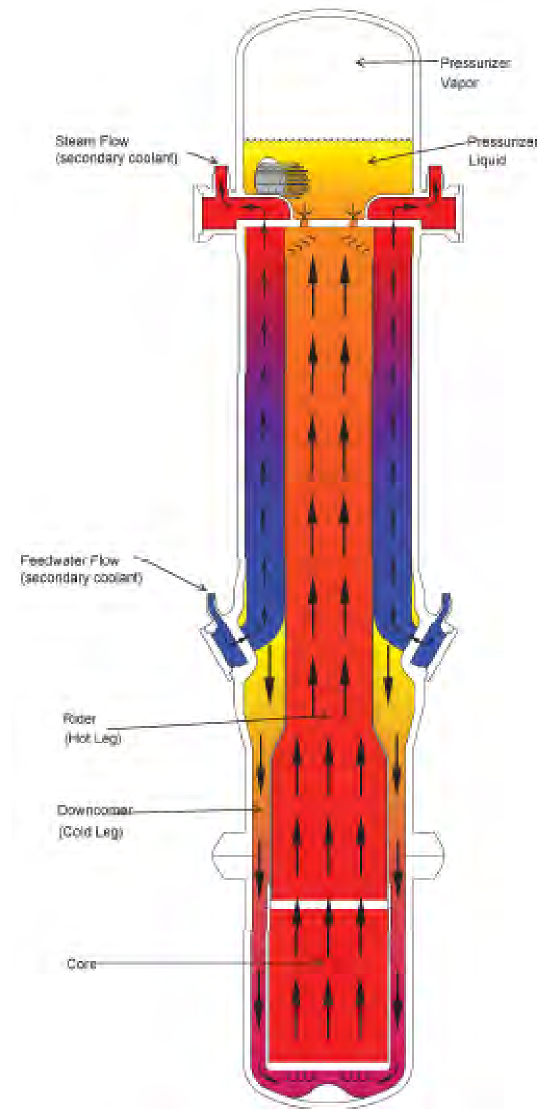
- Turbulence buffeting (TB)
- Vortex shedding (VS) lock-in with structural resonances
- Fluid-elastic instability (FEI)
- Acoustic resonance (AR)
- Leakage flow instability (LFI), and flutter and galloping (F/G)

Components Evaluated

- Helical Coil Steam generator (HCSG) tubes & supports
- SG inlet flow restrictors (SGIFR)
- Control rod drive shaft (CRDS)
- In-core instrument guide tube (ICIGT)
- Primary and secondary coolant piping up to the NPM disconnect flanges including the isolation valves.
- CRAGT and CRAGT support
- CRDS support
- Control rod assembly (CRA) card
- Upper core plate and support
- Pressurizer spray nozzle
- SG steam plenum
- Core barrel
- Lower core plate
- Lower riser
- Reflector Blocks
- Upper rise and hanger
- CVS Injection line
- Flow diverter

Risk Areas

- Natural circulation, low power operating levels, flow rates 5-25 times lower than those in typical PWRs
  - TB loads are much lower
  - Components normally evaluated for TB are not susceptible to damage in NuScale design
- However, some structural components may be susceptible to FIV in spite of lower flow velocities
  - Long, thin rods and tubes
    - HCSG
    - ICIGT
    - CRDS
- AR also possible in DHRS piping side branches
- LFI unlikely in most regions due to low pressure differences
  - Exception: SG inlet flow restrictors designed and evaluated by testing



NuScale's Screening Studies

- Methodologies from the following sources
  - ASME B&PV Code III Appendix N-1300 (Flow induced vibration of tubes)
  - Blevins, Flow-induced vibration, 2nd edition
  - Au-Yang, Flow-induced vibration of power and process plant components
- Computational fluid dynamics (CFD) analyses of flow velocities
  - Simplified model to support thermal hydraulic analyses; structural details (including steam generator and core) not included
- Finite element (FE) models of individual components with simplified boundary conditions to estimate lowest resonance frequencies
- Damping assumptions vary, ranging from 0.5% to 1.5%
  - Per RG 1.20 damping above 1% needs rigorous substantiation because higher damping will artificially increase margin

Low Margins of Safety

- HCSG tubing
  - <10% margin against FEI
  - <20% margin against VS
- RVI
  - CRDS < 25% margin against VS
  - ICIGT < 25% margin against VS
- DHRS piping ~ 20% margin primary AR
  - No margin for secondary AR
- SG inlet flow restrictor – low, but unquantified LFI risk
  - Design chosen based on testing of several concepts
- Issue: there are non-conservatisms in the FIV analyses that may outweigh the conservatisms
  - Currently being addressed by NuScale via RAIs

General Analysis Concerns

- Flow modeling
  - Non-conservative flow velocities assumed
    - Empirical forcing function models call for peak (free stream) velocities, not the average velocities used by NuScale
    - Spatial variability possible, ignored so far
- Structural modeling
  - Coarse meshing (biases resonance frequencies high)
  - Idealized pinned boundary conditions at interfaces with non-negligible clearances, fluid loading must be conservative
- If VS does occur, forced response analyses are needed to assess possible impacting and wear
  - Stress amplifications/weld factors not included
  - Average crossing frequency used for impact and wear assessments sometimes lower than first structural resonance frequency

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General Testing Concerns

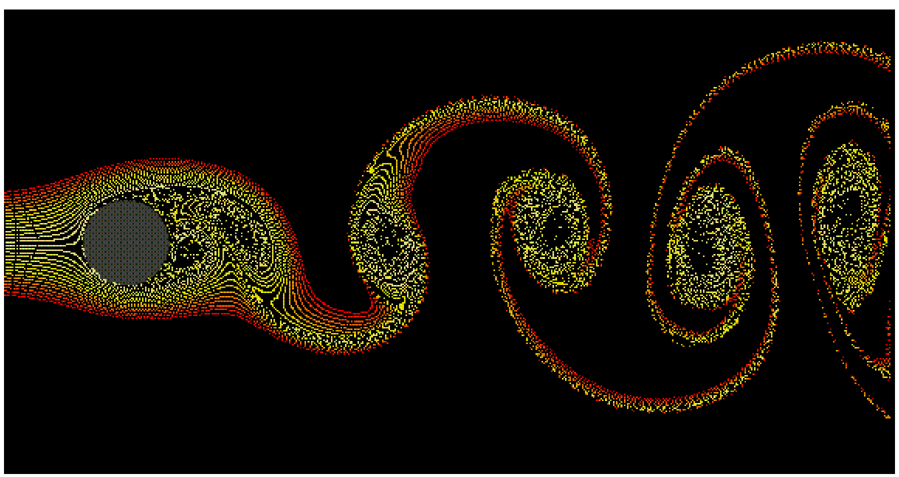
- Less benchmarking and testing than usual for a DCD
  - Unique new design with no operating history or similar predecessor
  - Pre-operational FIV testing infeasible due to natural circulation design
  - However, turbulent buffeting benchmarking unnecessary due to very low flow rates
- Staff proposed the following for reasonable assurance against significant FIV
  - Preliminary/validation testing focuses on key FIV mechanisms with low margins of safety and high uncertainty
  - Initial startup testing will focus on identifying any unexpectedly high FIV
    - Not intended for specific mechanisms or benchmarking

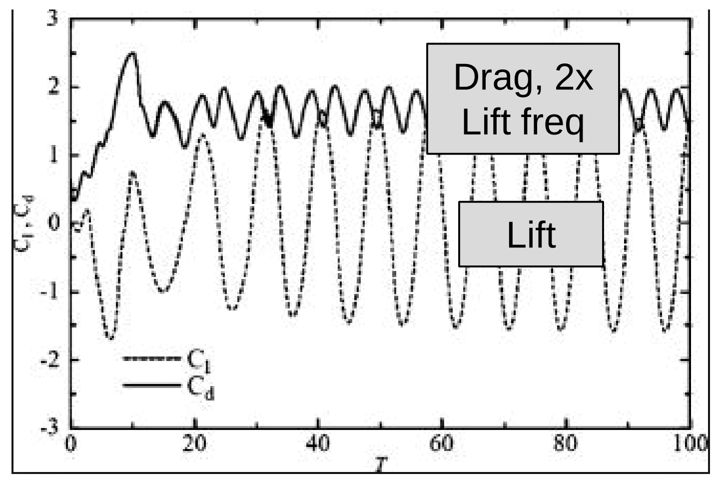
HCSG Tubing and RVI Vortex Shedding and Lock-In

Open Items 03.09.02-01, 02, 03

Vortex Shedding/Lock-In

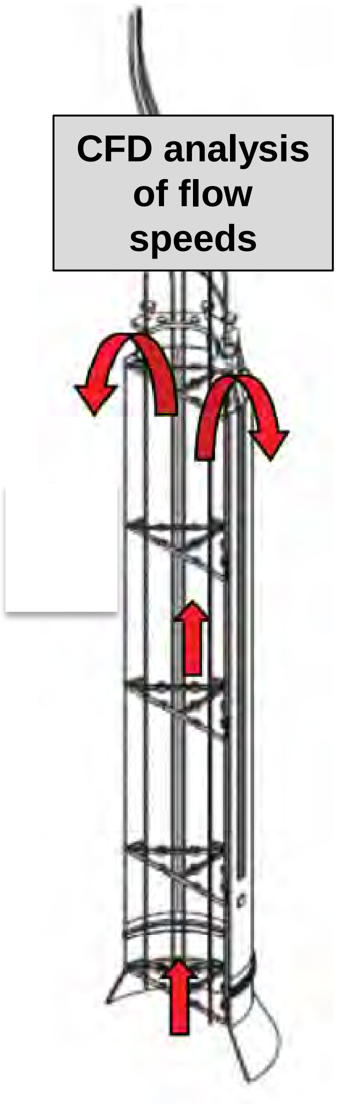
- Occurs when:
  - Vortex shedding (VS) frequency aligns with a structural resonance, and
  - Impedance of the structural resonance is low
    - Low damping, low modal mass (‘Damping Parameter’)
- Maximum VS frequency for NuScale cylinders: St = fD/U ~ 0.25
- At lock-in, vibration amplitude can exceed cylinder diameter
  - CRD shafts and ICIGTs would strongly and repeatedly impact supports
- ASME N-1324.1 provides criteria (a) – (d) for lock-in avoidance

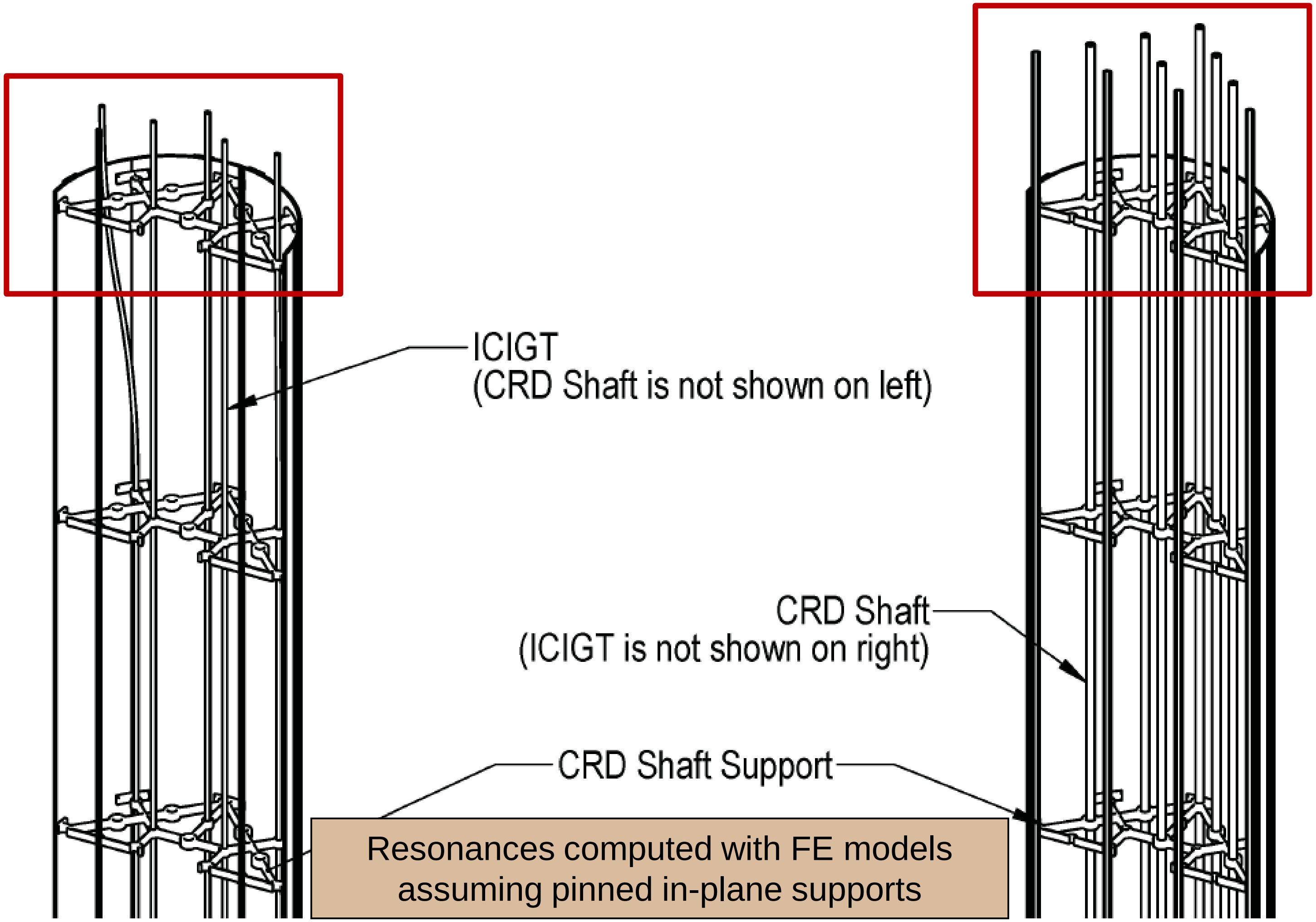




CRDS and ICIGT VS (OI 03.09.02-03)

Top sections of ICIGT and CRD Shafts exposed to cross flow; vortex shedding/lock-in is possible





Resonances computed with FE models assuming pinned in-plane supports

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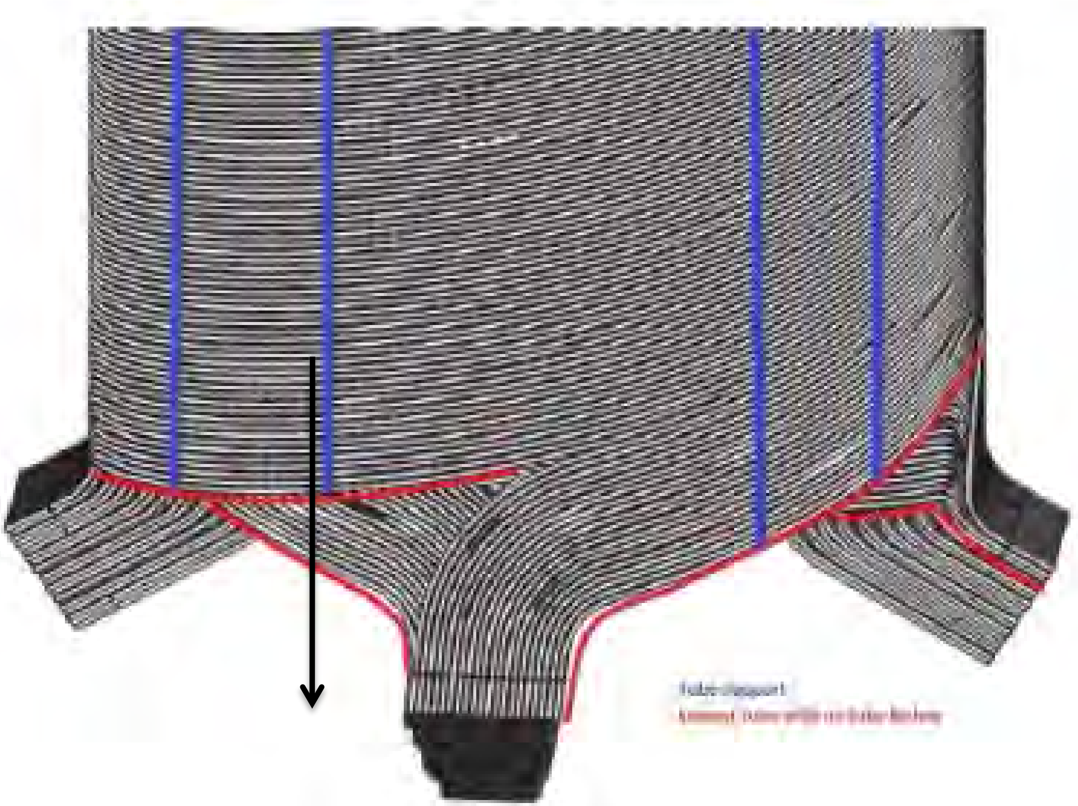
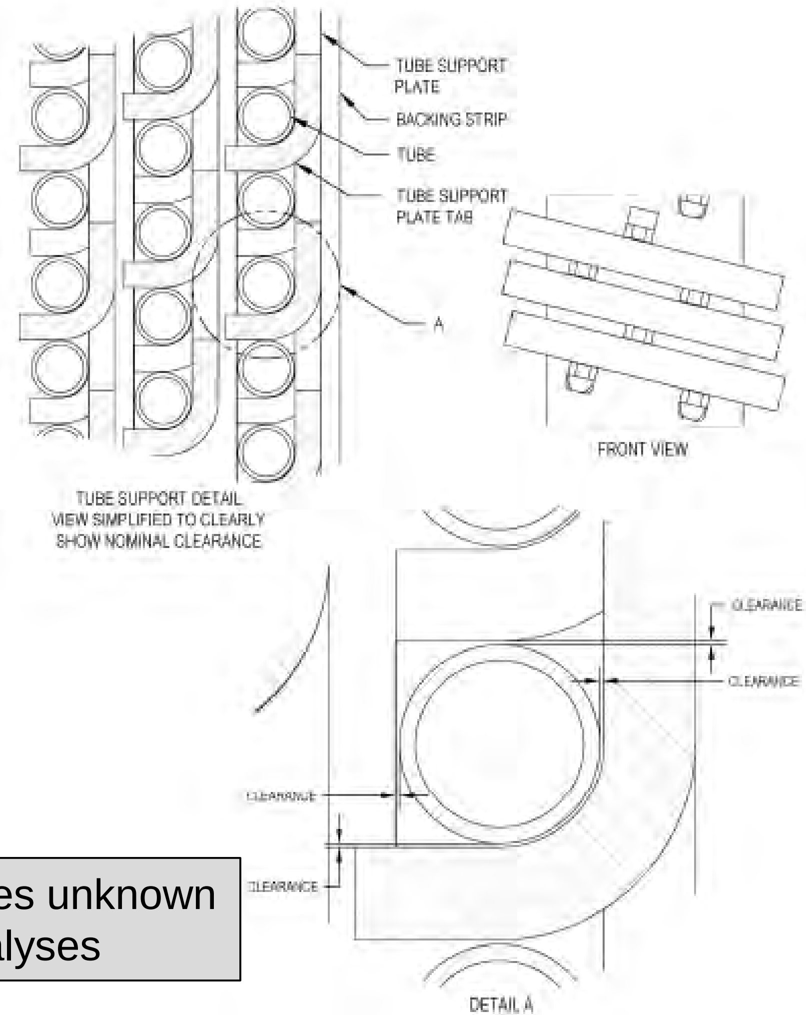
HCSG Lower Tubing VS (OI 03.09.02-01)

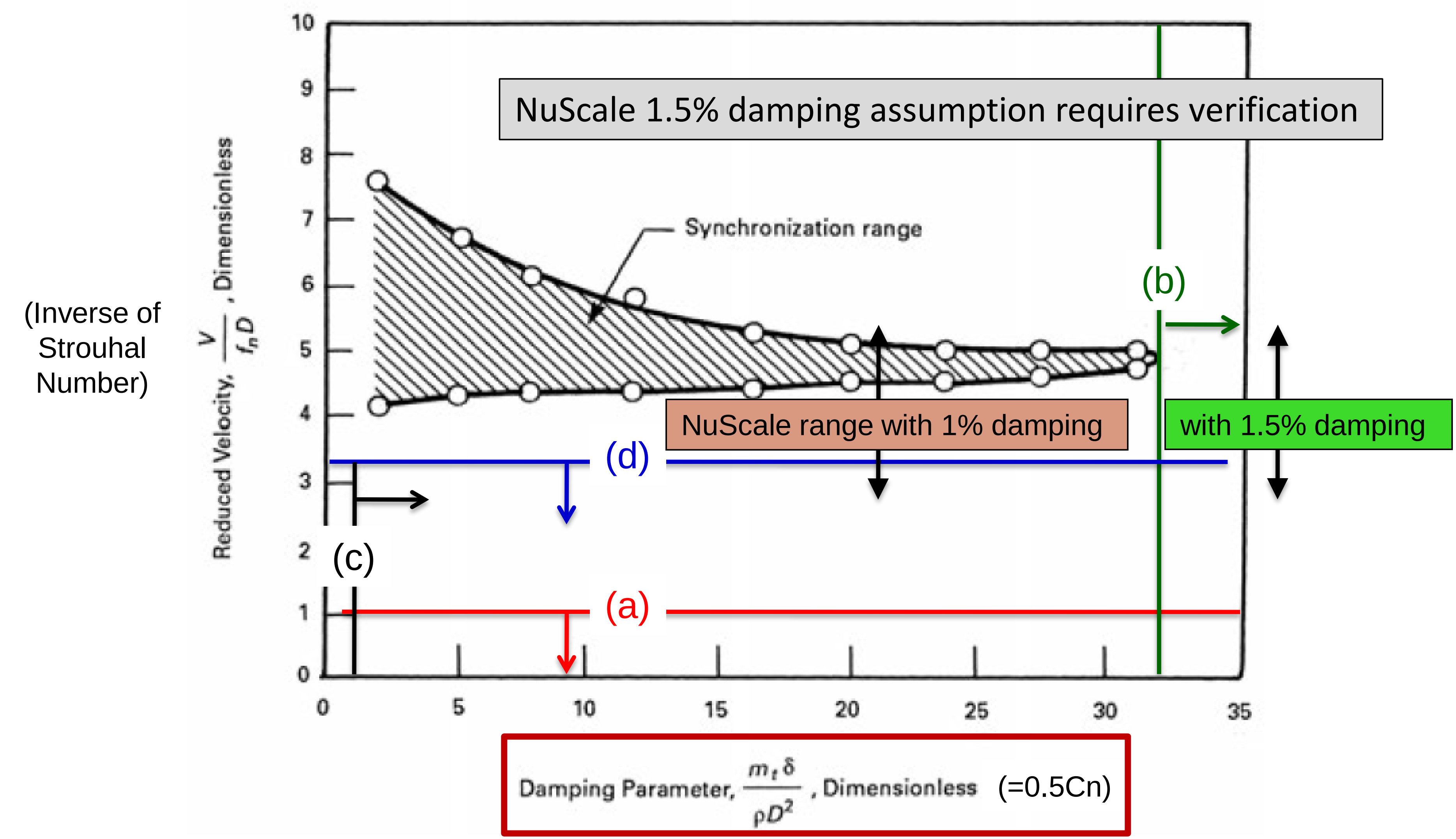
Bottom SG tubes exposed to cross flow, vortex shedding/lock-in is possible





0.009” nominal clearances specified, operating clearances unknown
NuScale assumes <u>fully active supports</u> for FIV analyses

ASME VS/Lock-In Avoidance Criteria



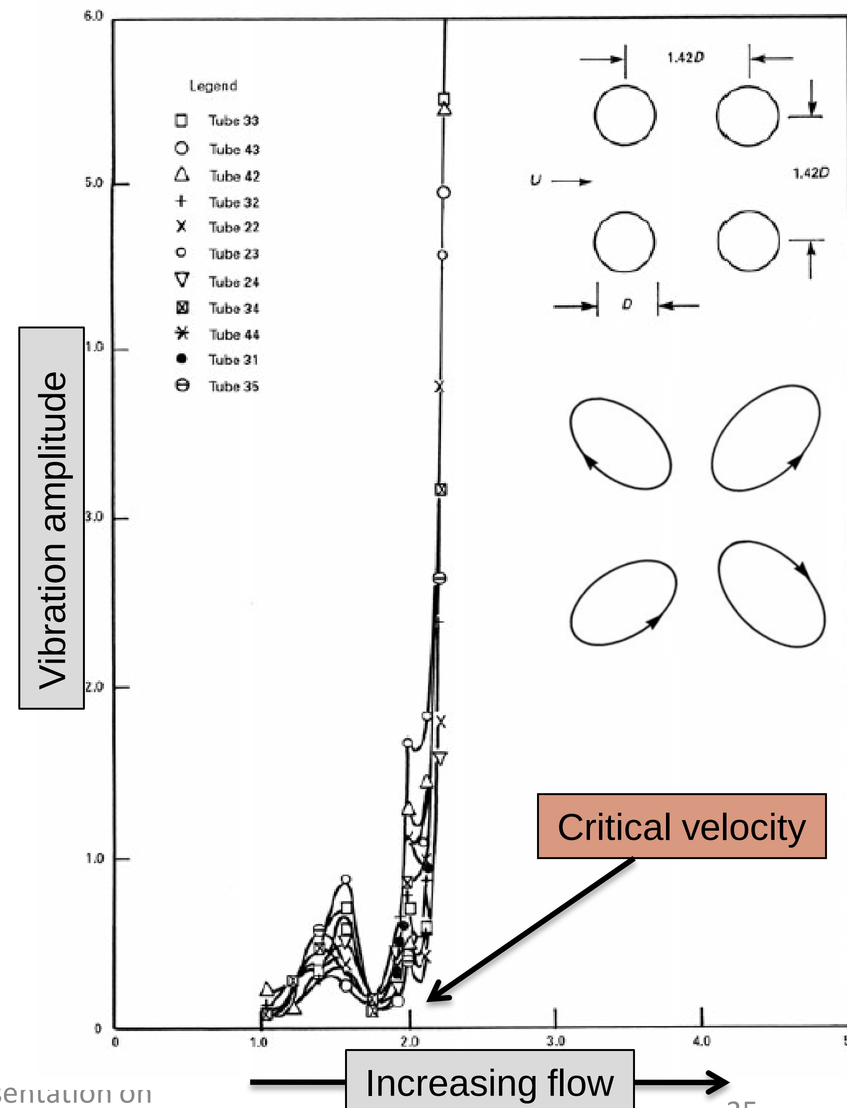
HCSG Tubing Fluid Elastic Instability

Open Items 03.09.02-1 and 03.09.02-2

HCSG Fluid-Elastic Instability

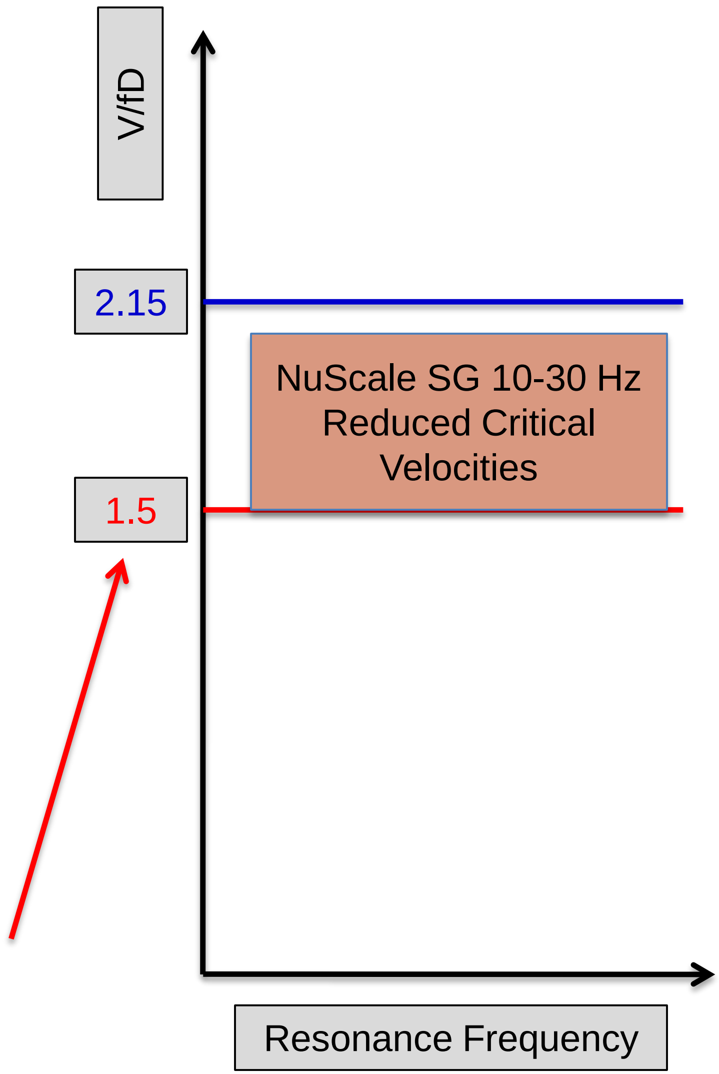
- Occurs when:
  - Vortex field around multiple tubes couples constructively with structural motion
  - Flow velocity exceeds critical value
    - Extremely high displacements, usually limited by contact with nearby tubes
- Estimate reduced critical velocity via empirical fit with two constants C and a to measured data for different tube array types
  - Critical velocity depends on mass-damping

$$V_c / f_n D = C\left[m_t\left(2\pi\xi_n\right) / \rho D^2\right]^a$$



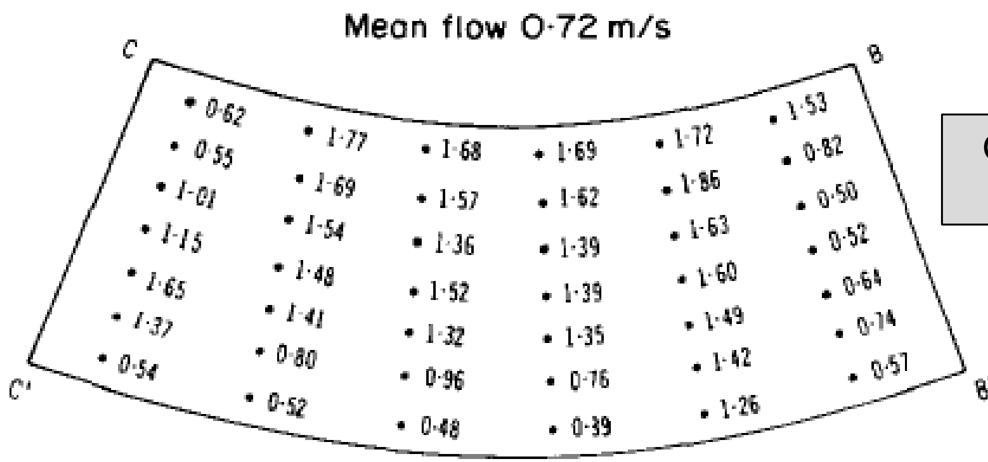
HCSG Fluid-Elastic Instability

- NuScale computes critical velocities for first 50 modes of HCSG tubes
  - Estimates resonance frequencies and mass-damping of each mode
  - Considers Connors (straight tube array) and Chen, 1983 (helical array) empirical constants
- Critical velocities are compared to gap velocities estimated from CFD bulk flow and blockage of tubes
- HCSG reduced critical velocities (V/fD) with fully active supports range from ~ 1.5 – 2 for modes with resonance frequencies less than 30 Hz
  - Connors reduced critical velocity is 2.15, so small margin exists
  - Actual reduced critical velocity should be around 1.5 per Chen; tubes may experience onset of FEI between 10 and 30 Hz



HCSG Fluid-Elastic Instability

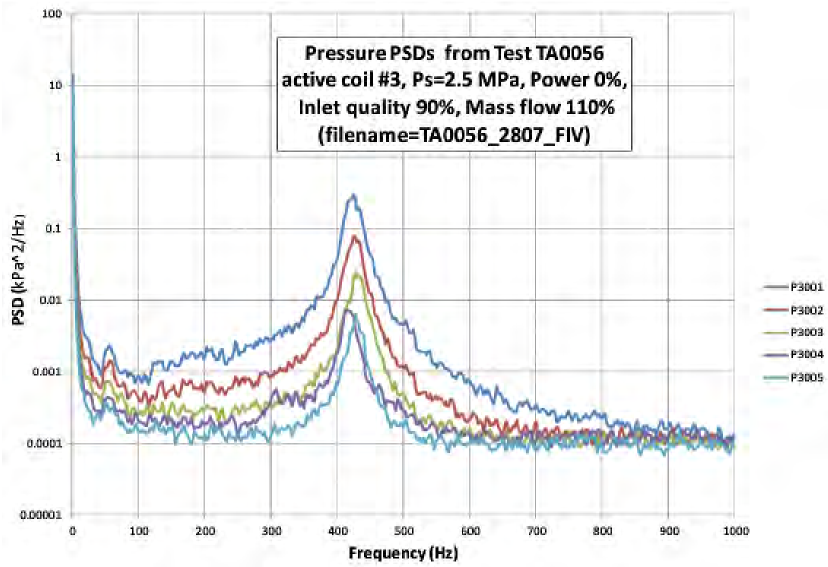
- If any supports are inactive, longer sections means lower resonance frequencies, so more modes will be susceptible to FEI
- Non-conservatisms:
  - SG tube damping is assumed to be 1.5%, greater than RG 1.20 specified 1%
  - No mesh convergence studies, biasing resonance frequencies higher
  - Lower (non-conservative) average velocity through region used for analyses with partially inactive supports
  - Actual flow velocities will vary throughout the annulus; regional velocities may be higher



Chen, JSV, 91 (4), 539-569, 1983

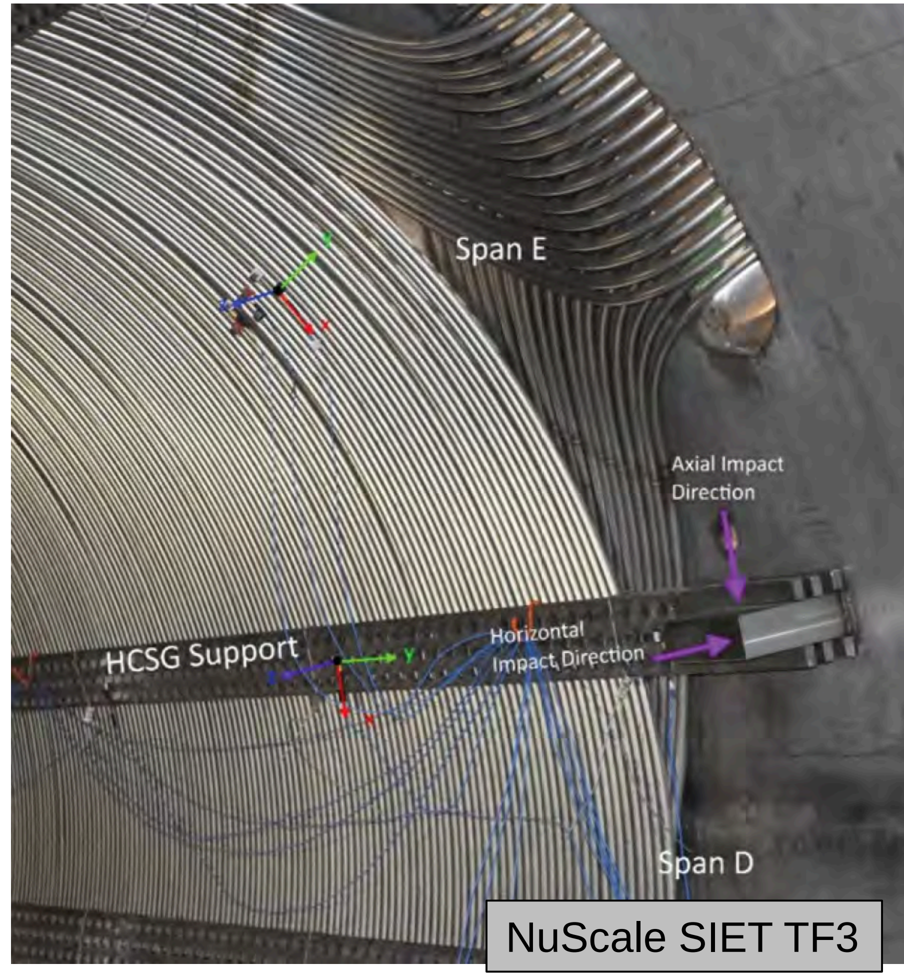
HCSG SIET FIV Testing

- Previous HCSG tests primarily for thermal-hydraulic assessments, but limited FIV data also acquired
- TF-1 – single tube internal flow
  - Unexpectedly high internal loading peak, not yet accounted for in FIV analyses
- TF-2 – tube arrays, but with limited instrumentation and non-prototypic supports
  - Encouraging data show no indications of FEI (or VS)
- Due to low VS and FEI margins, <u>TF-3 Design Validation Testing Underway</u>



HCSG SIET FIV Testing

- Prototypic construction
- Heavily instrumented
- Modal dynamic tests (near term, will support final SER)
  - Staff goal: on-site audit of testing and results to ensure:
    - Structural boundary condition modeling is appropriate
    - Structural damping assumptions are substantiated
- Flow tests (unknown time frame, will likely not support final SER)
  - Staff goal: ensure flow test procedures are sufficiently robust to provide reasonable assurance any VS/lock-in or FEI will be found and mitigated
  - Decision for TF-3 results for DC review

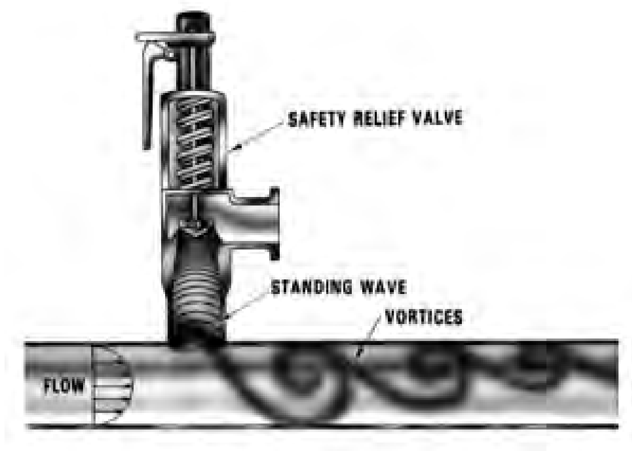


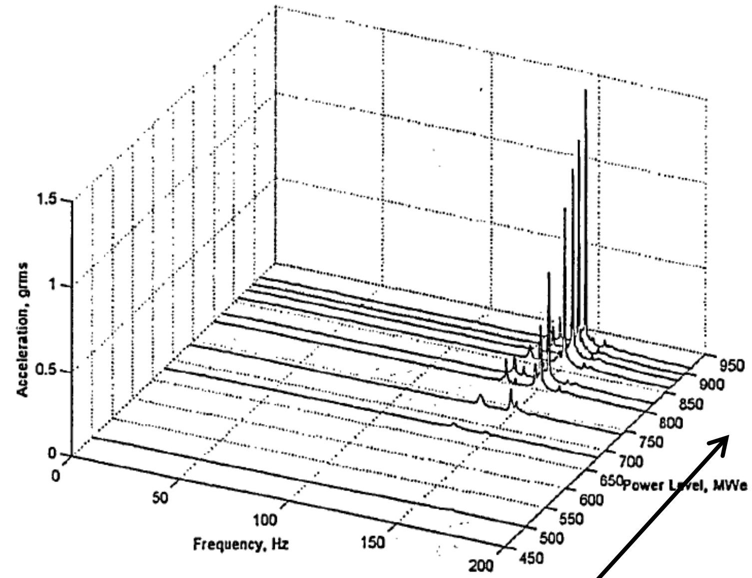
DHRS Piping Acoustic Resonances

Open Item 03.09.02-4

Piping Acoustic Resonances

- Occur when:
  - A flow instability over a cavity opening locks into an acoustic resonance (usually within the cavity)
- Instabilities occur in harmonics
  - Primary (and strongest): half wavelength across cavity
  - Secondary (weaker): full wavelength across cavity
    - Occurs at half the flow speed of primary
  - Higher harmonics generally too weak to lock-in
- Resulting acoustic pressures can be severe

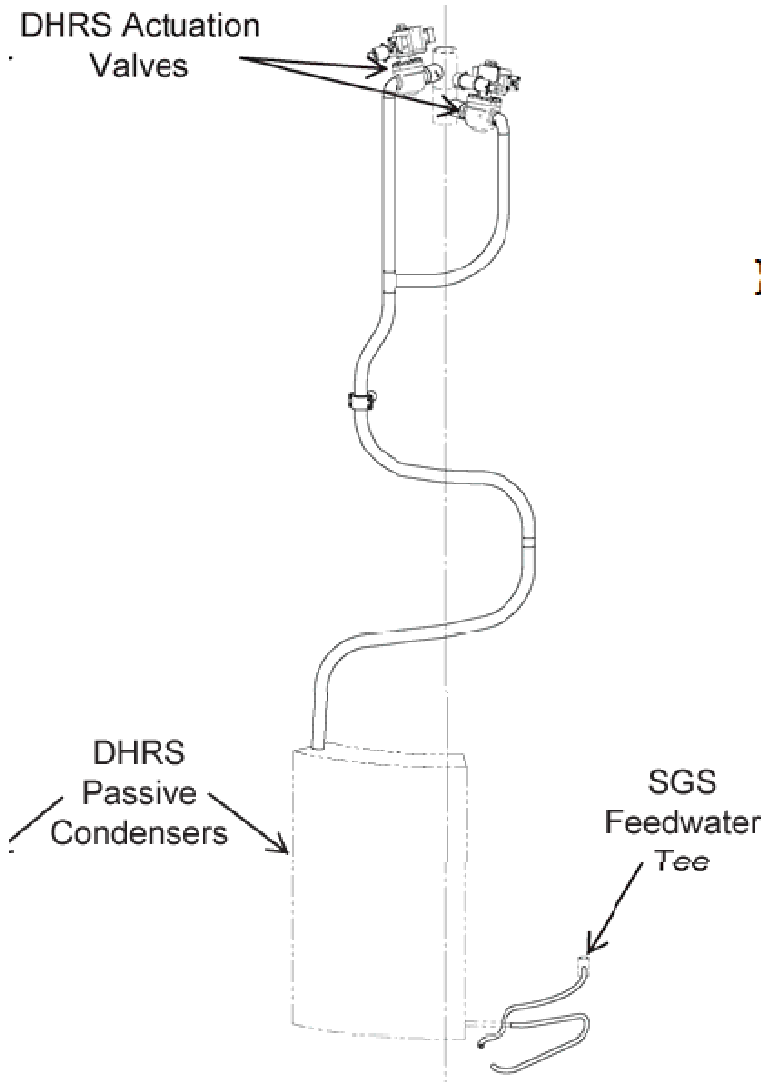




Increasing flow speed

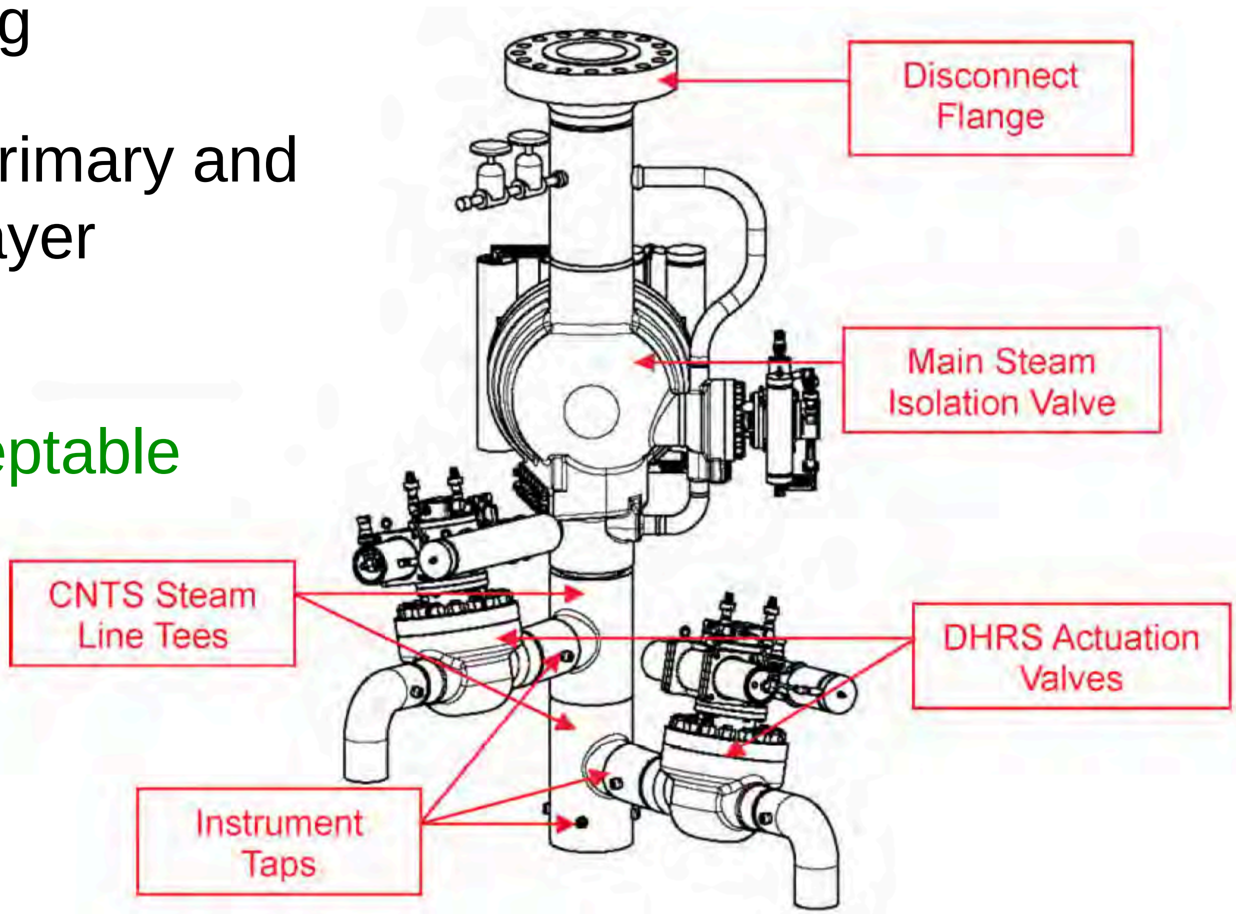
DHRS Piping Acoustic Resonances

- Per test data in open literature, AR is possible if $0.35 < fD/U < 0.62$
  - f is the acoustic cavity resonance frequencies
  - D is the cavity diameter
  - U is the flow rate
- Two possible AR locations:
  - Closed side branch from containment system steam tee to the DHRS actuation valve
    - fD/U=0.75, ~ 20% margin
  - DHRS condensate line from SG system feedwater tee to DHRS passive condenser
    - fD/U=1.33, high margin against primary instability



DHRS Piping Acoustic Resonances

- Proposed instrumentation for initial startup testing
- Will monitor both primary and secondary shear layer instabilities
- RAI response acceptable



Leakage Flow Instability (LFI)

Open Items 03.09.02-05 and 03.09.02-06

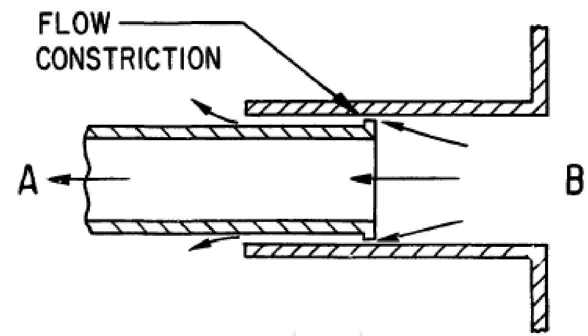
Leakage Flow Instability

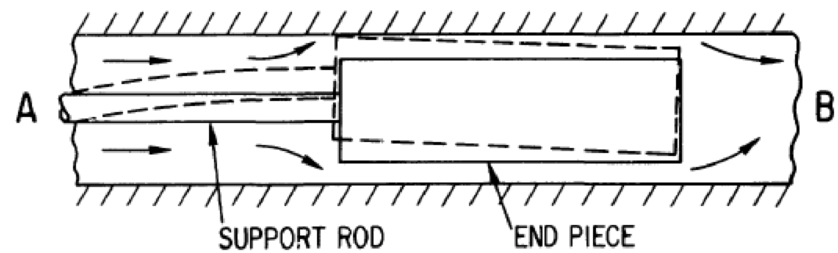
- Occurs in:
  - Narrow gaps between components with significant pressure drops
    - Control rods in guide tubes
    - Inlet flow restrictors

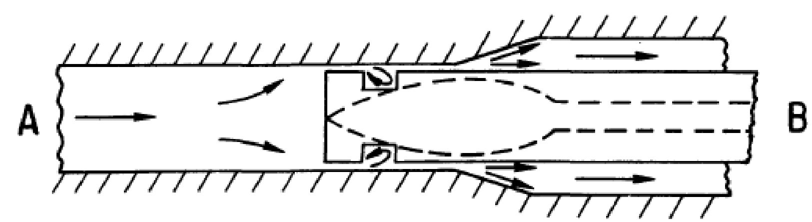
- Damage occurs when flow instability locks in to structural resonance(s)
  - <u>Occurs even with low flow rates</u>

- Has been significant in existing reactors
  - e.g., control rods in guide tubes

- Guidance from ANL report (Mulcahy, ANL-83-43)
  - Qualitative: obstructions should be downstream
  - Quantitative: every situation unique, requires measurements to assess risk

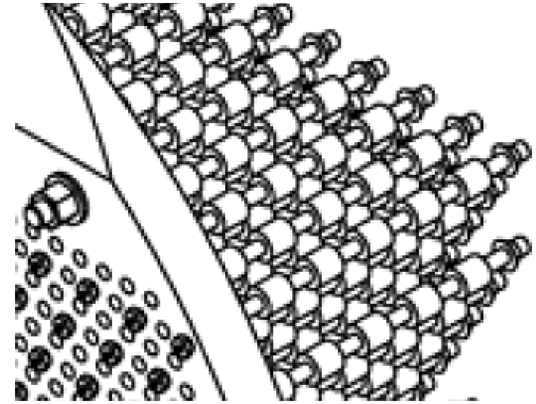


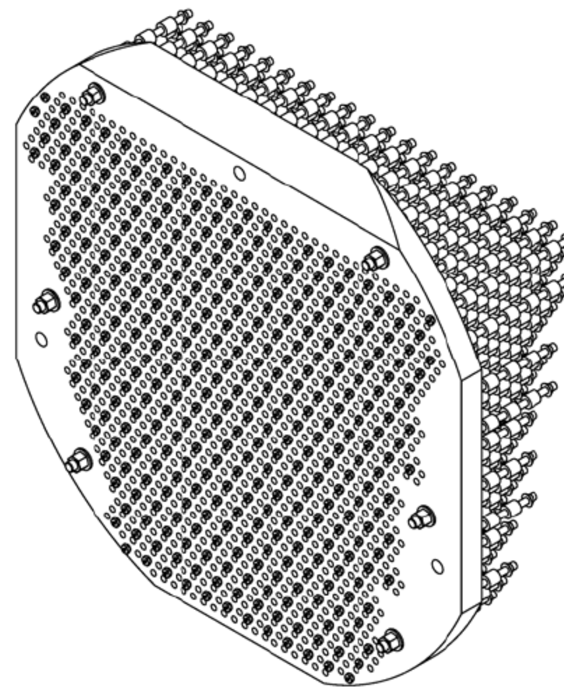


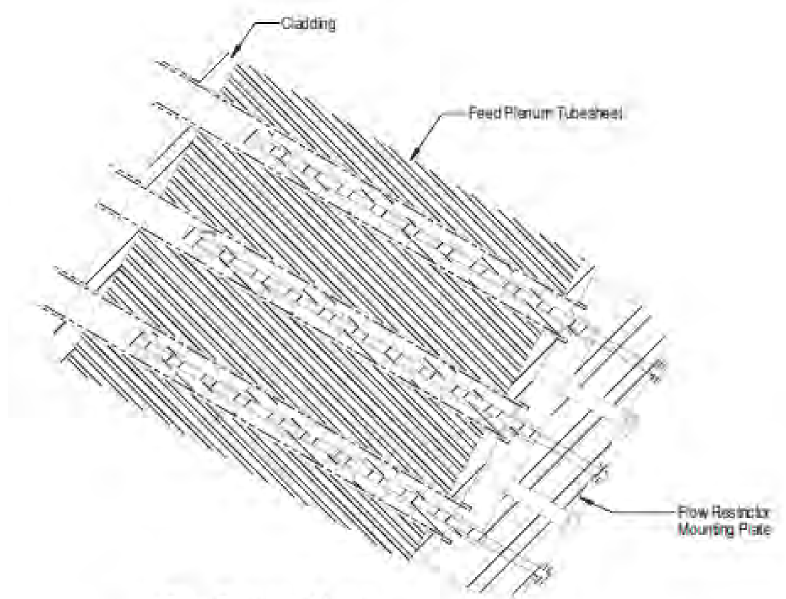


Leakage Flow Instability

- Individual flow restrictors will be placed within each SG tube inlet
  - Purpose: mitigate another instability: Density Wave Oscillation (DWO) within SG tubes
- No analysis of LFI mechanisms
- However, multiple designs tested by NuScale



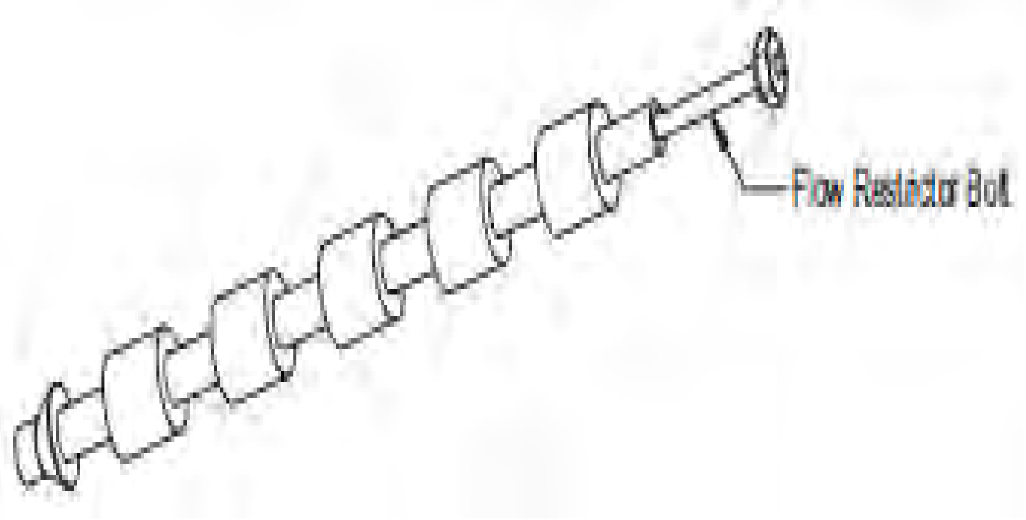




Flow Restrictor Mounting

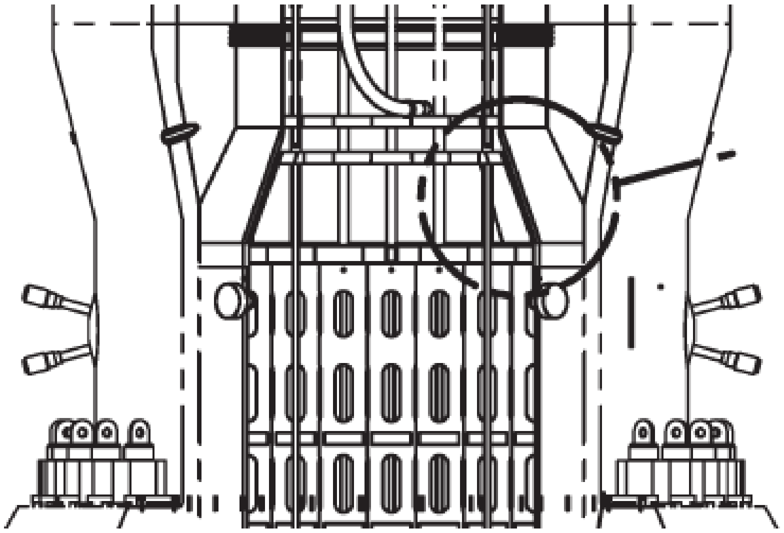
Leakage Flow Instability

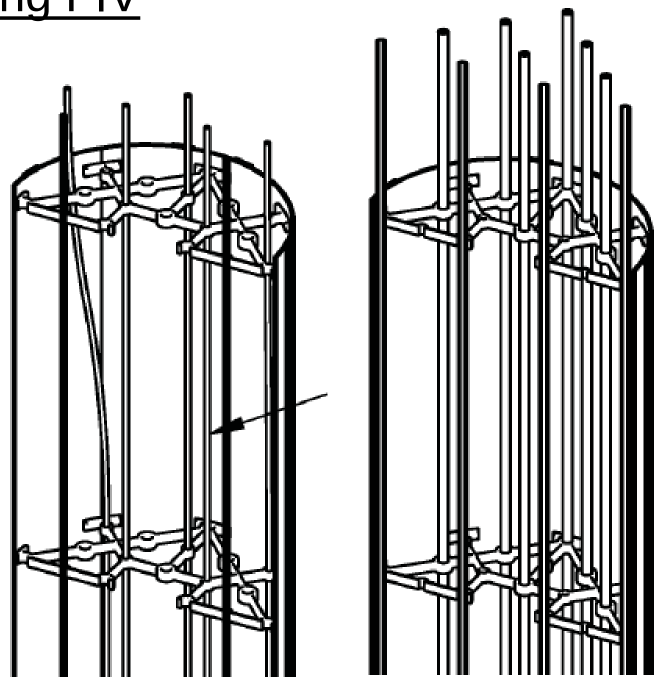
- Design least susceptible to LFI chosen
  - Test results show no sign of LFI or any other significant FIV
  - Minor changes made to final design
- <u>Design Validation testing planned prior to initial startup in separate facility</u>
  - Test plan provided in MIP, reviewed by staff and found to be reasonable
  - Lessons learned from design tests will be applied to validation testing
  - Report with results will be issued to the NRC after Design Certification – May impose license condition or COL item

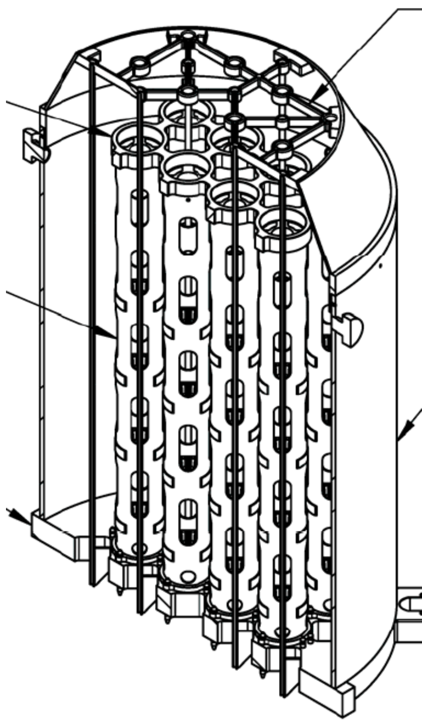


LFI – Other Components

- Other NuScale RVI with potential leakage flow paths indicated to have very low pressure differentials
  – Analyses are ongoing for CRDS, ICIGT, and CRAGT flow gaps
  – Hold down forces on upper/lower riser tapered joint preclude any gap flow
  – <u>Instrumentation being developed for initial startup testing to detect any unexpected LFI, or other strong FIV</u>







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Protecting People and the Environment

Planned Measurements and Inspections

Open Item 03.09.02-07

Measurements

- Described in MIP
- Awaiting final SIET TF-3 flow testing plan
  – Ensure sufficient rigor to provide reasonable assurance there will be no significant VS and/or FEI in HCSG in prototype and subsequent plants
- Awaiting instrumentation and pre-test predictions for initial startup testing
  – Limited in scope due to low risk of significant FIV
  – Goal: capture and localize any unexpectedly high FIV so that it may be mitigated

Inspections

- Components evaluated in the analysis program are inspected
- Components most susceptible to FIV are examined in limiting and representative locations
- Visual inspections are performed using VT-1 and VT-3 per ASME B&PV Code Section XI
- Staff finds the inspection methods and areas are consistent with RG 1.20

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Summary of Open Items

- **03.09.02-01 (RAI 427-9408, Question 03.09.02-74, HCSG)**
  - Address design concerns for HCSG tube supports, cantilevers, and welds
  - Analysis concerns
    - Non-conservative flow velocities used for HCSG FIV analyses
    - Provide FE mesh convergence studies
    - Justify idealized structural boundary conditions
    - Substantiate assumed damping higher than 1% (associated with tube boundary conditions and assumed tightness of fit)
    - Clarify if simplified random analysis methods used
  - Analysis results
    - Justify average crossing frequencies much lower than fundamental resonance frequencies
    - Provide VS forced response/fatigue calculations
  - SIET Testing
    - Account for strong spectral peaks in secondary flow pressures in TF-1 measurements
    - Address resonance peaks in TF-2 measurements
    - Provide TF-3 testing plans

Summary of Open Items

- **03.09.02-02 (RAI 386-9316, Question 03.09.02-52 Modeling procedures and SIET TF1 and TF2)**
  - Validate FE modeling procedures, perhaps using SIET TF1 and/or TF2 measurements
- **03.09.02-03 (RAI 427-9408, Question 03.09.02-73, CRDS and ICIGT)**
  - Lack of testing to assess possible VS/lock-in
  - Non-conservative flow velocities used for ICIGT and CRDS FIV analyses
  - Lack of mesh convergence studies
  - Justification of idealized boundary conditions
  - Clarify if simplified random analysis methods used
  - Average crossing frequencies much lower than fundamental resonance frequencies
  - VS forced response/fatigue calculations not provided
  - Provide test plan for factory testing

Summary of Open Items

- **03.09.02-04 (RAI 386-9316, Question 03.09.02-54 – AR)**
  - Assess possibility of AR due to second order shear instabilities (NOW RESOLVED)
- **03.09.02-05 (RAI 386-9316, Question 03.09.02-55 – SGIFR LFI)**
  - Provide test plan for final SGIFR testing (NOW CLOSED) – May impose COL item or license condition to review final design test results
- **03.09.02-06 (RAI 427-9408, Question 03.09.02-76 – RVI LFI)**
  - Provide details of LFI screening of RVI
- **03.09.02-07 (RAI 427-9408, Question 03.09.02-77 – Initial startup testing)**
  - Provide expected vibration levels, complete instrumentation, specifications, final test conditions, pretest predicted vibration and pressure levels, and acceptance criteria

Next Steps

- On-site audit of HCSG SIET TF-3 build-up and dynamic testing
  – Summer 2019
- Evaluation/closure of RAI responses
- Review of updated CVAP
  – Will include revised FIV margins of safety
- Review of updated MIP
  – Will include finalized test plans:
    - SIET TF-3
    - SGIFR
    - Initial Startup Testing instrumentation and pre-test predictions

Abbreviations

ACRS - *Advisory Committee on Reactor Safeguards*

AR - *Acoustic Resonance*

ASCE - *American Society of Civil Engineers*

CRDS – *Control Rod Drive Shaft/System*

CVAP – Comprehensive Vibration Analysis Plan

CRAGT - *Control Rod Assembly Guide Tube*

COL - *Combined License*

DC - *Design Certification*

DCA - *Design Certification Application*

DHRS – *Decay Heat Removal System*

ECCS – *Emergency Core Cooling System*

FE - *Finite Element*

FEI - *Fluid-Elastic Instability*

FIV - *Flow-Induced Vibration*

Abbreviations (cont'd)

- HCSG - *Helical Coil Steam Generator*
- ICIGT – *In Core Instrumentation Guide Tube*
- ISRS - *Instructure Response Spectra*
- LFI – *Leakage Flow Instability*
- MIP – *Measurement and Inspection Plan*
- RG - *Regulatory Guide*
- RXB - *Reactor Building*
- RVI – *Reactor Vessel Internals*
- SER - *Safety Evaluation Report*
- SG - *Steam generator*
- SGIFR – *Steam Generator Inlet Flow Restrictor*
- SRV - *Safety Relief Valve*
- SSE - *Safe Shutdown Earthquake*
- TB – *Turbulent Buffeting*
- VS – *Vortex Shedding*