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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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OPEN SESSION

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TUESDAY

JUNE 18, 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., Peter Riccardella and Jose March-Leuba, Co-Chairs, presiding.

COMMITTEE MEMBERS:

PETER RICCARDELLA, Co-Chair

JOSE MARCH-LEUBA, Co-Chair

RONALD G. BALLINGER, Member

DENNIS BLEY, Member

CHARLES H. BROWN, JR. Member

MICHAEL L. CORRADINI, Member

VESNA B. DIMITRIJEVIC, Member

JOY L. REMPE, Member

GORDON R. SKILLMAN, Member

MATTHEW W. SUNSERI, Member

ACRS CONSULTANT:

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

MIKE SNODDERLY

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P R O C E E D I N G S

8:29 a.m.

MEMBER CORRADINI: Okay, why don't we have the meeting come to order. This is a meeting of the ACRS NuScale Subcommittee. My name is Mike Corradini, Chairman of the NuScale Subcommittee. Members currently in attendance today are Vesna Dimitrijevic, Dennis Bley, Charlie Brown, Jose March-Leuba, Joy Rempe, Matt Sunseri, Pete Riccardella, Gordon Skillman and Ron Ballinger, and our consultant, Dr. Steve Schultz. I did.

The Subcommittee will review the staff's evaluation of Chapter 3, Design of SSCs, Chapter 6, Engineered Safety Features, and Chapter 15, Transient Accident Analysis of the NuScale Design Certification Application and NuScale's Topical Report, TRO-51649417, Evaluation and Methodology of Stability Analysis.

Today, we have members of the NRC staff and NuScale to brief the Subcommittee. The ACRS was established by statute and is governed by the Federal Advisory Committee Act or FACA. That means that the committee can only speak through its published letter reports. We hold meetings to gather information to support our deliberations, and interested parties who wish to provide comments can contact our office requesting time after the meeting announcement is published in the *Federal Register*.

That said, we set aside ten minutes for extemporaneous comments from members of the public attending or listening to our meetings. Written comments are also welcome. The ACRS section of the U.S. NRC's public website provides our charter, bylaws, letter reports and full transcripts of all full and subcommittee meetings, including slides presented here.

The rules for participation at today's meeting were announced in the *Federal Register* on May 24th, 2019. The meeting was announced as an open/close meeting. We may close the meeting at the end of appropriate parts of the sessions to discuss proprietary matters, and presenters can defer questions that should not be answered in public session to that time.

I'll go off script and note that we'll make sure the NuScale and staff warn us if we're straying into proprietary matters, and we can make note and hold it til the closed session.

We received a written statement from Mr. Michael Derivan. Mr. Derivan's comments will be attached to this meeting's transcript and they may be found in the NRC's document control system under ADAMS accession number ML19168A174. 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 microphones located throughout the meeting room when addressing the Subcommittee. Participants should first identify themselves and speak with sufficient clarity and volume so they can be readily heard. I'll note everybody with your appliances, please turn off your phones or at least put them on mute or vibrate so we don't get disturbed.

We have a bridge line established for the public to listen to the meetings. To minimize disturbances, the public line will be kept in a listen-in only mode. We'll now proceed with the meeting and I'll call upon Marty Bryan, Marty, of NuScale to begin today's presentations. Is the green light on somewhere, behind all your computers?

Overview of Chapter 3

MR. BRYAN: Yes, good morning. I'm Marty Bryan. I'm the NuScale Licensing Project Manager for Chapter 3, and we've reorganized it a little bit out of numerical order, just to help with the flow of the presentation. So this morning as you see the sections here, Patrick Conley, J.J. Arthur, Josh Parker and Storm Kauffman will be leading us through. So we're going to start out with Patrick Conley in Section 3.2.

MR. CONLEY: Okay, good morning. My name is Patrick Conley. I'm the programs engineer for NuScale Power. I'm going to cover Section 3.2, 3.10 and 3.11, respectively.

Next slide. So 3.2 is classification of SSC, System Structures and Components. Our SSCs were classified according to the size and category requirements, in accordance with Reg Guide 1.29 Rev. 5, the quality groups for Reg Guide 1.26 Rev. 4, and rad waste classifications were in accordance with Reg Guide 1.143 Rev 2.

The SSCs were classified as A1, safety-related and risk-significant, A2, safety-related but not risk-significant, B1, non-safety but risk-significant, and B2, non-safety and non-risk significant. There were COL items that required the applicant to identify and classify any site-specific SSCs according to the requirements set forth in the DCA.

MEMBER SKILLMAN: Patrick, before you change. For the record, what is the determining characteristic for risk-significant versus not risk-significant?

MR. CONLEY: We used our PRA insights and that, along with our operating experience, and we had a D-RAP expert panel. We were able to determine the total determination of risk significance for each proponent. So there was a formula of many things that went in, but PRA was part of that factoring.

MEMBER SKILLMAN: Thank you.

MR. CONLEY: Any other questions on 3.2?

(No response.)

MR. CONLEY: Next slide, please. 3.11 or excuse me, 3.10, seismic and dynamic qualifications of mechanical and electrical equipment. This chapter addresses the dynamic and seismic qualifications of Seismic Category 1, mechanical and electrical equipment and supports. The methods and procedures that we used were in accordance with Reg Guide 1.100, Rev. 3, which invokes IEEE 344, the 2004 edition.

There were three COL items for the applicant to establish a program, develop the records and also to submit a program prior to installation of any of the SSCs in plant. 3.11, environmental qualification of mechanical and electrical equipment. Our EQ program complies with the DSRS 311, which deviates a little bit from NUREG 800, as you're well aware.

The EQ program includes equipment in 10 C.F.R. Part 49 scope, certain PAM equipment specified in Reg Guide 1.97, and active mechanical equipment with safety-related functions. Our program meets the requirements of the generic design criteria 124 and 23.

You could look in the GCA, you'll see that Table 3.11-1 lists the harsh (phonetic) equipment that is subject to environmental qualification, and environmental qualification conditions considered include AOOs, normal, accident and post-accident conditions, and those are respectively represented in Appendix 3 Charlie of the DCA.

There are four COL items to provide site-specific EQ program qualification and documentation. Thanks.

MR. ARTHUR: My name is J.J. Arthur. I'll be presenting 3.9, 3.12 and 3.13. So Section 3.9.1 addresses analysis methods for Seismic Category 1 components and supports, including ASME boiler pressure vessel code Division 1, Class 1, 2 and 3 components, subsection NG for core support structures and subsection NF for supports.

The bulk of Section 3.9.1 is devoted to a description of all of our service level A, B, C, D and test events, which define the thermohydraulic conditions for the NuScale power module and represent bounding, representations of NPM operation that are using the components for these evaluations.

This section also contains a catalogue of the software used by NuScale in the dynamic and static analysis of mechanical load stresses and definitions, and also in the hydraulic transient load analysis of the Seismic Category 1 components. These are listed here on the slide, but I'm not going to read them to you.

Finally, Section 3.9.2 was covered with you just over a month ago, so we're not -- I'm not discussing that today.

Safety-related pressure retaining components, core support structures and component supports are designed and constructed in accordance with the rules of the ASME boiler and pressure vessel code, Section 3, Division 1. Almost all of NuScale's ASME Section 3 components are illustrated on this image shown on the slide, where you can see the containment vessel, reactor pressure vessel and the reactor vessel internals.

Section 3, piping, is covered in Section 3.12, which we'll talk about in a little bit. Can you go back a moment?

MALE PARTICIPANT: Sure.

MR. ARTHUR: My last bullet there. Section 3.9.3 contains descriptions of all the loads considered in the design and the components, component supports and core support structures, as well as the load combinations and stress limits applicable for all the service conditions.

Control rod drive system is composed of the pressure-retaining housing, control rod drive shaft, which attaches to the control rod assembly hub, and external water cooled electromagnetic coils, which provide for movement of the control rod assembly in and out of the core.

Portions of the control rod drive system, which form reactor coolant pressure boundary are designed and constructed in accordance with the rules of the boiler and pressure vessel code, Division 1, Subsection NB for Class 1 components. While the NuScale controller drive mechanisms are very similar to those used in existing PWRs, the NuScale CRDMs have a couple of unique features, namely a remote disconnect mechanism and longer than typical drive shafts.

NuScale has conducted rod drop testing and confirmed that the system can perform at safety function under conditions that result in misalignment of the control rod drive shaft support structures. Furthermore, a COL applicant is required to implement a control rod drive system operability assurance program that includes a series of tests including performance testing, stability testing, endurance testing and production testing.

These test are intended to demonstrate acceptable performance of the CRDS with respect to wear, functioning times, latching and the ability to overcome a stuck rod.

CO-CHAIR MARCH-LEUBA: I know we're not talking seismic explicitly, but this is a component of concern, a very long drive shaft. Has misalignment caused by long shaft been considered?

MR. ARTHUR: Yes. We used seismic results to inform the misalignment that we tested. It's supported at six locations along the upper riser, and so we misaligned those in the test facility informed by the seismic analysis.

CO-CHAIR MARCH-LEUBA: Okay, thank you.

MEMBER REMPE: Excuse me for a minute.

Someone must have an open line, perhaps from NuScale at this time, and we are getting feedback. So please, everyone on the line mute your phones, okay? Thank you.

DR. SCHULTZ: J.J., the prototype testing. Can you amplify that a bit as to who's doing what in the testing program, and what stage is it in at this point?

MR. ARTHUR: So we don't have any further testing happening right now. The COL item captures that the COL applicant will establish all of that with the vendor in the future. The only testing we did for the design certification is the raw drop testing.

DR. SCHULTZ: From the tests that you've done, how would you describe the scope of what you're asking the COL applicant to do and what you're asking us to wait for, if you will, what you're putting off to the COL stage?

MR. ARTHUR: Yeah. I guess --

DR. SCHULTZ: It sounded like a pretty expansive testing program that you've described so far.

MR. ARTHUR: Yes, it is, and I can't speak to a lot of the details with more than I've said. If someone in Corvallis is there? That's the

CRDM engineer. I don't know who's on the line.

MR. BRYAN: I do know in the Chapter 4 we've provided quite a bit of information. I was looking back at those slides yesterday. So there's quite a bit of detail there. But I think in summary, we did cold testing, so there's a lot of hot testing still to be done at the COL stage.

DR. SCHULTZ: I see, okay. I'll go back and look at that slide set too. Thank you.

MR. ARTHUR: Moving on to reactor vessels internals, the NuScale RVI assembly is composed of several subassemblies, which are located inside the reactor pressure vessel.

Primary functions of the internals are to provide structure to support, properly orient, position and seat the fuel assemblies, provide support and properly align the control rod drive system, and provide a flow envelope to direct the natural circulation primary coolant flow from the reactor core to the steam generators and back to the core.

Core support structures and internal structures are designed and constructed in accordance with the rules of ASME Code, Division 1, Subsection NG.

2012 edition of the ASME OM code was used to develop in-service testing requirements specified in 3.96. In addition, NuScale has applied an alternative authorization to also invoke the requirements of the 2017 edition, mandatory Appendix IV for the performance assessment testing of power operating valves.

Pumps, valves and dynamic restraints that are required to perform a specific function in shutting down the reactor to a safe shutdown condition, maintain the safe shutdown condition or mitigation consequences of an accident are required to be in the in-service testing program in accordance with ISTE 110-0 of the ASME OM code.

The NuScale design has no safety-related pumps, motor operated valves or dynamic restraints. Our IST program contain 39 valves for NPM, which includes 26 that are hydraulically operated. These are the containment isolation valves and the DHRS actuation valves, the five ECCS valves, two air-operated valves, four pressure relief valves and two check valves.

In addition, we have an augmented valve testing program that contains an additional 12 valves for NPM that either provide non-safety backup to the safety-related function or are non-safety related and provide augmented quality function.

Section 3.9 has 12 COL items that include provision of final design reports for the ASME Code Section 3 components, the establishment of site-specific pre-service and in-service inspection programs, and items related to the comprehensive vibration assessment program. Section 312 covers all of the ASME Code Class Form 2 and 3 piping systems, piping components and associated supports.

NuScale design has a relatively small amount of -- relatively small amount of relatively small diameter piping, Code Class 1, 2 and 3 piping. The largest piping connected to the reactor coolant system is two inch nominal pipe size, and the largest Section 3 piping in the design is our 12 inch diameter main steam line.

MEMBER CORRADINI: The largest two inch is CVCS or the RVV, the safety relief valve? I can't remember which one.

MR. ARTHUR: The piping is CVCS.

MEMBER CORRADINI: CVCS.

MR. ARTHUR: Yes.

MEMBER CORRADINI: And the largest the ECCS valves is the RRVV or the safety relief? That's what I can't remember.

MR. ARTHUR: I believe it's the safety relief. I'll look to Zack.

MEMBER CORRADINI: That's what I remember. That's --

MR. ARTHUR: Yeah, I think that's correct.

MEMBER CORRADINI: Okay, thank you.

MR. ARTHUR: Stress analysis in accordance with the ASME Code and NRC requirements has been performed for the high energy piping within the NPM, in support of Section 3.6 which Storm Kauffman will address later in our presentation. Finally, we have screened all of our piping for thermal stratification and thermal oscillations using the EPRI criteria.

This evaluation resulted in the identification of one potentially susceptible location. CFD analysis was performed to demonstrate that stratification does not occur, and that the temperature fluctuations in the decay heat removal system align, and the associated containment penetration cause thermal stresses that are below the endurance limit for the materials of the piping, the rods and the containment vessel. This slide shows --

CO-CHAIR RICCARDELLA: Are there any plans for monitoring for thermal striping or any of that sort of thing?

MR. ARTHUR: Aside from an in-service inspection that we plan to do.

MEMBER BALLINGER: What are you defining as the endurance limit?

MR. ARTHUR: The endurance limit is the stress below which there are no concerns with fatigue, for an infinite number of cycles without compromising the material.

CO-CHAIR RICCARDELLA: We know that. What's the value?

MEMBER BALLINGER: Yeah. For these materials as soon as you put water in there, there is no endurance limit. So it's defined as some number of cycles, usually 10 to the 7th, maybe 10 to the 8th or thereabouts.

MR. ARTHUR: Yeah. I don't recall what we used in this case.

CO-CHAIR RICCARDELLA: For stainless, I think the ASME Code goes out to 10 to the 9th.

MALE PARTICIPANT: 10 to the 9th.

CO-CHAIR RICCARDELLA: Yeah. They just, a new rev of the ASME Code. Okay.

MEMBER SKILLMAN: For that location that

has been identified as potentially susceptible, what consideration is given to having amend flow orifice or a triple flow, so that there's just a constant, almost inconsequential heat loss that nevertheless stratification stopped.

MR. ARTHUR: I'm not sure we could do that. This is the DHR condensate line, where it attaches to the steam generator from the feed water line. I guess I'll say the analysis we've done has shown us that while it was potentially susceptible, we're not concerned about it any longer.

MEMBER SKILLMAN: Thank you.

CO-CHAIR RICCARDELLA: Under what mode of operation is the concern? Under normal operation or under DHRS operation or --

MR. ARTHUR: Yes. The potentially susceptible situation is during normal operation.

MEMBER CORRADINI: So we're talking at the bottom of the condensate line connecting to the steam generator?

MR. ARTHUR: Correct.

MEMBER CORRADINI: Okay, thank you.

MR. ARTHUR: This image shows all of our -- almost all of our Section 3 piping. The middle image shows the piping inside the containment vessel.

Most of what you see there is main steam feedwater and CVCS, and then the image on the right shows the piping on the containment vessel head, and then also attached to the side of the containment.

CO-CHAIR MARCH-LEUBA: One concern I've always had with this small pipes connecting the vessel to the containment is the seismic problem when you have two humongous masses moving at different frequencies, and clearly having some displacement. I see that you are going with the straight shots and everything is curved. Has it been considered that it has enough elasticity not to break in a seismic event?

MR. ARTHUR: Yes. All of our analysis includes consideration of the seismic anchor motions --

CO-CHAIR MARCH-LEUBA: I would think that that is the limiting by far.

MR. ARTHUR: Yes, yes, that's correct.

CO-CHAIR MARCH-LEUBA: You have two humongous masses going like this.

MR. ARTHUR: Right.

CO-CHAIR RICCARDELLA: So the ASME Code for -- permits socket welded piping for I think two inch diameter and less. At a prior meeting I asked the question about your small bore piping and I got the response that you're not using any socket welds?

MR. ARTHUR: That's correct. We don't have socket welds.

CO-CHAIR RICCARDELLA: You're using full penetration welds?

MR. ARTHUR: That's right.

CO-CHAIR RICCARDELLA: Thank you. Just wanted to confirm.

MR. ARTHUR: The design analysis of pressure boundary threaded fasteners complies with the ASME code for Class 1, 2 and 3. Not applying any code cases, but following the code as written. We have one COL item in this section for a site-specific in-service inspection program for threaded fasteners.

(Off microphone comment.)

CO-CHAIR RICCARDELLA: Thanks, J.J. Now Josh Parker will take us through the 37386.

MR. PARKER: Good morning. My name is Josh Parker and I'll be going through, as Marty said, 3738 and 33 through 35. Before I get started through the bulk of my presentation, I was going to present this cross section through some of our major structures. So what you see here is the -- a cross section through the reactor building, which is the building in the middle with the red roof on it.

To the right of that or to the east is the control building. Both of these structures are Seismic Category 1 buildings on the site, and then to the left or to the west of the reactor building is our rad waste building, and that is a Seismic Category 2 structure and also a Rad Waste Category RW2A, per the requirements in Reg Guide 1.13.

So we'll go into Section 33 now on the wind and tornado loadings. This was discussed as a part of Chapter 2 of the FSAR back in December. But our design basis wind load is 145 mile an hour, three second gusts and used Exposure Category C. The reactor building, control building and rad waste building are all assessed using an importance factor of 1.15.

For tornado loads, we have conservatively assumed the tornado characteristics of Region I that are defined in Reg Guide 1.76, and this is the highest wind speed and this is all the characteristics that are defined in the Reg Guide. Similarly, the design basis hurricane wind speed NSO (phonetic) criteria have been taken directly from Reg Guide 1.221. All this results in wind analysis complying with the requirements of GDC 2 and 4.

Then in this section we have a COL item

that confirms that site-specific adjacent structures will not collapse or have an adverse effect on the Seismic Category 1 structures. As for 3.4 --

MEMBER CORRADINI: So let me ask the question before we move on. So the limiting missiles are not from wind and tornado?

MR. PARKER: The limiting missiles?

MEMBER CORRADINI: The loadings and missiles that would be generated by them are not from wind and tornado, the way the design is set up. Am I understanding correctly?

MR. PARKER: We have assessed for the -- we have assessed wind and tornado missiles, and we've assessed for turbine missiles, and that will be discussed in 3.5.

MEMBER CORRADINI: Okay, fine. Thank you.

MR. PARKER: For flood design, first internal flooding. Our internal flooding is done by a level by level and room by room area analysis for both the reactor building and the control building for postulated flooding events. So we do that by first assessing the water systems that might be in an area, and the volume of water and the flow rate and then the time to isolate that system if there was a postulated break.

Then we identify the -- any equipment that need protecting in those rooms, and then can determine what the mitigation, mitigating features are to protect that particular system. So that could be watertight doors, it could be elevating the system, what have you, and that's all been described in 3.4 of the FSAR. Next slide. So the previous slide was on internal flooding --

MEMBER CORRADINI: So let me ask a different question related to that. You guys in the design already know what's on what floor?

MR. PARKER: Yes.

MEMBER CORRADINI: So what's at the bottom floor?

MR. PARKER: Rad waste systems.

MEMBER CORRADINI: Okay, and then?

MR. PARKER: CVCS.

MEMBER CORRADINI: And then?

MR. PARKER: Electrical and I&C.

MEMBER CORRADINI: So where's the batteries?

MR. PARKER: That's on the 75 foot elevation of the reactor building, one floor below grade.

MEMBER CORRADINI: Okay, thank you.

MR. PARKER: And our favorite system, the boron system, EAS?

MEMBER CORRADINI: That's the 50 foot elevation. It's two floors below grade.

MR. PARKER: Two floors below the water level, is that -- or below grade?

MEMBER CORRADINI: Below grade.

MR. PARKER: Water level is higher?

MEMBER CORRADINI: Well, if we're talking water level in the pool, that water level is higher. But the water level we look at is for any --

MR. PARKER: You're talking about flooding?

MEMBER CORRADINI: Yes, right.

MR. PARKER: Yeah. So any piping systems that are running through the building, we postulate grades for all of them, and then assess the equivalent that's around it.

CO-CHAIR MARCH-LEUBA: So I'm just a little -- I only had one cup of coffee this morning. The BAS is then a 55, which is like ten feet below grade?

MR. PARKER: So we establish grade at 100 feet at 100 feet as nominally -- that's what we call our grade level --

(Simultaneous speaking.)

CO-CHAIR MARCH-LEUBA: 45 feet?

MR. PARKER: So the 50 foot elevation is actually two floors below grade. So nominally each floor is about 25 feet in elevation.

CO-CHAIR MARCH-LEUBA: Two floors, two big floors?

MR. PARKER: Two big floors, right. Okay. So that was internal flooding. Now with respect to external flooding, we defined our maximum flood elevation for potential flooding as one foot below grade, and the maximum water elevation of two feet below grade. So these all become -- as a result, there is no dynamic flood loads on these structures.

But these saturated soil pressures and water loads are used in the static and dynamic analysis of the structures that's a part of 3.7 and 3.8. Similarly for precipitation, the rates are given there, and those become input to the structural analysis loads that we used for the design of the structures.

Lastly on 3.4, we're satisfying GDC 4 and 2. We also look at the interaction between non-Seismic Category 1 with Seismic Category 1 structures to assess for any potential credible flooding source.

As we talked about previously, all of the equipment that needs protection is in the reactor building or control building. So those buildings have been assessed and we provide those mitigation features there.

Then we have seven COL items to confirm site-specific conditions, and the -- and materials for the structure used.

MEMBER BALLINGER: I had a curiosity question. Could you back up one slide? On the extreme snow load, 75 PS, pounds per square foot?

MR. PARKER: Yes.

MEMBER BALLINGER: Water is 62.4 pounds per cubic foot. So that would be 65 pounds per square foot. Where did the 75 pounds come from?

MR. PARKER: So these are --

MEMBER BALLINGER: I mean how do you get 75 pounds?

MR. PARKER: Yeah. Well, these come from ASME 7.

MEMBER BALLINGER: It just struck me as weird.

MEMBER SKILLMAN: Well, for 14 feet of snow.

MEMBER BALLINGER: Yeah, I guess. Yeah,

14 feet of snow.

CO-CHAIR RICCARDELLA: No, it's ice. It's ice.

MEMBER BALLINGER: So divide by ten roughly, right, for the snow?

MEMBER SKILLMAN: It's ice.

CO-CHAIR RICCARDELLA: Depends on where you are.

MALE PARTICIPANT: Well, ice would be pure water, yeah.

CO-CHAIR RICCARDELLA: If you're in New England, you divide by 2.

CO-CHAIR MARCH-LEUBA: The roof is very flat. I mean it just accumulates the snow and that's --

MR. PARKER: The reactor building roof is relatively flat and yes, I mean there may be some slight sloping to it. There likely would be for drainage, so there wouldn't be any ponding.

CO-CHAIR MARCH-LEUBA: So you're in Northern Minnesota. You will accumulate a lot of snow there, right?

MR. PARKER: Potentially.

MALE PARTICIPANT: With global warming, no.

CO-CHAIR MARCH-LEUBA: I know it's higher.

(Simultaneous speaking.)

MR. PARKER: All right. Moving on to missile analysis now. So here again we're meeting GDC 2 and 4. As I mentioned before, our rad waste building is classified as RW2A, and Reg Guide 1.143 indicates that that would have us also use the same missiles as a Seismic Category 1 structure. So our rad waste building is designed for the same missiles as the reactor building and the control building.

The missiles considered are the five kind of categories that are listed here on the slide, internally generated turbine missile, tornado and hurricane missiles. They also list site for aircraft hazards, but we don't postulate any missiles from them, giving the siting criteria we've established in Chapter 2.

Then there's -- we have a beyond design basis aircraft impact assessment that was covered in 19.5 the last time.

MEMBER SKILLMAN: When you say you've excluded what I think you said is the wind-driven missiles --

MR. PARKER: We have not excluded those.

MEMBER SKILLMAN: Because I'm thinking of a Buick coming through the roof or a Silverado coming through the truck bay at 35 and 50 miles an hour.

MR. PARKER: So the wind-driven missiles are an automobile, a sphere and essentially a wooden shaft like a hole. All those have been considered.

MEMBER SKILLMAN: Thank you.

MR. PARKER: Next slide. So for the NuScale design, we had no credible missiles inside containment, and as we just talked about, the wind and hurricane missiles have been defined by the criteria in Reg Guide 1.76 and 1.221 respectively.

All of our safety-related and risk significant components are located inside the reactor building or the Seismic Category 1 portion of the control building, and a function of those structures is to act as a barrier to protect safety-related systems. We have -- our analyses have shown they can serve that function against all postulated missiles.

We have four COL items to confirm the site-specific missile analysis in the missile criteria. So next on turbine missiles, a little more detail here.

MEMBER BLEY: Can I stop you before you get started on this one?

MR. PARKER: Sure.

MEMBER BLEY: I have a lot of concerns on this one, and let me just put a few things out there before you start. Some of this may apply more to the staff than to you, but I think that it applies to both. Early on we had a lot of plants designed with unfavorable positioning of the turbine, and then once we started talking about turbines, we mostly tried not to point them at anything we care about.

You've decided to claim that's something you care about and to use the -- not a probabilistic argument, which might be pretty good given the size of your rotors. I think you have monoblocks, right.

But you decided to use the barrier approach. I'm going to ask the staff. I don't know if anybody's used that before in the turbine missile licensing process. If we have -- that gives us a little more basis. If we haven't, going through your report, you throw out equations for penetration depth and other things. I didn't see a source for those in your report.

If we go to NRC's reg guide, they point to a 19, I think it's 87 or 97 reference. When you go through that, it points out that most of the data they used to come up with -- and they came up with five different formulations or they found five different formulations, I think it was five, from the munitions area looking at penetration of -- intentional penetration of missiles.

So I suspect everything from that kind of analysis, that when they think of being conservative, they want to be sure no missiles get through. For our case, we want to make sure the missiles don't get through.

So whatever was conservative over there is probably non-conservative over here. One, where did your formulations come from, what's the basis for them? Two, do you think they're conservative?

I was involved in a probabilistic analysis of this problem some years ago. It's very complex if you look at all of the different angels you can come from and all the different masses and momentums, the key thing. The reg guide cites a penetration velocity that echoes just a wealth of physics built into it. It's not just a number.

So if you can tell me why you have really good confidence, I'd appreciate it. It strikes me that if this is unique, as I expect it is and even if it isn't, I would have expected in a technical report or something to really lay out how you did this analysis, because proving that you can't get through the barrier is crucial, and from what I've read, it's hard to be convinced that you really thought of everything and looked at it thoroughly. Go ahead.

MR. PARKER: All right. So with respect to our turbine missile approach or our barrier approach, you're right. I don't believe there's been another vendor that's done it this way. We reviewed the paper you referenced. There are a couple of things about that paper.

One is that one of the out, the going-in assumptions is that both the missile and the target are infinitely rigid, and the paper makes the point that if there is any deformation, that those equations were drastically over-predict the amount of penetration. They're also based on a --

MEMBER BLEY: But we don't know what drastically means, but go ahead.

MR. PARKER: They're also based on a certain penetration depth to barrier thickness values. So we opted to use a finite element analysis to predict the amount of penetration distance that our missile would go into our barriers, and we validated --

MEMBER BLEY: That is from like the basic physics approach?

MR. PARKER: Yeah, we validated --

MEMBER BLEY: Other than these correlations that's in another place?

MR. PARKER: Correct, that's right. We validated that, what the software was producing with this test data.

MEMBER CORRADINI: What test data?

MR. PARKER: I don't have that, those references in front of me.

MEMBER CORRADINI: So to follow Dennis' question, I think, is there a topical or technical report that we can look at? I think that's where --

MR. PARKER: No.

MALE PARTICIPANT: The answer is no.

MR. PARKER: There's not. No, this is all in our FSAR and our supporting calculations.

MEMBER CORRADINI: So the supporting calculational document?

MR. PARKER: Yes.

MR. BRYAN: Yeah. I would note for the reason you stated, the staff had quite a few of the same questions. We had a number of interactions, and then in June we submitted an extensive document with that background or basis for how we got there.

MEMBER BLEY: Is that on the docket?

MR. BRYAN: Yes sir. So it's pretty comprehensive.

MEMBER BLEY: We'd like to see that Mike, if you can find that for us. Not today, you know, some time later. We'll look at it before they come back.

MS Is that the enclosure to that RAI response or is it --

MR. BRYAN: Yes, it's part of the RAI responses.

MEMBER BLEY: Oh okay. So maybe we found it yesterday.

(Simultaneous speaking.)

MEMBER CORRADINI: Is that what you requested?

MEMBER BLEY: Well, yes I did.

MEMBER CORRADINI: Okay.

MEMBER BLEY: The backup document, I haven't looked at it yet. But is that the document that really provides the full engineering analysis?

MR. BRYAN: Yes, it is. We extracted from all the supporting documents, and then the staff has audited those. But it's quite a bit of detail in that RAI response in June.

MEMBER BLEY: I hope so, because this is

-- I mean this is kind of new ground for the NRC, and it's a place we've got to make sure we've done it right.

MR. BRYAN: The staff had the same questions, yeah.

MEMBER BLEY: Including your analysis must have looked at -- okay. I'm assuming you didn't have any missiles go all the way through?

MR. PARKER: Not for the reactor building, and for the control building, it didn't go through the exterior wall. For the control building our barriers, the exterior wall and the grade floor elevation. So we didn't see missiles going through our credited barriers.

MEMBER BLEY: Okay, that's good. I really want to look at that. From the study I was involved in some years ago, the really interesting thing that came out of that was, and this was going through heavy reinforced concrete, if you had enough energy to get through the wall, even though your velocity is lower inside, some of the cases we looked at you didn't have enough energy to get out the opposite wall, and then it looks like a pinball machine in there.

It just spins all around, crashing into wall and cutting everything to pieces. I look forward to looking at that detailed analysis. If you saw -- did you do a probabilistic analysis?

MR. PARKER: No. We defined these three missiles here, the blade, a blade with a portion of the rotor --

MEMBER BLEY: We don't care about a blade.

MR. PARKER: The blade, the portion of the rotor, and then half of the last stage of the rotor.

MEMBER BLEY: Okay, and that -- half of the last stage is new. You hadn't looked at that before?

MR. PARKER: That's right.

MEMBER BLEY: Okay. Well half is pretty big, and you know most of the stuff I've seen, you usually get two to four or something like that big pieces out. So half is -- I mean three to four. So half is pretty big, and you assumed the velocity --

MR. PARKER: So the velocity was based on a 3600 RPM turbine and then we also looked at varying overspeed conditions.

MEMBER BLEY: Okay, and did you assume overspeed trucks would keep it from going too fast?

MR. PARKER: We varied the overspeed. I mean we defined our destructive overspeed at 160 percent. But we looked at it all the way up to 210 percent.

MEMBER BLEY: Okay, okay. Well, I look forward to seeing that detail. Go ahead.

MEMBER SKILLMAN: I'd like to ask a basic question. Given all the effort in industry over the last probably 30 years on this topic, as I imagine our footprint east-west, with modules 1 through 12 adjacent to each other kind of in the center and going west, the control building unit to the east, rad waste on the left, here you have these two turbine buildings that are basically on the same footprint as the reactor building.

Why wouldn't you just have oriented the turbines to have shafts north and south, so that you would not be dealing with this?

MR. PARKER: It really just kind of became to an optimization of piping layout at the time when the site was arranged.

MEMBER BLEY: So minimizing piping or something along those lines?

MR. PARKER: Right.

MEMBER BLEY: Because we asked about this in an earlier meeting. You put it off until now, and the arguments for orienting them this way weren't really clear from that meeting. I guess I'm not 100 percent clear here, but a little extra piping to avoid this problem would have been a nice idea.

MEMBER BALLINGER: You use a monoblock rotor, right? So what is the definition of half of last stage turbine rotor disc? It's a monoblock rotor. There's no disc.

MR. PARKER: I'll defer that to --

MEMBER BALLINGER: Have I got the design wrong?

MR. HOUGHTON: This is Zack Houghton, the mechanical design engineering manager, and I was also here presenting Chapter 10, where we discussed this topic a bit as well.

MEMBER BLEY: Yeah.

MR. HOUGHTON: For the monoblock rotor, it looks at a section that would be where the roof section of the blade is. So that's where I think it's referring to disc, when it looks at a chunk of the rotor from that section.

MEMBER BLEY: Okay.

MR. HOUGHTON: As far as some of the other questions that we heard, the reason for the orientation the way it is, one of the things that we looked at is not just length of piping but also balancing, right, so that we get equivalent operation between all the turbine generators.

If we stack them all width-wise, they would be further out than the reactors are themselves, right? The width of the turbine island would be larger than the reactors. So your modules for Reactors 1 and 6 would be much further away than for Modules 3 and 4. So we looked at all the impacts of going with the barrier approach per Reg Guide 1.115.

I'll note that even in the favorable orientation, we would still have to account for high trajectory blades. That's a requirement of the regulation, and it also --

MEMBER BLEY: I know it is, but you know that goes away on probability basis.

MR. HOUGHTON: Right. So the probability basis --

(Simultaneous speaking.)

MEMBER BLEY: Just the solid angles, then you drop one back down to get it.

MR. HOUGHTON: And even for a favorable orientation, we still have to do the probability analysis to look at, you know, to ensure that the missile generation frequency would be acceptably low. We did talk to turbine vendors about what a probability analysis for our size machine would look like, and they confirmed that they could meet the requirements of the probability analysis, whether it was in the favorably or unfavorably oriented position.

MEMBER BLEY: I would -- that's what I would expect with a monoblock rotor. But I haven't seen any data on those. I don't know what our experience has been with these.

(Simultaneous speaking.)

MEMBER CORRADINI: I was going to say, with the monoblock rotor I would expect --

MALE PARTICIPANT: Zero.

MEMBER CORRADINI: Zero.

MEMBER BLEY: Yeah, or something close. Anyway --

MR. HOUGHTON: And we would still have to look at blade failure. So even in the monoblock rotor, you're still looking at the root section, right, and it's -- even in a monoblock rotor --

MEMBER BLEY: What do the blades go through? Come on.

MR. HOUGHTON: What's that?

MEMBER BLEY: The blades don't go through much.

MR. HOUGHTON: Sure.

MEMBER BLEY: They might get out of the

machine, but they don't have much energy.

MR. HOUGHTON: The rotor fragment is the more limiting missile. Yes, we agree.

MR. PARKER: Okay. I think we've covered everything in this slide through their discussions, so let me go to the next slide. So -- kind of continuing on. Our barriers are designed through the SRP. We do look at both the local effects of the missile as well as the global effects from the missile, which is really where we see more of the last stage of the rotor, since it's much larger than the others.

So we determined the required thickness for the penetration, perforation and scabbing. Given the wall thickness of the reactor building and the wall thickness and slab thickness of the control building, we have shown adequate protection. I mean there is a number of conservatisms in the analysis that we haven't considered.

For example, we don't take credit for any of the reinforcement of the walls. We don't take credit for the thickness of the turbine casing. None of that is included in the analysis.

MEMBER BLEY: You just left me confused. The barrier wall you did, of course, that's what you're hitting.

MR. PARKER: But yeah. We assumed --

MEMBER BLEY: You're assuming there was no steel in it?

MR. PARKER: We assumed that the missile hit between layers of reinforcement.

MEMBER BLEY: What kind of spacing is there in your reinforcement?

MR. PARKER: I mean it's six to 12 inches, depending on the wall. So again, that's a conservative --

MEMBER BLEY: All you analyzed was the concrete wall?

MR. PARKER: That's right.

MEMBER BLEY: Hmmm.

MR. PARKER: And still showed that the missile didn't penetrate.

MEMBER BLEY: We'll look forward to looking at that.

(Off microphone comments.)

MR. PARKER: Next slide. So at this point we'll transition into seismic design. First of all, our input parameters. So this is our certified seismic design response vector, our CSDRS, this figure on the lower left. This is our input specter for all the Category 1 SSCs in our design, and that's on the lower left.

In addition, we have a CSDRS that we've also added some high frequency content to. So that is our CSDRS HF or that's the figure that's on the lower right. These are both our input specters for our Category 1 structures.

MEMBER CORRADINI: So, you're looking at the maximum of those two?

MR. PARKER: We look at both and, then, we take the bounding case.

MEMBER CORRADINI: Okay.

MR. PARKER: Yes.

MEMBER CORRADINI: Thank you.

MR. PARKER: Both these spectra are developed based on industry data and they meet the requirements of GDC 2 and 10 CFR 50, Appendix S.

Next slide, yes.

So, the figure on the screen here shows our spectra in orange and, then, also, the spectra from Reg Guide 1.60, anchored at 0.3g. We can see that the spectra that we've used is both higher in its ZPA -- we have a 0.5 ZPA -- and is more broad than the typical spectras used for design certifications.

Next slide.

Moving on with our other seismic inputs, we have five seed time histories that are compatible with their CSDRS and an additional seed time history compatible with the CSDRS HF.

We have four generic soil profiles that we've used in the analysis, and they vary from soft, which is a soil type 11, to very hard rock, or soil type 9.

The damping values for the analysis models follows the guidance of Reg Guide 1.61. And as previously stated, we only have two Seismic Category I structures. They're the reactor building and the control building. The rad waste building is Seismic Category II, given its proximity to the reactor building. And so, it's included in our multi-building analyses models that are developed in SAP2000 and SASSI. And I'll discuss those in the new few slides.

So, first, SAP2000, we use that to model the static analyses for the structures. These models have both material properties for both uncracked and fully cracked concretes. In the case of the reactor building, we also have a submodel of the NPM that's based on the work that's performed in ANSYS. And that was discussed as a part of 392 and the separate NPM Seismic Technical Report.

And then, finally, we create individual

models of these buildings that we call single building models. And then, we also create a model that has all three of the reactor building, control building, and rad waste building. And we call that triple building model. We create individual models and triple building models for it in SAP2000.

MEMBER CORRADINI: The purpose is interaction between the buildings?

MR. PARKER: That's correct.

MEMBER CORRADINI: Okay. Thank you.

MR. PARKER: Yes. Given their proximity to each other.

Next is SASSI. We use SASSI2010 for the dynamic soil structure interaction analysis. The SAP2000 models that we talked about in the previous slide are converted to SASSI. In doing so, the models have exactly the same node numbers, coordinates, element types, and material properties. And the figure on the upper right shows the triple building SASSI model for our design, and we have individual SASSI models as well.

We perform our SSI analysis using the extended subtraction method. In this approach, the interaction nodes are taken at the skin of the excavated soil model, which is the six sides, and we also have a plane in the middle. So, that's our extended subtraction method. Or we also call that the 7P method. And that's done for development of ISRS and for the dynamic forces and moments in the structures.

Next slide.

Next, I'll talk about our use of ANSYS in the structural analysis. We have three main purposes for using ANSYS in structural design.

The first is the study of Fluid-Structure Interaction. So, given that neither SAP2000 nor SASSI has the ability to model fluid elements, we use ANSYS to determine the hydrodynamic pressure in the reactor pool.

MEMBER CORRADINI: So, just educate me.

MR. PARKER: Yes.

MEMBER CORRADINI: So, that's the reason you go with ANSYS inside, because it has fluid elements?

MR. PARKER: That's right.

MEMBER CORRADINI: But the reason you don't use ANSYS outside is because the other two have an advantage in some manner?

MR. PARKER: Well, so SASSI is what we use for the dynamic analysis. So, that models the soil structure interaction.

MEMBER CORRADINI: Okay. And that's unique to that tool?

MR. PARKER: That's unique to that tool.

MEMBER CORRADINI: Okay.

MR. PARKER: And SAP2000 was used mainly just out of convenience. It's a common tool for structural analysis --

MEMBER CORRADINI: Okay.

MR. PARKER: -- building structural analysis.

MEMBER CORRADINI: But the reason to switch is strictly the fluid model?

MR. PARKER: Well, so we have three main purposes for ANSYS. One is for the fluid analysis, so to get those dynamic pressures in the pool and, also, to determine the sloshing wave height.

MEMBER CORRADINI: Right.

MR. PARKER: We use it for that FSI approach. We also use it for non-linear stability analysis. And I'll talk about that a little bit when we talk about 385.

MEMBER CORRADINI: Okay. Thank you.

MR. PARKER: And then, the third approach --

MEMBER BLEY: Well, I'm sorry. On the sloshing --

MR. PARKER: Yes?

MEMBER BLEY: -- how much can you actually slosh out and where does it go? Does it come right back in?

MR. PARKER: So, the sloshing height is about 2 feet. It's about 24 inches.

MEMBER BLEY: Uh-hum.

MR. PARKER: And our freeboard is about 6 feet. So, there's no water that sloshes out.

MEMBER BLEY: None should come out?

MEMBER CORRADINI: Nothing comes out?

MR. PARKER: Nothing comes out.

MEMBER CORRADINI: I guess just thinking out loud, the fact that you have a fluid element and it can take zero shear, and its load, its unusual load on structures would be probably the most important thing versus the sloshing, assuming you've got a high enough wall.

MR. PARKER: Sure. Yes, in that case.

MEMBER CORRADINI: Okay. All right. Thank you.

MR. PARKER: Then, our last use of ANSYS is determining the thermal analysis and hydrogen line break analysis. So, we actually use 3D solid concrete elements and explicit rebar and pool elements. And then, we do cases for operating thermal and accident thermal to determine the strains in the rebar and pool liner during the thermal and hydrogen line break.

So, on 373, our subsystem analysis, we have four subsystems that we've considered in the DCA. The NPM, which is described in the NPM Seismic Technical Reports; the crane; and the fuel racks, which are discussed as a part of Chapter 9, and then the bioshield.

So, the bioshield is the main subsystem we talk about in 373. It's a non-safety-related, not-risk-significant Seismic Category II component. Its major functions are for fire protection, radiation protection, ventilation, and to support personnel access, as they're located over on top of the modules.

MEMBER CORRADINI: So, just so I understand, the bioshield is the wall facing the pool. There's two concrete partitions or walls that separate the modules, and then, there's a concrete wall which is your missile protection on the outside. So, the bioshield is on the inside facing the other modules?

MR. PARKER: Correct. And it has a concrete portion that's horizontal.

MEMBER CORRADINI: Right, but it lays on top.

MR. PARKER: And it's this steel portion that we kind of see here, which is on the lower right of this screen.

MEMBER CORRADINI: Is this the new design that we --

MR. PARKER: This is the new design. The previous design had hinged panels.

MEMBER CORRADINI: Right, right, right. Okay.

MR. PARKER: This is the latest design that is --

MEMBER CORRADINI: So, it's louvered?

MR. PARKER: Yes, essentially, it's louvered. It has these kind of alternating HDPE panels with a steel frame that allows for continuous passive airflow.

MEMBER CORRADINI: Since you've mentioned airflow, are you allowed to say in open session what the height of these things are?

MR. PARKER: Yes, I think so. It's about 25-ish feet.

MEMBER CORRADINI: No, no, each of the louvered panels.

MR. PARKER: Oh, each of the louvered panels? Yes, that probably would be getting into the proprietary --

MEMBER CORRADINI: Okay, fine. We'll come back to it later.

MR. PARKER: Yes.

MEMBER CORRADINI: Thank you.

MR. PARKER: Okay. Next slide.

Lastly, with respect to --

MEMBER SKILLMAN: Yes, let's go back just for a second.

MR. PARKER: Okay.

MEMBER SKILLMAN: You're clear that it's non-seismic, not safety. So, let's take module 2, which is one east from one.

MR. PARKER: Correct.

MEMBER SKILLMAN: And we're going to refuel two. This is an approximately 75-ton, 150,000-pound load, right? As I read your book, that's what I remembered, 75 --

MR. PARKER: That's right.

MEMBER SKILLMAN: -- 145,000 pounds.

MR. PARKER: Okay.

MEMBER SKILLMAN: And so, there might be some adjustment for the new shield, for the new curtain. So, the No. 2 bioshield will be lifted and placed either on one or three, correct?

MR. PARKER: Or on the other side.

MEMBER SKILLMAN: Oh, okay. What ensures that in that configuration this device placed on probably an identical device does not become either an unacceptable load or under seismic acceleration does not become dislodged or dislodge the one that is holding it?

MR. PARKER: So, that is a design basis consideration, that one bioshield will be placed on top of another. So, we've performed an analysis for that very case.

MEMBER SKILLMAN: Is the maximum stacking two --

MR. PARKER: Yes.

MEMBER SKILLMAN: -- to only one?

MR. PARKER: Right, you would just stack one on top of --

MEMBER SKILLMAN: Only plus one?

MR. PARKER: Correct. And they're bolted down. One bolts down on top of the other.

MEMBER SKILLMAN: So, it's fixed?

MR. PARKER: Correct.

MEMBER SKILLMAN: When it's in that

configuration?

MR. PARKER: That's right.

MEMBER SKILLMAN: Oh. Thank you.

MEMBER SUNSERI: And the airflow was considered?

MR. PARKER: Yes. Yes. There's --

MEMBER SUNSERI: There's two of them right now.

MR. PARKER: Right. Yes. And so, it's not flush up against it. There is some space between one and the other.

MEMBER SUNSERI: Okay.

MEMBER CORRADINI: So, remind us, since we were discussing privately, without the bioshield, I'm worried about radiation exposure to personnel in the refueling area?

MR. PARKER: Right.

MEMBER CORRADINI: That's the issue, due to scattering of radiation coming out of a module, bouncing around, and --

MR. PARKER: And someone being across the pool.

MEMBER CORRADINI: Right.

MR. PARKER: Yes.

MEMBER CORRADINI: But someone wouldn't be across the pool under operation. They would be down in the refueling area. There would be nobody across the pool per se, right?

MR. PARKER: Well, unless we're refueling.

MEMBER CORRADINI: Okay, but that's down -- the crane does not have personnel on the crane as it's moving the NMPs around.

MR. PARKER: But we move -- there's going to be people on top of the module taking off piping and disconnecting the module --

MEMBER CORRADINI: Oh, oh.

MR. PARKER: -- during the refueling process.

MEMBER CORRADINI: Excuse me. I'm sorry. Got it. Okay.

And then, the final question is, is there a louver at the top panel for some sort of air movement at the very top?

MR. PARKER: I believe there is some gap up at the top to allow for ventilation.

MEMBER CORRADINI: okay. Right. We'll come back to seismic. Thank you.

MR. PARKER: Sure.

MEMBER CORRADINI: Thank you.

MR. PARKER: So, lastly, with respect to

3.7, we did conduct a number of sensitivity studies, given the uniqueness of our design. The cases are listed here, but we looked at a case where the dry dock is empty, looked at a case where we had less than 12 modules. We looked at soil separation. We looked at -- I talked before about our extended subtraction use in SASSI. So, we looked at that as a comparison of the direct method and, also, what we call a 9P. So, we're adding planes and assessing the adequacy of the 7P as well as non-vertically propagating shear waves.

As to the results, in all cases the design basis reinforcement was adequate for the sensitivity studies considered. There was some slight ISRS modifications in a couple of points, but those were also very minor.

And then, in the 3.7 area, we have 15 COL items, mainly to confirm the site-specific adequacy of the inputs and comparison of the site-specific inputs at a particular site.

DR. SCHULTZ: So, Josh, this is a site seismic input data characterization, principally 15 different elements that you're going to address --

MR. PARKER: That's the majority, yes.

DR. SCHULTZ: Is there analysis that would be needed or required also?

MR. PARKER: There will be, yes. Yes. So, we'll look at the site-specific input motion as well as the swell properties, and then, confirm that ISRS at a variety of locations is still bounded as well as the design of the critical sections in the structure.

DR. SCHULTZ: Based on what you've done for sensitivity studies, does the COL applicant need to redo sensitivity studies, as you see it?

MR. PARKER: I don't believe so. I mean. we do have COL items that confirm that. So, there are COL items to confirm, for example, soil separation.

DR. SCHULTZ: I was looking at that one, too.

MR. PARKER: Yes. So, there's a number of COL items that confirm these as well.

DR. SCHULTZ: But things like less than 12 modules, for example --

MR. PARKER: There's a COL item for that as well. The COL will need to determine the module loading sequence.

CO-CHAIR RICCARDELLA: But if the site-specific in-structure response spectra were below your design spectra --

MR. PARKER: Yes.

CO-CHAIR RICCARDELLA: -- why do you have to do additional COL analysis?

MR. PARKER: Well, you need to -- in that case, you wouldn't do anything.

CO-CHAIR RICCARDELLA: So, would you call it a tiered evaluation --

MR. PARKER: Sure, yes.

CO-CHAIR RICCARDELLA: -- that's being done?

MR. PARKER: Sure, yes. I mean, you would only need to make modifications if you weren't bounded.

CO-CHAIR RICCARDELLA: Okay. Thank you.

MR. PARKER: All right. Next, we'll move on to 3.8. And I'll talk briefly about the containment, and that will be talked in more detail later on when we discuss Chapter 6.

But the containment in the NuScale design is a steel vessel designed per SME Section 3, Division 1, Subsection NB. It's approximately 75-feet tall and nominally 15 feet in diameter. It's obviously slightly larger at the bolted flanges. It's operated in the reactor pool in the reactor building, and the internal design pressure is 1,050 psia and design temperature, 550 degrees F.

MEMBER CORRADINI: Maybe this just is Code. The outside design pressure is 60? Is that just a Code requirement? I'm trying to figure out how I get to 60.

MR. ARTHUR: A Code requirement to establish an external design pressure?

MEMBER CORRADINI: Yes.

MR. ARTHUR: Yes.

MEMBER CORRADINI: Okay. All right. And then, there were some RAIs and staff questioning about internal pressure of 1050. Has that been settled or are you still in discussions with staff about that? I'm going to ask the staff the same question because I got confused with the RAIs. No offense, but there's a lot of RAIs flying back and forth that I can't figure out.

MR. ARTHUR: The original internal design pressure was 1,000.

MEMBER CORRADINI: So, it's up 50?

MR. ARTHUR: Correct.

MEMBER CORRADINI: The staff analyzed and confirmed? Or they're in the middle of reviewing that?

MR. ARTHUR: You'll have to ask them. I

don't know about that.

MEMBER CORRADINI: Okay, fine. That's good. That's fine. Thank you.

MR. PARKER: The last thing I'll say here is that the CNV Ultimate Pressure Integrity Technical Report documents are in conformance to the guidance of NUREG-6909 and determined the ultimate pressure capacity of 1,240 psia.

MEMBER CORRADINI: Maybe this is not the right place, but we had somewhere -- I can't remember where it was. Maybe it was in one of the chapters. You're asking for an exemption for containment leak rate on the whole containment, is that correct?

MR. PARKER: I think I'll defer that to the Chapter 6 conversations. Is that --

MEMBER CORRADINI: Oh, good. All right. Let's do that. That's where I remember reading it. Thank you. Sorry.

MR. PARKER: That's all right.

All right. Next slide.

So, transitioning back to the structures, so in 3.7, we talked about the design loads. And in 3.8, we discussed the design of the structures. The structures are Seismic Category I and, therefore, designed to ACI-349 and AISC N690. All the normal and extreme loads have been included for the Code-required load combinations. And this allows us to determine the maximum demand load for each of the structural elements in the structure.

We have six COL items to confirm here in 3.8 on the site-specific acceptability of the structural design as well.

Next slide.

So, the detailed information on the design of the Category I structures is discussed and provided in Appendix 3B of the FSAR. And that's where we define the critical sections for each of the structures. The critical sections are the portions of the building that perform a safety-critical function or are subject to large stresses or demand forces. They might be difficult to construct or just generally representative of the building. So, we wanted to include at least one slab, one wall, a beam, a buttress, a pilaster, and then, also, wanted to ensure that a bay wall or a pool wall in the pool was considered as well as the supports for the NPM.

As referenced in the previous slide, the load combinations determine the maximum demands for each section. And then, after determining a reinforcement pattern, we can substantively determine the capacity. And so, we go section by section within the structure to ensure that the capacity is always greater than demand, so that the structure is adequate.

Lastly, in 3.8, is 3.8.5, which is on the foundations. And I'll focus here on stability, which is the ability of the structures to resist sliding and overturning.

So, given the high water table that we assumed and the high input spectra, we ended up using a non-linear analysis for sliding for both the reactor building and the control building. And in doing so -- that was one of the uses of ANSYS -- we saw sliding of less than an eighth of an inch for either structure. So, very, very small values. We also used a non-linear analysis for overturning for the control building and, again, saw even smaller numbers here of about a 64th of an inch of total uplift in the structure. So, very, very small numbers. And this allowed us to conclude that the structures adequately resist sliding and overturning.

So, with that, I'll --

MEMBER CORRADINI: Can you remind me, sliding or overturning due to seismic events?

MR. PARKER: Correct.

MEMBER CORRADINI: Oh, okay, fine.

MR. PARKER: Yes.

MEMBER CORRADINI: Thank you.

CO-CHAIR MARCH-LEUBA: I don't know if this question belongs to you, but have you guys considered expansion of the concrete over time? I'm sure you're going to use the correct concrete, but, even good concrete moves.

MR. PARKER: Sure.

CO-CHAIR MARCH-LEUBA: Is that part of this design?

MR. PARKER: At this point, we were just looking at nominal dimensions for the sections. So, our wall thickness if 5 feet. We assume it's 5 feet in the analysis. And there's ACI guidances for tolerances.

CO-CHAIR MARCH-LEUBA: You are aware that there are some operating plants that have issues with expanding concrete?

MR. PARKER: Sure.

CO-CHAIR MARCH-LEUBA: And one thing that they were worried about is where two buildings get close together, they can start touching. Have you brought enough clearances for just in case?

MR. PARKER: Yes. So, the buildings,

between the control building and the reactor building is about -- I think it's 30 or 35 feet. And then, between the rad waste building and the reactor building is another 20 feet. So, there's quite a bit of space between --

CO-CHAIR MARCH-LEUBA: Sometimes there are structures that are placed from a different wall. Some of the issues shield, things like that. Just make sure you consider that because it's been a big problem somewhere else.

MR. PARKER: Sure.

MR. BRYAN: Before you get started here -- thank you, Josh.

Before we get started, I want to make one clarification on the socket weld. J.J., do you want to address that?

MR. ARTHUR: Yes. We got a note back from our office. So, there are no socket welds in the 2-inch-diameter piping, but we do allow socket welds for less than three-quarter-inch. There's some tubing connecting to the ECCS valves that we do allow socket welds.

MR. BRYAN: Thank you.

Okay. With that, Storm Kauffman will take us through 3.6.

MR. KAUFFMAN: All right. The remaining section in Chapter 3 is 3.6, "Protection Against Pipe Rupture Effects".

I'm Storm Kauffman. I will take you through that for a few minutes.

While we followed the Standard Review Plan, there are a number of considerations in the NuScale design that require some interpretation because of the differences in how NuScale's plant is put together. Examples listed on this slide include the smaller-diameter and shorter pipe lengths involved in the NuScale plant; the fact that high-energy line breaks inside the containment vessel are limited to 2-inch-diameter piping; the containment being designed, as Josh just described, to ASME Code Section -- sorry -- Section III, Class 1. The containment operates in a vacuum, and it's immersed in a pool of water.

High-energy line break and moderate-energy line break response is passive. It does not require electrical power or motor-operated valves.

There are no concerns from a GSI-191 standpoint with stripping of piping insulation. In fact, there's no piping insulation inside containment where the ECCS is located.

The largest-diameter piping is the main

steam piping and the feedwater piping inside the containment vessel that's designed to leak before break criteria. In the area under the bioshield, it's designed to Branch Technical Position 3-4 break exclusion criteria, which I'll discuss in a minute.

The shorter lengths of piping and the small footprint of the plant result in more congestive piping arrangements that have to be considered.

All of the containment isolation valves and decay heat removal system valves are located outside containment.

And, of course, we have to deal with the fact that the nuclear power modules are moved and the F12 modules in relatively-close proximity.

Next.

We divided our high-energy line break or piping break analysis into three regions of the plant. The reason for this is that those three regions have different environments. The systems that are higher energy, high or moderate energy, are different, and there are potential target SSCs that are different and have different characteristics.

The three particular regions are inside the containment vessel where we looked at specific locations and arrangements, and how those affected the possible pipe break locations and their consequences. As I said, main steam and feedwater are designed to satisfy leak before break. The remaining piping is relatively short length and it's all small bore. There is no containment penetration area inside the containment vessel.

We followed Branch Technical Position 3-4, "Guidance for Determining Break Locations". Basically, we account for small-diameter tourmaline breaks, and we design the remaining piping to avoid intermediate breaks.

In the area under the bioshield, there are specific locations and arrangements considered and possibility of break locations is excluded by designing to Branch Technical Position 3-4 in the containment penetration area to prevent terminal end breaks, and throughout the area under the bioshield to Branch Technical Position 3-4 to prevent intermediate-sized breaks. However, BTP 3-3 requires that we consider non-mechanistic breaks of the main steam system and feedwater system, which I'll discuss in a minute.

Finally, the third area --

MEMBER CORRADINI: This is inside the hut?

MR. KAUFFMAN: Yes.

MEMBER CORRADINI: Outside the containment, outside the hut?

MR. KAUFFMAN: Correct.

MEMBER CORRADINI: Okay.

MR. KAUFFMAN: Under the bioshield that you were asking about.

The third area of the plant that we evaluate is the reactor building. There's piping present in the pipe galleries and in lower floors for the CVCS system. The exact piping arrangements are a COL item, and therefore, we took an approach of doing bounding analyses to assure that a pipe break in any location, given that the arrangement is not yet determined, would, in fact, have acceptable consequences. And I'll describe how we do that next.

Non-mechanistic breaks. The largest pipe in the main steam system or the largest pipe in the NuScale plant is the main steam piping, which is 12-inch NPS. BTP 3.3 requires that a non-mechanistic break in the containment penetration area of 12 -- sorry -- of 1 square foot be considered. That's larger than the flow area of a NuScale main steam pipe. So, it's a little bit illogical to apply that.

We looked at the likelihood of main steam and feedwater system piping breaks and concluded that, because our piping was made of stainless steel, which is more corrosion- and failure-resistant, and also because it's smaller, we should still consider main steam and feedwater system non-mechanistic breaks, but we should scale the size.

Basically, what we did is we took the main steam piping size, compared that to main steam piping in large PWR plants, and scaled down to about from 1 square foot to 12 square inches for the non-mechanistic main steam break in the NuScale plant. We also have a scaled-down feedwater piping break.

So, we analyzed for those piping breaks under the bioshield. We don't have piping breaks, double-ended circumferential piping breaks because of the design to Branch Technical Position 3-4.

MEMBER CORRADINI: Can you say that all over again? All right. Let me say it back to you. So, you basically scaled off of piping sizes that are larger and got down from 144 square inches to 12?

MR. KAUFFMAN: Correct.

MEMBER CORRADINI: Based on what?

MR. KAUFFMAN: I was agreeing -- we did that because we considered our piping is less susceptible to corrosion-induced failure than the large PWR piping to which the non-mechanistic break normally applies.

MEMBER CORRADINI: Yes, qualitatively, I've got that part. I'm trying to understand how you got down by a factor of 12.

MR. KAUFFMAN: I tried to show it graphically here. The AP1000 main steam pipe maximum diameter I believe is 38 inches.

MEMBER CORRADINI: Right.

MR. KAUFFMAN: So, 38 inches. If you look at 1 square foot flow area in relation to the flow area of 38-inch pipe --

MEMBER CORRADINI: You ratioed it off of flow area?

MR. KAUFFMAN: Yes.

MEMBER CORRADINI: And then, one last question. And then, let's just leave that for the moment. And so now, this 12-square-inch pipe just comes apart and that's what is your non-mechanistic break?

MR. KAUFFMAN: It doesn't come apart in a circumferential break. It just --

MEMBER CORRADINI: But in an area?

MR. KAUFFMAN: Yes, you just open up a hole.

MEMBER CORRADINI: Got it. Okay, fine.

Thank you.

CO-CHAIR RICCARDELLA: You don't consider a guillotine break of the pipe?

MR. KAUFFMAN: Not under the bioshield because we design to meet the criteria for break exclusion of Branch Technical Position 3-3.

CO-CHAIR RICCARDELLA: Got it. And again, you did leak before break for the dynamic loading to eliminate having to consider the dynamic loading? This is just to consider the LOCA transient, basically, right?

MR. KAUFFMAN: Leak before breaks only applied inside containment. We're talking outside containment here.

CO-CHAIR RICCARDELLA: Ah, okay. Okay.

MR. KAUFFMAN: And I get to re-explain this again using this slide.

So, starting inside containment, we've excluded main steam and feedwater piping breaks by meeting leak-before-break criteria. We consider the terminal end breaks in accordance with Branch Technical Position 3-3, and we exclude intermediate breaks, in accordance with Branch Technical Position 3-3.

In the area under the bioshield, we

consider leakage cracks and non-mechanistic breaks. And in the rest of the reactor building, we basically consider any break in any location that are high-energy or where medium-energy piping exists.

CO-CHAIR MARCH-LEUBA: If you are moving one element to refueling position, one module, and it drops and it falls right against another one, will it be a possibility of breaking those lines? It's a lot of mass heatings on the side.

MR. KAUFFMAN: I am not knowledgeable about what our assumptions are on drops during movement. I have to defer to --

CO-CHAIR MARCH-LEUBA: We assume they happen with very low probability, but it's, technically, they can happen --

MR. KAUFFMAN: Correct.

CO-CHAIR MARCH-LEUBA: -- by 10 to the minus 11.

MR. KAUFFMAN: I mean, in the limit, if you have to consider it, then you might possibly have a break, but I did not look at that.

CO-CHAIR MARCH-LEUBA: That's one thing I would consider to be -- either that or the seismic loads -- okay.

MR. KAUFFMAN: I understand the question.

CO-CHAIR MARCH-LEUBA: I'll leave it to the guys who know more about this than I do.

MR. KAUFFMAN: Okay.

Next, please.

All right. I'm going to step through the various phenomena that result from a high-energy break from blast waves through subcompartment pressurization, and there are a number of stops in between.

So, starting with blast waves, when you assume an instantaneous opening of a break, you get a very rapid mass and energy dump into the surrounding ambient environment. Because of that assumption, you can form a blast or shockwave. We've concluded that only steam-filled lines can generate a blast. The fluid in a two-phase blowdown can't get out fast enough. There is no blast if the break opening time is more than a few milliseconds. So, in a way, it's an artifact of the assumption that we analyze for. And the NuScale smaller-diameter piping reduces the mass and energy output, and therefore, the severity of any blast wave that does form.

MEMBER CORRADINI: So, just let me ask, is this set of assumptions common?

MR. KAUFFMAN: It's --

MEMBER CORRADINI: In other words, steam only, two-phase not, smaller because of smaller pipe size, is that a set of assumptions for other blast wave analyses?

MR. KAUFFMAN: Well, we had the small piping. So, other people do not look at the smaller-diameter piping.

MEMBER CORRADINI: That part I get, but your first two assumptions, if only steam-filled lines generate a shockwave, two-phase discharges do not, and then, the break or the opening times.

MR. KAUFFMAN: The opening time is an NRC guidance item that you have to assume break opening time of no more than 1 millisecond unless you provide a technical justification. So, most break analyses assume 1 millisecond or instantaneous.

MEMBER CORRADINI: Okay, fine.

MR. KAUFFMAN: And as far as the steam-filled lines generating blast, I don't recall if other applicants have made that assertion.

MEMBER CORRADINI: Okay, fine. Thank you.

MR. KAUFFMAN: To do the blast wave analysis, we concluded that we needed to do 3-dimensional computational fluid dynamics. We used the CFX code. We qualified or verified and validated the use of the code using eight test problems. We did have a simplified plant geometry, but one that was representative of the important structures.

We developed a force-time history for three degasification line break cases inside containment and three main steam line break cases inside the reactor building. In all cases, the forces from the blast were very brief and acceptably low. They pass very quickly and die off fairly quickly with distance.

MEMBER CORRADINI: And probably you didn't worry about obstructions that would break apart the shockwave energy?

MR. KAUFFMAN: That's why we did 3-dimensional --

MEMBER CORRADINI: Oh, so you did?

MR. KAUFFMAN: -- CFD.

MEMBER CORRADINI: Okay, fine.

MR. KAUFFMAN: Because without the 3-dimensional CFD, if you do 2Ds --

MEMBER CORRADINI: Sure.

MR. KAUFFMAN: -- CFD --

MEMBER CORRADINI: It's symmetric.

MR. KAUFFMAN: Well, you get a shadow, and you don't get the wraparound that an actual blast wave does.

MEMBER CORRADINI: Okay. Thank you.

MR. KAUFFMAN: Next, please.

The next phenomena is pipe whip. In this case, the smaller-diameter and shorter-length piping in the NuScale plant is important again, because the impact energy depends on the fluid conditions inside the pipe and the length of piping. Smaller pipes have lower energy content. And the congestion of the pipes results in a shorter length that can whip, and therefore, basically, there's less of a windup and less kinetic energy on impact.

So, our process was to determine, first, if the fluid energy within the pipe was sufficient to form a plastic hinge and cause the pipe to whip. If it was, we looked at motion of the pipe whip in the plane, the pipe geometry. If any essential SSCs were too far away to be struck, we were done; they were out of range. If a large, robust SSC nearby served as a barrier and prevented the pipe from striking anything more vulnerable, we were also done. The best example of that is inside containment. If it runs into the reactor pressure vessel wall, a 2-inch-diameter pipe, which is the largest pipe that we have to consider a rupture for, is not going to continue. It's going to basically come to rest against the reactor pressure vessel. And finally, if an impact was possible, we looked at the impact force and would show that it is, in fact, acceptable.

Next.

One special case that we looked at was the piping gallery where there were a large number of parallel piping runs of main steam piping and feedwater piping. The Branch Technical Position 362 -- sorry -- Standard Review Plan 362 says that you do not have to consider that a pipe of a given size and pipe thickness causes a pipe rupture in a pipe the same size of pipe thickness or larger. In other words, a main steam pipe, if it rips, would not cause a secondary failure in another main steam pipe because they're the same size and thickness.

However, in the pipe gallery, we do have smaller-diameter piping in the main steam piping, like the feedwater system. So, we wanted to be sure that we bounded the effects of possible secondary ruptures. Therefore, when we did the analysis for blast waves, when we did the analysis for subcompartment pressurization, we looked at a main steam pipe failure resulting in a whip, causing a failure of a feedwater pipe or an adjoining main steam bypass line, which is also smaller, and used the blowdown characteristics from that secondary rupture added to the first, to make sure that, again, we're bounded for any future analysis that the COL applicant may do, once the piping arrangements are finalized.

Pipe whip summary. As I said, the impact force depends on the thrust force of the break exit, which is dependent on the size of the piping. The NuScale force, just because of the large --

MEMBER CORRADINI: Can you go back?

MR. KAUFFMAN: Yes.

MEMBER CORRADINI: We're just trying to understand the picture.

So, the thick pink is the missile shield or is the wall that acts as a missile shield? Or is the missile shield outside of that?

MR. KAUFFMAN: The missile shield is the outside wall of the building, correct?

MR. PARKER: Yes. That's the, well, that's the --

MR. KAUFFMAN: That's the wall of the pool, what you see there. So, that's in the center of the building.

MR. PARKER: So, farther down the page would be the exterior --

MEMBER CORRADINI: Okay. So, the exterior wall is outside of that? That's what I was --

MR. KAUFFMAN: The exterior wall is down here.

MEMBER CORRADINI: Okay. That's what we were trying to understand.

CO-CHAIR RICCARDELLA: This is a plant view here?

CO-CHAIR RICCARDELLA: Looking down.

MR. KAUFFMAN: Yes, looking down.

CO-CHAIR RICCARDELLA: Okay.

MEMBER CORRADINI: Okay. That answers the question. Thank you.

MR. KAUFFMAN: You're welcome, yes.

Pipe whip, as I said, the impact force depends on the thrust force of the break exit. Because of the smaller-diameter pipe, NuScale's thrust force is about 5 percent of that of the large main steam pipe breaks in large PWRs. Again, the short length of the pipe limits the impact energy. We know the piping arrangements in the containment vessel. So, we analyzed those for explicit pipe with motions. We made conservative assumptions in the reactor building.

Basically, we're dealing with two types of targets. We've got metallic components in pipes in the containment and under the bioshield and we have concrete structures under the bioshield and in the reactor building. For the concrete structures, we looked at penetration of the walls due to pipe whip, using a formula from Sandia by Young, and we also looked at spalling, and concluded that the concrete surfaces would not be penetrated by pipe with impact.

MEMBER SKILLMAN: Hey, Storm, on your first caret there, thrust force at break exit, 5 percent. Isn't the steam pressure here at about 550 psi?

MR. KAUFFMAN: Yes.

MEMBER SKILLMAN: And let's take a normal PWR. It's 1050?

MR. KAUFFMAN: Yes, that also plays into it.

MEMBER SKILLMAN: So, this is not really 5 percent. It's 50 percent for the same diameter pipe?

MR. KAUFFMAN: No. No, because the thrust force is integrated over the cross-sectional area of the pipe. So, it's the pressure times the area.

MEMBER SKILLMAN: If I have a 2-inch pipe at 1,000 psi, a 2-inch at 500 psi, one's 50 percent of the other.

MR. KAUFFMAN: Right, but we --

MEMBER SKILLMAN: You're trying to tell me this is not a 36-inch line; it's a 10-4. Got it.

MR. KAUFFMAN: Okay.

MEMBER SKILLMAN: I think that's a nice statistic, but it's a stretch, just saying as a steam plant operator of many years.

MR. KAUFFMAN: Well, it's important because the image you can have is taking a chopstick and hitting something with it and taking a log and hitting something with it. The NuScale pipe is more like the chopstick; it carries less energy.

MEMBER SKILLMAN: I understand what you're communicating.

MR. KAUFFMAN: Okay.

MEMBER SKILLMAN: I'll grant you that. I'm just saying that there's another way to look at it, in which case it's about half, not 5 percent, not 1/20th.

MR. KAUFFMAN: I'll move on.

MEMBER SKILLMAN: Thank you.

MR. KAUFFMAN: Thank you.

Next slide.

Okay. The next one I wanted to discuss is jet impingement. Impingement pressure energy depends on the fluid conditions, the degree of spreading of the jet, once it exits the pipe break, and the distance from the break exit to the target.

There are different behaviors, whether the jet is issuing a steam or it's two-phase liquid or, for that matter, as subcooled, highly subcooled liquid. In the case of steam, I'll talk about that in a minute. In the case of two-phase liquid, we use the approach of NUREG-2913 to estimate the pressure-versus-distance relationship and the spread or zone of influence of the jet.

For the overall process, the steps we went through were to determine the pressure threshold for damage to essential SSCs. The big difference that the NuScale plant has is that we don't have insulation stripping as a concern. And therefore, we don't have to worry about a jet delivering a few psi to some component and stripping insulation from it. Instead, we have hard targets, and those have a lot more robust capability to withstand the pressure forces of a jet.

In looking at the zone of influence, we took a conservative approach inside containment. We assumed that the zone of influence is the hemisphere in front of the original location of the pipe. So, if the pipe was pointing straight forward, we considered a zone of influence anywhere in the hemisphere forward-pointing.

The jet spreading half-angle is something that the Standard Review Plan requires be addressed. There's concerns that the ANS standard that is referenced, or has been referenced for years, is non-conservative. It specifies, roughly, a 45-degree spreading angle. We used 30 inside containment. That was more conservative from the standpoint of maintaining a higher pressure versus distance downstream than if you used the 45. It was also supported by the CFD calculations we did for blast effects, where we found that the spread of the jet as it formed after the blast departed was on the order of 60 degrees.

When the jet issues from the pipe exit, if it impinges upon a large structure such as the containment vessel wall, then, basically, it stops. It can't go through the wall. And if there any essential SSCs that are close enough to be within the range of the jet, then we have to consider whether or not the impingement pressure is acceptable.

Next, please.

MEMBER BALLINGER: By the way, it's not

2913. It's NUREG-CR-2913.

MR. KAUFFMAN: Yes. Thank you.

Jet shape and pressure I'll review then. I described the zone of influence for inside containment. For pressure, I also mentioned 30 degrees. I forgot to mention that, after five diameters downstream, we take that back to a 10-degree spread, so it stays more concentrated.

In the reactor building, the arrangements for the piping do not get finalized. So, a jet could end up pointing anywhere, and that's what we assumed. We also assumed that the target SSC is only 2 L over D from the break exit. So, there's basically no reduction in pressure with distance.

Next.

There's one specific portion of 362 that requires evaluation of the potential for amplification of the pressure forces due to a resonance. There's been a lot of research on this subject. It's pretty much concluded that to have a resonance, you must have a symmetric or axisymmetric jet and you have to develop what one set of researchers called a phased lock, which basically is a stable condition that you form a standing wave between the exit of the jet and its target.

In a high-energy line break in a plant like NuScale's, there are multiple reasons why resonance will not occur. Those include the fact that the fluid is not a single-phase gas. It's condensable fluid, and the proximity to condensation or evaporation results in a damping effect that basically takes energy out of the jet. The distorted exit geometry violates the axisymmetric condition and prevents developing a phase lock. The absence of a large, flat impingement surface limits the ability to return the energy to the break exit nozzle area, and also eliminates the possibility of a resonance.

The instability of the jet, the fact that the pipe is whipped and is moving again disrupts the stability necessary to form a resonance. There are intersecting obstacles that disrupt the jet, and there also are fluid spray conditions that reenter the jet to disrupt it. And finally, there's a frequency mismatch with the structures. The jet has a frequency on several kilohertz, the tens of kilohertz range, which is not going to excite the structures which have natural frequencies on the order of hertz or tens of hertz.

Next, please.

DR. SCHULTZ: Storm?

MR. KAUFFMAN: Yes?

DR. SCHULTZ: This last evaluation with the bullets, is that a rationale that demonstrates that the research papers don't apply or have you done an analysis that demonstrates it because of these elements?

MR. KAUFFMAN: The research papers apply to single-phase gas jets. There is no research paper that I could find that ever reported finding a resonance or amplification condition for a steam jet.

MEMBER CORRADINI: For a what?

MR. KAUFFMAN: A steam jet.

MEMBER CORRADINI: A steam jet?

MR. KAUFFMAN: Anything that involves being close to saturation conditions. And, in fact, there's been a lot of work because the gas jets cause problems called screech and are something that the industry has been trying to avoid in a number of applications. There's been a lot of research on how to avoid the amplification, and they do things such as put little tabs on the exits of the nozzle to disrupt the axisymmetric conditions. They also inject fluid or liquid into a single-phase gas jet to help damp the oscillations. So, the things that the researchers have done to prevent or reduce the amplification in single-phase gas jets are the sort of things that exist naturally in a high-energy line break.

DR. SCHULTZ: And so, here are a number of reasons/rationale why it's not a concern for the application here?

MR. KAUFFMAN: Yes.

DR. SCHULTZ: Thank you.

CO-CHAIR RICCARDELLA: Excuse me. Could you go back a couple of slides to slide 46, please?

MR. KAUFFMAN: Okay.

CO-CHAIR RICCARDELLA: That top bullet, would you explain the meaning of NPS 2 only; therefore, no longitudinal breaks?

MR. KAUFFMAN: Okay. Inside the containment vessel, there is main steam and feedwater piping that is leak before break --

CO-CHAIR RICCARDELLA: Okay.

MR. KAUFFMAN: -- leaving only 2-inch NPS piping.

CO-CHAIR RICCARDELLA: Yes, yes.

MR. KAUFFMAN: By the guidance for when you have to consider longitudinal breaks, 2-inch or NPS 2 piping does not have to be considered for a longitudinal break. So, we considered circumferential offset breaks, but not longitudinal.

CO-CHAIR RICCARDELLA: That's NRC guidance?

MR. KAUFFMAN: Yes.

CO-CHAIR RICCARDELLA: Okay. What is it, a Reg Guide or an SRP?

MR. KAUFFMAN: I think it's in Standard Review Plan 362. It may be in the Branch Technical Position 3-4. Right now, I'm not sure which.

CO-CHAIR RICCARDELLA: Okay. Thank you.

MR. KAUFFMAN: Okay. So, in summary, to go back over jet impingement, the jet impingement depends on conditions that are less severe for the NuScale plant design. We've looked at spreading angle for the underexpanded jet and the distance to the target SSCs. Concluded that there is acceptable performance, in other words, no potential for unacceptable damage to essential SSCs inside containment. There are no jets under the bioshield because non-mechanistic breaks don't have to consider jets. And there's no dynamic amplification of jets.

Finally, in the reactor building, we've done a bounding analysis to account for whatever final piping arrangements the COL applicant comes up with.

The last area of effects of subcompartment pressurization. For this, we performed GOTHIC analyses for the various high-energy line breaks in different areas of the plant. Inside the containment vessel, any high-energy line break transient is bounded by the ECCS initiation transient, and therefore, did not require further evaluation.

Under the bioshield, the high-energy line breaks are limited to the non-mechanistic breaks of the main steam and feedwater system. We provided passive venting. That was what was being shown on slide 31 with the louvers on the bioshield. So, that passive vent path limits the temperature and pressure buildup under the bioshield. There's a structural limit of 1 psi that's met, and the temperature envelope is used, then, to determine the environmental qualification envelope for equipment under the bioshield.

In the reactor building, we also have vent paths to limit overpressurization or differential pressures across walls and floors. We looked at various breaks for the piping systems in different locations in the building, and have shown that the structural limit of 3 psid is met throughout the building.

Kind of the wrapup of the wrapup for the pipe rupture hazards assessment, the break locations in the NuScale plant are limited by application of break exclusion criteria from Branch Technical Position 3-4 and application of leak before break.

We evaluated blast effects using 3-dimensional computational fluid dynamics. We evaluated pipe whip, jet behavior, including jet spreading and jet impingement. We've looked at subcompartment pressurization using GOTHIC.

And finally, we have three COL items that require that the final pipe rupture hazards assessment confirm our conclusions and that the final arrangements determined by the COL applicant also satisfy our criteria.

One remaining slide to complete Chapter 3 is leak before break. NuScale, as I said, applied leak before break to the 12- and 4-inch, 4- and 5-inch large-diameter piping inside the containment vessel. This piping is all, basically, stainless steel 304 dual-certified, which is corrosion, erosion, and fatigue, and, in general, failure-resistant.

In accordance with NRC criteria, we applied a margin of 10 on detectable leak rate and a margin of 2 on flaw size. We detect leakage by changes in containment vessel pressure and by changes, accumulation of condensate in the containment evacuation system sample vessel that's constantly drawing a suction on the containment vessel.

And at that point, we're into the acronyms.

CO-CHAIR RICCARDELLA: Storm, compared to a normal PWR or BWR, your leak detection capability is very small, right?

MR. KAUFFMAN: Is capable of detecting very small leaks, yes.

CO-CHAIR RICCARDELLA: Very small leaks, yes.

MR. KAUFFMAN: The containment evacuation system sample vessel is kind of an integrating measure. It's constantly accumulating condensate. So, you can measure the rate of rise. And the containment vessel pressure, because we're at a vacuum, it's very sensitive to anything that puts more mass in the containment.

CO-CHAIR RICCARDELLA: So, refresh my memory, what is the gpm detectability that you claim?

MEMBER CORRADINI: If it's public.

MR. KAUFFMAN: That I don't know.

CO-CHAIR RICCARDELLA: Okay. It's not public?

MEMBER CORRADINI: If it's public, do you remember what it is?

MR. KAUFFMAN: That's right.

MEMBER SKILLMAN: I think it is public, and it's in, actually, 5.2, and this is what is written: "The containment pressure monitoring instrumentation is capable of detecting a minimum leak rate from the RCS of .007 gpm and the containment pressure will increase" -- and this is the kicker -- "containment pressure will increase .1 psi in less than one minute with a 1-gpm leak from the RCS."

This is where I was digging. This is 5.2.5.1, Revision 2. So, it's really, really tight, remarkably tight. And it's that vessel that Storm just discussed, the CES, the evacuation system vessel, which is a constant draw into the -- I guess it's a flask of some sort. It's measured --

CO-CHAIR RICCARDELLA: So, that's the more sensitive of the two?

MEMBER CORRADINI: You're going to turn to somebody else to tell you the answer to that?

MR. KAUFFMAN: Yes. Yes, I'm not the guru on the containment evacuation system.

CO-CHAIR MARCH-LEUBA: I would expect the pressure to be the most sensitive of all because you're under vacuum. You put a gram of water and, poof, the pressure goes up.

The problem is if you develop a second leak, then the pressure is already high. You have your vacuum system going and you cannot detect it. So, you have to rely on the drain. One leak, pressure; two leak, drain.

MEMBER SKILLMAN: Yes, which one is it, is the question. You know, what's going on in there? You can get one or two.

CO-CHAIR MARCH-LEUBA: Once you have one, essentially, you're blind. You don't know if you have two or three or four.

MEMBER SKILLMAN: Or a simmering relief valve.

CO-CHAIR MARCH-LEUBA: Yes, but you're in a blind condition.

MEMBER BLEY: You're not so blind that you don't know there's a leak.

MEMBER SKILLMAN: Pardon?

MEMBER BLEY: I said, you're not so blind that you don't know there is a leak.

MEMBER SKILLMAN: That's correct, yes.

MEMBER CORRADINI: Other questions for NuScale?

Oh, excuse me.

MR. HOUGHTON: I didn't hear a specific question there. I was just stepping up in case we had a point-specific on the containment evacuation.

(Laughter.)

MEMBER CORRADINI: Yes, stand at the ready.

Any questions? Any further questions by the members?

DR. SCHULTZ: Just one question. Storm, you talk about the three COL items. The way you describe the analyses and the evaluation you've done, you really don't expect that it will be difficult -- that the arrangements would be any different than it would be standard practice for the COL applicant to check the boxes, if you will, to demonstrate that the evaluations are met?

MR. KAUFFMAN: Correct. We did as much as possible at this stage to assure that the COL applicant would have a clean path to show acceptance.

DR. SCHULTZ: Good. Thank you.

MEMBER CORRADINI: Mike?

MR. SNODDERLY: I just wanted to apologize; we didn't have the sign-in sheets out the first thing this morning. So, they're out now. And so, if people could remember in the audience from the public and from staff to sign the sign-in sheets, we'd appreciate it.

MEMBER CORRADINI: Okay, okay. All right.

Other members, questions?

Otherwise, we'll take a break. 10:30. Thank you.

(Whereupon, the above-entitled matter went off the record at 10:14 a.m. and resumed at 10:30 a.m.)

MEMBER CORRADINI: Do you want to start us off?

MS. VERA: Sure, sure.

Good morning, everyone. My name is Marieliz Vera. I'm Project Manager for Chapter 3 for this application.

Today, we're going to present a Chapter 3, "Design of Structural Components, Equipment, and Systems".

The first couple of slides is the list of technical reviewers. Chapter 3 is a long one. So, we have a couple of reviewers here. We have most of them in the audience.

We're going to focus our presentation today in sections that have open items or that we know that the ACRS has shown some interest. So, here we have the sections that are not going to be presented because they don't have any open items. But if you have any questions, here we have the staff to respond to any questions you have on any of these sections.

So, we're going to start our presentation with Section 3.6.3, and we're going to have a couple of groups presenting. So, you're going to see a little bit of movement here. We're trying to make it as smooth as possible.

And the first section we're going to present is leak before break, evaluation procedures, with Eric Reichelt.

MR. REICHELT: Good morning.

My name is Eric Reichelt, and I'm a Senior Materials Engineer in the Office of New Reactors. This week I'm in the Division of Engineering, Safety Systems and Risk Assessment, the Materials and Chemical Engineering Branch. I am the technical reviewer for Section 3.6.3, "Leak Before Break Design," for the NuScale DCD.

I would like to give you a brief presentation on the work that has and will be performed for this particular section. The LBB approach for new reactors to use bounding limits established during the design certification phase, and to verify the final as-built design during the construction phase using ITAAC, was approved by the Commission in its SRM for SECY-93087.

The use of LBB applications has been utilized in previous design certification applications. However, the leak-before-break evaluation for the NuScale design posed some unique challenges because of the piping diameters and complex nature of the piping layouts involved. In addition, the sensitivity of the NuScale leakage detection system, which you heard about in Section 5.2.5, is also significantly different from traditional large lightwater reactor designs and will play a major role in a NuScale design being able to meet established LBB criteria.

Because of these complexities, the technical review is being performed by Engineering Mechanics Corporation of Columbus, otherwise known as EMC2, our contract for this review, who as an organization has decades worth of experience with the NRC-endorsed approach to leak before break.

And at this time, I would like to introduce the EMC staff who has assisted the NRC for the review of NuScale LBB analysis, and in my opinion, are the leading experts in the field of leak before break and fracture mechanics.

To my right I have Dr. Gery Wilkowski, who's the President of EMC2. Next is Dr. Prabhat Krishnaswamy, Senior Research Leader and principal investigator for the NuScale project. And finally, Dr. Mohammed Uddin, who is the principal engineer.

Next slide.

The NuScale design requested to apply leak-before-break methodology to main steam piping and the feedwater piping systems. The main steam piping is NPS 8 and NPS 12. The feedwater piping is NPS 4 and NPS 5.

Unique aspects for NuScale is the curved piping system and making sure fabrication, i.e., pipe-bending, limits cold-working to an acceptable limit, and the methods and criteria to evaluate LBB are consistent with the guidance in SRP-363 and NUREG-1061, Volume 3.

Next slide, please.

The staff has reviewed the applicable NuScale DCD subsections in Section 3.6.3. We're reviewed the DCD references for applicability and use. We've held public meetings with the NuScale staff about technical issues and RAIs leading to proposed DCD markups. The technical issues and response by

NuScale to RAIs were acceptable and were, therefore, closed.

The staff is currently reviewing the methodology and will perform a confirmatory analysis on the feedwater system proprietary information and data which has been provided by NuScale, and EMC2 is in the early stages of reviewing the information and eventually performing the confirmatory analysis.

The staff will also review the methodology and will perform a confirmatory analysis on the main steam lines when NuScale provides the proprietary information and data, which is tentatively scheduled for July 15th, 2019. Based on the confirmatory analysis, this is being tracked as a confirmatory item.

This concludes our presentation at this time.

MEMBER CORRADINI: Questions by the Committee?

Okay. Keep on going. I love your name. Sorry.

MS. VERA: The structural people, please come.

Okay. Now we're going to continue with mostly structural, Section 3.5.1.3, "Turbine

Missiles," and Section 3.5.3, "Barrier Design Procedures," that John Honcharik and BP Jain will start.

MR. HONCHARIK: Good morning.

My name is John Honcharik, and I'll talk about turbine missiles. Basically, the regulatory basis for it is GDC 4 requires the SCCs, important safety be protected against environmental effects, including missiles.

The safety-related, risk-significant SCCs in the NuScale design are located in the rack building and the control room building. The turbine generators are unfavorably oriented with respect to the reactor building and control room building.

To meet GDC 4, NuScale proposes to use the installed or existing structures to protect these essential SCCs.

MEMBER BLEY: John, is this, indeed, the first time anyone has used that approach for a turbine missile protection?

MR. HONCHARIK: Yes. To my knowledge, yes, that is correct.

MEMBER BLEY: Okay.

MR. HONCHARIK: It's the first time.

Next slide.

And right now, basically, I'm looking at the turbine missile parameters, and BP, to my right, is going to be looking at assessing how those missile parameters will affect the structures to protect the essential SCCs.

And for my section, there's one open item. Basically, NuScale didn't use the full spectrum of turbine missiles, including the size, weight, and speed. They said they were using a turbine blade, which basically isn't a missile because you could throw blades most of the time.

We're concerned with the rotor. And we basically said up to half of the last stage of the rotor.

NuScale --

MEMBER CORRADINI: What is that based on? Half of the last stage of the rotor, and what's the weight of that?

MR. HONCHARIK: Yes, basically, from past experiences of turbine shafts failing, there are pretty big sizes, including up to a half of the rotor, half of the last stage of the rotor.

MEMBER CORRADINI: Right, but these are monoblock rotors?

MR. HONCHARIK: Correct. Well, at this

time, they are. When we first proposed this, we weren't sure what they were.

MEMBER CORRADINI: All right. Okay, but what I'm trying to get at is, I'm trying to unwrap. So, the open item is what they've assumed so far is not considered bounding or it's not an appropriate spectrum but it is bounding? That's what I'm not sure about.

MR. HONCHARIK: It was not an appropriate spectrum.

MEMBER CORRADINI: Okay.

MEMBER BLEY: And therefore, was not bounding.

MR. HONCHARIK: Right.

MEMBER CORRADINI: So, their 3,000-pound portion of the rotor is not big enough?

MR. HONCHARIK: Well, they just submitted that a couple of weeks ago.

MEMBER CORRADINI: Oh, okay. Excuse me. Okay, fine.

MR. HONCHARIK: Yes.

MEMBER BLEY: You have not yet had a chance to really review that, have you?

MR. HONCHARIK: No, we have not.

MEMBER CORRADINI: Okay. That helps me.

I didn't put the 2 and 2 together. Thank you.

MR. HONCHARIK: Yes, initially, they came in with an analysis. We did an audit, and they only had an analysis of the blade, and a blade with a part of the rotor, but it was only like 30 pounds.

MEMBER CORRADINI: Sure. Got it.

MR. HONCHARIK: So, we said, no, that's not a turbine missile.

MEMBER CORRADINI: Okay.

MR. HONCHARIK: You've got to include something -- you have to come up with -- you know, because part of this turbine missile for barriers, you have to come up with an actual missile. So, what is the bounding missile. Okay? Based on some operating experience, you could see that it could be up to at least the last stage of the rotor. And the last stage is usually the larger piece.

MEMBER SKILLMAN: John, are you convinced that that disc portion, which is the last piece of the rotor, is bounding?

MR. HONCHARIK: Yes, pretty much. I mean, based on what we've seen and, also, the size of their rotor -- you know, it's only a 50-megawatt. And usually, the last stages are a lot bigger than the rest of it.

MEMBER SKILLMAN: All right. But you're comfortable that that piece is bounding? That's where I'm going with my question.

MR. HONCHARIK: Yes.

MEMBER SKILLMAN: Okay. So, the path forward is demonstrating that that piece not penetrating is sufficient for this adverse design?

MR. HONCHARIK: Yes, not penetrating or backscabbing, which BP will talk about that.

MEMBER SKILLMAN: Okay. Thank you, John.

MEMBER CORRADINI: Does that consider the generator, too?

MR. HONCHARIK: No, this is just the turbine, not the generator.

MEMBER CORRADINI: Okay.

MEMBER BALLINGER: Has there ever been a failure of a monoblock rotor in any way?

MR. HONCHARIK: Well, monoblock rotors have only been, I guess -- what? -- maybe 15 years or so, 20. So, right now, there has not been any that we know of. I mean, for nuclear applications.

MEMBER CORRADINI: But nothing makes this unique to nuclear in terms of being a monoblock rotor. They're used in other -- in fossil facilities, I would assume, I thought.

MR. HONCHARIK: They've been using --

MEMBER CORRADINI: Monoblock rotors in small fossil plants? No? I thought so.

MR. HONCHARIK: Yes, they have.

MEMBER CORRADINI: I'm looking at Ron because --

MR. HONCHARIK: Yes, they have.

MEMBER BALLINGER: Yes, I just don't know that there's ever been a failure.

MEMBER BLEY: I don't, either, but the Applicant chose to not look at the probabilistic side of this and to look at the missile protection by the barriers. So, I think --

MEMBER BALLINGER: Because there have been breaks -- breaks generally, one in like 1960, one in, say, the 1970-something, where the composition of the material and the cleanliness of the material underwent big changes.

MR. HONCHARIK: Right.

MEMBER BALLINGER: And so, if you look at failures of rotors as a function of time, and you put those in those bins, you'll find out that failure of rotors after 1975 for a manufacturer, very rare.

MR. HONCHARIK: Well, yes, but a lot of that had to do with implementing turbine rotor integrity, making sure the processes that you use to make the rotor --

MEMBER BALLINGER: Yes.

MR. HONCHARIK: -- make sure you didn't have impurities, right? Because a lot of them, you know, Hinkley and stuff had mainly sulfide inclusions, right? And that's what basically -- and that was a solid rotor, right?

MEMBER BALLINGER: That was lousy material.

(Laughter.)

MR. HONCHARIK: Correct. But, right now, NuScale does not credit anything.

MEMBER BALLINGER: Okay.

MR. HONCHARIK: They have no overspeed protection that is credited.

MEMBER BALLINGER: Yes.

MR. HONCHARIK: There is no material properties that are credited, inspections, or anything.

MEMBER BALLINGER: Well, it's extremely conservative. So, their analysis would be extremely conservative.

MR. HONCHARIK: Right. Yes. So, that's why, basically, they're assuming it's going to fail, and now you have to have that missile to see whether or not it could be prevented to penetrate or backscab, and so on, on the barrier.

And like I said, we just received this. So, right now, we're currently reviewing this information. And basically, it is that RAI responses. There's no Technical Report attached to it. So, if you look for a Technical Report, it's probably not going to be there.

And I don't know if you had any other questions on the turbine missile size.

MEMBER BLEY: Well, you're looking at me --

MR. HONCHARIK: Yes.

MEMBER BLEY: -- and I had a lot of questions. But I think, until we see those enclosures -- and we just learned there's about eight enclosures, and no idea which of those have the technical information. So, we haven't seen any of it yet.

MR. HONCHARIK: Yes.

MEMBER BLEY: We need to look at that first.

MR. HONCHARIK: Okay.

MEMBER BLEY: If you have anything to say about what's in those enclosures and where you stand, that would be interesting, but I don't think you need to do that yet, if you haven't --

MR. HONCHARIK: Yes, right now, we really haven't had time to look at it, other than it's responses to the RAIs and not a Technical Report.

MEMBER BLEY: This is more a process question. Given this is the first time anybody has taken this approach, I guess I'm a little surprised they didn't, and you didn't ask for a Technical Report on this, but I guess you have the analysis now to look at.

MR. HONCHARIK: Well, I think part of reviewing this, part of the RAI was provide the basis for this, okay, determine the spectrum missiles and, also, the analysis of the barrier, and document it and provide the analysis. So, I think we're still looking for that. Okay? So, it's not to say that, you know, because, typically, for even a probabilistic way, you have a Technical Report that supports that basis.

MEMBER BLEY: Yes.

MR. HONCHARIK: So, we're going to have to look and to see whether or not they have sufficient information. If not, we're going to have to ask for more, whether that's a Technical Report or they have to provide more information.

MEMBER BLEY: We look forward to hearing about it after you've had a chance to look and find out where it's going.

MR. HONCHARIK: Okay. And with that, I turn it over to BP to address the battery analysis.

MR. JAIN: Okay. So, this is BP Jain, obviously, and I'll be talking about the barrier design for the postulated missiles. There are three missiles NuScale has postulated, as NuScale has said before. One is a turbine blade, which weighs about 32 pounds. The second to the turbine blade was a rotor fragment, 52 pounds, and half of the last-stage rotor which weighs about 3,000 pounds.

MEMBER BLEY: Can you explain to us why there's different velocities used for all three cases?

MR. JAIN: No, I cannot. Structurally, our job is just to look, given a missile, what it will do to the barrier. So, that's where I come in.

MEMBER BLEY: So, who are the guys who look at whether the missile's postulated correctly?

MR. JAIN: My friend on the left.

(Laughter.)

MEMBER BLEY: Okay.

MR. HONCHARIK: Yes. I was going to do, I was doing the turbine missile spectra, which includes the weight, velocity, and speed. So, I think, right now, I still haven't looked at it. But, depending on -- the blade is further out. So, it's actually spinning at a little higher RPM, I mean a higher velocity.

MEMBER BLEY: Wait a minute.

(Laughter.)

MR. HONCHARIK: The same RPM, but a higher velocity --

MEMBER BLEY: Okay.

MR. HONCHARIK: -- because of the radius, the distance out.

MEMBER BLEY: Yes.

MR. HONCHARIK: So, that's why I think you'll see different speeds.

MEMBER BLEY: So, that's a blade velocity on the first two?

MR. HONCHARIK: Correct.

MEMBER BLEY: And they said they did it at normal running speed at 150 percent and I think 220 percent, or something like that?

MR. HONCHARIK: Yes, they did like a sensitivity analysis. I think they said their design overspeed is 160 percent, which would be destructive overspeed.

MEMBER BLEY: Uh-hum.

MR. HONCHARIK: So, that's what pretty much we're looking for as the bounding case, but they also went up to like 220 percent.

MEMBER BLEY: Okay. So, you have all of those to look at? Thank you.

MR. JAIN: Okay. So, given these missiles, the staff used the guidance in 3.5.3 to the design.

MEMBER BLEY: I'm sorry, this is just -- it's an interesting point, because if that's the proposed destructive overspeed, but nobody's oversped one of these and caused it to break, if, in fact, it holds together better than you expected, it doesn't come apart until a much higher RPM, it's sure not conservative to just look at that destructive overspeed.

MR. HONCHARIK: Correct, and that's why part of the RAI was provide your basis for determining their design overspeed. Because, like you said, depending on the material that you're using, you could have a higher destructive overspeed. If it's a crappy material, you may have a lower destructive overspeed versus a better.

MEMBER BLEY: Okay.

CO-CHAIR MARCH-LEUBA: And how are you going to handle that? Because this is like the better you design your turbine, the worse your building is going to be, right?

MR. HONCHARIK: Right.

MEMBER BLEY: So, you take no credit for the overspeed trip, yes.

MR. HONCHARIK: Right, or the materials. But they did do a sensitivity analysis up to 220, which more than likely 220 is extremely high. I mean. I don't think they've had one or two destructive overspeeds where it was, I think, to 165, or something like that.

MEMBER BLEY: But all of those were keyed --

MR. HONCHARIK: Most of those were keyed, correct.

MEMBER BLEY: I think not most. I think all of them.

MR. HONCHARIK: Right. Well, for destructive overspeed.

MEMBER BLEY: Yes.

MR. HONCHARIK: There have been others that, like the other ones where I said that it was due to material.

MEMBER BLEY: Yes.

MR. HONCHARIK: And it actually flew apart because of the inclusions and lower fracture toughness.

MEMBER BLEY: Uh-hum. Okay.

MR. JAIN: Okay. So, the staff used the guidance and started with 3.5.3 to review the better design procedure for local and global loading effects. And acceptance criteria, whether it meets the barrier design is primarily to see if the wall is thick enough to provide the backface scabbing.

Now in its SER, the staff is required to make the following review findings: one, that the procedure used for barrier design for the impact of design basis turbine missiles are acceptable. And acceptance criteria is, obviously, the staff guidance is one of the documents staff uses for that.

And the other part is that information presented in the DCD provides a reasonable assurance that the reactor control building walls provide adequate protection to essential SSCs.

So, those are the statements, the findings we have to arrive at in our Safety Evaluation when we are all done.

When we were looking at it, we did not

find complete and technically-sufficient information in the DCA. So, the staff asked questions, and we have a submittal and the staff is going to be reviewing that information.

MEMBER BLEY: So now, instead of just using the previous guidance you had, you're going to have to have somebody, if I understood them correctly, look at their finite element analysis.

MR. JAIN: I'll come to that. Okay?

MEMBER BLEY: Okay. You're going to come to that? Very good.

MR. JAIN: Yes. So, on the next slide, I'm going to talk about some of the overview of the key design issues, what makes this open item.

Well, one of the key reasons, obviously, we didn't have the information to look at. So, we can't come to a conclusion in terms of a safety finding. But whatever we looked at so far, there were certain areas which staff had concerns with. And so, I'll go one by one.

One, the finite element procedure that NuScale had used to calculate the penetration depth. Staff has never looked at it before. There's no precedence of using that approach for better design for these high-speed missiles. So, staff will review the validation and verification of the procedures, the approach, and the computer program used to arrive at those conclusions.

The other issue staff have, that the local and global damage assessment was not provided for all the three missiles. So, the staff did look at -- they did provide the local damage for the first two missiles, but there was no global assessment.

MEMBER CORRADINI: What do you mean by a global assessment?

MR. JAIN: Global assessment means -- local is like punching and scabbing. It's a local panel. But when you talk about global, it's when you apply the force. There has to be an energy balance. So, you apply kinetic energy, half mv-squared that needs to be dissipated.

MEMBER CORRADINI: Somewhere?

MR. JAIN: And so, deformation of the overall --

MEMBER CORRADINI: So, you want to know what went in equals what --

MR. JAIN: Right.

MEMBER CORRADINI: -- got somewhere else?

MR. JAIN: Exactly.

MEMBER CORRADINI: Okay, fine. Thank you.

DR. SCHULTZ: Bhagwat, are you anticipating a need for you to perform confirmatory analysis or do you feel that, in what you've seen so far that is being supplied, you can do the review of what the Applicant has done and --

MR. JAIN: Well, staff intends to review what the Applicant has presented. At this point, we don't plan to have any confirmatory analysis, unless we find things which really require that.

DR. SCHULTZ: You've got a lot of careful review that needs to be done --

MR. JAIN: That is correct.

DR. SCHULTZ: -- given that it's a new application, and you've got the validation and verification to look at, too.

MR. JAIN: New application, new approach. There's no precedence.

MEMBER BLEY: And in all the parameters in that finite element method, there's going to be a lot of uncertainty. So, how they dealt with that is something I hope you're looking at closely.

MR. JAIN: Right. And we have asked some of those questions, and I understand that NuScale has responded to those as well.

The other concern we had, or I guess the design issue was the penetration depth. And that, it's counterintuitive to the point that what they have reported, that if you have a softer missile or deformable missile, the penetration of that is reduced by one-third compared to the missile that was rigid. Now, granted, there will be a reduction in penetration depth because the missile gets deformed and not all the energy is input to the barrier. But, from what we have seen in the literature, generally, the reduction is of the order of 30-40 percent. Again, that's just a number. But this is almost 300 percent. So, we need to understand whether their models are okay or the validation with the test results, and so on, that they all makes sense.

MEMBER SKILLMAN: BP, do you have the tools, does the staff have the tools that are required to come to a determination on this topic?

MR. JAIN: Well, I mean, the NRC has infinite resources. So, to answer your question, we have the resources, but whether we will go that route, it's premature at this point.

MEMBER CORRADINI: But I just want to make sure I understand the bullet. The bullet is I have a 60 percent reduction in depth of penetration versus what you see in the literature of 30 to 40 percent?

That's what I hear you saying.

MR. JAIN: The concern we have --

MEMBER CORRADINI: If I'm one-third of the depth, that's a 60 percent reduction, a 66 percent reduction, versus what we see in the literature of 30 to 40 percent?

MR. JAIN: Well, it's not 66 percent. It's a lot more. For a rigid missile, the penetration is almost 60 inches. And for a deformable missile, it's 20 inches. It's one-third.

MEMBER CORRADINI: Right. I understand. Okay. What you use as the denominator is not what I would use as the denominator.

MR. JAIN: Oh, okay. All right.

MEMBER CORRADINI: That's all I'm trying to get at.

MEMBER BLEY: I'm just a little curious because I haven't thought about this, either. From what they told us, they assumed there is no reinforcing steel. So, they're looking at just a concrete wall being hit by a solid steel missile.

MR. JAIN: Uh-hum.

MEMBER BLEY: And I don't have a clue about how deformable a solid steel missile is running into concrete with no reinforcing that.

MR. JAIN: Well, that's what we want to understand because rigid and deformable is a relative term compared to the barrier.

MEMBER BLEY: So, there's got to be some good basis for it.

MR. JAIN: Right. When a missile becomes rigid compared to the barrier, that's what we intend to look at here.

MEMBER BLEY: I look forward to seeing what you come up with.

MR. JAIN: Because that's where we are at. That gives you a snapshot of some of the design issues we are looking at. So, we will review this design information and conduct audits, and whatever we need to, to make safety findings.

MEMBER BLEY: Okay. So, we won't see the results of your work until sometime later when you come back with an SER with no open items, I guess?

MR. JAIN: That is correct.

MEMBER BLEY: Okay.

DR. SCHULTZ: The same as leak-before-break evaluation.

MEMBER BLEY: Yes.

MR. JAIN: And now, I will pass it on to Sunwoo for Section 3.7.

MR. PARK: Okay. Good morning.

My name is Sunwoo Park, Structural Engineer at NRO. I'm going to cover Sections 3.7.2 and 3.7.3 of the staff's SER with the open items.

Section 3.7.2 addresses the Seismic System Analysis, which covers two Seismic Category I structures, the reactor building and the control building and, also, the rad waste building, which is a Category II, is also covered, but only from the perspective of a Seismic II over I and the structure-soil structure interaction on the perspective.

Section 3.7.3 addresses the Seismic System Analysis, covering, for example, the bioshield and the other subsystems, such as the NuScale Power Module, the reactor building crane are covered in other sections of the staff's SER.

The Phase 2 SER had a four open items in this area, and two of them have been resolved to date with the excellent effort and the cooperation of the Applicant. But, still, the two, the other open items, one in 3.7.2 and the other in 3.7.3, still remain open as of today.

The additional information staff has requested has already come in, which is in evaluation by the staff. And the next couple of slides will discuss more details of these two open items.

The only open item in Section 3.7.2 concerns seismic load events for NuScale Power Module and reactor building interface supports which are determined by analysis of the reactor building SASSI model and the NPM ANSYS model. Here SASSI and ANSYS refers to specific computer programs that are used in the seismic analysis of a reactor building and the NPM.

Then, the Applicant has expanded analysis cases to include 130 percent of NPM nominal stiffness as a way of accounting for the potential stiffness variation of the NPM system in a seismic analysis that involves the NPM.

The Applicant also has adopted a new methodology for modeling hydrodynamic mass of pool water, with a purpose of incorporating pool water mass into the analysis model more realistically.

The Applicant has provided information from a new set of analysis which is currently in staff evaluation.

And then, the last bullet needs to be updated because we have received the additional information recently which is about seismic loads on the pool walls from the analysis, which is, again, currently in staff evaluation.

Next slide.

The open item in Section 3.7.3 is about the bioshields which are Seismic Category II, concrete covers placed on top of each NPM as additional radiological barrier. A bioshield is removed during NPM refueling activities, and the removed bioshield is stacked on top of an adjacent bioshield. During the December 2018 regulatory audit, staff identified issues concerning seismic design of with respect to the bioshield. And Applicant, subsequently, has revised the design approach to address identified issues and, also, they provided information on new seismic analysis and the design of bioshields, which is currently in staff evaluation.

Next slide.

Okay. In summary, Applicant has been undertaking actions for timely resolution of the open items. And with the exception of two open item, staff has found that the NuScale DCA demonstrates compliance with the applicable regulatory requirements for a seismic system and the seismic subsystem analysis.

That is about it for my part. Any questions?

Thank you.

MR. ROCHE-RIVERA: Thank you, Sunwoo.

Good morning. My name is Robert Roche-Rivera. I'm a Structural Engineer in the Office of New Reactors, and I will be presenting the staff's review of DCA 3.8.4.

Next slide.

Section 3.8.4 focuses on Seismic Category I structures other than containment, which includes the reactor building and the control building. The staff's review scope includes Section 3.8.4 of the DCA and the associated Appendix 3B.

The staff performed its review in accordance with DSRS Section 3.8.4. We held biweekly public meetings with the Applicant to discuss technical issues and resolutions to RAIs. We conducted an audit of the design report supporting the information submitted in the DCA and RAI responses.

And Phase 2 SER identified five open items. All since that time of submission of the Phase 2 SER have been resolved. And we will be presenting the resolution of the representative open items in later slides.

Next slide, please.

So, as part of our review, we compared the Applicant's design procedures and associated results to the applicable Code and standards, acceptance criteria, and allowable. Based on our review, we found the loads and load combinations considered and the structural capacity determinations to be in accordance with the applicable Code and/or standard. We found the displacement and results with the applicable Code and/or standard allowable, and we found that the structural capacities are greater than the design basis demands. Based on our review, we concluded that the Applicant's methods for demonstrating the design adequacy of the structures are consistent with the NRC's regulatory requirements.

Next slide.

This is a representative open item. For this section, it's open item No. 3.8.4-1, related with RAI 8171, Question 3.8.4-13. Upon review of Rev 0 of the application, we did not find an evaluation, a design evaluation of temperature demands and/or pressure demands. And we issued an RAI requesting such evaluation. In response to our question, the Applicant performed the design evaluations with consideration of temperature, both operating and accident temperature, and accident pressure demands for the reactor building; and also provided results, including strength results for concrete rebar and liners, steel liner. And all those results were shown to be below the allowable limits from the applicable Codes and standards.

This concludes my presentation.

MS. VERA: Any questions for the structural?

MEMBER CORRADINI: Members, any questions? No. Keep on going. Next group.

MS. VERA: Okay. So, for the mechanical part, we're going to start with Nick Hansing for Section 3.9.4. "Control Rod Drive Systems".

MR. HANSING: Good morning. My name is Nick Hansing. I'm the leader here for Section 3.9.4. I'd like to begin by going over some of the key design considerations and features that made the staff's review.

On the NuScale design, the pressure housing and electromagnetic components are very similar to the existing fleet, in that the pressure housing is designed and constructed to ASME Boiler and Pressure Vessel Code Class 1 requirements.

There is a long drive shaft and remote disconnect mechanism which are unique for this design. The design standards and testing programs were emphasized in this review.

To talk about the drive shaft first, NuScale considers the drive shaft as an internal structure and as designated Seismic Category I. DCA stipulates additional design requirements for this SSC as well. An example of one of these additional design requirements would be some of the service-level loading combinations and material testing.

For the remote disconnect mechanism, the other unique feature, the Applicant confirmed that the remote disconnect coil is always de-energized during normal operations and remains in this state during a reactor trip. This mechanism was tested during the key feature mockup testing for its full design life of 150 cycles, which is five times the estimated cycles expected, with satisfactory performance.

Next slide, please.

Drop testing, which was mentioned earlier in the Applicant's presentation. The staff reviewed this testing. It was Appendix-B-compliant with prototypical components. The staff independently verified the dimensions of the design documents, the as-built dimensions of the test facility. Important dimensions like the diametrical gap were consistent. So, it was a very good representation of the actual facility.

Up to 1 inch of misalignment was possible at each interface, and the testing considered manufacturing tolerances and seismic displacements in the calculation of the amount of misalignment.

CO-CHAIR MARCH-LEUBA: I assume you're talking misalignment in the horizontal direction?

MR. HANSING: Correct.

CO-CHAIR MARCH-LEUBA: Did you test anything with bending when it is sideways and diagonal?

MR. HANSING: I believe that the testing facility did have mid-span deflection of the fuel as one of the attributes that was controlled in the --

CO-CHAIR MARCH-LEUBA: The concern is that this system is the only system that can shut down the reactor. I mean, it has to work extremely reliably. For example, some fuels have spacers that they're designed to collapse when you have horizontal loads. And then, the control rods still have to come in. And it's very long.

MR. HANSING: Yes.

CO-CHAIR MARCH-LEUBA: And it has to work.

MR. HANSING: Yes, it does.

CO-CHAIR MARCH-LEUBA: It does not have a backup. So, you did consider all those, right? Can

I sleep comfortably tonight that you looked at it?

MR. HANSING: Yes, I believe that the misalignment testing was very thorough and did look at these aspects.

So, with this testing, 14 different configurations were used. There was a number of variables reduced using fabrication constraints and testing worst-case scenarios. If the interface points were within the same weldment, they would probably move very similar to each other. So, that could be a reduced number of variables to look at.

And in this case, the tested displacement exceeded the maximum expected displacement by more than a factor of two. So, there was sufficient margin in their testing.

The most limiting drop, the maximum displacement and longest drop time was bounded by the performance assumed in the Safety Analysis for control rod drop time.

Additionally, the operability assurance program, which was also mentioned earlier, included performance testing, stability testing, endurance testing, and production testing. And this will be completed by a COL applicant.

The DC Applicant has provided an overview of the program in the DC application with a proposed COL item to implement the program and provide a summary of the testing program and results for staff review.

DR. SCHULTZ: What are the components of the production testing? What does that refer to?

MR. HANSING: The production testing includes some of the ASME Code testing, the hydrostatic tests, and I believe there's some material testing associated with that as well.

DR. SCHULTZ: And going back to your earlier discussion -- no, on this slide, on slide 27 -- you've got 14 configurations, maximum expected displacement by more than a factor of two. What does that refer to? The factor of two applied to what? Not the 1-inch misalignment?

MR. HANSING: So, they had calculated the maximum expected displacement between the manufacturing tolerancing stackup and the maximum expected seismic displacement from their calculations, and the value that they used was more than two beyond what the summation of those factors would be.

DR. SCHULTZ: All right. Thank you.

MR. HANSING: Thank you.

MR. SCARBROUGH: Okay. All right. I'm

Tom Scarbrough. I'm responsible for Section 3.9.6, which is the functional design and qualification and preservice testing and inservice testing of pumps and valves.

Section 3.9.6, we reviewed their function design qualification and preservice testing, PST, and inservice testing, IST, for the NuScale safety-related valves. The DCA provides a full description of the PST-IST programs with a few SER confirmatory items that are relatively minor. There are some table adjustments that need to be made, and things of that nature, and some clarifications of the section. But, overall, it's pretty close to being a full description. So, we're real close there.

There are first-of-a-kind emergency core cooling system valves, and there are first-of-a-kind containment isolation valves. We'll be looking at those and we're going to be talking quite a bit more about those during the closed session this afternoon.

They are currently conducting ECCS valve design demonstration testing at Target Rock right now. And that's ongoing, and we'll give you some more details about that this afternoon as well.

The SER itself has two open items in this section. There's the ECCS valve design, of course, that we're still currently reviewing, and then, also, the ITAAC. And there's a specific ITAAC, which is the valve installation verification ITAAC. As you know, there's several of these valve systems that have long hydraulic runs associated with them, and the proper installation of those valves is crucial to ensuring that these valves, all the various valves, the ECCS valves, the CIVs, the VHRS actuation valves, work properly, because of those long hydraulic runs.

So, we're working with NuScale on that. They provided us a proposal. We had some suggestions to maybe streamline it with some more of the details in Tier 2. And they're working on that now. So, we think that's very close to being resolved as well.

Okay. So, next slide, please. Is it going to change? There we go.

Okay. So, this is really high-level --

MEMBER BLEY: I'm sorry.

MR. SCARBROUGH: Yes, sure, go ahead.

MEMBER BLEY: Back to the ECCS valve design demonstration testing --

MR. SCARBROUGH: Uh-hum.

MEMBER BLEY: -- do you have the test plan? How extensive is that test plan?

MR. SCARBROUGH: Oh, it's very extensive.

We're going to talk about that quite a bit this afternoon.

MEMBER BLEY: We'll do that in closed session?

MR. SCARBROUGH: Yes. I talked to NuScale yesterday. They showed me their slides. They lay out that whole test plan in their slides --

MEMBER BLEY: Okay.

MR. SCARBROUGH: -- which is great.

MEMBER CORRADINI: More to come.

MR. SCARBROUGH: More to come, right. Right.

MEMBER CORRADINI: As you can tell, we're a bit interested.

MR. SCARBROUGH: We are, too.

(Laughter.)

MEMBER CORRADINI: Okay.

MR. SCARBROUGH: Okay. So, as you know, there's the reactor vent valves. There's three of those, and they're on top of the reactor vessel. And then, there's two reactor recirculation valves, RRVs, which are on the side.

MEMBER CORRADINI: Can you hold on now?

MR. SCARBROUGH: Yes.

MEMBER CORRADINI: So, I'm not sure if

somebody's online in the closed -- in the open line, but you need to mute your phone because we hear clicking.

Thank you.

MR. SCARBROUGH: Thank you.

MEMBER CORRADINI: Tom?

MR. SCARBROUGH: Thank you.

Okay. And we'll get more into this, the precise operation of these various design arrangement valves. But there's a main valve. There's an inadvertent actuation block -- it's called an IAB valve. And then, there's a solenoid trip valve, and there's a solenoid reset valve. And they're all connected by this long hydraulic tubing, and we're going to get into that this afternoon.

The valve design demonstration testing is being conducted to demonstrate that 50.43(e) is satisfied. That has to do with new design features, safety features in passive plants. And so, NuScale is conducting that, and we'll talk more about that this afternoon. But we've been following it very closely. We were just there last week and we may be there again. So, that's where we are with that.

MEMBER BLEY: This has a close link to Chapter 6 --

MR. SCARBROUGH: Yes, it does.

MEMBER BLEY: -- which we'll do later.

MR. SCARBROUGH: Yes.

MEMBER BLEY: Will you be back for that one?

MR. SCARBROUGH: Yes, I will.

MEMBER BLEY: Okay.

MR. SCARBROUGH: Yes. And some of the Chapter 6 people are helping us with this as well. And so, we're working together. It's a team effort.

Okay. So, function design qualification. In the big picture, the DCA specifies ASME Standard QME-1-2007 for the qualification of all the safety-related valves, as accepted in Reg Guide 1.100, Revision 3. So, we're comfortable with that. There's a lot more testing going on besides what's happening this month. You know, there's full qualification testing that has to be conducted under QME-1. So, we're comfortable with that.

Also, the ITAAC, there's a functional qualification ITAAC for safety-related valves which points to the qualification report that's specified in the QME-1 standard. So, there's a tie between the functional qualification ITAAC and QME-1 itself. So, we're comfortable with that as well.

We've conducted audits of the valve design specifications. We have a few open items, and most of them had to do with there were some portions of the design specifications that were not quite complete yet. There were some open items in the specifications. And we wanted to make sure the specifications were complete, that we were seeing a full-fledged version of those specifications. And we've heard that NuScale has completed that. So, we'll conduct an audit sometime to follow that up and make sure that was all, that they were all completed.

There are no safety-related motor-operated valves. So, none of those for me to be worried about here. There's no safety-related pumps and no safety-related snubbers. So, just valves in this reactor.

Okay. The next slide.

Okay. Containment isolation valves. These are interesting valves. Basically, they're hydraulic-operated and have ball valves in them. Okay? So, we're using the containment isolation for ball valves.

MEMBER CORRADINI: Is that unusual?

MR. SCARBROUGH: Yes, usually they're sealed.

MEMBER CORRADINI: They're gate valves?

MR. SCARBROUGH: You know, the gate valves. They seal very strongly and they're held closed by the neck-gear mechanism, right? So, that's something we've talked to them about in our audits, is what type of ball valves. There is a new valve design. For butterfly valves, you can have a triple offset which you can actually torque into it, because butterfly valves typically aren't very good for containment isolation, either. But if you have a triple offset where the disc plane is slightly off-center from the piping angle, then you can actually torque these into the seat. And that's what we've been talking about; maybe these ball valves may be the same design as that, so they can torque them into the seat.

MEMBER BLEY: The same valves they use for the main steam isolation valves?

MR. SCARBROUGH: Yes, yes. These are, yes, these are all the same type. Now, actually, how they're set up is a little different, and I'll get to that. But there's 16 of the primary system containment isolation valves, PSCIVs, and they have one valve body, but two valves, two ball valves in the same valve body with two separate, completely separate actuators that go out to separate skids. And so, they're completely separate that way.

But they're closed by hydraulic pressure and they have a hydraulic fluid. But they're opened by that, right, but to close, they have nitrogen bottles that close them, right? And we can talk more about the actual specifics this afternoon, maybe during the closed session, but I don't get too far into the proprietary portion.

MEMBER BLEY: If I'm moving to the closed session, tell me to wait.

MR. SCARBROUGH: Okay.

MEMBER BLEY: But I have two questions about this one. So, usually, a ball valve has equal forces on both sides, so that there's no torque from flow if they're partially open. But if they're offset, as you were saying, the flow can actually assist in closing them? Is that --

MR. SCARBROUGH: Well, these actually, you're able to torque them into the seat. And one thing we've talked about, it's a rack-and-pinion design that's going to be operated.

MEMBER BLEY: Yes.

MR. SCARBROUGH: And the question is, are you going to size the rack-and-pinion such that it's a locking gear train, right, or are you going to torque them into the seat sufficiently, that that force trying to open them cannot do it? And actually, we've talked about that, and that's something they're still thinking about, what the detailed design is going to be.

MEMBER BLEY: You just added another question for me.

(Laughter.)

MR. SCARBROUGH: Okay.

MEMBER BLEY: So, this afternoon will you have pictures of the details of these? Are these unique valves or have they been used in other industries for a long time?

MR. SCARBROUGH: Ball valves are used quite a bit.

MEMBER BLEY: I know, but --

MR. SCARBROUGH: But this use is unique.

MEMBER BLEY: Okay.

MR. SCARBROUGH: We can see if we can track some down for you. We had a lot of ECCS valves discussion, but we didn't have any pictures of the ball valves.

MEMBER BLEY: Okay.

MR. SCARBROUGH: We'll see if we can track some down. Okay?

MEMBER BLEY: We're actually going out there next month.

MR. SCARBROUGH: That's great. I think that's really good. It's really good to see how it all works.

So, that's the primary system containment isolation valves. Now there are six secondary system containment isolation valves, and these are the feedwater and the main steam isolation valves. And the feedwater, there's two of those, and 4 NPS, and then, there's four main steam isolation valves and those are the 12-inch NPS.

And the feedwater valves also have a nozzle check valve. They only have one valve, one ball valve, but, then, they have a nozzle check valve also after that. And nozzle check valves are a little unique. We're starting to see some of them like in AP1000 and such, but nozzles --

MEMBER BLEY: I've never -- I'm not familiar with those at all.

MR. SCARBROUGH: Oh, okay. They're, rather than having swinging, they actually have a -- it's almost like a --

MEMBER CORRADINI: Plugs.

MR. SCARBROUGH: A plug, it's like a plug.

But it's streamlined internal. And it's sort of streamlined internal to the valve, and it moves back and forth. And sometimes there's a small spring in there to keep it in the right position. And under AP1000, we had them do some testing. There were some testing done at, I think, Utah State out there with them, so to qualify those for AP1000. So, we have some experience with them.

MEMBER BLEY: I guess I'm familiar with those in very small-diameter, low-flow kind of situations. Is this again a unique use?

MR. SCARBROUGH: Right, using this large -- they -- typically the operating plants don't -- you don't see these in this large size. I have heard they've been using some non-safety-related systems and things, but that's why we are so interested in what the AP1000 was. It was sort of unique. They're sort of in the whole passive core cooling system for the AP1000.

So we wanted to make sure we had a good understanding of how they operated and whether they're qualified. So they also will be qualified per QME-1 as accepted in Reg Guide 1.100. We did conduct an audit of the CIVs and the reactor safety valves, and there are a few closeout items there. The same thing.

There are some specs. There were some open items in the specs, and they're working on those to get those closed out. So we'll -- as soon as they let us know, we'll finish that up. So that's where we are with those.

And that's the end of my planned presentation for the open session. We have more for you during the closed session. Is there any high-level questions that we could answer for you on this right now?

(No audible response.)

MR. SCARBROUGH: No? Okay. And Renee Li is going to take over for Section 3.6.2.

MS. LI: I'm Renee Li and I review Section 3.6.2, and the review is to ensure that the NuScale designs provide adequate protection against effects of postulate pipe rupture and make the GDC for --

The review concentrates on those area that were outside the staff guidance of BTP 3-4. That include two isolation valve outside containment for penetration piping and also the voltage connection of the ECCS valve to the reactor vessel nozzle. I also review a technical report that address applicant's pipe rupture analysis and associate results. And we have one SER open item related to the bolted connection of the ECCS valve to the reactor vessel.

So upon mentioning that the NuScale design provide two isolation valve in a single valve body outside the containment and the single valve body is welded to the containment vessel nozzle safe end. And those NuScale's isolation valve configuration deviate from the BTP 3-4 containment penetration area. The specific -- the accessibility of this tube of outside containment, rather than one in/one out would be provide in Chapter 6 presentation. And in my presentation I focus on the breaks exclusion for those containment configuration.

So as you may know, in Branch Technical Position 3-4 it defines the containment penetration from the inboard containment isolation valve through the penetration to the outboard containment isolation valve. And that also the BTP 3-4 provides staff's guideline that including certain design stress and the fatigue limit and augment inspection to ensure the extremely low probability of pipe failure in this area and usually refer as spray exclusion area.

Since the NuScale configuration is beyond the BTP 3-4 guideline, the staff ask a question for NuScale to justify the application of break exclusion in the area identify in the FSAR. In its RAI response the applicant state that the design stress and the fatigue limit criteria for the piping segment within the NuScale specific break exclusion area are consistent with the Branch Technical Position 3-4 guideline, and they also in the technical report I mention earlier. It shows the results of their calculation. And the results are within the relevant BTP 3-4 stress and the fatigue limit for break exclusion.

In addition, augment 100 percent biometric in-service examination requirement for all the welds within the break exclusion area is consistent with Branch Technical Position staff guideline, and detail of the ISI described in FSAR Section 6.2 and 6.6. So with this the NRC staff found the applicant justification for its of breaks exclusion area acceptable.

Next slide, please? So the next area is related to the break exclusion of RVV and RRV, the bolted connection to the reactor vessel. As Tom mention earlier, that the RVV and the RRV allowed the natural circulation of the emergency core cooling, and each of those RVV and RRV is bolted directly to the reactor vessel nozzle. The NRC staff's key concern is that this bolted flange connection must not fail to cause a LOCA. So the staff issue RAI requesting NuScale to provide a justification for why the probability of gross rupture is extremely low if NuScale desire to treat the bolted connection as a break exclusion area.

The applicant response the RAI and codified that for both bolted connection, the NuScale design classified them as break exclusion area and they also provide the following justification: First, the bolting stress design criteria per ASME Section III NB 3230 meet the intent of BTP 3-4 stress acceptance guidance for typical piping system. Second, they also demonstrate why the cumulative usage factor of 1.0 is acceptable.

For that first is the ASME Section III NB 3230.3(c) for the fatigue evaluation require a fatigue strength reduction factor of not less than four being apply. So that fatigue strength reduction would provide a conservative safety margin for the 40 design. And also they identify that a list of the phenomena which would -- which might adversely affect fatigue evaluation and then went through each phenomenon to demonstrate why those phenomena are not susceptible, why the NuScale bolt connection are not susceptible for those phenomena.

Here I'd like to point out the BTP 3-4, the fatigue usage factor criteria of 0.1, you know, some technical -- I'd like to provide some technical rationale. So in BTP 3-4 the postulate break is really a mechanistic approach and based on the ASME design requirement that including the stress and the fatigue usage factor. So for the stress is 80 percent of the ASME design stress allowable. And for the fatigue usage factor the design is 1.0, but in BTP 3-4 is -- use 0.1 as threshold and if stress fatigue is considered, then it would be a COF equal to 0.4.

So you see a -- I would say a larger margin that -- for the COF. The basis for the 0.1 is to ensure a conservative margin on the cycle is made available to take into account for example potential 40 design, improperly control fabrication, or installation error, an expect mode of operation, and certainty in the vibration load and other degradation mechanism such as corrosive environment, water hammer. So those are the phenomenon that I mention early that NuScale provide a justification of why those phenomena are not applicable to this particular RVV, RRV bolt connection.

Next slide. So --

CO-CHAIR RICCARDELLA: Wait a second. I

have a few thoughts on that. The fatigue strength, some of them were positive; some of them were negative. The fatigue strength reduction factor for bolting is in there because you have the threads and the threads give you a high stress concentration factor. So it's -- as opposed to a welded joint. But that's kind of a -- so you can't take the full credit for the fatigue strength reduction factor of four as providing margin because you got the stress raiser.

On the other hand, isn't -- in my mind there's just some natural redundancy in a bolt that -- in a bolted flange connection compared to a welded connection. In other words, to get a rupture you have to fail essentially all the bolts. I mean, it's -- you're more likely getting a leak, but I think you're less likely to get an actual rupture because you'll -- if you can -- if it starts leaking, you detect a leak and then you do something about it --

MS. LI: Yes.

CO-CHAIR RICCARDELLA: -- if you fail one or two bolts, right?

MS. LI: I agree. Actually I cover that because to justify the COF of 1.0 --

CO-CHAIR RICCARDELLA: Yes.

MS. LI: -- in addition to the two I

already cover --

CO-CHAIR RICCARDELLA: Yes.

MS. LI: -- the next slide is also NuScale design employ augment fabrication and the in-service exemption requirement. The in-service inspection, the UT exam for the bolt would be at least once every 10 years.

CO-CHAIR RICCARDELLA: Okay.

MS. LI: And also the bolting design adapt NUREG-1389 guideline with highly SS -- SCC-resistant material. That's the alloy-718. And lastly, the highly-sensitive leakage detection system that sensitive to leak rate as low as 0.001 gallon per minute.

So with all the five justification the NRC staff found that applicant justification provide a reasonable assurance of the -- for break exclusion.

MEMBER CORRADINI: But just so I'm on the same page, I see what you're saying. I'm fine with that. But on the other hand, the RRV is assumed to open up because from the standpoint of the transient accident analysis this is one of their breaks that they consider anyway.

MS. LI: Yes, I think Chapter 6 may cover that.

MEMBER CORRADINI: This maximizes pressure in containment.

MS. LI: Right, but I think in here the focal point is not to break the --

MEMBER CORRADINI: I'm with you. Okay.

MS. LI: Okay. So with that, NuScale has not yet complete the stress and the fatigue calculation for this particular bolt connection, and we going to audit that when the information become available and to close this open item. So that --

DR. SCHULTZ: Did they have a complete -- in terms of the augmented fabrication examination requirements has that been completed? Have they described that completely to your satisfaction, the fabrication examination requirements?

MS. LI: Right. Yes.

DR. SCHULTZ: Okay. Thank you.

MS. LI: So that conclude my presentation.

MEMBER CORRADINI: Thank you.

MR. TSIRIGOTIS: Good morning. My name is Alexander Tsirigotis and I'm a mechanical engineer in the New Reactors Office. This will be a brief presentation of the review the staff performed for the NuScale design certification Sections 3.8.2, 3.9.3, and 3.9.5.

The staff performed a design specification audit as part of its review. There the staff found that the stress analysis and fatigue evaluations of the reactor pressure vessel, containment vessel and reactor internals were not available. The position in SECY 90-377 is that this level of detail should be completed and included at the design certification stage. Because these evaluations were not available, this remains an open item.

NuScale is updating the stress analysis and performing fatigue evaluations at critical locations in accordance with ASME Boiler and Pressure Vessel Code Section III. The calculations for the -- to complete the stress analysis and the fatigue evaluation are scheduled to be available for audit by the end of July, therefore this remains an open item.

The staff also reviewed the design of the trip/reset valve of the ECCS valve. You've heard previously my colleagues discussing the ECCS valves and you will hear more about this later.

This trip/reset valve serves as both RCS and containment pressure boundary. The next slide will show a schematic presentation of the boundaries for this valve. According to the ASME Boiler and Pressure Vessel Code Section III examination of the valve body is required during fabrication. The NRC staff questioned the adequate level of quality for the design of this valve. The NRO office though finally accepted the ASME Boiler and Pressure Vessel Code Section III as the acceptable level of quality.

Information on the function, operation, performance and testing of the ECCS valves will follow in the closed session as Tom mentioned earlier.

And this sketch here shows the schematic of the trip/reset solenoid valve and it shows the boundaries, the pressure boundaries. Any questions?

CO-CHAIR MARCH-LEUBA: Yes, on the fatigue analysis these valves are set on their operating position and kept there for two years. They only move once every two years during refueling. So what fatigue analysis are you doing?

MR. TSIRIGOTIS: The fatigue analysis that I mentioned earlier --

CO-CHAIR MARCH-LEUBA: Yes.

MR. TSIRIGOTIS: -- is for the containment vessel, the reactor vessel --

CO-CHAIR MARCH-LEUBA: Oh, not for the valves?

MR. TSIRIGOTIS: -- and the internals.

CO-CHAIR MARCH-LEUBA: Okay. I hope

somebody's going to explain to us how this valve works, because I'm looking at the design that you --

MR. TSIRIGOTIS: In the closed session --

CO-CHAIR MARCH-LEUBA: -- and I don't see nothing.

MR. TSIRIGOTIS: -- you will see more about this.

CO-CHAIR RICCARDELLA: Would you put that sketch back up, please?

MR. TSIRIGOTIS: Yes.

CO-CHAIR RICCARDELLA: So this is the containment valve attached in the containment. So then there has to be a break exclusion zone here, right, where you can't assume a break?

MR. TSIRIGOTIS: Well, did you -- I don't know if Renee spoke about the containment, about the break exclusion area. I think she did. Yes.

CO-CHAIR RICCARDELLA: I just wanted to confirm. In this sketch right here we're talking about the area between the valve and the containment as a break exclusion zone, correct? Renee?

MEMBER CORRADINI: We're confused.

MR. TSIRIGOTIS: I can see that. These valves are very small to begin with, right? I mean, as you see there the size is --

CO-CHAIR RICCARDELLA: Yes. Okay.

MR. TSIRIGOTIS: Now I am not quite sure, but I think the break exclusion area is for piping and these valves have tubing, small tubing.

CO-CHAIR RICCARDELLA: Okay.

MEMBER CORRADINI: So --

MR. TSIRIGOTIS: To postulate breaks below one inch, I believe that BTP 3-4 does not postulate breaks below one inch.

MEMBER CORRADINI: Yes, help us out, Tom.

MR. SCARBROUGH: These -- those tubes are all inside that valve, that body. Safe-in -- there's a safe-in and those tubes run through there and then they connect to the solenoid valves. The rest valve is on the top and the trip valve is on the bottom, how they're set up. And we'll show those in the closed session --

MEMBER CORRADINI: Okay.

MR. SCARBROUGH: -- exactly how those are.

MEMBER CORRADINI: So you're going to educate us in closed session?

MR. SCARBROUGH; With NuScale's help, yes. Right.

CO-CHAIR RICCARDELLA: CORRADINI: Good. Okay. Thank you.

MR. SCARBROUGH: Yes.

MR. TSIRIGOTIS: Thank you, Tom.

MS. VERA: This concludes Chapter 3.

PARTICIPANT: It appears you're done.

MS. VERA: Yes.

PARTICIPANT: Thank you, gentlemen, ladies.

MEMBER CORRADINI: Questions the members?

(No audible response.)

MEMBER CORRADINI: Okay. We are going to postpone any closed session until the end of the day. And we will take up Chapter 6 next, but we're close enough to lunch I'll declare victory and we'll go to lunch. Can we -- I assume we can start earlier.

PARTICIPANT: Yes.

MEMBER CORRADINI: So why don't we come back at quarter to 1:00? We have an hour. Have a good lunch.

(Whereupon, the above-entitled matter went off the record at 11:48 a.m. and resumed at 12:49 p.m.)

CO-CHAIR MARCH-LEUBA: Rebecca, you're up.

MS. NORRIS: Good afternoon, everyone. Thank you for having us. This presentation is the NuScale presentation on FSAR Chapter 6, Engineered

Safety Functions. This is in support of the staff discussion including open items.

The presentation team for today is Dan Lassiter, a mechanical design engineer; Zack Houghton, mechanical design engineering manager; and myself, Rebecca Norris, the licensing supervisor for this chapter.

Today we will be covering Chapter 6 as shown with the following notes: As you can see, 6.7 does not apply to the NuScale design, so we will not be covering that. Also 6.3, we have detailed valve drawings and cutaways in the closed session later today, so we're going to be giving a brief overview in this open session then later providing more details.

We also will be giving an overview of Section 6.2.1 and 6.2.2., but we will not be going into the analysis portion. This is because we share the RELAP analysis with Chapter 15. The Chapter 15 presentation is tomorrow, so it will be covered then.

Also as a final note, in the back of your packet there are quite a few backup slides. We tried to organize them in the same order as the main presentation. We do not intend to cover them for time constraints, but if you have any questions on them, feel free to ask.

And with that, I will turn it over to Zack for the presentation.

MR. HOUGHTON: All right. Thank you, everyone. Good afternoon.

So first we'll start with Section 6.1, Engineered Safety Feature Materials. So components for the engineered safety features have been selected to be compatible with the environmental conditions of normal operation, maintenance, testing and accidents. Components for the NuScale design of the engineered safety features have a 60-year design life. One material choice of note that we put on this slide is the use of a forged XM-19 material in the core vicinity of the lower containment vessel.

MEMBER BALLINGER: Is this with -- well, XM-19, Nitronic 50, whatever you want to call it, is it -- is this section size bigger than usual for use of this material? In other words, is this an unusual size for XM-19?

MR. HOUGHTON: I believe it is larger than previous experience, but I'll consult with our materials expert H.Q. Xu, who should be on the line.

H.Q., do you have any more insight on that question?

MR. XU: Yes, this is H.Q. Yes.

MR. HOUGHTON: Yes is the answer.

MEMBER BALLINGER: I guess that's an answer.

(Laughter.)

MR. HOUGHTON: And it was chosen specifically because it has shown good resistance to neutron embrittlement. So the lower -- the containment valve being a metal vessel, I think the members are familiar with it, and we'll see it more in future slides. But that's why we made the specific material choice there for the lower containment vessel in the core region.

MEMBER BALLINGER: It's also got twice the yield strength of regular stainless steel.

MR. HOUGHTON: And we do not allow any protective coatings within the containment vessel or for any components within the containment vessel.

MEMBER CORRADINI: So there's no organics at all?

MR. HOUGHTON: Correct.

MEMBER CORRADINI: Okay.

CO-CHAIR MARCH-LEUBA: When you say on, you mean inside, or outside, too?

MR. HOUGHTON: Inside of containment.

CO-CHAIR MARCH-LEUBA: It is more like in?

MR. HOUGHTON: Yes.

All right. For Section 6.2.1 the primary functions of the containment itself are to provide enclosure and support of the reactor pressure vessel, the reactor coolant system and the associated components. It is a credited fission product release barrier. It provides containment of postulated mass and energy releases inside containment, so that would be a steam or feed line break or a chemical volume control system line break. It supports operation of the emergency core cooling system by providing retention of reactor coolants during ultimate heat sink operation and also by providing a heat removal pathway to the ultimate heat sink.

It's designed to support heat removal capability to maintain pressure and temperature less than the design allowables; that is in accordance with GDC-50, and also to rapidly reduce the peak pressure and temperature to less than 50 percent within 24 hours. And as Rebecca mentioned earlier, the actual details of the pressure temperature analysis will be with Chapter 15 tomorrow.

All right. And again, hopefully the members are familiar with the containment design at this point. We can see the figure on the left of this slide here is the -- showing the outline of the containment vessel and the reactor vessel associated components located with that. The containment vessel is designed and stamped as an ASME Class 1 vessel for NuScale. It is 1,050-PSI design pressure, 550-degree design temperature. It's located directly within the ultimate heat sink, which is the reactor building pool. And as mentioned, we use an XM-19 for the lower containment material and an SA-508 Grade 3, Class 2 for the upper containments.

CO-CHAIR RICCARDELLA: So you're going from a low-alloy steel to a stainless steel? You have a bi-metallic weld there?

MEMBER BALLINGER: Just a flange.

MR. HOUGHTON: The difference is made at the flange connection.

MEMBER BALLINGER: Oh, it's at the flange? I see. Okay.

MEMBER CORRADINI: So where is the water level?

MR. HOUGHTON: So the water level is located --

MEMBER CORRADINI: The ultimate -- or the --

(Simultaneous speaking.)

MR. HOUGHTON: The water level is located right below the head region of the containment vessel.

PARTICIPANT: That's the pool water level.

MR. HOUGHTON: The pool water level, that's correct.

MEMBER CORRADINI: Okay.

MR. HOUGHTON: During normal operation the containment is maintained at a vacuum, so there would be no water inside of containment.

MEMBER CORRADINI: Have you guys decided what you're going to define as vacuum? Has that been decided, or is that private -- proprietary what you choose finally as the value?

MR. HOUGHTON: Well, our tech specs require that we operate sufficiently low to maintain our containment evacuation system and containment leakage detection operable. The exact value that we would be operating at in normal operation will vary cycle to cycle, plant to plant based on if you have any in-leakage or out-leakage, of course, and the actual performance of the containment evacuation pumps themselves. So we don't have a set vacuum limit for normal operation.

MEMBER CORRADINI: Oh, there isn't? Where is the containment evacuation system taking its suction?

MR. HOUGHTON: So it is taking its suction at the head of the containment, in the upper region. And so all of the piping and electrical penetrations are located on the head of the containments. One of the pipelines at the head of containment is connected to a containment evacuation pump, which would be located out in the reactor building gallery area.

MEMBER CORRADINI: So one of the members at lunch after a refreshing meal at the cafeteria --

(Laughter.)

MEMBER CORRADINI: -- had a question about dynamic effects about measuring leakage. So what -- you tell us when there's an appropriate time to ask because I'm trying to understand where an accumulation of condensate would be, how you then -- either by measuring pressure or by measuring moisture condensate from essentially pulling a vacuum how you determine that and how the dynamics of it would be affected. Because we couldn't answer it at lunch.

MR. HOUGHTON: Okay. I'll give a high-level overview and then please ask any clarifying questions.

MEMBER CORRADINI: At the appropriate time in your discussion, whenever it is.

MR. HOUGHTON: This is probably as good a time as any. I really don't have any slides that talk specifically about the evacuation system. That would have been presented with Chapter 9.

MEMBER CORRADINI: Yes, I know, but --

MR. HOUGHTON: So we have --

MEMBER CORRADINI: -- we need to be reminded.

MR. HOUGHTON: Sure. So we have two methods for detecting any leakage into containment. There's the pressure of the containment itself and then there is the condensate collection vessel in the containment evacuation system, which is a very sensitive method for seeing if we have any condensate collected.

MEMBER CORRADINI: But the second method, which is essentially measuring what you condense by taking sampling, or by taking a continuous pool like the containment, just have some time window at which you do the averaging over. That time window can't be as small as a minute. It strikes me that that's too small.

So what is the time window of averaging? It's like it's got to be a moving average that you're checking and then monitoring and deciding if something happened.

MR. HOUGHTON: I don't have a specific answer for that on the time window, for the amount of time. That's something I'd have to go discuss with the team and come back with a --

MEMBER CORRADINI: Okay.

MR. HOUGHTON: -- better answer on. And I don't think we have anybody on the line --

(Simultaneous speaking.)

MEMBER CORRADINI: Or we can put it down on a list of things to chat with you about in July.

MR. HOUGHTON: Yes, that's --

MEMBER BALLINGER: There has to be a tech spec on unidentified leakage.

MR. HOUGHTON: Correct.

MEMBER BALLINGER: So you have to meet that somehow. And if each plant -- each module may be slightly different, then that implies that the vacuum -- the system is maybe slightly different. You have to meet the tech spec on unidentified leakage.

MR. HOUGHTON: Correct.

MEMBER BALLINGER: Yes.

MR. HOUGHTON: Yes, and there are specific timelines for the amount of time that is allowed to identify a specific leakage amount. I don't recall the exact numbers off the top of my head, but we're well within the regulatory requirements for how quickly we can detect leakage, our design being a metal containment vessel it maintain the vacuums. We have very sensitive measuring --

(Simultaneous speaking.)

MEMBER CORRADINI: Right, but the reason the question got generated over lunch was because the containment vessel has no insulation and it's always condensing there's got to be -- you'd get a pulse -- if you had a leak, you'd see a pulse in pressure and an equivalent increase in measure condensate, because I assume the evacuation pumps are not changing flow rate. They're pulling at some fixed rate. So I get to a different quasi-equilibrium pressure which tells me that somehow I have an additional leak that I can't detect, and that is either above or below the tech spec limit.

MR. HOUGHTON: Correct, and there's also tech spec limits on being able to maintain the containment evacuation system operable. And so to support that there are graphs in Chapter 5 which dictate the pressure versus temperature that you would see inside containment. Basically we're trying to avoid a scenario where we have condensate collecting inside the containment because that then wouldn't make it out to our collection vessel. So we preclude operation within a window that would have condensate on the containment vessel.

MEMBER CORRADINI: And that's in Chapter 6?

MR. HOUGHTON: The graph is in Chapter 5. That's correct.

MEMBER CORRADINI: Okay. Thank you.

CO-CHAIR MARCH-LEUBA: Yes, see, now that we've gone through that, I'm thinking that this leak detection only works if the leak rate is larger than the capacity of the vacuum to remove it. If you have -- I always assume that the vacuum pumps pull vacuum and then they stop.

MR. HOUGHTON: Yes.

CO-CHAIR MARCH-LEUBA: But if you have been continuously running --

MR. HOUGHTON: The vacuum pumps are continuously running, so if the leakage rate were beyond the capability of the vacuum pumps, then you would have a build up in pressure and then we would eventually exceed our tech spec --

(Simultaneous speaking.)

CO-CHAIR MARCH-LEUBA: But if it's smaller than the vacuum pumps, then you will not see it?

MR. HOUGHTON: Then we will see it. So the containment evacuation system is always running. Any vapor in the containment space would be pulled out, condensate would be collected off of the --

CO-CHAIR MARCH-LEUBA: Oh, so condensate is collected by the vacuum pumps, not at the bottom of the containment?

MR. HOUGHTON: By the vacuum pumps. That's correct. There's a -- excuse me, by the vacuum system. So the vacuum pumps are constantly taking suction. There's a condenser on that system.

CO-CHAIR MARCH-LEUBA: So the condenser is downstream of the -- is on the high-pressure side of the vacuum pump?

MR. HOUGHTON: Yes, and so we would collect any condensate out of that. That's how we measure the specific --

(Simultaneous speaking.)

CO-CHAIR MARCH-LEUBA: I misunderstood. I thought you were condensing it and collecting it in the bottom of the vessel.

MR. HOUGHTON: No. No, we preclude condensing inside of the vessel with our operating limits.

CO-CHAIR MARCH-LEUBA: Oh, you hope you don't.

MR. HOUGHTON: Yes.

CO-CHAIR MARCH-LEUBA: So as long as the leak is smaller than the capacity of the vacuum pumps, you will be okay if you continue to operate? You will detect it and you will be able to continue to operate?

MR. HOUGHTON: Correct. And as long as the leak is lower than our allowable leakages per our tech specs.

CO-CHAIR MARCH-LEUBA: Yes. All right.

CO-CHAIR RICCARDELLA: I'm sorry. I had something I wanted to ask about.

The selection of the materials for the containment -- would you go back to the previous slide? So you said you picked the FXM-19 because of better radiation resistance, but yet the reactor vessel, which is at a higher radiation field, is still a low-alloy steel, right?

MEMBER BALLINGER: Yes, the reason I asked the question, it's otherwise known as Nitronic 50. That material is used in light water reactors in in-core or -- you know, inside the vessel for a lot of things. So that's that.

It also has twice the yield strength.

CO-CHAIR RICCARDELLA: I would assume though that the radiation, the neutron embrittlement concern is greater at the vessel? Higher neutron flux than at the containment, right, down in that region?

MR. HOUGHTON:

MR. HOUGHTON: That's true. And so we do all -- I mean, we take appropriate precautions for dealing with neutron embrittlement at the reactor vessel, but that uses the SA-508 material. We have coupons in place in the Surveillance Program for monitoring embrittlement.

CO-CHAIR RICCARDELLA: Well, I guess I was just curious as why you felt it was necessary to go to this other material here.

MR. HOUGHTON: It was a design decision that we made for a number of reasons, one of which though was questions on whether or not we would be able to get appropriate coupon data from test coupons located in the containment vessel to support that containment vessel material.

CO-CHAIR RICCARDELLA: Okay.

MR. HOUGHTON: So it was trying to avoid the concern of embrittlement of that vessel because of its unique conditions.

CO-CHAIR RICCARDELLA: And are the

radiation levels close? I mean, they are fairly close together, so maybe they're not that big a difference.

PARTICIPANT: It's one over R.

MR. HOUGHTON: Yes, I don't have the exact numbers for the difference between the two.

CO-CHAIR RICCARDELLA: And then on the low-alloy steel on the containment is that clad on the outside?

MR. HOUGHTON: Correct.

CO-CHAIR RICCARDELLA: Inside and outside or just --

MR. HOUGHTON: Yes.

CO-CHAIR MARCH-LEUBA: Okay.

MR. HOUGHTON: It is clad on both the inside and the outside.

CO-CHAIR RICCARDELLA: Okay. And the reactor vessel is clad inside and outside, too?

MR. HOUGHTON: Correct.

CO-CHAIR RICCARDELLA: Okay. Thank you.

MEMBER SKILLMAN: Did the design of the reflector have anything to do with choice of material for the containment vessel? In other words, are you depending on that reflector to reduce the neutron fluence sufficiently to not have to consider NVT on the containment vessel?

MR. HOUGHTON: I don't think I'd be prepared today to give a complete answer on design decisions around the reflector and how that related back to the containment vessel. Certainly we do consider the reflector when we're accounting for neutron embrittlement. Design decisions of the reflector blocks, not something I could address completely today. If you have more interest, it's something we could discuss when the membership is out in July.

MEMBER SKILLMAN: One more question: A couple weeks ago, maybe a month or two ago, Dr. Ballinger and Dr. March-Leuba and I were really kind of on the warpath for safety level of boric acid addition and CVCS, but as we've explored more thoroughly the importance of leak detection, it's almost as if the containment evacuation system may have much more of a primary role in terms of alerting the operators to an emerging issue in terms of leakage.

So I understand you use the PRA to set the safety requirements for the -- or to establish what are the safety requirements for the system for what is nuclear safety and what is not, but does this warrant a more thorough examination of those decisions given the importance of your CES, your evacuation system? I mean, that's going to be the leading indicator of a leak.

MR. HOUGHTON: I would certainly agree with the importance of that system and, you know, you look at our -- the design of our containment being a smaller volume system maintained at a vacuum. I believe the sensitivity of the system to leakage is going to be a very important operating point for the plant and something the operators will care greatly about, so I fully agree with that statement. And it's -- we have built in the appropriate redundancy into the system to make sure that we can maintain that equipment operational. It's -- the system is classified the way it is because it is not credited for dealing with any accident scenarios.

CO-CHAIR MARCH-LEUBA: Yes, but one poor man's definition of safety grade is if it is used to maintain the tech specs, it is safety grade, right or not?

MR. HOUGHTON: I don't believe that's --

CO-CHAIR MARCH-LEUBA: Or it's only --

(Simultaneous speaking.)

MR. HOUGHTON: -- definition of safety-related.

MEMBER SKILLMAN: It's the right feeling, but it's not the right legal test.

MR. HOUGHTON: Okay. So we'll move onto slide 8 if there are no further questions.

So we handed out a hard copy of this slide as well because it's very difficult to read the figure, but it's an important figure.

So this figure shows the class breaks per Reg Guide 1.26 of the components that make up the containment isolation system. So there's the containment itself, but then these are the containment isolation valves located on the head.

The areas show in blue crosshatching are the ASME Class 1 isolations. And again, that's per Reg Guide 1.26. And then in the green crosshatching that shows the piping and the valves that are designated as ASME Class 2. And I'll say -- I'll point out as well downstream -- going back to the ASME Class 1 isolations, downstream of the second outboard isolation to the next isolation is designated as ASME Class 3, and that's again in accordance with Reg Guide 1.26.

One point that will come up later in the slides but I'll point out now, because the design of the isolations which isolate lines that connect directly the containment are similar to the design of those valves which isolate the reactor coolant system, those valves are all designed, constructed and stamped in accordance with ASME Class 1. That doesn't show up on this figure. I'll describe that in more detail later, but it's essentially the valves that you see on the right side of the screen. The main steam and the feedwater valves, those are designed in accordance with ASME Class 2.

CO-CHAIR RICCARDELLA: Help me understand. Most existing reactors, PWRs and BWRs, when they have double containment isolation valves, there's one inside and there's one outside, right? And your system is designed where they're both on the outside. Help me understand the technical differences between those two in terms of isolating the breaks and break exclusion zones.

MR. HOUGHTON: We'll have a slide on that --

CO-CHAIR RICCARDELLA: Okay.

MR. HOUGHTON: -- in a few more slides.

CO-CHAIR RICCARDELLA: All right. Thank you. I'll wait.

MR. HOUGHTON: Okay. The COL item for Section 6.2.1 is really related to the transient analysis and nothing that we've discussed so far.

All right. Section 6.2.2, Containment Heat Removal. Again, the containment itself is the containment heat removal component for the NuScale design, so we don't have a separate active system of containment spray or ice condensers or any -- these other designs for removing heat from the containment. The designs have active equipment.

As mentioned, the pool water is just below the upper head during normal operation, and that provides greater than 30 days cooling volume.

The containment steel walls allow direct passive heat transfer to the pool. For our limiting peak pressure cases the containment pressure is reduced to less than 50 percent of peak pressure within two hours. We have requested exemption from General Design Criteria 40 which requires testing of containment heat removal systems. Because there are no active components there's no real operability or performance testing to be done for our containment heat removal components. And the containment removes heat from module as the accident progresses with no operator action.

MEMBER CORRADINI: So there's no testing of DHRS? Maybe I'm not following.

MR. HOUGHTON: This is just the containment heat removal, so this is the containment itself.

MEMBER CORRADINI: Not the DHRS?

MR. HOUGHTON: This is not -- does not include the DHRS.

MEMBER CORRADINI: Okay. Sorry.

MR. HOUGHTON: All right. So Section 6.2.4 on Containment Isolation. So the containment isolation valves meet the General Design Criteria 55.

MEMBER CORRADINI: Can we go back?

MR. HOUGHTON: Yes.

MEMBER CORRADINI: So there's no testing of it. What is the inspection process that's necessary for -- because you rely on it for long-term cooling, which means you've got to make sure the inner surface is relatively continuously clean so that you don't have to take account of some sort of following factor of the inner surface, or the outer surface. So what is done there in terms of inspection upon refueling?

MR. HOUGHTON: So there are in-service inspection requirements within the in-service inspection, the ISI Program, for doing visual examinations of containment surfaces during refueling outages.

MEMBER CORRADINI: Is there some limit as to if you see fouling that you have to take -- go inside and do cleaning?

MR. HOUGHTON: That would have to be in the applicant's Inspection Program where they would just make that determination on exactly when cleaning will happen versus not. I'll say in our analyses we do account for fouling. It's one of the factors that we look at in our analyses.

MEMBER CORRADINI: But you don't account for any sort of cleaning operation on an every-two-year basis? That's not accounted for? What I guess I'm trying to get at is there will be an accumulation of a fouling factor and you don't take account of the fact that it needs to be cleaned off in terms of performance? You take what you'd consider to be a conservative value for fouling? You see my question?

MR. HOUGHTON: Yes, I see your question. I'd want to confirm with the Chapter 15 folks to make sure that we've got a --

MEMBER CORRADINI: Okay.

MR. HOUGHTON: -- complete answer there. But there is an in-service inspection requirement to review the visual that the actual In-Service

Inspection Program would be developed as a COL item. And that would require cleaning as needed of the vessel.

MEMBER CORRADINI: Okay. Thank you.

MR. HOUGHTON: All right. Next slide? So I was discussing, the containment isolation valves, they meet GDC-55, 56 and 57 with two exemptions being requested. The first is for locating both of the containment isolation valves outside of the containment for GDC 55 and 56 penetrations. And the second is with regard to the decay heat removal system which is unique in that it constitutes a closed system both inside and outside of the containment. So that's not clearly addressed by the General Design Criteria, so we have an exemption for locating the isolations separately.

To support this exemption the piping between the containment vessel and the containment isolation valve is designed to meet the break exclusion zone criteria. I mentioned previously the designed and stamped primary system containment isolation valve is Class 1; the secondary system containment isolation valve is designed and stamped as ASME Class 2. And as the staff mentioned previously in their Chapter 3 review, the containment isolation valves are hydraulically operated and all fail closed.

To the member's question on the basis for this, the NuScale containment environment is unique. It's a high-pressure, steam environment during accident conditions. The containment design is unique in that it is a metal vessel. So we determined that the most appropriate design choice would be to put both isolation valves outside of containment but as close to the boundary as possible and have taken appropriate cautions to eliminate the potential for break in that area.

MEMBER CORRADINI: Remind me, the cartoon shows as if they're bolted to the vessel. That's not correct?

MR. HOUGHTON: They are welded to the vessel. So this next slide shows that. So this is looking at a representative primary system isolation on the containment head. So you can see the nozzle integral to the containment head itself, safe end welded to the nozzle, and then the containment isolation valves welded to the safe end.

CO-CHAIR RICCARDELLA: Would you go through the materials for these various components, or is that in closed session?

MR. HOUGHTON: I do not have something on the materials for these components. That information is in the design certification document though.

MEMBER BLEY: You refer to these, and there's a bunch of other valves in Chapter 6 that get referred to as hydraulically operated. They're hydraulically moved to their position for normal operation of the plant, but they're -- are these spring-closed or are they --

MR. HOUGHTON: So they use a --

MEMBER BLEY: -- some other mechanism?

MR. HOUGHTON: -- gas bottle that essentially is the --

MEMBER BLEY: So they're gas-closed?

MR. HOUGHTON: -- spring that closes a valve, yes.

MEMBER BLEY: Okay.

MR. HOUGHTON: So they're pneumatic-hydraulic --

(Simultaneous speaking.)

MEMBER BLEY: So, yes. And when we look, like in the PRA, and we're going to talk about this when we come visit you, the part of the valve that moves them to the safe position is the part we're really interested in. So using hydraulically-operated failure rates for valves like this isn't appropriate.

You want to use the, either air-operated or spring-operated, which is true for a bunch of other valves we'll look at later, failure rates. Just wanted to mention that we'll be looking at that more late.

MR. HOUGHTON: Thank you.

MEMBER SKILLMAN: Zack, what is the hydraulic medium? What is the composition of it?

MR. HOUGHTON: So it is -- we do not have the composition of the medium specifically identified in the design certification or specifically selected today. It's required to be compatible with the pool environment and non-combustible. Tentatively we've been using Houghto-Safe, which is a -- no relation and a currently available hydraulic fluid that's used in industry today.

MEMBER SUNSERI: It would also have to be resistant to radiation changes, too, right?

MR. HOUGHTON: It'll have to be resistant for the radiation in the environment that's in, but, no, we're at the top of the containment vessel so a ways a way.

MEMBER SUNSERI: Yes, but still I mean, they've got a bioshield and all that stuff in there.

MR. HOUGHTON: Sure.

MEMBER SKILLMAN: Is it hydrocarbon? Does it have a carbon derivative, carbon base?

MR. HOUGHTON: I don't have that answer off the top of my head. Do we have somebody in Corvallis that would have the answer to that question?

MEMBER BLEY: We'll be interested in this --

(Simultaneous speaking.)

PARTICIPANT: -- give us a moment.

MEMBER BLEY: We'll be interested in the same question on other hydraulically-operated valves.

MR. HOUGHTON: Okay. It may be something for us to discuss when the staff's out, members are out in July.

MEMBER CORRADINI: Is somebody on the line that wants to help out here?

MR. McGEE: We'll have to get back to you on this one.

MEMBER CORRADINI: Okay. Thank you.

CO-CHAIR RICCARDELLA: What is that in-service insert all about?

MR. HOUGHTON:

MR. HOUGHTON: Oh, thank you. So because these valves are the containment isolation boundary they have to be local leak rate tested in accordance with Appendix J, which I'll discuss in a future slide.

You can see on the figure that there's an in-service insert shown installed on the inboard side of the containment isolation valve set.

MEMBER BLEY: Can somebody point to that?

MR. HOUGHTON: Yes, so it's -- so here's the in-service insert in the figure. So this would be the normal operating configuration for leakage rate testing. During and outage this insert would be removed. A testing insert would be installed that creates a leakage path barrier and that way we can do testing of the individual valves.

PARTICIPANT: So that's a bolted flange?

MR. HOUGHTON: Correct.

MEMBER BLEY: How often do you do that kind of in-service testing, or do you expect it to be done?

MR. HOUGHTON: Two years and as needed, right? As needed after maintenance.

MEMBER BLEY: Okay. But at least every two years?

MR. HOUGHTON: Yes, I'll look for Gary to confirm that.

Do we do local leak rate testing every two years regardless of maintenance?

MR. McGEE: Yes, right now we would do

Option A of Appendix J and then eventually we would -- I would imagine go to Option B, performance-based, but initially the plant is going to be at Option A, which would be every two years.

MEMBER BLEY: Okay. And what kind of a functional test do you do afterwards to make sure you can pass water when you need to? That you put the right --

(Simultaneous speaking.)

MR. McGEE: -- what you asked.

MEMBER BLEY: -- insert back in after the test.

MR. McGEE: Well, the inserts are clearly marked. I mean, you'd have administrative control.

MEMBER BLEY: They always are, but if you don't test, you're going to have one in the wrong -- wrong one in there one day.

MR. McGEE: Right, well, you're not going to flow your process fluid when you start up. You're not going to flow CVCS. You're not going to flow -- you're going to -- operations is going to know right away that they're not --

MR. HOUGHTON: Yes, we'd have to go look at each of the penetrations, but I believe Gary's --

MEMBER BLEY: So if I understood Gary

right, all of these during the startup would have flow through them so you'd have an actual verification that the right insert is in the --

MR. HOUGHTON: I think that would be generally true. We'd want to go through line by line. To my knowledge I don't recall this question coming up in the past, so I --

MEMBER CORRADINI: But there's got to be some sort -- I think where Dennis is going is there's got to be some sort of testing upon restart of the module to make sure that you have what you thought was the proper --

MEMBER BLEY: Well, I am, yes. I mean, and years ago we didn't do that kind of testing and sure enough we had things left in place that shouldn't have been left there.

Yes, sir?

MR. RAD: Hey, this is Zach Rad, Director of Reg Affairs. So to your question, generally speaking there is testing of your process systems after you've put them back together following an outage, but it's important to note that with that insert in, this is the safest condition it could be in relative to nuclear safety is that the containment is isolated.

MEMBER BLEY: I have to think that -- and you probably ought to think that all the way through. I mean, that's where you want it to be, under say accident conditions, but if most of these are one-way and some are another way, you might not be right.

MR. RAD: I didn't mean to imply that it was desirable.

MEMBER BLEY: I know.

MR. RAD: I just mean to imply that --

MEMBER BLEY: But I'm not sure it's always the safe thing.

MR. RAD: -- it's not going to prevent the operation of a containment isolation valve from a safety function perspective.

MEMBER BLEY: That's for sure.

MR. HOUGHTON: We understand the question and it's something we'll take away and consider internally.

MEMBER BLEY: Okay.

CO-CHAIR RICCARDELLA: So this is typical of all of the Class 1 penetrations?

MR. HOUGHTON: Correct. Yes. For all of the Class 1 penetrations, yes.

CO-CHAIR RICCARDELLA: And so looking at this, the break exclusion zone has to be from the valve all the way to the containment vessel, right?

MR. HOUGHTON: That's correct.

CO-CHAIR RICCARDELLA: And you can not take a break in there?

MR. HOUGHTON: That's correct.

CO-CHAIR RICCARDELLA: And you're doing some special materials, inspections and things for that, right?

MR. HOUGHTON: The materials, the inspections are all listed in the design certification, but we have applied the break exclusion zone criteria, that full section.

All right. Next slide?

MS. NORRIS: Now seems like a good time to bring up an answer. We had a question on the fouling of the containment heat transfer surfaces. So I've been referred to Section 6.2.2.4 of the FSAR, and this addresses some of this. So hopefully that answers your question.

MEMBER CORRADINI: I'll look. Thank you.

MEMBER BLEY: Say those again?

MS. NORRIS: 6.2.2.4 of the FSAR.

MR. HOUGHTON: All right. So we covered this a little bit in the staff's presentation on Chapter 3, but just to cover it again for the containment isolation valves. So every containment isolation valve has solenoid valves that actually control the hydraulics. Those are located remotely on a hydraulic control skid.

Each module has two separated dedicated hydraulic control skids which are located in separate areas of the reactor building and each hydraulic control skid operates one division of valves. So in that last figure that we saw each of the valves shown in that one body would have their own dedicated solenoid valves which are located on dedicated control skids that support each module and are located in separate areas. So we've remained divisional separation as much as we can given the unique consideration of our design of our isolation valves being in the same area. And then this figure just shows what I described in a bit more detail.

MEMBER REMPE: So I'd like to follow up on your follow-up about Section 6.2.2.4, because it refers us to Section 6.2.1.6 without any definition of what periodic inspections is. And if I look at 6.2.1.6, all it says is we're going to have periodic; again, undefined, in-service inspection to ensure compliance with GDC-39, just -- that's for surface fouling.

Is that enough of an answer for you, Mike?

MEMBER CORRADINI: I'm still trying to find it. I'm sorry.

MEMBER REMPE: I mean, didn't you want to know how often they're going to inspect and --

MEMBER CORRADINI: I want to know -- I'm just trying to understand; and maybe this is for the accident analysis portion that's closed, what they used to bound it? And if it's bound, then what are they looking for to decide I'm out of bounds; I've got to go clean it?

MR. HOUGHTON: Understood.

MEMBER CORRADINI: But I found it on 6.2-25 and it doesn't say very much. It's a paragraph.

MEMBER REMPE: And it refers -- 6.2.2.4, which is where she referred us, says go to 6.2.1.6, which again doesn't say what you're asking about, what is the -- when do you start cleaning up?

MEMBER CORRADINI: So it doesn't have to be answered today, but to the extent that you take credit for some sort of fouling, I'd like to know what it is and how do you know you've exceeded it?

MR. HOUGHTON: Understood.

MEMBER SUNSERI: Let me ask a follow-up question. I'm trying to remember. So when you refuel though, you flood the containment, right? So it's flooded and drained each refueling outage, right, at a minimum?

MR. HOUGHTON: Yes.

MEMBER SUNSERI: Okay.

MR. HOUGHTON: All right. So moving on to Section 6.2.5, Combustible Gas Control. So for the NuScale design, again it would be the unique containment arrangement with the containment held at vacuum conditions during normal operation. We have designed for the complete effects of hydrogen burn or detonations within the containments. We have requested an exemption from 10 CFR 50.44(c)(2), which is a requirement for an inerted atmosphere or limited hydrogen concentration.

The design itself provides a mixed volume. Just inherent to its design of having no sub-compartments it relies on natural convection for emergency core cooling system operation which creates a mixing environment within the containment.

So we do however provide equipment for monitoring of in-containment hydrogen concentrations and we are capable of continuously monitoring hydrogen and oxygen gas concentrations after both design-basis and beyond-design-basis accidents. There are analyzers provided within the process sampling system. I will note however that the exact performance requirements of that monitoring equipment is still something that is under discussion with the staff.

MEMBER CORRADINI: So is there a standing hydrogen and oxygen concentration just due to the fact --

SIRI ON CELL PHONE: Okay. I found this on the web for hey, Sara, standing hydrogen --

(Laughter.)

MEMBER CORRADINI: I didn't do anything.

PARTICIPANT: Does she have an answer?

MEMBER CORRADINI: She didn't.

PARTICIPANT: Thank you, Alexis.

MEMBER CORRADINI: Thank you, Alexis. Siri.

So there's a standing hydrogen and oxygen concentration in the atmosphere, is that correct?

MR. HOUGHTON: That is not correct. So the atmosphere is a vacuum under normal operations, so it would be air --

MEMBER CORRADINI: But you have a relative humidity in there all the time, so you have a standing concentration due to essentially decomposition, radiolitic decomposition, don't you?

MR. HOUGHTON: Well, no. So we're maintaining in a vacuum continuously during operations. So are you speaking specifically during operation or during an accident scenario?

MEMBER CORRADINI: Well, let's start during operation.

MR. HOUGHTON: So during operation we're maintaining in a vacuum continuously. So anything that you have in there would be negligible, I'll say, volumes or mass of anything left behind.

MEMBER CORRADINI: Okay. And now if I take it to an accident, I'm kind of going with your previous slide where you said you considered detonation. So I'm trying to understand how you evolve hydrogen concentration and what's the associated oxygen concentration.

MR. HOUGHTON: So there would be some amount of oxygen that would come out from dissolution, as you mentioned. For our analysis -- okay, so in an accident condition the regulations require you to consider 100 percent fuel-clad water interaction.

MEMBER CORRADINI: Right.

MR. HOUGHTON: So hydrogen is very dominating in what would have to be a severe accident case for our plant to generate hydrogen under those conditions because we maintain the core covered for all design-basis accidents.

For the purposes of our integrity evaluation we assume that there was sufficient oxygen to detonate the hydrogen under all conditions, however, in actual design you'll have very limited amounts of oxygen available. It's a very hydrogen-limited scenario for our plant.

MEMBER CORRADINI: So you're making the assumption there's sufficient oxygen? You're not -- there's not a calculation that computes what it would be?

MR. HOUGHTON: Correct.

MEMBER CORRADINI: Okay. All right. Thank you.

MR. HOUGHTON: And the only oxygen would be from air and leakage or dissolution, so relatively small amounts.

And then last bullets, we do include connections for clean up of the environment after an accident, should it be needed. So there's connections to bring in skidded equipment as needed.

Next slide.

MR. SEXTON: Hi, this is Colin Sexton. If I could just step in real quick and discuss options.

When we do our calculation for how large of a combustion could occur, the main element of that calculation is how much oxygen could be present, because oxygen is a limiting ingredient out of oxygen and hydrogen in our combusting calculations. And the elements that contribute to that oxygen in our calculations are initial oxygen that could be present in the containment and radiolitic production of oxygen.

MEMBER CORRADINI: So, Colin, nice to hear from you. So what is that value at -- so it's a calculation in terms of that concentration and then an assumption of combustion?

MR. SEXTON: Yes, what we do in our calculation is we estimate the maximum amount of oxygen that could be present by being quite conservative on how much could be there initially and how much could be developed radiolitically. And then we calculate how big of an explosion you could generate with that oxygen by optimizing the hydrogen conditions to match. So oxygen limiting, and we use the oxygen as effectively as we can to generate a limiting combustion event in our analyses.

MEMBER CORRADINI: Okay. I think I got it. Thank you. Thank you very much.

MR. HOUGHTON: Okay. So containment leakage testing. All the containment isolation valves and the passive containment isolation barriers -- so these would be manways and bolted joints.

MEMBER CORRADINI: So can we go back? I'm sorry. I'm reading my notes.

So there's an exemption -- if I'm remembering correctly, there's an exemption on something to do with sampling. Does this somehow intersect with combustible gas control, that you're sampling to determine what your concentration is in containment for hydrogen monitoring?

MR. HOUGHTON: That is the portion that's still under discussion with the staff.

MEMBER CORRADINI: So tell me more about that since I can't remember. I wrote a note to myself that I should ask.

MR. HOUGHTON: It's related to our alternate source term discussions that are ongoing. I wouldn't want to give any preliminary information in that area.

MEMBER CORRADINI: So this is more a matter of the method in which you're going to do the sampling so you can determine the concentrations?

MR. HOUGHTON: The open questions are

around the exact conditions under which an operator would have to go out, establish the equipment, what sort of environment are they going to see, how are we accounting for those aspects of taking the sample.

MEMBER CORRADINI: Well, I know it's an open item, but it's still being discussed with staff. There's not been a response back in terms of the open item, or is it still being reviewed by staff? What's the status of this?

MR. HOUGHTON: I'll look for the -- it's being reviewed?

MEMBER CORRADINI: Well, if -- I'll ask the staff the same question. So if we're going to go -- if you're going to go look for the staff, we'll just get them next.

MR. HOUGHTON: Okay.

MEMBER CORRADINI: Okay. Thank you.

MR. HOUGHTON: Thank you.

All right. So Section 6.2.6. So we designed all of the containment barriers to support local leak rate testing, and local leak rate testing would be done per 10 CFR 50, Appendix J for all type B and type C tests. So those are -- passive barriers such as manways, bolted connections would constitute type B testing. Type C testing is your valves, right, the containment isolation valves that we discussed.

We have requested exemption from GDC-52 for doing the type A integrated leak rate testing, again because of the unique aspects of our containment design. Some of the aspects of our design which support that exemption are that it's fabricated and tested similar to reactor pressure vessel requirements because it is a -- stamped as a Class 1 vessel. All known leakage pathways are tested through local leak rate testing. We have a comprehensive In-Service Inspection Program meeting Class 1 requirements for all those welds, so those welds that we saw that make up the connection to the valves. And then the containment is also hydrostatically tested.

MEMBER CORRADINI: So the argument is that you basically are always monitoring? So you know what your leak rate is just by the continuous monitoring of the evacuation system? I'm trying to understand the --

MR. HOUGHTON: That's a correct statement, but it really doesn't play heavily into our exemption request because we still have to do testing under different conditions. The monitoring that we're doing under normal operation is a different set of conditions than we would see during design-basis accident --

MEMBER CORRADINI: Sure.

MR. HOUGHTON: -- conditions, which is what we would have to try to replicate in a type A test.

MEMBER CORRADINI: But the argument is by your individual isolation valve by isolation valve testing that meets the standard of knowing what your leak rate would be?

MR. HOUGHTON: Correct.

MEMBER CORRADINI: Okay. Thank you.

MR. HOUGHTON: So going to the next slide. So there are some additional commitments that we made to support this exemption. As I mentioned, the containment is hydrostatically tested, however, the controls over hydrostatic testing are a little bit different. We test at a -- you test at a higher pressure under hydrostatic conditions.

The code for hydrostatic testing doesn't have requirements on bolt pre-load, bolt gasketing, so we created a separate -- what we call a pre-service design pressure leakage test that is controlled by ITAAC that sets additional controls over bolt pre-load, gaskets used configuration so that prior to operation we're doing a test to show that no leakage pathway develops under limiting conditions.

We have done an analysis to support our design certification and we have a COL item to follow up on this analysis for the as-built containment to show that we maintain flange contact pressure at accident temperature concurrent with peak accident pressure conditions.

So we use a finite element analysis model to look at the prying that would happen at every one of our bolted joints and ensure that we maintain appropriate contact where the gasket itself is. And that's important because it supports our position that the leakage that we see under type B or type C testing conditions is identical to what we would see under type A conditions.

MEMBER SKILLMAN: Are all your bolted connections -- you have a fiber flange, a flexitallic gasket. What kind of gasket do you have such that you're determining whether or not that gasket is threatened by that pressure?

MR. HOUGHTON: The exact gasket selection is in the technical report where we talk about the details about our assumptions for this analysis. That's also part of the purpose of the COL item is to ensure that the actual -- if there's changes in the final selected gasket material type, that that all gets considered in this analysis.

MEMBER SKILLMAN: Thank you.

MR. HOUGHTON: And then finally we made a commitment to verify the preload that was -- that's been applied. So after maintenance, that sort of thing, we'll ensure that we've got the appropriate preload because it is important for that analysis that we've done.

Next slide. And here are the COL items.

CO-CHAIR RICCARDELLA: But you actually hydrostatically test that containment pre-service at 1,000 PSI, some really high pressure like --

MR. HOUGHTON: Correct, at pre-service, yes.

CO-CHAIR RICCARDELLA: No in-service?

MR. HOUGHTON: No, we do not make a commitment for in-service hydrostatic testing.

CO-CHAIR RICCARDELLA: Or leak tests? No in-service leak tests?

MR. HOUGHTON: Local leak rate testing only.

PARTICIPANT: That's the type A they're asking for the exemption.

CO-CHAIR RICCARDELLA: Okay. Got it.

MR. HOUGHTON: All right.

MS. NORRIS: So coming back real fast to the hydraulic fluid question, so we have a vendor-specified fluid, which is Fyrquel, F-Y-R-Q-U-E-L. But this is not specified in the DCA. This just is what is suggested. That is a hydrocarbon.

So the words in the DCA are in Section 5.6.1 and 5.6.2. Basically the hydraulic actuator shall use water or fluid demonstrated compatible with reactor building pool water cleanliness requirements as provided in the reference so that fluid leakage in the pool is not detrimental. And 5.6.2. says any hydraulic fluid including corrosion inhibitors if added shall be non-flammable and non-combustible. So that's what we have in the DCA.

MEMBER SKILLMAN: Fyrquel. The reason I asked; and I mentioned this I think at a first meeting, you've got a -- what, a 7½-8 million gallon pool, 12 reactors?

MR. HOUGHTON: Correct.

MEMBER SKILLMAN: You're going to keep the temperature in that pool to probably 100, under a 100 so you're not steaming inside your building. You're going to work to keep your relative humidity low. When we did the de-fueling at TMI-2 we used hydraulic fluid. We thought that would be dandy. What we didn't realize is that through the course of events we actually had coliform in the water because we used a clean water supply, but it was not pure from coliform.

And the de-fueling at TMI-2 stopped for several months because we grew leafy green vegetables in the reactor that completely blocked all of our activity. We were on standby for probably three months. And what happened was the hydrocarbon; and Fyrquel is one of them, became a food source for the material that was in the water and we had to use hydrogen peroxide to purify the water.

And so if there is any hydrocarbon exposure, I would suggest that that is something you want to act on now. You do not want to continue with any hydraulic fluid that can add hydrocarbon to your pool. You want either having clear water with boric acid or some other substance, but not anything that would feed a hydrocarbon base. That is a multi, multi-million dollar decision.

MS. NORRIS: Yes, sorry about that. As we were talking corrections came in. It is not hydrocarbon. It is in fact phosphate-based, probably for those reasons. Also that's not from the FSAR. It's from the design spec, the sections I gave you earlier.

MEMBER SKILLMAN: Thank you.

MEMBER BLEY: Zack, can I take you back to the containment evacuation system?

MR. HOUGHTON: Yes.

MEMBER BLEY: I'm looking at the drawing. And you come up out of the containment and you go into those vacuum pumps and into a condenser and drain, but it also shows line coming with valves open into the vacuum pumps from the service air system. Is that some kind of cooling for the vacuum pumps or something? You're not mixing these streams. That wouldn't make sense.

MR. HOUGHTON: Let me --

MEMBER BLEY: From SAS, which I'm guessing is service air system.

MR. HOUGHTON: Yes, I believe that's only for startup, although there may be some other unique conditions where we use that. So I'd want to go confirm with the systems engineers.

MEMBER BLEY: Okay. And have them take a look at the drawing that's in the FSAR, because that shows the valves open from those systems, which is a little funny.

Now you asked -- people asked you about a fairly complex way to collect the sample over time. As I read Chapter 6, or 9, it seemed as if you just let it collect until you get some amount and you're keeping a time run on that so you can get a leak rate. And then I guess if it starts to fill up, you would dump it. That's what it looks like, except you didn't -- it also says you can do grab samples in there, which would mean you'd have to drain it and then do a --

(Simultaneous speaking.)

MR. HOUGHTON: Right, and the grab samples would be for chemistry purposes, but we do let it collect. When it --

MEMBER BLEY: Over time?

MR. HOUGHTON: -- as it fills up it would have to be dumped.

MEMBER BLEY: Yes, you don't have some kind of meter --

(Simultaneous speaking.)

MR. HOUGHTON: But we can monitor it as it's collecting to see --

MEMBER BLEY: Yes.

MR. HOUGHTON: -- what the rate is.

MEMBER BLEY: That's what I thought. Okay. Thank you.

MR. LASSITER: And I would add that the containment pressure is a fast response leakage mechanism, but the --

MEMBER CORRADINI: Well, but I --

(Simultaneous speaking.)

MR. LASSITER: -- would build over time as a longer average.

MEMBER CORRADINI: We're talking -- but I guess I'm back to the question that kind of led from Member Ballinger, which is if the leakage is below your evacuation pump capacity, I wouldn't see it. I'd see it in the collection. If it's above the evacuation pump capacity, I'd see it in the pressure. At least that's how I'm thinking this things functions.

MR. LASSITER: And if the pressure continues to build, we have --

MEMBER CORRADINI: Right.

MR. LASSITER: -- automatic trip on containment pressure.

MEMBER CORRADINI: Okay.

MR. HOUGHTON: And also mention it's not exactly a 0.1 --

MEMBER CORRADINI: Oh, I understand.

MR. HOUGHTON: -- sort of problem, right?

As leakage increases your pump performance changes depending on now much volume it's removing.

MEMBER CORRADINI: Sure.

MR. HOUGHTON: Yes.

MEMBER CORRADINI: Okay. Thank you.

MR. HOUGHTON: All right. Section 6 --

MR. ARTHUR: This is J.J. Arthur again. I can answer the question about the service air connection. That's actually used to remove liquid from the containment after refueling, so we actually use pressurized air to drive out water.

MEMBER BALLINGER: Oh, back to the vacuum system again.

MEMBER CORRADINI: Are you happy?

(Simultaneous speaking.)

MEMBER BLEY: -- this drawing and that didn't -- I understand that, but that didn't seem to match up to this, but maybe you can help me.

MEMBER BALLINGER: The vacuum system connects to the top of the containment, right?

MR. HOUGHTON: Correct.

MEMBER BALLINGER: You've got this giant condensing wall that's the containment. What if water condenses down into the bottom?

MR. HOUGHTON: So that was the -- that

refers back to the graph that I mentioned that's in Chapter 5 that describes -- we -- if we maintain the pressure low enough inside containment, then you won't have condensation --

MEMBER BALLINGER: Right.

MR. HOUGHTON: -- on the walls, right? So we have a limit on vacuum pressure in order to support the containment evacuation system monitoring feature.

MEMBER BALLINGER: So you are not going to allow condensation on the walls?

MR. HOUGHTON: Correct, we'll maintain the pressure below the saturation pressure that --

MEMBER BALLINGER: Okay.

MR. HOUGHTON: -- corresponds to the temperature of the pool.

MR. LASSITER: That's why have a safety-related pressure trip below atmospheric.

MR. HOUGHTON: All right. Section 6.2.7 covers --

MEMBER SKILLMAN: Let me ask one more question: So let's take a module. You remove it from its operating bay. You take it to the refueling stand. You disassemble it. And all of that equipment is now bathed in borated water, 800 PPM or -- it's a number like that. Over the course of the life of the plant will not that boric acid become a coating on the inside and the outside, particularly on the inside of that vessel? And is there a wash-down before you reassemble and bring that vessel back into its operating bay?

MR. HOUGHTON: Well, that goes I think to the other member's question. So the exact frequency that -- at which we would have to do a cleaning isn't defined today, but there's an inspection requirement to look for exactly the collection that you're describing and there would be a requirement to clean it off if you were --

MEMBER SKILLMAN: Okay.

MR. HOUGHTON: -- building up a scale.

MEMBER BLEY: The kind of thing I've been stewing over; and we talked to the PRA people about this, is -- and I hadn't thought about this scenario, but if you begin to get some leakage -- I mean, you walk around a current PWR and anywhere boric acid solution leaks out and sees the atmosphere you start getting crystallization and this gooey stuff.

I don't know why you wouldn't get that inside on the containment surface. And if then this expanded to be an accident, then would you get the same heat transfer? And that's certainly not reflected anywhere in the PRA if that's a potential problem. We were worried about other kinds of contamination on the surface as well, but that's an obvious one that --

(Simultaneous speaking.)

CO-CHAIR MARCH-LEUBA: Boric acid is very volatile and certainly at atmospheric pressure it will be a crystal, maybe not at that --

MEMBER BLEY: What it leaves behind --

CO-CHAIR MARCH-LEUBA: Nothing. It becomes a gas and it goes to the pump. Maybe. I mean, I know it boils off at -- it evaporates at 300 degrees CM in --

(Simultaneous speaking.)

MEMBER BLEY: Oh, I believe that, yes.

CO-CHAIR MARCH-LEUBA: Yes, so the pressure.

MEMBER BLEY: I don't know what the containment temperature is.

CO-CHAIR MARCH-LEUBA: It's --

(Simultaneous speaking.)

MEMBER CORRADINI: Containment -- well, it can't be any -- it can't be much higher than the pool temperature.

MEMBER BLEY: I wouldn't think so.

CO-CHAIR MARCH-LEUBA: Yes.

MEMBER BLEY: So that --

CO-CHAIR MARCH-LEUBA: I don't know what --

MEMBER BLEY: -- whether it comes out at --

(Simultaneous speaking.)

CO-CHAIR MARCH-LEUBA: I --

(Simultaneous speaking.)

MEMBER BLEY: -- it leaves behind a hell of a residue.

CO-CHAIR MARCH-LEUBA: I don't know what evaporation temperature is at, but whatever pressure they operate at, but it is a possibility, I guess.

MEMBER CORRADINI: But I thnk what you're getting at is kind of what I was getting at, is at the end of cycle before I would go through a refueling, which they would then fill it with water, everything would dissolve, everything would be nice and rosy again. There would be some sort of fouling that you'd have to consider.

MEMBER BLEY: And it seems like over time there might be --

CO-CHAIR RICCARDELLA: Well, that leads I guess to a question. What sort of In-Service

Inspection Program are you doing on the CNV?

MR. HOUGHTON: So there's a requirement to do a visual inspection of the CNV to look for exactly this. There's also a requirement to have a Boric Acid Control Program, right? So applicants will have to be inspecting, looking for the boric acid crystals that might get left behind from a leak, for example. So there's requirements in the In-Service Inspection Program and in the Boric Acid Control Program to look for these sorts of indications.

CO-CHAIR RICCARDELLA: So visual at every outage, but then is there also a volumetric inspection as part of this slide that's up here now?

MR. HOUGHTON: There's volumetric inspections of welds as described by the In-Service Inspection Program.

So Section 6.2.7 covers fracture prevention of the containment vessel. As mentioned, the containment vessel meets the relevant parts of GDC-116 and 15 and is there for ASME Section III, Subsection NB. The ferritic pressure boundary materials meet Section III, Subsection NB fracture toughness requirements. And again, as discussed the austenitic stainless steel XM-19 is not subject to neutron embrittlement at the fluence levels we've --

(Simultaneous speaking.)

CO-CHAIR RICCARDELLA: Again, following on what I said, it's -- then is it also Section 11, IWB in your In-Service Inspection Program, which is like what -- the way you would inspect the reactor vessel?

MR. HOUGHTON: That's -- the inspections for the containment vessel are all captured in Table 6.2-3, so I'd refer you there to see the exact requirements.

CO-CHAIR RICCARDELLA: Okay.

MR. HOUGHTON: Okay. And with that I'll turn it over to Dan to discuss emergency core cooling.

MR. LASSITER: All right. Thanks, Zack.

So we're going to talk about the NuScale emergency core cooling system design. I know there's been some questions about detailed figures on the valves, and those figures will be in the closed session this afternoon, but there will be some simplified figures in this presentation.

The NuScale --

CO-CHAIR MARCH-LEUBA: Can you talk closer to the microphone?

MR. LASSITER: Sure.

CO-CHAIR MARCH-LEUBA: Otherwise, I won't hear you.

MEMBER BLEY: And speak a little louder, if you can.

MR. LASSITER: Okay. The NuScale ECCS system cools the core when it cannot be cooled by other means such as loss of coolant accidents. As Tom described earlier, the ECCS consists of five valves: three reactor vent valves on the top of the reactor vessel and two reactor recirculation valves on the sides. The ECCS works in conjunction with the containment isolation system to retain the required inventory for operation. ECCS valves are normally held closed by electrical power and the ECCS is actuated on a high containment level and loss of AC power for 24 hours.

CO-CHAIR MARCH-LEUBA: Wait, the RRVs are held open by electric power, AC power and pressure difference. The IAB keeps them closed, right?

MR. LASSITER: The IAB is not normally acting to -- acting in the ECCS valve system. They're normally open and unengaged. The only thing keeping the main valves closed is the trip solenoid valves staying closed.

CO-CHAIR MARCH-LEUBA: So you can trip the AC power and then even though the pressure difference is larger than the set point, it would open?

MR. LASSITER: Okay. Well, it is conditional on the prevailing reactor pressure as to whether the IAB is --

(Simultaneous speaking.)

CO-CHAIR MARCH-LEUBA: So the IAB is not holding it closed, but it will the moment you remove the power?

MR. LASSITER: Dependent on the reactor pressure, correct.

CO-CHAIR MARCH-LEUBA: All right. So it's both AC power and pressure that keeps them closed?

MEMBER CORRADINI: So let me ask the question a little differently because eventually you're going to explain this to us and after you give us the quiz we'll pass.

If the IAB fails in the direction of allowing the ECCS upon a loss of power to open, the effect is a higher containment pressure with the blowdown for a brief time. That's the only difference that I can see. Am I missing something? Maybe we're into accident analysis and you can tell me to wait until tomorrow, which I'm happy to do or wait until closed session.

But I'm trying to understand a failure of the IAB upon a demand to open essentially allows for just a starting point of a higher stored energy and pressure in the primary system.

MR. HOUGHTON: It would be dependent on the exact event that you're looking at. So one of the limiting events is inadvertent opening of a relief valve.

MEMBER CORRADINI: Right.

MR. HOUGHTON: So multiple valves opening --

MEMBER CORRADINI: Right.

MR. HOUGHTON: -- would be a -- critical heat flux limits or something that we also look at in addition to the peak pressure.

MEMBER CORRADINI: Right. Okay. But memory -- well, I don't want to get ahead of ourselves. We're going to be back there tomorrow. So, okay. Fine.

MR. HOUGHTON: Correct.

MEMBER BLEY: You said they're electrically. The solenoid-operated FSAR says they're hydraulically operated.

MR. LASSITER: They're hydraulic or pilot-operated main valves, but the actuator which controls the hydraulic fluid --

MEMBER BLEY: Oh, that's right.

MR. LASSITER: -- is a solenoid, correct.

MR. HOUGHTON: Well, see this in more detail in the closed session in future slides.

MR. LASSITER: Yes, maybe we'll --

MEMBER CORRADINI: Yes, well, you can just tell us to wait.

MR. LASSITER: Okay.

MEMBER CORRADINI: That's acceptable. And I have a feeling --

(Simultaneous speaking.)

MEMBER BLEY: -- that's probably a lot easier, yes.

MR. LASSITER: Yes, I would defer these -- this review of the detailed valves to the closed session.

Next slide, please?

CO-CHAIR MARCH-LEUBA: Oh, no, no, no. Don't move. This is something I should have brought up during Chapter 19, but let me just put it on the record.

It actuates on high CNV level the moment the level goes a little bit higher than the valves, but that level is measured by this famous laser-guided --

MEMBER REMPE: Radar.

CO-CHAIR MARCH-LEUBA: -- radar --

MEMBER REMPE: Yes.

CO-CHAIR MARCH-LEUBA: -- radar bouncing back and forth from the top of the vessel. Now my suspicion; and I haven't seen any detail, is that those radar-level sensors are digital and they will -- they would have a microprocessor on them that would process the radar signal to see the phase delay on the bouncing back.

MR. LASSITER: Right.

CO-CHAIR MARCH-LEUBA: And therefore once you consider common-cause failure between all those four microprocessors that those four level sensors may fail at the same time. And why is this a problem? Because for two years that radar is out of sensor. I mean, there is no level and you're sending the radar, nothing bounces back. So you are not testing -- those level sensors are working for two years and you're hoping that because you tested them during refueling they still work now.

And I wish I had asked this question before during the Chapter 19 evaluation, but the common-cause failure of the instrumentation is not beyond the realm of possibility, and especially a complex instrumentation that has its first nuclear application. Now your question is a Chapter 19 guy, but --

MR. HOUGHTON: We understand the question and we'll take that back and discuss with both chapter 19 and our I&C counterparts as well to fully consider that.

CO-CHAIR MARCH-LEUBA: And I know that Dr. Corradini would like us to help your design by myself. I'm looking for some analog switches, that red line, and then put the radar as a non-safety grade system to measure the level.

MR. HOUGHTON: We appreciate the comment.

CO-CHAIR MARCH-LEUBA: But it is something that we may have to come back to chapter 19 when we come back to chapter 19.

MEMBER DIMITRIJEVIC: Chapter 19 only goes through there to assign the failure rates. How the valve operates, that belongs here. And we agreed to discuss operation of this with either a visit or here, so the chapter 19 just models reality. We have to know what reality actually is so we know what will be appropriate model for it.

MEMBER REMPE: Along with those other concerns, degradation with radiation? Is that so? Is that a concern?

CO-CHAIR MARCH-LEUBA: The main concern I have is that for two years that level sensor doesn't measure nothing.

MEMBER REMPE: Right.

CO-CHAIR MARCH-LEUBA: If it fails, you will know it failed.

MEMBER REMPE: And what causes it? Well, many things -- is a radiation environment, there's it's a lot of things. I'm very interested in --

CO-CHAIR MARCH-LEUBA: My main concern is for two years you have a radar that is blinking and not receiving anything back. And then you hold that in a severe accident and it probably will work, but --

MEMBER BLEY: It's not seeing water, but it should be seeing the metal at the bottom of the vessel.

CO-CHAIR MARCH-LEUBA: Somebody that know how it works it should --

MEMBER BLEY: Just a heads up, when we get back to chapter 7, we didn't talk about that during chapter 7.

MEMBER REMPE: Yes, there was no evidence about this. We learned about it after chapter 7.

MR. LASSITER: We'll take that question back to our instrumentation design group.

MEMBER CORRADINI: It will be one of the questions in July.

MR. LASSITER: Okay. Moving on, there are a couple of slides animating the ECCS system and the NuScale module. It's getting ahead of me a bit here, but on the ECCS demand, the ECCS valves are opened. Steam is relieved out of the top reactor vent valves. In a situation such as a loss of coolant accident, a CVC pipe break which is an un-iceable pipe break inside containment, high containment level is reached. By that time reactor pressure has decreased to some level and ECCS is actuated.

Go to the next slide, please.

CO-CHAIR MARCH-LEUBA: What signal opens the vent valves?

MR. LASSITER: MPS, our module protection system, sends a signal to actuate all the ECCS valves into the open position. They're actuated by removing power to the solenoids.

CO-CHAIR MARCH-LEUBA: Yes.

MR. LASSITER: And the signals are high containment level. That's really the only signal other than a loss of AC power --

CO-CHAIR MARCH-LEUBA: How are you going to get to that level unless you have the vent valves going?

MR. LASSITER: If there's a pipe break.

CO-CHAIR MARCH-LEUBA: In the LOCA.

MR. LASSITER: Correct.

CO-CHAIR MARCH-LEUBA: That would be the only time the ECCS would be actuated.

MR. LASSITER: Correct.

MEMBER CORRADINI: So let me just repeat what you said just so we're on the same page. I would expect with any sort of LOCA inside containment, I'd see a pressure rise that can't be maintained by the evacuation system. Therefore, it would be a high pressure signal or a rate of pressure rise that would essentially trigger the RVVs. Am I wrong?

MR. HOUGHTON: Well, before you get to that point, you'd have a containment isolation and a reactor trip on other signals.

MEMBER CORRADINI: Thank you.

MR. HOUGHTON: So if you continue to lose volume into the containments and raise level, then once level is high enough is when you get the ECCS actuation.

MEMBER CORRADINI: Okay, and it's the RVVs and the RRVs will then open in some sequence.

MR. LASSITER: Right now, they open

simultaneously.

MEMBER CORRADINI: Okay.

MEMBER BLEY: But they don't open until there's a enough water that you have water to move back into the vessel through the ECCS valve.

CO-CHAIR MARCH-LEUBA: This movie is kind of flaky because you seem to open the valves before the level loops and to open the valves you need to have the water outside. Your level opens valves and the water -- the level wasn't really dropped inside.

MR. HOUGHTON: Correct.

MEMBER CORRADINI: Let me just go back to my question so that I'm not -- because you said it much better than I did which is I had some sort of detection of a pressure rise or a rate of pressure rise above some sort of specification. And then I go to containment isolation, reactor trip, turbine isolate, turbine trip, reactor trip. And then I would assume I sequentially first open the RVVs and then after some delay time open the RRVs. Otherwise, I'd start blowing down in both locations and losing inventory. But I guess I don't care because I get to some sort of level which would cause a recirculation anyway. Am I there?

MR. LASSITER: Yes.

MEMBER CORRADINI: So all five of them upon a signal will open?

MR. LASSITER: Correct.

MEMBER CORRADINI: Okay.

CO-CHAIR MARCH-LEUBA: But that signal being the level in containment is higher.

MEMBER CORRADINI: No, it would be a pressure rise.

CO-CHAIR MARCH-LEUBA: What is it?

MR. LASSITER: I think what Zack was describing earlier is that earlier in the sequence there would be a reactor trip and a containment isolation based on other signals such as high containment pressure or low pressurizer level. And if the event progresses and the core is not being -- able to be cooled by other means, DHR, for example, and there's water spilling into containment, once there's sufficient level for ECCS operation, that's where ECCS is actuated by the ECCS.

CO-CHAIR MARCH-LEUBA: As it says in the previous slide, the only thing that opens the five valves is high level in containment, nothing else opens it.

MEMBER BLEY: Loss of all electrical power.

CO-CHAIR MARCH-LEUBA: Loss of DC power.

MR. LASSITER: Correct.

CO-CHAIR MARCH-LEUBA: If it is good for you, you should encourage it.

MEMBER BLEY: Or an operator can open --

MEMBER CORRADINI: An operator can override and do a manual actuation.

MR. LASSITER: Operator can actuate ECCS from the control room.

MEMBER BLEY: And that's like one switch that opens both valves.

MR. LASSITER: Correct.

MR. HOUGHTON: But the IAB would go off if it was above the IAB pressure. This is all going to be -- we'll see that the pictures that will show this, I think in more detail in the closed session where you can actually see the components and the pressure chambers and how -- the operation of the valve --

MEMBER CORRADINI: Thank you for explaining. Appreciate it.

MR. LASSITER: Move onto the next slide of the animation here. So I think we're aware now but the steam vents into the containment region, condenses in the containment due to the containment being submerged in the reactor pool. Liquid level rises above the core and also above the reactor recirculation valves.

MEMBER CORRADINI: Can you point to recirc valves? I think I see them. Oh, here they are. Thank you.

MR. LASSITER: Okay, reactor recirculation valves and the water level was rising up in that animation there.

You can go to the next slide.

MEMBER CORRADINI: While you're still there -- oh, well.

MR. LASSITER: So the cold water comes back into the reactor vessel by the reactor recirculation valves down the downcomer, reenters the core, the inlet to the core. You see that again the liquid travels through the riser. It's steamed off back through the reactor vessel vent valves, the ECCS vent valves on the top and this establishes the long-term cooling of ECCS in two phase natural circulation cooling system.

Next slide.

MEMBER CORRADINI: So let me ask again if we're straying into chapter 15 just tell us to wait until tomorrow.

But the real cartoon, the actual cartoon of the lower part of containment is a much smaller gap than all the things we've seen in the animation. So how much water does it take to essentially fill up to the RRV?

In this picture, it strikes me as about one tenth of the total free volume to fill to the RRV. And in all the animations, it looks more.

CO-CHAIR MARCH-LEUBA: The more important question is how high above the fuel is the inside level when the outside level reaches the set point?

MR. LASSITER: I'll answer the question this way. The reactor recirculation valves are about six feet above the top of the core. And the level of -- you know, the settling level of retained coolant inside the reactor module is dependent on the accident. So I think I'd like to defer the question about the level of water to tomorrow to chapter 15 which is tomorrow, I believe.

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

MR. LASSITER: Because it varies depending on if it's a -- what type of break is occurring.

MEMBER CORRADINI: Right, but the actual volume of water doesn't change, so I'm just looking for the volume up to the RRVs which is the trip point.

MR. HOUGHTON: We don't have that exact number. That's something we can look at, but your point is correct that in some of the artistic renderings that we see that clearly shows more space there, but in the figure that you held up that was shown earlier in the slides here which is a very small volume in there that's for two reasons, right, it's the exact number we can come back to it.

MEMBER CORRADINI: That's fine.

MR. HOUGHTON: But it's small for two reasons, right, so that we can quickly correct condensate and get volume into the lower region, but it's also there to minimize the heat retransfer resistance once we're in ECCS operation, so we keep a very short distance there between the reactor vessel and the containment vessel in that lower region.

MEMBER CORRADINI: Okay.

MEMBER BLEY: Mike, which figure did you hold up?

MEMBER CORRADINI: Slide seven of his packet we just had which is an engineering drawing.

MR. LASSITER: Correct, taken from an engineering drawing.

MEMBER REMPE: I'm sure you'll talk about it tomorrow, but do you have an idea of the uncertainty in all the condensation heat transfer? I mean do you think your numbers are accurate to plus or minus 20 percent, 50 percent?

MR. LASSITER: That is a question for the chapter 15 folks tomorrow.

MEMBER REMPE: Okay.

MEMBER CORRADINI: They're taking a lot of penalties that a best estimate wouldn't take. That, at least from what we've seen in the PRA calculation when we went to NuScale at the offices, I see a lot of penalties.

MEMBER BLEY: Can you go back to seven? You can just it a seven and it will jump there. Yes, at least you learned one thing today.

(Laughter.)

MEMBER BLEY: Since this is a better rendering and I did like it because the picture is a little different, just above the reactor core out inside in the containment it's kind of something sticking out, annular ring around the reactor vessel down lower. Right there. What is that?

MR. LASSITER: This is a lower reactor flange, so that's where the refueling --

MEMBER BLEY: Okay, those are vertical bolts?

MR. LASSITER: Correct.

MEMBER BLEY: That I'm seeing in there. Okay. So that's where you open it up to get down to front. Okay. And some of the other pictures look like a bit flow restricter, but on this picture it's essentially maintaining the same annulus that you have down below it it looks like. So I probably doesn't figure in very much.

MEMBER CORRADINI: Since we're on that is the RRV above or below that flange? It's above the flange. I don't see it in the cartoon.

MR. LASSITER: It's above the flange.

MEMBER CORRADINI: Okay. Thank you.

CO-CHAIR MARCH-LEUBA: I'm looking at figure 15.6-62 about how much water is in the vessel. And when you open the RRV and you let the cooling rate, the inside and the outside, they reach ten feet above the core, core top, to guard the fill.

MEMBER BLEY: Inside.

CO-CHAIR MARCH-LEUBA: Yes, approximately, because if it did have flow there will be a one foot difference between left and right.

Roughly, the cooling units are about ten feet above the part.

MR. LASSITER: Okay, next slide here.

So only two vent valves and one recirculation valve is required for operation, at least in chapter 16 analyses. Core stays covered for all design basis events and ECCS, as we have discussed, is automatically actuated when required by the module protection system, based on the high containment level where loss of power for 24 hours. So operator action is not required. However, operators can manually actuate ECCS from the control room. The ECCS valves form part of the reactor coolant pressure boundaries, so they're designed and fabricated in accordance with class 1 as ASME class 1 components.

CO-CHAIR MARCH-LEUBA: If an ECCS valve fails to open, is there anything the operator can do to push it, to drive it to position?

In the cartoons that you show, no. The answer is no.

MR. LASSITER: No, they are instructed to not wait until there's an active ECCS demand before they attempt an override, but physically, you know, it's not turning a knob, turn something harder in valve --

CO-CHAIR MARCH-LEUBA: Other than going there with some scissors and cut the power to the solenoids, he cannot drive it in with a wrench.

MR. LASSITER: No.

CO-CHAIR MARCH-LEUBA: Because the reason it would fail would be you have corrosion inside stopping the motion, failing of two solenoids would de-energize it is unbelievable.

MR. LASSITER: Okay, and we haven't talked about this yet, but an auxiliary function of the ECCS is to provide low temperature over pressure protection for the reactor vessel in the start up portion or I guess I should say in the low pressure start up or shut down of the module. And there's a --

MEMBER BLEY: You don't have a brittle failure problem, no. That's when people used to use LTOP all the time.

MR. LASSITER: It precludes the module from getting into that region.

MEMBER BLEY: Yes.

MR. LASSITER: So there's a pressure temperature curve built into the module protection system logic accordingly to prevent over pressurization at low temperature.

MEMBER BLEY: So it is for brittle failure protection.

CO-CHAIR RICCARDELLA: It might be a very generous curve, but it still is at pressure temperature.

MEMBER BLEY: I think that would be very generous. I mean it seems to me that it would probably generous to the point you didn't even need it except for operational reasons where you might not want to over prep.

MR. LASSITER: The curve for that is in chapter 5. I don't have the table off the top of my head.

MEMBER BLEY: Table 5.

MR. LASSITER: Next slide. This is a simplified diagram of the ECCS valves. We're talking about valve operation and configuration. This figure in particular looks more like the recirculation valve on the side of the reactor vessel.

We have -- I'll point with the mouse here. We have the main valve attached to the reactor vessel. We have the inadvertent actuation block device which has been discussed previously attached to the main valve or as part of the main valve design. We'll see the detailed drawings.

MEMBER BLEY: It's actually a separate valve, at least it looks like from the cartoon in chapter -- we'll see that later. I said separate within a housing it's a separate valve.

MR. LASSITER: Correct. There's a trip line tubing going through the containment region and the trip reset actuator assembly which is attached to the containment vessel wall which was discussed in chapter 3 at some extent.

MEMBER CORRADINI: So is the reason you need an outside containment for maintenance and servicing?

MR. LASSITER: Primarily it's for solenoid cooling.

MEMBER CORRADINI: Oh, it is.

MR. LASSITER: So we would either need to bring cooling water into the vacuum region of containment or put them on the outside and submerge in the reactor pool to provide cooling to the solenoids which are --- especially for the trip valve which is normally energized to stay closed.

MEMBER CORRADINI: So all five have to be sitting inside the swimming pool.

MEMBER BLEY: Wait a minute, all five of those pilot valve assemblies.

MR. LASSITER: Correct.

MEMBER CORRADINI: All five of the things that are S and S. IABs are inside. S and S are outside. P is with a real valve.

MEMBER BLEY: IAB that's not solenoid operated?

MEMBER CORRADINI: No, it's simply a pressure differential measure. You pressurize it closed and then you wait for a pressure to allow to open, a pressure difference --

MEMBER BLEY: I'm showing what it accomplishes.

MEMBER CORRADINI: I'm showing what it accomplishes. You're asking me how a valve actually works. It's something different.

MR. LASSITER: Yes, we'll look at the cut away in the closed session. I think that will be much more clear.

Next slide.

MEMBER BLEY: So is it just air cooled outside of containment?

MR. LASSITER: Water, because they're submerged in the reactor pool.

MEMBER BLEY: Oh, they're in the pool. That's right. Thank you.

MEMBER CORRADINI: What is the failure that you're concerned about that you can't have it? Is it just a high temperature limit on the solenoid materials? I guess I don't understand.

MR. LASSITER: It's longevity of the solenoids for operational purposes.

MEMBER BLEY: They just keep heating up in a vacuum.

CO-CHAIR MARCH-LEUBA: If it's in a vacuum, you can only cool them by radiation.

MR. LASSITER: It would be too hot.

MEMBER BLEY: They get really hot. Typically, a solenoid when it gets too hot, whatever kind of insulation, whatever it is, it'll disappear.

MEMBER BALLINGER: You'll get gamma heating as well.

MEMBER BLEY: Slightly more than I know.

MEMBER BALLINGER: And in a vacuum, you get a heat record. There ain't nowhere for it to go.

MR. LASSITER: Okay, on to the next slide. So as we talked about removing power to those solenoid actuators, the trip valves are located out on the containment vessel wall. That actuates the main valve and the main valves operate using reactor coolant pressure and they're also assisted in movement by a spring, an internal spring.

MEMBER BLEY: So in our closed session, I'd be interested in -- it states it the other way in chapter 6 is they're spring operated with some hydraulic assist. The proportion of that would be of some interest.

MR. LASSITER: The proportion is largely hydraulic. The spring is a low-pressure spring.

MEMBER BLEY: So you really don't need the spring.

MR. LASSITER: Around 30 psi, but we need the spring to hold the valve open indefinitely. So after the vessels have depressurized and there's no driving hydraulic force to open the valve.

CO-CHAIR MARCH-LEUBA: I guess we'll go through this maybe closed session or maybe tomorrow, but if your level on the containment is high enough to treat ECCS, your delta P is not that much.

MR. LASSITER: So it's event dependent as to how much delta P is --

CO-CHAIR MARCH-LEUBA: Delta P will be negligible. If you rely on that to be existing, I think you went into the view of all the analysis more accurately.

MR. LASSITER: I think we should defer that question to --

(Simultaneous speaking.)

MEMBER BLEY: I'm not frowning because I

disagree, but my worry is that the simplified using of hydraulic evaporated valve data has given us a biased estimate in the PRA.

CO-CHAIR MARCH-LEUBA: I cannot wait until --

MEMBER BLEY: And over confidence, yes.

CO-CHAIR MARCH-LEUBA: I cannot wait to see all of these hands.

MR. LASSITER: Okay, continue here. Each ECCS main valve includes an inadvertent actuation block device. You know the primary purpose of ECCS is cooling in postulated accidents. But now due to the design of ECCS, an inadvertent actuation block device is engineered to prevent ECCS main valve opening due to an unexpected failure, such as solenoid or power failure while at operating conditions.

The inadvertent actuation block device is not normally engaged and it's strictly a differential pressure device and we'll see detailed figures in the closed session and some diagrams of its operation, but it does not come into play for any valid ECCS demands in chapter 15 analyses. It stays open. It's only for an unexpected failure during operational conditions.

MEMBER BLEY: Pilot valves that live outside the containment are solenoid operated into their normal position, right? And spring operated to their failed, but desired safety position.

MR. LASSITER: That's correct. Well, even the small valves are solenoid operated valves. They're just on a much smaller scale, but there is a spring. The spring assist is also there. They're based on some --

MEMBER BLEY: So the solenoid is not driving them in both directions.

MR. LASSITER: No, the solenoid is energized to close, and fail to open which when the fail to open is by spring and hydraulic actually.

MEMBER BLEY: On both?

MR. LASSITER: Even on the small ones.

MEMBER BLEY: Okay. You'll show us more later.

MR. LASSITER: Yes.

MEMBER BLEY: I sure didn't get that. Okay.

MR. LASSITER: Okay, let's go to the next slide.

So another -- talk about the regeneration in section 6.3. And the regeneration in the NuScale containment is limited by restricting of insulation, paint, and coatings in containment. There is a debris effects evaluation performed, see section 6.3.3.1. And the debris assumed in that evaluation is based on latent debris from refueling or maintenance activities. Therefore, we have a COL item for the applicant to implement a cleanliness program to limit debris within the containment and support the assumption of that evaluation.

And the ECCS is capable of post-accident extended long term cooling beyond 72 hours.

MEMBER CORRADINI: So let's talk about that for a minute. So I want to ask the staff as well. I want to make sure. Have you responded to their RAIs on potentially boron non-equilibrium concentrations within the core and outside the -- in the containment? 8930 is the RAIs that I have written down to remind myself to ask about.

MR. LASSITER: I think that's being addressed by our chapter 15 analysis team.

MEMBER CORRADINI: So you escape again. But not for so long.

But I want to make sure because I thought this is part of the requirement for the long term cooling for ECCS actuation. Because if I get into a situation that all goes well in the short term, I eventually have the situation where I can essentially create a -- not a disequilibrium, but a difference in concentration inside the core and outside in containment in terms of boron concentration. Am I not correct?

CO-CHAIR MARCH-LEUBA: I think we can discuss this at length tomorrow. So they can get the 24-hour relief.

MEMBER CORRADINI: Okay, fine.

CO-CHAIR MARCH-LEUBA: But we will discuss at length tomorrow.

MR. LASSITER: But the long term cooling topical report that's addressed --

MEMBER CORRADINI: The RAI that I was tracking was 8930. I'm looking at staff, are they nodding that I have the right RAI?

MS. KARAS: This is Becky Karas from Reactor Systems. Yes, that's correct and that is going to be described as part of the chapter 15 presentation.

MEMBER CORRADINI: Okay, so this is the one that you guys will not talk about today that you'll talk about tomorrow. So everybody gets a day. Great. Thank you. Move on.

MR. LASSITER: Next slide. This is COL item for section 6.3. If there are no questions, we'll move on to 6.4.

Switch gears to control room habitability system. The control room habitability system provides stored breathing air and pressurization to the control room in the event that the normal control room ventilation is unavailable. It's designed to support 20 personnel for 72 hours. The air inventory is monitored by pressure and temperature instrumentation. The control room habitability system is automatically actuated by the plant protection system or manually inside the control room envelope.

CO-CHAIR MARCH-LEUBA: Just out of curiosity how big are those times? I mean 72 hours worth of air for 20 guys.

MEMBER CORRADINI: A lot.

CO-CHAIR MARCH-LEUBA: Because I know when you go diving you carry something like this and it lasts you for an hour.

MR. LASSITER: There are -- I'm trying to remember the quantity of tanks, but they are about five feet by a foot in diameter. And there are excess inventory to take some out of service.

CO-CHAIR MARCH-LEUBA: You have confidence that's sufficient?

MR. LASSITER: Yes. The design includes

a connection to offsite air supply source if needed past 72 hours. However, power is expected to be available to the normal control room ventilation system after 72 hours.

MEMBER CORRADINI: The requirement is 72 hours.

MR. LASSITER: Correct.

MEMBER CORRADINI: Okay.

MR. LASSITER: Next slide. This is a simplified diagram of the control room habitability system. Air bottles providing the stored breathing air to the control room envelope. There's automatically actuated supply valves. There's a manual valve as well and there are the -- I guess I could point with the mouse -- and the balancing of pressure relief valves. These are solenoid valves which similarly relate to the ECCS and actually are held in the closed position by energizing to close and to fail to open.

DR. SCHULTZ: Is the control room normally pressurized?

MR. LASSITER: The control room is normally pressurized to 1/8th inch water gauge by the normal control room ventilation system.

DR. SCHULTZ: Thank you.

MR. LASSITER: And the control room ventilation system operates in conjunction with the normal control room ventilation system to close the normal supply and return dampers which are shown at the bottom of this figure here.

Go to the next slide.

Due to the fact that NuScale does not rely on operator action for chapter 15 analyses or postulated events, the control room habitability system is classified as non-safety related. However, augmented quality requirements are applied such as all the components which store the air and supply the air are seismic category one. And the COL item which requires the applicant to periodically test and inspect the integrity of the control room envelope as well as the control room habitability components.

The control room habitability system is automatically actuated on loss of AC power to both normal control room ventilation, air handler units for ten minutes; high radiation downstream of the normal control room ventilation intake filtration train; or loss of AC power to the four EDSS battery chargers.

Next slide.

Here are the COL items for section 6.4. That concludes my presentation for section 6.4.

MR. HOUGHTON: All right, so section 6.5 covers fission product removal and control. So we do not have any credited fission product removal systems other than the containment and other components that we've discussed already.

We don't credit the reactor building or any of the HVAC systems in our dose analyses and we do not employ a containment spray system in the NuScale design.

Section 6.6, in-service inspection and testing covers ISI, in-service inspection program, for class 2 and 3 components based on 10 CFR 50.55(a)(g)(3). Pre-service inspections and in-service inspections are performed on class 2 and 3 components in accordance with boiler pressure vessel section 11 requirements and I'll add a clarification here. The containment vessel inspections are performed for class 1 requirements and those are described in table 6.2-3.

The in-service inspection program is discussed during the chapter 3 discussion this morning. It includes augmented volumetric and surface inspections to protect against postulated piping failure areas. So that's specifically looking at the high energy fluid system piping and break exclusions on piping. You'll see those augmented inspections.

There are COL items for development of the actual programs by the applicant. And that concludes the presentation.

MEMBER CORRADINI: Questions by the members?

Okay, no questions. Before we have the staff come up with chapter 6, thanks to NuScale. Let's take a break. We'll be back at 20 til.

(Whereupon, the above-entitled matter went off the record at 2:26 p.m. and resumed at 2:40 p.m.)

MEMBER CORRADINI: Okay, let's get back in session.

Omid, you're up.

MR. TABATABAI: Yes. Good afternoon. My name is Omid Tabatabai. I'm a senior project manager in the Office of New Reactors and I just want to first thank the members for giving us an opportunity to present to you our draft phase 2 SER for chapter 6, engineered safety features.

The reason I mention draft phase 2 is because we still have some open issues that we're still working with NuScale to find resolutions to. So they're not quite open items as we are used to defining them that we haven't mutually agreed --

MEMBER CORRADINI: They're uncommon open

items.

MR. TABATABAI: Yes.

MEMBER CORRADINI: Uncommon.

MR. TABATABAI: Uncommon open items.

MEMBER CORRADINI: So I counted three uncommon open items in chapter 6 that I want to make sure I understand. So when we get to the right point I want to make sure we understand them.

MR. TABATABAI: Sure.

MEMBER CORRADINI: Thank you.

MR. TABATABAI: Thank you. Staff's agenda today, I'm just going to recognize the team who worked on chapter 6 for the staff provide an overview of the NRC staff's presentation and because we want to focus this presentation on chapters -- I'm sorry, subsections that have open items or exemption requests. So we'll focus only on section 64.5.6, 6.3 and 6.4. So we don't have really because of the time and important issues that will be discussed in chapter 3 and chapter 6 issues, we're not going touch on 6.1, material issues, or other things that we don't have any open items for.

This is a list of NRC staff reviewers who were heavily involved in the review of chapter 6. This is just a representation of the staff here who are presenting to you today.

As I mentioned earlier, we won't focus on section 6.2 which is containment systems; 6.3, emergency core cooling system; and control room habitability. These are the sections that either have open items or exemption requests in them.

As we have already informed the members, we won't discuss 6.2.1 containment structures, 6.2.2, containment heat removal systems as part of chapter 15 presentation later this week. And during NuScale's presentation, it required a few questions with respect to these sections that were deferred to to tomorrow and Thursday's presentation.

MEMBER CORRADINI: So let me stop you here and make sure I understand. So today, we're not going to hear about 6.2.1 nor 6.2.2?

MR. TABATABAI: That's correct.

MEMBER CORRADINI: But the uncommon open items are part of 6.2 --

MR. TABATABAI: 6.2.1.

MEMBER CORRADINI: 6.2.1. I'm sorry.

MR. TABATABAI: That's correct.

MEMBER CORRADINI: 6.2.1. I get all my subsections mixed up. Are we even going to discuss those today? Are we going to be discussing those tomorrow?

MR. TABATABAI: They will be discussed tomorrow.

MEMBER CORRADINI: Okay. The reason I'm asking is I read in two places in there that the staff has done audit calculations and I'm curious about are we going to see the audit calculations and what tools were used with the audit calculations. So that is planned for tomorrow.

MR. TABATABAI: That's correct.

MEMBER CORRADINI: Okay, good.

MR. TABATABAI: And also, I just want to make sure you know that there are some open items, uncommon open items for today's presentation as part of chapter 6 which Shanlai will be talk about later on.

MEMBER CORRADINI: Okay. Thank you.

MR. TABATABAI: Just for your information, we issued about 34 requests for additional information which included 114 questions. And we have received about 111 of those questions responded to as of today.

So with that, I'm going to turn the microphone to Clint Ashley to discuss 6.2.4.

Clint?

MR. ASHLEY: Thank you, Omid. Good

afternoon. As Omid indicated, my name is Clint Ashley. And I'll be presenting the staff's review of section 6.2.4, containment isolation. There are actually no open items in this review area, but there are four containment isolation related requirements where NuScale seeks an exemption.

MEMBER CORRADINI: You have to speak up, Clint. Even though you've got your green thing on. We want to hear your words.

MR. ASHLEY: So this slide just contains a containment isolation regulatory basis and related requirements. I don't intend to read these aloud. They're just listed here for information purposes.

And in an upcoming slide, I'll talk specifically about those requirements where NuScale requests an exemption.

This slide shows the review guidance applicable to containment isolation. Again, I just show this for information purposes. And I don't intend to read these aloud.

Next slide, please.

So NuScale's DCA contains several exemption requests associated with containment isolation requirements. They were touched on earlier by Zack Houghton from NuScale.

Just to remind each of us about what these requirements are, for GDC 55, the provisions require in part that each line that is part of the reactor coolant pressure boundary and penetrates primary reactor containment shall be provided with isolation valves.

GDC 55 specifies the location of the valves with one inside and one outside containment. And you can think of the systems that traditionally fall within GDC 55 are like the CVCS system.

The provisions in GDC 56 are nearly identical to GDC 55 except that GDC 56 applies each line that is connected directly to the containment atmosphere. And for both of these GDCs, redundant barriers are required to account for a single act of failure in the isolation provisions. And this is typically achieved by providing two barriers, two isolation valves in series.

NuScale's GDC 55 and 56, piping penetration lines, provide a containment isolation design consisting of two automatic isolation valves located outside containment in series and therefore the applicant requests an exemption request from the requirements of GDC 55 and 56 just regarding valve location.

MEMBER CORRADINI: Historically, have there been any containments with outside isolation valves such as this?

MR. ASHLEY: There have been exemption requests for having isolation valves located both outside.

MEMBER CORRADINI: Okay, so that's happened in the past?

MR. ASHLEY: It has.

MEMBER CORRADINI: Okay. And you approved them?

MR. ASHLEY: I'm sorry?

MEMBER CORRADINI: And the staff approved them?

MR. ASHLEY: The Agency at the time approved them, that's correct.

MEMBER CORRADINI: I assumed that. I should have asked that. Thank you.

CO-CHAIR RICCARDELLA: But there are exemptions and they're limited, right?

MR. ASHLEY: They are. They are limited. And we'll talk more about that in the next few slides.

The provisions in GDC 57 require, in part, that each line that penetrates the primary containment is neither part of the reactor coolant pressure boundary or connects directly to the containment atmosphere. So we're looking at things like confluent cooling water, cooling systems, those things that aren't directly tied in the reactor coolant pressure boundary or the containment atmosphere. Those typically just have at least one isolation valve. And as specified in GDC 57, the valve is located typically outside containment.

In NuScale there are two independent decay heat removal system trains, each with a decay heat removal system steam supply line and a decay heat removal system condensate return line. And the applicant poses the use of closed loop decay heat removal system outside containment as an alternative to the isolation valve requirement. So therefore, because of the difference, they've sought an exemption request for that particular requirement.

And the last item on the slide is a TMI-related requirement. It's 10 CFR 50.34(f)(2)(xiv)(E). And it requires containment isolation systems that include automatic closing on a high radiation signal for all systems that provide a task to the environment.

The NuScale design provides signals to automatically close all systems that provide a task to the environment, but they don't use the high radiation signals. And so as a result, the applicant requests an exemption from the TMI-related requirement.

Next slide, please.

The purpose of this slide is to identify that the Commission will not consider granting an exemption unless special circumstances are present. Special circumstances in part are present whenever application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. For the exemptions that we just discussed on the prior slide, NuScale's position is that special circumstances are present in that the underlying purpose of the requirement or rule is achieved through alternate containment isolation provisions. And we'll talk more about how NuScale meets the underlying purpose of the requirement due to the alternate needs on the next few slides.

So just to tee up in order to assess whether special circumstances exist, it's important to understand the underlying purpose of the requirements. For GDC 55, 56, and 57, the underlying purpose is to provide containment isolation capability that supports the safety function of containment, to provide a barrier to the uncontrolled release of radioactivity to the environment. And generally, this requires a redundant isolation barrier such as achieved by either a valve or a closed system.

The underlying purpose of the TMI-related requirement is to limit radiological releases by ensuring containment isolation for systems that provide a task to the environment. And this is most notably associated with NuScale's containment evacuation system.

Okay. Two more slides. This slide summarizes the staff finding for GDC 55, 56, and 57. And I'd like to emphasize that the design specific review standards section 6.2.4 does discuss adding two isolation valves outside containment. And I think Zack Houghton from NuScale teed up this important issue about the uniqueness of the NuScale's containment design, the harsh environment inside containment.

And if you were going to have these two isolation valves outside the design specific review standards speaks to basically assessing that region between the containment and the first outboard isolation valve. And you want to look at that region with the lens of standard review plan 3.6.2 and associated Branch Technical Position 3-4 and adopt this break exclusion zone. So that's why GDC 55, 56, and 57 have satisfied the alternate containment isolation position.

Next slide, please.

For 50.34(f)(2)(xiv)(E), the NuScale applicant requests an exemption from this requirement as applied to the containment evacuation system. In the DCA, the applicant describes the NuScale design meets the underlying purpose of the rule by isolating the containment evacuation system using two automatic containment isolation signals. The first one, not necessarily in this order, but one would be a containment vessel pressure signal and the second one would be low pressurize the level. And that's not all the signals that are available, but these are the two that would be called upon in the event of trying to contain that radiation or radiological release inside containment.

So the applicant goes on to explain that the NuScale design differs from the traditional large light water reactors designs because reactor core uncovery and the resulting core damage cannot occur without reaching a low pressurized level contained in isolation set points. Therefore, an event similar to

TMI Unit 2 accident is precluded by the NuScale plant design.

The applicant also states that in the NuScale plant design the pressurizer is located well above the level of the reactor core, not connected to the reactor vessel by piping. And any decrease in reactor vessel and inventory to the level of the core would result in complete emptying of the pressurizer and initiation of the pressurizer level contained in isolation signal. And as such, the applicant describes the automatic isolation of the containment evacuation system on a high radiation signal is not required to meet the underlying purpose of the rule, in part, mainly in part, because there's alternate means that would preclude that release of radiation to the environment.

But based on that information and the unique NuScale design, and our review of the TMI Unit 2 related events, we felt very comfortable that alternate means are provided to prevent radiological releases to the environment and in this case, automatic isolation on a high radiation signal is not required to meet the underlying purpose of the rule.

For both of these exemption requests, the staff finds that NuScale's exemption meets the requirements as defined in 50.6. That concludes my presentation.

MEMBER SUNSERI: So just -- I mean maybe I'm reading more into it, but you say it meets the requirements for an exemption, but are you accepting it?

MR. ASHLEY: Well, I think it's ultimately the Commission. I certainly recommend it. They meet the requirements.

MEMBER SUNSERI: Okay. I get it. All right. Thanks. But technically, there's nothing.

MR. ASHLEY: Yes.

MEMBER SUNSERI: Okay.

MS. GRADY: Good afternoon. I'm Anne-Marie Grady and I'm the reviewer for combustible gas control and also for the next presentation containment grate testing.

The regulatory basis for -- I should start off by saying this section combustible gas control has one exemption request which is what I'm going to talk about now and it also has one open item.

The regulatory basis for combustible gas control is to have -- is to meet GDC 41, 42, and 43 of being able to have atmospheric clean up and in section 4, of combustible gas control system and also to meet

10 CFR 50.44(c). 50.54(c) has five aspects that it controls. It requires a mixed containment atmosphere. It requires that hydrogen be controlled below ten percent for severe accident. It requires equipment survivability. It requires containment, hydrogen, and oxygen monitoring post-accident. And it also requires containment integrity post-accident.

And I should also emphasize because this has been -- this has changed in the early 2000s, we're only talking about a severe accident. This regulation used to apply to the design basis accident. Now it's an accident basically that can have 100 percent zirconium oxidation and release all of the hydrogen that it does produce.

Some key design features to evaluate in looking at the exemption request is that the NuScale containment is kept at a very low pressure, sub-atmospheric, during normal operation which severely limits the amount of actual hydrogen -- I'm sorry, actual oxygen in the containment which is a key factor.

NuScale's containment is designed to accommodate a bounding combustion event resulting from hydrogen generation at 72 hours without a loss of integrity or loss of supporting system structures and components.

NuScale has asked, requested an exemption to not have a system to control the hydrogen in the containment and that is primarily because the way that this plant is operated and it really is going to be an oxygen-starved environment, so even though there's significant hydrogen produced from the oxidation of the zirconium, there's barely ever enough oxygen to support combustion. So it almost behaves as if it were a BWR, although certainly not described as such. So it's oxygen starved. There's sufficient hydrogen, but so what?

Now NuScale would say that they don't need a system to control that because they can, in fact, meet the conditions of the combustion and/or detonation in the containment within the first 72 hours considering that there is barely enough oxygen to support the combustion and/or detonation.

Next slide, please.

We focused on the fact that combustible gas control does meet design of NuScale, does meet the other aspects required by 50.44(c). It does have a mixed containment environment. It does provide hydrogen and oxygen monitoring. I'll get to that in a minute. And it does have equipment survivability program and the containment integrity is maintained even with a bounding accident within the first 72 hours.

Therefore, staff recommends that the exemption request be granted because a system to control the hydrogen in the containment is not necessary within the combustion and/or detonation would not be supported.

CO-CHAIR MARCH-LEUBA: If you have the right LOCA all that 100 percent hydrogen mass can move to outside of the containment into the main building. What would be the concentration of hydrogen there?

MS. GRADY: The analysis that was done, the calculations that were done for this design were for LOCA with an intact containment.

CO-CHAIR MARCH-LEUBA: So the LOCA bypasses containment?

MS. GRADY: No. Intact.

CO-CHAIR MARCH-LEUBA: Or intact containment.

MS. GRADY: Intact. So there is nothing leaving the containment except what's --

CO-CHAIR MARCH-LEUBA: Are there some circumstances where highly can move outside containment, what would happen then?

MS. GRADY: There has been an accident analyzed in chapter 19 which was a bypass accident, but I'm not talking about that today. I'm talking about only the LOCAs that could take place in inside containment, containment -- unless the combustion or detonation failed the containment. We're talking about an intact containment.

CO-CHAIR MARCH-LEUBA: We are asking for an exception to the rule. I'm asking is not the equivalent to this rule for the particular conditions of NuScale hydrogen on the pool building instead of hydrogen in containment?

MS. GRADY: I'm sorry, Dr. March-Leuba, but I don't understand your question.

MR. LU: Actually, I can help a little bit. Based on LOCA analysis, at this point there is no issue with the core recovered and then based on what was presented to us. It is unlikely to have that the design base, the LOCA generated --

CO-CHAIR MARCH-LEUBA: What you're saying --

MR. LU: Chapter 19, you're correct.

CO-CHAIR MARCH-LEUBA: You're saying there is no way to get that much hydrogen anyway.

MS. GRADY: Well, that's --

MR. LU: For design basis. Yes.

MS. GRADY: From the perspective of Chapter 15, he's absolutely right. But that has no bearing on combustible gas control regulation. Because it says, you shall model an accident with 100 percent zirconium oxidation, and produce the hydrogen that would be produced.

That's the requirement. It's not a deterministic accident.

CO-CHAIR MARCH-LEUBA: The requirement says that you should produce 100 percent hydrogen and put it in containment. That's what the requirement says.

MS. GRADY: It says model that --

(Simultaneous speaking.)

CO-CHAIR MARCH-LEUBA: I'm asking what are they thinking -- if they had been thinking about NuScale, would it have had a requirement that says also generate 100 percent hydrogen and put it on the building?

MEMBER CORRADINI: No. I'll say no. That's beyond the design base. That's a Chapter 19 question.

CO-CHAIR MARCH-LEUBA: I'm thinking of a picture that is what everybody's doing for Fukushima, of a place that was not supposed to have hydrogen blowing up.

MEMBER CORRADINI: But with current operating plants, that's not a requirement. So it would seem -- I'm going to speak only for myself. That would seem inappropriate for asking this of NuScale if I'm not asking this for current plants.

MS. GRADY: I know what you're asking now. Combustible gas control is combustible gas control in containment, that's the regulation. Now if you're asking me what might occur --

MEMBER CORRADINI: But I think where Jose's going to is what you talked to us about a month ago --

MS. GRADY: Yes.

MEMBER CORRADINI: -- which is nevertheless, you are still worried about hydrogen control within the bioshield relative to --

MS. GRADY: Absolutely.

MEMBER CORRADINI: -- a hydrogen control issue.

MS. GRADY: Yes.

MEMBER CORRADINI: Even though it's not a Chapter 15 issue, it's still an issue.

MS. GRADY: Yes, and that's being

evaluated in Chapter 19.

CO-CHAIR MARCH-LEUBA: And the bioshield was modified to move the hydrogen into the building. What I'm asking is, what will be the concentration -- that's a very big building. Is the concentration there that undetonable?

(No audible response.)

CO-CHAIR MARCH-LEUBA: You don't know.

MS. GRADY: It is a big building, and that's probably a saving grace because there's not that much hydrogen. And there's a significantly large atmosphere. So the concentration would not be a problem in the building.

CO-CHAIR MARCH-LEUBA: I don't think reaching concentration is a problem. But the better answer is to say, even if you put 100 percent hydrogen in the building, nothing happens. So I'm saying that doesn't apply.

MS. GRADY: It would be a low concentration of hydrogen and probably would not lead to combustion.

CO-CHAIR MARCH-LEUBA: I'll be very surprised that it does.

MEMBER CORRADINI: I want to take us back to -- I want to make sure we're all on the same page.

So within containment, I think Anne-Marie has answered the question. But I think where Dr. March-Leuba is going is, we still want to somehow get back to the staff's eventual review --

MS. GRADY: Yes.

MEMBER CORRADINI: -- findings on the appropriateness of the bioshield and any sort of hydrogen concentrations within the bioshield above the container.

MS. GRADY: That's correct.

MEMBER CORRADINI: And that's yet to be -- that's in the process of review.

MS. GRADY: Right. We discussed it preliminarily when we were here with Chapter 19. We will be answering the question in Phase 4, yes. But that's a specific event identified by NuScale.

DR. SCHULTZ: As you do that are you going to go further than the region of the bioshield, because the design of the bioshield reportedly is going to take care of that concentration possibility. Are there other areas that we ought to be concerned about?

MS. GRADY: Well, looking at that from the perspective of multi-module effects, that whether or not there could be combustion under the bioshield and/or detonation under the bioshield, could lead to impacting another module. That's the scope of the review.

DR. SCHULTZ: Thank you.

MS. GRADY: Okay, now the open item. One of the -- as I mentioned earlier, one of the requirements for 50.44(c) is to have hydrogen and oxygen monitoring. And NuScale originally intended to provide the hydrogen and oxygen monitoring in containment post-accident by using an aspect of the sampling system, the post-accident sampling system.

Now, the sampling system is an online system during normal operation. It runs continuously. It records in the main control room. It tells the operators what's going on inside the containment atmosphere. That system can be, in fact, reinstated after an accident.

And in this case, we have to postulate an accident, 50.44(c), that can be reinstated by -- after they have a LOCA, you have the containment isolating, the isolation valves close.

In some time, and it doesn't have to be quickly. In some time, within the first 72 hours, the containment isolation valves could be opened up again in the containment evacuation system and the containment flooding and drain system and establish that flow path again to sample the containment atmosphere. And that was how NuScale proposed to meet the requirements of 50.44(c)(4). So they would be monitoring the containment atmosphere post-accident.

Now NuScale has come in, as you know, with an accident source term topical report, which has associated with it a post-accident sampling exemption request. And they have, as of today, requested that they be able to not meet the dosage requirements for establishing this flow path, this hydrogen and oxygen monitoring.

So we are discussing how this function, monitoring hydrogen and oxygen in containment post-accident, can be reinstated safely. That is under evaluation now, there's been an RAI issued asking them to provide the information to us that this can be done safely, and we're waiting for the response. And their RAI number is 9682, and that's -- so it's an open item.

MEMBER CORRADINI: That's one of the three yet to be responded to in terms of the tallying that Omid showed at the beginning. That still --

MS. GRADY: That has not been responded to yet, but yes.

MEMBER CORRADINI: Okay. Thank you.

MEMBER SKILLMAN: Anne-Marie, I find the wording in that write-up at your fourth bullet interesting. RAI has been issued to determined that this configuration could be safely established.

Why isn't the word, that, if?

MS. GRADY: Oh, I don't know. I guess it was arbitrary.

MEMBER SKILLMAN: I'll give you four inches of lead on each one of those pipes, so maybe six based on real experience. I don't think you want to be drawing a sample after DBA unless you have really shielded all the piping, all the tubing, and everywhere that tubing and piping might go. You do not want to tap into that containment if you've had an accident. That's all I'm going to say.

MS. GRADY: Okay, I will take that under consideration in the review of the response.

MEMBER SKILLMAN: There's real operating experience on that.

MS. GRADY: Hence, the question has asked for a dose analysis to, in fact, establish this configuration. We are mindful of that but thank you.

MR. TABATABAI: I tried to actually quickly change, that, with, if, but I forgot this is a PDF file so I couldn't do it, sorry.

MEMBER CORRADINI: Nicely done.

MS. GRADY: I won't forget.

Okay, so this is containment leakage testing, and this is another topic that NuScale has an exemption request. And the exemption request is that, as you know, containment leak rate testing is required. It's required by Appendix J to do the integrated leak rate test on the containment, which is Type A testing.

It's required to do the local mechanical joints testing on the containment which are the Type B testing, the piping -- the flange openings and also the electrical penetration assemblies and the hatches. And it also requires Type C testing which is on the containment penetrations, which is essentially testing the containment isolation valves.

NuScale proposes to test the containment isolation valves. The Type C testing, as required by Appendix J, proposes to do Type B testing on the mechanical joints, which is the flanges and the electrical penetration assemblies.

But NuScale has requested not to do a Type A test, which is typically a test on the entire vessel, which would be a pneumatic test, and NuScale has several reasons for requesting not to do it. But primarily, their basic reasoning is that they would -- could meet the underlying reason or goal of demonstrating the total containment leakage by other means.

And the other means are that they would, first of all, have met -- that this is an ASME, Section 3, Class 1 vessel, and all that that implies. That they would be doing inservice inspection on this vessel, which is significantly different from the existing plants because they -- the vessel is inspectable both inside and outside at 100 percent of its surface, which is different from the plants that tend to rely on the Type A testing for results.

That they -- let's see, they are -- they have also -- the ASME code requires that they do a hydrostatic test and they will do that as a matter of design, but they have also promised -- proposed they do a pre-service design pressure test. And that really is a pressure test that has more stringent requirements than the ASME hydrostatic test because the ASME hydrostatic test on the vessel would in fact allowed leakage at the mechanical joints on the containment, and that would still pass the test.

They are basically looking as a strength test, as a material flaw test, as anything like that on the entire vessel, and not so much focusing on that. So the pre-service design pressure test that NuScale proposes to perform, once for each containment vessel, is to have a hydrostatic test as the hydrostatic test would be per the code, hold the pressure for 30 minutes, and then if there is any leakage -- not just some, any leakage at all, the vessel would not pass the pre-service design pressure test. It would be an indication that the joints weren't tight enough, and they would have to go back and evaluate what that really means.

Okay, that test is a NuScale proposal. We have accepted that idea, and it's a test that would be ITAACed as well separate from any other ITAAC that is involved in this vessel.

Now, significant time on NuScale's part, I'm sure, certainly on mine was to what you evaluate the design of the mechanical joints. As many of you people know here, the typical leakage -- the preponderance of the leakage from a containment is from the penetrations and the mechanical joints. NuScale has done an analysis of their mechanical joints per the ASME code.

They have analyzed the bolt preload values so that the flanges would be tightened down sufficiently. They have established what the -- using ANSYS and a finite element analysis. They have established what the minimum bolt preloads would be for all these mechanical joints.

Then they have -- recognizing the fact that -- obviously, a phenomenon called prying that could occur. I mean, these flanges have many bolts. And if one were to fail, the pressure -- the internal pressure could, in fact, cause the increase in pressure to the adjacent bolts, and before you know it you've unzipped the flange. So they haven't stopped with the minimum preload per the ASME code, they've also determined what the maximum preload would be for the bolts they have designed and the materials they have chosen.

And then establishing a bolt preload for each of the different joints, and they're all different. The preloads are all different.

And I should say one thing about the mechanical joints. There are circular closures. They have two concentric O-rings with mechanic -- with metal seals that are set in grooves. They're all specified individually because they are different sizes. They bolt preloads are all different.

And then they did a finite element analysis on all of these mechanical joints to determine what the gap would be after imposing pressure -- internal pressure, which would represent two different LOCAs in the containment.

And one of them was -- one of the scenarios they analyzed just from -- well, on sensitivity -- one of them was the RRV inadvertent actuation, and the other one was the CVCS line breaking side containment. And they got pressures and temperatures resulting from those accidents. They analyzed the joints.

They calculated what the gaps would be on the flanges, and they showed flange by flange that the gaps would be very small. They would be very small typically on the inside of the flange, but the gap would not proceed past the first seal in the flange joint and progress to the outer one.

So, in other words, the design tells them that under these pressure and temperature conditions that the flanges would not open, would not leak. That's what they say. And that is why they have proposed to have the pre-service design pressure test to confirm what they've already analyzed and shown in their analysis.

Since these features -- the features of the fact that this vessel is an ASME design since they have done an analysis that shows that leakage is entirely unlikely. Since the other types of testing -- types being types and testing will still be done pneumatically per Appendix J.

And since -- I think that's the only other feature I want to mention. Then I would recommend that the Commission grant an exemption request from Type A testing.

And if you have any questions?

(No audible response.)

MS. GRADY: Thank you.

MEMBER CORRADINI: I'm happy. I'm happy. I'll let the members speak for themselves. Thank you.

MR. LU: Okay. Shanlai Lu from staff, Reactor System Branch. I'm going to cover ECCS system 6.3.

I think the staff is expecting the committee to have questions, as Michael mentioned already at the beginning that you really understand the open items, especially those uncommon open items.

So we do have limitations here. This is an open session, so staff and NuScale are preparing to have a closed session this afternoon after this one.

We'll talk about details of the valve designs so I can answer any questions about how the ECCS system functions and why it works.

And then after that one and tomorrow, you can see a lot of my open items in Section 6.3, they're actually related to the ECCS performance evaluation, LOCA and non-LOCA, and then also single-filler stuff there.

So it's all -- and, you know, will be presented tomorrow. And tomorrow there is also two sessions. There is an open session. There is also a closed session. So while it's our staff's intention to brief and inform the committee as much as possible to really understand all those issues we have on our plate.

So the strategy I have for this presentation is I'm going to go through a top-down approach instead of going through issue-by-issue explained one by one. What is the problem -- it goes through top-down so that you are saved within the limited amount of time you can see what's the big picture of the puzzle looks like.

And then, in this open session, if you have questions, you can ask, but if there is detailed information, we need to defer to the closed session to do that.

And then there a lot of questions will be related to Chapter 15 analysis, which we're going to cover tomorrow. Including myself, I am going to support other reviewers to do that.

So with that, I'm going through the first slide. Okay, NuScale has already covered this one. And you can see that from the ECCS system designs, that they are extremely simple, three RVVs, two RRVs, and each ECCS has its own IAB trip valve and trip reset valve. That's it.

But on top of that, there are additional -- its containment functions as the part of ECCS. Once the valve opens, containment provides the cooling, and then directly to the pool. So not only does it works with the LOCA event, it works for all other long-term cooling events too, once the ECCS is actuated.

So with that one -- and then, I'm going to point out that that's a unique feature of that -- you can stick them back -- a unique feature of this ECCS system. No other design or possibly line has such a simple ECCS system. It's very unique.

But another unique feature is that it does not have additional neutron poison injection. That was required by GDC 27. That's the reason for a while that the staff has already engaged with the committee to talk about that one.

And that's also unique because no other PWRs do not have that feature. They all can inject additional boron poison into the reactor during a LOCA event or ECCS actuation including AP1000 capacity line too.

So that will lead to the next --

MEMBER BLEY: Can I ask you a question about --

MR. LU: Sure.

MEMBER BLEY: -- how the design works? I don't think we talked about this.

MR. LU: Okay.

MEMBER BLEY: The vendor -- so under our LOCA we had a long talk about how much you flood up --

MR. LU: Right.

MEMBER BLEY: -- open up those valves, and everything works fine. For long-term cooling otherwise, if you use this system --

MR. LU: Right.

MEMBER BLEY: -- are there requirements on when you open the RRVs and the RVVs?

MR. LU: Right.

Okay, actually that's part of Chapter 15 zero scenario for non-LOCA.

MEMBER BLEY: That's part of Chapter 19.

MR. LU: Yes, also.

MEMBER BLEY: What's the design scenario?

MR. LU: Okay. All right. So there are quite different scenarios that will lead to ECCS actuation. One case that definitely is a LOCA, you have, you know, a serious accidental line break. As we just discussed, about that you had dumped enough water into the containment, the water level flood of containment, and that will trigger the ECCS actuation.

There was one point here I think somebody asked a question about how is IAB is going to function if you have very quick blowdown, the system pressure quickly goes down below the setpoint of IAB, for example, 1,100 pounds. Then IAB does not do anything, valve opens -- all five ECCS valves are supposed to open, but of course, you can assume certain valve failures.

But in reality, they all need just two RVV and one RRV. So once you blast it open, you dump the inventory from the vessel to the containment, then you establish the natural circulation.

Actually, I think NuScale had a very

beautiful --

MEMBER BLEY: They did a good job on the LOCA one, I don't remember --

MR. LU: So for the non-LOCA side, for example, if there is an event and then I think we can talk a bit more tomorrow. And then, for example, let's assume the DC power everywhere -- we would lose AC power 24 hours later. It's actually -- there's a timer that actuates the ECCS.

If at that time, the system pressure is higher than the 1,100 pounds of pressure, IAB would block all five of them.

MEMBER BLEY: All five of them.

MR. LU: It can add a blast of water into the containment. However, during 24 hours, normally we expect the DHRS would start to function and start to cool them down. And then by the time you really reach 24 hours -- depending on scenario -- you analyze -- then there is a possibility most likely your pressure will be lower than the 1,100 pounds setpoint, IAB will not block your blast open all five valves at that point. So that's for the long-term cooling.

Okay, any questions for this slide?

(No audible response.)

MR. LU: Okay, we're good. Let's go --

the regulatory basis -- so I'm going through the top-down approach. So I'll start from regulatory basis and that I think you guys are all familiar with that one. So I'm not going to spend time to explain one by one.

MR. LU: But I'm going to summarize, the GDC 2 and GDC 4 have been covered by the environmental review of Chapter 3, the sharing of structure system and components as covered by the pools, how many modules, you know, are shared. It's not a big deal.

The electrical power system requirements and then that's the part you probably want to pay attention to. But I think the electrical branch, I've already covered that one because this one does not have the safety grade of DC power.

Neutron poison addition and the appropriate shutdown margin for stuck rods. I think that's part where lead to one of the uncommon open items here. Because if it does not have Neutron poison addition capability and resolving the re-critical so there's no -- if it reaches the re-critical reason sometimes there's no shutdown margin for stuck rods.

So that's the part the staff has been working with NuScale and back and forth for a while.

And also I think the other staff from the entry system has already briefed the meeting about that.

All right. Can we just touch a few more bullets there? Long-term cooling, okay, 50.46(b), long-term cooling, although we did not and, you know, have any specific issue related to rule or regulation, but to really make sure it works, the system can remove not only decayed but also if you have re-critical, lots of power, how are you going to remove the power? That's the issue there.

Related to the testing, and I think Tom already covered that we are, you know, observing and auditing the NuScale's ECCS performance testing program. So it's ongoing. Actually, it's going according to the plan and then we are going to go back to re-observe the test.

Okay. Next slide. This part I don't want to touch too much because that's a typical DSRS for 623. We modify that one based on -- back to 2014 there is a white paper about the specific NuScale ECCS system. So based on that, we revised the DSRS. We'll follow that one.

And the TMI action item, one of them related to boron, boron dilution from precipitation, so that's a part that I'm going to cover a little bit.

There are two exemption requests here. The first one actually as I mention it because the NuScale ECCS system design is passive, simple and what they claim there is a lot of margin because two freezable level is always above the core for the largest design basis LOCA. So, therefore, the core never experienced a reflood dry out. So those are the issues back to 2014.

There was a white paper submitted by NuScale to the staff. We reviewed that. But basically, we claim at that time was there was no IAB design so basically they said because of this over a combination of the walls and the containment concept and that those are the items they are going to seek the exemption request from us as part of 50.46, Appendix K requirements.

And we actually at that time, you know, we made a decision very quickly is that, yes, sounds good.

So it's not a final approval, but we basically gave them our feeling because of, you know, our recommendation or our comments, yes, looks good to us. Because we had exposure to their testing program. We had exposure at that time whether they predict a two freezable level.

As long as they can support that claim that at the end for the most limiting case, they do not have core uncovered, I think this part makes sense. Of course, there is no pump. And a pump needed to be there.

So this partial exemption at this point as part of the ECCS performance evaluation and we feel comfortable and we recommended the approval of this exemption request. And we are going to cover a little more tomorrow as part of a LOCA analysis and LOCA topical analysis and summary.

GDC 27, okay. We haven't talked about the detail analysis of this one, especially we have a proprietary session tomorrow to talk about the staff calculations, evaluations.

And as I mentioned right at the beginning, the unique features of this ECCS system, it does not have neutron poison injection addition capability in comparison with all other PWRs in the world.

So what is going to be the impact on this design that they already told us it's going to become a re-critical. And on top of re-criticality, they felt that they are not going to change the CHF. So staff is evaluating their space safeguard analysis to see whether it's sufficient enough for us to prove this exemption or not.

So that's where we have uncommon open items. All right. Now --

MEMBER BLEY: And that one is really breaking new ground.

MR. LU: That's their side. Yes.

MEMBER BLEY: Yes.

MR. LU: That's the reason it's not something --

MEMBER BLEY: For you, too, I mean, you've never proved something quite like this.

MR. LU: We were aware about that issue --

MEMBER BLEY: Oh, you knew it was coming, no?

MR. LU: -- three months before this DCD submittal. So we asked --

MEMBER BLEY: But you never issued a license allowing a condition like this?

MR. LU: We in the past history, never.

MEMBER BLEY: Never, ever.

MR. LU: Never. And actually one of the -- for the Fukushima event, I think all the world regulatory agency grouped together and developed two common positions. Although we have to deal with our kinds of different governments differences, but there was only two. Remove decayed heat and maintain core subcritical during downtime cooling.

So I'm not so sure. But I'm not only a reviewer. I think also, you know, it's another reviewer's responsibility to call the final shot on that one. But that's my review on the ECCS system.

So I'm going to go through from a top down perspective the pieces which can give you a color of the puzzle of what sections impact ECCS performance.

So the first one, reactor coolant boundary, as Tom has already covered that one. And so open item is 3.96-1 related to ECCS performance testing needed. So we are doing that. And actually there is a lot of good activity.

And then the conceptual design last testing last year went well. And so we are expecting hopefully this one goes through easily, too. But it's ongoing so we cannot say one way or another at this point.

All right. Low temperature over pressure protection. In Chapter 5 of the presentation, HR has asked a question. I was not present. And the reviewer passed me the question.

So I think I can answer that one. And then the concern at that time was, oh, you have IAB.

Are you going to block it at the ECCS actuation to protection their low temperature over pressure.

And then we checked with NuScale and they provided the actual protection curves. And then the LTOP through ECCS system is enabled only below 750. So therefore IAB should not interfere during the start-up. So we are still having some conversation with them over the come down with the power. So that might be the case.

MEMBER BLEY: That's all automatic here. It's not --

MR. LU: Yes. It's automatic. It's not menu. It's not menu. But there is also a menu operation to main they are supposed to follow to protect.

All right. All right. Core, you know, I think the open Item 6.3, that's wrong. It's the core cooling related to inadvertent opening of ECCS valves. And then it's right now it's pending on the LOCA topical report at 50.6 review, Open Item 6.3-1. Core cooling secondary --

MEMBER CORRADINI: I don't think I understand that.

MR. LU: Okay.

MEMBER CORRADINI: I'm sorry.

MR. LU: Sure. Go ahead, Mike.

MEMBER CORRADINI: Is your tracking as an open item because it's a matter of the modeling tool has yet to be upgraded because it's related to 15.02-2, which is essentially the NRELAP Version 1.4 that is yet to be reviewed. Is that not correct?

MR. LU: You're correct.

MEMBER CORRADINI: Okay.

MR. LU: And actually that's the ECCS performance valuation. At 6.3, it's a summary of all those sections, supporting sections and then the performance evaluation satisfies all those basic criteria.

MEMBER BLEY: Okay. Well, and the first half of this --

MR. LU: Right.

MEMBER CORRADINI: -- is this inadvertent ECCS, is that one of the LOCAs that's analyzed in the technical report?

MR. LU: Yes, sure.

MEMBER CORRADINI: It is, isn't it?

MR. LU: Yes. It's one of the -- they claim that the -- the NuScale claim is one of the transit.

MEMBER CORRADINI: There is --

MR. LU: Non-LOCA transit.

MEMBER CORRADINI: -- we're going to go back to it tomorrow. But there are six open items related to the model, the tool. And 15.02-2 is one of those in terms of essentially there's a new version of -- if I understand correctly, there's a new version of NRELAP that staff is still reviewing compared from 1.3 to 1.4 and that affects essentially all the accident analyses. Thanks.

MR. LU: Okay. All right. That's a similar issue related to non-LOCA because during non-LOCA the ECCS system at a certain point will be actuated to maybe the non-LOCA. So therefore that part is open.

Those open items are what we consider with a clear path forward. Okay. And core cooling loss of coolant accident also we consider with a clear path forward. It is related to the Item 3.62-1 and also 15.0 on 2-2. It's a LOCA topical report.

Okay. The core cooling long-term cooling return to power and boron transport. That's the section of 15.06 and staff is going to cover that in detail. That's what we considered an uncommon open item.

Shared system, that part is closed. Let's move forward. Next slide.

MEMBER BLEY: Are you going to get to that one tomorrow or that will be later?

MR. LU: Actually, the staff will prepare a lot of material for tomorrow --

MEMBER BLEY: All right. Okay.

MR. LU: -- as part 15.0.6 and actually my management had asked me to talk about the boron transport, too. So I will be back here tomorrow, too.

MEMBER BLEY: Okay, good.

MR. LU: So if you have any question, you can ask me at that time, too. All right. Closed. And then power requirements related to non-safety graded EC power system. And I think there is a pass forward there already.

Instrumentation, Jose, you asked a question about the level and initiating our LOCA review. And you asked a question how you did you measure the level and then how did you trigger your protection system?

And the initial that they had a vessel with level sensors there. So our questioning at that time was through the Digital I&C Chapter 7 guises. Oh, you have a high dose, high temperature, high pressure. Are you sure your readout where your type of detector would ever function in that vessel.

CO-CHAIR MARCH-LEUBA: My comment was more related to the possibility of digital common cause failures, which we always worry because we cannot quantify them easily.

MR. LU: Yes.

CO-CHAIR MARCH-LEUBA: They go undetected for two years because you're not using the sensor.

MR. LU: Yes. I think that's the similar aspect. So later that they switched to all containment. And that particular issue, I recall that I was talking to the digital I&C guys -- at that time it's part of inspection, the maintenance and the in-service testing.

So how they're going to do that -- I think at this point Chapter 7. Okay.

CO-CHAIR MARCH-LEUBA: Yes. We keep it in mind for Chapter 7 and maybe we will repeat tomorrow again.

MR. LU: Okay. System boundary closed. And when NuScale presented that one, we had the trip set valve and then the reset valve all enclosed as a part of containment boundary.

Testing is batching qualification part of Tom's performance testing. We are going to go through the process to address a lot of issues related to testing.

The environmental requirements is closed as I mentioned earlier and actually the Chapter 3 guys presented on that one.

System reliability, failure mode and the effects analysis also related to the task. And we want to make sure what we, you know, suspect are the issues. For example, water hammer, what's going to happen there? So that's the part of a testing we're going through this week and next week.

Single failure and it's an open item right now. And we are going to follow Commission's decision.

Technical specification, closed. Chapter 7 --

MEMBER CORRADINI: Let's stop there for a second. So that is still pending?

MR. LU: I have not seen the Commission's final decision paper yet.

MEMBER CORRADINI: Thank you.

MR. LU: Okay. Any questions on this page? Next slide.

All right. Let's talk about now that's the top down view of all the issues inter-related to

ECCS evaluation. And now as part of DCD 6.3, they submitted the in-vessel downstream effects as part of the DCD submittal.

Actually, they had a lot of good details inside of that Section .3 write-up. So we reviewed that one. As it was, you know, discussed this morning that they only used the reflector metallic installations. And there is no fiber, no insulation nor a particular insulation, no chemical buffer.

So the fiber and particulate and all those are intentional and some are insulation within the zoning infinites are excluded. So, therefore, all they have is they slosh the containment through the cleaner process and then it's unlikely there is only going to be even latent debris. But they are digging post their COL item to ensure the cleanness. I think that's sufficient.

Debris transport for the given assumed latent debris, we assume 100 percent transportable from containment to the RRV to the inlet of the core.

The good news here is that they are using AREVA fuel. So the AREVA provided them actually our reference testing results. And so the results show there was no problem for the given debris loading they are expecting.

If you really pay attention to the number, 5.6 gram per assembly fiber, it's lower than 15 gram. We are approving the operating fleet.

So they only have 5.6 for the given COL action item, which is already conservative from my perspective. So therefore on top of that they assume very conservative chemical participants although they do not have chemical buffer.

So where does it come from? They just want to make sure it's conservative and staff has no more questions. I did not have more questions after reading. That's why it's approval. So I don't have a problem with that one. So that part is done.

MEMBER CORRADINI: What is this -- I think I understand what you're saying here in terms of their assumptions. But how do they arrive at these values?

MR. LU: Okay. Arrive at the values as part of the -- they just assume for the certain amount that's part of cleaning program there is a -- I think there is a merit about a specific number.

And then checking certain amount of latent debris they could even find in the containment and then so they just justified that they do not have a certain amount of total number of fiber in particular. That's part of what the cleaning program would do.

And then you divide the total number by the total number of assemblies. That's what you get for the fiber.

CO-CHAIR MARCH-LEUBA: Is it a speck on the pool that the pool water's cleanliness? We talked this morning about algae growing on certain pools.

MR. LU: Okay.

CO-CHAIR MARCH-LEUBA: Because the only way that debris can make it after refueling --

MR. LU: Right.

CO-CHAIR MARCH-LEUBA: -- is if it comes with the water in the pool and then you evaporate the water and leave the debris behind, correct?

MR. LU: Yes.

CO-CHAIR MARCH-LEUBA: Because all I was going to see how --

MR. LU: Okay. Yes. I think my understanding, when I was reviewing that part -- that's a good question. Thank you.

They can drain the water through the -- they have to drain the water eventually, right? So by the time they really drain that one, and I'm not so sure there's any, you know, significant amount of, you know, debris would be there anymore.

CO-CHAIR MARCH-LEUBA: There is no drain

at the bottom. They are pushing from the top and eventually they evaporate.

MR. LU: Right. Right. Yes, but --

CO-CHAIR MARCH-LEUBA: So any solids will settle down.

MR. LU: Yes. That's what I was thinking about since they are moving the entire containment from one side to another side. By the time it gets sloshed all around, they stir up all the dirt if there is any.

CO-CHAIR MARCH-LEUBA: Yes.

MR. LU: And then they have clean-up system continuously. So that part you can also assume, but with the margins we have we're talking about -- let's say if we're down more area, they only can because they have 5.6 gram per assembly fiber. That's three times. That means your cleaner's program, let's see, if you really have this scenario, you can still survive.

CO-CHAIR MARCH-LEUBA: I'm pretty sure that if you start growing a green algae on top of the pool that --.

MR. LU: Yes. I would say that nobody wants to see muddy water, you know, in a containment pool. All right. Next slide.

Water hammer, okay. It's part of the requirement that we review the water hammer for the ECCS system. But in a standard ECCS system, you have large pipe. You have a 90 degree bend. When you have water coming in, there's a possibility you have water hammer.

For this one, the people were asking me why do ask questions? And then that was because between the actuator line and the IAB you have a very long line. And from RRV to RVV to that the trip set valve there is a long line.

MEMBER CORRADINI: It's a sampling line. It's a small line.

MR. LU: It's a small line. That's exactly the case. So we asked as part of Tom's performance testing requirement. And we actually imposed that it was specific to maintain the pressure of the trip actuator line, trip set off line, make sure that it's high temperature, high pressure temperature up to 400 degree Fahrenheit.

So when trip valve opens, boom. We have a critical flow going through. So what's going to happen to those vents? So the good news at least at this point we observed one of the tests that did not show water hammer after they put quite a few vents there. So we are waiting for these testing results and then waiting for their analysis coming back to show, yes, it's not a problem.

So this particular item, and it's part of open item associated with Tom's performance testing, and I think we have a clear pass forward. So next slide.

Conclusion. This particular section and I'm not surprised because that's brand new in the latest design so we have open items there. But we have seven open items.

And right now we have three open items of what we consider uncommon. But actually the second one, the second paper, we got an IAB single filter where we cannot discuss tomorrow. We're going to follow the Commission's direction.

So ECCS demonstration testing, although we are still saying it's not done yet, but we hope this testing will go well and that we observe that one. And it looks like NuScale already responded to staff's requests. They did what we asked during the audit process in March and then they assembled the testing program well. And so we hope that will be okay.

Boron dilution during long-term cooling is a sticking issue. So we are going to talk about it tomorrow. We have detailed slides and discussion about that.

So that's what I got. Any other questions?

CO-CHAIR MARCH-LEUBA: Maybe I missed the presentation, but on the reliability of the valves, there are three, five valves in the system?

MR. LU: Okay.

CO-CHAIR MARCH-LEUBA: And they're each kind of mobile design?

MEMBER BLEY: Yes.

CO-CHAIR MARCH-LEUBA: I mean, complicated enough so when I see a drawing I don't see how it works.

MR. LU: That's a perfect question. I think that will be covered by our closed session.

CO-CHAIR MARCH-LEUBA: The closed session. But you --

MR. LU: Because until you see the details.

CO-CHAIR MARCH-LEUBA: Can you tell us the conclusions in the open session? I mean, don't give us the details, but what are the conclusions?

MR. LU: The conclusions, the reason I'm still in the open item, actually it's Tommy that's still only in open item stages because we cannot give a person a conclusion until we see something docketed from NuScale.

CO-CHAIR MARCH-LEUBA: Fair enough.

MR. LU: Any other questions?

MEMBER CORRADINI: Other questions? Turning to 6.4.

MS. HART: Yes. I'm Michelle Hart. I'm with radiation protection and accident consequences. And I'll be talking about the control room radiological habitability.

The main regulation that we look at is GDC 19 which has a design criterion of 5 rem TEDE for the duration of the accident.

However, I do have an open item 6.4-1 because NuScale did request exemption from the control room design criteria in GDC 19 to instead propose a NuScale principle design criterion 19.

The effect of that proposed exemption does not change the dose requirements. It does make a chance to the remote equipment outside the control room that would refer to being able to take it to safe shutdown instead of cold shutdown and hot shutdown.

Although there is no change to the dose requirement, I am tracking it as an open item in 6.4

for the radiological habitability in that I would have to refer to PDC 19 if we were referring to that instead of GDC 19.

Next slide, please. The next topic is that the dose to control room operators from accidents is analyzed in DCA, the FSAR Chapter 15. The specifics of the radiological consequence analysis are presented in Chapter 15.

Those do include several design basis accidents and also includes direct dose contribution from the core damage event from filters and outside radiation cloud.

Open Item 6.4-3 is that there are revisions to the dose analyses that were provided in conjunction with Revision 3 to the accident source term methodology topical report, which was just received in late April 2019.

At the time of writing of this Phase 2 draft SER we did not have that information in hand. And so that review is certainly not complete. We've just gotten it. There are not a lot of changes from what was seen before. But the review is just not complete at this time.

So to go to the big topic in control room radiological habitability, one of the things that

NuScale did propose is that the post-accident control room habitability is not a safety-related function. And that is because there are no operator actions that are required or credited to mitigate the consequences of design basis accidents. And also there are no post-accident long-term monitoring that is necessary to be taken from the control room.

Open Item 6.4-2, this relates to a discussion that Anne-Marie had brought up earlier. Because they are required to monitor oxygen and hydrogen concentrations in the containment, they do have to set up that monitoring pathway. They have to reopen the containment to be able to have those monitors take those concentrations.

It is not the same thing as sampling. It's in the same system as sampling. They do have an exemption request from taking samples altogether.

MEMBER CORRADINI: So I don't think I appreciate that subtlety. Can you try that one on me again?

MS. HART: So a TMI related action item requirement in 10 CFR 34(f)(2), I think it's (viii), would have you take samples quickly after an accident. And they have requested an exemption from that regulation. We have not completed the review of that.

And so that's the sampling requirement. And so that may be sampling for chemical purposes and also may include combustible gas sampling.

However, for the requirement for 50.44(c), they have to monitor combustible gasses and that's continuously monitor. It's in the same pathway that they have in their systems. It's in the process sampling system and it's online monitor.

So we have some questions about whether there are actions they have to take from the control room for 6.4, the open item. Are there actions that they would have to take from the control room to be able to set up that pathway to be able to do combustion gas monitoring?

MEMBER CORRADINI: So they're not taking a sample. They're going to have to use the same flow pathway for some sort of monitoring post-accident?

MS. HART: That's correct. And so we still have concerns about -- in Chapter 12, we still have some concerns about if there are local actions they have to take, is there an impact to the dose --

MEMBER CORRADINI: Okay.

MS. HART: -- to the operator to take those actions?

MEMBER CORRADINI: I remember that from

Chapter 12.

MS. HART: Right.

MEMBER CORRADINI: Okay.

MS. HART: But from Chapter 6 perspective, are there actions they have to take in the control room to be able to do this? So this goes directly to that discussion as to whether control room habitability is a safety related function or not.

MEMBER BROWN: Before you leave --

MS. HART: Mm-hmm.

MEMBER BROWN: -- put aside the sampling thing. What does it mean to the control room to be either safety related or not safety related? I'm having a hard time --

MS. HART: Right.

MEMBER BROWN: -- figuring out why the operation of the control room -- regardless of what operator actions are required or not required is not a safety-related --

MS. HART: Right.

MEMBER BROWN: -- need to me. I don't understand the difference, okay? What does it mean to be non-safety related? Does that mean they can go home and have a beer or abandon --

MS. HART: What I can tell you is that

NuScale has stated that within the context of having to have control room radiological habitability be supplied by safety related equipment that that post-accident radiological habitability in the control room is not a safety function because there are no operator actions that are required for the control room to keep.

MEMBER BROWN: They can leave because it doesn't have to be a radiation -- you don't have to be able to stay in there during the accident. That's the way I read your --

MS. HART: They do have systems to provide that. They do not need to be safety related to assure it to that higher mandate that a safety-related function would have you take. I know it's a very fine point. We've had a lot of discussions about this.

MEMBER BROWN: Well, there's a lot of fine points. I mean, here we can stay critical for days after a plant shuts down without having to -- I mean stack on another one. We don't have to be radiologically secure in the control room.

MS. HART: Right. Certainly, their plan is not to evacuate the control room. Their plan is to stay in --

MEMBER BROWN: I understand the plan, but

--

MS. HART: Right. There are no actions that are necessary from the control room. So if they did have to evacuate --

MEMBER CORRADINI: So let me turn Charlie's question around a different way. If it were safety related, what would be required that they're not providing already now?

MS. HART: Safety related ventilation systems.

CO-CHAIR MARCH-LEUBA: And maybe you want to back up. I mean you have to have two systems. And right now you have only one, right?

MS. HART: You may only need one system. But it needs to be safety related.

CO-CHAIR MARCH-LEUBA: Do they have --

MS. HART: It's like all the other plants have one safety related --

CO-CHAIR MARCH-LEUBA: Typically, if you leave the room --

(Simultaneous speaking.)

MS. HART: -- control room habitability system. It's a --

CO-CHAIR MARCH-LEUBA: Requires.

MS. HART: -- filtration system or like in a thousands of cases --

MEMBER CORRADINI: The filtration and air supply system they have is provided. It's just not safety related.

MS. HART: That is correct.

CO-CHAIR MARCH-LEUBA: I think it's an excellent solution that NuScale has proposed to have compressed air and feed them -- the operators as if they were diving under water. I mean, you just use your compressed air, which we know is clean and it hasn't been affected by the accident. The question is do you want it classified as safety related or not?

MS. HART: Correct. So their systems look similar to other systems we've seen in other facilities. You know, we do have bottled air systems in some currently operating plants.

The AP1000 also has a similar safety related, in their case, bottled air system that they rely on for control room habitability.

CO-CHAIR MARCH-LEUBA: But my concern of your Open Item 6.2 is -- I'm not concerned about control room habitability. They have air, and they have a system.

But if you decide that something is a safety-related action, then you need a minimum DC power and operability of those valves and operability of your instruments. So if you decide that is a safety-related action, it's a cascade of events.

MS. HART: That is correct.

CO-CHAIR MARCH-LEUBA: It is not just the bottled air.

MS. HART: And that's why it's still under discussion at this point.

CO-CHAIR MARCH-LEUBA: It's a big decision.

MS. HART: Correct.

DR. SCHULTZ: Michelle, you said earlier, at least I heard earlier and then we're going to talk about it tomorrow, the expectation or the requirement is still that the control room dose will be less than 5 rem.

MS. HART: That is correct.

DR. SCHULTZ: Okay. So we'll talk more about that.

MS. HART: Okay. So to get to that point that we were just talking about, the NuScale design does have two systems that they use for ventilation and/or habitability in their control room. And the credit they've taken in those dose analyses is for these systems that are not engineered safety features.

And so in effect the control ventilation system and the control room habitability system, the normal ventilation system and the bottled air system are backup systems for each other. They're both designed to be reliable and capable of operation during accident conditions. And in this case the accident duration is 30 days as for all other plants so far.

They are independent and diverse systems. The bottled air system does not use the same injection points as the duct work for the ventilation system.

They both have automatic initiation with different signals based on the radiation. And for the habitability system, the bottled air system as they described earlier, they do have initiation signals on that based on power loss. There's a couple different flavors of that and also there's a backup manual for those systems.

So in the dose analyses, they analyzed two different dose analysis cases. For the one case, they assume that the bottled air system operates for 72 hours. Then the control room ventilation system in the supplemental filtration mode comes on after that and operates through the rest of the duration of the accident. And that's 30 days.

In the second case, they assume that the normal ventilation system stays in supplemental filtration mode for the duration of the event. And in both cases, they meet the dose criteria for all accidents analyzed, all those DBAs in Chapter 15.

So we did have some concerns about the apparent reliance on both systems over the duration of the accident to meet the control room requirements in that first case that they analyzed where the --

MEMBER CORRADINI: It's only the first case that --

MS. HART: That's correct, yes. The second case it's only the one system that they're relying on.

For that first case, the control room habitability system does operate for 72 hours. And then the control room ventilation will come in filtration mode. So we asked them some questions about the sensitivity if those systems would fail to operate as expected.

And so they provided us some sensitivity analyses where the control habitability system, the bottled air system, operates for the first 72 hours and then the control room ventilation system fails and the control room habitability system does not come back in. So there's no ventilation filtration after the 72 hours.

They also provided a more extreme case where neither the bottled air system nor the control room ventilation supplemental filtration system operate.

And all of those show that they meet the dose criteria. They do have higher doses than are reported in the DCD, of course. But they still do show doses that are less than 5 rem criterion.

DR. SCHULTZ: Michelle, where are these analyses reported? Was that in response to an RAI?

MS. HART: It's in response to an RAI, yes. And unfortunately, I don't have the RAI responses on me.

MEMBER BROWN: When you say they're higher, what do you mean they're higher?

MS. HART: They're higher than 2.14 rem.

MEMBER BROWN: But what does that mean? Are they still considered --

MS. HART: They're still less --

MEMBER BROWN: Are they 25?

MS. HART: They're less than 5 rem.

MEMBER BROWN: They're less than 5.

MS. HART: But they're higher than what

are reported in the design certification documents. So therefore what's reported in their FSAR is not the result of their sensitivity analysis is what I'm trying to get across.

DR. SCHULTZ: The 5 rem is accumulated over the 30 day period?

MS. HART: That's correct. And does include direct dose from the filters as well.

MEMBER BROWN: If you could give us reference to that RAI, not now. Tomorrow is fine.

MS. HART: Sure.

MEMBER BROWN: I presume you'll be back for Chapter 15.

MS. HART: You may be disappointed in my Chapter 15 presentation. I will just let you know that right now.

MEMBER BROWN: I understand why. Well, tomorrow would be fine.

MS. HART: I'm just saying. You may not have as much information as you would hope for.

MEMBER CORRADINI: But I think if we could get the RAI, then I think that's what Steve's --

MS. HART: Yes. Perhaps by the end --

DR. SCHULTZ: Yes. I would like to see that comparison. It would be interesting.

MS. HART: Right.

DR. SCHULTZ: There are a lot of different assumptions that are in tomorrow's evaluation. There's a couple --

MS. HART: Correct.

DR. SCHULTZ: As you say things have changed in the analysis, input --

MS. HART: Right.

DR. SCHULTZ: -- assumptions have changed.

MS. HART: And I myself have done a lot of sensitivity analyses to try to push the boundaries and see -- and, of course, you can push it to where it is much greater than 5 rem if you come up with the correct reasoning. If you make some other assumption changes, especially in the core damage event --

DR. SCHULTZ: Okay.

MS. HART: -- for any accident other or any analysis other than the core damage event. They're not going to approach the control room dose criterion.

DR. SCHULTZ: Thank you.

MEMBER CORRADINI: Thank you.

MEMBER BROWN: Are there requirements for the core damage event?

MS. HART: I'm sorry. Say that again.

MEMBER BROWN: You said they could be higher than 5 if you had a core damage event.

MS. HART: No, if there were changes to the dose analysis, if there were -- because we are still evaluating the control room -- I mean, excuse me, the core damage event, it's not a final.

There were some questions that we had about containment leak rate. There were some questions that we had about the particular assumptions of the release rate to the containment, if those would have to be changed because it's 2.14 rem and with those criterion it's 5 rem. There's not a huge amount of margin. So there's a potential that they could increase to be over the top of that.

MEMBER BROWN: Over the 5.

MS. HART: Correct. But right now, I don't think that that's the case. And their dose analyses and my confirmatory analyses but the current dose analyses do not show that that is the case.

MEMBER CORRADINI: Okay. Further questions by the members because I want to go to public comments and then we want to have enough time for our extended closed session. Okay. Any more questions by the members? All right.

Why don't I ask then if there are members of the public that want to make a comment in the room. And then we'll open the phone line. Nobody in the room? The phone line is open. Could we ask anybody on the open phone line for the public to please at least acknowledge that you're out there.

MS. FIELDS: This is Sarah Fields so.

MEMBER CORRADINI: Okay. Thank you very much.

MS. FIELDS: There are listeners out there.

MEMBER CORRADINI: Good. Go ahead with your comment, please, Ms. Fields.

MS. FIELDS: A couple of comments. The first are concerns. According to the NRC schedule for the review -- there's a lot of background noise here. I don't know what that is.

MEMBER CORRADINI: We can barely hear you. You're going to have to speak -- we can barely hear you.

MS. FIELDS: There was some loud background noise. Can you hear me better now?

MEMBER CORRADINI: I think if you would get closer to the microphone. If you have a receiver, pick up the receiver so we can hear you a little better, please?

MS. FIELDS: Is that better?

MEMBER CORRADINI: Yes.

MS. FIELDS: Okay. I'm talking into my mobile phone. I'm concerned that it appears that the NRC staff intends to commence the design certification rulemaking next December.

It's not been officially put on the website part of the schedule. But it was commenced before the ACRS review of the advanced SCR with no open items and before the final SCR with no open items.

So I'm concerned that rulemaking will commence before making a public comment before the ACRS completes its review and before the NRC completes its final SCR approval. So that's my concern.

MEMBER CORRADINI: Okay.

MS. FIELDS: I know you're not going to --

MEMBER CORRADINI: May I repeat your comment so I understand it? You're concerned about the time scale of the final rule versus ACRS review. Is that what you're saying?

MS. FIELDS: Right.

MEMBER CORRADINI: Okay. Fine. I have your comment. Thank you.

MS. FIELDS: And then another comment is

it's pretty clear there's some internally important open items that have yet to be resolved. And also there are a number of extensions left and assuming that the NRC assumes most of those extensions that I wonder what the NRC is going to do if after a period of operational history of this design, it has no operational history.

So if after a period of operational history, the operators in the NRC find that some of these assumptions in these extension requests don't hold true during operations and there may be other things that don't hold true for this design.

And I think the NRC should make clear to the public what will happen if and when one or more aspects of this design and one or more aspects of the bases for the exemption requests don't pan out in reality.

Those are my comments. Thank you very much.

MEMBER CORRADINI: Thank you. Is there anyone else on the line that wants to make a comment? Okay. Hearing none, why don't we close the public line, please? And at this point, we're going to go into closed session. So I'm going to thank the staff --

MR. TABATABAI: Before we go into the closed session, I just want to -- I think Michelle has an update for an RAI number that Mr. Schultz asked.

MEMBER CORRADINI: Okay.

MR. TABATABAI: And also I just want to make sure that you're aware that the staff, for the purpose of the Chapter 6 presentation, we are not going to present any slides during the closed session. We don't have any proprietary slides to show. But we'll be in the audience to answer any questions.

MEMBER CORRADINI: That's fine. So what's the information about the open item that you wanted to give us?

MS. HART: Okay. So you were asking about the RAI where I ask about the sensitivity analysis for the dose analysis. And it was RAI 9534 and it was specifically Question 06.04-4. And there were several responses to that.

MEMBER CORRADINI: What's the question again, please?

MS. HART: It was a question about the sensitivity analysis if you --

MEMBER CORRADINI: But the number again?

MS. HART: I'm sorry, 06.04-4 and that was in RAI 9534.

MEMBER CORRADINI: Okay. Thank you.

MS. HART: Yes.

MEMBER CORRADINI: All right. So with that I'm going to ask the staff to please -- and NuScale to make sure the room is closed so that we have only the appropriate people in the room. And we'll take a few minutes to kind of do that. And then can we close the public line? And then the -- I assume NuScale's private line to its subject matter experts will remain open.

(Whereupon, the above-entitled matter went off the record at 4:15 p.m.)

LO-0619-65896

June 11, 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 NuScale Subcommittee Presentation: FSAR Chapter 3, Design of Structures, Systems, Components and Equipment,” PM-0619-65894, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) NuScale Subcommittee Meeting on June 18, 2019. The materials support NuScale’s presentation of Chapter 3, “Design of Structures, Systems, Components and Equipment,” of the NuScale Design Certification Application.

The enclosure to this letter is the nonproprietary version of the presentation titled “ACRS NuScale Subcommittee Presentation: FSAR Chapter 3, Design of Structures, Systems, Components and Equipment,” PM-0619-65894, Revision 0.

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

Sincerely,

Zackary W. Rad
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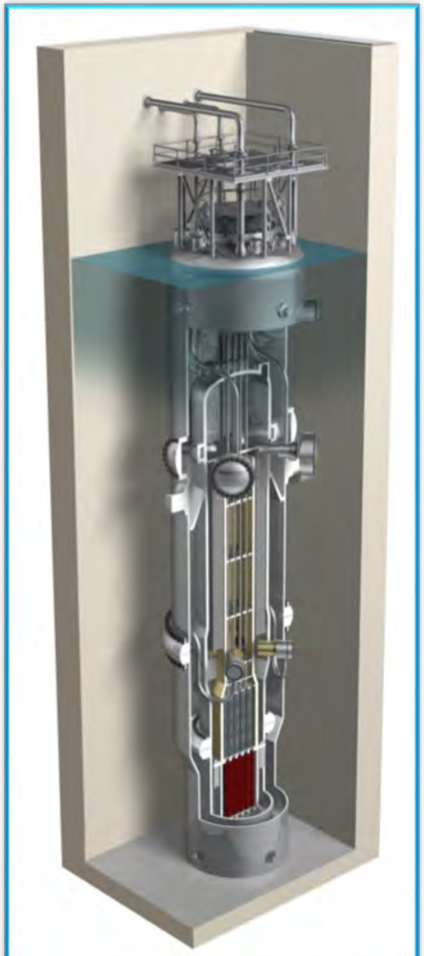
Enclosure: “ACRS NuScale Subcommittee Presentation: FSAR Chapter 3, Design of Structures, Systems, Components and Equipment,” PM-0619-65894, Revision 0

Enclosure:

"ACRS NuScale Subcommittee Presentation: FSAR Chapter 3, Design of Structures, Systems, Components and Equipment," PM-0619-65894, Revision 0

ACRS NuScale Subcommittee Presentation



NuScale FSAR

Chapter 3

Design of Structures, Systems, Components and Equipment

June 18, 2019

PM-0619-65896
Revision: 0

NUSCALE
Power for all humankind

Presenter

Marty Bryan

Licensing Project Manager

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Programs Engineer, Section 3.2, 3.10, 3.11

J. J. Arthur

Manager, Structures and Design Analysis, Sections 3.9, 3.12, 3.13

Josh Parker

Supervisor, Civil/Structural Analysis, Sections 3.3, 3.4, 3.5, 3.7, 3.8

Storm Kauffman

Consultant, Section 3.6

Purpose

Provide an overview of FSAR Chapter 3 to the ACRS NuScale Subcommittee

3.2 – Classification of Systems, Structures, and Components

- SSC classified according to
  - Seismic category (RG 1.29 R5)
  - Quality group (RG 1.26 R4)
  - Radwaste classification (RG 1.143 R2)
- SSC classified as
  - A1 is safety-related and risk significant
  - A2 is safety-related and not risk-significant
  - B1 is nonsafety-related and risk-significant
  - B2 is nonsafety-related and not risk-significant
- COL Item requires applicant to identify classification of site-specific SSC

3.10 – Seismic and Dynamic Qualifications of Mechanical and Electrical Equipment

- Addresses seismic and dynamic qualifications of SC-I mechanical and electrical equipment and supports

- Methods and procedures meet RG 1.100 R3 and IEEE 344-2004

- Three COL Items to develop site-specific seismic and dynamic qualification program and equipment qualification database

3.11 – Environmental Qualification of Mechanical and Electrical Equipment

- EQ program complies with DSRS 3.11.

- EQ program includes equipment in 10 CFR 50.49 scope, certain PAM equipment specified in RG 1.97, and active mechanical equipment with safety-related function
  - Meets GDC 1, 2, 4, and 23
  - Table 3.11-1: harsh equipment list subject to EQ
  - Environmental conditions considered include AOOs and normal, accident, and post-accident (See Appendix 3C)

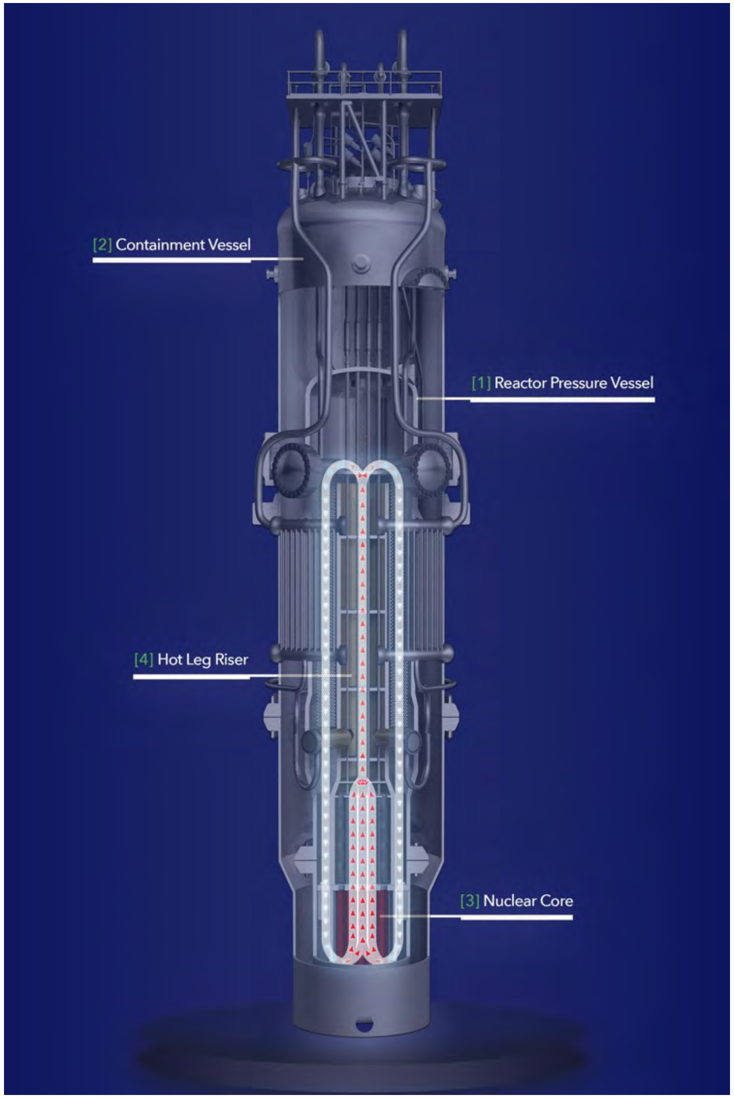
- Four COL Items to provide the site-specific EQ program

3.9.1 – Mechanical Systems and Components

- Addresses analysis methods for SC-I components and supports
- Operating condition categories ASME Service Level A through D and Test Conditions apply to Class 1 & 2 components, CNV, supports, RVI, piping and valves inside and outside containment
- Dynamic and static analyses used ANSYS, AutoPIPE, NRELAP5, RspMatch2009, SAP2000, SASSI2010, SHAKE2000, EMDAC, and Simulink
- NuScale technical reports for NPM Seismic Analysis, CVAP, and Short-Term Transient Analysis reviewed in May 16, 2019 ACRS meeting

3.9.3 - ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

- ASME Code Class 1, 2, and 3 Components are designed and constructed in accordance with Section III the 2013 edition of the ASME Boiler & Pressure Vessel Code.
- Section 3.9.3 defines loading combinations, system operating transients, and stress limits for these components.



3.9.4 – Control Rod Drive System

- The control rod drive system (CRDS) is composed of:
  - a pressure-retaining housing, a control rod drive shaft with a coupling for attaching to the control rod assembly (CRA) hub,
  - external electromagnetic coils with cooling loop heat exchangers
  - The RCPB pressure boundary parts of the CRDS applicable requirements of ASME BPVC, 2013 Edition, Section III Subsection NB
- The control rod drive mechanisms (CRDMs) provide:
  - Means for CRA insertion and rod position indication to the module control system
- Unique CRDS features subject to prototype testing described by Section 4.2.4

3.9.5 – Reactor Vessel Internals

- Reactor Vessel Internals (RVI) composed of subassemblies which:
  - Support and align the reactor core system including the CRAs, and include the guide tubes
- RVI channels the reactor coolant from the reactor core to the steam generator (SG) and back to the reactor core
- RVI core support structures and internal structures comply with ASME BPVC Section III, Division 1, Subsection NG.

3.9.6 – Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints

- ASME OM-2012 Code Edition was used to develop inservice testing requirements.
- Pursuant to 10 CFR 50.55a(z), an Alternate Authorization was used to apply Mandatory Appendix IV (AOV Testing) of the ASME OM-2017 Code Edition for *performance assessment testing* of power operated valves.
- The NuScale design includes no safety-related pumps, MOVs or dynamic restraints (snubbers).
- The IST program contains 39 valves per NPM. This includes 26 HOVs (22 are CIVs), 5 ECCS valves, 2 AOVs, 4 PRVs, and 2 CKVs. [FSAR Table 3.9-16]
- The Augmented Valve Testing Program (Augmented IST) contains 12 valves per NPM. These valves do not meet the criteria of ISTA-1100, but do have an augmented quality function. [FSAR Table 3.9-17]

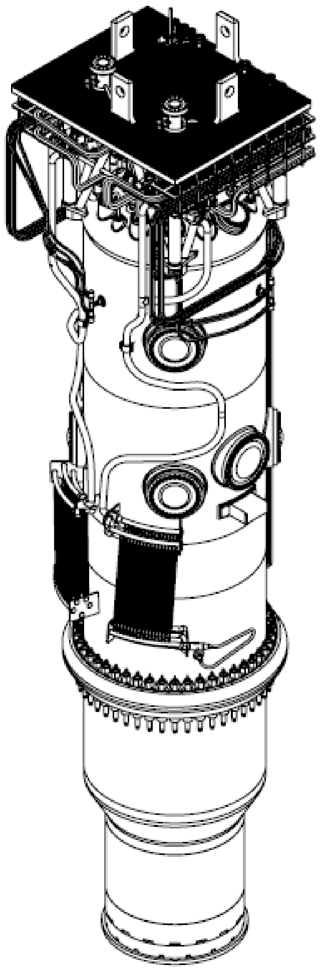
COL Items in Section 3.9

- 12 COL Items to assure compatibility of design with site-specific conditions and equipment, describe programs, and provide test procedures

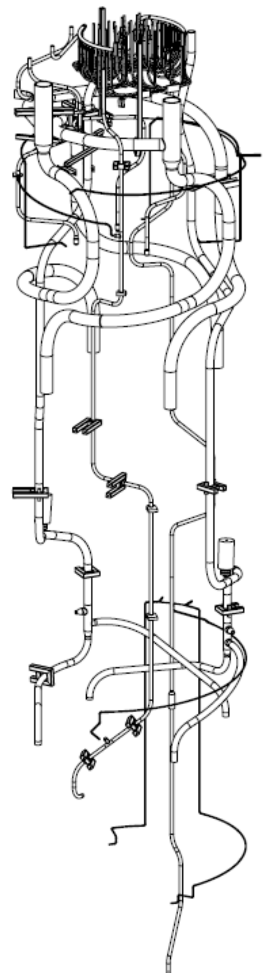
3.12 – ASME Code Class 1, 2, 3 Piping Systems, Piping Components and Associated Supports

- The NuScale design includes less ASME Section III piping than traditional LWRs.
  - The NPM does not include large reactor coolant piping. ASME Class 1 piping systems are specified as NPS 2 or smaller.
  - Maximum pipe size in the NPM is NPS 12 (Main Steam Lines)
  - Snubbers are not used within the NPM.
  - Stress analysis has been performed for high-energy lines (>NPS 1) within the NPM to support Section 3.6 (i.e., postulate HELBs)
  - NPM piping screened for thermal stratification and thermal oscillations using criteria developed by EPRI.
    - The DHRS condensate branch connection to the feedwater line was identified as being potentially susceptible. CFD analysis was performed to demonstrate that stratification does not occur and that the temperature fluctuations in the DHRS line and the associated containment penetration cause thermal stresses that are below the endurance limit for the materials of the piping, welds, and CNV

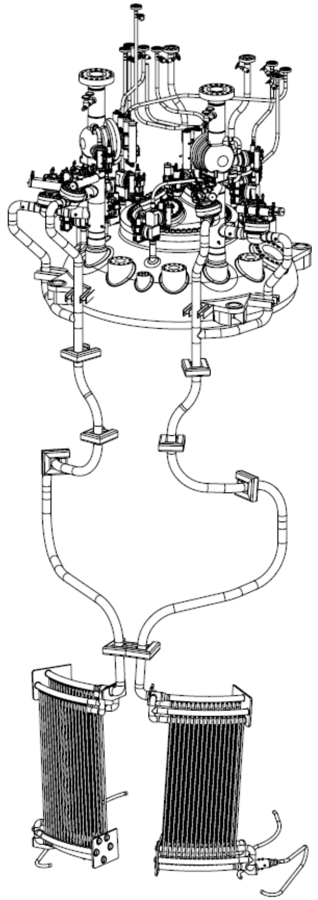
3.12 – ASME Code Class 1, 2, 3 Piping Systems, Piping Components and Associated Supports (cont'd)



NPM



NPM Piping
Inside the CNV

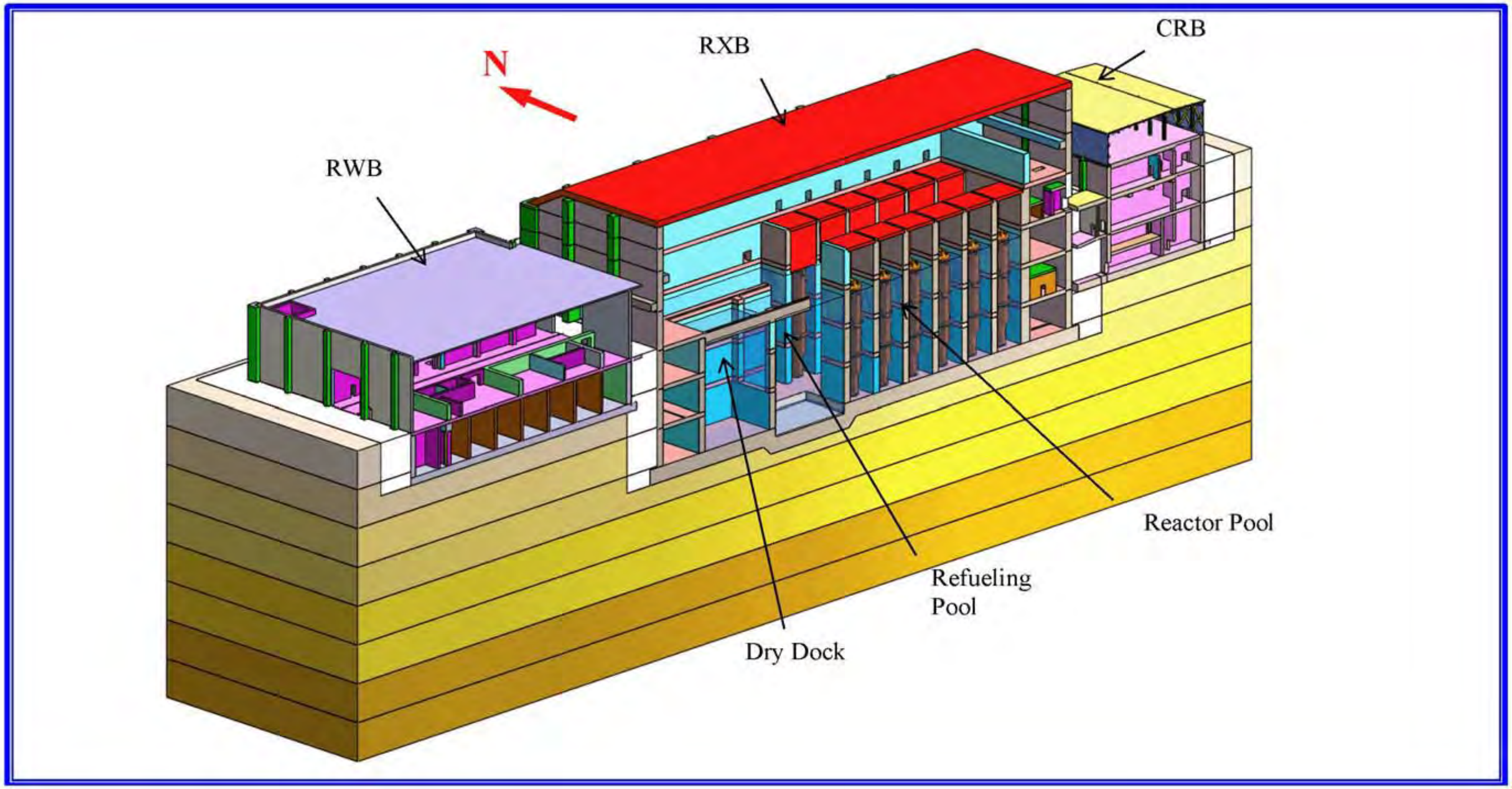


NPM Piping
Outside the CNV

3.13 – Threaded Fasteners (ASME Code Class 1, 2, and 3)

- Meet 10 CFR 50.55a
  - No code cases per RG 1.84 R36 used
- Meet GDC 1, 4, 14, 30, and 31
- ISI meets 10 CFR 50.55a
- COL Item to provide ISI program for threaded fasteners

NuScale Plant Structures



RWB is SC-II and RW-IIa

RXB and CRB are SC-I

3.3 – Wind and Tornado Loadings

- Wind Loads
  - Design basis severe wind, 3-second gust = 145 mph (ASCE/SEI 7-05)
    - Importance factor of 1.15 for RXB, CRB, RWB
    - Exposure Category C
- Tornado and Hurricane Loads
  - Design basis tornado: 230 mph (RG 1.76 R1)
  - Design basis hurricane: 260 mph (RG 1.221 R0)
- Design complies with GDC 2 and 4
- COL Item confirms site specific structures will not collapse and adversely affect RXB or SC-I portion of CRB

3.4 – Water level (Flood) Design

- Flooding analysis is conducted on a level-by-level and room-by-room basis for the RXB and CRB for postulated flooding events
- The RXB and CRB flooding analysis consists of the following steps:
  - Identification of potential flooding sources
  - Identification of rooms/areas that contain equipment subject to flood protection
  - Estimation of flood depth in the identified rooms/areas
  - Determination of the need for protection and mitigation measures for rooms

3.4 – Water Level (Flood) Design (cont'd)

- Protection from external sources
  - Probable Maximum Flood
    - 1 foot below grade elevation
    - Max groundwater elevation: 2 feet below grade elevation
    - There are no dynamic flood loads on the structures given the below grade flood elevation
    - The ground water and soil pressure are included in the static and dynamic loads
  - Precipitation
    - Rainfall rate: 19.4 in/hr and 6.3 inches for 5 minutes
    - Roof snow load: 50 psf
    - Extreme snow load: 75 psf
    - Bounding rain and snow loads are considered in the structural analysis

3.4 – Water Level (Flood) Design (cont'd)

- Design satisfies GDC 4 and GDC 2
- Interaction of non-SC-I structures with SC-I structures assessed/analyzed to ensure no credible potential
- RXB and CRB flooding analyses protection of safety-related equipment
- Seven COL Items confirm site-specific conditions, programs, locations, and no adverse impact to RXB and SC-I portion of CRB

3.5 – Missile Protection

- Design meets GDC 2 and 4
- SC-II RWB also classified RW-IIa (RG 1.143 R2) and designed for same missiles as SC-I structures
- Potential missiles considered
  - Internally-generated missiles
  - Turbine missiles
  - Tornado/extreme winds missiles
  - Site proximity missiles
  - Aircraft hazards

3.5 - Missile Protection (cont'd)

- No credible missiles inside containment
- Tornado/wind missiles from RG 1.76 R1 and RG 1.221 R0
- Safety-related and risk-significant SSC are located inside SC-I RXB and SC-I portions of CRB
- Four COL Items to confirm site-specific missile analyses

3.5.1.3 Turbine Missile Protection

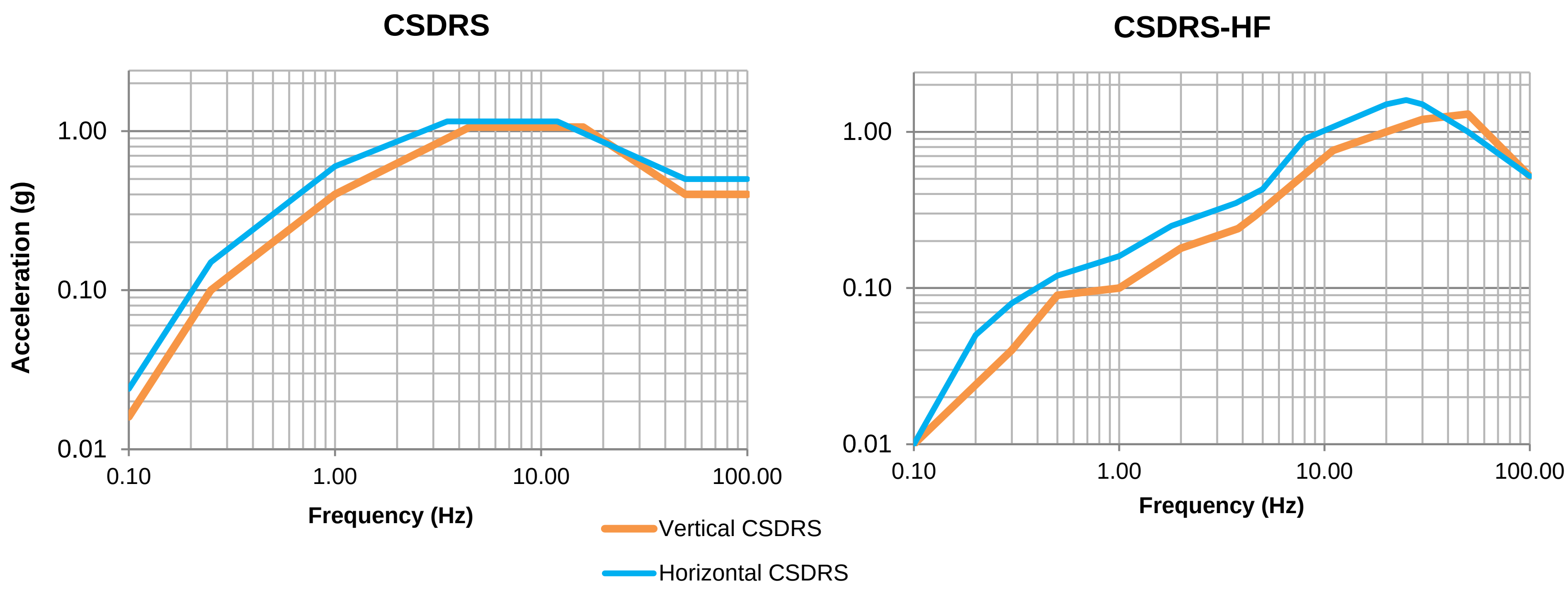
- NuScale plant designed per RG 1.115 using barrier approach
- Three different missiles analyzed
  - Turbine blade, 32.6 pound
  - Turbine blade with rotor fragment, 52.6 pound
  - Half of last stage turbine rotor disk, ~3,000 pound
- Velocity of missiles based on 3600 revolution per minute turbine speed
  - Varying overspeed conditions considered- up to 160% in accordance with RG 1.115

3.5.1.3 Turbine Missile Protection (cont'd)

- Barriers designed per SRP 3.5.3
  - Local missile effects
    - Penetration
    - Scabbing
    - Perforation
  - Global missile effects
- Reactor Building and Control Building structures analyzed as turbine missile barriers;
  - Provide adequate protecting for essential SSCs housed within buildings

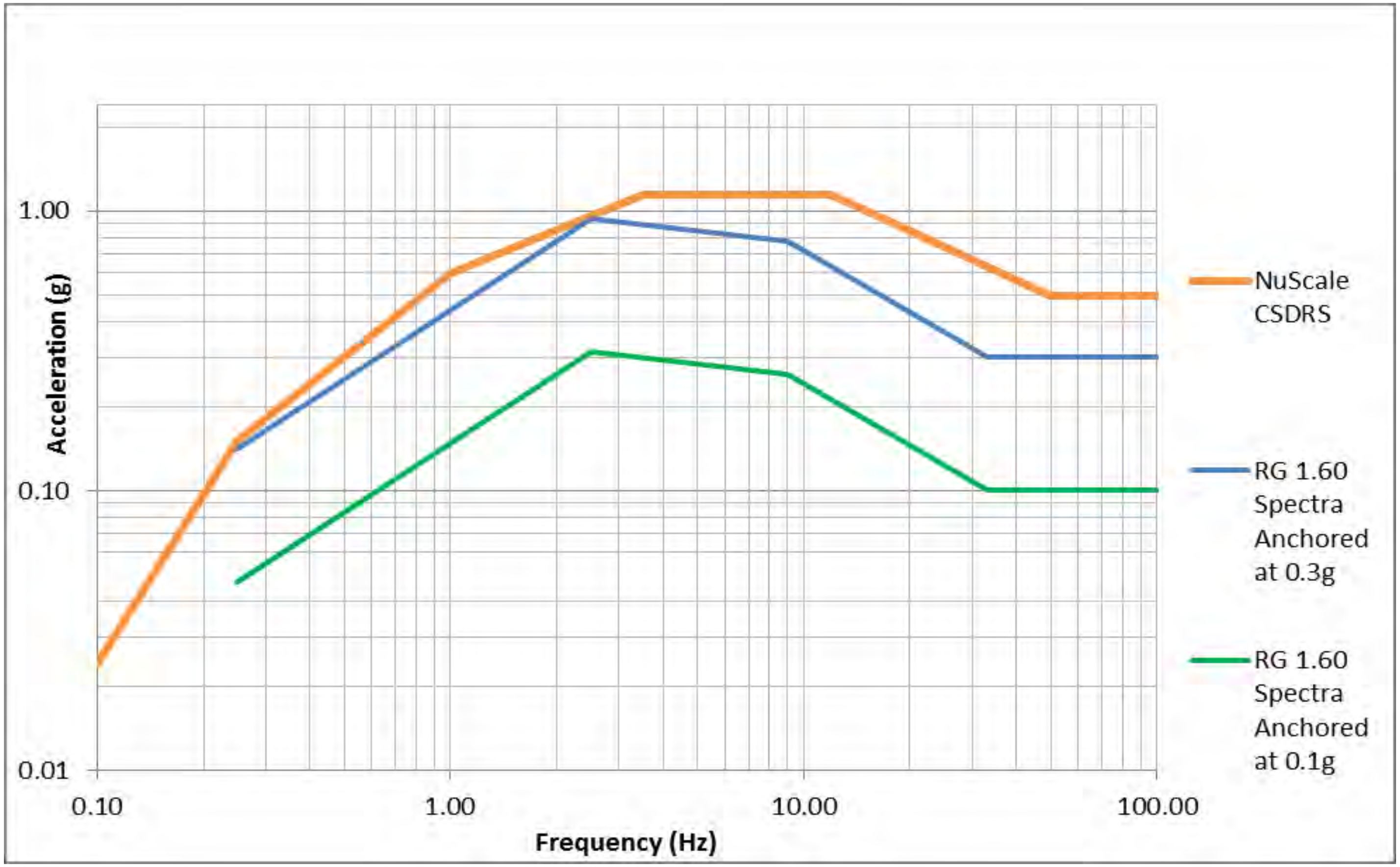
3.7 – Seismic Design

- Design meets GDC 2 and 10 CFR 50, Appendix S
- CSDRS is design basis for SC-I SSC
  - SC-I structures also evaluated using CSDRS-HF with more high frequency content than CSDRS



3.7 – Seismic Design (cont'd)

- NuScale spectra higher and broader than previous design certifications

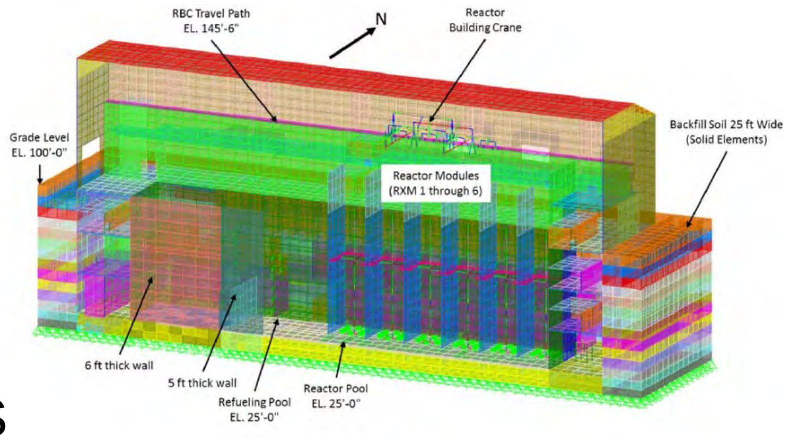


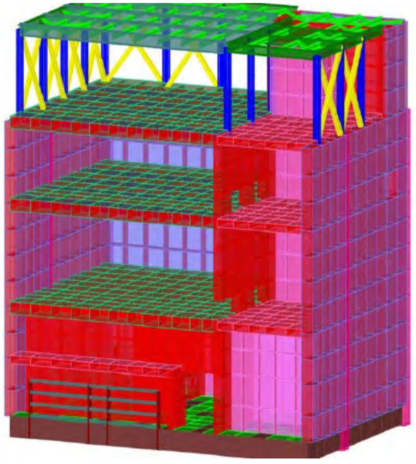
3.7 – Seismic Design (cont'd)

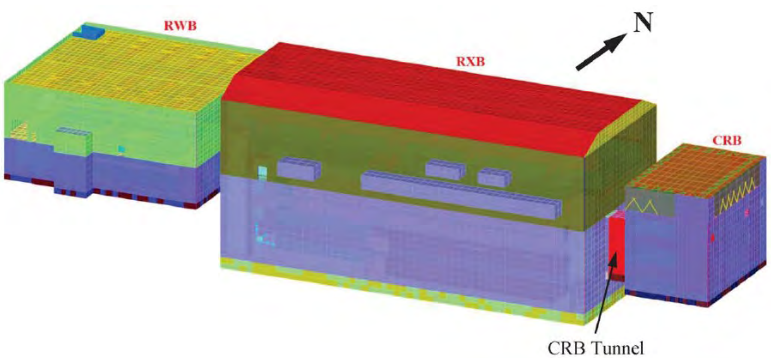
- Time Histories
  - Five seed time histories compatible with the CSDRS
  - Separate seed time history compatible with the CSDRS-HF
- Soil Profiles
  - Four soil profiles - Soft Soil [Type 11], Firm Soil/Soft Rock [Type 8], Rock [Type 7], Hard Rock [Type 9]
- SC-I and SC-II SSC analyses use RG 1.61 R1 damping values
- Only two SC-I structures: RXB and CRB
  - RWB is SC-II and also RW-IIa (high hazard) (RG 1.143 R2)
  - Buildings Analyzed using SAP2000, SASSI2010, and ANSYS

3.7 – Seismic Design (cont’d)

- SAP2000
  - Used to develop the RXB and CRB finite element models for static analyses
  - Consider both uncracked and fully cracked material properties
  - NPM submodel is input based on ANSYS model
  - Single building models and triple building models developed







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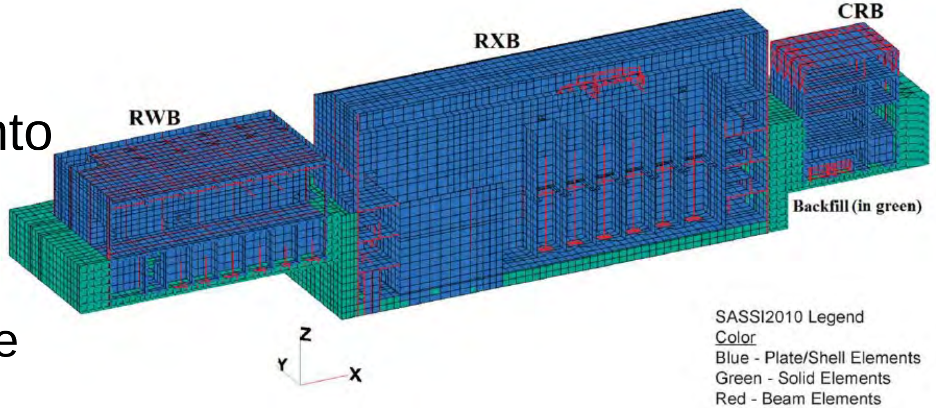
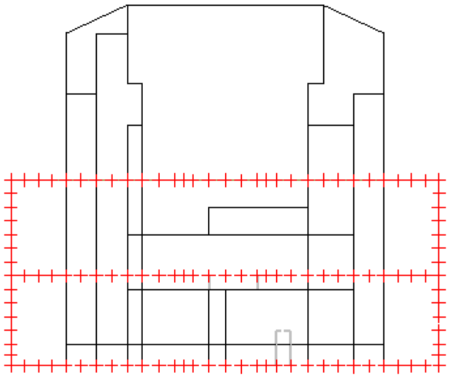
Revision: 0

Template #: 0000-21727-F01 R5

3.7 – Seismic Design (cont'd)

- SASSI2010
  - The SAP2000 model is converted into SASSI model
    - The models have the same node numbers, node coordinates, and finite elements in terms of types, numbers, sectional, and material properties
  - Models use the extended subtraction method to perform soil structure interaction analysis
    - Interaction nodes are nodes on the 'seven planes' (i.e., six sides [east, west, north, south, top, and bottom]) and the middle plane, of the excavated soil model





7P Interaction Nodes (in Red)

Cross-Section through RXB

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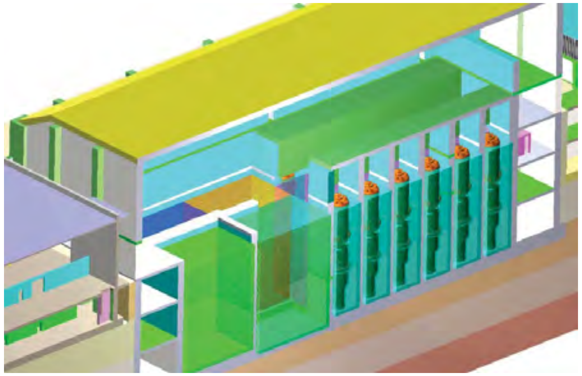
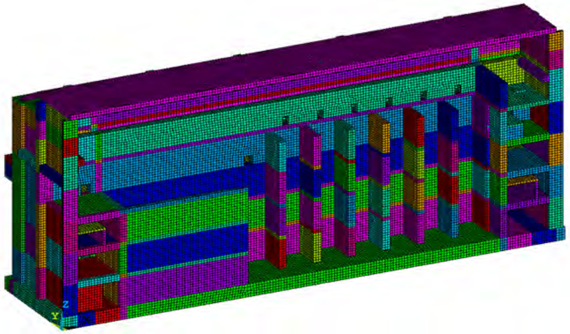
3.7 – Seismic Design (cont'd)

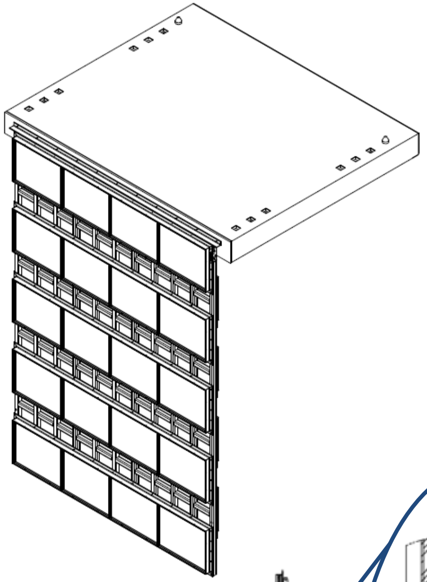
- ANSYS
  - – Necessary since neither SAP2000 nor SASSI have explicit fluid element formulations
    - Used to determine the hydrodynamic pressures on the reactor pool walls and foundation from a Fluid-Structure Interaction analysis
    - Used to determine the sloshing wave height
  - – Used for RXB and CRB stability analyses
  - – Used to develop thermal and stress analysis models of the RXB
    - Obtained bounding rebar and pool liner strains from load combinations that involve thermal and HELB loads

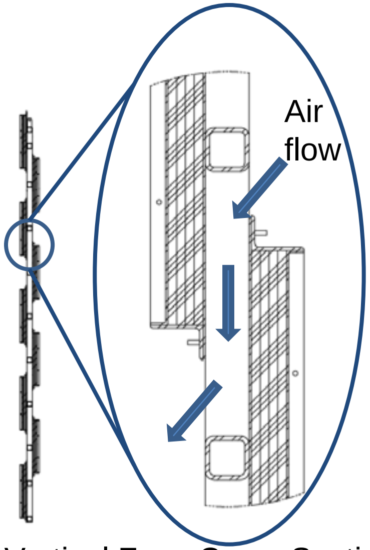




3.7 – Seismic Design (cont'd)

- Subsystem analysis
  - Four subsystems evaluated in the DCA
    - NPM- Described in the NPM Seismic Technical Report
    - RBC- Described in Section 9.1.5
    - Fuel Racks- Described in Section 9.1.2
    - Bioshield
  - Bioshield
    - a nonsafety related, not risk significant SC-II component
    - Major functions include fire protection, radiation protection, ventilation, and support personnel access





Bioshield Vertical Face Cross Section

3.7 – Seismic Design (cont'd)

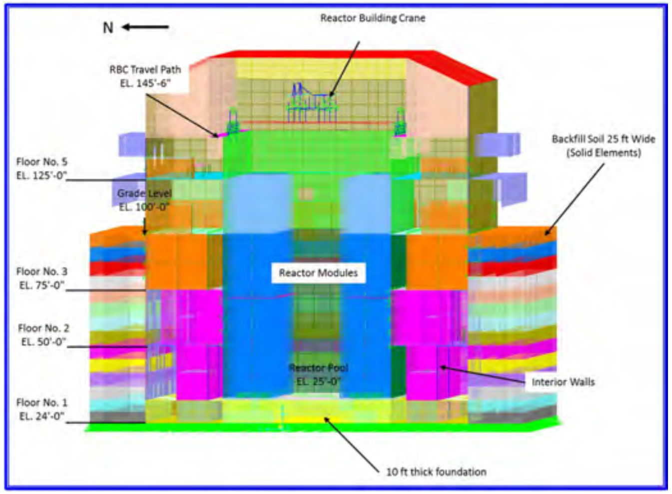
- Sensitivity studies
  - Multiple sensitivity studies were evaluated such as:
    - Empty dry dock
    - Less than 12 modules
    - Soil separation
    - Extended subtraction method to direct method
    - Non-vertically propagating shear waves
  - Results were shown to be bounded or design basis was modified
    - Compared in-structure response spectra and design forces and moments
- Fifteen COL Items to confirm that the site-specific design is bounded by the DCA evaluation

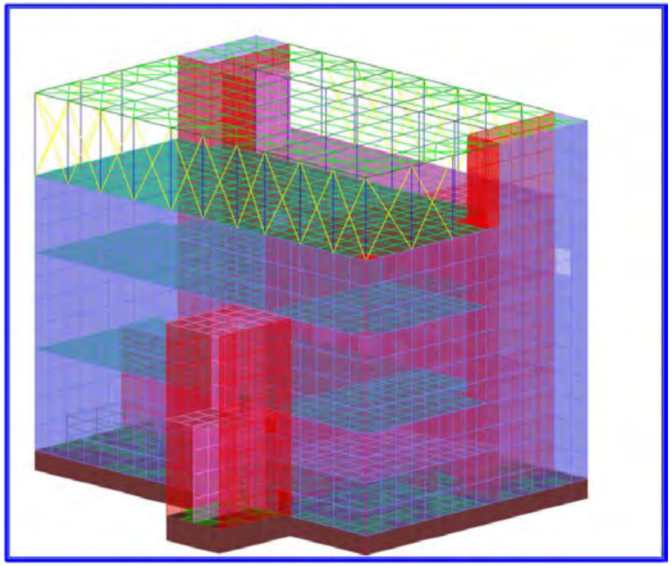
3.8.2 – Steel Containment

- The NuScale containment is a steel pressure vessel designed to the requirements of ASME III, Division 1, Subsection NB.
  - The CNV has an upper and lower section connected with an approximate 218-inch diameter bolted flange.
  - The CNV is housed in the reactor pool within the RXB
  - The internal design pressure of 1,050 psia bounds all service level pressures except for hydrostatic test conditions. The external design pressure is 60 psia. The design temperature is 550 deg-F.
  - An analysis conforming to the guidance provided in Appendix A of NUREG/CR-6906 was performed to determine the ultimate pressure capacity of the CNV (1240 psia)
    - Documented by Technical report TR-0917-56119, CNV Ultimate Pressure Integrity

3.8.4 – Design of Category I Structures

- RXB and CRB meet ACI 349 and AISC N690 requirements
- Designed for normal loads, severe environmental loads, extreme environmental loads, and abnormal loads (e.g., high energy pipe break)
- Six COL Items to confirm acceptability of RXB and CRB, describe program for monitoring and maintaining SC-I structures, and evaluate construction elements



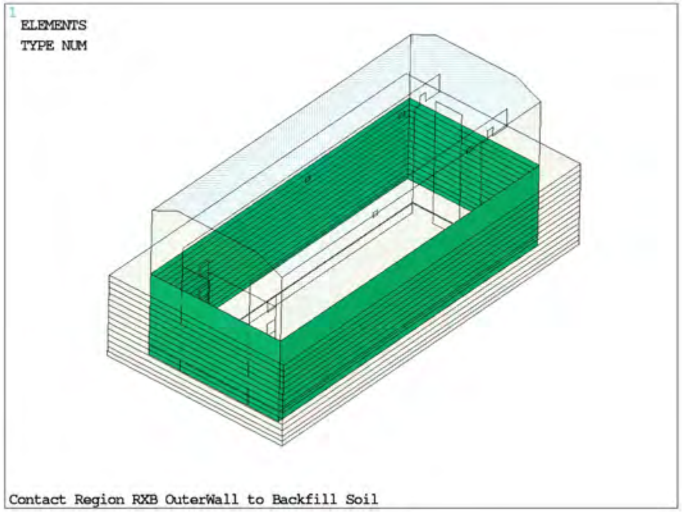


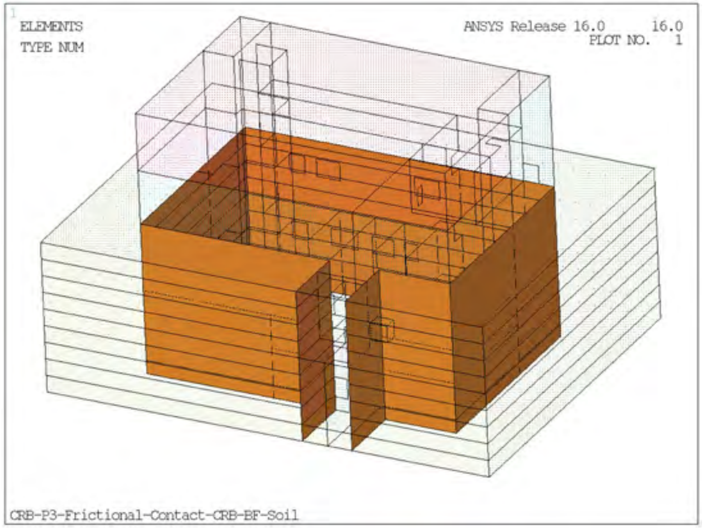
3.8.4 – Other Seismic Category I Structures (cont'd)

- FSAR Appendix 3B summarizes RXB and CRB structural design and analyses for Critical Sections
- 15 Critical Sections for the RXB, and 7 for the CRB
  - Perform a safety-critical function
  - Are subjected to large stress demands
  - Is a feature that is difficult to design or construct
  - Are considered to be representative of the structural design
  - Selection of walls, slabs, beams, buttresses, pilasters, NPM bay wall, and NPM supports
- Design Evaluation of Critical Sections demonstrate the acceptability of the design
  - Compare the demand on the section to available capacity

3.8.5 - Foundations

- Stability (i.e., sliding and overturning) is checked per SRP 3.8.5
- Nonlinear sliding analyses were performed for both RXB and CRB.
- Nonlinear overturning also performed for CRB.
- Results demonstrate that the effect of sliding and uplift is small and acceptable for deeply embedded structures





3.6 – Protection Against Pipe Rupture Effects

Key NuScale Differentiators

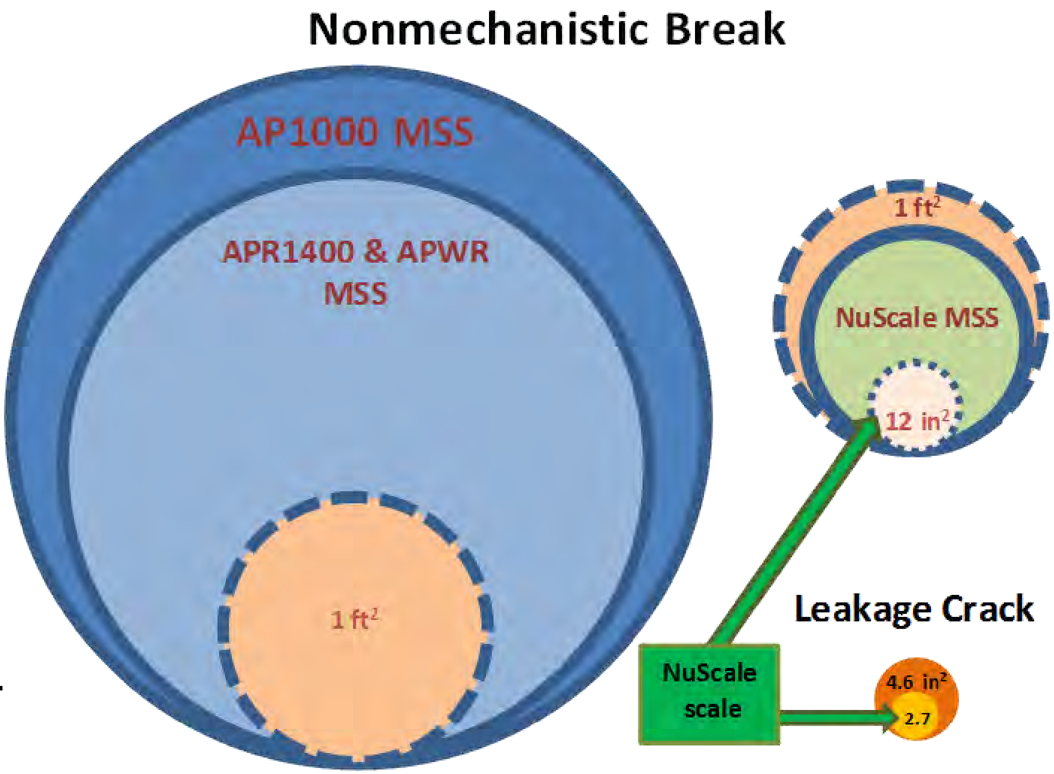
- Piping small diameter and short compared to typical LWR
- HELBs inside CNV limited to NPS 2 piping
- HELB/MELB response is passive
- Containment immersed in pool of water
- CNV designed/fabricated to ASME Class 1 (not a building)
- Containment at a vacuum
- No containment spray or emergency make-up
- Piping inside containment is not insulated
- Insulation outside CNV cannot clog ECCS
- MSS & FWS piping inside containment satisfies LBB
- MSS & FWS piping in containment penetration area is RCS-like
- Essential SSCs inside CNVs qualified to 1050 psia saturated steam
- Congested arrangements
- CIVs and DHRS actuation valves outside CNVs
- NPM moved during refueling / disconnect flanges
- Main control room in building from high energy piping
- Up to 12 NPMs operating in proximity

Locations of Possible Line Breaks

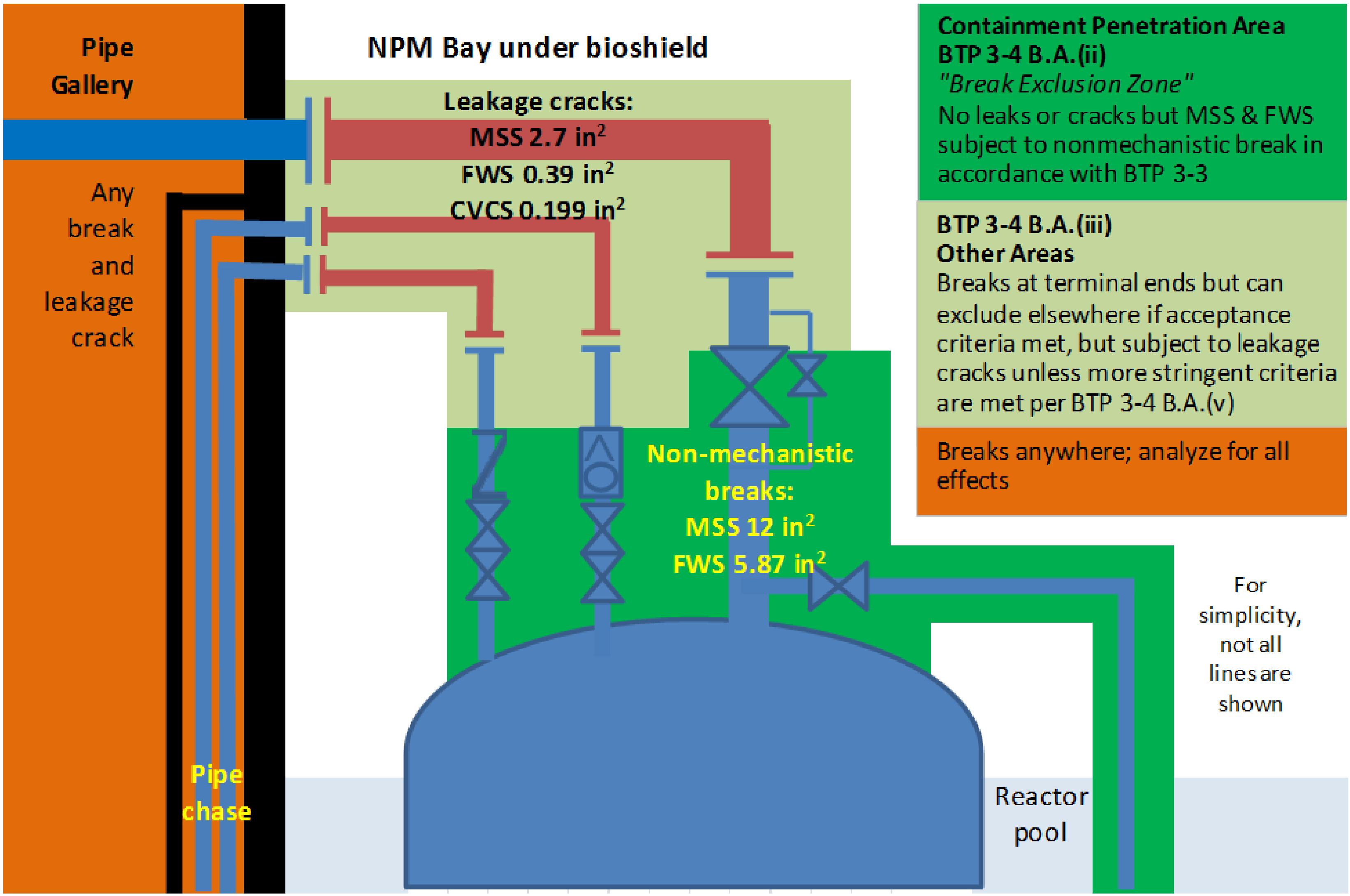
- Analyses divided into 3 regions because of different:
  - Environment
  - Systems that are high or moderate energy
  - Potential target SSCs
- The regions are
  - Inside CNV: specific locations and arrangements
    - Main steam (MSS) and feedwater (FWS) piping designed to satisfy LBB
    - Small amount of other piping, all small bore – no containment penetration area
      - Evaluate effects of breaks at terminal ends
    - Follow BTP 3-4 B.A.(iii) guidance for determining break locations
  - In NPM bay under bioshield: specific locations and arrangements
    - Containment penetration area out to weld between last valve & pipe
      - Non-mechanistic break of MSS or FWS
  - In RXB where piping present: generic break and target locations
    - Bound possible breaks
    - Future finalization of pipe routing will not affect conclusions

Non-Mechanistic Breaks

- MSS and FWS piping generally
  - Largest, high-energy piping near the containment boundary
  - Single CIV outside containment (GDC 57)
  - Piping in other plants usually made of less-corrosion-resistant material (carbon or low-alloy steel), which have greater susceptibility to degradation than stainless steel



Break Exclusion



Blast Wave Summary

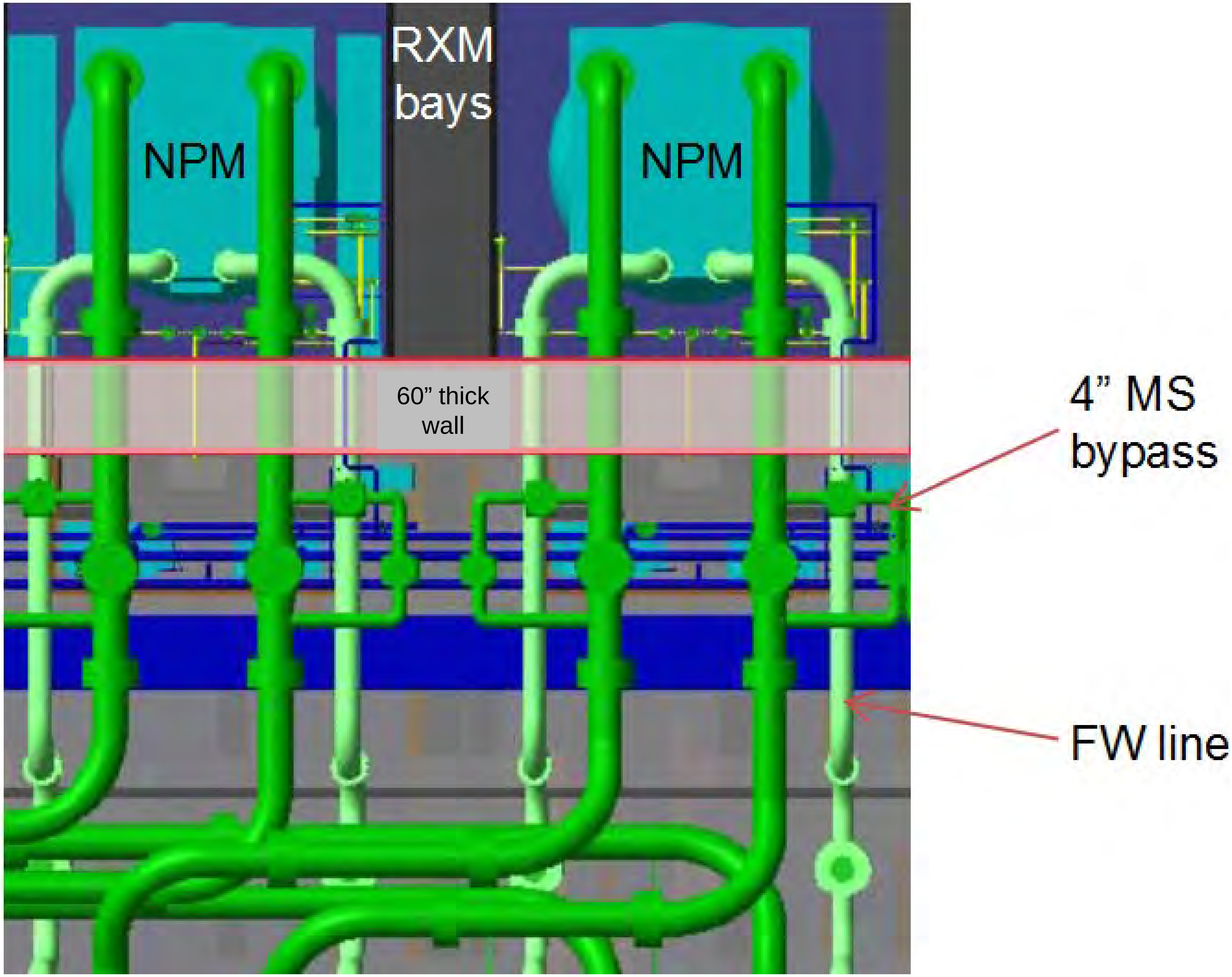
- Assumed instantaneous break opening
  - Only steam filled lines generate a blast
  - No blast if break opening time more than a few milliseconds
  - Small piping reduces mass & energy available
- Accelerated material forms region of high density
  - Higher density wave speed exceeds ambient speed of sound
- 3D computational fluid dynamics using CFX V18.0 code
  - Verification and validation: 8 test problems
  - Simplified plant geometry
  - 3D needed to capture geometry of wave propagation / reflection
  - Force-time history for
    - 3 degasification line cases in CNV
    - 3 MSS line cases in RXB
- Blast forces very brief and acceptable

Pipe Whip

- Impact energy depends on fluid conditions & length
  - Small pipes have low energy content
  - Congested areas limit length of whip
- Determine
  - If fluid energy sufficient to form plastic hinge (i.e., thrust force)
  - Motion of whip in plane based on pipe geometry
  - If essential SSCs are too far away to be struck
  - If nearby SSC serves as barrier
  - Impact force if pipe whip impact occurs

Pipe Whip-caused Secondary HELB

SRP 3.6.2 – an unrestrained, whipping pipe need not be assumed to cause ruptures or through-wall cracks in pipes of equal or larger NPS with equal or greater wall thickness.



Pipe Whip Summary

- Impact force depends on
  - Thrust force at break exit (NuScale force ~5% of large PWRs)
  - Short length of moment arm to plastic hinge
  - Distance through which pipe whips
    - Know piping arrangements in CNV
    - Conservative assumptions in RXB
  - Energy absorbed by pipe on impact
- Two types of targets
  - Metallic components/pipes
  - Concrete in RXB: penetration (Young (Sandia) formula) & spalling
- Conclusions
  - Same or larger size pipe does not break or crack (SRP 3.6.2)
  - Targets in CNV except RPV, CNV, & CRDMs out of range
  - Concrete wall penetration minor & wall thickness prevents spalling

Jet Impingement

- Impingement pressure energy depends on fluid conditions, spreading of jet, & distance to target
  - Small pipes have low energy content
- Jet behavior phase dependent
  - Steam: limited spread
  - 2-phase: NUREG-2913
- Determine
  - Pressure threshold for SSC damage
  - Zone of influence (ZOI)
  - Jet spreading half-angle
  - If nearby SSC serves as barrier
  - If essential SSCs are close enough to be above pressure threshold

Jet Shape & Pressure Overview

- Inside CNV (NPS 2 only – no longitudinal breaks)
  - ZOI: forward facing hemisphere out to threshold
  - Pressure: half-angle of 30˚ to 5 L/D, then 10˚
    - Greater spreading in low pressure ambient
    - More conservative than Modified Moody Model 45˚ half-angle
- In RXB
  – ZOI: anywhere because piping arrangements not finalized
    - Assumes target SSC is 2 L/D from break exit
  – Pressure: full break exit pressure (no expansion)

Jet Dynamic Amplification

- SRP 3.6.2 identifies concern that jet impingement on target can induce resonant condition
- Numerous research papers show resonance requires
  - Axisymmetric jet
  - A “phase lock” in boundary shear layer at jet exit
- Occurs in gas jets but not seen in steam or 2-phase jets
- Resonance will not occur due to jet impingement
  - Condensable fluid in jet attenuates oscillations
  - Distorted exit geometry violates axisymmetry
  - Absence of a large, flat impingement surface sufficiently close and perpendicular to the jet axis eliminates return of acoustic energy
  - Instability of jet exit separation and angle prevents phase lock
  - Presence of obstacles or intersecting flow disrupt jet axisymmetry
  - Frequency mismatch with structures

Jet Impingement Summary

- Impingement pressure depends on
  - Thrust force at break exit; NuScale thrust force small
  - Spreading angle of underexpanded jet
  - Distance to target SSCs
- Two types of targets in CNV (no thermal insulation stripping)
  - Metallic components/pipes
    - Safety-related components (e.g., valves, CRDMs) go to safe position on loss of power or pressure holding them open
  - Cables
- No dynamic amplification
- Conclusions:
  - No safety-related SSCs within range in CNV
  - RXB structure satisfactory by analysis bounding future final design

Pressurization Summary

- GOTHIC analysis of subcompartment pressurization caused by HELBs in the different areas of plant
- In CNV: HELB transient bounded by ECCS initiation for which safety-related SSCs are qualified
- Under bioshield: HELBs limited to nonmechanistic breaks of MSS and FWS and to leakage cracks
  - Passive vent to pool room limits pressure and temperature
  - Envelope of results used to create EQ temperature envelope
  - Structural limit of 1 psid met
- In RXB: various HELBs limiting in different areas
  - Vent paths limit room pressures
  - Structural limit of 3 psid met

PRHA Conclusions

- Break Locations limited by break exclusion and LBB
- Blast – 3D CFD determined loading is low and brief
- Pipe Whip – SSCs out of range or can withstand impact
- Jet Behavior – components and cabling have good resistance to damage, and conservative modeling showed no adverse effects
- Subcompartment Pressurization – GOTHIC analysis used to determine need for venting, to set EQ envelope, and to show structural limits met
- COL items confirm final arrangements satisfy criteria

Analyses of the various external effects of HELBs confirmed acceptability of the NuScale design

Leak Before Break

- Applied to NPS 12 steam and NPS 4 and 5 feedwater lines inside CNV
  - SA-312 TP304/304L dual-certified stainless steel
  - Corrosion, erosion, fatigue, other failure mechanisms evaluated
  - Margin of 10 for leak rate and 2 for flaw size
  - Leak detection by change in
    1) CNV pressure
    2) CES sample vessel condensate level

Acronyms

COL	combined license
CNV	containment vessel
CRB	Control Building
CSDRS	certified seismic design response spectra
CSDRS-HF	CSDRS - high frequency
CVAP	Comprehensive Vibration Assessment Program
DBT	design basis tornado
ECCS	emergency core cooling system
EQ	environmental qualification
FIV	flow-induced vibration
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
HE	high energy
HELB	high-energy line break
ISI	in-service inspection
LOCA	loss-of-coolant accident
ME	moderate energy
NPM	NuScale Power Module
PAM	post-accident monitoring
PDC	principal design criteria
PMF	probable maximum flood
PMP	probable maximum precipitation
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	Regulatory Guide
RPV	reactor pressure vessel
RVI	reactor vessel internals
RWB	Radioactive Waste Building
RXB	Reactor Building
SC-I	Seismic Category I
SC-II	Seismic Catogory II
SSC	structures, systems, and components

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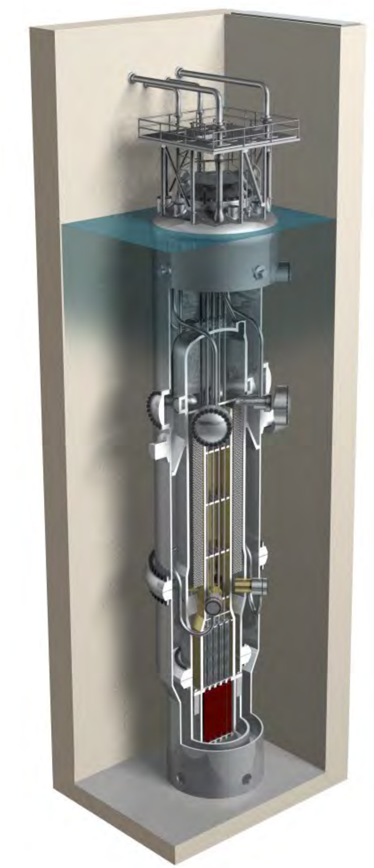

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Presentation to the ACRS Subcommittee

NuScale Design Certification Application Review

Safety Evaluation Report

Chapter 3: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Project Manager: Marieliz Vera

June 18, 2019

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  - **Gregory Cranston,** Lead Project Manager
  - **Marieliz Vera**, Chapter Project Manager

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Sections with no Open Items

- 3.2.1 – "Seismic Classification"
- 3.2.2 – "System Quality Group Classification"
- 3.3.1 – "Severe Wind Loads"
- 3.3.2 – "Extreme Wind Loads (Tornado and Hurricane Loads)"
- 3.4.1 – "Internal Flood Protection for Onsite Equipment Failure"
- 3.4.2 – "Analysis Procedures"
- 3.5.1.1 & 3.5.1.2 – "Internally Generated Missiles (Outside and Inside Containment)"
- 3.5.1.4 – "Missiles Generated by Tornadoes and Extreme Winds"
- 3.5.1.5 – "Site Proximity Missiles (Except Aircraft)"
- 3.5.1.6 – "Aircraft Hazards"
- 3.5.2 – "Structures, Systems, and Components to be Protected from Externally Generated Missiles"

Sections with no Open Items (Contd)

- 3.6.1 – “Plant Design for Protection against Postulated Piping Failures in Fluid Systems Outside Containment”
- 3.7.1 – “Seismic Design Parameters”
- 3.7.4 – “Seismic Instrumentation”
- 3.8.5 – “Foundations”
- 3.9.1 – “Special Topics for Mechanical Components”
- 3.10 – “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment”
- 3.12 – “ASME BPV Code Class 1, 2, and 3 Piping Systems and Associated Support Design
- 3.13 – “Threaded Fasteners—ASME BPV Code Class 1, 2, and 3”

3.6.3 – Leak Before Break Evaluation Procedures

Eric Reichelt

Section 3.6.3 - Leak Before Break

- NuScale DCD requested to apply Leak Before Break (LBB) methodology to Main Steam Piping and Feedwater Piping Systems.
- Main Steam Piping (MSS) is NPS 8 and NPS 12. The Feedwater Piping (FWS) is NPS 4 and NPS 5.
- Unique aspect for NuScale is curved pipe system, and making sure fabrication (pipe bending) limits cold-working to an acceptable limit.
- The methods and criteria to evaluate LBB are consistent with the guidance in SRP 3.6.3, and NUREG-1061, Volume 3.

Section 3.6.3 - Leak Before Break

- Reviewed applicable NuScale DCD subsections in Section 3.6.3.
- Reviewed DCD references for applicability and use.
- Held public meetings with NuScale staff about technical issues and RAIs leading to proposed DCD markups.
- The technical issues and response by NuScale to RAIs were acceptable and were therefore closed.
- The staff is currently reviewing the methodology and will perform a confirmatory analysis on the FWS proprietary information and data provided by the applicant on June 8, 2019.
- The staff will review the methodology and will perform a confirmatory analysis on the MSS when NuScale provides the proprietary information and data scheduled for July 15, 2019.
- Based on results of the confirmatory analysis, this is a confirmatory item.

Section 3.5.1.3 – Turbine Missile and Section 3.5.3 – Barrier Design Procedures

John Honcharik
BP Jain

Regulatory Basis and Use of Barriers

- 10 CFR Part 50, Appendix A, GDC 4, requires SSCs important to safety to be appropriately protected against environmental and dynamic effects, including the effects of missiles that may result from equipment failure.
- Safety-related and risk-significant SSCs for the NuScale design are located within the RXB and CRB.
- Turbine generator rotor shafts are unfavorably oriented such that the RXB and CRB are within the turbine low-trajectory hazard zone.
- To meet the requirements of 10 CFR Part 50, Appendix A, GDC 4, NuScale proposes to use installed or existing structures for protecting essential SSCs that meet the acceptance criteria in DSRS Section 3.5.3.

Determining Turbine Missile Parameter on Barriers

- Open Item 03.05.01.03-1 : The staff determined that NuScale had not used the full spectrum of turbine missiles (size, weight and speed) which includes up to half of the last stage of the rotor in the barriers analysis.
- NuScale provided additional information in June, and the staff is currently reviewing the information on spectrum of turbine missile parameters.
- Section 3.5.3 addresses the verification of the barriers used for turbine missile protection.

Section 3.5.3 - Turbine Missile Barrier

Open Item 03.05.03-1

- Postulated Turbine Missile spectrum - an open item 3.5.1.3-1
  1. Turbine Blade : Weight: 32.6 lbs ;Velocity 784 mph,
  2. Turbine Blade with rotor fragment : Weight 52.6 lbs ; velocity 996 mph
  3. Half of last stage rotor: weight 3079 lbs; velocity 350 mph
- Staff Guidance in SRP 3.5.3 used to review Barrier Design Procedures for the local and overall damage.
- Acceptance Criteria – Concrete barriers should be thick enough to prevent backface scabbing.
- Staff is required to make the following review findings.
  - The procedures used for barriers design for the impact of design basis turbine missiles are acceptable.
  - Information presented in the DCD provides a reasonable assurance that Reactor and Control building walls provide adequate protection to essential SSCs from the impact of the design basis turbine missiles.
- Complete and technically sufficient information not available in DCD for staff to make safety findings. NuScale submitted supplementary info in June.
- Staff is reviewing NuScale's response to barrier design issues.

Section 3.5.3 - Turbine Missile Barrier

Open Item 03.05.03-1

- Overview of Key Barrier Design Issues:
- NuScale used FEM procedure for penetration depth calculation instead of SRP 3.5.3 empirical NDRC formula.
- Local and global damage assessment for the full turbine missile spectrum was not provided
- FEM procedure is not previously reviewed by the staff for high speed turbine missile impact
  - ➢ Staff to review Validation and Verification of the FEM approach

  Computer Program - not reviewed for turbine missiles
  - ➢ Benchmarking of computer program against relevant test data required
- Penetration depth from the impact of deformable missile is significantly reduced to almost 1/3 of that from the missile impact (e.g., reduced to 20” from 54.6”)
  - ➢ Validation by test and/or analytical results required

Section 3.5.3 - Turbine Missile Barrier

Conclusions

- Staff will review the barrier design information and conduct audits as required to insure compliance with the regulatory requirements

- NuScale application to demonstrate compliance with regulatory requirements for barrier design

Sections 3.7.2 – Seismic System Analysis and 3.7.3 – Seismic Subsystem Analysis

Sunwoo Park

Sections 3.7.2 and 3.7.3 - Overview

- Review Scope
  - **3.7.2 Seismic System Analysis** – Seismic Category I structures: Reactor Building, Control Building
  - **3.7.3 Seismic Subsystem Analysis** – example, Bioshield
- Phase-2 SER had 4 Open Items
- 2 OIs resolved to date
- 2 OIs require additional information and evaluation
  - 1 in Section 3.7.2 – to be discussed in the following slides
  - 1 in Section 3.7.3 – to be discussed in the following slides

Section 3.7.2 – Seismic System Analysis

Open Item 03.07.02-1

- Seismic load demands for NPM-RXB interface supports are determined by analysis of the RXB SASSI model and NPM ANSYS model
- Applicant expanded analysis cases to include 130% NPM nominal stiffness
- Applicant adopted new methodology for modeling hydrodynamic mass of pool water
- Applicant provided information from new analysis, which is currently in staff evaluation
- Applicant is to provide information about seismic loads on pool walls from new analysis

Section 3.7.3 – Seismic Subsystem Analysis

Open Item 03.07.03-1

- Bioshields are Seismic Category II concrete covers placed on top of each NPM as additional radiological barrier
- Bioshield is removed during NPM refueling, and removed Bioshield is stacked on top of adjacent Bioshield
- During December 2018 audit, staff identified issues concerning seismic design of stacked Bioshields
- Applicant revised the design approach to address identified issues
- Applicant provided information on new seismic analysis and design of Bioshields, which is currently in staff evaluation

Sections 3.7.2 and 3.7.3 - Conclusion

- Applicant is undertaking actions for timely resolution of the open items.

- NuScale DCA demonstrates compliance with regulatory requirements for seismic system and seismic subsystem analysis.

Sections 3.8.4 – Other Seismic Category I Structures

Robert Roche-Rivera

Section 3.8.4 – Overview

- Review Scope
  - DCA Tier 2 Section 3.8.4, Other Seismic Category I Structures
  - DCA Tier 2 Appendix 3B, Design Reports and Critical Section Details
- Reviewed DCA Tier 2 Section 3.8.4, associated Appendix 3B, and the listed tables and figures in accordance with the DSRS Section 3.8.4 acceptance criteria.
- Held bi-weekly public meetings with the applicant to discuss technical issues and resolutions to RAIs
- Conducted audit of design reports supporting the information provided in the DCA
- Phase 2 SER had 5 Open Items
  - All OIs have been resolved

Section 3.8.4 – Other Seismic Category I Structures

- Reviewed and compared the applicant’s design procedures and associated results to the applicable code and or standard acceptance criteria and allowable
  - Loads and load combinations considered and structural capacity determinations are in accordance with the applicable code and or standard
  - Displacement and strain results meet the applicable code and or standard allowable
  - Structural capacities are greater than design basis demands (i.e. D/C ratios less than 1)
- Concluded that the applicant’s methods for demonstrating the design adequacy of the structures are consistent with the NRC’s regulatory requirements

Section 3.8.4 – Other Seismic Category I Structures (Cont'd)

- Open Item 3.8.4-1: RAI 8971, Question 3.8.4-13
  – Staff finding:
    - Design evaluation for temperature (operating and accident) and accident pressure demands not provided.
  – Resolution:
    - Applicant performed design evaluations of temperature and accident pressure demands and other concurrent loads for the RXB
    - The results demonstrated that the concrete, rebar, and liner strains are below the strain limits

Section 3.9.4 – Control Rod Drive Systems

Nicholas Hansing

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Key Design Considerations and Features

- Pressure housing/ electromagnetic components very similar to existing fleet
  (Pressure housings designed & constructed to ASME BPV Code Class 1 requirements)
- Long drive shaft and remote disconnect mechanism are unique
- Design standards and testing programs were emphasized in the review

Drive Shaft

- ASME BPV Code, Subsection NG, component (internal structure), and is seismic Category I
- Additional design requirements are specified in the DCA, such as Service Level loading combinations

Remote Disconnect Mechanism

- Applicant confirmed the remote disconnect coil is always deenergized during normal operations and remains in this state during a reactor trip
- Mechanism was tested during Key Feature Mock-Up Testing for its full design life of 150 cycles (5x estimated cycles expected) with satisfactory performance

Section 3.9.4

Drop Testing

- Appendix B compliant testing with prototypical components
- Staff independently verified dimensions of design documents to as-built dimensions of test facility – important dimensions like diametrical gap were consistent
- 1” misalignment possible at each interface- testing bounded manufacturing tolerances & seismic displacements
- 14 configurations used, number of variables reduced using fabrication constraints & testing worst-case scenarios – tested displacement exceeded the maximum expected displacement by more than a factor of two
- The most limiting drop (maximum displacement and longest drop time) was bounded by the performance assumed in the safety analysis for control rod drop time.

Operability Assurance Program

- Includes performance testing, stability testing, endurance testing, and production testing.
- Will be completed by COL applicant,
- DC applicant has provided overview of program in DC application with a proposed COL Item to implement the program and provide a summary of the testing program and results.

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Section 3.9.6 – Functional Design and Qualification, and Preservice Testing and Inservice Testing of Pumps, Valves, and Dynamic Restraints

Thomas Scarbrough

Section 3.9.6

- NRC staff reviewed functional design and qualification, and preservice testing (PST) and inservice testing (IST) program description for NuScale safety-related valves.
  - DCA provides full description of PST/IST programs with SER Confirmatory Items.
- First-of-a-kind (FOAK) Emergency Core Cooling System (ECCS) Valves and Containment Isolation Valves (CIVs).
- ECCS Valve Design Demonstration Testing in 2019.
- SER Open Items for ECCS valve design and ITAAC.

ECCS Valve Design

- 3 Reactor Vent Valves (RVVs) and 2 Reactor Recirculation Valves (RRVs) allow natural circulation for emergency core cooling.
- Each RVV and RRV has FOAK design arrangement of a main valve, inadvertent actuation block (IAB) valve, solenoid trip valve, and solenoid reset valve connected by hydraulic tubing.
- Proprietary design and operation of IAB valve can be discussed in a closed session.
- ECCS Valve Design Demonstration Testing to satisfy 10 CFR 50.43(e) initiated June 2019.
- Ongoing NRC audit.

Functional Design and Qualification

- DCA specifies ASME Standard QME-1-2007 for qualification of safety-related valves as accepted with conditions in RG 1.100 (Revision 3).
- ITAAC acceptance criteria for functional qualification of safety-related valves require Qualification Report (specified in QME-1 standard).
- Safety-Related Valve Design Specification audits conducted in 2017 and 2018 with follow-up items for close-out in 2019.
- No safety-related motor-operated valves, pumps, or snubbers.

Containment Isolation Valve Design

- Hydraulic-operated actuators with ball valves.
- 16 Primary System Containment Isolation Valves (PSCIVs) have FOAK design with two actuators and ball valves for same valve body.
- 6 Secondary System Containment Isolation Valves (SSCIVs) have one actuator and ball valve.
- CIVs will be qualified in accordance with ASME QME-1-2007 as accepted in RG 1.100 (Revision 3).
- NRC audit in 2018 with follow-up items for close-out in 2019.

Section 3.6.2 – Introduction Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Renee Li

Section 3.6.2

- This review is to ensure that NuScale's design provides adequate protection against the effects of the postulated pipe ruptures and meet the GDC 4 requirement.
  - The review concentrated on those areas that were outside the guidance of BTP 3-4
    - Two isolation valves outside containment for penetration piping
    - Bolted connection of the ECCS Valves to the RV head
- Reviewed TR-0818-61384-P, Revision 1 that addresses applicant's pipe rupture hazards analysis and the results.
- One SER Open Item related to the bolted connection of the ECCS Valves to the RPV head.

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Break Exclusion Areas for NuScale Plant Design

- NuScale two isolation valves outside containment configuration deviates from the BTP 3-4 containment penetration area.
  - Evaluation will be provided in the Chapter 6 presentation.
- Applicability of break exclusion to the containment penetration areas.
  1. Design stress and fatigue limit criteria for the piping segments within the NuScale's break exclusion area are consistent with BTP 3-4 guidelines and the associated calculation results are within the relevant BTP 3-4 stress and fatigue limits for break exclusion.
  2. Augmented 100 percent volumetric in-service examination requirement for all the welds within the break exclusion areas is consistent with BTP 3-4 staff guidelines.
- The NRC staff found the above applicant's justification for its application of break exclusion areas acceptable.

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Break Exclusion of RVV and RRV Bolted Connection

- Applicability of break exclusion to the ECCS Valve connection to the RPV:
  1. Bolting stress design criteria per ASME BPV Section III, NB-3230 meet the intent of BTP 3-4 stress acceptance guidance for typical piping system
  2. Cumulative usage factor of 1.0 is acceptable
     - ASME Section III, NB-3230.3(c) fatigue criteria applies a strength reduction factor of no less than 4.
     - Bolted connections are not susceptible to the phenomena (e.g., unexpected modes of operation vibration, stress corrosion cracking, water hammer,…etc.) that might adversely affect fatigue evaluations.
       - Basis for recommending CUF of 0.1 in BTP 3-4.

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Break Exclusion of RVV/RRV Bolted Connection (Cont.)

3. Augmented fabrication and in-service examination requirements.
   - In-service inspection, Ultrasonic examination of the bolt at least once every 10 years.
4. Bolting design adopted NUREG-1339 guidelines with highly SCC resistant material (Alloy 718)
5. Highly sensitive leakage detection system (sensitive to leak rate as low as 0.001 gallon per minute)

- NRC staff found that the applicant's justification provides reasonable assurance
- NRC audit of the applicant's stress and fatigue calculations to close this SER Open Item.

3.8.2 – Steel Containment, 3.9.3 – ASME BPV Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures 3.9.5 – Reactor Pressure Vessel Internals

Alexander Tsirigotis

Design Stress/Fatigue Analyses

- During a Design Specification audit, stress analyses and fatigue evaluations of the RPV, CNV and RVI were not available.
  - SECY 90-377 establishes position that this level of detail should be complete at DC stage.
  - Open Item
- Stress analyses are being updated.
- NuScale evaluated fatigue locations
  - Selected most critical locations to conduct ASME BPV Code, Section III fatigue analysis
  - All locations scheduled to be available for audit by end of July

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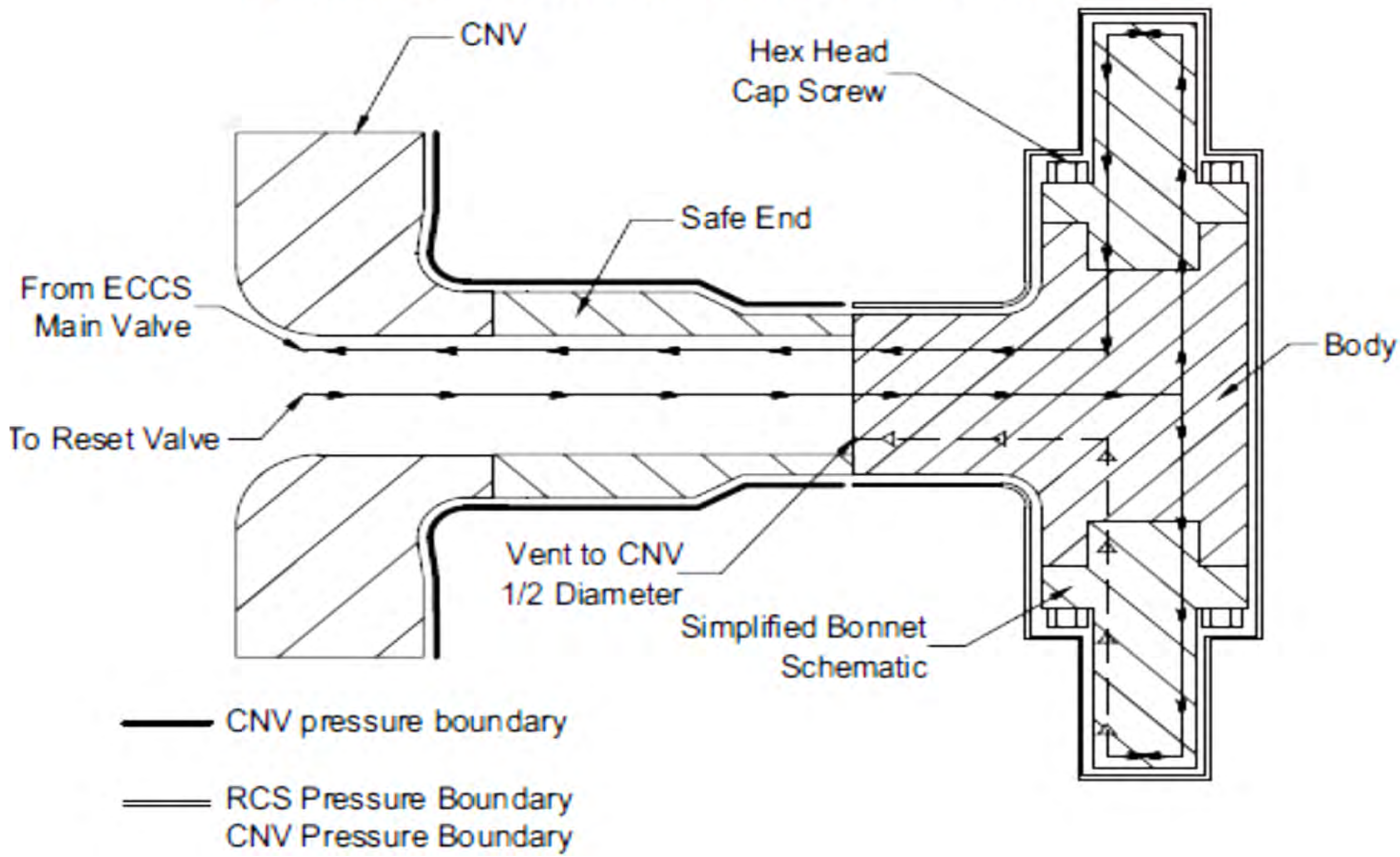
RCS/CNV Pressure Boundary

- Valve body of ECCS Valve trip/reset valve serves as both RCS and CNV pressure boundary
- ASME BPV Code, Section III requires surface examination of valve body
- NRC staff questioned adequate level of quality for design
- NRO office accepting ASME BPV Code, Section III as acceptable level of quality

ECCS Trip/Reset Solenoid Penetration

Figure 3.8.2-10: ECCS Trip/Reset Actuator Valve Pressure Boundary



Abbreviations

ACRS	*Advisory Committee on Reactor Safeguards*
AOV	*Air-Operated Valve*
ASME	*American Society of Mechanical Engineers*
BPV	*Boiler & Pressure Vessel*
BTP	*Branch Technical Position*
CIV	*Containment Isolation Valve*
COF	*Coefficient of Friction*
COL	*Combined License*
CNV	*Containment Vessel*
CRB	*Control Building*
DC	*Design Certification*
DSRS	*Design Structure Response Spectra*
DCA	*Design Certification Application*
ECCS	*Emergency Core Cooling System*
FEM	*Finite Element Method*
FWS	*Feedwater Piping*

Abbreviations

FOAK	First of a Kind
FSAR	Final Safety Analysis Report
HOV	Hydraulic-Operated Valve
IAB	*Inadvertent Actuation Block*
IST	*Inservice Testing*
ITAAC	*Inspections, Tests, Analyses, and Acceptance Criteria*
LBB	*Leak Before Break*
MSS	*Main Steam Piping*
MOV	*Motor-Operated Valve*
NDRC	*National Defense Research Committee*
NRO	*NRC Office of New Reactors*
OM	*Operation and Maintenance*
PST	*Preservice Testing*
QA	*Quality Assurance*
RAI	*Request for Additional Information*
RG	*Regulatory Guide*
RCS	*Reactor Coolant System*

Abbreviations

RPV	Reactor Pressure Vessel
RRV	*Reactor Recirculation Valve*
RVV	*Reactor Vent Valve*
RXB	*Reactor Building*
SER	*Safety Evaluation Report*
SRP	*Standard Review Plan*
SSCs	*Structures, Systems, and Components*

LO-0619-65928

June 14, 2019 Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Subcommittee Presentation, NuScale FSAR Chapter 6, Engineered Safety Features," PM-0619-65926, Revision 0

The purpose of this submittal is to provide presentation materials for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) NuScale Subcommittee meeting on June 18, 2019. The materials support NuScale's presentation of Chapter 6, "Engineered Safety Features," of the NuScale Design Certification Application.

The enclosure to this letter is the nonproprietary presentation entitled "ACRS Subcommittee Presentation, NuScale FSAR Chapter 6, Engineered Safety Features," PM-0619-65926, Revision 0.

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

If you have any questions, please contact Rebecca Norris at 541-602-1260 or at rnorris@nuscalepower.com.

Sincerely,

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

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Enclosure: "ACRS Subcommittee Presentation, NuScale FSAR Chapter 6, Engineered Safety Features," PM-0619-65926, Revision 0

Enclosure:

"ACRS Subcommittee Presentation, NuScale FSAR Chapter 6, Engineered Safety Features," PM-0619-65926, Revision 0

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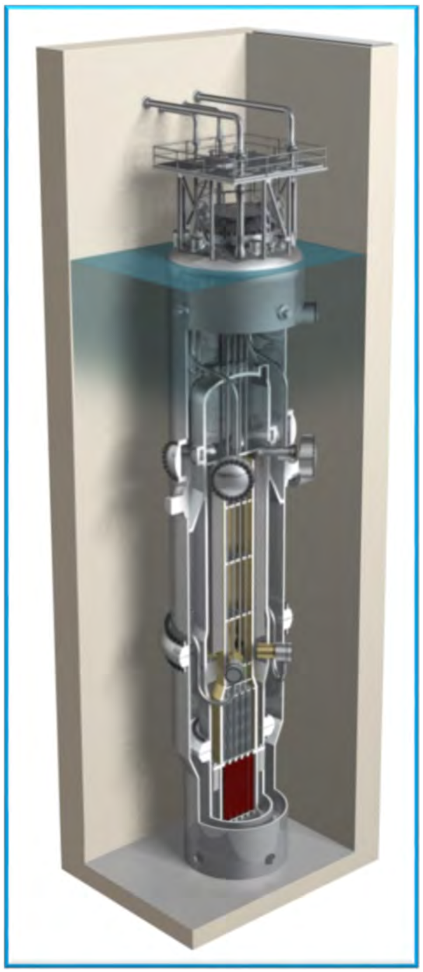
ACRS Subcommittee Presentation



NuScale FSAR

Chapter 6

Engineered Safety Features

June 18, 2019

PM-0619-65926
Revision: 0

Template #: 0000-21727-F01 R5

Presentation Team

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Chapter 6 – Engineered Safety Features

Section	Title
6.1	Engineered Safety Feature Materials
6.2	Containment System • Performance analysis results presented with FSAR Ch15
6.3	Emergency Core Cooling System • Additional figures will be provided in closed session
6.4	Control Room Habitability
6.5	Fission Product Removal and Control Systems
6.6	Inservice Inspection and Testing of Class 2 and 3 Systems and Components
6.7	Main Steamline Isolation Valve Leakage Control System

Objective

Provide a technical summary of NuScale engineered safety features. All containment peak pressure and temperature analyses related to 6.2.1 and 6.2.2 will be presented with Chapter 15 on Wednesday.

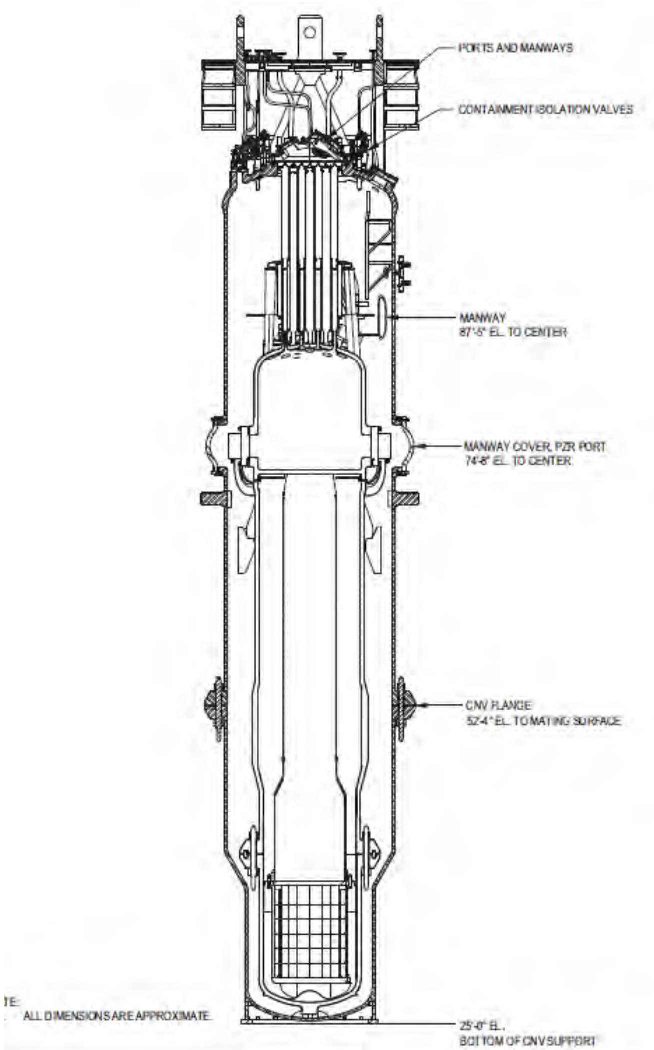
6.1 – Engineered Safety Feature Materials

- Components compatible with environmental conditions of normal operation, maintenance, testing, and accidents
- 60-yr component design life
- Grade SA-965, FXM-19 material provided in the core vicinity
  - Demonstrates good resistance to neutron embrittlement
- No protective coating allowed on the containment vessel (CNV) or any components within CNV

6.2.1 – Containment Functional Design

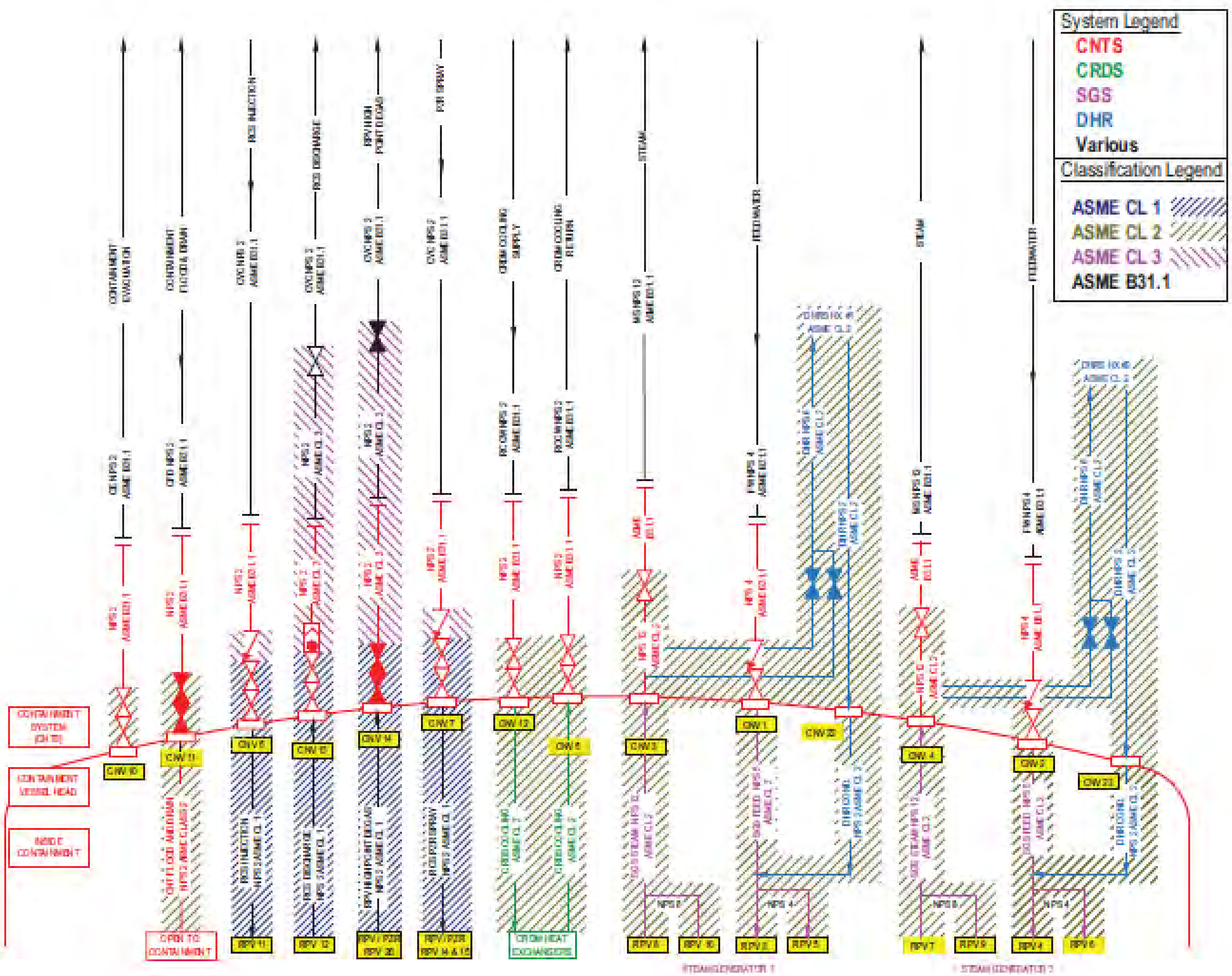
- CNV Primary Functions
  - Enclosure of the RPV, RCS and associated components
  - Containment of fission product releases
  - Containment of the postulated mass and energy releases inside containment
  - Operation of the ECCS by the retention of reactor coolant and the transfer of sensible and core heat to the UHS
  - Heat removal capability to maintain peak P/T less than design (GDC-50)
  - Reduction of peak P/T to < 50% within 24 hr (GDC-38)
- CNV peak P/T analysis discussed with Chapter 15

NuScale Containment Overview



- Designed and stamped as Class 1 ASME vessel
- 1050 psia design pressure
- 550 F design temperature
- Located directly within the ultimate heat sink
- SA-508 Grade 3, Class 2 upper containment material
- FXM-19 lower containment material
- Piping and electrical penetrations located above pool level
- ~15’ shell diameter

Containment Class Breaks



6.2.1 – COL Items

Item Number	Description
COL Item 6.2-3	Perform an analysis that, in consideration of the as-built containment internal free volume, demonstrates that containment design pressure and temperature bounds containment peak accident pressure and temperature. The evaluation value for containment internal free volume must include margin to address the complex shapes of internal structures and components and manufacturing processes.

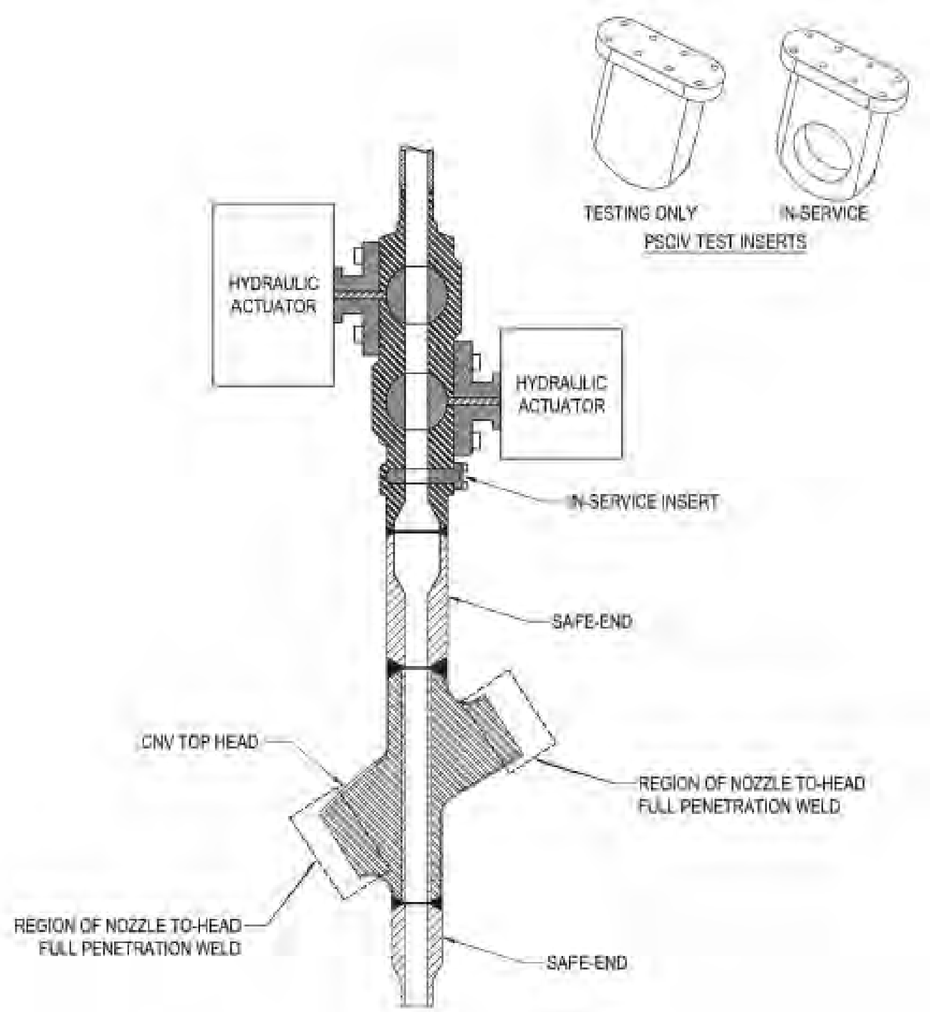
6.2.2 – Containment Heat Removal

- Pool water level just below CNV upper head during normal operation (> 30 days cooling volume)
- CNV steel wall allows direct, passive heat transfer to pool
- For limiting peak pressure case, CNV pressure reduced to < 50% of peak within 2 hr
- Exemption from GDC 40 requested - testing of containment heat removal system justified in FSAR Section 3.1.4
- CNV removes accident heat from NPM as accident progresses; no operator action required

6.2.4 – Containment Isolation System

- Containment isolation valves (CIVs) meet GDC 55, 56, & 57 with the following exemptions requested:
  - Both automatic CIVs located outside of the CNV for GDC 55 & 56 penetrations
  - Closed system piping both inside and outside of the CNV for GDC 57 penetrations (DHRS)
- Piping between CNV and CIV designed to break exclusion zone criteria
- Primary system containment isolation valves (PSCIVs) constructed and stamped as ASME Class 1 valves
- Secondary system containment isolation valves (SSCIVs) constructed and stamped as ASME Class 2 valves
- CIVs hydraulically operated, fail closed

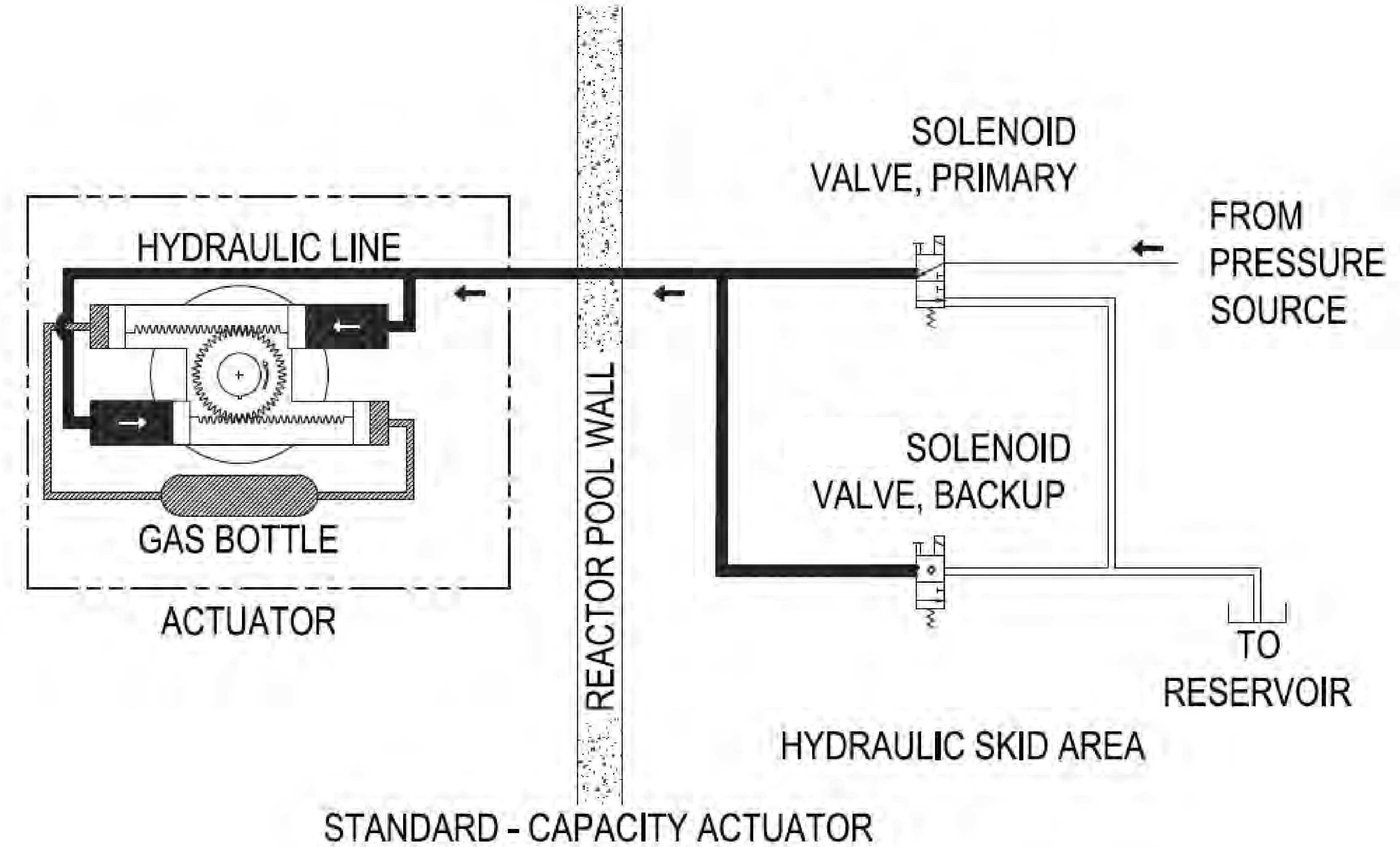
Primary CIV Arrangement



6.2.4 – Containment Isolation System (Cont)

- Solenoid valves for each CIV are located remotely on a hydraulic control skid
- Each module has two separate dedicated hydraulic control skids, located in separate areas of the Reactor Building (RXB)
- Each hydraulic control skid operates one division of valves

Actuator Hydraulic Schematic



6.2.5 – Combustible Gas Control

- Containment integrity analyzed for effects of complete hydrogen burn and detonations
- Exemption requested from10 CFR 50.44(c)(2) requirement for inerted atmosphere or limited hydrogen concentration
- Design provides a mixed CNV volume during design basis events (DBEs) and beyond design basis events (BDBEs)
  - No sub-compartments: design relies on natural convection

6.2.5 – Combustible Gas Control (cont)

- Equipment provided for monitoring in-containment hydrogen concentration – capable of continuously monitoring hydrogen gas concentrations after DBEs or BDBEs
  - Hydrogen and oxygen analyzers provided in the process sampling system
  - Monitoring equipment performance requirements still under discussion with NRC staff
- Design includes connections for cleanup equipment to scrub CNV atmosphere prior to release

6.2.6 – Containment Leakage Testing

- All CIVs and CNV isolation barriers designed to support local leakage testing (GDC 53, 54)
- Follows 10 CFR 50, Appendix J for Type B and Type C testing
- Design supports exemption from GDC 52 (Type A ILRT) testing (see TR-1116-51962, Rev 0 and FSAR 6.2.6.1)
  - Fabrication and testing similar to reactor pressure vessel (RPV) requirements
  - All known leakage pathways tested through local leak rate testing
  - Comprehensive inservice inspection (ISI) meets ASME Class 1 requirements
  - CNV hydrostatically tested

6.2.6 – Containment Leakage Testing (cont)

- NuScale commitments made to support the GDC 52 exemption include:
  - CNV preservice design pressure leakage test described by FSAR Section 6.2.6.5.2 (ITAAC #23)
  - Demonstration that the as-built containment maintains flange contact pressure at accident temperature concurrent with peak pressure conditions (COL Item 6.2-2)
  - Containment flange bolting preload verification

6.2.6 – COL Items

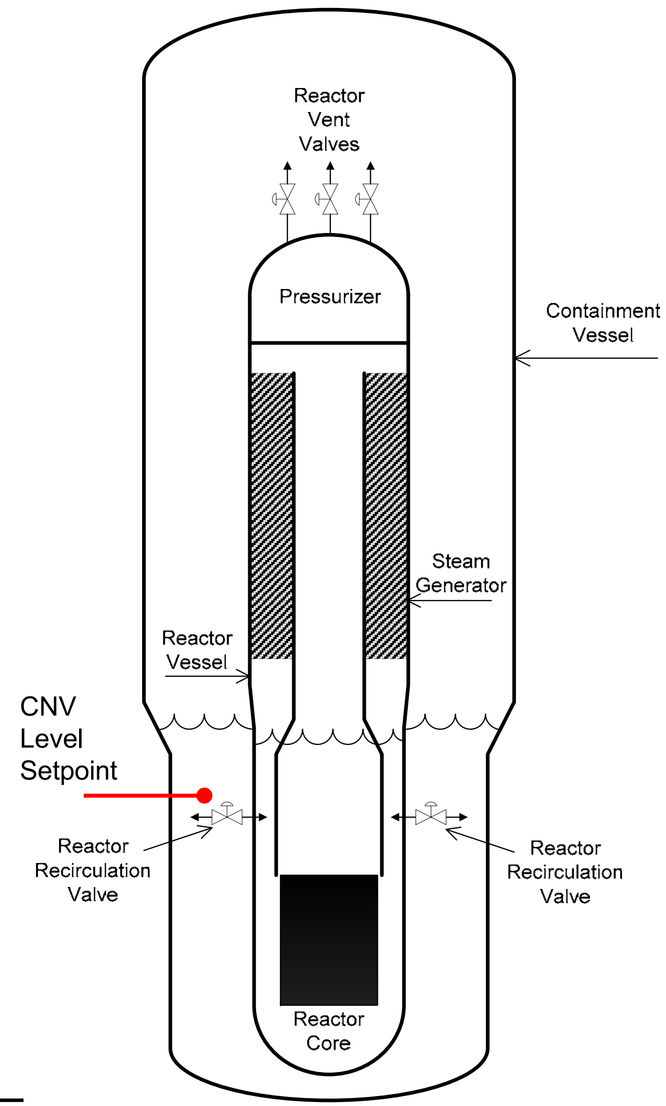
Item Number	Description
COL Item 6.2-1	Develop a containment leakage rate testing program that will identify which option is to be implemented under 10 CFR 50, Appendix J. Option A defines a prescriptive-based testing approach whereas Option B defines a performance-based testing program.
COL Item 6.2-2	Verify that the final design of the containment vessel meets the design basis requirement to maintain flange contact pressure at accident temperature, concurrent with peak accident pressure.

6.2.7 – Fracture Prevention of CNV

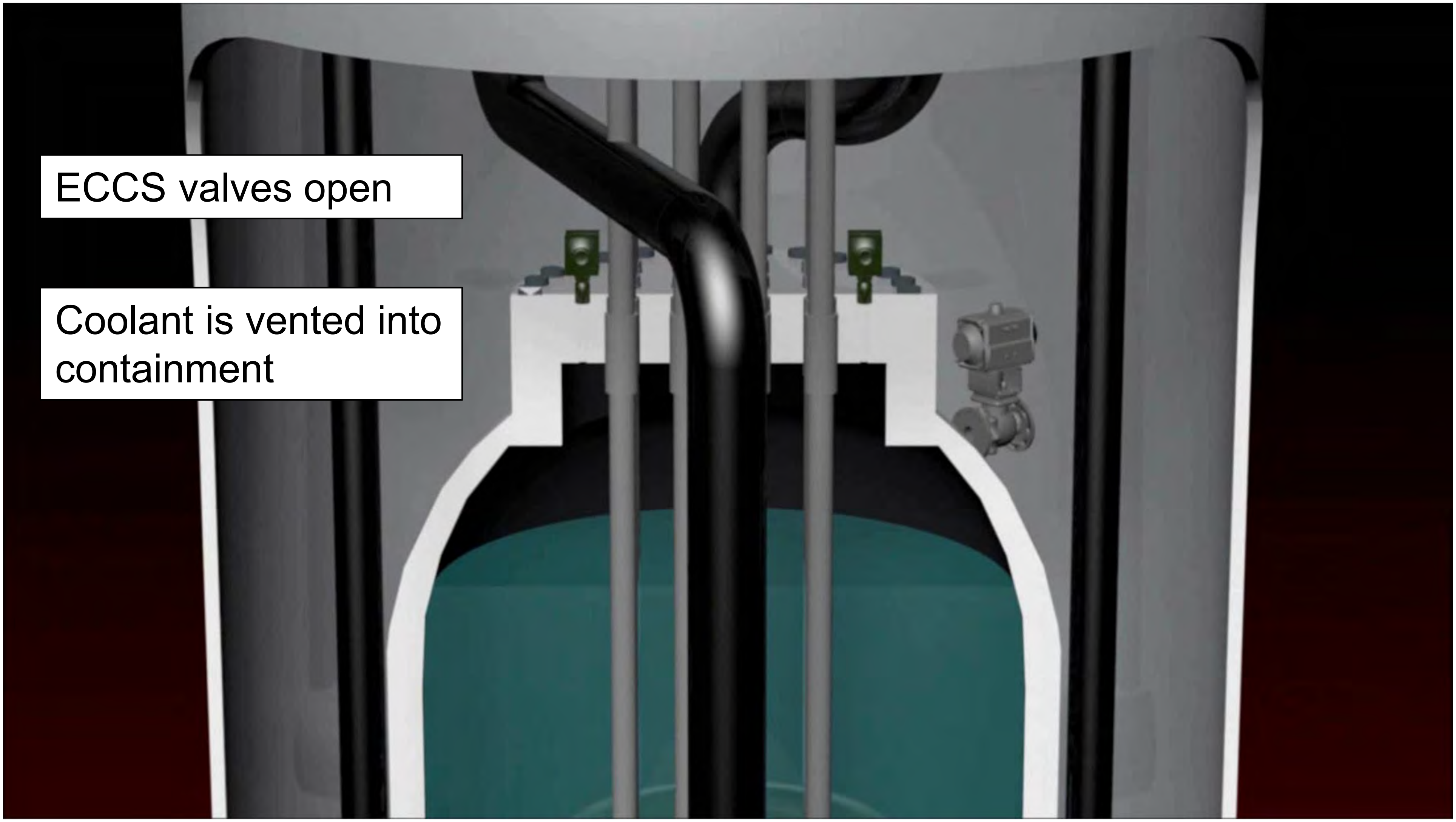
- CNV meets relevant parts of GDC 1, 16, 51 (thus, ASME Section III, Subsection NB)
- Ferritic CNV pressure boundary meet ASME Section III, Subsection NB fracture toughness requirements
- Lower CNV made of austenitic stainless steel (grade FXM-19) which is not subject to neutron embrittlement at CNV fluence levels

6.3 – Emergency Core Cooling System (ECCS)

- Cools core when it cannot be cooled by other means, such as loss of coolant accidents
- Main components are pilot-actuated valves
  - 3 reactor vent valves (RVV)
  - 2 reactor recirculation valves (RRV)
- Actuation signals
  - high CNV level
  - loss of AC power for 24 hr



Emergency Core Cooling System (cont'd)

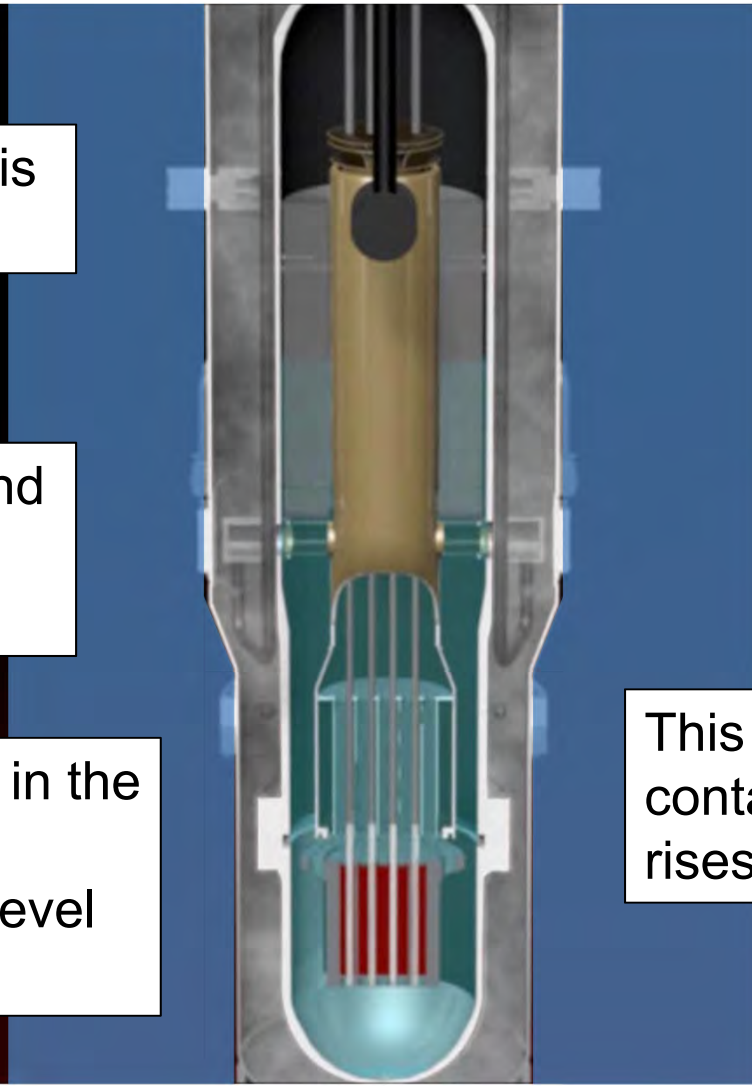


Emergency Core Cooling System (cont'd)

Containment vessel is surrounded by UHS

Steam condenses and liquid collects in containment vessel

As RPV level lowers in the downcomer region, containment vessel level rises



This continues until containment vessel level rises above RRVs

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Emergency Core Cooling System (cont'd)

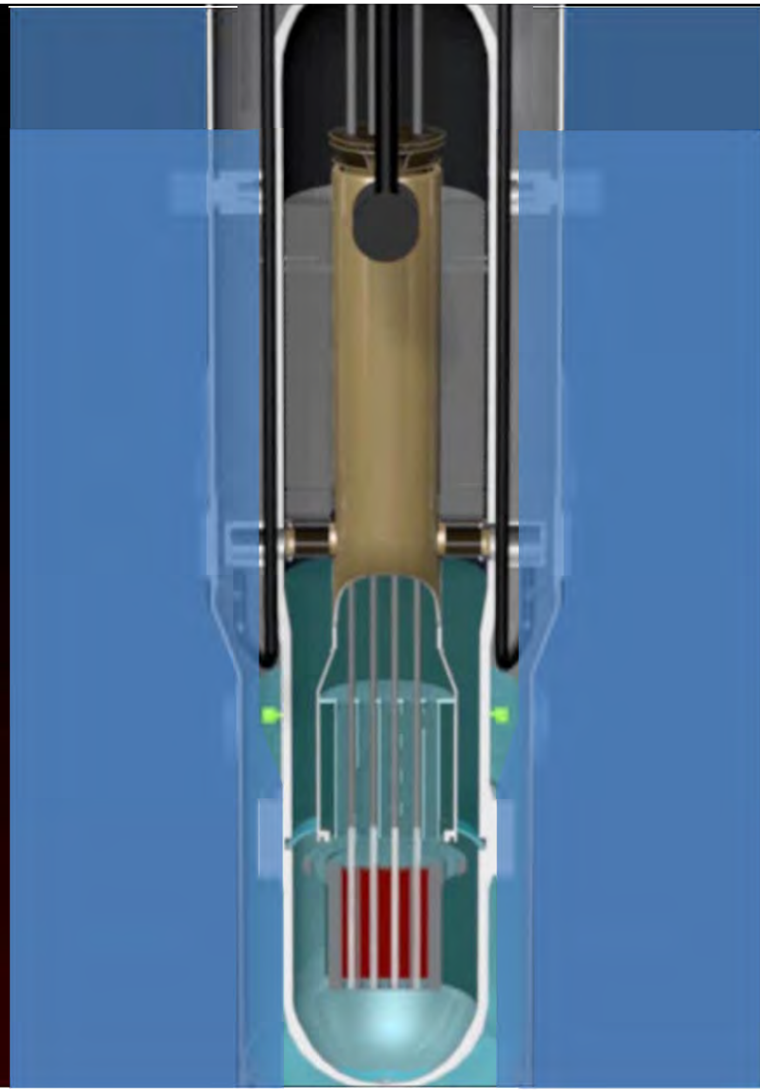
Coolant flows from the containment vessel into the RPV

Flows to bottom of RPV and back to the core

Heated by the core

Flows up to top of RPV

Coolant again exits through RVVs



Coolant level remains above the core, and containment level remains above the RRVs

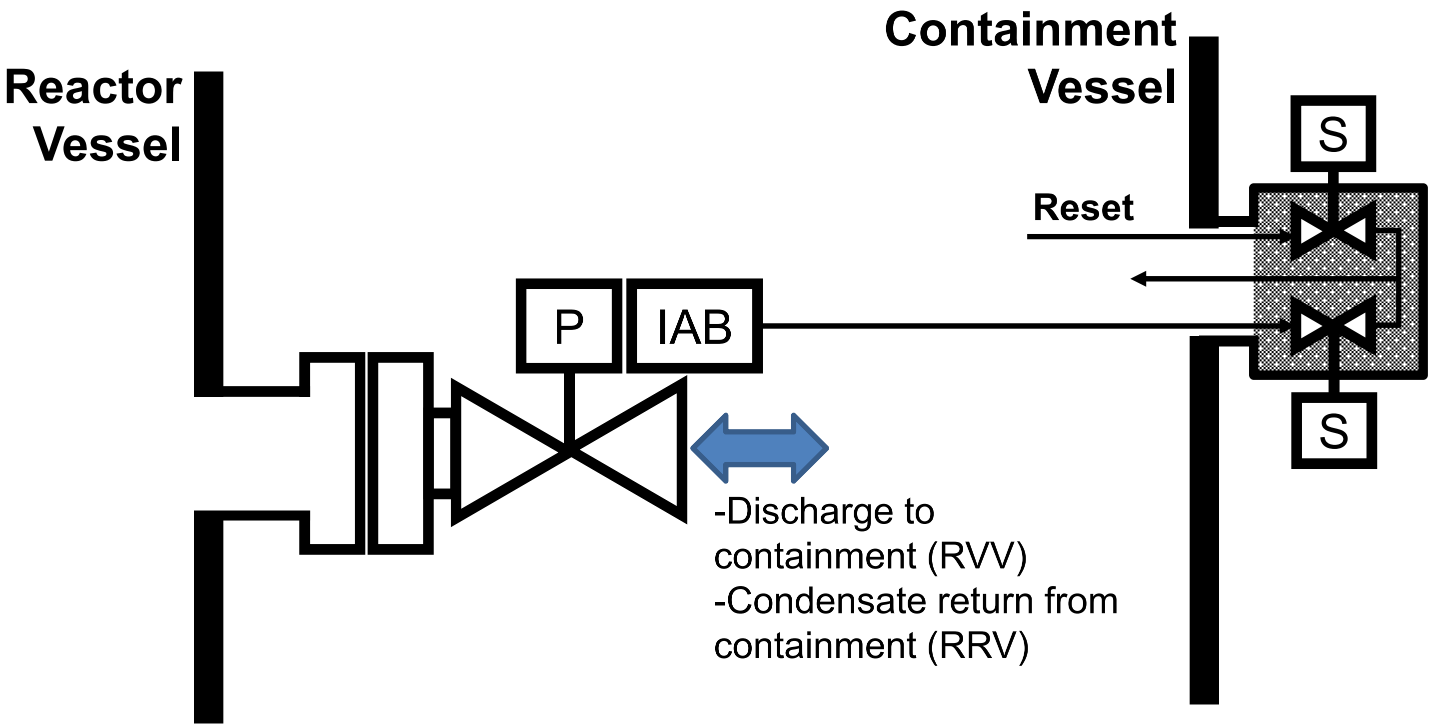
Heat is transferred
- Core to coolant
- Coolant to containment vessel
- Containment vessel to UHS

Emergency Core Cooling System (cont'd)

- Only 2 RVVs and 1 RRV required for operation
- Core stays covered for all design basis events
- ECCS does not require operator action or support from non-safety systems. Operators can manually actuate ECCS from control room.
  - Operator alerted if a valve fails to move to its safety position
  - Engineered safety features actuation system (ESFAS) design prohibits manual override of ESFAS signal
- Components form part of reactor coolant pressure boundary (RCPB), ASME Class 1 components
- Auxiliary ECCS Function
  - Low temperature overpressure (LTOP) protection for RPV
  - Module protection system (MPS) actuation logic based on pressure-temperature limits for RPV

Emergency Core Cooling System (cont'd)

- Simplified main valve and pilot actuator assembly configuration



ECCS Valve Operation

- Removing power to the solenoid actuators actuates the main valves which operate using RCS pressure
  - Spring ensures the ECCS valves stay open after depressurization
- Each ECCS main valve includes an inadvertent actuation block (IAB) device
  - Operates solely on differential pressure between reactor vessel and containment
    - The IAB is normally not engaged, only engages on high differential pressure
    - The IAB does not actuate for any design basis ECCS demands
  - Prevents inadvertent opening of an ECCS main valve until the differential pressure between the CNV and RCS reaches a predetermined range

Emergency Core Cooling System (cont’d)

- Debris generation is limited by restricting use of insulation, paint, and coatings in containment
  - Debris effects evaluated, see FSAR Section 6.3.3.1
- COL Item 6.3-1 requires COL applicant to implement cleanliness program to limit debris within the CNV
  - Support debris evaluation assumptions
- ECCS capable of post-accident, extended long-term cooling beyond 72 hours (PDC 35 and GDCs 36 and 37)

6.3 – ECCS COL Item

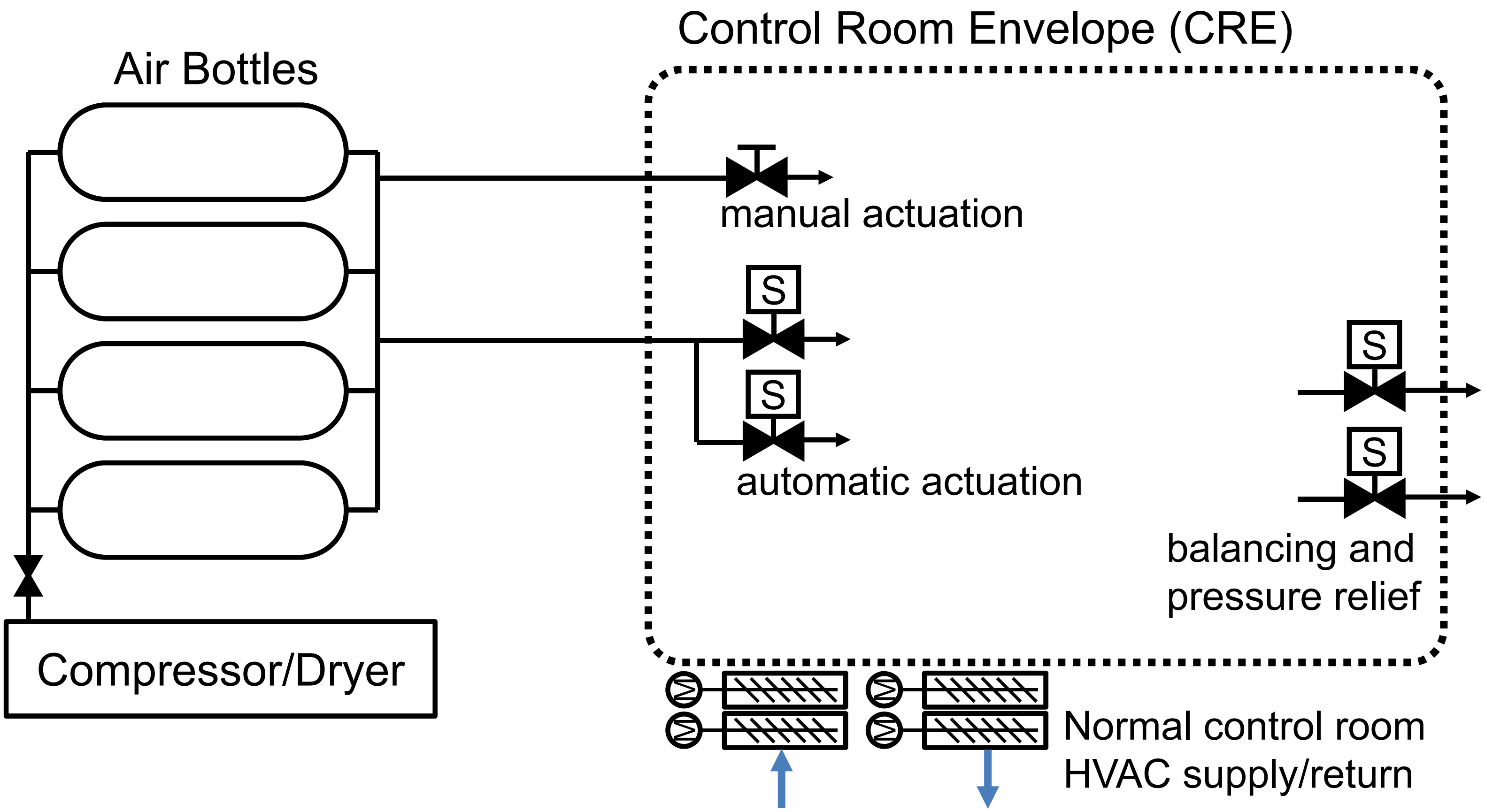
Item Number	Description
COL Item 6.3-1	Describe a containment cleanliness program that limits debris within containment. The program should contain the following elements: • Foreign material exclusion controls to limit the introduction of foreign material and debris sources into containment. • Maintenance activity controls, including temporary changes, that confirm the emergency core cooling system function is not reduced by changes to analytical inputs or assumptions or other activities that could introduce debris or potential debris sources into containment. • Controls that limit the introduction of coating materials into containment. • An inspection program to confirm containment vessel cleanliness prior to closing for normal power operation.

6.4 – Control Room Habitability

- The control room habitability system (CRHS) is a passive system in standby during normal plant operation
- Provides breathable air and positive pressure to the control room if normal control room area ventilation is unavailable (FSAR 9.4.1)
  - Supports 20 personnel for 72 hours without electrical power
  - Air inventory monitored by pressure and temperature instrumentation
- Automatic valves and manual actuation line provided
- CRHS stored air and supply components are Seismic Category I
- Design includes connection to offsite air supply source if needed past 72 hours

6.4 – Control Room Habitability

- Simplified CRHS diagram



Control Room Habitability (cont'd)

- Classified nonsafety-related, however, augmented quality requirements are applied
- COL Item 6.4-5 requires applicant periodically test and inspect system
- CRHS automatic actuation upon
  - Loss of AC to both normal control room HVAC system (CRVS) air handler units for 10 minutes, **or**
  - High rad downstream of CRVS air filtration unit, **or**
  - Loss of AC to all four EDSS-C battery chargers

-

6.4 – COL Items

Item Number	Description
COL Item 6.4-1	Comply with Regulatory Guide 1.78 Revision 1, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."
COL Item 6.4-5	Specify testing and inspection requirements for the control room habitability system and control room envelope integrity testing as specified in Table 6.4-4.

6.5 – Fission Product Removal and Control

- Reactor building and associated HVAC not credited in dose analysis
- No containment spray system

6.6 – Inservice Inspection and Testing

- ISI program for Class 2 & 3 systems/components based on 10 CFR 50.55a(g)(3)
- Preservice inspections and ISIs are performed on ASME BPVC Class 2 and 3 components in accordance with ASME BPVC Section XI
  - Inspections performed per Class 1 requirements
- ISI Program includes augmented volumetric and surface inspections to protect against postulated piping failures
  - High energy fluid system piping
  - Break exclusion zone piping

-

6.6 – COL Items

Item Number	Description
COL Item 6.6-1	Implement an inservice testing program in accordance with 10 CFR 50.55a(f).
COL Item 6.6-2	Develop preservice inspection and in-service inspection program plans in accordance with Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), and establish the implementation milestones for the program. The COL applicant will identify the applicable edition of the ASME BPVC used in the program plan consistent with the requirements of 10 CFR 50.55a. The COL applicant will, if needed, address the use of a single in-service inspection program for multiple NuScale Power Modules, including any alternative to the code that may be necessary to implement such an in-service inspection program.

Acronyms (page 1 of 3)

ANSI - American National Standards Institute
ASME - American Society of Mechanical Engineers
BDBE - beyond design basis event
BPVC - Boiler and Pressure Vessel Code
CIV - containment isolation valve
CNV - containment vessel
CNTS - containment system
COL - combined license
CRDS - control rod drive system
CRE - control room envelope
CRHS - control room habitability system
CRVS - normal control room HVAC system
DBE - design basis event
DHR(S) - decay heat removal (system)
ECCS - emergency core cooling system
ESFAS - engineered safety features actuation system

Acronyms (page 2 of 3)

FSAR - Final Safety Analysis Report
GDC - General Design Criteria
HVAC - heating ventilation and air conditioning
IAB - inadvertent actuation block
ILRT - integrated leak rate testing
ISI - inservice inspection
ITAAC - Inspections, Tests, Analyses, and Acceptance Criteria
LTOP - low temperature overpressure
MPS - module protection system
NPM - NuScale Power Module
NPS - nominal pipe size
PDC - principal design criteria
PSCIV - primary system containment isolation valve
P/T - pressure/temperature
PZR - pressurizer
RCPB - reactor coolant pressure boundary

Acronyms (page 3 of 3)

RCS - reactor coolant system
RG - Regulatory Guide
RPV - reactor pressure vessel
RRV - reactor recirculation valve
RVV - reactor vent valve
RXB - Reactor Building
SGS - steam generator system
SSCIV - secondary system containment isolation valve
UHS - ultimate heat sink

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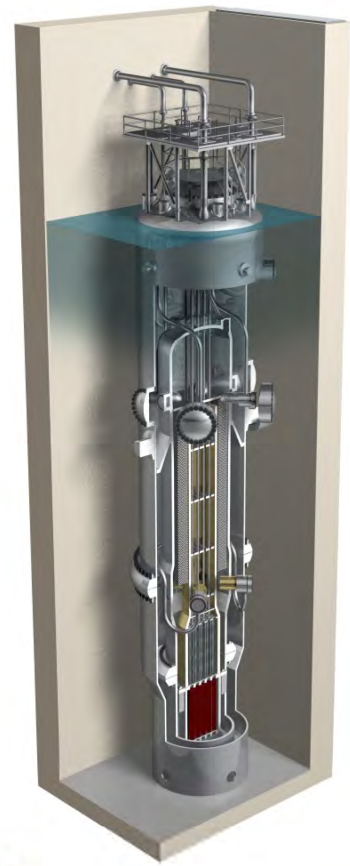

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Protecting People and the Environment

Safety Evaluation with Open Items: Chapter 6, “Engineered Safety Features”

NuScale Design Certification Application

ACRS Subcommittee Meeting
June 18, 2019

Agenda

- NRC Staff Review Team
- Overview of the NRC Staff's Presentation
- Section 6.2.4
- Sections 6.2.5 & 6.2.6
- Section 6.3
- Section 6.4
- Abbreviations

NRC Staff Review Team

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  - Robert Vettori, NRO
  - Nicholas McMurray, NRO
  - Anne-Marie Grady, NRO
  - Ryan Nolan, NRO
  - Shanlai Lu, NRO
  - Michelle Hart, NRO
  - Nan (Danny) Chien, NRO
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  - Syed Haider, NRO
  - Tom Scarbrough, NRO
  - Nick Hansing, NRO
  - Carl Thurston, NRO
  - Peter Lien, RES
- Project Management
  - Omid Tabatabai, Senior Project Manager
  - Greg Cranston, Lead Project Manager

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Overview of the NRC Staff's Presentation

- The focus of staff's presentation today is on SER Sections that involve Open Items and/or Exemption Requests, namely, Sections 6.2, "Containment Systems," 6.3, "Emergency Core Cooling System," and 6.4, "Control Room Habitability."
- The NRC staff's will present its evaluation of FSAR Sections 6.2.1, "Containment Structure," and 6.2.2, "Containment Heat Removal," as part of Chapter 15 presentations June 19-20
- There are several Open Issues in Section 6.3 that do not have a path forward to resolution. These OIs are directly tied to open issues in Chapter 15.
- For Chapter 6, the staff issued 34 RAIs (114 Questions). We have received responses to 111 Questions.

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Staff's Evaluation of Section 6.2.4, "Containment Isolation System"

Clint Ashley
Reactor Systems Engineer, NRO

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Regulatory Basis

- GDC 1 - Quality standards and records
- GDC 2 - Design bases for protection against natural phenomena
- GDC 4 - Environmental and dynamic effects design bases
- GDC 5 - Sharing of structures, systems and components
- GDC 16 - Containment design (essentially leak-tight barrier)
- GDC 54, GDC 55, GDC 56, GDC 57 - Provisions for piping systems penetrating containment
- 10 CFR 52.47(a)(8) - Technically relevant items in 10 CFR 50.34(f)
- 10 CFR 50.63 - Loss of all alternating current power
- 10 CFR 52.7 - Specific exemptions (refers to 10 CFR 50.12)

Review Guidance

- NuScale Design Specific Review Standard (DSRS)
  - Section 6.2.4, "Containment Isolation System."
- Regulatory Guide (RG)
  - RG 1.141, "Containment Isolation Provisions for Fluid Systems."
  - RG 1.151, "Station Blackout."
- U.S. Nuclear Regulatory Commission technical report designation (NUREG)
  - NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident."
  - NUREG-0737, "Clarification of TMI Action Plan Requirements."

Exemption Requests

- NuScale's DCA (Part 7) contains exemption requests associated with the following containment isolation requirements:
  - GDC 55
  - GDC 56
  - GDC 57
  - 10 CFR 50.34(f)(2)(xiv)(E)

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10 CFR 50.12 - Specific Exemptions

- Pursuant to 10 CFR 50.12 in part, the Commission may grant exemptions when special circumstances are present.
- According to 10 CFR 50.12(a)(2)(ii), special circumstances are present whenever, “application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.”

Underlying Purpose

- The underlying purpose of GDCs 55, 56, and 57 is to provide containment isolation capability that supports the safety function of containment to provide a barrier to the uncontrolled release of radioactivity to the environment.
- The underlying purpose of 50.34(f)(2)(xiv)(E) is to limit radiological releases by ensuring containment isolation for systems that provide paths to the environs.

Staff's Review

- For GDC 55, 56 and 57, the staff finds that the NuScale design accomplishes the containment safety function by providing a containment isolation capability comparable to that required by the GDCs and therefore, the underlying purpose of the GDCs is met.
- The staff finds NuScale's exemption request meets the requirements for an exemption as defined in 10 CFR 50.12.

Staff's Review (Cont'd)

- For 50.34(f)(2)(xiv)(E), the staff finds that the NuScale design isolates systems that provide a path to the environs before core damage or degradation occurs, preventing significant releases from the containment, and therefore, the underlying purpose of the rule is met.
- The staff finds NuScale's exemption request meets the requirements for an exemption as defined in 10 CFR 50.12.

Staff's Evaluation of Section 6.2.5, "Combustible Gas Control in Containment"

Anne-Marie Grady
Reactor Systems Engineer, NRO

Regulatory Basis

- GDC 41, 42, 43 – Containment atmosphere cleanup (and inspection and testing), requires control of hydrogen or oxygen that may be released into containment following postulated accidents to ensure containment integrity is maintained and supporting SSC safety functions can be accomplished

- 10 CFR 50.44(c)(2) – requires that the plant accommodate hydrogen generation up to 100 percent fuel clad-coolant reaction while limiting containment hydrogen to less than 10 percent and maintain containment structural integrity and other accident mitigation features

Key Design Considerations and Features

- The NuScale containment is kept at a very low pressure during normal operation, limiting initial oxygen inventory

- The NuScale containment is designed to accommodate a bounding combustion event resulting from hydrogen generation at 72 hours without a loss of integrity or loss of supporting SSC functions

- NuScale has requested an exemption from 50.44(c)(2), which would require the design provide a hydrogen control system to limit hydrogen concentrations below 10 percent

Exemption Request

- Staff review focused on:
  - Hydrogen conditions in the CNV during the postulated accident
  - Demonstration of adequate containment mixing
  - Equipment survivability in the event of the postulated hydrogen combustion
- NuScale analyses demonstrated the containment would survive a bounding combustion event inside the containment at 72 hours
- Staff confirms the calculation for a limiting pressure pulse inside the containment resulting from a hydrogen combustion event at 72 hours for a very short duration, and containment integrity would be maintained.
- Staff recommends granting the Exemption Request
- Providing a system to control the hydrogen concentration is not necessary to serve the underlying purpose of 10 CFR 50.44(c)(2), which is to prevent a loss of containment integrity

Open item

- To fulfill the requirement to provide hydrogen and oxygen monitoring that is functional and reliable, NuScale uses the process sampling system.
- The system is isolated on receipt of a containment isolation signal (would occur post-accident), but the applicant states that CNV isolation valves for CES and CFDS could be opened in the event of a need for sampling once containment pressure is low enough.
- Separate exemption request (post-accident sampling) raised questions related to this configuration.
- RAI 9682 has been issued to determine that this configuration could be safely established in the context of a beyond-DBA in order to meet the requirements associated with monitoring for 10 CFR 50.44(c)(4). This is **Open Item 6.2.5-1**.

Staff's Evaluation of Section 6.2.6, "Containment Leakage Testing"

Anne-Marie Grady
Reactor Systems Engineer, NRO

Exemption Request Regarding Containment Leakage Testing

- NuScale requested an exemption from the following regulations:
- 10 CFR 50, App. A, GDC 52, Capability for Containment Leakage Rate Testing, requires the capability to perform containment periodic integrated leakage rate testing (ILRT) (Type A) at containment design pressure.
- 10 CFR 50, App. J—Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, requires Type A tests preoperationally and periodically. thereafter.
- NuScale asserts that the CNV ASME design, analysis for leak tightness, 100% vessel inspectability, and pre service design pressure hydrostatic leakage test would satisfy the underlying goal of demonstrating CNV allowable leakage.

Exemption Request Regarding Containment Leakage Testing (Cont'd)

- NuScale CNV is an ASME Section III, Class 1 pressure vessel
- CNV leakage analysis, design specifications, capability for 100% vessel inspection, examination, and testing, provide assurance that the leakage integrity of the CNV is maintained.
- NuScale analyzed the CNV bolt design for the flanged openings, using ANSYS, based on the seal design and specification.
- NuScale calculated flange contact pressures and corresponding flange gaps, based on CNV internal accident pressure and temperatures.
- Staff reviewed these calculations during an audit.

Exemption Request Regarding Containment Leakage Testing (Cont'd)

- NuScale asserts that the CNV design combined with the Types B and C test results are sufficiently representative of accident conditions to demonstrate that the TS leak rate, La, would not be exceeded.
- In addition to testing required by ASME, NuScale proposes a preservice design pressure test to confirm the expected performance of the CNV design. This would be verified by ITAAC.
- The staff recommends approving the exemption request to not require Type A testing nor to require design capability for ILRT as required by GDC 52.

Staff's Evaluation of Section 6.3, "Emergency Core Cooling System"

Shanlai Lu

Senior Reactor Systems Engineer, NRO

NuScale ECCS Design Features

- 3 Reactor Vent Valves (RVV)
- 2 Reactor Recirculation Valves (RRV)
- Each ECCS valve has its own IAB, trip valve, and trip reset valve
- Containment functions as part of ECCS

Regulatory Basis

- GDC 2 - Design bases for protection against natural phenomena
- GDC 4 - Environmental and dynamic effects design bases
- GDC 5 - Sharing of structures, systems and components
- GDC 17 - Electrical power system requirements
- GDC 27 - Neutron poison addition and appropriate shutdown margin for stuck rods
- GDC 35 - Safety function of transferring heat from the reactor core following LOCA
- GDC 36 - Permit appropriate periodic inspection of important components
- GDC 37 - Permit appropriate periodic pressure and functional testing
- PDC 35 - Similar to GDC 35 without the requirements related to electric power
- 10 CFR 50.46(b)(5) - Long-term cooling
- 10 CFR 52.47(b)(1) - Inspection, Tests and Analyses and Acceptance Criteria
- 10 CFR 52.80(a) - COL application contain the proposed inspection, tests and analyses
- 10 CFR 50.34(f) - Reactor coolant system safety and relief valves

Review Guidance

- NuScale Design Specific Review Standard (DSRS)
  - Section 6.3, “Emergency Core Cooling System”
- Regulatory Guide (RG)
  - NUREG-0737, “Clarification of TMI Action Plan Requirements”

Exemption Requests

1. NuScale requested partial exemptions from certain requirements in 10 CFR 50.46 and 10 CFR Part 50, Appendix K, specifically, from the following requirements:
    a. Swelling and rupture of the cladding and fuel rod thermal parameters
    b. Pump modeling
    c. Refill and reflood heat transfer
    d. BWR specific items

2. GDC 27 requires neutron poison addition via the ECCS. Current NuScale ECCS design does not have the capability to inject additional neutron poison.

- The staff's evaluation of exemption requests are ongoing. Detailed discussions will be provided during Chapter 15 ACRS presentations.

Technical Evaluation

Sub-section Title	Issue Description	Resolution Status
Reactor coolant boundary	ECCS performance testing needed	Open Item 3.9.6-1
Low Temperature Over Pressure Protection (LTOP)	ECCS provides protection up to 750 psig	Closed. Section 5.2
Core Cooling - Inadvertent ECCS Actuation	Contingent on LOCA TR and 15.6.6 review	Open Item 6.3-1
Core Cooling – Secondary System Pipe Break Inside Containment	Contingent on Non-LOCA TR and 15.2.8 review	Open Item 15.0.2-4
Core Cooling – Loss of Coolant Accident	LOCA TR and ECCS Valve Flange Design	Open Item 3.6.2-1 Open Item 15.0.2-2
Core Cooling – Long Term Cooling	Return to power and boron transport. Section 15.0.6	Open Item 15.0.6-5
Shared Systems	Only the reactor pool is shared	Closed

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Technical Evaluation (Cont'd)

Sub-section Title	Issue Description	Resolution Status
Power Requirements	Use of non-safety graded DC system	Open Item 15.0.0.5-1
Instrumentation	Valve position and solenoid power indication in the control room	Closed. Chapter 7.
System Boundary	No direct fluid mass exchange with the reactor pool	Closed
Testing, Inspection and Qualification	Demonstration tests are on going.	Open Item 3.9.6-1
Environmental Requirements	EQ, ASME compliance, Seismic qualification and missile protection	Closed (Section 7.2.2, 3.2, 3.3, 3.6, 3.10 and 9.5)
System Reliability	Failure mode and effects analysis	Open Item 3.9.6-1
Single Failure	IAB single failure assumption	Open Item 15.0.0.5-1
Technical Specifications	ECCS Specific Items	Closed. Chapter 16

Technical Evaluation (Cont’d)

In-vessel Debris Downstream Effects Evaluation

Debris Generation:

Only Reflective Metallic Insulation is used in the containment and there is no chemical buffer. Only latent debris is considered to be present

Debris Transport:

100% of latent debris is assumed to arrive at the core inlet

Fuel Bundle Head Loss:

Only 5.6 g/assembly fiber, 33.4 g/assembly particulate are estimated to arrive at the inlet of the core. Without a pH buffer, conservative amount of chemical precipitates is assumed.

Based on relevant AREVA fuel bundle head loss testing and LOCA deposition model, it is concluded that there would be no adverse impact on long term cooling due to limited amount of latent debris

Water Hammer Evaluation

- During the ECCS actuation, the ECCS actuator lines and trip set valves experience two-phase choke flow and sudden depressurization.
- Staff issued RAI 9469 (Question 31517, ADAMS Accession No. ML18162A351) to better understand the capability of ECCS actuator hydraulic line design, analysis, and tests, and to evaluate the ability of ECCS to withstand potential water hammer during actuation.
- Additional tests are planned by NuScale to assess the likelihood of water hammer phenomenon. The test facility includes a long actuator hydraulic line with several 90 degree bends and pressure sensors.
- This part of the staff's evaluation is being tracked and documented in Chapter 3 (Open Item 03.09.06-1).

Conclusions

- Seven Open Items have been identified by the staff
- The NRC staff and NuScale are actively engaged to develop a path toward resolution for the following three Open Items. More discussions during Chapter 15 presentation.
  - Open Item 03.09.06-1: ECCS demonstration testing
  - Open Item 15.0.0.5-1: SECY paper regarding IAB single failure assumption
  - Open Item 15.0.6-5: Boron dilution during long term cooling

Staff's Evaluation of Section 6.4, "Control Room Habitability"

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Control Room Radiological Habitability

- GDC 19 dose criterion
  - 5 rem TEDE for duration of accident
- Open Item 6.4-1:
  - NuScale requested exemption from control room design criteria in GDC 19 to instead have NuScale principal design criterion (PDC) 19
    - Effectively no change to dose requirement
  - Review is not complete

Control Room Radiological Habitability (Cont'd)

- Dose to control room operators from accidents is analyzed in DCA FSAR Ch. 15
  - Includes direct dose contribution from core damage event
- Open Item 6.4-3:
  - Revisions to dose analyses in DCA FSAR Ch. 15 and referenced accident source term methodology topical report received in late April 2019
  - Review is not complete

Control Room Radiological Habitability (Cont'd)

- Post-accident control room habitability is not a safety related function
  - No operator actions are required or credited to mitigate the consequences of design basis events
  - No post-accident long-term monitoring from the control room is necessary
- Open Item 6.4-2:
  - Staff to evaluate questions on potential operator actions from control room to achieve post-accident monitoring of containment atmosphere hydrogen and oxygen concentration

Control Room Radiological Habitability (Cont'd)

- Credit in dose analyses for control room ventilation and habitability systems that are not engineered safety features:
  - CRVS and CRHS backup systems for each other
    - Both designed to be reliable and capable of operation during accident conditions
      - Assumed accident duration is 30 days
    - Independent and diverse systems
    - Both have automatic initiation with different signals for each system

Control Room Radiological Habitability (Cont'd)

- CRVS and CRHS post-accident operation modeled in 2 dose analysis cases:
  - CRHS operation for 72 hours, then CRVS in supplemental filtration mode for 72 hours through 30 days
  - CRVS operation in supplemental filtration mode for entire 30 days
- Both cases meet dose criteria for all accidents analyzed

Control Room Radiological Habitability (Cont'd)

- Resolution of staff's concerns about apparent reliance on both CRHS and CRVS over the duration of the accident to meet control room requirements
  - NuScale sensitivity analyses met dose criterion
    - CRHS operates for 72 hours only, then CRVS fails
    - Neither CRHS nor CRVS supplemental filtration operate
  - Staff confirmatory assessment of sensitivity analyses shows similar results
  - Staff confirmed NuScale design does not need to rely on CRVS operation after CRHS is exhausted

Review Status/Conclusion

- Review not complete
- Staff cannot yet reach a conclusion whether the control room radiological habitability requirements are met until resolution of Open Items 6.4-1 through 6.4-3

Abbreviations

ACRS	Advisory Committee on Reactor Safeguards
ASME	American Society of Mechanical Engineers
CES	Containment Evacuation System
CFDS	Containment Flooding and Drain System
CFR	Code of Federal Regulations
CNV	Containment Vessel
CRHS	Control Room Habitability System
CRVS	Control Room Ventilation System
COL	Combined License
DBA	Design-Basis Accident
DCA	Design Certification Application
DSRS	Design Specific Review Standard
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GL	Generic Letter
ILRT	Integrated Leakage Rate Testing
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
MSS	Main Steam System
NRC	Nuclear Regulatory Commission
NRO	NRC Office of New Reactors
PDC	Principal Design Criteria
RG	Regulatory Guide
SER	Safety Evaluation Report
SSC	Structures, Systems, and Components
TEDE	Total Effective Dose Equivalent