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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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692ND MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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WEDNESDAY

FEBRUARY 2, 2022

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The Advisory Committee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 8:30 a.m., Joy L. Rempe, Chairman, presiding.

COMMITTEE MEMBERS:

JOY L. REMPE, Chairman

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RONALD G. BALLINGER, Member

VICKI M. BIER, Member

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VESNA B. DIMITRIJEVIC, Member*

GREGORY H. HALNON, Member

JOSE A. MARCH-LEUBA, Member

MATTHEW W. SUNSERI, Chairman

ACRS CONSULTANT:

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

QUYNH NGUYEN

MIKE SNODDERLY

*Present via teleconference

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

8:37 a.m.

CHAIRMAN REMPE: Good morning. This meeting will now come to order. This is the first day of the 692nd meeting of the Advisory Committee on Reactor Safeguards.

I'm Joy Rempe, Chair of the ACRS. Member in attendance are Ron Ballinger, Vicki Bier, Greg Halnon, Walt Kirchner, Jose March-Leuba. We've got an echo, hold on.

(Off record comments.)

CHAIRMAN REMPE: Okay. Jose March-Leuba, Dave Petti and Matt Sunseri. Member Charles Brown will be joining us shortly, he had some traffic issues coming in.

And we expect, also, that Member Vesna Dimitrijevic will be joining us shortly. However, I note we do have a quorum.

Today the Committee is primarily meeting in person with some ACRS Staff, NRC Staff and participants attending virtually.

The ACRS was established by the Atomic Energy Act and is governed by the Federal Advisory Committee Act. The ACRS section of the U.S. NRC public website provides information about the history

of this committee and documents, such as our charter, bylaws, federal register notices for meetings, letter reports and transcripts of open portions of our full and subcommittee meetings. Including slides presented at these meetings.

The Committee provides its advise on safety matters to the Commission through its publicly available letter reports.

The federal register notice announced this meeting was published on December 28th, 2021. This announcement provided a meeting agenda, as well as instructions for interested parties to provide written documents or request opportunities to address the Committee. We have received no written comments or requests to make oral statements from members of the public regarding today's session.

A communications channel has been open to allow members of the public to listen. Periodically, the meeting will be open to accept comments from participants listing to our meetings.

Written comments, if not provided previously, may still be forwarded to Mr. Kent Howard, the designated federal officer, for today's meeting. During today's meeting, the Committee will consider the following topics.

North Anna subsequent license renewal application, the NuScale topical report on the building design and analysis methodology for safety related structures. And the proposed rule language for 10 CFR Part 53 Subpart F, related to Staffing, personnel qualifications, training and human factors.

With respect to the NuScale topic, some portions of the session may be closed to discuss information that's proprietary to the licensee and its contractors pursuant to 5 U.S.C. 552b(c)(4).

A transcript of the meeting is being kept. And I request that speakers identify themselves and speak with sufficient clarity and volume so that they can be readily heard. Additionally, participants should mute themselves when not speaking.

At this time I'd like to ask other Members if they have any opening remarks? Seeing none, I'd like to ask Member Sunseri to lead us through our first topic for today.

MEMBER SUNSERI: Thank you, Chair Rempe. Good morning. During this session we will hear from Members of the NRC Staff and Dominion Energy on the subsequent license renewal application for the North Anna Power Station.

Our plant license renewal subcommittee met

with these folks on December 15th, 2021 and conducted a thorough review of the topic. It's been our intention to apply an internal ACRS lessons learned for subsequent license renewal applications to shift the Subcommittee review to the full committee when the application doesn't have any outstanding items or open issues.

This action would have the benefit of reducing Staff, Applicant and ACRS resources on some redundant aspects of the review. This was not the case for this review as we had two unresolved different views at the time of this subcommittee.

These different views have subsequently been dispositioned, and I anticipate that we will hear from Staff on this today. I look forward to the future subsequent license renewal applications that we can apply our lessons learned.

At this time, I'd like to invite Mr. Brian Smith, director of division of new reactor licensing for any comment. Brian, are you there?

MR. SMITH: Yes, sir, I'm here. Thank you, Member Sunseri. And also, thank you, Chairman Rempe and Members of the ACRS.

As you said, my name is I'm Brian Smith, I'm the director of the division of new and renewed

licenses here in NRR.

We sincerely appreciate the opportunity today to present to the ACRS the results of the Staff's review on the fourth application for subsequence license renewal. This application was submitted by Virginia Power & Electric, or Dominion, for the North Anna Power Station, Units 1 and 2 located in Louisa, Virginia.

By way of background. North Anna Units 1 and 2 received approval for their initial license review from the NRC on March 20th, 2003. The NRC review at that time was performed using guidance prior to the issuance of the generic aging lessons learned report, or the GALL report.

The NRC guidance for license renewal over the years has evolved through enhancements and improvements based on the lessons learned from NRC reviews from both domestic and international industry operating experience as well.

The GALL report has gone through two revisions. An additional interim Staff guidance was issued following Revision 2.

The guidance for subsequent license renewal is contained in the GALL SLR report. This was built from previous guidance and included additional

focus and enhancements were necessary on aging management and time limiting aging analyses for the operation in the 60 to 80 year period.

In the Staff's presentation today, you will hear about some of the specific SLR issues as applied to the North Anna review, and those requested by the Subcommittee. Including the two differing professional reviews.

The NRC project manager for the North Anna subsequent license renewal application review is Lois James. Lois will introduce the Staff who will be presenting or addressing the questions regarding the Staff's review.

Part of the management team here with me today is Lauren Gibson, the chief of the license renewal project branch, and branch chiefs for Staff involved in the technical review, including Joe Colaccino and Steve Bloom.

We also have with us representatives from Region II, Stewart Bailey, chief of the reactor project Branch IV, and the division of reactor projects Region II. And Jen England, senior resident inspector at North Anna.

I'd like to note that the Staff completed its review with no confirmatory or open items in the

safety evaluation report. The Staff will provide an overview of its safety review and highlight a few technical areas that may be of interest to the ACRS.

Finally, we will address any questions you may have on the Staff's presentation. We look forward to a productive discussion today with the ACRS.

At this time I'd like to turn the presentation over to Mr. Paul Phelps, Dominion engineering director for SLR to introduce his team and commence the presentation.

MR. PHELPS: Thank you, Brian. Good morning. My name is Paul Phelps and I am the engineering director responsible for the North Anna Power Station subsequent license renewal, or SLR project.

We appreciate the opportunity to speak with the ACRS full committee today on Dominion Energy's application for subsequent license renewal. This is a very important day and we appreciate the support and look forward to presenting the SLR application highlights to the committee.

By way of my background, I have been in the nuclear industry for 30 years. I am responsible for various SLR related projects that are currently under development in Virginia.

We have stood up an organization, not only to perform the requisite work for the re-licensing, for re-licensing the station, but we also have a larger organization that is currently working on projects to improve the safety, reliability and aging management for North Anna Power Station through various modifications.

I want to take the time to introduce the team assembled to present the North Anna SLR application. Paul Aitken is the engineering manager responsible for the development of the North Anna SLR application. Paul also led the team for the Surry subsequent license renewal project.

Keith Miller is one of the team's project managers. He has served in different roles within the SLR team and is currently the project manager for aging management programs.

Along with the team I introduced we have senior station leaders on the virtual call as well. I would like to recognize Lisa Hilbert, who is the new site vice president at North Anna Power Station.

In addition, we also have technical Staff available in the virtual audience, or in the room with me, should we need some assistance on any questions you may have during our portion of the presentation.

If needed, they will identify themselves and address your questions.

Next slide. I want to cover the agenda for today's meeting. We will discuss the station overview performance, SLR application development, SLR aging management programs, technical topics and closing remarks.

Next slide please. Here is an overview of the station and the 50 mile radius. North Anna Power Station is located in Louisa County Virginia adjacent to Lake Anna. Lake Anna was constructed to serve the needs of the station by damming the North Anna River.

The area includes both populated and industrialized areas, as well as expansive rural areas. And spans from Northern Virginia to the suburbs south of our State Capitol, Richmond, and from the upper Chesapeake Bay to the area west of Charlottesville.

Included in this area are many military installations and airports providing international travel.

Next slide please. North Anna is a Westinghouse three-loop pressurized water reactor within an output net capacity of over 1,900 megawatts. Together these two units are capable of producing

approximately 15 percent of Virginia's electricity needs.

Unit 1 started commercial operational in 1978 and Unit 2 started commercial operation in 1980. The independent spent fuel storage installation facility recently had the site-specific licensed renewed in 2018.

A 4.3 percent power uprate was implemented in 1986, prior to the initial license renewal. The renewed licenses for both Surry North Anna Power Stations were issued in March of 2003. Lastly, North Anna entered the period of extended operation in 2018 and 2020 for Units 1 and 2 respectively.

Next slide please. Here is some high level information on the performance of North Anna. To note, North Anna operates on a 18-month refueling frequency. The plant capacity factor has been good, as reflected in the bullets above.

As far as the regulatory oversight process, North Anna is in column one and has been there since 2013. Additionally, ROP indicator status has been all green since 2000 for both Unit 1, in 2010 for Unit 2.

Next slide please. There has been significant capital investments made to North Anna

since the first renewed license was issued in 2003. As I mentioned in my opening remarks, Dominion Energy will continue to invest in North Anna to maintain safety and plant reliability for the current and subsequent period of extended operation.

Here is the partial list of some of the major projects that have been completed at North Anna. I would like to highlight a few.

North Anna has performed a reactor vessel head replacement on both units, replaced the reserved station service transformers, which supply power to the emergency buses, replace the main transformers on both units, replace the main generators on both units, and perform reactor vessel upflow conversion on both units.

At this time I will transition the presentation to Paul Aitken to provide an overview of the SLR application development process.

MR. AITKEN: Next slide. Thanks, Paul. And good morning to everyone. We're on Slide 7 now.

I'll be providing a quick overview of the SLR application development process and other considerations for the ACRS Committee today.

Surry and North Anna are considered sister plants, based on the similarities and design and

operation. I believe that the similarities between North Anna and Surry allowed for a more efficient review by the NRC Staff since many of the aging management programs and supporting materials were either the same or very similar.

For those three programs we are proactive and implemented as many enhancements as we could for North Anna that were identified during the Surry application preparation and review. For every procedure enhancement that we are able to implement during North Anna application submittal, we increased our alignment with the GALL, SLR and decreased the number of enhancements the station would need to implement upon issuance of the subsequent licenses.

In addition, the same Dominion project team identified and examined any lessons learned from the Surry review. Including audit questions and REIs and incorporated those improvements into the North Anna application as appropriate. Together this resulted in reducing the overall number of REIs.

And lastly, through the issuance of the four interim Staff guidance documents prior to the North Anna submittal, the number of aging management programs, with exceptions, were reduced by nearly half in North Anna. Keith Miller will be providing some

additional context for the Committee following my presentation.

Next slide please. I'll provide a brief summary on the differences between the first license renewal and subsequent license renewal for the Committee. For scoping and screening there are minimal changes in the overall process approach. This is primarily because of the established industry criteria hasn't changed very much from first license renewal.

One area that we expect it to have adjustments was related to scoping and screening for Alpha 2. That's non-safety, which can affect safety related equipment.

This was due to the criterion guidance evolving since first license renewal, as Brian mentioned in his opening comments. North Anna, as well as Surry, were pre-GALL plants so we were in the same situation of updating the methodology and scoping of plant additional equipment.

In the area of aging management reviews, the expansion and the number of aging effects we had to address significantly increased due to the vintage of the previous application and the evolution of the GALL over the years. During the aging management

review, our alignment with the GALL was 99.7 percent with the use of industry footnotes alpha through echo for the aging management reviews.

The biggest difference with an aging management programs. Currently for first license renewal we have 25 aging management programs. Moving into subsequent license renewal there will be 48 aging management programs.

In terms of aging management program enhancements, these will be tracked within the Dominion Energy commitment tracking system following issuance of the renewed licenses.

Some commitment items have already been addressed and Dominion Energy will ensure their proper resources are in place to implement the commitments on or ahead of schedule.

At this time, I'll hand the next portion of the presentation over to Keith Miller to discuss aging management programs and technical topics.

MR. MILLER: Thank you, Paul. Operating experience was reviewed for a ten year period to inform the aging management programs.

In addition to operating experience, recent license REIs associated with the Surry, Turkey Point and Peach Bottom SLR projects, as well as recent

first license renewal projects, were reviewed for insights. Our project team also participated in the Turkey Point, Peach Bottom and Oconee industry peer reviews to provide AMP insights and share constructive comments.

Prior to submittal of application effectiveness of aging management activities was assessed using evaluation elements identified in NEI 14-12, which provides industry guidance on AMP effective missed reviews.

All first license renewal aging management activities were continued and incorporated into SLR AMPs. None were discontinued.

Next slide please. This slide gives you a picture of the AMPs consistency with the GALL SLR. Looking at the lefthand column there are 41 existing AMPs that resulted from combination subdivision process of the first license renewals AMPs.

The SLR existing AMPs are augmented by seven new AMPs. The remainder of the column is to provide some perspectives on our consistency with the GALL SLR.

You can see there that approximately half of the 48 SLR AMPs are consistent with GALL about enhancement. About a third of the AMPs are consistent

with enhancement. Three AMPs are consistent with exception and four are consistent with exceptions and enhancements.

So this in particular was a major improvement from Surry. We went from 12 AMPs with exceptions for Surry down to seven AMPs with exceptions for North Anna.

On the next slide I'll provide some context on the effectiveness of the first license renewal AMPs. First license renewals AMPs have been, and will continue to be, assessed for AMP effectiveness.

AMP reviews, including in NEI 14-12, an AMP effectiveness review confirmed implementation of the first license renewal commitments and performed assessments of inspection schedules, inspection results and trending data. Any identified gaps were addressed, were included in the corrective action program.

Program owners receive periodic training and are required to complete AMP effectiveness reviews every five years, as well as perform systematic operating experience reviews on an ongoing basis to inform the aging management programs and augment AMP effectiveness.

And just as an indication of regulatory acceptability of the Dominion Energy aging management programs, the IP 71003 Phase two inspection identified no findings or concerns in the fourth quarter of 2017.

Next slide please. In the subcommittee meeting, we presented in some detail how Dominion addressed four technical topics reflected on this slide related to concrete and containment degradation, reactor vessel internals, reactor vessel support steel and reactor vessel embrittlement.

To summarize, we have developed our various aging management programs to be consistent with the GALL SLR guidance for each of these technical topics. There has been no loss of license renewal intended functions due to concrete aging since entering the period of extended operation.

Dominion Energy has proactively addressed alkaloid sulfur reaction aging by implementing the EPRI alkali silica reaction inspection guidance. As we discussed in the subcommittee meeting, ASR has not been identified at North Anna based on inspections to date. With the exception of precast concrete poles that support overhead electrical circuits from the reserve station service transformers to the turbine building.

There is a design change currently being implemented that either replaces or refurbishes the precast concrete poles. These poles were fabricated offsite and their concrete is not representative of the concrete used in other structures.

Next, the concrete, the containment concrete biological shield wall, gamma and neutron irradiation remains conservatively below GALL SLR radiation exposure levels throughout the subsequent period of extended operation. Also, recent examinations of the concrete, of the containment liner to concrete slab interface in 2016 did not identify any degradation.

North Anna will manage the reactor vessel internals consistent with MRP 227, Rev 1 Alpha inspection and evaluation guidance that was issued in December 2019 with its NRC safety evaluation for the first period of extended operation. The reactor vessel internals program also incorporates recent NRC interim Staff guidance for updated aging management criteria for reactor vessel internal components in PWRs.

In addition, examinations for ten SLR reactor vessel internal components are also incorporated into the PWR vessel internals program,

consistent with MRP 2018-22 interim guidance, MRP 2019-9 interim guidance and WCAP 17451, Rev 2 guidance.

For reactor vessel support steel, Dominion Energy determined that peak stresses for design basis loads associated with the Units 1 and 2 reactor vessel support assemblies are below the critical stress limits for calculated through wall and/or surface flaws based on projected fracture toughness through the subsequent period of extended operation.

And lastly, we have developed fluence projections for 80 years. Satisfactorily evaluated reactor vessel material properties for 80 years. And will perform surveillance capsule testing for each unit prior to, and during, the subsequent period of operation.

The applicability of existing heat up and cool down curves can be extended to 72 effective full power years based upon using updated material property data and application of the K1c methodology is currently included in the ASME code.

I will now turn the presentation back over to Paul Aitken for closing remarks.

MR. AITKEN: Okay, thanks, Keith. I first want to commend the NRC Staff on their efforts over

the last couple of years. The Staff has worked very hard in reviewing the North Anna SLR application in conducting the various public meetings with provided the appropriate forum for stakeholder involvement.

I want to emphasize that Dominion Energy has been engaged in integrative of the work leading up to the GALL SLR issuance through our highly experienced team. We have been heavily invested, along with others in the industry, over the last couple of years, to ensure we have the appropriate guidance and have explored areas for optimization with the NRC Staff.

Dominion Energy has developed a high quality application for North Anna that benefit from the insights and lessons learned gained from the Surry application that I mentioned earlier.

As Paul Phelps mentioned in the opening, Dominion Energy will continue to invest in North Anna now and into the future to ensure the continued safe and reliable operation for 80 years. This ends our prepared remarks and I would like to express our appreciation to the ACRS Committee for this opportunity.

MEMBER SUNSERI: Thank you, Paul. I would just comment, it's pretty remarkable that you've

entered the first extended period of operation and had several inspections with no findings. How do you attribute that success?

What are you doing, at least in your opinion, that's maintaining the plant so well and establishing such a high record with the inspection?

MR. AITKEN: I would say that starts at the top. With Lisa and Fred Mladen, her predecessor, setting the expectations and the tone on implementation and execution.

And then through the involvement of the SLR team coming in and looking at the work that was done for first license, because we turn over a lot of stones, as you know.

MEMBER SUNSERI: Right.

MR. AITKEN: So, we didn't have a lot of, you know, we found some things, but when identified we brought it to the attention of the station and things were addressed. So the station is highly reactive and was willing to listen and act. And I think that was, really set the tone for a successful project.

MEMBER SUNSERI: Now that's good. At least in my mind it goes beyond just this subsequent license renewal. It's the care in which you take care of the facility asset. And it's probably every

employee's engagement in some way so. I mean, it's a real confidence builder from my perspective.

Let me turn to the other members. Any members have any comments or other questions for Dominion?

MEMBER HALNON: I have a couple of questions.

MEMBER SUNSERI: Go ahead, Greg.

MEMBER HALNON: On Slide 8, this is Greg Halnon by the way. On Slide 8 you said that the first license renewal only had 25 AMPs and the second license renewal 48.

And on Slide 10 you said that there is 41 existing programs. Could you just explain the difference between those numbers for me?

MR. MILLER: Yes. This is Keith Miller from Dominion again. So, there were 25 programs associated with first license renewal.

Some of those programs were subdivided into other programs and subsequent license renewal space. So an example, I can give a couple examples.

There were, the chemistry aging management activity in first license renewal space was subdivided into water chemistry and subsequent license renewal, as well as lube oil and one other AMP as well. So

while there were 25 programs on paper for first license renewal, many of those same activities are being continued in subsequent license renewal space.

Now, there were seven new programs in subsequent license renewal space that we really didn't have a corollary for in first license renewal. Four of those were related to electrical programs and three were mechanical.

MEMBER HALNON: Okay. So it's only seven new programs were written, the other, say 25 or 41, whatever you want to call it, they were existing programs that you're already tracking --

MR. MILLER: Correct.

MEMBER HALNON: -- information. Okay. So you'll have that much data already in an established database.

The second question, since subsequent license renewal period of operation is down the road a ways, what are your top concerns relative to equipment issues that you might be looking at 20 years down the road here?

MR. MILLER: Looking into the future, based on the inspections and condition monitoring that we're doing, I can't say anything jumps out. I think selective leaching in the industry is something that

we're dealing with. In particular, in buried pipes. So I would say that would be in the area that's probably the one that jumps to the top.

And as far as the internals, I think that's working its way through EPRI and the NRC. I think even Rev 2 has been provided to the NRC with additional guidance for 80 years of operations. And that will be reviewed. So we feel good about that. We've had good inspection results at both Surry and North Anna.

So, I guess to answer your question simply would be the selective leaching and buried pipe.

MEMBER HALNON: Okay. So how, you probably could, well, you got a cathodic protection program, how is that, how is the data in the cathodic protection looking?

MR. MILLER: So we --

(Simultaneous speaking.)

MR. MILLER: Yes, I'm sorry, I didn't mean to cut you off.

MEMBER HALNON: No, go ahead. Go ahead. You got it.

MR. MILLER: Yes, we do have a cathodic protection system. We do have a couple of subsystems that have been not meeting expectations, and there are

enhancements in the application to get those restored to working condition.

MEMBER HALNON: Okay. And then I assume those obviously will be in the corrective action program?

MR. MILLER: That's correct.

MEMBER HALNON: On the radars of the NRC.

MR. MILLER: It is definitely on the radar. Right.

MEMBER HALNON: Thank you.

MEMBER SUNSERI: Any other Members? Yes, Dr. Steve Schultz, our consultant, has a question. Go ahead, Steve.

DR. SCHULTZ: This is Steve Schultz. Paul, I think the question is for you. You mentioned that the program managers have a responsibility to assess their programs and report on a five year interval. Is that done altogether or is each program responsible for their own schedule associated with that?

And if it's the latter, how is that reporting coordinated within the plant?

I know there's a lot of crosstalk between the program managers, but can you describe that in more detail?

My second question is, you mentioned in the presentation that the major inspection associated with the original license renewal was done in 2017. How are you coordinating with the NRC on what you would expect to be, this may be a question for the Staff, but how you would coordinate your program manager's assessments and reports with the NRC inspections, what you would anticipate in the near future associated with the current license renewal?

MR. MILLER: This is Keith Miller again. On the first question related to AMP effectiveness reviews, historically the station has done those all at once for all the programs. So it would be one concerted effort to do those, to do those reviews at once rather than kind of staggering them on a program basis.

Administratively it's a lot easier to do it that way. I think typically they actually track it through the corrective action program so the initiated condition report, and then individual actions for each of the program owners to do the reviews.

And then as far as future NRC interaction inspection I would probably turn to the NRC Staff on that. All those, the NRC inspections are coordinated through the licensing group at the station.

But I think that would be really, I guess, spearheaded by the NRC Staff as opposed to the station initiating it. But maybe you could, maybe I'm not understanding the question.

DR. SCHULTZ: That's a fine answer to the question. I'll wait for the Staff to respond to that.

When was your last report published, associated with the current license review? Or is this a new program, the five year commitment that you have within the station?

MR. MILLER: The five year commitment has been existing. It's procedurally driven. It's, as stated in the plant procedure, these reviews are done every five years.

Looking back, I think that the last one was done in 2019 time frame. Or 2020. It's been relatively recent. But I would have to go back to get the exact year for you on that.

DR. SCHULTZ: That's fine. Thank you.

MR. AITKEN: Can I --

DR. SCHULTZ: And as you, go ahead.

MR. AITKEN: Yes, this is Paul. Just to add on to what Keith was saying.

So yes, we were cut off so I couldn't answer your question, Keith jumped in. So yes, what

we try to do is all of these at one time. And we use the guidance, you heard us reference the NEI document 14-12.

And that was an industry initiative that was put forward back as we were starting subsequent license renewal to address the elements of an effectiveness review. What should be some of the objectives that should be considered.

So as we start off, we setup the objectives for all the aging management programs, for everybody to looking for the same types of things. And that's all correlated at the end and assessed.

So as Keith said, I think there was, yes, maybe 2019. Our last one was done. And you find administrative things in a procedure that doesn't have this or that. And we put that in a corrective action program and get that tidied up.

But there was nothing significant that came out of it. And I think that was just reinforced by the region inspection.

Because Unit 2 entered the period of extended operation in August of 2020 and ironically we submitted just a couple of days later. So we went ahead and did that work, just to give us the confidence that as we proceeded through the

application submittal and the review that we had confidence in the aging management program implementation from first license renewal.

DR. SCHULTZ: So you've had your corrective action program associated with the last review. If you take a look at what was proposed as corrective action from that review, I presume at this point they're relatively all completed?

What's your record on --

MR. AITKEN: Yes, generally --

DR. SCHULTZ: -- resolving those comments? Those corrective actions?

MR. AITKEN: Yes. I don't, I can't speak with hundred percent confidence, but just through the track record at North Anna they're quick to resolve.

Because a lot of the things were just administrative, as I mentioned. So they report in, they're tracked by management and things are driven to closure. So I have a high degree of confidence. If not all, most are actually completed by this time.

DR. SCHULTZ: Thank you.

MR. AITKEN: Yes, sir.

MEMBER SUNSERI: Thank you. Now at this time we will transition to the NRC Staff. Brian, I'll ask you to queue up your team.

MR. SMITH: All right.

MS. JAMES: Good morning.

MR. SMITH: Lois, go ahead.

MS. JAMES: Good morning, Chairman Rempe, and Members of the ACRS. My name is Lois James and I am one of the senior licensed renewal project managers in the office of nuclear reactor regulations, or NRR.

We are here today to discuss the Staff safety review of the North Anna Power Station, or North Anna, subsequent license renewal review application, or SLRA, and documented in the safety evaluation report, or SER.

Joining me today are On Yee, technical reviewer for the license renewal review, Steve Bloom, chief of the corrosion and steam generator branch in NRR, and Jen England, North Anna senior resident inspector. Also joining us today are members of the technical and regional Staff, including Lauren Gibson, chief of the license renewal projects branch, Angie Buford, chief of the vessels and internals branch, and Joe Colaccino, chief of the structural civil and geotech engineering branch.

I went one too far. We will begin today's presentation with an overview of the North Anna licensing history before moving on to the North Anna

aging management programs. We will then discuss a few technical areas that we believe are of interest to the ACRS, and hear from Region II on the inspections and plant material conditions before sharing the Staff's conclusion of the safety review.

The North Anna SLRA, oops. North Anna Units 1 and 2 were initially licensed in April 1978 and August 1980, respectively. In May 2001 the Applicant, Virginia Electric Power Company, or Dominion, submitted the initial license renewal application.

The initial license, renewed licenses were issued on March 2003 extending the expiration date to April 2038 and August 2040 for Units 1 and 2, respectively.

On August 24th, 2020 Dominion submitted an SLRA for North Anna Units 1 and 2. The application was accepted for review on October 15th, 2020. And the final safety evaluation report was issued on January 3rd, 2022 with no open or confirmatory items.

The North Anna SLRA described a total of 48 AMPs. Seven new and 41 existing. This slide describes the Applicant's original disposition of these AMPs, as stated in the SLRA in the left column. The final disposition, as documented in the SER, is

the right column.

All of the AMPs were evaluated for consistency with a generic aging lessons learned subsequent license renewal review, or GALL SLR report. As a result of its review, the Staff documented one Staff identified exception in the neutron fluence monitoring program in the SER. This accounts for the difference in the existing programs. One program moved from consistent to consistent with enhancements and/or exceptions.

Based on the review, the results of the audit, and additional information provided by the Applicant, the Staff concluded that the Applicant's aging management program activities, and results, were consistent with the criteria of the standard review plan and requirements of 10 CFR Part 54.

Since the final SRA was issued, or SER was issued with no open or confirmatory items, we will highlight some specific areas of the review we thought would be of interest to the ACRS. The first two topics are technical areas of general interests. The last topic is a plant-specific item that was identified during the Staff's review and received two nonconcurrences.

I will turn the presentation over to On

Yee to discuss the first two topics, then Steve Bloom will discuss the last. On.

MR. YEE: Good morning. My name is On Yee. This next topic on RPV embrittlement has two aspects.

The first is related to the adequacy of the Applicant's reactor vessel material surveillance program, which is addressed on this slide, and the second is related to the disposition of the vessel embrittlement time limited aging analyses for the 80 year operating period. Which will be addressed on the next slide.

This slide identifies the peak reactor vessel fluence levels for each unit on the left, which are in access of seven times ten to the 19th neutrons per centimeter squared on the vessel inter-diameter surface.

The Applicant has identified the withdrawal and testing of one capsule at each unit to assess vessel embrittlement at the 80 year fluence levels. As indicated, testing of these capsules will occur within the next five years.

On the right side of the slide we plotted the completed surveillance testing and the plant testing for North Anna Unit 1. The graph for Unit 2

is nearly identical to this.

The axes on these graphs are calendar years on the x-axis and neutron fluence on the y-axis. The blue dash lines represent the 60 year vessel fluence and the end of the current 60 year license. The solid green lines represent the same quantities for 80 years. The solid black data points provide the neutron fluence and the test date for the prior tested capsules.

The orange symbols represent the Applicant's change in testing of a capsule to obtain the 80 year fluence. In this case, the Applicant will test the capsule located at a higher neutron flux location and will acquire data for a 80 year fluence level earlier than originally planned.

Next slide please. For the TLAs, time limiting aging analyses, certain RPV materials at Unit 2 were projected to exhibited upper shelf energies at or below the 50 foot pound limit required in Appendix G, to 10 CFR Part 50.

As a result, the Applicant performed equivalent margin analyses for these RPV materials, as well as additional RPV materials, out of conservatism for both units. Regarding pressurized thermal shock, the Applicant projected its PTS evaluations to the end

of the subsequent period of extended operation and demonstrated all of the RPV materials meet the screening criteria in 10 CFR 50.61.

The Applicant has demonstrated upper shelf energy and pressurized thermal shock TLAs have been projected to the end of the subsequent period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

Next slide please. This topic is related to radiation of reactor vessel internals. There is no Staff approved topical report or generic methodology for aging management of PWR vessel internals for 80 years of operation.

The GALL SLR report, as updated by recent interim Staff guidance in 2021, finds it acceptable for the Applicants to use the Staff approved guidance for six years, and MRP 227 Revision 1-A in combination with the plant specific gap analysis to identify appropriate changes to the inspection program for 80 years.

For the assessment of neutron fluence projections for the vessel internal components at 72 effective full power years, the Staff reviewed the fluence projection methodology and exposures to the internals as part of its review of the aging

management program and the Applicant's gap analysis.

From its audited information, the Staff verified that the neutron fluence methodology and the calculated fluence values for the PWR vessel internals are acceptable. The Staff also verified that component specific fluence exposures for 72-EFPY were adequately addressed in the Applicant's gap analysis and aging management program.

The Staff determined the PWR vessel internals programs is consistent with the GALL SLR report, as modified by the recent interim Staff guidance and adequately addresses inspection of PWR internals during the subsequent period of extended operation.

I will now turn the presentation over to Steve Bloom. Next slide please.

MR. BLOOM: Good morning. My name is Steven Bloom. The plant-specific buried and underground piping and tanks aging management program addresses the buried gray cast iron piping in the fire protection system. This piping had six ruptures prior to 2003, with the root caused identified as cracking initiated due to manufacturing defects due to cyclic loading that occurred during pump start testing.

The plant changed the test procedures to

limit the pressure transient in the downstream piping, and no ruptures have occurred since 2003. Multiple inadvertent pipe starts have occurred since 2003 within out any additional ruptures occurring.

The 14-12 iron piping material is also subject to loss of material due to selective leaching. And the material was identified in Dominion documents as brittle from impact testing.

Next slide please. The Applicant provided two enhancements to the buried and underground piping and tanks program directed towards management of selective leaching in Enhancement 5, and also cracking due to cyclic loading in Enhancement 6.

For these enhancements, the Applicant committed to excavate six gray cast iron locations of each unit, in each ten year operating period prior to entering the subsequent period of extended operation. So, a total of three periods under which they would be excavated.

Five of these excavations will be ten foot blanks of piping. The remaining excavation will be either a pipe or component location, for example, a hydrant location, with a focus on identification of selective leaching in that fifth location.

For the piping locations, the Applicant

will use visual and magnetic particle testing to inspect for cracking on both the inside diameter and outside diameter of the piping. If cracks are identified, the Applicant will perform a radiographic non-destructive examination on these areas.

In addition, if no pipings are identified, a one foot axial piece will be removed and de-structurally examined for loss of material due to selective leaching. If cracking is identified, then a one foot section will be selected and destructively tested to identify cracking due to cyclic loading, and also inspected for loss of material due to cyclic loading.

If the cause is from manufacturing flaws and not age related degradation mechanism, then the results from these inspections will be documented. If the cyclic loading cracking is identified as due to aging, then the Applicant will perform additional evaluations that will cover through the end of subsequent period of operation.

If these evaluations project a loss of function prior to the end of subsequent license, a period of extended operation, this finding will be entered in the Applicant's corrective action program for an evaluation of extent of condition and extent of

cause and to identify any needed follow-on actions.

The Staff found these enhancements to be adequate to find reasonable assurance that the piping will perform its intended function during the period of extended operation.

Next slide please. At this point I would like to note that we did receive two nonconcurrences during the review. Management has addressed both nonconcurrences and the resolution of the issues are identified in the final safety evaluation package in ADAMS.

In addressing the first nonconcurrence, additional justification was added to the Staff's findings in this section of the SER. These changes were included in the draft SER.

In evaluating the second nonconcurrence, management agreed that the FSAR supplement did not provide, did not sufficiently match the program description that Dominion provided in the supplements and previous REI responses. The Staff issued an REI to Dominion to update the FSAR supplement. Dominion provided its response and the Staff updated the discussion in the final SER.

I will now turn over the presentation to Jen England, who will discuss the regional inspection

activities and plant material conditions. Next slide please.

MS. ENGLAND: Good morning. As mentioned, my name is Jennifer England. I am the Acting Senior Resident Inspector at North Anna. With me is Kenya Carrington, the North Anna Resident Inspector, and Stewart Bailey, our Branch Chief.

We are here to provide Region II's review and assessment of the implementation of the Aging Management Programs, Material Condition, and Overall Regulatory Assessment of North Anna Units 1 and 2.

The License Renewal Inspection Program and the Reactor Oversight Baseline Inspection Program are both used to inspect aging management activities at North Anna.

I will start with the activities performed under the License Renewal Inspection Program, then discuss the ROP inspections, and follow up with the material condition of the plant.

In order to assess the adequacy of the license renewal program for the initial period of extended operation Inspection Procedure 71003 recommends using a four-phase approach to license renewal inspections.

This slide lists the specific license

renewal inspections that have been performed at North Anna. The Phase 1 and 2 inspections were performed for both units on the dates listed with no findings identified.

Phase 3 inspections were not required because no findings were identified in Phase 1 or 2. Finally, the Phase 4 inspections, which typically occur five to ten years into the period of extended operation, have not yet been performed at North Anna as the period of extended operation began in 2018 for Unit 1 and 2020 for Unit 2.

Next slide, please. In addition to the inspections mandated by the License Renewal Inspection Program, inspectors use several ROP baseline inspection procedures to evaluate the implementation of aging management activities.

One example is the baseline inspection of the In-Service Inspection Program. This inspection is performed each refueling outage and provides the inspectors the opportunity to review and assess inspections credited for aging management.

The second example is the heat sink inspection, which provides the inspectors an opportunity to review the service water system, including heat exchangers, the service water intake

structure, and both above ground and buried or inaccessible piping and components.

All of these activities are within the scope of license renewal. Additionally, the Design Basis Assurance Inspection, or DBAI, includes a review of the aging management activities for the safety structure systems and components selected.

At North Anna the regional inspectors have found no violations or findings of greater than green significance as a result of the inspections performed using these procedures.

Also of note, the triennial fire protection procedure has recently updated to review aging management of this equipment.

Next slide, please. Currently, North Anna Units 1 and 2 are in the licensee response column and have all green findings and performance indicators.

This indicates that the licensee has been able to effectively identify conditions adverse to quality and correct them in a timely manner.

We reviewed all inspection findings and trends over the last ten years to gain insight related to aging components. We have identified two findings with an aging management aspect.

In 2013 the NRC issued a finding for

failure to replace capacitors in the turbine control system. Specifically, Capacitor C in the speed amplifier card failed leading to a reactor trip.

The cause of the failure was due to aged degradation of the capacitor. Since the time of this failure this control system has been replaced with a digital system for both units.

In 2021 the NRC identified a green finding in a non-cited violation for inadequate instructions for handling of aged cables. The procedure did not identify the allowable bend radius for battery cabling.

If a cable is bent with a small radius the cable insulation can crack over time. The licensee has corrective actions to address the radius of cables.

Next slide, please. I will now speak to the material condition of North Anna from the resident inspector viewpoint. We have no concerns with the overall material condition of the plant that need to be addressed outside the baseline ROP.

The license has been successful at completing large capital improvement projects that maintain or improve the material condition of its structure systems and components.

The License Renewal Program inspections did not identify any substantial weaknesses in the station's performance in managing the effects of aging at the site.

The inspectors will continue to inspect and assess the licensee's ability to manage the aging through the baseline inspections. Are there any questions?

MEMBER SUNSERI: This is Matt. I have one for you, Jen. You know, this is I guess directly related to your experience being on the site and being the person that gets to see the equipment performance and how it's being maintained and everything through direct observation, which a lot of us don't get that opportunity, right, we just see it remotely or read about it in reports.

So regarding the fire protection system piping and, you know, there were different views on how to address that, from your observation of how that equipment is being maintained and what is expected to be done with it going forward do you see those as effective measures for managing the aging of that fire protection piping?

MS. ENGLAND: Yes, I believe it is effective. Our technical experts from the regional

office have also looked into this issue and we will continue to monitor it.

I don't know if anyone else would like to add anything from our side.

MEMBER SUNSERI: It's been a number of years since there has actually been a failure, I think 2003, something like that.

MS. ENGLAND: That's correct. And there has been several inadvertent starts which have not resulted in any additional failures.

MEMBER SUNSERI: Okay. All right. Thank you.

VICE CHAIRMAN KIRCHNER: Matt?

MEMBER SUNSERI: Yes, Walt?

VICE CHAIRMAN KIRCHNER: Yes, this is Walt Kirchner. Just following on, looking ahead, not North Anna per se, but is the Staff seeing this as a common problem throughout the license renewals and subsequent license renewals for the plants?

It seems to me, if I memory is correct, we saw this on Surry. I am returning to the buried gray cast iron piping. Does that suggest that some kind of guidance going forward for new plants or reworking existing plant fire mains and such should be looking at different materials?

MR. BAILEY: So this is Stewart Bailey. I was wondering if I could ask NRR to chime in on that. There is a broader industry experience related to failure of these systems, but I don't know exactly what the Agency is doing about it at this time.

MEMBER BALLINGER: This is Ron Ballinger. I'll just make a comment. Gray cast iron pipe, buried piping, is not just a problem in the nuclear industry, it's a problem everywhere.

If you go to the American Water Works website you'll discover that. Piping, gray cast iron piping with selective leaching, which is hard to determine where it is, 80 to 100 years before you start getting failures.

And so what has happened here is they have just accelerated the problem by pump starts, and so I think the inspection program that they have identified is as good as it's going to get.

Will there be random failures going forward? Yes. Well, you can't 100 percent inspect everything and sooner or later they're going to have a failure but it's not for lack of them trying to get a handle on it.

But what Walt's saying is people are replacing the pipe. But replacing pipes at plants

where they are buried in concrete in some cases is a really tough problem to do. It's as good as they're going to get.

MR. SMITH: Hi. This is Brian Smith. Thanks for asking the question and raising the issue. We recently decided to start an update to the GALL SOR report, a revision to that similar to what we did for just the regular GALL report some years back.

As part of that we had planned to do a lessons learned. So this can be one of the items that we'll focus on as we go forward and in developing the updated guidance. So, thank you.

MEMBER BALLINGER: Yes. There are EPRI programs that are ongoing right now to develop techniques for detecting selective leaching as a function of chemistry and the like.

It's a tough -- You know, you see it, you know it's there, but to decide whether a material will be susceptible is a little bit trickier. So there are EPRI programs that are ongoing in this area as well.

MEMBER SUNSERI: Okay. Very good. Do we have any other questions for the Resident Inspector before we move on?

MEMBER HALNON: Yes, I have a couple.

MEMBER SUNSERI: Okay. Go ahead, Greg.

MEMBER HALNON: So, Jen, we talked earlier with the Staff, the North Anna Staff, about the protection program. Is it on your radar screen? Have you been looking at that since they've said they felt themselves that they had not met expectations?

MS. ENGLAND: Yes. We are always focused on service water piping and any time we can get access into a manhole, this is just something that all residents spend their time on. Yes, so we are definitely focused on that.

MEMBER HALNON: Okay. Many plants have cable vaults outside in the yard that often get flooded and have to be pumped out.

Is there any concern there with the, I hesitate to say climate change, but with the increased weather activity that we are having on the east coast relative to their program and how they make sure that there is not a wetted cable issue that's out in the yard?

MS. ENGLAND: Yes. Like I said, every time you have the ability to access a manhole, this is very typical for us to go in there and look at it for things such as wetted cables.

We also do look at it annually as part of our external flooding procedures. I do not -- I can't

speak to climate change, but from all the things that I have looked at, so their last two inspections of manholes and things of that nature, I don't see a significant negative trend, but, obviously, we are always looking at it and we are always asking questions.

We're inspectors so we're never actually satisfied.

MEMBER HALNON: Okay. So I guess I'll ask, you can just make a judgment, do they have a robust program to make sure that those manholes are pumped out and dried out after weather?

MS. ENGLAND: Yes. So it looks like any manhole that -- Every manhole is inspected on a 3-year basis. Any manhole that any water is found in, more than 12 inches, they will install a sump pump from what I can tell, which appears to be pretty robust in my opinion.

MEMBER HALNON: Okay. Thank you.

MEMBER BROWN: Hi. This is Charlie Brown. I had one question I guess, and maybe I should have brought it up at the Subcommittee but I didn't.

As a result of the earthquake some years ago, obviously buried piping is susceptible to damage, possibly even damage that's not evident immediately or

even a few years later, was there any effort to try to assess whether an enhanced inspection program should be applied to this buried cast iron piping as opposed to just the routine?

I don't know if that's the Staff or Dominion to answer that, but that was the only question I had.

MS. ENGLAND: I do not have the answer to that. I don't know if someone else would like to comment.

MEMBER BROWN: Is there a reason that there wouldn't be any need to do that? I'm not a stressed, you know, piping expert, but it was pretty -- I felt it all the way up at my house. My house was rattling and shaking.

MEMBER SUNSERI: Yes. I don't know where I've heard this, but I know that North Anna went through a very extensive review, like a top to bottom review.

MEMBER BROWN: Well I remember that, yes, for the, I mean after the earthquake.

MEMBER SUNSERI: But I don't know specifically about the pipes.

MEMBER BROWN: I was just looking at downstream.

MEMBER SUNSERI: Yes.

MEMBER BROWN: You know, you might not find anything today but is there something incipient in --

(Simultaneous speaking.)

MEMBER SUNSERI: Maybe we could ask Dominion Energy if they have a response to that question. Paul, if you're still on?

MEMBER BROWN: You're still there, Dominion?

MR. PHELPS: Sorry. We have to come off mute here. So, yes, we had, you know, we're going back in time here so it's difficult, but there was obviously a confirmatory action whether it was a series of activities and actions and I just don't recall.

All I can maybe state is the fact that since the earthquake, ten years, 11 years now we're at, we haven't had any issues. So I think if there was something that may have caused some disturbance to the buried pipe, as Jen said, in some of the autostarts that we've had and the normal testing that we would have done, we probably would have seen something by now.

So that might not be the best answer, but

that's probably all I can offer at this point.

MEMBER BROWN: Okay. I just wondered if somebody assessed any. We're looking at another 20 years, and so that's why I asked the question. If that's what we got, that's what we got. Thank you.

MR. PHELPS: Yes, sir.

MEMBER SUNSERI: I would imagine that the inspection program that they have if it would see, you know, other things from that if it existed.

MEMBER BROWN: Yes, I would hope so. I just -- It just dinged my brain when I was looking through it again, do we need to do any, do we need to think about it a little harder going out for another 20 years recognizing we didn't see anything in the past ten.

MEMBER SUNSERI: Yes. But that --

(Simultaneous speaking.)

MR. PHELPS: So, Charlie, this is Paul. Yes, I'm sorry. Yes, one of my tech staff said that we've done six or seven fire protection buried pipe inspections in the teen years and we have not seen any issues, so I just want to offer that up to the Committee.

MEMBER BROWN: All right. No, that's a reasonable approach to seeing if you had anything in

the near future with whatever is coming up, so thank you.

MEMBER SUNSERI: Any other questions for the Resident before we move on?

(No audible response.)

MEMBER SUNSERI: All right. Steve Schultz?

DR. SCHULTZ: This is Steve Schultz. Jen, a question for you, or a comment first. When you present the slides about the results of inspections stating no findings, no findings, no findings, I think it's important to understand how detailed your inspections, in fact, have been and how well the documentation of those inspections has been completed.

Very thorough reports of very intense inspections, so that result is a good one to see here. The other is a question, and following on to my comments to Dominion, you've focused on the fact that the ROP baseline inspections and your other resident activities really do focus continuously on the aging management programs.

But would it be worthwhile to coordinate a particular license renewal related inspection following the licensee's approach to do their five year evaluation and reporting and then evaluate the

corrective action program that would result from that and its effectiveness?

MS. ENGLAND: We get to select the samples that we look at on a periodic basis and, yes, I agree that would probably be a very good sample to take. Does that answer your question?

DR. SCHULTZ: Yes. Thank you.

MS. JAMES: This is Lois James. If I could add, Jen, if I am not mistaken when we do our Phase 4 that is going to look at effectiveness which will particularly prompt us to look at how they have looked at their effectiveness.

So I think that, you know, it will definitely caught, more than likely get caught in the resident review, but also get caught when we do the Phase 4.

DR. SCHULTZ: Thank you for the additional comment, Lois.

MS. CARRINGTON: This is Kenya Carrington, the Resident. I will also add that as the licensee has been going through their subsequent license renewal we have engaged with the Staff and Headquarters just to try to gain a better understanding on our part and additional insights as we go out into the field and perform our inspection

and vice versa.

DR. SCHULTZ: I am glad to hear that. The licensee talked a lot about their training program. Well not a lot, but they mentioned, and I know they do a lot of work associated with the training of the program managers for the aging management program, so I am glad to hear that is happening on the Staff's side also. Thank you.

MEMBER SUNSERI: Thank you all. Jen and Kenya, we appreciate the fact that you are there onsite and obviously we have a lot of interest in your perspectives on things, so thank you for your comments today.

I will turn it back to you to either continue with your part or pass it on to the next person.

MS. ENGLAND: All right. Thank you. I would like to hand the presentation back over to Lois James.

MS. JAMES: Thank you, Jen. In conclusion, for the SLRA safety review the Staff finds that the requirements of 10 CFR 54.29(a) have been met for the subsequent license renewal of the North Anna Power Station's Units 1 and 2.

We will now be happy to answer any

additional questions you may have. Thank you.

MEMBER SUNSERI: Thank you, Lois. Members, any other questions for Staff, I guess or Applicant at this point since we're wrapping it up?

(No audible response.)

MEMBER SUNSERI: All right. So no questions for this. We'll go the public line I guess for comments at this point in time. So, members of the public, this is your opportunity to provide a comment.

Unmute your line by star 6 and provide your name and comment. So the line now is open. Let's see if -- Well, any -- We'll give it one more or a few more seconds.

(No audible response.)

MEMBER SUNSERI: All right. Then I guess, Chair Rempe, this concludes our part of the presentation here, so we can head back to you.

CHAIRMAN REMPE: Thank you. I appreciate you for doing this. I know we had a late start getting it done on time.

It's my understanding you have a letter that you are ready to read in?

MEMBER SUNSERI: Yes. Based on the review and the fact that (audio interference) information

since that time and by the presentation today we do have a letter report that, draft letter report, that was prepared to send to the Board.

CHAIRMAN REMPE: Great.

MEMBER SUNSERI: It's ready to go.

CHAIRMAN REMPE: Okay. So at this point I think we can go off the record and ask the court reporter to please come back at 1:00 p.m. eastern time and we'll go back on the record.

(Whereupon, the above-entitled matter went off the record at 9:50 a.m. and resumed at 1:00 p.m.)

CHAIRMAN REMPE: Okay, my computer tells me it's 1 o'clock, so we're going to come back into session.

And at this point, I would like to invite Member Sunseri, to make a statement.

MEMBER SUNSERI: Thank you, Chair Rempe.

I had previously announced during the subcommittee review of the North Ana subsequent license renewal, that I was recusing myself from metal environmental fatigue, and reactor vessel embrittlement issues, due to professional affiliation outside of the ACRS.

I intended to make that same statement prior to the start of full committee. I just

overlooked it in my notes. But I have been abiding by the recusal.

In addition to that, for the same professional affiliation reason, I will be recusing myself from the complete review of the NuScale topical report on buildings design analysis methodology for safety related structures.

Thank you.

CHAIRMAN REMPE: And, I'm going to do a sound check because of some concerns about certain members not being audible.

Could you hear that, court reporter, what Member Sunseri said?

(No audible response.)

CHAIRMAN REMPE: Great.

So, at this time, I'd like to ask Member Kirchner to lead us through the discussion on NuScale topical report on the building design and analysis methodology for safety related structures.

Walt?

VICE CHAIRMAN KIRCHNER: Thank you, Madam Chairman. Good afternoon everyone.

This afternoon we will hear from representatives of NuScale, and the NRC staff, on the NuScale topical report, the building design and

analysis for safety related structures.

We had the benefit of detailed presentations by both parties, at our NuScale subcommittee meeting on January 19.

Today, we will hear summary presentations on the matter, and I have a draft letter report to read into the record, for the Committee's consideration and deliberation, after the presentations.

With that, I am going to turn to Fehmida Mesania from NuScale, to start us off this afternoon.

Fehmida, go ahead.

DR. MESANIA: Thank you. Good morning, good afternoon. Can everyone hear me okay?

CHAIRMAN REMPE: Yes, we can.

DR. MESANIA: Thank you.

So, please, would you mind going to the next slide, please, number two?

Thank you.

So, again, my name is Fehmida Mesania. I'm a licensing engineer with NuScale. We thank you for the opportunity to present NuScale topical report on the building design and analysis methodology for safety related structures.

For today's meeting, my colleagues Rim

Nayal and Giulio Flores, will support all presentations.

Slide three, please.

Thanks.

So, the proposed agenda for today's presentation includes a purpose, the brief introduction, followed by a technical discussion on the following subjects.

The building design of steel composite walls in reinforced concrete members; and, design methodology, as it's related to in-structure response spectra and effective stiffness.

Slide four, please.

Thanks.

The purpose of this meeting is present the ACRS scope committee, the technical content of the topical report as outlined by the agenda.

And, also provide the public a general understanding of the NuScale methodology.

There are several considerations that are unique to the steel composite walls, and our focus is to highlight the methodology for building design for SC1 and SC2 safety related structures, related to RC and SC walls that are applicable to the NuScale SMR design.

It is worth mentioning that the design applicability and constructability (audio interference) are not part of the scope of this topical report, but are being considered and investigated as part of our SDA submittal.

Slide five, please.

Thanks.

So, plate reduction. So, this topical report presents a design methodology implementing new industry standards for nuclear facilities.

The methodologies presented in our topical report are in compliance with the requirement of several standards and codes, including NC918, 360-16, both recently endorsed by R.G. 1.2043. But we are also using 349.13, and 360-16.

For the NuScale methodology, is consistent with the structure acceptance criteria presented in the Nuclear Scale Design specifically deal with standards, the EDSRS, sections 384 and 372.

By meeting this requirement and criteria, the methodologies provided an added assurance that the SSEs will perform their safety function.

In addition, the topical report clarifies methodologies for the interaction of SC walls, with traditionally constructed, reinforced concrete members

such as facemasks, slabs and rules.

This methodology is intended, but is not required, to be used in conjunction with another NuScale topical report related to improvements in frequency domain soil structure fluid interaction analysis.

The NuScale plans to apply the methodology described here, to our SCA submittal where design details will be provided.

Are there any questions or comments so far?

(No audible response.)

So, hearing none, I would like to thank you for your attention, and I will hand it over to Rim Nayal, to continue our technical presentation.

DR. NAYAL: Good morning, and good afternoon.

This is Rim Nayal, I'm structural engineer with NuScale.

In the next few slides, I'll present building design methodology for steel plates composite walls.

Next slide, please.

So, this slide presents, provides a general description of SC walls. SC walls are

fabricated from steel plates, referred to as face plates, with a concrete core between the plates.

As shown in the sketch here, stud anchors are welded to the faceplate at certain spacing in both directions, to develop composite actions between the steel and concrete.

The steel plates are also connected to each other with ties at specific spacing, to resist the out-of-plane sheer loading.

Next slide, please.

This slide presents a discussion and advantages associated with using SC walls. And, it highlights some of the areas requiring special attention when utilizing them in construction.

SC walls are known to offer higher loading capacity and ultimate strength, compared to other types of construction.

And, SC walls also provide higher resistance to blast and earthquake.

Major advantages of SC walls is modularity, where SC walls are fabricated offsite, and transported to the field for installation. Which helps save time and reduce cost.

These modules are sized taking into account multi aspect of the design, including

limitation on dimensions and weight for transportation.

Another advantage of modular construction is that the design can be optimized to have limited number of conflagrations, in terms of layout that are repeated throughout the building.

Some areas requiring special design or treatment in SC walls construction are discussed here. First one is SC wall connection to other structural members.

Design and specifically, construction of these connections, can become challenging given potential steel congestion and exclusivity (phonetic) issues.

The second item is corrosion of face plates for below grade walls. These two areas will be discussed later in this presentation.

Next slide, please.

So, as far as construction material, carbon steel is mainly used everywhere in the design, except for face plates in contact with borated water. For these modules, stainless steel face plates are utilized.

For modules used in the full area connecting the stainless steel face plates to other

carbon steel components, involves performing the similar metal welds.

Welding requirements associated with the use of the similar welds are outside the scope of this topical report, and will be addressed in the future.

However, as in general, here are some actions that help with the concern with the similar metal welds.

Like utilizing industry experience on welding parameters and techniques that are appropriate for the welded metals. In addition, controlling the environment of these welds can help alleviate this issue.

Controlling the environment may be like controlling concrete chemistry to avoid corrosion of these similar welds, and also the use of leak chases, and the module splices to prevent borated water leaks from reaching to these dissimilar welds, metal welds.

Next slide, please.

As far as design of ASC walls, the walls are designed to the rules of NAEIC N690, 2018 edition, and AIC 360-16.

Design is based on load and resistance force design. Loading combinations are based on AIC's N690, and the dimensionals and material properties

limits from N690 are applied, which applies to wall thickness for interior and exterior walls, limits on minimum and maximum faceplate thickness, stud and ties minimum and maximum spacings, and et cetera.

The required strength is calculated by dividing walls into panels, based on code panel size limits for interior and for connection regions.

Five element responses are then averaged over these panels, to obtain demand.

Available strength for each type of response and load interaction, are calculated based on code equations.

And, based on that, demand to capacity ratio is for governing load cases, are calculated to ensure acceptable design.

As a result of the global design, faceplate thicknesses are determined along with anchor and tie, tie dimensions, and spacings.

Next slide, please.

So, back to the subject of corrosion of face plates for below grade walls.

The topical report discusses NuScale's defense in depth approach for corrosion mitigation, of the carbon steel face plates based on environments.

This can vary from applying coating as a

minimum on the face plates below grades, to applying controlled load strength material, or shotcrete, for environments with high chloride, or hydrogen sulfide.

Or even using backfill with controlled pH and chloride limits to reduce impact from corrosion, corrosive soil.

Specific corrosion and aging management plans for NuScale design, are outside the scope of this topical report, and will be addressed in the future.

Next slide, please.

This slide provides a brief discussion on design of connections between SC walls, and other structural components like basement, reinforced concrete slabs, and other SC walls.

The connections are designed to the rules of AIC N690. The code's philosophy is to ensure that these connections are stronger than the weaker of the connected members.

This ensures ductile behavior of the structure.

For calculating the connection required strength, full-strength connection design is used for most connections. That is, the connection is designed to 125 percent of the nominal strength of the weaker

of the connected members.

Use of over strength connection is limited to cases where the connected members have low demand-to-capacity ratio.

Calculation of the available strength of the connection is in compliance with the requirements of N690, AIC 360-16, and ACI 349.13.

Connection detailing can be achieved using different types of connectors and mechanisms, to transfer forces and moments between the connected members. Like using studs, anchor rods, continuity plates, et cetera.

Detailing of the connections for the NuScale design is also outside the scope of the topical, this topical report, and it will be addressed in the future.

In general, examples from the design guide 32 shows some connection configuration, that can give some idea about the design.

This concludes my presentation about SC walls, and we can take any question right now, or otherwise, we can continue to the next section of this presentation, which is about reinforced concrete.

MEMBER HALNON: Yes, I had one question, this is Greg Halnon.

When you say that these are corrosion management and details are outside the scope, how will the, I mean will this topical report provide acceptance criteria, or some kind of criteria, so that when we do see those other topical reports, we will know the connection back to this topical report?

I guess my question is more, how are we going to know if you address it in future ones, or just letting it go now?

DR. NAYAL: So, I mean first of all, the design of these connections is still under development. And, course it's going to be presented in the future when it's needed, before the final design is you know, is completed.

VICE CHAIRMAN KIRCHNER: I was actually more interested in corrosion management. I mean I'm assuming you're assuming there's no corrosion? In your account.

DR. NAYAL: Could you repeat the question?

VICE CHAIRMAN KIRCHNER: I'm more interested in the corrosion management, the previous slide, where you had said that that was outside the scope and it will be in future application.

DR. NAYAL: Right.

So, right now, I mean there are so many

options that can be done for that, to mitigate the corrosion, or to take some actions to alleviate that concern.

Right now in the, like the detailed, this is mainly design and design methodology topical report. So, we think that dealing with the corrosion and, you know, and aging management, that will be addressed somewhere else.

VICE CHAIRMAN KIRCHNER: I guess it's that somewhere else --

(Simultaneous speaking.)

DR. NAYAL: Yes --

VICE CHAIRMAN KIRCHNER: -- will be some --

DR. NAYAL: -- it will be site specific. And, it will be communicated at the time an application is submitted.

VICE CHAIRMAN KIRCHNER: Okay, so I guess we'll just have to remember that it's a hanging topic.

Thanks.

DR. NAYAL: Thank you.

Okay, so next slide, please.

Next section of this presentation is to discuss the methodology for reinforced concrete members in building design.

Next slide, please.

Basement, all other slabs, and roof slabs in the NuScale design, are constructed of reinforced concrete. The design methodology for reinforced concrete is based on the requirements of ACI 349.13, and ACI 318.08, for each structural element, as applicable.

Design is based also on load and resistance for design. Loading combinations are based on ACI 349 load combinations, and core dimensional limits as applicable, are applied. Like thickness, spacings between reinforcements, and many other details.

The required strength is based on section cuts at different locations of the member, and available strength for each type of loading and interaction are calculated based on code equations.

Demand-to-capacity ratio governing failure modes are then calculated to reflect design margins.

The next few slides will go in a little bit more detail over the design.

Next slide, please.

So, on this slide we're focusing on major actions that we look at when designing slabs in basement.

This includes designing for in-plane shear, and evaluating interactions between axial force with out-of-plane moment, axial force with in-plane moment, and axial force with out-of-plane shear.

The table shows what checks are typically looked at for different numbers. For example, while in-plane moment is not important for basement, it has to be looked at for the slabs.

Next slide, please.

So, for each member, the demand is calculated from section cuts taken at different locations of the slab.

The topical report provides guidance on where to take these section cuts for different types of structural elements, and in addition to that, we look at the contour plot of the stresses, to confirm the critical locations of these section cuts.

And if needed, additional section cuts are considered to obtain the demand.

So, for example, on the left side, the sketch shows the vertical deformation of the slab. This deformation generates out-of-plane moment, and out-of-plane shear forces in the slab.

So, the section cuts recommended are at the locations where these actions are expected to be

critical.

Similarly, the sketch on the right showed the diaphragm action deformation where the slab transferred loads, horizontal loads, to the adjacent supporting walls.

And, based on that, section cuts critical locations are recommended where the design to be checked.

And, as I said, these sketch confirm later when we look at the contour plots of the stresses.

Next slide, please.

So, the capacities of the members are for different design actions, are then calculated based on code equations.

Where it comes to the design, demand of the members is calculated for different loading combinations, and for all time points of the seismic events.

For example, the sketch to the top, here, shows an example of demand for interaction equations. Each one of these blue dots represents member of demand at certain time points of the seismic event.

To perform the design, we envelope that demand of the member and for all of these points, and then we compare it with the capacity curve that's

shown in the bottom of the slides, to decide whether the member has enough capacity or not.

And, then based on that, we calculated the maximum demand-to-capacity ratio and to assess the safety margins, and whether the design is sufficient, or whether it, change is needed.

And, we repeat that for different types of checks, and for different members.

Next slide, please.

Finally, the reinforcement of the member is determined after calculating the needed demand for each action.

The demand is added based on what type of reinforcement is needed to resist that demand. For shear enforcement, the maximum demand of the two directions is used for both, for both of these directions.

This summarizes the slides on reinforced concrete design, and if there are any comments or questions, I'll be happy to take them now.

(No audible response.)

DR. NAYAL: And if not, I'll hand over to Guilio, to go over the next part of the presentation.

(No audible response.)

VICE CHAIRMAN KIRCHNER: Go ahead, Guilio.

(No audible response.)

CHAIRMAN REMPE: So, we're not hearing anyone speaking. So, perhaps your microphone is on mute, Guilio?

MR. SNODDERLY: We did a sound check --

(Simultaneous speaking.)

CHAIRMAN REMPE: Is anyone one from --

MR. SNODDERLY: Yes, we did a sound check with Guilio and it worked. So, maybe he lost his connection.

Liz English or Mark Chitty, could someone give Guilio a call, or see what, what the problem is?

MR. CHITTY: We can.

PARTICIPANT: Yes, we're communicating with him now. Thanks, if you just give us a minute?

MR. SNODDERLY: Very good.

(Pause.)

MR. FLORES: Hello?

PARTICIPANT: We hear you.

(Audio interference.)

CHAIRMAN REMPE: We can hear you but it's not very clear.

MR. FLORES: Okay, so let me call the number.

(Pause.)

CHAIRMAN REMPE: So, I assume the folks from NuScale know that we aren't hearing anything in the room. If someone could confirm that and let us know.

PARTICIPANT: Yes, thank you, we're aware. We're working on it. Appreciate it.

CHAIRMAN REMPE: Thanks, we'll be patient.

PARTICIPANT: Thank you.

MR. NGUYEN: So, the fastest way in troubleshooting is to log off, and then log back in.

MR. FLORES: Hello, can you hear me now?

(Audio interference.)

CHAIRMAN REMPE: Yes, but we have an echo.

MR. SNODDERLY: Can you mute your computer? If you're using your phone and your computer, you need to mute one or the other. And, I suggest your computer.

Guilio, are you there?

(No audible response.)

MR. FLORES: Okay, can you hear me?

MR. SNODDERLY: Yes, you're good now.

MR. FLORES: Okay, yes. Now let me open my computer. I had to close it because it was sounding like echo.

(Pause.)

MR. FLORES: Okay, yes, sorry for the inconvenience.

So, I am Guilio Leon Flores, I am a structural engineer by NuScale Power, and I'm going to present a summary of the topical report sections 4 and 5.

Section 4 describes the methodology to obtain in the structure response spectra for equipment design, and member forces for design of seismic category 1 and 2 structures.

If we go to the next slide, please.

I'm going to start by summarizing some important seismic design concepts used in our design methodology.

Buildings are designed for a specific target performance, or a specific level of deformation and damage, designed by a limit state category.

For nuclear plants, the criteria to achieve a specific target performance come from ASC 43.

Four limit states are defined A, B, C, and D, which range from significant to negligible damage.

NuScale buildings are designed for limit state D. In other words, structures are designed to remain essentially elastic, with negligible damage.

Next slide, please.

(Audio interference.)

MR. FLORES: Sorry.

CHAIRMAN REMPE: There's still an echo.

MR. FLORES: Oh, it's starting, okay.

Okay, how about now?

MEMBER SUNSERI: That's better, Guilio.

MR. FLORES: Okay, sorry.

Okay, yes, this slide.

In agreement with the reinforced concrete design philosophy, concrete is designed to take compression loads, and it is considered to retract under tension loads. Reinforcing the steel on the other hand, is designed to take all tension loads.

For this reason, it is expected that some structure members will experience cracking under the safe shut down, earthquake, or SSE. But this cracking should be minimal since structure design, according to (audio interference), is expected to result in essentially elastic behavior.

And, this is accomplished following design code requirements, such as the use of lower material and member strength, use of load factors to increase the magnitude of design loads.

Due to all of these reasons, concrete

cracking under the SSE, if any, should not impair the member safety function.

Next slide, please.

Under the SSE, the stiffness of cracked members is reduced. On the other hand, the structure damping increases, because the energy dissipated by the structure increases.

The figure shows an example of a cantilever wall subject to a seismic force at the top. The wall stiffness is that slope of the force displacement curve.

So, when the force is small, the member is uncracked, and if the stiffness is uncracked or the gross stiffness.

Above a certain threshold, the member cracked, and the stiffness decreases with increasing force.

The safety of the stiffness, which is a red line in the figure, is an intermediate stiffness that's up by the code, to represent the cracked state for seismic analysis using linear elastic models.

(Audio interference) damping is also taken from the codes, and the standard.

If we go to the next slide.

So, as I explained, damping ratios and

effective stiffness used in linear elastic analysis, are co-related to the level of cracking expected during the SSE.

ASCE 4 introduces a concept of response level to help make this correlation. The response level is determined on a member-by-member basis, based on the member stresses.

So, if the member stress is smaller than the stress associated with cracking, response level 1 is assigned. Otherwise, response level 2 is assigned to the member, and the member is considered cracked.

Next slide, please.

This and the following slides, show our design process. We start with two models, one for seismic analysis, and another one for static analysis considering known seismic loads.

The first step is the determination of the member response level. To this end, seismic models are set to uncracked damping and uncracked stiffness, corresponding to a response level 1 for all members.

Stresses are then evaluated in all the members, considering the most critical seismic load combination. Stiffness and damping of the members that are cracked, are abated to the ones corresponding to a response level 2.

Next slide, please.

Seismic models with abated stiffness and damping are used for, in-structure response spectra calculation.

ISRS are calculated for 535 seismic design respond spectrum ground motion, and then they are averaged.

ISRS are also calculated for one CSDRS high frequency ground motion. Blast manual 15 percent, peak broadening is then applied.

ISRS for notes on specific flow regions are then enveloped. The process is repeated for all selected 4 to 5. And, the final ISRS are the envelope considering all soil profiles.

MR. SNODDERLY: Excuse me, NuScale, I need you to hold for a second.

I just was notified by our executive director in the room, that we lost communications with the room.

So, now they can't hear you, so they're trying to reconnect. So, now we waited for you, now you'll have to wait for us.

But --

(Simultaneous speaking.)

MR. FLORES: Sure.

MR. SNODDERLY: -- so, please be patient.

MR. FLORES: That's fine.

MR. NGUYEN: Thanks, Mike, I'll physically check.

MR. SNODDERLY: Okay, thanks.

(Pause.)

MR. SNODDERLY: Hello, Member Halnon?

(No audible response.)

MR. SNODDERLY: Member Halnon?

(No audible response.)

MR. SNODDERLY: Court reporter, are you there?

(No audible response.)

MR. SNODDERLY: All right, some people, okay, great.

So, we've lost the room so we're, I'm letting some people back in.

Can I ask who just joined us?

(No audible response.)

MR. SNODDERLY: Okay, the ACRS has lost connection. Hang on, they're getting it back.

All right. So, Greg just notified me that they've, okay.

MEMBER HALNON: Yes, we're trying to get it back.

MR. SNODDERLY: Okay.

MEMBER HALNON: Quynh's working on it, so just give us a couple seconds.

MR. SNODDERLY: Okay. Now --

(Simultaneous speaking.)

MEMBER HALNON: You know that patience we had a few minutes ago?

MR. SNODDERLY: Yes.

MEMBER HALNON: We need it the other way now.

MR. SNODDERLY: I already used that line, Greg.

MEMBER HALNON: Yes, right.

MR. SNODDERLY: We're on standby.

(Pause.)

MEMBER HALNON: Should see us any second, Mike.

MR. SNODDERLY: So, I just saw Dave Petty came in. I'm seeing the individual members coming in, but hopefully people in the room can hear us.

CHAIRMAN REMPE: Okay, somebody who's got their mic on, turn it off.

MR. SNODDERLY: That's, well, I have mine on. Do you want me to turn my mic?

CHAIRMAN REMPE: No, I think you're fine.

We had members in the room who were on their mics.

MR. SNODDERLY: Okay.

CHAIRMAN REMPE: Okay, I want to apologize for internet interruptions, and I hope that you can, didn't go too far before we lost the connection.

MR. SNODDERLY: Yes, so right now I believe we're on slide 25, and I'll ask NuScale to start there.

And, then why don't you let us know if that's something you haven't heard before, or that said something.

VICE CHAIRMAN KIRCHNER: We were on 21.

MR. SNODDERLY: 21?

PARTICIPANT: Yes, Mike, we were on 21.

MR. SNODDERLY: Okay, please, NuScale, can you please go to slide 21?

MR. FLORES: Okay, sure.

MR. SNODDERLY: Okay.

MR. FLORES: Yes, this is part of the enforced concrete design philosophy. So, I was saying the concrete is designed to take compression loads, and it is considered to be cracked under tension loads. Reinforcing the steel on the other hand, is designed to take all tension loads.

For this reasons, it is expected that some

structural members will experience cracking under the phase shutdown earthquake.

But this cracking should be minimal since the structural design is according to limit the (audio interference). And, it is expected to result in essentially elastic behavior.

And, this is accomplished following design code requirements, such as the use of lower material and member strengths, use of (audio interference) factors to increase the magnitude of design loads.

Due to all these reasons, concrete cracking under the SSE if any, should not impair the member's safety function.

Next slide, please.

Okay, so the stiffness of cracked members is reduced. On the other hand, structural damping increases because the energy dissipated by the structure increases.

The figure shows an example of a cantilever wall that is subject to a force at the top, and to the right, there is a force displacement incur where the effective stiffness is shown as the red line in the figure, and is an intermediate stiffness set up by the codes to represent the crack states for seismic analysis using linear elastic models.

Structural damping is also set up by the codes, and the standard.

Next slide, please.

As explained before, damping ranges and effective stiffness in linear elastic analysis, are correlated to the level of cracking expected during the SSE.

ASC 4 introduces the concept of response levels to help make this correlation. The response level is determined on a member-by-member basis, based on the member's stress.

So, if the member stress is smaller than the stress associated with cracking, response level 1 is assigned. Otherwise, response level 2 is assigned, and the member is considered cracked.

Next slide, please.

This and the following slides, show our design process. We start with two models, one for seismic analysis, and another one for static analysis considering known seismic loads.

The first step is the determination of the member response level. To this end, seismic models are set to uncracked damping, and uncracked stiffness corresponding to response level 1 for all members.

Stresses are then elevated in all the

members, considering the most critical seismic load combinations.

Stiffness and damping of the members that are cracked, are updated to the ones corresponding to response level 2.

The next slide shows the process for the ISRS calculation, that seismic models with updated stiffness and damping, are used for the ISRS in the structural respondent specter calculation.

ISRS are calculated for 535 seismic design respondent specter ground motions, and then they are averaged.

ISRS are also calculate for one CSDRS high frequency ground motion. Plus/minus 15 percent peak broadening is then applied.

ISRS from notes on the specific flow ratios are then enveloped and the process is repeated for all selected soil profiles.

Final ISRS are the envelopes considering all soil profiles.

The next slide shows the process for the member design in seismic models with updated stiffness.

And, response level 2 damping are used to obtain seismic forces for five CSDRS ground motions,

and one CSDRS high frequency ground motion.

No seismic forces accompanying the seismic loads are obtained from the static model, with the stiffness matching the stiffness of the seismic model.

Demand-to-capacity ratios for the five CSDRS ground motions are averaged, and then enveloped considering all those combinations.

Demand-to-capacity ratios have to be lower than one, otherwise, the member section is increased or additional reinforcement is considered.

And, the process is repeated for all selected soil profiles.

Final demonstration of ratios are the envelopes considering all soil profiles.

Are there any questions?

VICE CHAIRMAN KIRCHNER: Guilio, this is Walt Kirchner. I have one question.

On the ASCE guidance, it's like how much cracking is acceptable? Where I'm going with this is there's cracking, and then there's just complete failure of concrete.

I'm presuming that ASCE set of adjustments, the R1 and the R2, obviously the concrete hasn't turned to rubble inside the steel composite wall. So, how much cracking is acceptable in the

codes?

MR. FLORES: Yes, so can we go to slide number 22, please?

Yes, so the figure, the force displacement code, may help explain this.

So, we are talking about the, really the elastic range, which is the first portion of the curve.

Now, the profile that you see there is a global capacity curve that defines the global capacity of the member.

So, the cracking of course, of the force in the first portion of the curve, which means the crack that may appear on the member, are very tiny. Maybe you may not see it.

And, in order to see the crack but maybe you're worried, you need to go to the curve towards the end. Then you start seeing the big cracks, and then you reach the capacity. But we are far from it. We're just in the beginning of the curve.

And, the concrete, I mean it cracks because it has a very low capacity for tension loads. But those cracks are very minor at this stage. It's not the end of the capacity, it's just the beginning.

I don't know if that's clear?

VICE CHAIRMAN KIRCHNER: Yes, okay. The presumption is that it's relatively small cracking, it's not gross failure of the concrete?

MR. FLORES: Yes, it's not failure at all, no. Failure as I mentioned, you need to go to the end of the curve, and we are not there because the end of the curve is to limit to state A, near collapse.

We are at the beginning, which is limit to state D, essentially elastic.

VICE CHAIRMAN KIRCHNER: Okay, thank you.

MR. FLORES: It shows with minor cracking there.

MEMBER BALLINGER: Remember, this is reinforced also.

VICE CHAIRMAN KIRCHNER: No, of course.

MEMBER BALLINGER: So.

MEMBER HALNON: I've got one quick question.

MR. FLORES: Yes.

MR. HALNON: This is Greg Halnon.

In the next, when you did your iterations, you said selected soil types. And, how broad is that relative to the potential soils that might be, a site might be built on?

MR. FLORES: Yes, it's very broad.

So, we have three soil profiles. One is very soft soil, and the other one is rigid soil, and the third soil is more rigid. It's like crack. It's like rock, sorry, rock.

So, we try to cover the different scenarios now from soft to very rigid soil profiles.

MEMBER HALNON: Okay, so essentially, you've covered pretty much most of the soil types that would be considered for an adequate site for one of these plants?

MR. FLORES: Yes.

MEMBER HALNON: I'm not, I don't want to put words in your mouth, but it sounds like you're confident that you're pretty well bounded.

MR. FLORES: Yes.

So, for the selective soil profiles we evaluated, the soil, different soil profiles across the U.S. And, that was, were used for in previous plant applications, as well.

And, because based on that, we selected those three soil profiles, try to cover the entire spectrum of soil conditions.

MEMBER HALNON: Okay, thanks.

MR. FLORES: You're welcome

If there are no more questions, we can

move to the last section of the topical report.

(No audible response.)

MR. FLORES: This last section describes the modeling approach, to represent effective stiffness in our building models.

Next slide, please.

For the NuScale design, finite element models are developed using ANSI phototropic material model. Steel plate composite walls and reinforced concrete members, are modeled using ANSI's traditional shell elements, and also 3-D solid shell element.

For shell elements, effective stiffness is implemented by adjusting material properties and model, in the model thickness to match the code specified values.

And, this is the same procedure outlined in AIST design guide 32.

For solid shells, since the model thickness cannot be changed without changing the geometry, two alternative methods are implemented.

Both methods are consistent with design guide 32 approach. And, I'll explain in the next slide.

In both methods, the solid shell element is subdivided into three layers. In method one, the

middle layer has thickness and properties to match code specified stiffness. And, the outside layers are dormant layers having negligible stiffness.

This method is more suitable for reinforced concrete members.

In method two, each layer thickness is equal to the actual steel plate, or concrete thickness. Material properties for each layer are chosen to match again, the code specified thickness.

This method is more suitable for SC walls.

Both methods, method one and method two, can be used in the same model. As I explained, they generate a Q and effective stiffness values matching the code.

This summarize our presentation, is the topical report. I don't know if there are more questions?

VICE CHAIRMAN KIRCHNER: Members, any further questions of NuScale?

(No audible response.)

VICE CHAIRMAN KIRCHNER: Okay, thank you very much. I think then we can turn to the NRC staff, and.

MR. FLORES: Oh, there was one slide but it's just a summary. We can go just one minute, yes.

VICE CHAIRMAN KIRCHNER: Okay.

MR. FLORES: So, in summary, the topical report includes the building design and analysis methodology implementing new industry standards, and novel structural systems such as the use of SC walls.

It is applicable to new generation of a small model reactor design, and a portion of this report will be used as part of the standard design approval application.

Thank you.

VICE CHAIRMAN KIRCHNER: Thank you.

I think we can turn to the NRC staff presentation at this point.

MR. SNODDERLY: Yes, Member Kirchner, it will be Demetrius Murray from the staff, to kick things off.

Thank you.

MR. MURRAY: Thank you --

(Simultaneous speaking.)

VICE CHAIRMAN KIRCHNER: Go ahead, Demetrius.

MR. SNODDERLY: NuScale, if I could, yes, thank you, relinquish the slides. Thank you.

MR. MURRAY: Can you see my slides?

MR. SNODDERLY: Yes, they're up. Thank

you.

MR. MURRAY: Thank you.

Okay, good afternoon. I would like to thank the ACRS, NuScale, and the general public, for entertaining the NRC for the presentation of our safety evaluation of NuScale's building design and analysis methodologies for safety related structures topical report.

In December of 2020, NuScale submitted Rev. 0 of their building design topical report, to the NRC. After acceptance of the topical report, the NRC issued multiple requests for additional information, RAIs, to NuScale in May and August 2021.

NuScale provided answers to the NRC's RAIs in June and September of the same year.

NuScale issued Rev. 1 of their topical report to the NRC on October 6 of 2021.

We are here today to further discuss the staff's advanced safety evaluation of the topical report.

The staff reviewers are Ata Istar, Dr. Amit Ghosh, and now retired Robert Pettis. I am the topical report project manager, Demetrius Murray, supported by senior project manager Getachew Tesfaye.

Before we transition to Ata to discuss our

safety evaluation, I would like to open, or give an opportunity for NRC management Michael Dudek, and Joseph Colaccino.

MR. DUDEK: So, just once again, thanks to the Committee for hearing us on this very unique topic.

I think we had a very good and collaborative subcommittee meeting, in which we got a lot of good feedback from the Committee.

We've taken a lot of those comments and concerns into account today, and we've added that as some of our specific talking points in going forward today.

And, I would welcome the Committee if they have any additional questions with what we're presenting. We are open books and we have the appropriate staff members on hand today, to address those concerns.

That's all for me. Joe, would you like to say something before we proceed?

MR. COLACCINO: Thanks, Mike.

All of what Mike said, plus in addition I just want to again, recognize the uniqueness of this review, and how we coordinated with the Office of Research's development and approval of Reg. Guide

1.243.

I believe we have staff members from Research here today, if there are any questions about that Reg. Guide.

Thank you.

MR. ISTAR: Good afternoon, or good morning. Can you guys hear me?

(No audible response.)

MR. ISTAR: Can you hear me?

VICE CHAIRMAN KIRCHNER: Yes, we hear you very well. Go ahead.

MR. ISTAR: Thank you. This is Ata Istar, structural engineer, NRC.

Before I start my presentation, I would like to recognize Bob Pettis, who retired last month and, or I should say two months from now, in December, end of December.

And, he was the lead for the review of this topical report, and he provided immense information, and for that, that he needs to be recognized for.

Based on the review of the TR, the staff concluded that the methodology presented in the NuScale TR, are acceptable to perform the building design and analysis for seismic category 1 and 2

nuclear safety related reinforced concrete, and steel plate composite structures, other than containment.

The methodology follows the implementation of the requirements of ACI 349.13, AIC and 690.18, appendix 9, which is endorsed by Reg. Guide 1.243.

The methodologies are also consistent with the applicable regulatory requirements of acceptance criteria, sections 372 and 384, in NRC NuScale design specific review standard.

Next slide, please.

I would like to provide information on the following items that are questioned during the ACRS subcommittee meeting.

The first one is the effects of corrosion on the ASC wall below grade. The second one is welding attachments to the ASC walls after concrete is placed. The third one is industry experience on the SC wall construction.

Next slide, please.

Based on the review of the TR, the staff requested the information related to corrosion effect on SC walls, as required in section B3.13, of AIC 360--16, which states the structure components shall be designed to tolerate corrosion, or shall be protected against corrosion.

Since the small modular rafter design may also consider the possibility of plant license extensions of up to 80 years, and the in-service inspection and repairs of below grade exterior SC walls base plates would be impractical for the duration of extended licensing period.

NuScale response included the following graded approach, and is described in revision 1 of the topical report.

First, the application of tar epoxy coating on the exterior below grade face plates. The second one, using control low strength material, or employing shotcrete cementitious material on the exterior low grade face plates.

Further, using backfill material with control pH and chloride limits.

Based on the reviews, staff conclude that the NuScale response is consist with section b3.13 of AISC 360-16.

Next slide, please.

VICE CHAIRMAN KIRCHNER: Ata, before you go on, this is Walt Kirchner.

MR. ISTAR: Yes, sir.

VICE CHAIRMAN KIRCHNER: This is an almost the application for the reactor building in

particular, that we saw in the NuScale DCA.

It's almost like a ship at sea, only the water now is inside the ship, not outside.

And, is this a place where cathodic protection could be used to mitigate issues with corrosion below grade?

MR. ISTAR: It could be, but at this point, the applicant is just picked up, picked up the cathodic protection mitigation, you know, mitigation application.

But pick up providing epoxy coating, external on the below grade face plates and the shotcrete using a cementitious material. And, also using a backfill material with control pH, and with chloride limits.

And, again, anything can be used. Once it goes into a site specific, issues are determined. I think numerous protection measures can be used, you know.

It's when --

(Simultaneous speaking.)

MEMBER BALLINGER: As a practical --

MR. ISTAR: Yes?

MEMBER BALLINGER: As a practical matter, cathodic protection really can't be used.

MR. ISTAR: Well.

MEMBER BALLINGER: As a practical matter, cathodic protection is not going to be. Because you've got this pool that's got a stainless steel wall on the inside, carbon steel plate on the outside.

The stainless steel will be a giant cathode. It's air saturated water. And, the outside will be steel, and if it's exposed to any kind of water or anything like that, that's just, you'd have to use the power plant itself to provide enough current to do cathodic protection.

MR. ISTAR: Okay. Sir, let me explain.

I think the pool wall stainless steel part of it, it should be stainless steel on both sides. And, that's the outside surface, so that faceplate I don't think, is exposed to the ground level.

So, there is another wall --

(Simultaneous speaking.)

MEMBER BALLINGER: No, what I meant was the --

MR. ISTAR: there is another wall --

MEMBER BALLINGER: -- the tie rods that go.

MR. ISTAR: There is another wall outside, which is a carbon steel.

MEMBER BALLINGER: Right, but they're

connected with tie rods.

MR. ISTAR: Correct, they are connected with tie rods, correct.

But the pool area, and that is the outside of the SC wall for that location, is, doesn't, outside faceplate is not exposed to the soil.

The applicant can confirm that, but based on my understanding, the outside wall of the reactor building is carbon steel on both sides. And, the outside face of, faceplate, is exposed to the soil.

And, it's very difficult to inspect for the duration of the license, and also repair.

But the cathodic protections could be used on the outside. But not on the stainless steel part of it. I agree with you on that.

Any other questions?

MEMBER BALLINGER: No, that's, okay. Just since we've wagged this issue, when, where would we normally see this taken up? When we get to an actual COL application where you have site specific information, and then these mitigating measures would be implemented?

And, that would be somewhere in the FSAR --

(Simultaneous speaking.)

MR. ISTAR: Correct.

MEMBER BALLINGER: -- or the submittal. Somewhere in the licensing application submittal?

MR. ISTAR: Yes.

MEMBER BALLINGER: Okay.

MR. ISTAR: Correct.

MEMBER BALLINGER: Thank you.

MR. ISTAR: Correct.

Next slide, please.

Based on the review of the TR, the staff also requested information related to the attachment to SC walls, and the effects of elevated temperatures in the concrete due to welding.

NuScale response was the attachment to SC walls are to be installed in the shop during the fabrication, or at the field prior to the concrete placement.

However, when the attachments are needed after the concrete is placed, the effects of elevated temperatures from weldings are minimized by the following generic industry guidelines.

The minimum faceplate thickness is limited to half-inch. The minimum concrete curing prior, period is 21 days. This requirement assures that the concrete has enough strength.

And, heat input to the concrete is controlled by setting a maximum base plate temperature to 25 degree Fahrenheit, above the minimum pre-temperature.

Or the calculations performed provide specific allowance.

Based on the review, the staff concluded that the NuScale's response is consistent with Reg. Guide 1.243.

Next slide, please.

During the ACR subcommittee meeting, there was a question related to industry experience on SC wall construction.

Similar SC wall application was reviewed and approved for AP 1000 shield building. Integrated design approach was implemented on the AP 1000 shield building design by performing the design detailed structural analysis of the SC walls and connections, performing testing of SC walls and connections, and performing inspections.

Finally, AICN 690-18 was recently endorsed by Reg. Guide 1.243, and this specification is acceptable for the design of SC walls and connections, for seismic category 1 and 2 nuclear safety related structures.

This will conclude my presentation. I can take any question you may have.

Thank you.

VICE CHAIRMAN KIRCHNER: Members, any questions at this juncture?

(No audible response.)

VICE CHAIRMAN KIRCHNER: Okay.

So, Ata, does that complete the NRC presentations?

MR. ISTAR: Yes, sir.

VICE CHAIRMAN KIRCHNER: Thank you. Thank you very much for responding to our questions during the subcommittee.

MR. ISTAR: I thank you, too.

VICE CHAIRMAN KIRCHNER: So, Madam Chairman, I think we're at a juncture where we should open the public line for comment.

(No audible response.)

VICE CHAIRMAN KIRCHNER: If there are any members of the public who wish to make a comment, please identify yourself and make your comment.

Press *6 to unmute your phone.

(No audible response.)

VICE CHAIRMAN KIRCHNER: Hearing none, any further comments from NuScale, or the staff?

(No audible response.)

VICE CHAIRMAN KIRCHNER: Hearing none, okay, thank you very much all, and Madam Chairman, I think we're ready to move on to read the letter into the record.

CHAIRMAN REMPE: Okay. Do we want to take a, first of all Tammy, are you out there?

MS. SKOV: I'm right here.

CHAIRMAN REMPE: Great. As always, we can depend on you.

Do we want to take two minutes to take a break and we'll come back -- oh, first of all, we do need to ask for public comments.

VICE CHAIRMAN KIRCHNER: I just did.

CHAIRMAN REMPE: Okay, okay, sorry.

Okay, then let's take a 10 minute break, and we will reconvene at how about we'll give 2020, give you 13 minutes, and we'll read the letter in.

VICE CHAIRMAN KIRCHNER: Thank you.

(Whereupon, the above-entitled matter went off the record at 2:07 p.m. and resumed at 3:45 p.m.)

CHAIRMAN REMPE: Okay, it's 3:45 on the east coast and we are going to reconvene. And at this time I'd like to ask Member Halnon to lead us through our discussion on the proposed rule language for 10

CFR 53 Subpart F relating to staffing, personnel qualifications, training and human factors. Member Halnon.

MEMBER HALNON: Thank you, Chairman Rempe. The purpose of this session at the full committee is to review and discuss the proposed rule language as she mentioned, Subpart F of Part 53. We've had a couple of meaningful subcommittee discussions on the topic and it's been very beneficial and has enabled us to focus this discussion to a few topics that we can dive deeper into. We will also get into the review of the rule language as well.

But I wanted to make sure everybody knew it didn't preclude asking questions on different topics even though we're going to limit the discussion here to -- for different focus topics.

So, staff has been ready. They've got a good presentation coming up. I'm looking forward to it so I'm going to turn it over to Lauren Nist, and you can introduce your team and get started.

MS. NIST: Thank you, sir. Can you hear me okay?

MEMBER HALNON: Yes, we can.

MS. NIST: Okay, so good afternoon. I'm Lauren Nist, the chief of the operator licensing and

human factors branch and my staff's presenting today on the preliminary rule language for Part 53 Subpart F, staffing, personnel qualifications and training. Let's see.

So the presenters today are Jesse Seymour, Maurin Scheetz, and Theresa Buchanan, and they are all in operator licensing and human factors branch in NRR> all the slides are publicly available and in ADAMS at the ML number shown here. Next slide, please.

Although you -- the subcommittee and I think previously the full committee perhaps, you primarily heard from Jesse Seymour on all things related to Part 53 Subpart F before today. And I think he's done a wonderful job presenting the committee with information on our efforts in this area to date.

Jesse is one of the members and leaders of a team of staff who have been working on developing this subpart. And that team consists of the staff members listed here which you can see includes staff from both NRR and the Office of Research. Collectively these staff members have expertise in human factors engineering, nuclear power plant personnel qualification and training processes and methods, and experience in commercial nuclear power

plant operations and training. Next slide, please.

For today's meeting as Member Halnon said we prepared to focus on a couple of different areas based on discussion topics at the earlier subcommittees which are listed here. In addition to an overview of the preliminary rule language we'll also plan to talk about staffing considerations including treatment of the role that has been filled by the shift technical advisor at operating plants and its inclusion as it may be relevant for Part 53 plants. Also the concept of certified versus licensed operators, considerations for simulators, and plans for updating NRC regulatory guidance documents associated with Part 53.

So without further delay I will now turn the presentation over to Jesse. There was one key thing I wanted to say actually which was that we had planned on talking for 30 minutes or less, at least the staff had planned to do that to allow the rest of the time for questions and answers. And we do plan on stopping after talking about the key topics to allow the committee to ask questions and discuss with the staff.

So, now without further delay, Jesse, if you'd please take over. Thank you.

MR. SEYMOUR: Thanks, Lauren. I appreciate it. Liz, if we can move on to the next slide, please. Thank you. Okay. I'm Jesse Seymour and I'll start out the main part of our presentation by providing an overview of the structure and content of the preliminary proposed rule language that our team has developed within Part 53 Subpart F for application during the operating phase facility-wide.

Specifically, we'll be talking about requirements that currently span the 53.750 to .789 sections. And these requirements address the general areas of staffing, personnel qualifications, training, and human factors engineering, and they're divided up into four subsections at present. As we go through this I'll provide an overview of those subsections.

In light of having covered the full scope of these requirements in detail during previous meetings of the subcommittee today I'll be keeping things at a higher level and focusing on highlighting some of the more significant provisions of these preliminary requirements.

To begin with, the 53.750 through .759 section covers general requirements that apply to all operating license and combined operating license applicants and holders under Part 53. Within the

subsection two key sets of requirements are located at 53.753 and at 53.755.

First, 53.753 addresses technical requirements for operating license and combined operating license applicants. These includes human factors engineering design requirements, human-system interface design requirements, concept of operations, functional requirements analyses, and function allocation requirements, staffing plan requirements, and licensed and certified operator programmatic requirements.

Second, 53.755 section addresses certain conditions of licenses for operating license and combined operating license holders, and some key provisions -- and again, we're just highlighting some major areas -- some key provisions that are embedded in that section include provisions for not using licensed operators, and also provisions for load following.

Next, 53.760 through .769 section covers operator licensing requirements including those associated with training, examination, requalification, and simulator requirements. Separately, the 53.770 through .779 section covers operator certification programmatic requirements.

And lastly, 53.780 through .789 section covers general training and qualification requirements for other facility personnel.

We can move on to the next slide, please. So we begin by focusing on some of the key aspects of the first section, the 53.750 through .759 section. This section fulfills a role that's similar to certain aspects of the existing 50.34(f) post-TMI requirements, 50.54 conditions of facility licenses requirements, and the Part 55 operator licensing requirements.

However, a major difference from the current regulatory framework is that the requirements established in areas of human factors engineering, staffing and operator qualification within this preliminary rule language are now linked to design-specific safety functions and their fulfillment.

I'll explain what is meant at a high level here. First, human factors engineering is required or needed to support safety functions versus being generically applied and limited to a control room.

Secondly, operator staffing is required to the extent necessary to support design-specific means for safety function fulfillment versus relying upon a prescribed number of reactor operators and senior

reactor operators. This is an area that Maurin will expand upon shortly.

Third, the fundamental role of the licensed operator now centers around the management and fulfillment of safety functions in addition to manipulation of facility controls. Also included are requirements that must be met in order to justify not using any licensed operators as part of facility staffing. And again, I want to make the distinction that these criteria that we're about to talk about that simply establish whether licensed operators are required for a facility. Meeting these requirements doesn't imply (audio interference) staff. It simply means different requirements change, and we'll talk about how they change.

Facilities meeting these requirements would have the option to instead use non-licensed certified operators in lieu of licensed operators. And we'll delve further into the specifics of certified operators in the upcoming slides. At present, there's two different staff proposals for what these criteria would consist of.

The first proposal is a PRA-centered approach that would require that several criteria could be met in order to meet the overall criteria.

And these criteria include, first, not requiring any human actions for event mitigation as part of meeting safety criteria, achieving safety functions, or providing adequate defense-in-depth.

The second criteria would be that the PRA for the facility would need to demonstrate that the evaluation criteria for each event sequence could be met without human action for mitigation.

And third, the plant response to licensing basis events would need to not be reliant upon human actions to guarantee the performance of safety-related system structures and components. And examples of what we mean by that third criteria are that you would need to be relying upon either things like inherent safety characteristics, or robust passive characteristics, or other safety features that have provisions made to guard against human failures rendering those systems unable to fulfill their function. An example of that would be safeguards against system misalignments and so forth.

The second alternate proposal -- so this would be the second bullet. The second alternate proposal is derived from an integrated safety analysis based set of principles. And that would instead require that the design basis accident safety criteria

of Part 53 be met without mitigation by human actions, active engineering features, or by passive engineering features except for those passive features that have the capability to both survive the licensing basis events while also not being defeated by credible human errors. And again I use the term "passive" there, but built into that is that reliance upon inherent safety factors would also be an acceptable means of meeting that second criteria as well.

So that's a synopsis of key parts of the 53.750 to .759 section. With that being said I'll move into an overview of the other portions. We can go ahead and move on to the next slide.

So, sections 53.760 through .769 address operator licensing requirements for those Part 53 facilities that require licensed operators. I'd like to note that these sections propose a framework that includes an operator licensing pathway that is independent of, or borrows in part from that of Part 55.

Key aspects of the operator licensing requirements include, first, requiring training programs to be based on a systems approach to training, and to ensure that licensed operators possess the knowledge and abilities needed to protect

public health and maintain plant safety functions.

And secondly, incorporating facility developed and NRC approved examination programs that are tailored to the design-specific operator roles.

Section 53.770 through .779 cover the (audio interference) requirements for operator certification programs. At those facilities that are around to use certified operators in lieu of licensed operators. Certified operators are defined under Part 53 as being individuals certified to manipulate facility controls without being licensed by the Commission.

It is important to note that certified operators are not intended to be credited for fulfilling plant safety functions. But contrary, the designs of the facility to qualify to use certified operators would not have a human role in the mitigation of plant events. This is a topic that Theresa will be addressing in detail later on in this presentation. For now though, key aspects of the operator certification requirements include requiring training programs to be based on a systems approach to training, and to ensure that non-licensed certified operators possess the knowledge and abilities needed to protect the public health and perform job duties,

and also using facility developed and NRC approved tailored examination programs for the qualification of those operators.

Lastly, sections 53.780 through .789 cover training and qualification requirements for commercial nuclear plant personnel under Part 53. Section 53.781 in particular covers specific training and qualification requirements, and these -- key aspects of these include requiring training programs to be established with sufficient time to provide trained and qualified personnel to operate the facility prior to fuel load. And this is a flexibility beyond the existing 51.20 requirements.

Requiring the use of a systems approach to training which is consistent with our system requirements, and also requiring the training and qualification of supervisors, technicians, and other appropriate operating personnel to be accounted for. And importantly it should be noted that the categories of personnel described in the preliminary language are in general more generic than those of 51.20 in order to accommodate for greater flexibilities in roles and responsibilities.

So with that being said I'd like to go ahead and hand off to Maurin who will continue on with

the staffing part of the discussion.

MR. SNODDERLY: Thanks, Jesse. Can you hear me?

MR. SEYMOUR: Yes, I can hear you.

MR. SNODDERLY: Great. So, this is Maurin and I'm going to talk about our Part 53 staffing approach, and then the rule language. And then I'll talk a little bit about the shift technical advisor, and we'll have some time for discussion there.

So first as the next slide -- there we go, staffing approach. So, we wanted a staffing rule that works for new and novel concepts of operations, those that we expect to see for the wide variety of advanced reactor technologies, and we want something that allows more flexibility based on design-specific needs.

A key assumption we have is that prescriptive staffing requirements like those that we use for large light water reactors which assign a specific number of operators and senior operators to be onsite in the control room and at the controls. That doesn't make sense for Part 53.

We do not believe that rule-based prescriptive staffing levels would provide reasonable assurance of safe plant operation across different

designs and facility sites. Furthermore, the use of prescriptive staffing would likely force applicants to rely on the exemption process when those staffing levels didn't fit their design's needs. So we don't think that would be an efficient or practical approach to rely on exemptions.

We think that flexible staffing requirements, ones that are risk-informed, performance-based, and technology-inclusive are needed.

We anticipate that there will be designs where operators do not significantly influence safety outcomes. And because of this we thought about staffing needs for two groups of advanced reactor facilities. One group where operators have a safety role, and one group where operators do not have a safety role.

You'll notice that this staffing approach is consistent with our Part 53 approach for human factors as well as licensing or certifying operators, and that's because these areas are interrelated. In fact, the Part 53 requirements in these areas are integrated and supported by requirements for applicants to submit new types of information as part of their application.

Those new types of information that we're going to require are things like a concept of operations document, and a function requirements analysis, and information about function allocation. So for a given facility design this type of information gives the staff a broad understanding of the operator's role in facility safety, thus informing staffing plan needs.

Finally, we relied heavily on our experience that we gained in recent years from evaluating staffing plans as well as looking at how to eliminate the shift technical advisor position for the NuScale small modular reactor design. Next slide, please.

So that was our approach. Now I'm just going to summarize the Part 53 staffing plan requirements.

MEMBER BROWN: Can you go back a slide?

MR. SNODDERLY: Go back a slide.

MEMBER BROWN: This is Charlie Brown. Yes, back to slide 9. What is a novel concept?

MR. SNODDERLY: I think we just mean new. So different -- new and different conduct of operations. The end concept. Sorry, novel -- new or novel. Basically we're saying different than what

we're used to --

(Simultaneous speaking)

MEMBER BROWN: -- is fundamentally what you're saying. Is that right?

MR. SNODDERLY: Say it again? I talked over you.

MEMBER BROWN: No operators would be okay.

MR. SNODDERLY: That would be novel. I think we're just saying we expect different ones. There are so many different designs out there, their concept of operations are going to be varied. We're trying to come up with a way to require staffing for different types of ways to operate the plant.

MEMBER BROWN: Have you given any consideration that there ought to be some, how do I phrase this. Do you have a basis? That's not what I'm trying to say. Just saying novel concept without some idea of where you think a cutoff may be. You don't have any floor the way this is phrased.

MR. SNODDERLY: Well, I mean this is -- I think Part 53 has some broad requirements to come in the door so we're going to look at whatever the concept of operations are for those plants. I think Lauren has her hand up if she wants to try a stab at Charlie's question.

MS. NIST: Thanks, Maurin. I just wanted to say that this is talking about -- this bullet here is the staff's approach to coming up with a performance-based technology-inclusive staffing requirement.

MEMBER BROWN: I don't need the buzzwords. I'm not trying to be critical, but those buzzwords are used for everything. Performance-based, technology neutral, blah, blah, blah. And I'm not criticizing, that's just what's being used for everything as if that's the magic elixir that's going to give us --

MS. NIST: So (audio interference) just in the interest of proceeding on and clarity we're talking about what the staff's effort is and our work product is attempting to accommodate reviews that we might -- approach of how we might review concepts of operation for facilities that basically aren't large light waters. That's the point we're trying to get across with the bullet.

MEMBER BROWN: I understand that, but it seems to me you ought to have some idea of what would be acceptable -- the least acceptable thing you would accept I guess. The minimum thing you may accept. I mean, this is open-ended and to me there ought to be a game plan or some thought process to say what -- do

we really think you can sit these out with having nobody around for 500 miles. And if the answer to that is no then you try to put something into the staffing approach that gets that point across. I mean, should there be a minimum staff as opposed to zero. That's the only point I'm trying to make and this doesn't give the communication to whoever comes in to do a Part 53 that zero is not ever going to be the right answer. That's all I'm saying. This is -- it's totally open-ended.

MEMBER HALNON: Jesse, you wanted to weigh in?

MR. SEYMOUR: Yes, yes, I appreciate that. So, I just want to make a point, a clarifying point here that under the preliminary proposed structure that we're talking about here there is no outcome that ends with zero operations personnel being associated with an operating power reactor. There's always going to be some variation of licensed or certified personnel overseeing the plant operations. So zero is not an acceptable outcome with what we're doing here.

MEMBER BROWN: Is that clear from the writeup? The comment, I like the comment that zero is not the right outcome. Somehow that thought crossed -- this is my personal opinion obviously.

CHAIRMAN REMPE: Excuse me, we're getting a lot of background noise. So if you're not speaking please mute your computers or whatever.

MEMBER BROWN: I'm sorry, Joy.

CHAIRMAN REMPE: I just was going to tell you to go ahead.

MEMBER BROWN: My only reason for making the comment is that somehow this zero is not an acceptable outcome is -- should be communicated somehow as part of the Part 53 language. That's all I'm saying. There's -- that's the reason I made the comment. I've just never done anything without having an idea of where -- what the minimum I would accept in terms of -- it's like looking at the I&C system. I would never accept a system that allows total communication between all four independent, non-independent channels. That's insane. So to me staffing is --

MR. SEYMOUR: So I'm not hearing anything on my end. I'm not sure --

MEMBER HALNON: This is Greg online. We just lost our stuff again here. So if you all would just hang in and let us get our internet back here at the ACRS room we'll be back with you in a minute.

MR. SEYMOUR: Okay, thank you.

MEMBER HALNON: We're still working on it so everybody just stand by. Okay, we're coming back now. I take it back, we're not coming back now. Hold on.

(Simultaneous speaking)

PARTICIPANT: This is one of the things that happens where I don't think I'd ever be able to sit in a driverless car.

CHAIRMAN REMPE: You have no faith in the future, Derek. I think we're back.

MEMBER BROWN: You just made my point Derek.

(Simultaneous speaking)

MEMBER HALNON: Okay, this is Greg. We're going to continue on. Member Bier has a comment to make and then I'll summarize the theme of what we're listening to, and we'll let you guys go on from there.

MEMBER BIER: This is in a way a different version of Charlie's question which is what is the basis for evaluating proposals. So for instance, let's say we decide that X number of operators is acceptable for Y number of reactors of a certain type, right. One operator can handle however many.

So then five years from now they come back and say well, everything has gone really smoothly so

now we want twice that many reactors. And five years later they may come back and say well, everything is going super smoothly so now we want three or four times what you approved initially, and that could keep going until something doesn't work anymore. And is there a process by which there would be a way to say no, we approve this based on these assumptions and that's no longer valid as you increase. You need to up your operators.

MEMBER HALNON: Thanks.

MEMBER BROWN: Can I amplify what she said a little bit Jesse?

MR. SEYMOUR: Go ahead.

MEMBER BROWN: What just occurred -- what's been brought up in the past is that we'll have these reactors sitting out there, a small modular reactor with somebody in the middle of Wyoming with somebody back in New York controlling it via the internet. And we just lost control. I mean, somehow it's got to be communicated that you've got to be in a manner to always have control on demand. And I don't know what the right way to phrase it is, but that's the thought process that I think you ought to be trying to incorporate into the Part 53 language. Did I reflect your thought?

MEMBER HALNON: Yes. For the staff don't answer that now.

MEMBER BROWN: No reason to answer.

MEMBER HALNON: But what I wanted to say is that the theme that you're going to be hearing I believe, and we talked about it. I had to sit back and think about what the themes were at the subcommittee meetings was the -- and I don't know if I'm going to use the right word. It's sort of the permissive structure, the lack of a minimum, the lack of an expectation or whatever the case may be. We talked about with the STA. We'll talk about it some more.

So we'll work through that and try to clarify that comment for you so that you'll understand where we're coming from as we go through this. So I didn't want to bog it down right now.

MEMBER BROWN: That's fine. I just wanted to get that thought process.

MEMBER HALNON: You popped the bubble on where we wanted to go on some of these things. Go ahead.

MS. CUBBAGE: This is Amy Cubbage, NRC staff. I just wanted to interject if I may.

MEMBER HALNON: Go ahead, Amy.

MS. CUBBAGE: I wanted to reiterate what Jesse was saying, that under Part 53 we're currently not considering fully remote and autonomous operations which I think is what Charlie was alluding to.

MEMBER HALNON: Thanks, Amy. And it's not just the autonomous.

MEMBER BROWN: It wasn't just -- I'm sorry, Greg, go ahead.

MEMBER HALNON: No, it just wasn't autonomous. It was the minimum requirements.

MS. CUBBAGE: Understood, understood. I just wanted to clarify the idea of remote and fully autonomous is not being contemplated right now under the rule. And as Jesse indicated all of these scenarios would have some operations presence onsite. It's a matter of how many and what their certifications or licensing would be. And Jesse, please correct me if I'm wrong.

MR. SEYMOUR: No, that is the case. And I just, you know, while I have the floor here just for a second I just want to reinforce the fact that when determining the number of operators. And again, we're talking about licensed operator staffing and the general approach that Maurin is outlining here. This is a performance-based requirement. And Maurin will

talk about this. But an applicant is going to have to use human factors engineering based analyses and assessments to show the proof, to show the data for why that number of people is going to be capable of fulfilling the safety functions for that facility.

So not only do we not go to zero, but we also have essentially the state of the art of human factors engineering that has to be backing up the number that they're using.

MEMBER BALLINGER: This is Ron Ballinger. I've got to go look at the transcript for one of the previous Part 53 meetings, but I seem to recall the statement that Part 53 will not preclude autonomous and remote operation.

MEMBER BROWN: That's correct. I vaguely remember that, but I would not swear to it.

MEMBER BALLINGER: So what I'm hearing now is that's not the case.

MEMBER HALNON: And I think -- this is Greg. We're talking two different areas. We were talking in the technology world about how you would license the technology as opposed now we're talking how you staff the operators. So there is a disconnect between those two areas. If the expectation from this group is that you will not have autonomous reactor

operations that's not the expectation of the group writing the language for the earlier parts of the Part 53. We did have that discussion at least once if not twice during other subcommittee meetings, but not for the Subpart F. So that's a question.

The other piece of it is, again, it's not so much the autonomous operations, it's the lack of that expectation being very explicit in the rule language which is what you just said, Jesse, was there will not -- there's no expectation of zero operators, no expectation of autonomous or remote operation. Well, that's not in the rule language. That's your expectation, but that's not in the rule language. So that's the point, again, as we try to cut through all the examples. But that's the point.

CHAIRMAN REMPE: Greg, I'm going to interrupt here for a minute. Someone tried to respond with a technical response in the chat. That doesn't appear in the transcript and so it doesn't -- in fact, it just flashes on the screen and it's gone. So if the staff wants to make the comment they need to make it verbally so it's on the record.

MR. SNODDERLY: Right, Chairman, and I can do that. This is Maurin. I'm ready to keep going and I talk to what's in the chat. And what's in the chat

is this implication that the staffing plan, there's got to be somebody there. So if you look in the Subpart F discussion table from October 2021 when we put this proposed rule language out, or preliminary rule language out it discusses how the staffing requirements work.

And the implication of staffing requirements specified here for certified operators are that there must be a minimum of one individual fulfilling these requirements at all times with as many additional certified operators as needed above that minimum in order to implement these requirements.

An applicant will need to demonstrate how their proposed approach to certified operator staffing accomplishes this. It must also be recognized that even a fully autonomous reactor would still need to be under the cognizance of a certified operator. So in staffing land where the rule works is that there's at least one person there. We don't say that in the rule because that would be prescriptive, but there is an assumption. And we're going to look at that.

And that's where I was trying to get -- these areas are all interrelated. So you know, who are these people. Well, they're certified operators or they're licensed operators. And there's human

factors requirements to determine those needs and what kind of HSI they operate with, or just watch, monitor with, et cetera. So it's all interrelated.

I agree it doesn't really work with zero. I can't review a staffing plan that says zero. There's not going to be much to do. So it's kind of set up for this. And this does establish a minimum. So we're letting the design kind of lead that, what is the minimum for this design and prove it is kind of what -- that is our approach.

MEMBER BALLINGER: To me the word "cognizance" is important. That doesn't mean the operator has to be there.

MEMBER BROWN: There's another thought process with that. One of the things that needs to be considered in terms of operating any plant, and I can use only one example. I will just use one example, is when you have multiple plants and you've got a large amount of computer-based operations where they're monitoring and they're controlling and you've got one operator there ought to be -- this is me, not the committee talking. For any critical operation something has to start, or be stopped, or initiated, or what have you. There ought to be a manual backup somehow, and that's part of the human factors

engineering assessment is can that be done if the automation fails.

We do that with reactor trips. There's a manual backup, manual scram switch that goes directly to the scram breaker. You can directly go start a pump if you need to manually. It doesn't go through the computer system. So I don't know how that -- that's got to be factored in when you're looking at human factors engineering and the displays a person has even if it's only one. That doesn't mean -- it's just a thought process that you've got to consider as part of this whole staffing and operations setup. I'll stop right there. It's just a thought process.

MR. SNODDERLY: Okay, thank you.

MEMBER BROWN: Thank you.

MR. SNODDERLY: Thank you for that. Okay, so I think we were on the slide with the rule. I was stepping through the staffing rule. And first, the applicant needs to provide a staffing plan that describes the numbers, positions, and qualifications of reactor operators and senior reactor operators or certified operators across all modes of operation.

That staffing plan needs to describe personnel providing support in areas such as plant operations, equipment surveillance and maintenance,

radiological protection chemistry, et cetera. The list is on the slide, I'm not going to read it all. And this type of information, they support areas -- this information supports the staff's understanding of any non-traditional roles or collateral duties that those licensed operators or certified operators might have, and any associated workload with that.

And then finally facilities that are required to have licensed operators must provide a description of how the proposed licensed operator staffing would be sufficient to provide assurance that plant safety functions can be maintained. And this is performance-based. It needs to be supported by HFE, human factors engineering analyses and assessments.

MEMBER HALNON: This staffing plan is absolutely key obviously, and it's going to be -- I assume it's submitted and approved by you guys during the licensing process. Is there going to be a guidance document that puts out there what are the minimum requirements for a staffing plan to submit? Or are you going to let it be a free flow of whatever the licensee or the applicant wants to put in there? That's not the right way to put it. Are you going to have a guidance document that's going to give what one should look like, what should it contain, what level

of detail in that respect?

MR. SNODDERLY: We have a staff review guidance document under way for this to look at these staffing plans, how to -- what we're going to be looking for, how we're going to verify that they work, what kind of validation did the applicant do. So yes, that exists in the form of guidance.

But the requirement itself is right here. That has to describe numbers, positions, and qualifications across all modes. So this is what's required. It's pretty clear, but how the staff is going to look at it, yes, that will be available and it's -- like I said it's under way currently as an interim staff guidance white paper that we'll be putting out for comment sometime this spring I think.

MEMBER HALNON: Okay. I think we'd like to see that when you get it to a point where you want it to be. I think that will help alleviate any fears that we're not going to have operators onsite, or anything like that. Let's go ahead and move on unless there's other comments. Okay, go ahead.

MR. SNODDERLY: Okay. Next slide, please. So now I'm going to talk about the shift technical advisor, or the STA for Part 53. Initially the staff did not include a requirement for staffing the shift

technical advisor position in rule language. We arrived at this position in part because of commission policy that made the STA position an interim measure after the accident at Three Mile Island unit upgrades could be made to the control room human-system interfaces and operator training.

We also left the STA out of the rule because of the combination of other requirements for operator training, human factors engineering, human-system interface design, staffing analysis, and defense-in-depth provisions for Part 53 facilities that in essence take over from the STA.

We understand that the committee has the following concerns about the no STA approach. And they are reservation about blanket STA elimination under the rule, the value of having an independent individual for event assessment, a desire to maintain engineering expertise available, and the relevance of the role of the STA in light of uncertainties with new designs. So we understand those concerns and I'll say in light of these concerns and after additional discussion that we've had on this very challenging topic we are currently considering three different options for the next iteration of the rule language.

The first being the status quo which is no

requirement for staff. The second is STA is required with provisions for omitting it with justification. And that would be at plants with licensed operators. So the STA would still be required as part of that staffing. And then the third one is at all plants regardless of whether they have licensed or certified operators all plants must provide independent and readily available engineering expertise to the on-shift operators. So like I said we're still considering them. I'm going to open up the floor now for full committee discussion on this topic if I can.

CHAIRMAN REMPE: So before a member has a comment, we discussed earlier that the court reporter could identify which member is talking and they didn't need to identify themselves. Apparently my light, I'm the chairman, lights up anytime anyone in the room talks. And so a participant has indicated that it would be helpful for them if we could identify ourselves. It's not absolute all the time, but just to try and have them understand who's talking. Okay, folks? Thank you. Go ahead.

MEMBER HALNON: Thank you, Chairman. So we've arrived at this point after subcommittee discussions and providing comments back to where we have these three options. Again, this is Greg Halnon.

That didn't work very quickly, did it. Isn't two and three essentially the same? In other words, if you can do number three then you can eliminate the STA provision. I guess explain to me the difference between two and three.

MR. SNODDERLY: Okay. So, two is like we say today at our Part 50 plants, that's the STA body. There's licensed operators in the control room. The STA might be one of them, a licensed SRO or has other qualifications. So that's going with what we currently have for that position, that role, with the option to justify why you don't need it such as we saw for NuScale.

MEMBER HALNON: Wouldn't you use number three to justify why you wouldn't need it?

MR. SNODDERLY: Could you use number three to justify --

MEMBER HALNON: In other words, the way that we justify omitting the STA now is by saying we have engineering expertise within our licensed operators, and we have them available that they can jump into the STA role.

(Simultaneous speaking)

MR. SNODDERLY: I think in number two we're looking at the traditional STA, and number

three, these are standalone. These aren't like you can do any combination. If we went with number three it's having some access to engineering expertise. So you don't have to call that person the STA. They don't have to be the licensed operator. So different from two is that it could be somebody at a different location that you can get a hold of, this is one way I see it. To help you with an issue that you're seeing on shift.

MEMBER HALNON: Okay, I get it. So number three is just access to. I've got it. Okay.

MR. SNODDERLY: Yes.

MEMBER HALNON: So it doesn't have to necessarily be onsite, but you have somebody who they have on call that they can call at any time.

MR. SNODDERLY: Exactly.

MEMBER BROWN: I think the key is on call for each shift of operation. Not that you find them if they're there. They have to know they've got to be available. That's the way I would read that.

MEMBER HALNON: Yes, and this is Greg again. I assume that we would go further on the engineering expertise that they would have access to indications, display information that is independent than just a phone call where somebody is relaying

information to them.

MR. SNODDERLY: Right. I imagine some kind of awareness -- this is Maurin, sorry. Some kind of awareness -- we haven't gotten into the details of how this works or looks, but -- and I will point out that number three is for certified and licensed operator plants. So anything coming in.

And number two is those licensed operator plants where there are things that the operators have to do to maintain safety functions, keeping the traditional STA role to advise them.

MEMBER HALNON: Okay, thanks. Jesse?

MR. SEYMOUR: Thank you. I just wanted to add to the point. And again, with the understanding that option three would have to be further developed if we were to incorporate this.

We do have human-system interface design requirements that we adapted from the existing comparable post-TMI requirements. So when you go through and you look at certain indications and so forth that are required to be available to the operators, we do have things that cover possible indication of core damage states, establishes safety functions, and those kinds of important parameters.

And we don't make any distinction between

whether it's a licensed operator plant or a certified operator plant. So again, it wouldn't take too much to extend those types of provisions to whoever is fulfilling this engineering expertise type of requirement.

MEMBER HALNON: Thank you, Jesse. Go ahead on with your presentation.

MR. SNODDERLY: Okay, well that's all I had prepared for my part on the STA. It looks like we're ready to pass off to Theresa now. She's going to talk about certified operators. Over to Theresa.

MS. BUCHANAN: Yes, Maurin. Can you give me a sound check?

MR. SNODDERLY: Sound good.

MS. BUCHANAN: Thanks, Maurin. So I'm Theresa Buchanan and I'm here this afternoon to talk with you all about certified operators or certification of operators which is of course a brand new proposed process from how we're currently doing it.

I would like to note that I have a hard stop at 5 p.m. so hopefully I'll be able to get through all of my section. If not Jesse is standing by to step in and pinch hit for me. So I thank Jesse for being willing to do that.

So, certified operators. Giving you some background. The staff has proposed an option that allows for the use of non-licensed operating staff in place of licensed operators at plants that can meet stringent safety criteria. These individuals would still be required to complete a commission approved systems approach to training based training program and to demonstrate competence through an examination process that is approved by the Commission, but the certification of that competence would be through a facility program rather than through the Commission licensing process. Therefore, the expectation is that these operators would demonstrate a similar level of competence at performing those required actions determined from a job task analysis as would a licensed operator, although the domain of those actions may be different.

The stringent safety criteria that would need to be met to allow use of this option would ensure that the certified operator was not relied on to perform actions necessary to safely mitigate licensing basis events as opposed to a licensed operator who would be.

The certified operator therefore would not need to demonstrate competence in performing these

actions because they would not exist at that site. The certified operator would be responsible for administrative functions similar to what a senior reactor operator would be expected to perform as well as being responsible for operation of the facility in the following ways as applicable to the site design. Monitoring plant data and parameters, shutting down the reactor when required, dispatching maintenance or other operations personnel, implementing the emergency plan, and of course performing reactivity manipulations when required. Next slide, please.

All right, so now I'm going to talk about --

MEMBER BROWN: Can I ask you, can you go backwards?

MS. BUCHANAN: Certainly. Can we please go back one slide.

MEMBER BROWN: Slide 12. I just want to make sure I understood you. You said the certified operator would not be able to handle an accident situation.

MS. BUCHANAN: I'm saying that --

(Simultaneous speaking)

MS. BUCHANAN: Are you finished with your question?

MEMBER BROWN: Yes, I'm trying to make sure I understood what you said. I heard the words accident or -- I can't remember exactly how you phrased it. Safety function, that's what you used. You wouldn't be certified to be -- to safety functions like a licensed operator would be. If that's the case who would?

MS. BUCHANAN: The terminology I used is that the stringent safety criteria that Jesse was discussing earlier would be in place to ensure that certified operators were not relied upon to perform actions necessary to safely mitigate licensing basis events. The reasoning being was that other processes that don't rely on human actions would be the ones that would be relied upon to mitigate licensing basis events in order to meet the other requirements of Part 53.

MEMBER BROWN: So it would be all automatic in other words. It would not require human intervention to perform those safety functions.

MS. BUCHANAN: It would not require human intervention to perform those safety functions. I'm not sure I would agree with the use of the term "automatic" because as Jesse mentioned on the second optional criteria it relies on things like inherent

safety characteristics. So the use of physics, or gravity. So if you consider those to be automatic. I consider those to be more inherent to the design. Any other questions on this slide?

MEMBER BROWN: No, that's fine. Thank you.

MS. BUCHANAN: All right. Next slide, please then. All right. So now we're going to talk a little bit about -- so I looked at some of the background. Now here's some of the reasoning. We sort of got into it a little bit on the last slide.

If an operator is not needed to ensure safe mitigation of licensing basis events as I just mentioned then they would not be required to be licensed from a public health and safety standpoint.

Regulatory burden is also a consideration. These facilities would be expected to have finite resources. And if we're requiring additional burden for licensing of operators when not necessary from a safety standpoint then that could require the use of resources that would be better applied elsewhere from a safety standpoint. In other words, requiring licensing of operators when not necessary from a safety standpoint could actually have an adverse impact on safety even though that's counter-intuitive.

Note, the facility is still licensed and must still have and follow a commission approved training and examination program for these operators which would be monitored for compliance by the Commission through periodic inspection. Tasks identified by the job task analysis for the certified operator position would still be accomplished by rigorously trained and qualified individuals.

So we have some past precedent. Certified fuel handlers are an example of a similar process in that the position requires these individuals be trained and qualified to perform the tasks identified by a job task analysis, in their case involving movement of fuel in the spent fuel pool through to dry storage. The certified fuel handler is responsible to ensure reactivity is maintained, respond in the event of an accident in their case, for example, a damaged fuel rod or a drop in fuel in order to take actions to mitigate the event. Their actions may not be complex and the consequences of the accident may not be as significant as for a currently operating facility. When I talk currently I mean currently operating, current power operators, not advanced reactors.

But they are still there and they have been evaluated as being acceptably low to allow for a

certified fuel handler program by a licensed operator. These processes are not the same, but they are similar in that the certified fuel handler like a certified operator would be expected to monitor the progress of work, perform administrative functions, direct work activities, and oversee operations in the case of a certified operator that may impact reactivity in the spent fuel pool. Not on this --

MEMBER BROWN: I'm sorry, go ahead.

MS. BUCHANAN: Okay. Not on this slide, but another example that I wanted to provide is with fuel cycle facilities. Although the NRC does have the authority under the Atomic Energy Act of 1954 as amended to license operators at fuel cycle facilities this has not been done in practice. Instead, technicians and line operators at these facilities are trained and qualified in the required aspects of their job by the facility. This process is reviewed as part of our standard review plan for fuel cycle facility applications. Again, this is a similar process but there are some differences as there are safety consequences for these individual actions or inactions in operation of the facility.

In fact, 10 CFR Part 70.61 discusses what I refer to as safety tiers. An example would be a

high-consequence event, or credible intermediate consequence event. And they require engineering or administrative controls to limit the likelihood of these events. And these controls are applied to the worker's responsibilities as applicable.

A high-consequence event could result in an acute worker dose of 100 rem or more, a dose of 25 rem or more, or intake of 30mg or more of soluble uranium to any individual outside the control area, or certain acute chemical exposures to individuals.

A certified operator for a power facility would not be expected to be required to manipulate plant equipment in the event of an accident as we discussed earlier, although they would have other responsibilities such as notifications and administrative actions.

For reactivity manipulations and whether or not a non-licensed operator can safely perform those the staff's perspective is that if the design can meet the stringent safety criteria then the consequences of any errors performed by the certified operator during these reactivity manipulations would be bounded by the design, and this plus the previously discussed rigorous qualification requirements provides an adequate assurance of protection.

Finally, although it has been said that some current advanced reactor designs may meet the stringent safety criteria Jesse mentioned earlier, even if they currently cannot the intent of this rule is to make it robust enough to accommodate future designs which may be able to meet this criteria.

MEMBER BROWN: Can I ask a question?

MEMBER HALNON: Go ahead, Charlie.

MEMBER BROWN: It says whether the operators are licensed or not the facility would still be licensed by NRC. Does that shift the liability to make you responsible then if something went bad as opposed to the operators? Not the operators but the company or whoever, the licensee? Is that the same as for fuel facilities, that NRC assumes the liability because it's licensed by NRC even though the operators aren't? It's a little bit different. It's not a reactor plant, it's a fuel facility as opposed to there's -- consequences are somewhat less than you would from a power reactor.

MS. BUCHANAN: I agree that the consequences at a fuel cycle facility are somewhat less than that of currently operating power reactors, yes. There are some differences, and I did acknowledge the fact that these -- precedent are not

exact precedent. They are similar, they are not the same. I do have some information on the next slide that actually may address your question.

MEMBER BROWN: You answered my question. NRC effectively absorbs the liability when we're in this position for a power reactor. That's the way I would read this. Greg?

MEMBER HALNON: She made a point -- this is Greg -- made a point of saying that the facility is still licensed. So you're still going to have a chief nuclear officer, or somebody that is in charge of nuclear that's going to -- the officer will ultimately be held.

MEMBER BROWN: The owner --

MEMBER HALNON: The owners --

MEMBER BROWN: -- even though they're licensed by the NRC without the "operator" licensing certification.

MEMBER HALNON: I believe that would be the case in this situation.

MEMBER BROWN: Okay. All right, well thank you.

MEMBER HALNON: I mean, you'd have to follow up with the enforcement policy, but it will go after the officer of the company.

MEMBER PETTI: I just had a question and I don't know the answer. Does this new approach impact the ability of facility owner and Price-Anderson insurance and all that, is there a connection there that has to be thought of?

MS. BUCHANAN: Well, I will state that I don't have the answer to that as I'm not really involved in anything to do with the Price-Anderson Act associated with insurance. What I will say is that I actually have information on the next slide that's really related to this topic so if everyone is okay with it can I go to the next slide and then we can revisit this?

MEMBER PETTI: Sure.

MS. BUCHANAN: Okay. So let's go to the next slide, please. All right, so there are actually some prior concerns raised by the ACRS subcommittee. And some of these concerns include differences in accountability as compared to licensed operators which is kind of what was being alluded to earlier. Where is the accountability, as well as a lessened ability to resist coercion by inappropriate management action or orders, a redundancy to a scalable operator licensing provision, as well as lack of certification by an independent entity, i.e., NRC.

So we acknowledge that these concerns are valid. We continue to have the perspective that if the stringent safety criteria are met to allow the use of this provision the safety consequences of an inappropriate operator action or inaction are so low that these concerns would not be expected to have a significant impact on the safety of the facility.

Additionally, the framework allows for appropriate staffing requirements when there's no significant human role in safety. There's past precedent for using non-licensed personnel in similar roles although those roles have a different relation to safety for those types of industries.

The ability to meet the criteria and the desire of the industry for this provision has been expressed. It is important to note, and here's the piece I wanted to get to. It's important to note that the administrative section of technical specifications can assign and with the current operating reactor fleet have assigned responsibilities to management to allow for personal accountability so not just the facility is accountable.

A similar case could be applied here, codifying in technical specifications the certified operators by tech specs which are part of the license

for the facility, that certified operators have certain responsibilities they must meet.

So before I finish the slide did that answer the question that was raised earlier, at least help answer the question that was raised earlier?

MEMBER HALNON: I would say it addresses the question. I'm not sure it answers it.

MS. BUCHANAN: All right. I'll take addressing it.

MEMBER BIER: I'd also like to ask a quick follow-up. This is Vicki Bier. I just want to make sure I understand a question that was going through my mind while I was listening to you. Early on you said that a certified operator would have no safety responsibility essentially. And in my mind I was running through but actions they take could have adverse consequences for safety and who's going to be there to fix it if they do.

And the second part of your presentation kind of said well, those adverse consequences would have to be bounded by the design of the facility. Is that the answer to my nagging question, that there's a limit to the damage somebody can do?

MS. BUCHANAN: I wouldn't call that a nagging question. I would call that a very good

question and the answer is yes, that is part of it. I believe when -- and Jesse, please correct me if I'm wrong, but I believe when the stringent criteria that he was mentioning, especially on the second option, there's a statement there that has to do with the ability to withstand not just the event, but basically operator actions associated with the event. Jesse, did you have anything you wanted to add on that?

MR. SEYMOUR: Yes, Theresa, that's a good way to characterize it, and it's actually baked in to both options. It just gets there via different means. But the key thing is that whether it's through pointing to reliance on the types of robust passive features that would be resistant to incorrect human actions rendering them unable to fulfill their safety function, or things that are just inherent, or whether it be requiring -- perhaps through a human reliability analysis type assessment, going through and determining what the credible human errors of omission or commission could be. Installing engineered features to reduce the likelihood of those occurrences down to some acceptable level.

Again, those two options, they both accomplish that and via different means. But it's baked in to both of them for lack of a better way to

put it. Again, it's a key consideration that if you're assuming that people, they don't have a human safety role, it's not only for the actions that they need to take, it's for the things that they could do that potentially could remove the things that are credited for safe (audio interference).

MS. BUCHANAN: Thank you, Jesse. Does that answer the question or address the question?

MEMBER BIER: Yes, I'm happy with that answer for now. Thank you.

MEMBER BROWN: I'm going to calibrate that in just a slightly different way. This is just from past experience in that the guidance we used to give to our operators, you were the final -- the operators were the final step. They were the final safety step. They were responsible for the safety of the reactor, taking care of it. They were always there. We're effectively abandoning that thought process. I'm not saying it's not justified if it's bought off on, I'm just saying we've walked away from saying that the operator is the final protection against a major accident.

MS. BUCHANAN: I would characterize -- finish, please.

MEMBER BROWN: No, that's -- there's no

way to -- you could characterize it.

MS. BUCHANAN: Well, I would like to characterize it slightly differently. You can choose not to agree with this characterization, but the way I would characterize it is we would give the facilities an opportunity to prove to us that they don't need the operators as that last step to safety. But they would have to prove to us that those operators are not only not needed as last step to safety, but they can't do a wrong action that would put them into the line of safety.

And that's how I would characterize it. So I wouldn't say that we're abandoning that as a philosophy. I would say that we are modifying it to acknowledge the possibilities of future technologies that they could be designed in such a way as to eliminate the need for it.

MEMBER BROWN: The plant is so safe that it can't possibly hurt anybody.

MS. BUCHANAN: Well, that's why I said it's a very high bar.

MEMBER BROWN: Okay, thank you.

MEMBER HALNON: This is Greg. Before Theresa has to go I want a couple of questions. You mentioned that the lesson, the ability to resist

coercion by inappropriate management. I got that part. What would keep management from only certifying operators who let's say toed the line as opposed to operators that may not necessarily -- let's say independent thinkers as opposed to just toeing the line.

What would prevent them from only -- being very selective on their certified operators to not have to coerce them?

MS. BUCHANAN: Well, I would respond by saying it's primarily the other aspects of the rule language for certified operators. They're required to propose a training program which would include eligibility, experience requirements for entry into the program as well as what the contents of the program would be to output an operator. And they would be required -- we would then approve, review and approve it, the Commission would. They would be required to follow that program, and then we would be coming by in the -- behind them doing periodic inspections to ensure compliance with the program.

Now, could facilities look for certain personality traits that would fall within those experience and eligibility guidelines? Possibly. But they would still have to meet all of those guidelines

as well as the training program, a certification program, and periodic inspection by the Commission. So I don't think that that would be a significant concern, that they would only have compliant operators. Would that be a good term to use?

MEMBER HALNON: Yes, I understand what you're saying. Okay, then carrying this out further then, first of all I did want to make the statement that we are assuming that these are qualified proficient operators. They're not duds who get through this training program. So I wanted to say that. They're probably equivalent to the proficiency and abilities of a licensed operator. So we're not talking about people who are just going to go do bad things all the time.

However, first question, and this is -- should be an easy one. Do you anticipate that there would be full-time resident inspectors onsite at these new smaller facilities?

MS. BUCHANAN: That's beyond my purview. I wouldn't be able to answer that question.

MEMBER HALNON: My sense is no. If you're not going to have any operators there you're probably not going to have a full-time resident inspector. So the second question is can the NRC de-certify an

operator if they see something that they don't like?

MS. BUCHANAN: And Jesse, I'm going to ask for your assistance on this one, but the way that I am envisioning this program is that the certification is at the facility level, but it's covered by a commission reviewed and approved program. And if the facility were to fail to follow their program which would include de-certifying operators when necessary then they would of course be subject to enforcement action. And if they continued to fail to de-certify operators when necessary then of course the enforcement would continue to be escalated per that process. Jesse, did you have anything you wanted to add?

MR. SEYMOUR: Theresa, you characterized that well. Just very broadly, something that we for the purposes of this presentation can't speak to to any great extent is what the inspection program and what the oversight process will ultimately look like within this framework simply because it's beyond the scope of the piece of it that we're working on. To my understanding it really kind of goes a bit beyond where the status of the work is.

So again, if we're asking questions about will they have resident inspectors, and what type of

specific inspection activities will be there. We can't speak to that, but what we can say is that we're laying the groundwork here for what we anticipate the inspection needs and the regulatory hooks to be. And one of those which Theresa I thought put quite well is that the facility will be on the hook for maintaining the approved training program and also complying with the regulations that govern the certified operator process. So part of that would be removing people that aren't performing from performing their duties and remediating them. Again, that's casting a pretty wide -- painting that with a pretty wide brush. There's more specifics to it than that.

But if they're not doing that, again, our intention right now and the way that we foresee this is similar to how we do re-qualification program inspections. There would be regulatory touch points where we're looking at those programs. And again, if we found efficiencies there, in a similar manner to how, in the commercial world, we will issue findings under the ROP for the re-qualification training programs. We would have the regulatory hooks needed to issue enforcement. Those actions would address to that facility as licensee. And then from that point again if there was continued non-compliance with the

regulations, again we've seen how that works in enforcement space. Things will ratchet up from there.

MS. BUCHANAN: Thank you, Jesse. I've only got two sentences left on my spiel so can I finish those and then we can get back to your questions?

MEMBER HALNON: Yes, go ahead, Theresa.

MS. BUCHANAN: Thank you so much. As Maurin just finished discussing we are continue to evaluate the role of an STA or engineering expertise availability. If the STA role or that engineering expertise availability is included in the rule language this can also add -- provide additional assurance of safety for the plant.

Finally, a key requirement, this is kind of like a summary. A key requirement is the certification programs require the review and approval of the Commission, and compliance with the programs will be monitored by the Commission through periodic inspections. So that's the end of my prepared remarks for the certified operators section. So if we could return. I'm sorry, I didn't catch the name, but if we could return to the individual who said he had some questions for me. He was only able to ask one.

MEMBER HALNON: This is Greg, Greg Halnon.

You've answered most of them. I'm not going to press it beyond. I brought up the inspection piece because you relied on it in an earlier answer. That's one of the reasons I brought that up. But with lack of continuous oversight on the site like we have with our large light water reactors operator proficiencies at the request of and opinion of the management who may not be onsite.

And the other piece of this, the basis for the earlier question was the difference between the licensed versus non-licensed operators and the level of personal accountability they feel for that.

The last question I have for you is with all the stuff and all the conversation that we've had why are you doing this?

MS. BUCHANAN: Okay. So you threw me the softball earlier to throw me a hardball later. I got it. Why we're doing this. Well, we already got some feedback about the use of all those buzzwords, but unfortunately those buzzwords are actually very applicable in this situation.

Part 55 as written right now works for large light water reactors, but it's very prescriptive, and it's very technology-specific for large light water reactors. So we obviously have to

do something about that because it won't fit with advanced reactor designs. There are aspects of it that just won't.

So, okay, that follows into the why are we doing Part 53. Fine, now it's like why are we splitting between licensed operators and these non-licensed certified operators. Why not just do licensed operators.

We could do that. What we are looking at trying to do was trying to balance a regulatory burden versus a safety impact. So we're trying to reduce regulatory burden. This would be a reduction of regulatory burden because the facility would not need to be going through the Commission for the licensing of operators. They would still have some regulatory burden because we're not just handing this over to them and not ever looking at it again. That was a very good point you had earlier. We need to make sure that we have appropriate touch points to ensure that they're following their programs appropriately.

But it would reduce regulatory burden in this aspect, to allow them to focus their efforts elsewhere, their resources and efforts elsewhere. So really that's why we're doing this. We are attempting to balance the burden of a processes against the

safety. So if the safety case isn't there the burden shouldn't be there, and that's it.

MEMBER HALNON: Okay. Well, thank you for that. I wanted to just before you left give you the sense that I think this is the sense of the committee. I speak for myself and I'll look at the heads nodding or throwing something at me around the table.

I think we're concerned with the continued cumulative effect of this backing off of oversight through requirements in the regulatory language. Elimination of the STA. We made that comment and now you're coming back with some appropriate options. Not having a licensed operator. How about potentially not having any operators onsite at all? Having lesser scope simulators as we'll talk about in a minute. Having less re-qualification requirements from the standpoint of prescriptive two-year re-qualification exams, written exams and that sort of thing.

So the cumulative effect of all these feels like we're giving away the farm to the less oversight and less NRC I guess hooks is what I'm trying to say. So anyway, that's where the concern is. We'll continue to talk about it. We'll talk more about it probably with the reduced simulator scope although I think we'll get through that pretty

quickly. So that's where we're coming from and that's where you see some of the discussion coming from the members.

MS. BUCHANAN: Understood. I thank you for the insights, appreciate that. I'm going to -- Jesse, I'm going to go ahead and turn this over to you because I've got to head out of here. So I want to thank everyone for letting me come and present before you all. I really do appreciate having the opportunity to present the staff's perspective on those controversial topic.

MEMBER HALNON: Thank you, Theresa. Jesse, you're back up again.

MEMBER PETTI: So I had one comment sort of on the opposite side. I keep hearing this is a high bar. So, as I look at the advanced reactors I see very few that will not rely on an operator in some way once they do a full safety analysis, et cetera, et cetera. So I worry that we're going through a tremendous amount of effort for a null set. No one can get through this gate. Now, there may be some design out there, but at least all the ones that I know of that are out there, there has to be an operator involved.

It will be very different than the current

fleet in terms of the nature of the interaction like we even see in NuScale. But I worry -- there's also that. There's so much effort here and in the end the bar is -- may be so high that why are we doing it.

MEMBER HALNON: That's kind of where my question came in. I agree.

VICE CHAIRMAN KIRCHNER: I have the same feeling, Dave. As any -- this is one member's opinion so I have to be careful how I say this. If it's a power reactor of scale, I'm talking 100 megawatts or more, I don't -- like you say, I see the bar very high, and also I see lots of reasons why you would want and need an operator onsite full-time with such a facility.

So, if we put aside these micro reactors for a moment, I know the staff is to be commended for trying to span a spectrum that is like a research reactor to something that's, you know, on the other end is an LWR. But I have the same feeling as Dave, that the credible power production reactors, this is going to be a very high bar to demonstrate no one is needed and then we drop all of the requirements down to a certified level. (Audio interference) that person's only doing administrative functions. There's some language that's being used here that I don't

think is appropriate either.

MEMBER HALNON: And this is Greg. Remember we're talking about commercial reactors that have the sole purpose besides being safe to earn money for somebody. So.

MEMBER BIER: I have one other comment I want to make which is at the kind of 10,000 foot level. So when -- I guess it was Charlie who asked is there a floor. Is the floor zero or non-zero. And essentially Jesse's answer was no, the minimum is one, but we cannot say that because that would be prescriptive. And I'm like if the answer is the minimum is one why can't you say that and what is gained by obscuring that fact. And I realize -- it's not a criticism of Jesse. I realize there may be some administrative pressures being brought to bear, industry pressures or whatever, but that just concerns me if there is a plain English answer that we cannot write down.

MS. CUBBAGE: This is Amy Cubbage, NRC staff. I want to reiterate what the staff tried to say earlier. The preliminary proposed rule language does specify a minimum of one.

MEMBER BALLINGER: It specifies a minimum of one cognizant person.

MS. CUBBAGE: So maybe we need to take a look at that language, but the intent was one.

MEMBER BALLINGER: This is Ron Ballinger. To me cognizant doesn't mean onsite. And maybe that's appropriate.

MEMBER BROWN: And it doesn't mean a reactor operator either. It could be certified.

MEMBER HALNON: And this is Greg. It doesn't conform with what we heard earlier on other parts of Part 53 so you have to get together with -- and aligned with the rest of the Part 53 staff.

MS. CUBBAGE: Well, I'm communicating information from the Part 53 team. If somebody else would like to chime in. We're kind of talking amongst ourselves here. I'm seeing language that the licensee must provide for a certified operator to continually monitor, et cetera, et cetera, at a minimum, et cetera, et cetera. So I'm not seeing this cognizant language in the rule text. So I think we need to kind of take that back and make sure -- your comment is well taken, and make sure that the language is consistent. But that's what my awareness is of the language.

MEMBER BALLINGER: I'm just reading, I guess it's from the chat line from Lauren. It must

also be recognized that even a fully autonomous reactor would still need to be under the cognizance of a certified operator. This is Ron Ballinger.

CHAIRMAN REMPE: So Lauren, if you're there please speak up.

MEMBER BALLINGER: That's what I recall. Excuse me, I'm sorry.

CHAIRMAN REMPE: Yes, if you're there please speak up.

MS. CUBBAGE: We need to clarify that because that's not rule text. That was a table. So please go ahead, Lauren.

MS. NIST: Yes. So like Amy said a moment ago, the staff here in our caucus had been sharing some information to help address that. And again I had forgotten about the limitation on the use of the chat in this meeting and I had shared that.

So, what I shared was from a discussion table that was from October of last year on the rationale for some language that was being put forward. But it looks like in the preliminary, let's see, 53.755(i)(3) the licensee must provide for a certified operator utility to monitor the operation of field units. At a minimum the certified operator must have the following capabilities. Cognizant was not in

the rule text.

So I think Mr. Brown and Amy had just said that as well, so I don't think there's anything additional there from that chat message that hasn't been stated.

MEMBER BROWN: This is Charlie Brown. I didn't say cognizant. I haven't gone to the -- I was just looking at the certified word as opposed to reactor operator, licensed.

MS. NIST: Yes, that's right. That's what I -- my apologies. What I meant to say was that you had basically just stated what I said about the minimum operator could be licensed or certified I believe.

MEMBER BROWN: Yes, that's all. I just have difficulties with -- I just looked for a floor of what I would call a licensed operator. By the time we went through all the certified stuff. I understand what you're trying to do, and I understand the basis as you reiterated it very well. It's just I guess I really don't agree that you have somebody that's, quote, "certified" that can't take actions under an accident condition. That does not make a whole lot of sense to me. It may be where we end up, it just doesn't make sense to me.

MEMBER HALNON: This is Greg. Before this horse is absolutely dead let's go ahead and move on to the other topics, and then we can summarize some discussions at the end. If the team wants to continue on we'll be quiet for a minute. I can only guarantee a minute.

MR. SEYMOUR: Okay. So with that being said I'll continue on to the simulator scope discussion and then we can come back to those points. I think there's some clarification I can provide as well if we want to circle back around later.

So again, just a discussion of simulator scope. Specifically we'll be talking about that aspect of our training and qualification requirements. So now I'd like to discuss the topic of simulators and how we are preliminarily approaching that area under Part 53. So at present section 53.765(e) would establish simulation facility requirements for those plants that are required to have licensed operator staffing, and separately the 773(e) section would establish separate and somewhat less stringent simulation facility requirements for plants with certified operators.

Key aspects of both of those sections even though they are separate include the following.

First, full scope simulators are not mandated and partial scope simulators may, key word may be acceptable provided that the simulator scope is adequate to meet the intended usage. Alternatives to the use of simulators may be possible as well.

Simulation facilities for plants with licensed operators must be approved by the Commission if a facility licensee will rely upon them for training, experience requirements, or for initial or re-qualification examinations. Similar approval is not required for certified operator facilities though.

Also, facilities must demonstrate that adequate simulator scope is provided to support human factors engineering analyses and assessments in order to be able to use the simulation facility for conducting those analyses and assessments. We can move on to the next slide, please.

So in developing the preliminary rule language we -- and by that I mean our working group, we reviewed section 306 of the Nuclear Waste Policy Act and the Federal Register notice that's listed here on the side that was associated with the staff's implementation of the Nuclear Waste Policy Act simulator related provisions, and essentially the considerations for when they were first incorporated

into regulation.

One key observation that we made was that flexibilities were historically provided to allow for potential use of the plant itself, and/or plant referenced simulator, or some other type of simulation device such as a part-task or basic principles simulator for the conduct of the simulator portion of the operator test.

Another key observation was that we found that the NRC's stated intent during this time frame, and again this was written back around the 1987 time frame. The NRC's stated intent was not to permit the initiation of transience on the plant itself if the plant was going to be used as the quote unquote "simulation facility."

Rather, the use of the plant was envisioned as an option that might be used in conjunction with another simulation device or devices in lieu of a plant referenced simulator. Our current perspective based on our work thus far is that the Nuclear Waste Policy Act does not mandate that the NRC require that plants have simulators, but instead requires regulations to address the use of simulations in training.

The implication of this is that

flexibility exists to allow the use of the actual plant to simulate tasks for training and operator test purposes without necessarily having a separate simulation facility. In this case, specifically a simulator.

Prior concerns raised by the ACRS subcommittee have included the potential for reductions in training and evaluative efficacy, impacts on procedure quality, reduced support of analyses, and staff experience in approval of partial-scope simulators. If we can move on to the next slide, please.

The philosophical basis behind the preliminary rule language addressing simulators is that overall we feel that plant referenced full-scope simulators will remain the preferred approach and would represent the best route for meeting Part 53 requirements. So again, I'll say that we discuss these flexibilities but we still see that the most straightforward and preferred path to navigate that is with a full-scope plant referenced simulator.

Even with the flexibilities afforded we expect that the majority of Part 53 applicants will choose to go that route due to the regulatory certainty consideration associated with it, and

advances in technology lowering the associated cost.

I think it's important to keep in mind that what we are conventionally used to are these large simulation facilities that have analog controls. There's a lot of hardware involved, a lot of switches, gauges, things like that. And there's a substantial amount of installation and maintenance and upkeep costs associated with that.

As we look at modern designs we see the use of a lot of soft controls. So in this case we anticipate that with that progression we're going to see the cost associated with building a plant referenced full-scope simulator and maintaining it over the life of the facility come down. So again, these are the different things that shape our thinking here.

So as a point of consideration and a point of comparison I'd like to point out that the existing regulations, and by this I point to what we currently have in Parts 50 and 55. The existing regulations do not strictly mandate plant referenced full-scope simulators either because of how the Commission approved alternative structure. So again, if we truly look at how the Commission approved alternative structure, and we look at the genesis of that, that

there was the intent to -- I'll give an example. The research in test reactors, there was not a desire to change the status quo for them. So they were essentially grandfathered and allowed to continue with not having simulators.

But again, if we look at how the Commission approved alternative structure we can see that although there is another track that's built into this simulator process right now, full-scope plant referenced simulators were still adopted by all current power reactors. So again, just because the flexibility is there doesn't mean that it's necessarily going to be the most effective path.

So Part 53 language leaves alternatives to simulator usage, whether they be full-scope or otherwise. But, and this is very important here. The burden will still be on the applicant to demonstrate how a number of areas are supported by whatever approach they're proposing. And these areas would include showing the following.

So, they would have to show how licensed or certified operator training and examinations would be supported. The simulators used in this context would require sufficient scope and fidelity for operators to acquire and demonstrate the knowledge and

abilities needed for the job duties. They would also have to be able to demonstrate how experience requirements, and by that I mean specifically those associated with reactivity manipulations would be conducted. And also how human factors engineering analyses, assessments, and human-system interface design test bed needs would be met. So again, if an applicant were to opt to pursue one of these flexibilities, perhaps propose even to use the plant itself as a simulation facility in conjunction with maybe a part-scope simulator or something to that effect, this is the hurdle they would have to get over.

So again, it's not just a matter of I'm not going to build a simulator. You have to make the case for how you're able to do these things. And our belief is that there may be a very small subset of applicants for whom they do see a success path where they see that they can show this using an alternative to full-scope plant referenced simulator, that they can provide the required evidence to justify that, and that that will be a benefit to them as they go through the process.

But our sense looking at this is that the majority of facilities would look at this and that

they would still see the plant referenced full-scope simulator as being most advantageous to them over the long run.

So, the next slide would take us over to a discussion of guidance and our different guidance projects. So I'll pause there to see if there's any questions before we move on to that.

MEMBER HALNON: Okay, this is Greg. Anyone have questions on the simulator scope issue? I don't see any. Go ahead, Jesse, move on.

MR. SEYMOUR: Okay. So, this is the final part of this presentation. What I'd like to do now for the committee is I'd like to provide a summary of the areas of regulatory guidance that our team is currently working on. And I'll go through these. As you can see there's a number of areas. We're working on many fronts here.

So first, we are developing guidance to facilitate human factors engineering reviews being accomplished in a manner that is scalable. We have a working group that is developing this guidance with contract support from Brookhaven National Laboratory. Our current goal is to complete our draft of this guidance by the June time frame. So that would be this June.

Second, we are developing guidance for the review of the staffing plans that would be associated with the flexible operation staffing requirements that were preliminarily proposed in Part 53. So earlier when the committee was asking about this guidance and where it will exist, and when it will be ready, and so forth that's specifically what I'm talking about here.

So this guidance augments the existing NUREG-1791. And again, just for clarity NUREG-1791 is the human factors engineering based exemption review process that we use when applicants want to do something other than the staffing that's prescribed by 50.54(m). So again, if you want to propose a different staffing model than what is mandated by that regulation this is the human factors engineering based process that you use to justify that. So again, a different application of that. And this is guidance that supplements NUREG-1791 and repurposes its processes again just to a different end, in a very high level accomplishing the same thing, looking at what needs to be done and how many people and what roles do you need to do it, going through a very rigorous human factors engineering based structured methodology.

But again in this case we're augmenting

existing guidance for that purpose, and adapting it for use within the specific context of Part 53. Now, that's a project that we're doing in-house, so we're not relying on contractor support for that. We're using our own experts used to work through that.

And again, we are targeting June of this year. And again that's our current goal for having a draft of that guidance prepared.

Third, we are currently working with Idaho National Lab on the development of operator licensing examination guidance that can support the tailorable approach to operator licensing and certification examinations that this rule would permit.

And as before, in conjunction with Idaho National Lab we've got a pretty aggressive schedule for developing that. But we're currently aiming to have a draft completed by the June time frame for that guidance as well.

And also, something I wanted to point out as well too. On one of the first slides we showed the working group that we have very broadly. Again, just kind of the key individuals. There's more people even than that working on this stuff. But I want to say that we have quite a bit of interaction with the Office of Research on those projects.

So again, as I talk about the human factors engineering guidance, the staffing guidance, and the operator licensing guidance we have members of the Office of Research that are participating, on some cases even leading some of those efforts as we work through that. So again, a lot of close coordination, not only with contract national lab support, but also with our own Office of Research.

Separately we're also working on developing guidance for the review of a systems approach to training based training programs. And that guidance would support the review of non-accredited training programs and in general would serve as an update to the existing set of review guidance of NUREG-1220 and Inspection Procedure 41500. And during previous meetings of the subcommittee it was pointed out very accurately that those two references are dated. So we recognize that and what we're doing is working on updating and refreshing, and taking a good look at that guidance and reassessing what does it need to look like decades after it was originally developed. So again we're doing a pretty comprehensive overhaul of guidance for reviewing these staff-based training programs.

Our current development goal is

approximately one year. And that is so as to be able to support any near term applicants as they may be needed. So in case we get an applicant that comes in under Parts 50 or 52 that chooses to forego that accreditation, so as to be ready for that. But then on a longer time frame also have it ready for use within the context of Part 53.

So our current working group includes both headquarters and regional operator licensing staff, and also includes individuals with commercial instructor and operating experience. So again, commercial instructor experience very important here because they're kind of the end user of that process out in the field.

Lastly, it's important to note that ARCAP ISG Chapter 11 which covers organizational and human-system considerations is planned to be supplemented with guidance for Part 53 at an appropriate point in the future. So at present myself and many of the other members of the group that we talked about, Theresa Buchanan, Maurin, and so forth have worked to one degree or another on the current version of ARCAP Chapter 11 which is intended to address Part 50 and 52 applicants, fast reactor applicants, and to navigate some of the considerations associated with licensing

within that framework.

However, at a future date once Part 53 is sufficiently developed our intent is to augment that guidance. Now again, that document is broad in its scope, and we anticipate including within it the new guidance for other review areas beyond those key areas that I've already outlined here. And I'll just give some examples, but we foresee that eventually we would be expanding that guidance out to cover areas such as load following, the post-Three Mile Island human-system interface design requirements, and also potentially aspects of simulation facilities should we find that there's gaps with the existing guidance in those areas.

But again, the key thing is that that is an interim guidance document and over a very long time frame we anticipate that that guidance will eventually be translated into other review plans and documents as their permanent home.

So with that being said if we could just go ahead and move on to the final slide. This completes the prepared part of our presentation, and I'd just like to go ahead and open things up for questions and general discussion at this point.

MEMBER HALNON: Thank you, Jesse. Any

other lingering questions from the committee? Chairman Rempe.

CHAIRMAN REMPE: I'm sorry, I was just going to remind you to get public comments.

MEMBER HALNON: Thanks, appreciate it. I was going to forget. Okay, at this point then we'll open up the -- I have a question. Oh Amy, did you have something you wanted to say?

MS. CUBBAGE: Yes, I just wanted to follow up on one point that was made earlier. I don't want to leave anything incorrect on the record.

There was a statement about not wanting to have a certified operator there that's not allowed to take any actions. And I wanted to just state what the current preliminary proposed rule language says with regards to certified operators, that they must have the following capabilities. Number one, the ability to receive plant operating data including reactor parameters and information needed for the evaluation of emergent conditions, and importantly here, number two, the ability to immediately initiate a reactor shutdown from his or her location. So I wanted to reassure that these certified operators are not -- are going to be required to have the ability to take action to shut the reactor down. Thank you.

MEMBER HALNON: Thank you, Amy.

MR. SEYMOUR: If I could just clarify that point. Amy, thanks for pointing that out. It's a very good point. That's an area where we almost became deterministic in our thinking. And I'll clarify that.

You could go through analyses that show that the facility through inherent characteristics and so forth will find its way into a safe, stable state, you know, subcritical or whatnot. But even though, even though the analyses show that we still wanted that extra layer of assurance. So we said one of the requirements for the certified operator is that they have that ability to have vital plant information coming to them wherever they are located, and have that ability to initiate that reactor shutdown. So again, it's an area where we said for our own assurance we still want that extra measure there as well.

MEMBER PETTI: So Jesse, I'm a little confused then because earlier one of the criteria for a certified operator was not being involved in any safety -- can't remember the exact words, but any actions that would influence safety. Being able to shut down the reactor, that sounds like a safety --

MS. CUBBAGE: This is Amy Cubbage. Let me explain it in my words, and that's that if you do the analysis of the reactor and you have no operator actions that are credited for safety functions that's different than saying that the operator can't take an action if something didn't go as planned.

MEMBER PETTI: Okay.

MS. CUBBAGE: Did I get that right, Jesse?

MR. SEYMOUR: Yes, Amy. Again, it's kind of like how we think about having many layers to defense-in-depth. There's a difference between the operator being credited to do something and then having almost a beyond design basis capability that's built in there just for extra assurance. And that's really what we're talking about here. It's a prudent measure to have this. We think that's justified to maintain that capability.

MEMBER PETTI: Thanks, that helps.

MEMBER BROWN: The words that were used earlier, earlier said that the certified operator would not be capable of taking actions in an accident situation. Didn't say they couldn't shut down the reactor. I understood that they can shut down the reactor. My response, the reason I made the comment was there's many times when shutting the reactor down

is not the only thing that has to be made sure happens. So that was the reason for my earlier comment. And they've made the statement in earlier meetings that they could -- the certified operator could scram the plant, but whatever that means in the new plants. But what about auxiliary things that a real reactor, an actual licensed operator could take. Start this, do that, make sure such and such is happening, and make it a safe shutdown as opposed to just putting the rods on the bottom, or whatever the reactivity mode is.

MEMBER BIER: So, I would like to raise one other question which kind of comes back to a point earlier. This is Vicki Bier. And you said that -- I forget who, that the requirements for what level of staffing are needed would be performance-based. And I think within the NRC in general there's kind of a vagueness about performance-based that doesn't exist in some other fields because if you have performance-based pollution control you just go to the smokestack and measure. You don't need an analysis to tell you what the performance is.

But at NRC some performance is measured like how long the maintenance backlog is at a plant, and other performance is by analysis like how your

fire protection does you don't start a fire to test it. So coming back to my question about what if somebody had previously established a certain level of staffing and came back in and said by performance we can justify more reactors for the same staff because we've never needed our staff to do anything in the last five years. Would that count as performance-based or do they actually need to have an elaborate human factors analysis to justify that they don't need more operators or whatever?

MR. SEYMOUR: So in this case, the way that I understand I think the setting for that question is that this would be a facility that was licensed and then at some future point perhaps they're going to add additional modules or something like that that they would then look to modify that staffing.

So a key aspect of this is that the staffing plan that is submitted would be reviewed, and if we find it to be acceptable and approve it it would become part of the licensing basis for that plant. So even though that number was developed and reviewed and approved, and it didn't come out of a table in 50.54(m), what we would be doing is in the final stage of licensing that would become -- it would be combined.

So if they wanted to modify that what they would have to do is go through the license amendment process, and as we envision it it would be very similar to what they had to do prior to that. Again, you would have to go through the human factors engineering based analyses and assessments, do the validation work and actually show the proof for why the proposed revision to the staffing plan was going to be acceptable. And you wouldn't be allowed to do that until that licensing action had been approved by the NRC.

MEMBER HALNON: Jesse, how would you handle first of a kind technology then?

MR. SEYMOUR: So, with a first of a kind technology the key thing is that as you go through the staffing process what we're doing is we're borrowing from the framework of NUREG-1791. And what that's going to look for, again we'll see where -- there's challenges associated with this. This isn't easy. The NUREG-1791 process, it's structured. You begin with things like an operating experience review. So you look for existing operator experience which may be limited in this case. So again, that's an area of challenge.

As you move through that what you'll see

is that -- and again, I'll skip over -- I'm sorry? Okay, sorry, I thought someone was speaking. So what you'll see as you move through that is you'll see that there's an identification of what the safety functions are for that design. Again, achieved through analysis and safety functions, that's dealt with more broadly under the Part 53 framework.

There's an allocation where you determine who or what is going to be doing that, what level of feature and so forth. And then from there you develop a task analysis to figure out, okay, what are the tasks the operators need to be able to do.

Now, at that point you go through from there. And there's other stages, but ultimately what all that leads to is now that you've mapped out what needs to be done ultimately you're working towards the validation of that, and you're looking at really the proof that as you've identified those things that need to be done for that design, that the proposed number of people is going to be sufficient to accommodate that.

And typically that's going to take the form of a validation that's done in a simulator with scenarios, again, performance-based testing. Using this can you show that it all works. And again,

recreating things like workload under challenging conditions, making sure that under these emergency situations that the safety functions. And again, key wording that we use, that the safety functions needed for that plant can be fulfilled by the proposed number of people.

So again, what I don't want to do is gloss over the fact that first of a kind doesn't present a challenge. It does. What you're doing is you're approaching that with really the state of the art for how do you answer that question.

MEMBER HALNON: Okay. Dr. Desaulniers, you wanted to weigh in?

DR. DESAULNIERS: Well, I think Jesse went on to provide a little bit more clarity that I wanted to add in response to Member Bier's question with regard to what it meant to be performance-based. And Jesse was going into that there, that we wouldn't just be looking at past operating experience as a justification to infer what might be possible for performance in the future, but as Jesse was just noting typically this would be done in -- performance-based testing in a simulator where operations for the additional number of units if that was the question to be addressed here in the revised staffing assessment,

it would need to be shown through performance-based scenarios. So just wanted to be clear as I fully recognize there is differences in how that term is being used.

MEMBER BIER: Okay, thank you.

MEMBER HALNON: Yes, thanks. Any other questions from the committee? Okay, I'll open it up for the public comment. If you would please unmute, state your name and if appropriate, affiliation. Again, it's open for public comment. If you would unmute your mike that I think you do by *6, and state your name and if appropriate your affiliation.

Okay, hearing no public comments I wanted to thank the staff. Jesse, you and your team, Lauren, Maurin, all you guys. You did a great job of getting through this. We appreciate it. The slides were right on point. And we appreciate you addressing the comments of the subcommittee. We appreciate that very much. Chairman Rempe, how would you like to proceed from here?

CHAIRMAN REMPE: First, I'd like to note that we're going to go off the record, so the court reporter may cease recording things.

(Whereupon, the above-entitled matter went off the record at 5:36 p.m.)

Staff Presentation to the ACRS Full-Committee

NuScale Topical Report
Building Design and Analysis Methodology for Safety-Related Structures (TR-0920-71621, Revision 1)

February 2, 2022

Topical Report Review Chronology

- NuScale submitted topical report TR-0920-71621, Revision 0, “Building Design and Analysis Methodology for Safety-Related Structures,” on December 18, 2020.
- NRC issued RAIs -9833, -9834 on May 06, 2021. and RAI 9860 on August 06, 2021.
- NuScale provided responses to RAIs 9833 and 9834 on June 05, 2021, and RAI 9860 on September 05, 2021. The Staff found the responses acceptable.
- NuScale issued topical report Revision 1 of on October 06, 2021, that incorporated the RAI responses.
- NRC issued the advanced Safety Evaluation Report on November 30, 2021.

NRC Staff

Reviewers:

- Ata Istar, Structural Engineer, NRR/DEX/ESEB
- Amitava Ghosh, Ph.D., Geotechnical Engineer, NRR/DEX/ESEB
- Robert Pettis, P.E. (Retired) Sr. Reactor Engineer, NRR/DEX/ESEB

Project Managers:

- Demetrius Murray, TR Project Manager, NRR/DNRL/NRLB
- Getachew Tesfaye, Sr. Project Manager, NRR/DNRL/NRLB

Conclusions of Staff's TR Findings

- The methodologies presented in the NuScale TR are acceptable to perform building design and analysis for seismic Category I and II nuclear safety-related RC and SC structures other than containment.

- The methodologies follow implementation of the requirements of ACI 349-13 and AISC N690-18, Appendix N9, endorsed by RG 1.243.

- The methodologies are also consistent with the applicable regulatory requirements of acceptance criteria in NRC NuScale DSRS Sections 3.7.2 and 3.8.4.

Members Questions from ACRS Subcommittee Meeting

- Effects of corrosion on SC walls below-grade.
- Welding attachments to SC walls after concrete is placed.
- Industry experience on SC wall construction.

Effects of Corrosion on SC walls below-grade

- The staff requested information related to the corrosion effects on SC walls as required in provision B3.13 of ANSI/AISC 360-16.
- Section B3.13 of AISC 360-16 has a general requirement that states:

  "Structural components shall be designed to tolerate corrosion or shall be protected against corrosion."
- SMR designs may also consider the possibility of plant license extension of up to 80 years.
- In-service inspections and repairs of below-grade exterior SC wall faceplate would be impractical for the duration of the extended licensing period.
- **NuScale Response:** The following graded approach is described in the TR, Revision 1:
  - ✓ application of tar epoxy coating on exterior below grade faceplates.
  - ✓ using controlled low strength material or employing shotcrete cementitious material on exterior below grade faceplates.
  - ✓ Using backfill material with controlled pH and chloride limits.
- NuScale's response is consistent with Section B3.13 of AISC 360-16.

Welding attachments to SC walls after concrete is placed

- The staff requested information related to the attachments to SC walls and the effects of elevated temperature in concrete due to welding.

- **NuScale Response:** The attachments to SC walls are planned to be installed in the shop during fabrication or in the field prior to concrete placement. Design will employ a defense-in-depth control, based on industry wide engineering practices and more recent design experience related to installation of SC walls.

  When attachments are needed after concrete is placed, the effects of elevated temperature from welding are minimized by these generic industry guidelines:
  - The minimum faceplate thickness is limited to half an inch.
  - The minimum concrete age is 21 days. This requirement assures the concrete has enough strength.
  - Heat input to the concrete is controlled by setting the maximum base metal temperature to 25°F above the minimum preheat temperature, or a calculation is performed to provide specific allowances.

- NuScale's response is consistent with RG 1.243.

Industry experience on SC wall construction

- Similar SC wall application was reviewed and approved for AP1000 Shield Building.
- Integrated design process, which SC walls, were implemented on the AP1000 Shield Building design.
- For SC walls and connections, specification ANSI/AISC N690-18 was recently endorsed by RG 1.243.

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Thank You for Your Attention

Any Questions?

January 28, 2022 Docket No. 99902078

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

SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Full Committee Meeting: NuScale Building Design and Analysis Methodology for Safety-Related Structures," PM-112948, 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) meeting on February 2, 2022. The materials support NuScale's presentation of the topical report "NuScale Building Design and Analysis Methodology for Safety-Related Structures," TR-0920-71621, Revision 1.

The enclosure is the nonproprietary version of the presentation entitled "ACRS Full Committee Meeting: NuScale Building Design and Analysis Methodology for Safety-Related Structures," PM-112948, Revision 0.

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

If you have any questions, please contact Liz English at 541-452-7333 or at EEnglish@nuscalepower.com.

Sincerely,

Mark W. Shaver
Manager, Licensing
NuScale Power, LLC

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Enclosure: "ACRS Full Committee Meeting: NuScale Building Design and Analysis Methodology for Safety-Related Structures," PM-112948, Revision 0

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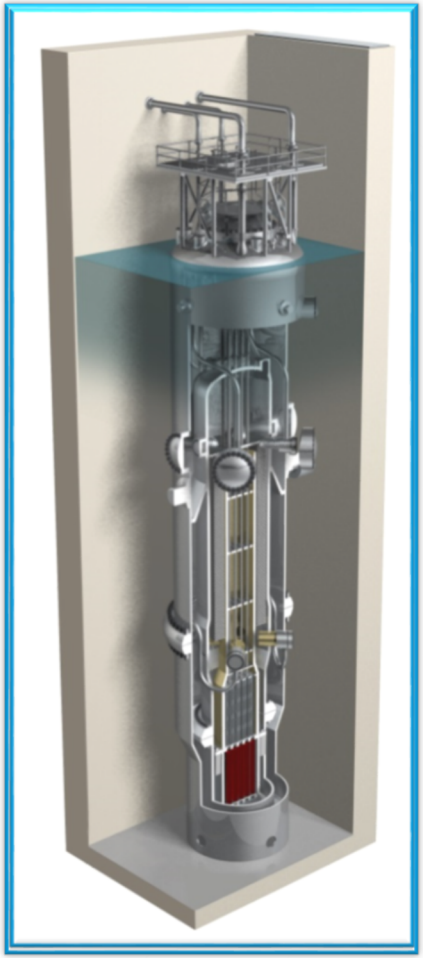
ACRS Full Committee Meeting



NuScale Building Design and Analysis Methodology for Safety-Related Structures

February 2, 2022

PM-112948
Revision:0

Presenters

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Engineer, Civil Structural

Giulio Leon Flores, P.E., S.E.
Engineer, Civil Structural

Agenda

- Purpose
- Introduction
- Technical Discussion
  - Steel-plate composite (SC) Walls
  - Reinforced concrete (RC) members
  - Effective stiffness modeling approach
  - In-structure response spectra (ISRS) and structural design process

Purpose

- Present technical content of topical report TR-0920-71621
- Provide the ACRS committee a general understanding of building design and analysis methodology for seismic Category I and II nuclear safety-related reinforced concrete (RC) and steel-plate composite (SC) structures applicable to NuScale design

Introduction – Topical Report

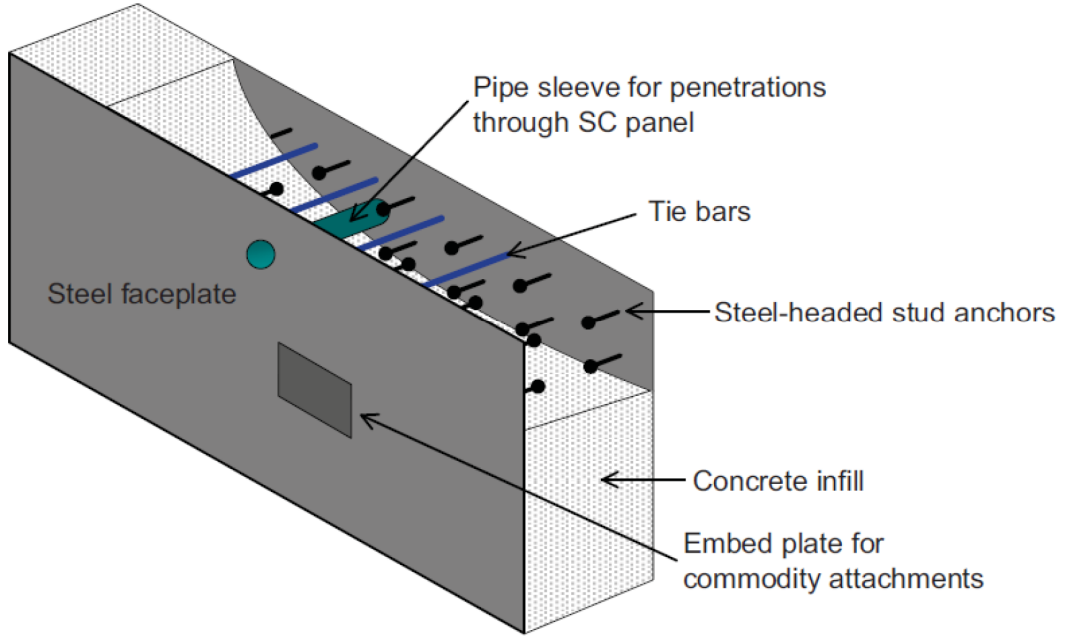
- Topical report presents a design methodology implementing new industry standards for nuclear facilities
- Applicable to new generation Small Modular Reactor (SMR) designs
- Complies with reinforced concrete and SC walls requirements (i.e. ANSI N690-18, ANSI/AISC 360-16, ACI 349-13, RG 1.243, RG 1.142, DSRS 3.7.2 and 3.8.4)
- Defines design methodologies to account for the interaction of SC walls with traditionally constructed RC members such as basemats, slabs, and roofs
- Implements the soil library methodology for complex structures as per NuScale topical report, “Improvements in Frequency Domain Soil-Structure-Fluid Interaction Analysis,” TR-0118-58005-P-A, Revision 2
- Topical report information will be used as part of Standard Design Approval Application (SDAA) submittal

Building Design – SC Walls

Rim Nayal, Ph.D., P.E.
Engineer, Civil Structural

Steel-Composite Walls

- Steel-Plate Composite Walls
  - Steel faceplates with concrete core
  - Anchors to ensure composite behavior
  - Ties to ensure integrity



SC Walls – Insight

- Advantages
  - ✓ Higher capacity and resistance to blast and earthquake
  - ✓ Modular construction
    - Fabrication and erection: time saving
    - Transportable: smaller and lighter modules
    - Design optimization: common module layouts repeated throughout the building
- Areas requiring special attention
  - ➢ Connection with other structural members
  - ➢ Corrosion of faceplates for below-grade walls

SC Walls – Material

- Material
  - Carbon steel (CS)
  - Stainless steel (SS) faceplates for ultimate heat sink (UHS) pool (borated water)
- Dissimilar-metal welds (DMWs): connecting CS and SS
- Welding requirements for DMWs are outside the scope of this topical report (LTR), general notes:
  - Utilize welding parameters and techniques based on industry experience appropriate for the welded metals
  - Concrete chemistry is controlled to avoid corrosion in DMWs
  - Leak chases at the module splices prevent borated water leakage inside the wall

SC Walls – Design

- SC Design methodology
  - *"Specification for Safety-Related Steel Structures for Nuclear Facilities,"* ANSI/AISC N690-18 & "*Specification for Structural Steel Buildings,"* ANSI/AISC 360-16
  - Load and resistance factor design (LRFD), load combinations based on AISC N690
  - Code limits on dimensional and material properties are applied
  - Required strength: averaging finite element responses
  - Available strength: for each type of loading and interaction
  - Demand-Capacity ratio for governing failure modes
- SC modules design parameters
  - Panel faceplate thickness
  - Anchor diameter/ tie dimensions and spacing

SC Walls – Corrosion

- LTR discusses multiple corrosion mitigation of below-grade walls based on the soil environment:
  - Minimum: Coating for below-grade protection of carbon steel
  - High chloride or hydrogen sulfide: Controlled Low Strength Material (CLSM) or shotcrete employed as a cementitious material
  - Controlled pH and chloride limits backfill placed and thoroughly compacted based on site-specific conditions
- Specific corrosion management plans are outside the scope of this LTR, subject of future applications

SC Walls – Connections

- Connection Design
  - AISC N690-18: Connections are designed to be stronger than the weaker of the connected parts to ensure ductility
  - Required strength: Full strength connection design, limited use of over-strength connection design
  - Available strength: Per ANSI/AISC N690-18, ANSI/AISC 360-16 and ACI 349-13 as required
  - Detailing: Connectors to transfer loads (e.g., studs, anchoring rebar, couplers, shear lugs, continuity plates)
- Detailing of connections is outside the scope of this LTR, will be presented in future applications
- Examples of connection configuration can be found in AISC DG-32

Building Design – RC Members

Rim Nayal, Ph.D., P.E.
Engineer, Civil Structural

Reinforced Concrete – Design

- RC members: basemat, intermediate floors, and roof slab
- RC design methodology
  - American Concrete Institute, ACI 349-13 *"Code Requirements for Nuclear Safety-Related Concrete Structures"* and ACI 318-08 *"Building Code Requirements for Structural Concrete"*
  - Load and resistance factor design (LRFD) and load combinations based on ACI 349
  - Code dimensional limits: spacing, thickness
  - Required strength: section cuts
  - Available strength: for each type of loading and interaction
  - Demand-Capacity ratio for governing failure modes

Reinforced Concrete – Design cont.

Design Conditions:

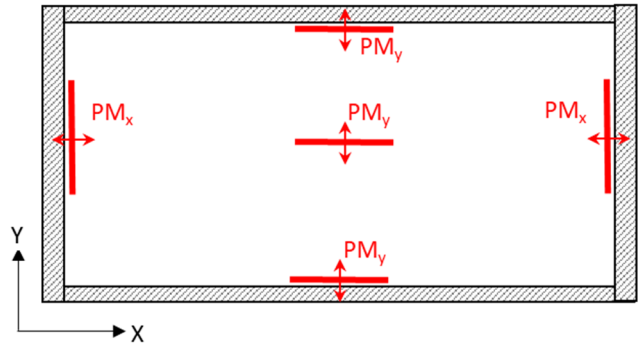
- Interaction of axial force – out-of-plane (OOP) moment
- Interaction of axial force – in-plane (IP) moment
- Interaction of axial force – out-of-plane shear
- In-plane shear

	IP Moment[1]	IP Shear[1]	OOP Moment	OOP Shear	Axial Load[2]
Slabs	✓	✓	✓	✓	✓
Basemat		✓	✓	✓	✓

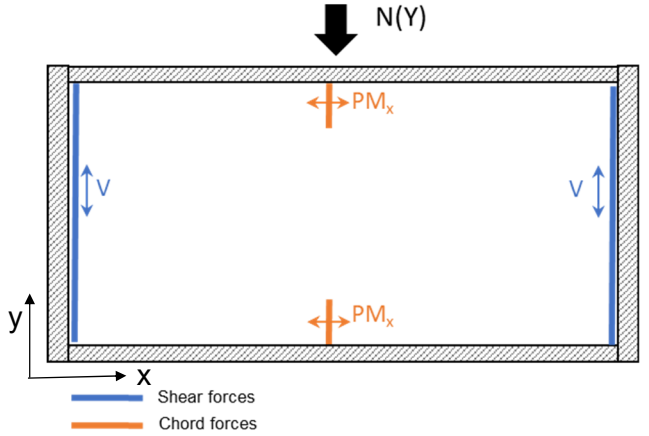
[1] Satisfies special seismic requirements of ACI 349 Chapter 21
[2] Axial load is used together with the IP and OOP actions

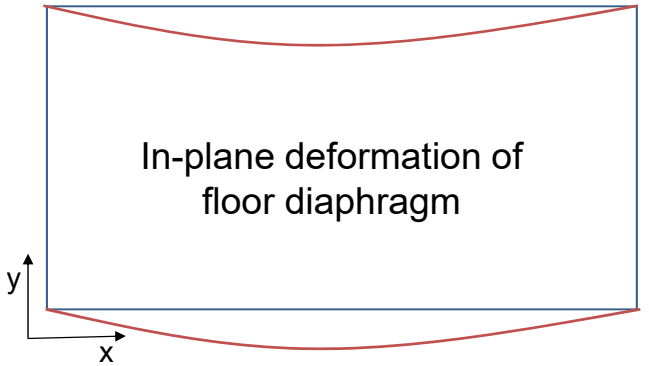
Reinforced Concrete – Demand

- Demand calculation:
  - Out-of-plane moments and axial loads
  - Out-of-plane shear
  - In-plane shear
  - In-plane moment



Vertical deformation of a floor slab

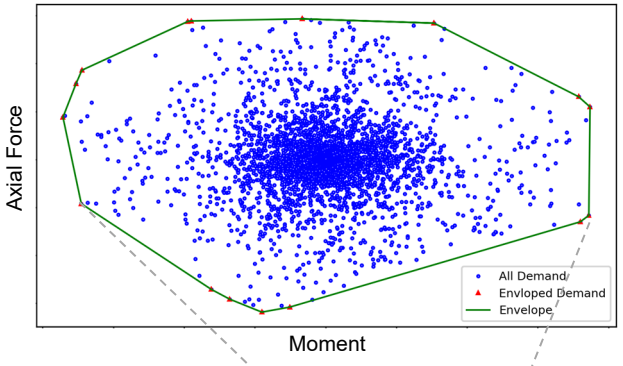


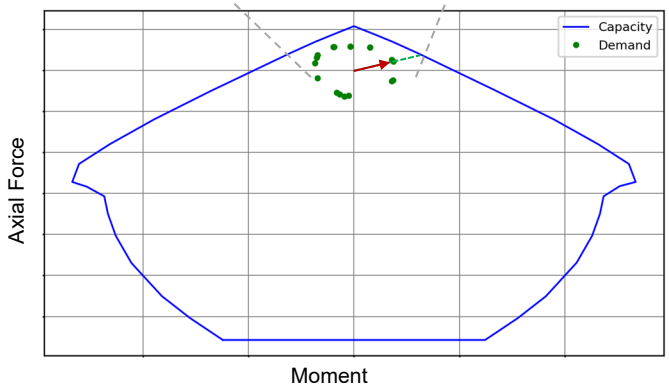


Reinforced Concrete – Capacity

- Member capacities obtained from ACI 349
- In-plane shear capacity according to Chapter 21 of ACI 349
- Interaction diagrams checks
- Demand-capacity ratio is calculated to assess safety margin in the design

Envelope of Demand





P-M Design Evaluation

$$DCR = \frac{D}{C}$$

Reinforced Concrete – Reinforcement

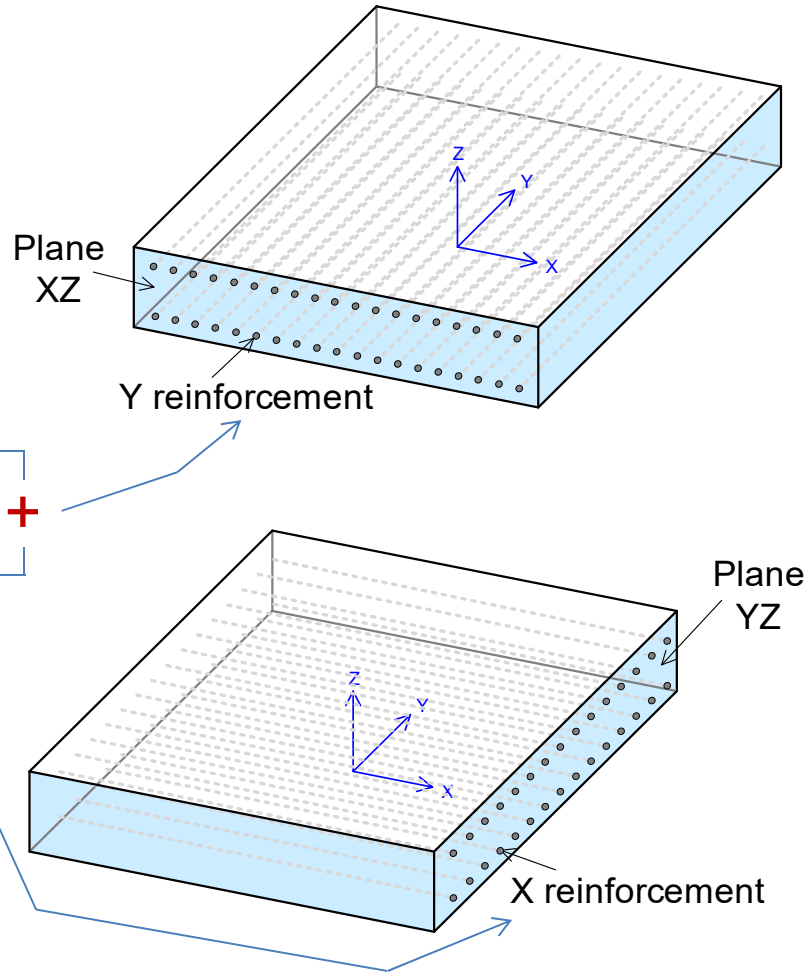
Total Slab Reinforcement

Section	Demand
XZ plane	IP shear
	OOP shear and axial load
	OOP moment and axial load
YZ plane	IP shear
	OOP shear and axial load
	OOP moment and axial load

use max each way

+

+



ISRS & Design Methodology

Giulio Flores, P.E, S.E.
Engineer, Civil Structural

Seismic Design Philosophy

- Buildings are designed for a specific target performance
  → level of deformation and damage, defined according to the Limit State (LS) categories
- ASCE 43-19 provides criteria to achieve a specific target performance
- NuScale Seismic Design Basis meets LS-D criteria for Structures, Systems and Components (SSCs)

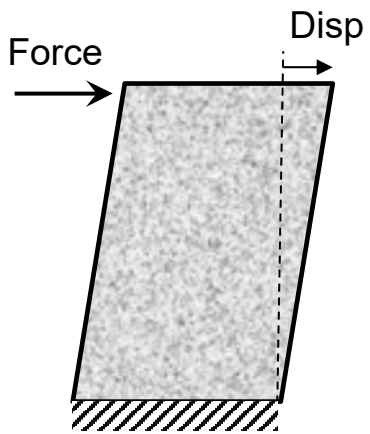
Limit State	Expected Deformation	Expected Damage
A	Large permanent distortion, short of collapse	Significant
B	Moderate permanent distortion	Generally repairable
C	Limited permanent distortion	Minimal
D	Essentially elastic behavior	Negligible

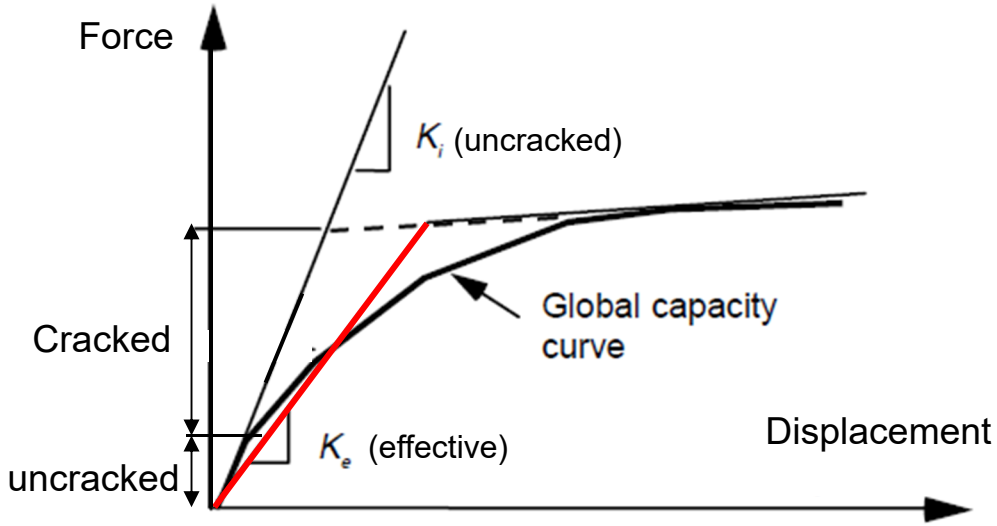
Seismic Design Philosophy

- In RC design, concrete takes compression loads while it is considered cracked under tension loads
- Reinforcing steel is designed to take all tension loads
- Concrete cracking is expected under SSE. However, structural design according to LS-D criteria is expected to result in essentially elastic behavior with negligible damage. In design codes, this is achieved through:
  - Use of lower material and member strengths
  - Use of load factors to increase the magnitude of design loads
- Thus, concrete cracking under SSE, if any, should not impair the member's safety function

Effective Stiffness Concept

- Due to cracking under SSE loading, the stiffness of cracked RC/SC members is reduced compared to their gross stiffness. Conversely, structural damping increases
- Effective stiffness and damping ratios are taken from codes and standards





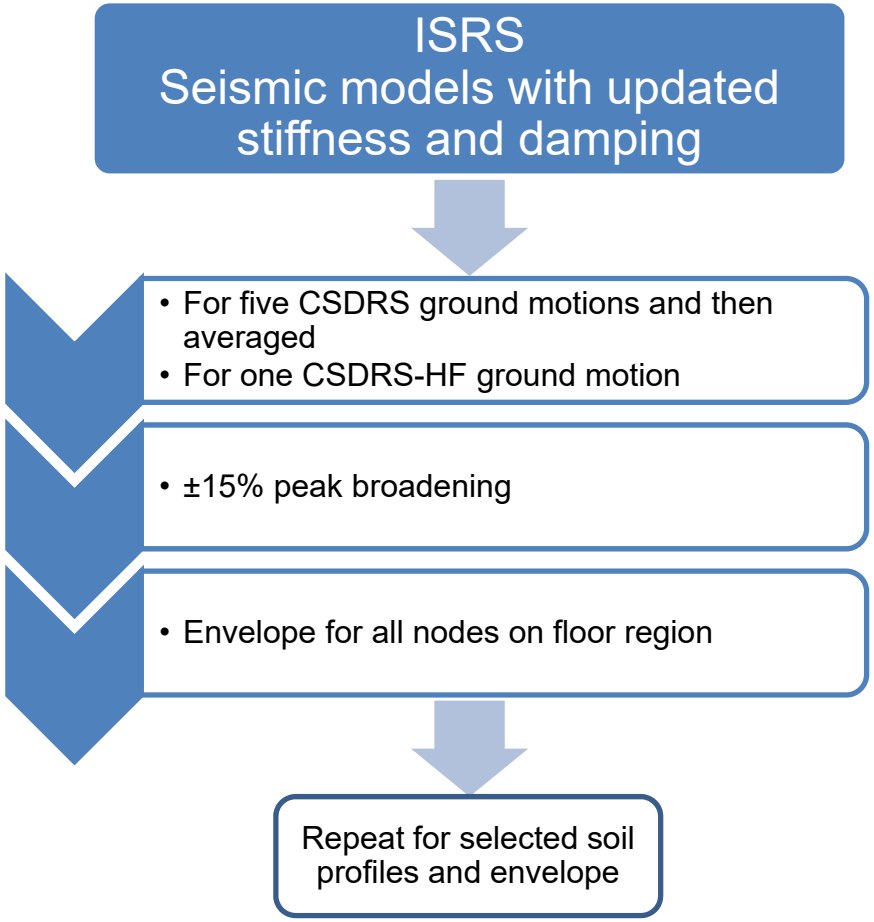
Effective Stiffness & Damping

- Damping values and effective stiffness used in linear elastic analysis depend on the level of cracking expected during the SSE
- In accordance with ASCE 4, a response level (RL) is determined on a member-by-member basis, based on the member stress, S:
  - If $S < S_{cr}$, RL-1 (uncracked) is used
  - If $S_{cr} \leq S$, RL-2 (cracked) is used
- For ISRS calculation, stiffness and damping are consistent with the expected RL
- For member design, stiffness is consistent with the expected RL and RL-2 damping values are used for all members as per ASCE 4

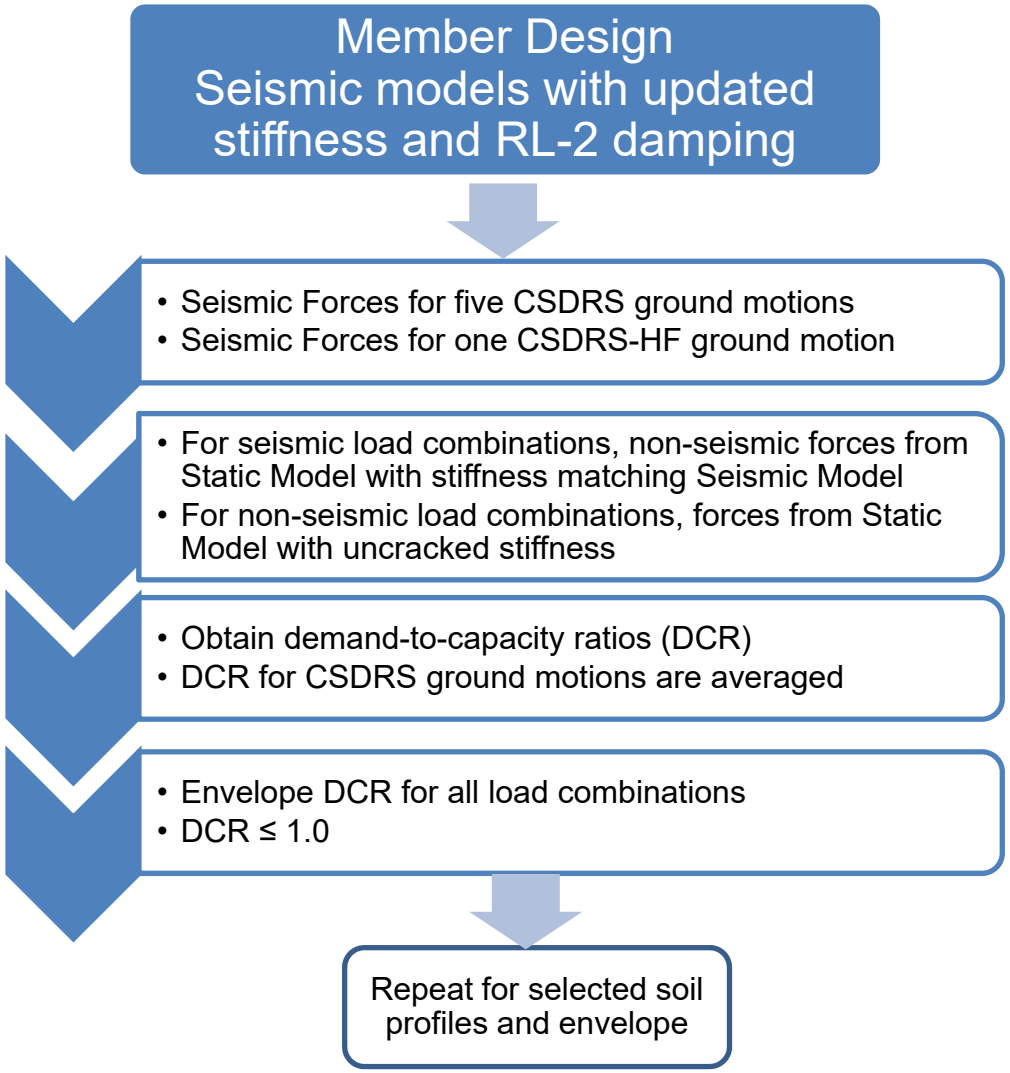
ISRS & Design Methodology

- Seismic and Static models are developed
- First step is determination of effective stiffness and damping for load combinations involving seismic loads,
  - Seismic models are set to uncracked damping and uncracked stiffness for all members, i.e. RL-1 is assumed for all members
  - Seismic shear and bending stresses are evaluated in all members considering the most critical seismic load combination
  - Stiffness and damping of members that are cracked are updated to the ones corresponding to RL-2

ISRS Methodology



Member Design Methodology

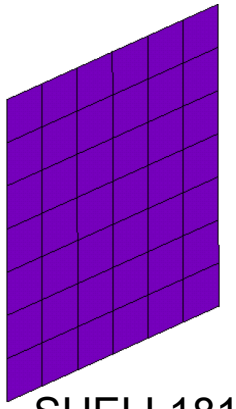


Effective Stiffness Modeling Approach

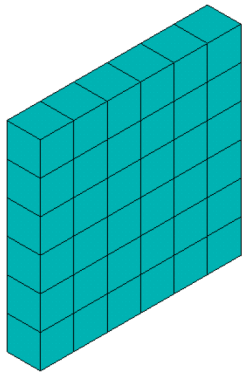
Giulio Flores, P.E, S.E.
Engineer, Civil Structural

Modeling Approach

- Finite element (FE) models are developed using ANSYS orthotropic material model
- Modeling of SC and RC walls and slabs can be performed using ANSYS SHELL181 or SOLSH190
- For SHELL181, effective stiffness is implemented by adjusting material properties and member thickness to match code-specified values (AISC DG-32 approach)
- Two methods are used for implementing effective stiffness values for SOLSH190. Both methods apply to RC members and SC walls and are consistent with AISC DG-32 approach



SHELL181

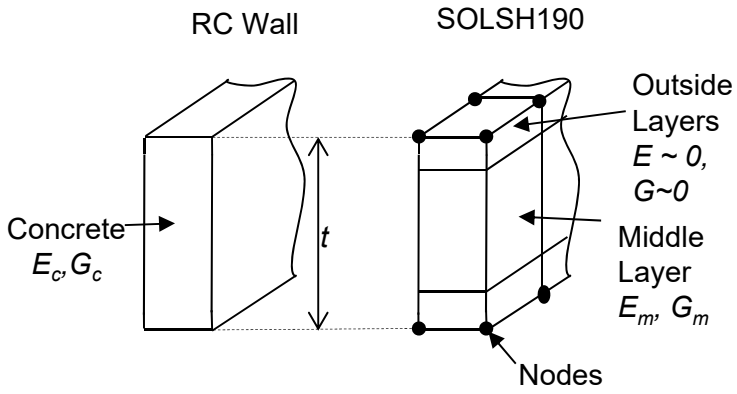


SOLSH190

Modeling Approach

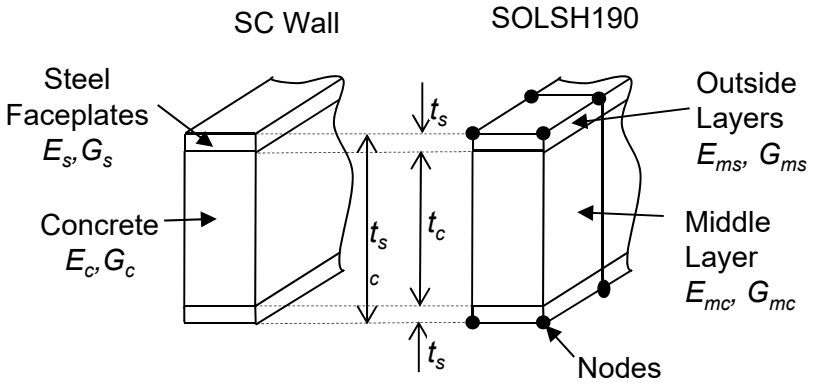
Method 1

- 3-layered SOLSH190
- Middle layer properties match code stiffness
- Outside layers are dummy
- More suitable to RC members



Method 2

- 3-layered SOLSH190
- Middle layer thickness matches concrete
- Outside layer matches faceplate thickness
- More suitable to SC walls



Summary

- Building design modifications
  - Use of SC wall panels
- Building design and analysis methodology
  - Topical report presents a design methodology implementing new industry standards for nuclear facilities
  - Applicable to new generation small modular reactor designs
- Implements the soil library methodology for complex structures as presented in the NuScale topical report, “Improvements in Frequency Domain Soil-Structure-Fluid Interaction Analysis,” TR-0118-58005-P-A, Revision 2
- Topical report information will be used as part of SDAA submittal

Acronyms

ACI	American Concrete Institute
ACRS	Advanced Committee on Reactor Safeguards
AISC	American Institute of Steel Construction
ANSI	American National Standards Institute
ASCE	American Society of Civil Engineers
CLSM	Controlled Low Strength Material
CS	Carbon Steel
CSDRS	Certified Seismic Design Response Spectra
CSDRS-HF	Certified Seismic Design Response Spectra - high frequency
DCR	Demand-to-Capacity Ratio
DG	Design Guide
DSRS	Design Specific Review Standard
DMW	Dissimilar-Metal Welds
FE	Finite Element
IP	In-Plane
ISRS	In-structure Response Spectra
LS	Limit State
LRFD	Load and Resistance Factor Design
LTR	Licensing Topical Report
OOP	Out-of-Plane
RC	Reinforced Concrete
RL	Response Level
SC	Steel-plate Composite
SDAA	Standard Design Approval Application
SMR	Small Modular Reactor
SS	Stainless Steel
SSC	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
UHS	Ultimate Heat Sink

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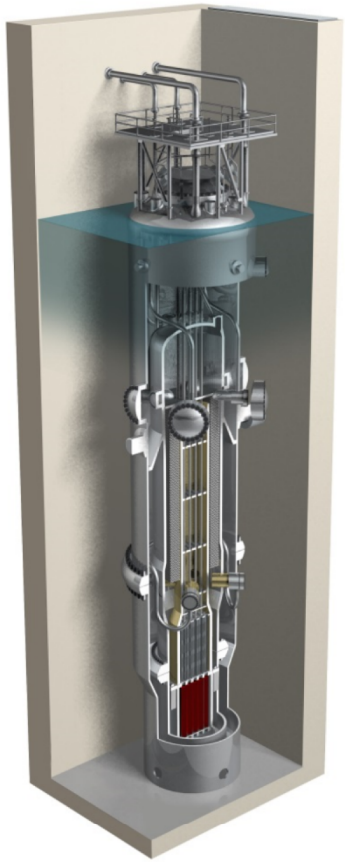

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Protecting People and the Environment

Advisory Committee on Reactor Safeguards (ACRS)
Full Committee

10 CFR Part 53 Subpart F
Staffing, Personnel Qualifications, Training, and Human Factors

February 2, 2022

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Agenda

3:45pm – 3:55pm	Opening Remarks & Staff Introductions
3:55pm – 5:00pm	Subpart F – Staffing, Personnel Qualifications, Training, and Human Factors
5:00pm – 5:15pm	Discussion

Welcome / Introductions

- **Welcome:**
  - Lauren Nist, Office of Nuclear Reactor Regulation (NRR)

- **Presenters:**
  - Jesse Seymour, NRR
  - Maurin Scheetz, NRR
  - Theresa Buchanan, NRR

- **Public Meeting Slides**:
  - ADAMS Accession No. ML22027A369

Subpart F – Staffing, Personnel Qualifications, Training, and Human Factors Requirements

Overview of Primary Staff Contributors (NRR & Office of Nuclear Regulatory Research)

- Theresa Buchanan, Senior Reactor Engineer (Examiner)
- Dr. David Desaulniers, Senior Technical Advisor for Human Factors and Human Performance Evaluation
- Dr. Brian Green, Human Factors Team Leader
- Dr. Niav Hughes Green, Human Factors Psychologist
- Dr. Stephanie Morrow, Human Factors Psychologist
- Lauren Nist, Branch Chief, Operator Licensing and Human Factors Branch
- Maurin Scheetz, Reactor Engineer (Examiner)
- Jesse Seymour, Reactor Operations Engineer (Human Factors)

Subpart F – Staffing, Personnel Qualifications, Training, and Human Factors Requirements

Presentation Topics

- Overview of Preliminary Rule Language
- Key Operations Staffing Considerations
  - Staffing Plans
  - Shift Technical Advisor
  - Certified Operators
- Simulator Considerations
- Regulatory Guidance Documents
- Questions

Subpart F – Staffing, Personnel Qualifications, Training, and Human Factors Requirements

Overview of §§ 53.750-789 Structure and Key Content

- §§ 53.750-759: General Requirements
  - § 53.753: Technical Requirements for operating license (OL) and combined license (COL) Applicants
    - Human Factors Engineering (HFE) design requirements
    - Human-System Interface (HSI) design requirements
    - Concept of Operations, Functional Requirements Analysis, and Function Allocation requirements
    - Staffing Plan requirements
    - Licensed & Certified Operator program requirements
  - § 53.755: Conditions of Licenses for OL and COL Holders
    - Provisions for not using licensed operators and criteria
    - Provisions for load-following
- §§ 53.760-769: Operator Licensing Requirements
  - Training, examination, requalification, and simulator requirements
- §§ 53.770-779: Operator Certification Requirements
- §§ 53.780-789: General Training and Qualification Requirements.

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Subpart F – Staffing, Personnel Qualifications, Training, and Human Factors Requirements

§§ 53.750-759: Staffing, Training, Qualifications, and Human Factors

- Fulfills role similar to aspects of § 50.34(f) post-Three Mile Island (TMI) requirements, § 50.54 conditions of facility licenses, and Part 55 operator licensing requirements
- Key areas now linked to design safety functions and their fulfillment:
  - HFE now required where needed to support safety function fulfillment
  - Operator staffing now required to the extent needed to support safety function fulfilment, versus reliance on prescribed numbers of operators
  - Licensed operator role centered on fulfilling/managing safety functions
- Includes criteria for when licensed operator staffing would not be required
  - 1st proposal - no mitigative actions by operators needed to meet safety criteria, safety functions, or provide defense in depth (as supported by probabilistic risk assessment) and structures, systems, and components performance needed for licensing basis event (LBE) response not reliant on humans
  - 2nd proposal - Design-basis accident safety criteria met without mitigative actions by operators, active engineered features, or passive design features (except those able to survive LBEs and resist credible human errors).

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Subpart F – Staffing, Personnel Qualifications, Training, and Human Factors Requirements

§§ 53.760 through 53.789 - Overview of Key Aspects

- §§ 53.760-769, Operator Licensing Requirements
  - Requires training programs to be based on systems approach to training (SAT) and ensure licensed operators possess the knowledge/abilities needed to protect public health and maintain plant safety functions
  - Incorporates facility-developed and NRC-approved examination programs that are tailored to the design specific operator roles
- §§ 53.770-779, Operator Certification Requirements
  - Requires training programs to be based on SAT and ensure non-licensed, certified operators possess the knowledge and abilities needed to protect public health and perform job duties
  - Uses facility-developed/NRC-approved, tailored exam programs
- §§ 53.780-789, General Training & Qualification Requirements
  - Builds upon the § 50.120, “Training Rule,” but adjusts timeframe for program establishment and updates personnel categories

Subpart F – Staffing, Personnel Qualifications, Training, and Human Factors Requirements

Part 53 Staffing Approach

- Accommodate novel concepts of operations and diverse technologies
- Prescriptive staffing ratios (like those for large light water reactors) may not be needed/appropriate to support safe operation
- Consider differences in staffing needs when:
  - Operators have a safety role
  - Operators do not have a safety role
- Conduct of Operations, Functional Requirements Analysis and Function allocation as input to staffing plan review
- Staff experience from recent review of NuScale small modular reactor staffing plans

Part 53 Staffing Plan Requirements

- Describe numbers, positions, and qualifications of reactor operators (RO) and senior reactor operators (SRO) (or certified operators) across all modes
- Describe personnel in other support roles (e.g., operations, maintenance, radiological protection, chemistry, fire brigades, engineering, security, and emergency response)
- Facilities with licensed operators: describe how the proposed staffing level is sufficient to provide assurance that plant safety functions can be maintained (must provide support via HFE-analyses and assessments)

Shift Technical Advisor (STA) Position in Part 53

- Prior concerns raised by the ACRS subcommittee have included:
  - Reservation about blanket STA elimination under rule
  - Value of having an independent individual for event assessment
  - Desirability of maintaining engineering expertise available
  - Relevance of role in light of uncertainties with new designs
- Considering three different options as part of staffing plan requirements:
  1. No requirement for STA
  2. STA required with provision for omitting STA with justification
  3. Requirement for engineering expertise that is independent from and readily available to the on-shift operators (for certified and licensed operators)

Subpart F – Staffing, Personnel Qualifications, Training, and Human Factors Requirements

Certified Operators – Background

The staff proposed the option of non-licensed, certified operators for facilities meeting specific requirements as an alternative to SROs & ROs

- Certified operator would be responsible for important administrative functions that would otherwise be performed by SROs
- Certified operator staffing would need to provide a continuity of responsibility for facility operations during the operating phase, including monitoring of fueled units with the following capabilities:
  - Receiving plant operating data and parameters
  - Ability to immediately initiate a reactor shutdown
  - Ability to promptly dispatch ops/maintenance personnel
  - The ability to implement any emergency plan responsibilities
  - Conducting reactivity manipulations that require human action

Subpart F – Staffing, Personnel Qualifications, Training, and Human Factors Requirements

Reasoning Behind the Certified Operator Alternative

- If a facility lacks an operator role in safety (e.g., an autonomous reactor design), then a key driver warranting federal licensing of individuals is removed (i.e., operator performance would not have a meaningful influence on public health and safety outcomes within that context)
  - Regardless of whether the operators were licensed, the facility itself would still be licensed by the NRC
- Important administrative job tasks that would remain still need to be accomplished by adequately qualified personnel.
  - Precedent shows that similar administrative tasks have been fulfilled by non-licensed personnel, such as Certified Fuel Handlers
- Durable rule should account for future safety advancements

Subpart F – Staffing, Personnel Qualifications, Training, and Human Factors Requirements

Certified Operators versus Licensed Operators Considerations

- Prior concerns raised by the ACRS subcommittee have included:
  - Differences in accountability compared to licensed operators
  - Lessened ability to resist coercion by inappropriate management
  - Redundancy to scalable operator licensing provisions
  - Lack of certification by an independent entity (i.e., the NRC)
- At present, the staff perspective remains that the certified operator alternative is appropriate based upon the following considerations:
  - The framework should be able to efficiently account for staffing requirements when there is no significant human role in safety
  - Precedent for using non-licensed personnel in comparable roles
  - Designers have indicated that they may be able to meet criteria and have expressed potential demand for such an alternative
  - Ability to administratively assign responsibilities to management
  - Effects of a potential STA or engineering expertise requirement
  - Regulatory approval and oversight of certification programs

Subpart F – Staffing, Personnel Qualifications, Training, and Human Factors Requirements

Simulator Scope – Background

- Part 53 preliminary establishes simulation facility requirements for plants with licensed operators, along with less stringent simulation facility requirements for plants with certified operators; some key aspects include the following:
  - Full-scope simulators are not mandated; partial scope simulators may be acceptable, provided that the scope is adequate to meet intended usage; alternatives to simulators are possible as well
  - Simulation facilities for plants with licensed operators must be approved by the Commission if the facility licensee will rely upon them for training, experience requirements, or for initial or requalification examinations
    - Equivalent approval not required for certified operator facilities
  - Must demonstrate that adequate simulator scope is provided to support HFE analyses/assessments in order to use a simulation facility for conducting these analyses/assessments

Subpart F – Staffing, Personnel Qualifications, Training, and Human Factors Requirements

Simulator Scope

- In developing preliminary rule language, staff reviewed Section 306 of the Nuclear Waste Policy Act (NWPA) and 52 FR 9453 which discussed implementation of the Act's simulator-related provisions:
  - Flexibilities historically provided to allow for potential use of the plant itself, and/or a plant-referenced simulator, and/or some other type of simulation device (such as a part-task or basic-principles simulator) for the conduct of the simulator portion of the operating test
  - The NRC's stated intent was not to permit the initiation of transients on the plant itself if used as a simulation facility; rather, the use of the plant was envisioned as an option that might be used in conjunction with another simulation device or devices, in lieu of a plant-referenced simulator
- Current perspective is that NWPA does not mandate NRC to require that plants have *simulators,* but instead requires regulations address the use of *simulations* in training; flexibility exists to allow the use of the actual plant to "simulate" tasks for training and operating test purposes without having a separate simulator
- Prior concerns raised by the ACRS subcommittee have included the potential for reductions in training and evaluative efficacy, impacts on procedure quality, reduced support of analyses, and staff experience in the approval of partial scope simulators

Subpart F – Staffing, Personnel Qualifications, Training, and Human Factors Requirements

Simulator Scope (cont'd)

- Philosophical basis behind preliminary rule language is:
  - Plant-referenced, full-scope simulators remain the preferred approach and would represent the best route for meeting Part 53 requirements
    - Staff expect majority of Part 53 applicants will have them due to regulatory certainty and technology lowering the associated costs
  - Existing regulations do not strictly mandate plant-referenced, full-scope simulators either, but still adopted by all current power reactors
  - Part 53 rule language leaves alternatives to simulator usage (full-scope or otherwise), but the burden will be on the applicant to demonstrate how the following are supported:
    - Licensed or certified operator training and exams; simulators used require sufficient scope and fidelity for operators to acquire and demonstrate knowledge and abilities needed for job duties.
    - Experience requirements (i.e., reactivity manipulations)
    - HFE analyses/assessments and HSI design testbed needs
  - Additional staff review guidance may be needed, such as to support reviews of partial scope simulation facilities

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Subpart F – Staffing, Personnel Qualifications, Training, and Human Factors Requirements

Regulatory Guidance Development Overview

- HFE Review Guidance
  - Supports scalable reviews, developed with Brookhaven National Laboratory support, goal is draft by June
- Staffing Plan Review Guidance
  - Facilitates review of staffing plans using NUREG-1791, goal is draft by June
- Operator Licensing Examination Review Guidance
  - Supports review of tailored programs, developed with Idaho National Laboratory, goal is draft by June
- SAT-based Training Program Review Guidance
  - Supports the review of non-accredited training programs; developed by staff
  - Updates the existing, dated SAT review guidance of NUREG-1220 & IP 41500
  - Current development goal is 1 year to support near-term applicants as needed
  - Team includes HQ and regional operator licensing staff (inc. former instructors)
- Advanced Reactor Content of Application Project, ISG Chapter 11, “Organization and Human-System Considerations,” as supplemented with guidance for Part 53
  - Will support other review areas beyond those covered above or by existing guidance (e.g., load following, post-TMI items, simulation facilities, etc.)

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Final Discussion and Questions

Acronyms and Abbreviations

ACRS	Advisory Committee for Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System
COL	Combined license
FR	*Federal Register*
HFE	Human factors engineering
HSI	Human-system interface
ISG	Interim staff guidance
IP	Inspection procedure
LBE	Licensing basis event

NRR	Office of Nuclear Reactor Regulation
NUREG	U.S. NRC technical report designation
NWPA	Nuclear Waste Policy Act
OL	Operating license
RO	Reactor operator
SAT	Systems approach to training
SRO	Senior reactor operator
STA	Shift technical advisor
TMI	Three Mile Island

North Anna Power Station

Units 1 and 2

Subsequent License Renewal Application

ACRS Full-Committee Meeting

February 2, 2022

Dominion Energy®

Agenda

- Station Overview/Performance
- SLR Application Development
- SLR Aging Management Programs
- Technical Topics
- Closing Remarks

North Anna Power Station

Dominion Energy

Station Overview

	Unit 1	Unit 2
Full Power License – 2,775 MW_t	April 1, 1978 (Operating License Issued)	August 21, 1980 (Operating License Issued)
Independent Spent Fuel Storage Loading (ISFSI)	Pad 1: 1998, Pad 2: 2008, Pad 3: 2021	
4.3% Power Uprate to 2,893 MW_t	1986	
First License Renewal Approval	2003	
1.6% MUR to 2,940 MW_t	2010	
Entered Period of Extended Operation	April 1, 2018	August 21, 2020
Current License Expiration	April 1, 2038	August 21, 2040

Dominion Energy

North Anna Performance

- North Anna operates on an 18-month refueling frequency
- Plant Capacity Factor:
  - 2019: U1 – 94.53% U2 – 90.38%
  - 2020: U1 – 101.7% U2 – 88.41%
  - 2021: U1 – 83.32% U2 – 102.21%
- Regulatory Status
  - ROP Actions Matrix Column 1
  - All ROP Indicators are Green

Significant Plant Modifications

North Anna Power Station	Unit 1	Unit 2
A/B RSS Transformer Replacement	2021	2021
C RSS Transformer Replacement	2019	2019
Reactor Vessel Upflow Conversion	1996	2018
Reactor Vessel Head Replacement	2003	2003
Main Transformer Replacement	2006	2006
Station Service Transformers Replacement	2010	2010
Main Generator Replacement	2014	2008
Underground Fire Protection Piping Replacement	2012	2012
Fire Detection System Replacement	2021	2012
Service Water Spray Array Piping Replacement	2009	2009
AL-6XN Service Water Charging Piping Replacement	2001	N/A
Service Water Instrument Air Compressor Dryer Replacement	2003	2003
Underground Security Diesel Generator Fuel Oil Supply Tank Replacement	2015	N/A
Flux Thimble Tube Replacements	2016	N/A

SLR Application Improvement

- Implementation of Lessons Learned from Surry
  - Fewer Than Half the Enhancements
  - More Efficient Review – Half the RAIs
- Improved Consistency with GALL SLR
  - Led Industry/NRC Effort to Identify and Issue ISGs
  - Reduced Number of AMPs with Exceptions

Integrated Plant Assessment

First License Renewal (FLR) and Subsequent License Renewal (SLR) deltas

- Scoping & Screening
  - Minimal Differences from FLR (pre-GALL)
  - Some updates required to address 10 CFR 54.4(a)(2)
  - Followed NUREG-2191 (GALL-SLR) and NUREG-2192 (GALL-SRP)
- Aging Management Reviews
  - NAPS FLR was pre-GALL, additional aging effects required disposition based on NUREG-2191 (GALL-SLR)
  - High SLR AMR Consistency (99.7% Notes A thru E)
- Aging Management Programs
  - FLR – 25 AMPs
  - SLR – 48 AMPs
- License Renewal Commitments
  - UFSAR Supplement (Appendix A)
  - Implementation activities have begun and will continue following issuance of renewed license

North Anna SLR AMP Considerations

- Incorporation of operating experience (OE):
  - Industry and plant specific OE reviewed for a 10-year period
  - Reviewed Surry and Industry RAIs for AMP insights
  - Participation in Industry Peer Reviews

- AMP Effectiveness Reviews performed on all first license renewal AMPs using elements of NEI 14-12

- All First License Renewal (FLR) AMPs will be continued and incorporated into SLR AMPs

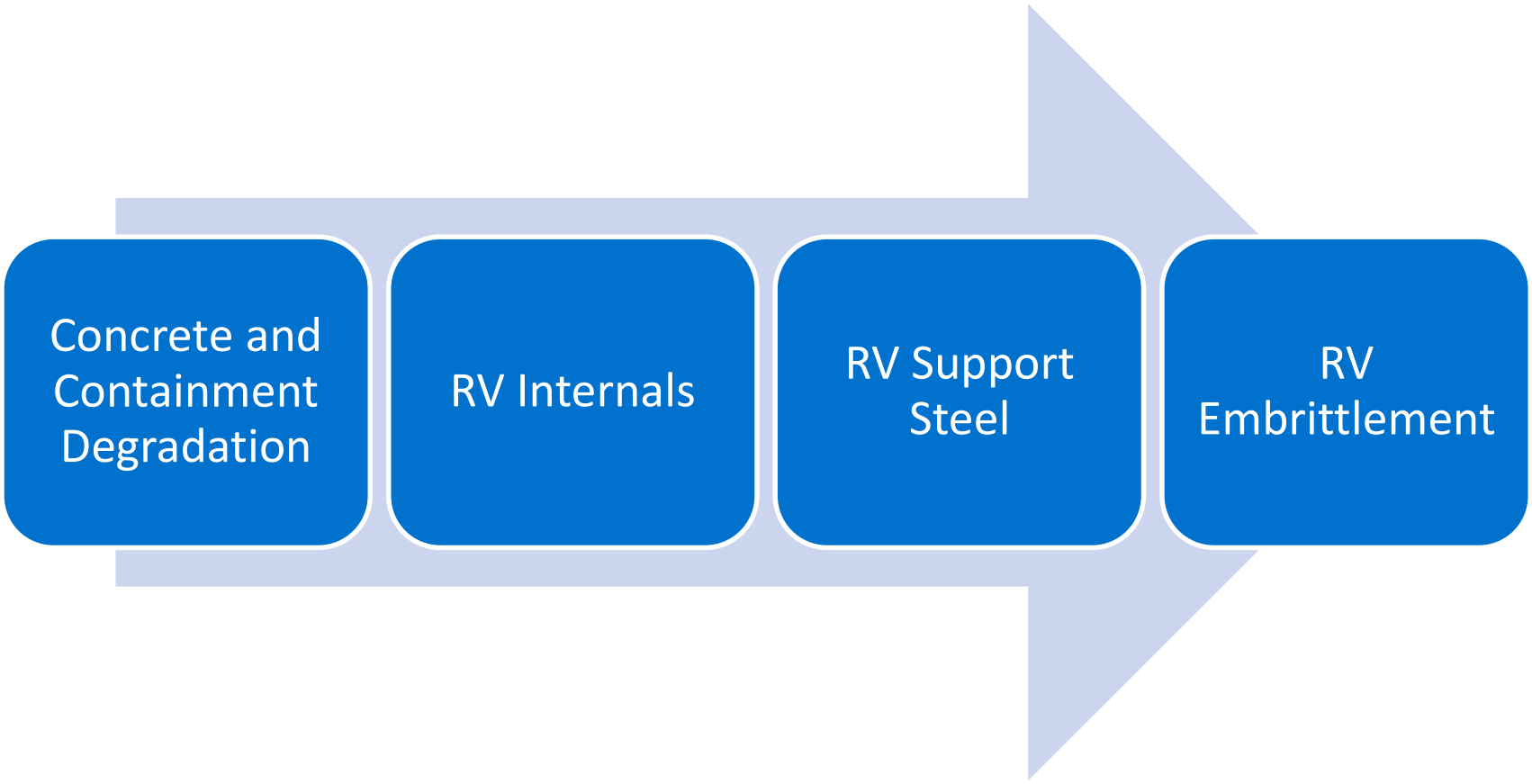
North Anna SLR – 48 GALL-AMPs

	Consistent with GALL-SLR	With Enhancement	With Exception	Exception and Enhancement	Plant Specific
Existing 41	**16**	**18**	**3**	**4**	**0**
New 7	**7**	**0**	**0**	**0**	**0**
Total 48					

FLR AMP Effectiveness

- FLR AMPs have been evaluated for AMP effectiveness
  - Identified gaps have been included in the CAP system as described in Appendix B
  - Periodic AMP effectiveness reviews are required to be completed by the program owners every 5 years
- OE is systematically reviewed on an on-going basis
- Training is conducted periodically for program owners
- IP 71003 Phase 2 inspection identified no findings or concerns in 4Q17

Technical Topics



Dominion Energy

Dominion Energy SLR Summary

- North Anna has a high degree of consistency with GALL-SLR, which resulted in a high quality SLR Application that includes lessons learned from Surry's SLRA
- North Anna SLRA was developed using the same highly experienced team used for Surry SLR
- Dominion Energy has committed future investments in people, program enhancements and equipment modifications for the SPEO

Advisory Committee on Reactor Safeguards

North Anna Power Station, Units 1 and 2 Subsequent License Renewal Application (SLRA) Safety Evaluation Report (SER)

February 2, 2022

Lois M. James, Senior Project Manager
Office of Nuclear Reactor Regulation

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Presentation Outline

- North Anna Licensing History
- North Anna Aging Management Programs
- Specific Technical Areas of Review
- Inspections and Plant Material Conditions
- Conclusion on North Anna SLRA Review

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North Anna, Units 1 & 2: Licensing History

Initial License Renewal

Unit	Initial License	Initial License Renewal Application	Renewed License	Expiration Date
1	4/1/1978	5/29/2001	3/20/2003	4/1/2038
2	8/21/1980	5/29/2001	3/20/2003	8/21/2040

Subsequent License Renewal

Application Submitted	8/24/2020
Acceptance Determination	10/15/2020
Draft Safety Evaluation Report with No Open or Confirmatory Items	10/18/2021
Final Safety Evaluation Report	1/3/2022

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North Anna
Aging Management Programs

SLRA - Original Disposition of AMPs	SER - Final Disposition of AMPs
○ 7 new programs • 7 consistent ○ 41 existing programs • 17 consistent • 24 consistent with enhancements and/or exceptions	○ 7 new programs • 7 consistent ○ 41 existing programs • 16 consistent • 25 consistent with enhancements and/or exceptions

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Specific Technical Areas of Review

- Reactor Pressure Vessel Neutron Fluence
- Irradiation Effects on Reactor Internals
- Buried Gray Cast Iron Piping

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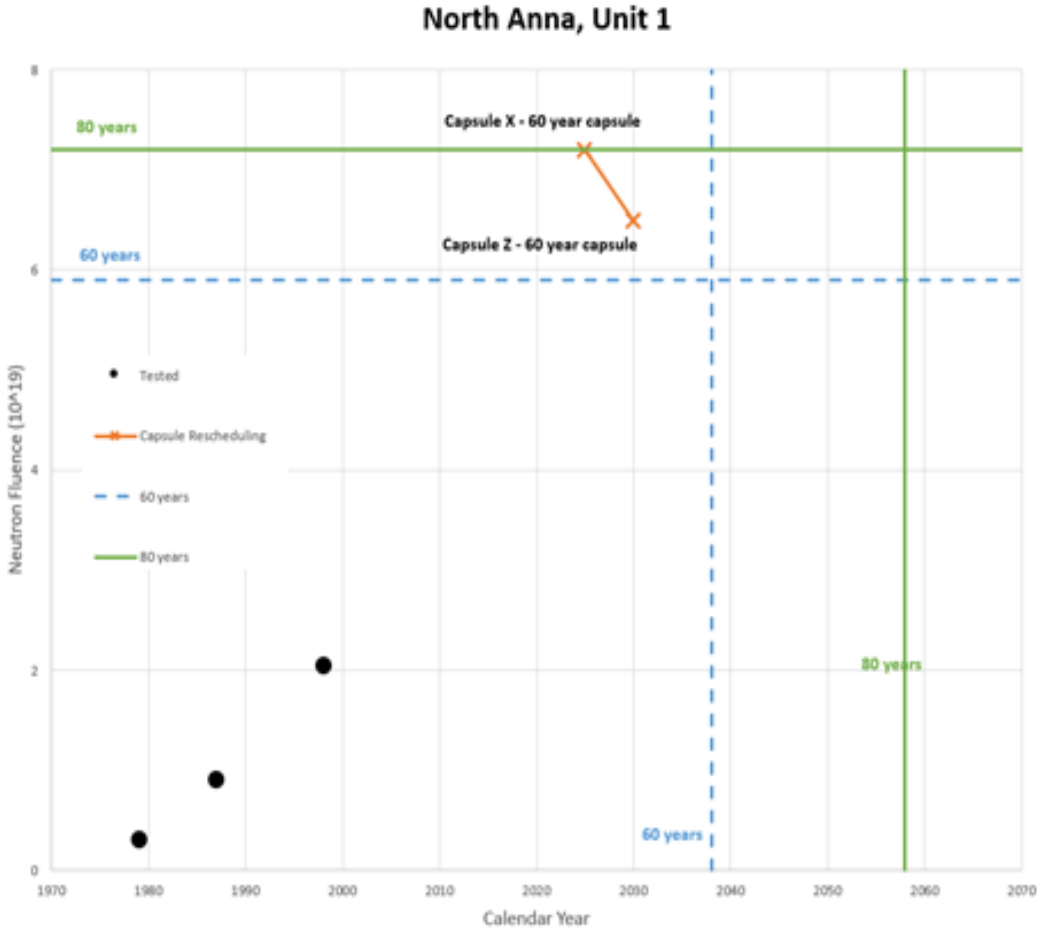
Reactor Pressure Vessel Neutron Embrittlement

- Unit 1 – 80-year projected peak neutron fluence of 7.20 x 10^{19} n/cm^2 (E>1.0 MeV)
  - Capsule X achieves 80-year projected peak neutron fluence at approximately 39.1 EFPY (~2025)
- Unit 2 – 80-year projected peak neutron fluence of 7.34 x 10^{19} n/cm^2 (E>1.0 MeV)
  - Capsule X achieves 80-year projected peak neutron fluence at approximately 39.3 EFPY (~2026)



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Reactor Pressure Vessel Neutron Embrittlement

- Upper Shelf Energy (USE)
  - Limiting USE value at 72 EFPY
    - 50.0 ft-lbs for the Unit 1 Inlet Nozzle Forging 11
    - 48.2 ft-lbs for the Unit 2 Intermediate Shell Forging 04
  - Applicant conservatively performed equivalent margin analysis (EMA) for:
    - Upper and Intermediate Shell Forgings
    - Inlet and Outlet Nozzle Forgings and Welds
  - Analyses have been projected to be no less than 50 ft-lbs or EMA demonstrated lower values of Charpy USE will provide margins of safety against fracture to the end of the SPEO
- Pressurized Thermal Shock (PTS)
  - Limiting RT_{PTS} value for base metal or longitudinal weld materials at 72 EFPY
    - 212.2° F - Unit 2 Lower Shell Forging 03
  - Limiting RT_{PTS} value for circumferentially oriented welds at 72 EFPY
    - 136.3° F - Unit 1 Intermediate to Lower Shell Circumferential Weld Heat # 25531
  - PTS analyses have been projected to be below screening criteria in 10 CFR 50.61 at the end of the SPEO

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Irradiation Effects on Reactor Internals

- Review Basis: The staff reviewed the 80-year neutron fluence values for the reactor vessel internals (RVI) components as part of the staff review of the RVI gap analysis that was included as part of SLRA AMP B2.1.7, PWR Vessel Internals.

- Staff Verification: The staff verified that the 72-EFPY neutron fluence values for the RVI components were adequately addressed in the applicant's gap analysis.

- Reasonable Assurance: The staff determined the applicant's PWR Vessel Internals Program (including gap analysis) adequately addresses the inspection of RVI components during the SPEO

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Buried Gray Cast Iron Piping in Fire Protection System

- 6 ruptures prior to 2003
- Identified as cracking due to cyclic loading – during pump start testing
- Changed test procedure in 2003 to limit pressure in downstream piping – no ruptures since then
- Multiple inadvertent pump starts have occurred since 2003 without failures
- Material is also subject to selective leaching
- Material is brittle

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Buried Gray Cast Iron Piping in Fire Protection System (cont)

- Enhancements 5 (for selective leaching) and 6 (for cracking) to the Buried and Underground Piping and Tanks Program
- 6 gray cast iron locations to be excavated each 10 years
  - 5 piping 10-foot lengths and 1 piping or component for Selective Leaching program
  - Visual and magnetic particle to inspect for cracking
  - Combination of radiography and destructive testing on a 1-ft section if cracking is identified, to determine cause
    - If defect is from manufacturing and not aging: document results
    - Cracking due to aging will result in crack growth and flaw stability evaluations to end of subsequent PEO
    - Projected loss of function will be entered into Corrective Action Program for extent of condition, extent of cause and further follow-on actions

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Buried Gray Cast Iron Piping in Fire Protection System (cont)

Two nonconcurrences were received during the review and concurrence process regarding the buried gray cast iron piping in fire protection system.

Both nonconcurrences were resolved in the final SER.

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Region II
AMP Inspections

License Renewal Inspection Program for Initial Period of Extended Operations

Inspection	Dates	Results
U1 & U2 IP 71003 Phase 1	September 19-22, 2016 ML16306A189	No Findings
U1 IP 71003 Phase 2	Nov 27-Dec 15, 2017 ML18029A029	No Findings
U2 IP 71003 Phase 2	March 11-15, 2019 ML19134A146	No Findings
U1 & U2 IP71003 Phase 4	TBD	

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Region II: AMP Inspections
ROP Baseline Inspections

Inspection	Date	Aging Management Program
IP71111.08 ISI	Annually alternate units	Augmented Inspection Activities Boric Acid Corrosion Surveillance ISI Program – Component and Component Support Inspections ISI Program – Containment Inspections ISI Program – Reactor Vessel Reactor Vessel Internals Inspection Steam Generator Inspections
IP71111.07T Heat Sink	2006, 2008, 2011, 2014, 2017, 2020	Service Water System Inspections
IP71111.21M DBAI	4Q 2018 4Q 2021	Ensure the selected SSCs that are subject (operating in the post-40-year licensing period) to aging management review pursuant to 10 CFR Part 54 are being managed for aging in accordance with appropriate aging management programs.
IP71152 PI&R Sample Age-related capacitor degradation resulted in a reactor trip	2Q 2018	Preventive Maintenance Program

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Region II
AMP Inspections

Resident Inspector Insight and Inspection Results

- No findings from License Renewal Program inspections
- 2013: Green finding for the failure to failure to establish and implement appropriate periodic preventive maintenance for replacement frequency of the C4 capacitor on the Speed Error Amplifier card B (1A08D). The C4 capacitor failed due to age related degradation. (FIN 05000339/2013007-01)
- 2021: Green NCV for an inadequate procedure for handling age degraded safety related cable. (NCV 05000338,05000339/2021010-03)

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Region II: Plant Material Condition + Conclusion

- Plant material condition is generally acceptable and meets regulatory requirements for systems, structures, and components.
- The inspectors found that the AMPs were being implemented in accordance with the license condition.
- The NRC will continue to monitor AMPs using the baseline Reactor Oversight Process.

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SLRA Review Conclusion

On the basis of its review of the SLRA, the staff determined that the requirements of 10 CFR 54.29(a) have been met for the subsequent license renewal of North Anna Power Station, Units 1 and 2.