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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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669TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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OPEN SESSION

+ + + + +

WEDNESDAY

DECEMBER 4, 2019

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ROCKVILLE, MARYLAND

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The Advisory Committee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2D30, 11545 Rockville Pike, at 1:00 p.m., Peter Riccardella, Chairman, presiding.

COMMITTEE MEMBERS:

PETER RICCARDELLA, Chairman

MATTHEW W. SUNSERI, Vice Chairman

JOY L. REMPE, Member-at-Large

RONALD G. BALLINGER, Member

DENNIS BLEY, Member

CHARLES H. BROWN, JR. Member
VESNA B. DIMITRIJEVIC, Member
WALTER L. KIRCHNER, Member
JOSE MARCH-LEUBA, Member
DAVID A. PETTI, Member

ACRS CONSULTANTS:
MICHAEL L. CORRADINI*
STEPHEN SCHULTZ*

DESIGNATED FEDERAL OFFICIALS:
KENT HOWARD
MIKE SNODDERLY

*Present via telephone

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P R O C E E D I N G S

1:00 p.m.

CHAIRMAN RICCARDELLA: The meeting will come to order.

Scott?

MR. BROWN: Sure. I would just like to announce for the Committee that we have a new member on the ACRS staff, Thomas Dashiell. Thomas comes to us to be our Conference Room Manager, which we badly need, as you all have seen. Thomas has served for years in the Navy, retired with honors from the Navy. We won't hold that against you, Thomas.

And following that, he's been here at NRC for 15 years as an AV Project Manager, IT Project Manager. While he was in the Navy, he worked directly under two Presidents. So, he comes with high credentials. And here at NRC, he worked the AV equipment for the Commission itself in the hearing rooms and in the auditorium. So, he comes with high skills and we're glad to have him on our staff.

So, we're glad you're here, Thomas. Thanks.

CHAIRMAN RICCARDELLA: Welcome, Thomas.

So, this is the first day of the 669th meeting of the Advisory Committee on Reactor

Safeguards. I'm Pete Riccardella, Chairman of ACRS.

ACRS was established by the Department of Energy Act and is governed by the Federal Advisory Committee Act, or FACA. The ACRS section of the U.S. NRC public website provides information about the history of the ACRS and provides FACA-related documents, such as our Charter, Bylaws, Federal Register notices for meetings, letter reports, and transcripts of all full and subcommittee meetings, including slides and presentations at the meetings.

The Committee provides its advice on safety matters to the Commission through its publicly-available letter reports. The Federal Register notice announcing the meeting was published on November 18th, 2019, and provided an agenda and instructions for interested parties to provide written documents or request opportunities to address the Committee, as required by FACA.

In accordance with FACA, there is a Designated Federal Official for the meeting. The DFO for today's meeting is Mr. Kent Howard.

During this meeting, the Committee will consider the following: Peach Bottom subsequent license renewal; NuScale Source Term Topical Report methodology; Susquehanna Atrium 11 fuel transition and application/Framatome, and preparation of reports.

As reflected in the agenda, portions of the NuScale and Atrium 11 sections may be closed in order to discuss the proprietary information designated as sensitive or proprietary information.

There is a phone bridge line. To preclude interruptions of the meeting, the phone will be placed in a listen-in mode during the presentations and Commission discussions. We have received no written comments or requests to make oral statements from members of the public regarding today's session. There will be an opportunity for public comment, as we have set aside 10 minutes in the agenda for comments from members of the public attending or listening into our meeting. Written comments may be forwarded to Mr. Kent Howard, the Designated Federal Official.

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

So, the first topic on the agenda is Peach Bottom Atomic Power Station subsequent license renewal application, and I will turn the meeting over to Matt Sunseri, who is Chairman of the License Renewal

Subcommittee.

VICE CHAIRMAN SUNSERI: Thank you, Chairman Riccardella.

As Pete mentioned, I'm Matt Sunseri, Chairman of the Plant License Renewal Subcommittee.

The purpose of this full Committee meeting is for Exelon Generation Company LLC and the NRC staff to brief the full Committee on the subsequent license renewal application for the Peach Bottom Atomic Power Station's Units 2 and 3. The Plant License Renewal Subcommittee previously met on November 5th of this year to discuss the matter.

At the conclusion of these presentations, we will be ready to start our Committee work on letter writing at your pleasure following this briefing. So, anytime after that.

There are members of both the NRC and Exelon staff listening in on the phone. So, this reminder about using the microphones is particularly important because they just can't hear us if we don't do that.

At this point, I'd like to turn to Meena Khanna to see if she has any opening remarks as well.

MS. KHANNA: Thank you. Thank you, Chairman Riccardella and Subcommittee Chairman

Sunseri, and Members of the ACRS.

I am Meena Khanna, Acting Deputy Director of the Division of New and Renewed Licenses, which is DNRL. We sincerely appreciate the opportunity today to present to the ACRS full Committee the results of the staff's review of the second application for subsequent license renewal and which is the first subsequent license renewal application for a boiling water reactor. This application was submitted by Echelon Generation Company LLC for the Peach Bottom Atomic Power Station, Units 2 and 3, located near Delta, Pennsylvania.

As Subcommittee Chairman Sunseri mentioned, we had the opportunity to present the results of the review of this application to the ACRS Subcommittee on Plant License Renewal approximately a month ago on November 5th. Subsequently, we issued the updated SER on November 19th.

By way of background, Peach Bottom Units 2 and 3 received approval for their initial renewed licenses from the NRC on May 7th, 2003. The NRC review at that time was performed using guidance developed prior to the issuance of the Generic Aging Lessons Learned Report, or the GALL report. The NRC developed guidance for review of subsequent license renewal applications, and it was issued in July 2017 as NUREG-2191, also referred to as GALL SLR, and NUREG-2192, SLR SRP, following extensive interactions with the ACRS. The staff performed its review of the Peach Bottom SLR application using these NUREGs.

The NRC Project Manager for the Peach Bottom SLR application review is Ms. Bennett Brady, seated behind me. Ms. Brady will introduce the staff, who will be seated at the table, that will be presenting or addressing questions regarding the staff's review of the Peach Bottom SLR application.

Part of the management team that are here with me today: to the left is Anna Bradford, the Director of the Division of New and Renewed Licenses. To my right is Eric Oesterle, Chief of the License Renewal Projects Branch. And in the audience are other DNRL and NRR technical review Branch Chiefs and their staffs that have been involved with the review. There may also be some technical staff on the phone.

In addition, we are fortunate to have representatives from Region I also on the phone that include Kevin Mangan, Senior Reactor Inspector, as well as Justin Heinly, Senior Resident Inspector at Peach Bottom.

The staff will provide an overview of its safety review which will include a discussion of the confirmatory item related to the core plate rim hold-down bolts, which, as we discussed at the ACRS Subcommittee meeting, was closed based on the supplemental information provided by Exelon.

Staff will also provide a discussion of the regional inspection of the Aging Management Program implementation for initial license renewal and address the material condition of the Peach Bottom facility.

We look forward to a productive discussion today with the ACRS and will address any questions that you may have.

At this time, I'd like to turn the presentation over to Mr. Michael Gallagher, Exelon Nuclear Vice President for License Renewal and Decommissioning, to introduce his team and commence their presentation.

Thank you.

VICE CHAIRMAN SUNSERI: Thank you.

And, Mike, one other thing I need to mention is that Members Riccardella and myself are going to recuse ourselves from any discussions on the metal and environmental fatigue issues and radiation embrittlement issues with the reactor pressure vessel and the sacrificial shield wall. That's just due to some outside business that we've done.

Thank you.

MR. GALLAGHER: Okay. Thank you, and thank you, Meena.

Good afternoon. My name is Mike Gallagher, and I'm the Vice President of License Renewal at Exelon. I have 38 years of nuclear power plant experience, all at Exelon, and have been working on our license renewal project since 2006.

Slide 1, please.

Before we get into today's presentation, I'd like to introduce the presenters.

To my right is Anna Krause, and Anna is our Senior Manager of Design Engineering for Peach Bottom. And Anna has 14 years of nuclear power plant experience.

To Anna's right is Paul Weyhmuller, and Paul is our License Renewal Technical Manager for the Peach Bottom project. Paul has 37 years of nuclear power plant experience, including working on Exelon's license renewal project since 2011.

And to Paul's right is Julian Laverde, and Julian is our Mechanical Design Manager for Peach Bottom. And Julian has nine years of nuclear power plant experience.

And to my left is Dave Distel, and Dave is our Project Licensing Lead. And Dave has 39 years of nuclear power plant experience.

In addition, here in the room we have our technical support personnel, and, also, as mentioned, on the NRC conference line, we have our Peach Bottom technical staff available to answer questions on the conference line.

And we also have with us here today Pat Navin, and Pat is our Site Vice President at Peach Bottom.

Slide 2.

So, this slide shows our agenda for the presentation. This is a similar presentation that we gave the Subcommittee and that we abbreviated somewhat to be focused on the main activities. Included in our presentation, we did include slides that we presented to the Subcommittee meeting as backup material. And again, we can go into any questions that the full Committee may have.

We believe we developed a robust, high-quality subsequent license renewal application, and we also have developed effective aging management programs to ensure the continued safe operation of

Peach Bottom.

We appreciate the opportunity to make this presentation and look forward to answering any questions you may have.

With that, I'll turn it over to Anna Krause.

Anna?

MS. KRAUSE: Thank you, Mike.

Slide 3, please.

Good afternoon. My name is Anna Krause, and I'm a Senior Manager of Design Engineering at Peach Bottom.

Peach Bottom Units 2 and 3 are GE boiling water reactors with Mark I containments that are jointly owned by Exelon and PSE&G and operated by Exelon.

The Peach Bottom Station is located in the Commonwealth of Pennsylvania, approximately 40 miles northeast of Baltimore, Maryland, and 60 miles southwest of Philadelphia, Pennsylvania.

On the aerial view of Peach Bottom, you can see the power block; the independent spent fuel storage installation pad; the north and south substations; the plant intake and discharge canal, which is the normal heat sink for the station, and the emergency cooling tower, which comprises the emergency heat sink for the station in the event that the normal heat sink is not available.

Slide 4, please.

Peach Bottom is operated on 24-month refueling cycles. The station capacity factor for 2018 was 94.2 percent, and then, year to date through October 31st is 96.2 percent.

Our regulatory performance as Peach Bottom is in action matrix column 1 and all ROP indicators are green.

Slide 5, please.

Now this slide shows the dates for thermal power license changes for Peach Bottom Units 2 and 3. We also show that the independent spent fuel storage installation was installed in 2000. And then, the current license expiration dates are August 8th, 2033, for Unit 2, and July 2nd, 2034, for Unit 3.

MEMBER REMPE: Anna, I thought there was a measurement uncertainty recapture in 2002, but it's not shown here. Is that true? The reason I'm asking is because I kind of looked ahead and it might be good for us to clarify that.

MR. GALLAGHER: Yes, that's a license recapture.

MR. WEYHMULLER: Yes, we did a -- it was called Appendix K in its day, where they did the measurement uncertainty recapture at that point. And then, subsequent to that, you see that we did the EPU modification. With that, the Appendix K mod was taken away, and they did the EPU project, and then, subsequently, followed back up with what was now known as MUR, or the uncertainty recapture, and reinstated, basically, what had been there in the past.

MEMBER REMPE: The reason I'm asking is I was involved in the EPU approval, and I remember that earlier letter, but it may come up in our deliberations on the letter today. So, thank you.

MR. WEYHMULLER: Okay.

MS. KRAUSE: All right. Moving to Slide 6, this slide provides an overview of significant plant modifications that have been implemented at Peach Bottom that address component aging and long-term operations.

Okay. I will now turn it over to Paul Weyhmuller, who will present to you the highlights of our subsequent license renewal application.

MR. WEYHMULLER: Thank you, Anna.

Slide 7, please.

Good afternoon. My name is Paul

Weyhmuller. I'm the Technical Manager for the Peach Bottom license renewal project. I will discuss the highlights of our subsequent license renewal application, focusing on application development, our new time-limited aging analyses, the overall GALL SLR consistency, a review of the aging management programs, the exceptions we have taken, and a summary of the first license renewal aging management program affecting these reviews that have been conducted.

Slide 8, please.

Exelon used industry and NRC guidance to make our application as consistent with GALL SLR as possible. Our submittal is based on the guidance provided in both NUREG-2191 and 2192.

In developing the Peach Bottom subsequent license renewal application, changes noted from first license renewal include:

For scoping and screening, we have updated our packages for plant modifications as well as to address NEI 17-01 guidance.

For aging management reviews, the first license renewal was pre-GALL. So, additional aging effects required assessment based on NUREG-2191 GALL SLR.

For aging management programs, we have 47 programs for subsequent license renewal utilizing the GALL SLR guidance. Activities from first license renewal have been addressed in subsequent license renewal programs.

Our aging management programs were developed incorporating lessons learned from previous Exelon projects as well as from benchmarking current industry applications. The aging management programs were also developed using insights from industry RAIs.

For time-limited aging analyses, the Peach Bottom subsequent license renewal application has reassessed the existing plant current licensing basis TLAAs. Additional TLAAs for repair or replacement activities not part of the first license renewal application have been added. There are a total of 35 TLAAs found in the subsequent license renewal application.

MEMBER BLEY: Before you go on, in the core plate replacement -- I may have asked this before, but I'm asking it again -- what was the main difference between Units 2 and 3? Why did 3 need the improvement?

MR. WEYHMULLER: There was cracking noted on Unit 3 attributed from early operation. That was thought to be the cause of why there were additional defects found in that piping system that warranted replacement. It got to be --

MEMBER BLEY: And none on Unit 2?

MR. WEYHMULLER: That's correct.

MEMBER BLEY: Thank you.

MR. WEYHMULLER: Okay. Slide 9, please.

As stated earlier, Peach Bottom subsequent license renewal application is based on GALL SLR. Peach Bottom aging management review achieved significant consistency with the GALL SLR, as reflected by the fact that 98.6 of AMR line items are covered by notes A through E.

There are 50 commitments for the implementation of subsequent license renewal at Peach Bottom, consisting of 47 commitments from the implementation of individual aging management programs and 3 additional commitments for OPEX actions and for the continued use of FERC inspections for specific water-controlled structures. These commitments will be captured within the subsequent license renewal UFSAR supplement, which is contained in Appendix A of the subsequent license renewal application.

These commitments are managed in accordance with Exelon's commitment tracking program, which is based on the NRC-endorsed NEI 99-04, "Guidelines for Managing NRC Commitment Changes Process".

The table shown on the slide provides a breakdown of aging management programs in regards to consistency with GALL SLR. The summary table also provides a numerical breakdown for existing and new AMPs.

There are only 11 programs with exceptions. For each exception, we have provided an alternative to the recommendation found in GALL SLR. Supporting technical justification has been provided and has been found acceptable, as identified in the SER.

Slide 10, please.

The Peach Bottom aging management program effectiveness reviews assessed first license renewal activities and included a detailed review of inspection schedules, results, and data, as well as a review of relevant operating experience within the corrective action program. All first license renewal programs were determined to be effectively implemented. A summary of each review is found in Appendix B of the subsequent license renewal application for each specific aging management program under OPEX Item No. 1.

In November of 2018, the NRC staff conducted an IP 71003 Phase 4 inspection, post-approval site inspection for license renewal at Peach Bottom. This inspection found no issues.

I will now turn the presentation over to Julian Laverde, who will discuss how we closed the one confirmatory item and a brief summary on the specific technical topics involved in subsequent license renewal.

MR. LAVERDE: Thank you, Paul.

Slide 11, please.

Good afternoon. My name is Julian Laverde, and I am the Site Mechanical Design Engineering Manager at Peach Bottom.

There was one confirmatory item involving a commitment for the BWR vessel internals aging management program. Additional information was required by the NRC staff to complete the assessment of the proposed enhancement for core plate rim hold-down bolts. This was addressed by revising the enhancement to provide the source document, BWR 25, Revision 1, which was used to determine the appropriate actions to be taken to address stress corrosion cracking of core plate rim hold-down bolts.

This issue has been resolved with the

submittal of a supplement to the NRC staff on October 9th, 2019, and the NRC has closed this item, as stated in the updated SER dated November 19, 2019.

Slide 12, please.

In the Subcommittee meeting, we presented how Exelon addressed the four technical topics related to SLR that were of interest to the NRC Commissioners during the NRC staff preparations for SLR. These topics were discussed in Staff Requirements Memo for SECY-14-0016. The four topics are: RPV embrittlement, IASCC of reactor vessel internals, concrete and containment degradation, and electrical cable EQ and condition assessment.

To summarize, we have constructed our aging management programs in these areas to be consistent with the GALL SLR guidance. For example, for RPV embrittlement, we have developed flows projections through SPEO, satisfactorily evaluated reactor vessel material properties through SPEO, and added a commitment to withdraw and test an RPV surveillance capsule for each unit.

For IASCC, we have confirmed the acceptability of existing BWR guidelines to manage the aging of reactor vessel internals to SPEO.

For concrete and containments, we have

reported that the concrete and containment at Peach Bottom are in good condition.

And finally, the EQ cable and condition assessment, we have updated analysis to show EQ cables have a qualified life greater than 80 years. And we continue to visually inspect and test, per GALL SLR recommendations.

I will pause here to see if we have any questions on these topics.

(No response.)

I will now turn the presentation over to Mike Gallagher for closing remarks.

MR. GALLAGHER: Okay. Thank you, Julian.

Slide 13, please.

This was our summary presentation of what we gave earlier to the Subcommittee. And as I stated before, we developed a comprehensive, high-quality subsequent license renewal application, along with robust aging management programs that will ensure the continued safe operation of Peach Bottom during the subsequent period of extended operation.

Pending any questions you may have, this concludes our presentation.

VICE CHAIRMAN SUNSERI: I didn't want to distract. I missed an opportunity to ask a question a little earlier. So, I'll ask you now, just for completeness, and I'll ask the staff also when it's their turn.

But the 71003 Phase 4 inspection, that seems like a significant activity to meet the effectiveness of the aging management program. And to have no findings, how extraordinary is that? I mean, in your experience working with your peers, is that a typical finding or is that an extraordinary finding?

MR. GALLAGHER: I mean, there have been several or many Phase 4 inspections done at other sites, and there have been findings, usually a green finding. And in ours, we didn't have that, not to say we didn't get any lessons learned at all from the NRC review. I think the staff, the regional staff did thorough reviews. We had well prepared for it, for the inspection. And we would have initiated any corrective actions for further improvements in our programs, and there were items like that that were identified and acted on. But there were no findings.

VICE CHAIRMAN SUNSERI: Yes. I mean, I asked the question because we don't get to go visit the sites and do the detailed reviews. So, we rely on staff's feedback for a lot of our information. We always want to push to make sure that these reviews are being done with the rigor and technical competence that we need to ensure the regulations are going to be met and that the applicants are upholding their end of the story also. So, this seems like a good news story to me, anyway.

MR. GALLAGHER: Yes, I think so.

VICE CHAIRMAN SUNSERI: Anyone else?

(No response.)

All right.

MS. KHANNA: So, we'll definitely address that. We've got the regional folks on the phone, and they'll be happy to address a little bit more details of the inspections.

Thanks.

VICE CHAIRMAN SUNSERI: Thank you.

All right. Well, we can swap out then.

MS. BRADY: Good afternoon, Chairmen and Members of the ACRS.

My name is Bennett Brady. I am the Project Manager for the safety review of the Peach Bottom Atomic Power Station, Units 2 and 3, subsequent license renewal application.

As you know from Meena, we are here today to discuss the NRC staff's safety review of the Peach Bottom SLRA, as documented in the Safety Evaluation

Report, or SER, as it's known, which was issued on November 19, 2019.

Joining me here at the table is Bill Rogers, Senior Project Manager in the Division of New and Renewed Licenses, or DNRL, who also assisted me in managing the project. In addition, joining us by telephone is Kevin Mangan, Region I, Senior Reactor Inspector, and Jon Greives, Region I, DRP Branch Chief, responsible for Peach Bottom.

I would suggest that we ask them, when we get to the end of our presentation, to address your question about how unusual this finding is.

Angela Wu, also a Project Manager in DNRL, will be controlling the slides.

Seated in the audience and joining us by phone are members of the NRR technical staff who participated in the review of SLRA and conducted the audits.

Next slide, please.

We will begin the presentation with a general overview of the staff's safety review, followed by an overview of SER Section 2 on scoping and screening; SER Section 3, aging management review, and Section 4, time-limited aging analysis. We will, then, discuss the closure of the confirmatory item, the Region I initial license renewal inspection that coincided with the staff's SLRA review, and the Resident Inspector's perspective on plant material conditions, and then, finally, the summary conclusion.

Next slide, please.

Peach Bottom Units 2 and 3 were initially licensed in October 1973 and July 1984, respectively. The licensee, Exelon Generation Company LLC, or Exelon, submitted the application for a subsequent license renewal in July 10, 2018.

Next slide, please.

As you've heard, the Peach Bottom SLRA is the second safety review performed by the staff using the GALL SLR and SRP SLR guidance issued in 2017. The staff's Peach Bottom SLR review was the same as that used for Turkey Point SLRA review. The staff identified and implemented several efficiencies as compared to the safety review of initial license renewal applications.

One of these efficiencies dealt with the conduct of audits. Instead of one large and lengthy onsite audit, the staff conducted two standard audits, an operating experience audit, and an in-office audit. The majority of audit activities and breakout discussions were conducted in-office through the use of portals and telecommunications.

During the Peach Bottom operating experience audit, the staff performed an independent review of plant-specific operating experience to identify pertinent examples of age-related degradation, as documented in the applicant's program corrective action program database.

During the in-office audit, the audit team first focused on two areas: first, the scoping and screening review and, second, the review of aging management programs, or AMPs; aging management review items, and the time-limited aging analysis.

For the Peach Bottom SLRA, the staff review was informed by the results of the Region I initial license renewal inspection, the IP003 Phase 4. This inspection was performed in November of 2018, as has been mentioned, and coincided with the SLRA review timeline. However, it should be noted that the Phase 4 inspection is related to the initial renewed license and is independent of the SLRA review. We will discuss this inspection more in detail later in our presentation.

Next slide, please.

The Peach Bottom SER with a confirmatory item was issued on October 7, 2019. The confirmatory item was related to the core plate rim hold-down bolts. During the ACRS Subcommittee meeting on November 5, 2019, the staff presented how this confirmatory item was closed on the basis of supplemental information provided by Exelon. Since that meeting, the staff has updated the SER to close the confirmatory item. The updated SER was issued on November 19, 2019, and details of the closure of this confirmatory item will be discussed later in this presentation.

During the staff's technical review of the SLRA, it issued 48 RAIs, four of which were followup RAIs. Although this was an early SLRA review, and new topics were reviewed for the 60-to-80-year time period, one might well have expected to have more RAIs than initial license renewal. However, this was a significant decrease in the number of RAIs from the recent initial license renewal application reviews. The staff believes that this was due to the high quality of the subsequent license renewal application.

Next slide, please.

In the next few slides, we will present the results of the staff's safety review, as described in the SER. SER, Section 2, describes the scoping and screening of structures and components subject to an aging management review. The staff reviewed the applicant's scoping and screening methodology, procedures, and the results. The staff review included, as required by the license renewal rule, the results of the integrated plant assessment, the safety-related SSCs, non-safety-related SSCs affecting safety functions, and SSCs relied upon to perform functions in compliance with the Commission's regulations for fire protection, environmental qualification, station blackout, and anticipated scrams without a scram.

Based on the staff's review, the results from the in-office audit, and review of additional information provided by the applicant, the staff concluded that the applicant's scoping and screening methodology and implementation were consistent with the SRP SLR and the requirements of 10 CFR Part 54.

Next slide, please.

SER, Section 3, and its subsections, cover the staff's review of the aging management programs for managing the effects of aging, in accordance with 10 CFR 54.21(a)(3). Sections 3.1 through 3.6 include the AMR items in each of the general system areas within the scope of license renewal, which is shown on this slide. For a given AMR item, the staff reviewed the item to determine whether it is consistent with the GALL SLR report. For AMR items not consistent with the GALL SLR report, the staff reviewed the applicant's evaluation to determine whether the applicant has demonstrated there is reasonable assurance that the effects of aging will be adequately managed, so that the intended functions will be maintained, consistent with the current licensing basis for the subsequent period of extended operation.

Based on this review, the results from the in-office audit, and additional information provided by the applicant, the staff concluded that the applicant's aging management review activities and the results were consistent with the SRP SLR and the requirements of 10 CFR Part 54.

Next slide, please.

The SLRA described a total of 47 AMPs, 11 new AMPs, and 35 existing. This slide identifies the applicant's original SLRA distribution of these AMPs in the left column and the final disposition, as documented in the SER, in the right column. All of the AMPs, with the exception of the plant-specific AMP, were evaluated by the staff for consistency with the GALL SLR report. As a result of the staff review, the applicant made several changes in the AMPs.

However, the distribution of AMPs did not change, as you will see comparing the left column and the right column. The plant-specific AMP was evaluated against the criteria contained in Appendix A1 of the SRP SLR.

Based on the staff's review, the results from the in-office audit, and review of additional information provided by the applicant, the staff concluded that the applicant's aging management program activities and results were consistent with the SRP SLR and the requirements of 10 CFR Part 54.

Next slide, please.

SER, Section 4, identifies time-limited aging analysis, or TLAAs. Section 4.1 of the report documents the staff evaluation of the applicant's identification of applicable TLAAs. The staff evaluated the applicant's basis for identifying those plant-specific or generic analyses that need to be identified as TLAAs and determined that the applicant has provided an accurate list of TLAAs, as required by 10 CFR 54.21(c)(1).

Section 4.2 and 4.7 document the staff's review of the applicable Peach Bottom TLAAs for the areas shown on this slide. Based on its review, the information provided by the applicant, the staff concludes that either one of three conditions are met:

(I) the analysis remains valid for the subsequent period of extended operation; (ii) the analysis has been projected to the end of the subsequent period of extended duration, or (iii) the effects of aging on the intended functions will be adequately managed for the subsequent period of extended operation, as required by 10 CFR 51.21(C)(1).

Based on the staff review, the results from the in-office audit, and the review of additional information provided by the applicant, the staff concluded that the applicant's TLAAs analysis and results were consistent with the SRP SLR and the requirements of 10 CFR Part 54.

Next, Bill Rogers will assess the closure of the confirmatory item and the Region I activities.

Thank you.

MR. ROGERS: Thank you, Bennett.

Good afternoon.

The SER with confirmatory item issued October 7th, 2019, included one confirmatory item associated with the BWR vessel internals AMP B.2.1.7. Specifically, the applicant had proposed an enhancement to perform one of two future activities post-licensing to address the potential for stress corrosion cracking of the core plate rim hold-down bolts and its mitigation.

The first option was to install core plate wedges, which the staff found acceptable. The second option was to submit an inspection plan to the NRC for future review and approval. Since the completed inspection plan as well as the acceptance criteria was not currently available during the staff's SLRA review, that is, it would be developed at a future date, this option did not satisfy the staff's need to complete its technical review prior to granting a new license.

In response to the staff's concern regarding the inspection plan, the applicant submitted a supplement to the SLRA which revised the enhancement to AMP B.2.1.7, to be in accordance with BWRVIP 25, Revision 1, to: one, install wedges or, two, install core plate rim hold-down -- excuse me -- inspect core plate rim hold-down bolts, or, three, demonstrate via analysis that the installation of wedges and inspection of the core plate rim hold-down bolts were not required. The staff determined each of the three options included in the SLRA supplement can be confirmed by inspection through the reactor oversight process and were, therefore, acceptable.

On the basis of this information, the

staff determined that its concerns related to the confirmatory item are resolved, as documented in the November 19, 2019, updated SER.

Next slide, please.

In conclusion, for the SLRA safety review, the staff finds that the requirements of 10 CFR 54.20(a) have been met for the subsequent license renewal of Peach Bottom Units 2 and 3.

Next, I'll discuss regional inspections and observations on the plant condition.

The Region conducts a license renewal team inspection, IP 71003 Phase 4, 5 to 10 years following the entry into the initial period of extended operation. The team examines a sample of AMPS to verify the effects of aging were being managed effectively to ensure structures, systems, and components in the scope of these programs maintain the ability to perform their intended functions.

I'll address the Peach Bottom IP 71003 Phase 4 initial license renewal inspection on the next slide. The Peach Bottom IP 71003 Phase 4 initial license renewal inspection was performed in November of 2018 on both Units 2 and 3. Exelon had committed to 35 aging management programs at Peach Bottom for the initial period of extended operation. Seventeen

AMPs were previously existing program in which no changes were required. Twelve programs were previously existing, but were enhanced. And there were six new AMPs created.

For the Phase 4 inspection, a sample of six of these AMPs were reviewed. The AMPs listed here on the slide were selected based on inspection procedure criteria such as new enhanced AMPs, AMPs impacted by internal or external operating experience, Resident Inspector input, AMPs not inspected by other baseline inspections, and risk insights.

In addition, the staff considered the applicant's periodic AMP effectiveness review, which is performed every five years. The applicant's reports on this activity were used by the staff in the AMP selection process and to provide insights on program performance.

The Region's inspection focuses on the program's detection of aging effects, monitoring and trending, corrective actions, and implementation of operating experience elements. The inspection team did not identify any findings and concluded that Exelon that was effectively implementing the AMPs review.

Next slide, please.

And before we go on, were there any questions on that specific topic related to the earlier question?

VICE CHAIRMAN SUNSERI: It just strikes me as, I guess, impressive that an inspection scope so big and of so many technical areas, and you have no findings. I mean, you could look at it, I want to look at as a glass half full; it was a very thorough inspection and they did a good job. Another way of looking at it, though, is you didn't look at it very good and missed something, right? So, that's what I'm trying to figure out.

MR. ROGERS: Okay. I'd like us to give the Region an opportunity to address that comment or question.

MR. GRAY: Thanks for that.

This is Mel Gray. I'm a Branch Chief in NRC, Region I, responsible for oversight of inspections in license renewal. And I have with me Kevin Mangan, and he was a team leader. But I'm going to turn it over to Kevin.

My opinion definitely is it was an invasive inspection that demonstrated licensee performance.

But go ahead, Kevin.

MR. MANGAN: Yes, so for that inspection, as you said, this is Kevin Mangan and I was the team lead.

That inspection is a one-week inspection with three inspectors. And as you said, we didn't identify any violations. Of note, it was the first Phase 4 inspection ever done in the United States. We have done a couple since then, one in Region I and I think one in Region II. There may be one or two others.

There were some violations identified in other inspections of this inspection, but here and, then, we also did Ginna, and that also identified no finding.

MEMBER KIRCHNER: Could I ask a follow-on question then, Bill?

I know at the Subcommittee meeting we heard good things about the applicant's preventive maintenance program, particularly with regard to cables. We heard about the diesel generator cables. So, fairly proactive.

If my notes are correct, the applicant, they changed out about 100 -- there are about 100 medium-voltage circuits and they replaced about half. So, I'm curious why you inspected medium-voltage cables rather than, say, I&C protection system cables.

MR. MANGAN: This is Kevin Mangan again.

So, for that particular AMP, a lot of the cables you've mentioned are, for the scope of the AMP, I think there was only 8 or 10 cables in scope. The cables that were replaced were not in the scope of the AMP. They were in the scope of license renewal, but were excluded because they were energized less than 25 percent of the time, which was the criteria when they first received their license renewal.

So, for the cables we looked at, which is limited scope, they are risk-significant and there were changes to the GALL from -- Peach Bottom was a pre-GALL plant. Through Rev. 1 and Rev. 2, they went from 10-year inspections to seven-year inspections, and that particular requirement that excluded cables that were energized less than 25 percent of the time was removed. So, those are some of the reasons why we looked at that, to see what kind of changes Exelon was making to the program to address the operating experiences of the GALL reports.

MEMBER KIRCHNER: Well, if I remember correctly from the Subcommittee meeting, and the applicant and your inspections, going back to the diesel generator cables, those are active less than 25 percent of the time. And yet, there was a problem there which the applicant addressed and corrected.

I'm not sure about that 25 percent of the time. It's still sticking in my mind as not a good criterion to use on cable inspection. So, this is a more generic question than just the applicant.

MR. ROGERS: So, I think it might be helpful to have one of the electrical reviewers address the change to the GALL and how that's been modified.

MR. SADOLLAH: Yes. Hi. This is Mo Sadollah at NRR, a Design Engineer.

So, that provision that was in the previous GALL revision, Rev. 0, subsequently, in Rev. 1 and Rev. 2, and then, ultimately, in the SMR, that was removed. So, that 25 percent threshold was no longer there. Whether the cables are energized or not, they're considered in the scope.

MEMBER KIRCHNER: That's what I was looking for. So, that's been removed?

MR. SADOLLAH: Yes.

MEMBER KIRCHNER: Okay. Thank you.

MR. SADOLLAH: Yes.

MR. ROGERS: Any additional questions on that topic?

(No response.)

Go to slide 14.

Okay. At the ACRS Subcommittee meeting on November 5th, 2019, the Senior Resident Inspector discussed the plant's performance and material condition. The Senior Resident stated that the NRC assessment of Peach Bottom was that the material condition of the plant was acceptable and meets regulatory requirements for systems, structures, and components, based on the inspection results and green performance indicators which resulted in both Peach Bottom units being in the licensee response column.

In addition, Resident Inspectors continue to inspect and assess the licensee's ability to manage the effects of aging through the baseline inspection program.

And again, if there are any additional questions related to plant material conditions or how this assessment was made, I would offer the question to the Region in that area.

VICE CHAIRMAN SUNSERI: I recall the discussion was very good at the Subcommittee. So, we got a really thorough briefing then.

MR. ROGERS: Good. Thank you.

And considering the NRC inspection

results, the inspectors found that the aging management programs were being effectively implemented in accordance with the facility's renewed license. And the NRC will continue to monitor AMP effectiveness using the baseline reactor oversight process.

And if there are no additional questions at this point, I'll turn the presentation over to Bennett for a summary conclusion.

MS. BRADY: The NRC has now completed its presentation of its conclusions from the staff's safety review of the Peach Bottom SLRA and the Region I conclusions on AMP inspections and plant license conditions.

At this point, we would be pleased to address any further questions that you may have.

VICE CHAIRMAN SUNSERI: Any additional questions or comments?

MEMBER BLEY: Yes, I have one. This is not related to this particular application, but from the NRC staff side, and the licensee using the new GALL, and your reviews, did you find places where you think you're going to need to make changes to the subsequent licensee renewal GALL? And could you tell us about any of those?

MS. BRADY: Yes. Right now, we are just

beginning the process. We've collected a lot of ideas/opinions on changes that should be made to the GALL SLR and SRP SLR. We will be reviewing the proposed changes. At some point in the future, there will be an Interim Staff Guidance with these changes. And they'll likely incorporate -- that would be one that would be considered to be modified.

MEMBER BLEY: Okay. Thanks. Any idea when that timeframe will come to pass?

MR. ROGERS: That person is sitting behind you.

(Laughter.)

MEMBER BLEY: Maybe they would like to comment.

MR. OESTERLE: Thank you, Bill.

This is Eric Oesterle from the NRC staff.

So, thanks for the question, Dennis.

Back in March of this year, we did have our first SLR lessons learned meeting from reviews of the first three applications to date, and we did identify a number of technical issues which we thought were ripe for considerations and inclusion perhaps in an update to the SLR guidance documents, one of which happened to be an issue regarding irradiated structural steel.

So, we have compiled a list of those technical issues, and, in fact, we're having our second SLR lessons learned public meeting on December the 12th. So, we're continuing to engage with the applicants and with industry to address these technical issues that have come up.

MEMBER BLEY: Thanks a lot. We look forward to seeing that whenever it comes to pass.

VICE CHAIRMAN SUNSERI: Yes. This is kind of a crystal-ball question, but would you anticipate that those improvements would help reduce the number of RAIs coming through the process?

MR. OESTERLE: Yes.

VICE CHAIRMAN SUNSERI: Because people will know in advance what they should be providing?

MR. OESTERLE: Eric Oesterle from the staff.

And, yes, that's one of the goals or one of the criteria for identifying some of these technical issues, if not as a new issues, but areas where clarification can be provided. One of the goals is to reduce the number of RAIs.

And to address a question that you had, Member Dennis, we're looking, currently looking at whether or not we're going to do an update of the entire document or whether or not we're going to group issues and put them out in three or four separate ISGs. But, tentatively, we're looking at the latter part of next year to start coming out with the drafts.

VICE CHAIRMAN SUNSERI: Okay. Any other questions?

(No response.)

So, while we're opening up the phone line for public comments, I'll turn it to the room and ask if there's any members of the public in the room that would like to make a statement or a comment. Now come to the microphone and state your name and your comment.

I can't see anyone.

MEMBER BLEY: No, nobody.

VICE CHAIRMAN SUNSERI: Okay. Thank you, Dennis.

And now, we'll go to the open public phone line for any comments. State your name and provide your comment, please.

(No response.)

All right. No comments. So, we'll close the phone line again.

And I just would like to extend our appreciation to the applicant and the staff for the thoroughness of your review and a very good Subcommittee where we reviewed these in great detail. And now, it makes the full Committee meeting almost seem anticlimactic, which I guess is a good thing, right? So, we did all the hard work and this is the fruit of the labor here.

So, thank you all for your comments, and I'll turn it back to the Chairman now.

CHAIRMAN RICCARDELLA: Thanks.

We're supposed to take a break at 2:30. We have until 2:45 until the next meeting --

VICE CHAIRMAN SUNSERI: Yes. So, we have a letter that we could read in, you know, do the read-in on. I mean, we could fit it in the 30 minutes.

CHAIRMAN RICCARDELLA: Okay.

VICE CHAIRMAN SUNSERI: So, are you going to pull that up? Got it. All right.

Thank you. You are excused. Thank you.

We'll need you again at 2:45.

(Whereupon, the foregoing matter went off the record at 1:59 p.m. and went back on the record at 2:45 p.m.)

CHAIRMAN RICCARDELLA: So, we'll reconvene the meeting.

And the subject is NuScale source term,

and the lead on this is Dave Petti.

MEMBER PETTI: So, we had the Subcommittee -- what? -- two weeks ago, the week in front of Thanksgiving, and discussed a lot of these issues in detail. There was only one area that came up sort of as a questionable one that I believe NuScale will talk about it in a high-level summary, and then, NRC will give a more complete, but a high-level overview, again, because most of us were in the Subcommittee meeting.

So, let's start with NuScale.

MR. MILTON: Sure. This is Mike Milton. I'm basically going to turn the slides and be here for moral support. Zack Rad, Director of Regulatory Affairs, is going to kick us off from Corvallis. And then, our team in Corvallis will lead the discussion. Okay?

Okay. So, I'll turn it over now to you, Corvallis. Is that correct? Please go.

CHAIRMAN RICCARDELLA: Corvallis, are you there?

MR. MILTON: I heard sound, too. It was very low.

Carrie, can you hear us in the room okay? Because we didn't hear anything coming from the phone line at the moment.

MS. FOSAAEN: Yes. We just need one minute here in Corvallis, if that's all right.

CHAIRMAN RICCARDELLA: Okay.

MR. MILTON: Okay.

MR. RAD: Okay. Good afternoon. This is Zachary Rad, Director of Reg Affairs for NuScale Power. I just have a few opening remarks.

Like we discussed at the Subcommittee meeting, we only intend to provide supplemental information on a single topic during this meeting, and not repeat our comprehensive presentation. So, as we discussed in the Subcommittee meeting, one of the topics that came up late in the review of the Accident Source Term Topical Report was associated with postulated leakage from the hydrogen monitoring system coincident with a beyond design basis severe accident. We're going to provide information regarding elements on the topic that hadn't been fully addressed during the Subcommittee meeting to ensure that the record accurately reflects our position.

So, as I noted in the Subcommittee meeting, the reason this topic is here for discussion in this forum is because it's a specific item we were unable to reach alignment on with the staff during the review, and the decision has been made to move forward by --

(Interference on the public line.)

MEMBER REMPE: Excuse me for just a minute, Zack.

If you are on the public line, unfortunately, we can hear everything you're saying. So, could we please ask you to mute your phones and we can hear the applicant. Thank you.

MEMBER BLEY: Go ahead, Zack.

MR. RAD: All right. So, I'm just going to take a few minutes to address our position in summary. Jim Osborn, who's here with me as well, will provide some supporting details.

So, as I just noted, late in the review the staff raised some questions regarding the inclusion of some postulated leakage from the hydrogen monitoring system, in addition to a severe or concurrent with a severe beyond design basis accident. And that's specifically estimation of the contribution from operational leakage.

It's our position that NuScale has addressed this topic consistent with the applicable regulations and guidance, and specifically, NUREG-0737, and within that, the provisions addressing control room habitability requirements and integrity of systems outside containment likely to contain radioactive material.

It's important that the existent systems, such as the hydrogen monitoring system, and their potential to contain active source term, was considered at the time the guidance was developed and addressed within the guidance.

So, the guidance as well as the operating fleet, and all previous applicants, addressed the topic by including these systems in a program to reduce leakage as low as practical. And this is an operating program. So, I think that that's also important to note. It includes testing during refueling outages and a variety of other provisions.

NUREG-0737 also addresses systems with known leakage, such as ESF systems, by specifically addressing those, where applicable, and those are addressed within the provisions, specifically control room habitability requirements. It's probably also worth noting that NuScale doesn't have any such systems.

So, NuScale addressed the topic in the same manner at the same level of detail, or even a greater level of detail, than previous applicants.

It's our position that there's no material difference in the NuScale design that makes existing guidance insufficient or diminishes the applicability of precedence. It's also important to note that, for NuScale, this is not a safety concern; it's not a design or a licensing basis issue for NuScale. It's just a matter of reasonable assurance; that is, that the guidance and precedent for design -- following guidance and precedence for a design with lower associated risk is sufficient for reasonable assurance.

So, with that, that's my summary. If there aren't any questions, I'm going to turn it over to Jim to address some supporting elements in more detail.

MR. MILTON: Yes, we can proceed.

MR. RAD: All right. Thanks.

MR. OSBORN: Good afternoon. This is Jim Osborn.

So, I want to preface the presentation and say that the purpose of the presentation is to convey the fact that NuScale has designed out a core melt scenario, and therefore, there is no design deficiency related to the hydrogen monitoring system. This was discussed in the earlier meeting a couple of weeks ago.

The first slide here talks about risk significance. So, the core damage frequency of the NuScale power module is very small. The sum of the internal events has a core damage frequency of on the order of 3E to the minus 10 module critical years per year. This is a significant margin of about five orders of magnitude to the NRC's safety goal.

So, accidents in which hydrogen monitoring could be used, i.e., those that have an intake containment that results in core damage, are even a lower frequency, on the order of E to the minus 11. But, even with a significant increase in consequences, the overall risk still remains small, considering the frequency of these events is so small. You see the equation up there for risk.

And I will quote from the last bullet on the slide. It says, "In any licensing review or other regulatory decision, the staff should apply risk-informed principles when strict, prescriptive application of deterministic criteria is unnecessary to provide reasonable assurance of adequate protection of public health and safety." This quote is from the SRM for SECY-19-0036, which was entitled, "Application of the Single Failure Criteria to the NuScale's

Inadvertent Actuation Block Valves".

But this directive should also be applied to other deterministic criteria like hydrogen monitoring system leakage. The core melt sequences in which hydrogen monitoring could even be utilized, i.e., there's an intact containment, can be considered negligible and, therefore, not risk-significant. Therefore, to provide reasonable assurance of adequate protection of the public health and safety, this incredible sequence would not need to be considered in a review using proper application of risk-informed principles.

Next slide, please.

The systems used for hydrogen monitoring are included in the leakage monitoring program. This program is one of the post-TMI action items that is intended to minimize the potential leakage from systems outside containment that may contain actual source term. NuScale is in compliance with this regulation. The implementation of this program ensures that these systems are essentially leak-tight and are available for use post-accident.

The seismic aspects of the next bullet will be addressed in a later slide. So, next slide, please.

But what if it's hypothesized deterministically that the hydrogen monitoring system does leak? There would be subsequent emergency response actions to isolate that leak. The particulars of this action would be the responsibility of the emergency response organization as an unplanned and unanticipated emergency action, for which there are no explicit dose acceptance criteria.

Recently, just two weeks ago, the NRC stated in the Brunswick SER for hardened vents that, "For plant personnel performing emergency response actions during a beyond design basis severe accident, there are no explicit dose acceptance criteria." The only purpose for the NuScale hydrogen monitoring system is for a beyond design basis severe accident. Therefore, the 5-rem limit of 10 CFR 50.34(f)(2)(vii) does not apply to the operator action of re-isolating the containment isolation valves used in hydrogen monitoring.

Next slide.

Based on the nuclear industry's low risk from severe accidents, which are even lower for the NuScale design, the NRC relaxed the regulatory requirements for hydrogen monitoring. As a severe accident monitoring system, it is not required to be safety-related or Seismic Cat 1 because there is no design basis accident that involves the hydrogen monitoring system.

So, for the NRC staff to compare NuScale's non-safety-related, non-seismic, Cat 1 design of a hydrogen monitoring system to other designs that are safety-related or Seismic Cat 1, is not commensurate with a risk-informed review. It is not appropriate for the NRC to relax requirements based on the risk significance and, then, penalize a design by deterministically presuming it will leak because it is non-safety or not Seismic Cat 1.

This application of risk significance is evident in the guidance provided in Reg Guide 1.183 related to offsite dose consequences for hydrogen purge operations for severe beyond design basis accidents. For the NRC to require NuScale to deterministically account for hydrogen monitoring system leakage runs counter to the application of its risk significance and does not reflect a risk-informed review.

Are there any questions?

MEMBER MARCH-LEUBA: Yes, this is Jose. Can you clarify something for me? The hydrogen monitoring system is non-safety grade and it is connected to the containment evacuation system, is that correct?

MR. OSBORN: So, yes, the hydrogen monitoring system is made up of three different systems: the containment evacuation, the sample system, and the core flood and drain. They are connected to the containment through containment isolation --

PARTICIPANT: I'm on this call and I can't hear anything from the actual meeting.

MR. OSBORN: The containment isolation valves are safety-related.

MEMBER KIRCHNER: Can you wait a moment? We're having a problem.

PARTICIPANT: I can hear you talking now, but I can't hear the ACRS meeting apparently.

CHAIRMAN RICCARDELLA: No, this is the ACRS meeting room. I think what you're not hearing is the NuScale remote call-in. So, we're trying to address that right now.

PARTICIPANT: Oh, okay.

CHAIRMAN RICCARDELLA: Steve, can you hear me? Steve Schultz?

DR. SCHULTZ: Yes. Yes, Pete.

CHAIRMAN RICCARDELLA: But you couldn't

hear NuScale talking from Corvallis?

DR. SCHULTZ: That's correct. Whenever you go to the phone line, we can't hear. The same thing happened in the Peach Bottom meeting.

CHAIRMAN RICCARDELLA: Okay. We're trying to work on it.

Mike, are you there?

DR. CORRADINI: I am here.

CHAIRMAN RICCARDELLA: And you can hear Corvallis, too?

DR. CORRADINI: At this moment I can only hear you.

CHAIRMAN RICCARDELLA: Yes, because they're not talking right now.

(Laughter.)

But, when they were talking, you could hear?

DR. CORRADINI: Yes, I could, sir.

CHAIRMAN RICCARDELLA: Are you on the closed line or the public line?

DR. CORRADINI: The closed line.

CHAIRMAN RICCARDELLA: Okay. I think the other people who are having problems are on the public line, not the closed line.

MEMBER MARCH-LEUBA: Okay. So, let's try

it again. Corvallis, this is Jose March-Leuba.

So, I was trying to confirm that the hydrogen monitoring system is connected to the CES, and you were describing the three systems that are interconnected.

MR. OSBORN: Yes, that's correct. Yes, that's correct.

So, to understand, the hydrogen monitoring system is portions of three systems. So, it's not in itself its own system. It's just a pathway utilizing three different systems.

MR. MILTON: Okay. Hang on a second, Jim.

So, it's a pathway utilizing three different systems, and the hydrogen monitoring system is actually a portion of three systems.

MEMBER MARCH-LEUBA: But all of those three systems are downstream of the containment isolation valves, which is the last safety-grade system that protects containment on a safety-grade basis, is that correct?

MR. OSBORN: I believe that's correct, yes.

MEMBER MARCH-LEUBA: All right.

MR. MILTON: We believe that's correct.

MEMBER MARCH-LEUBA: Yes. I realize that

the probabilities or the frequencies might be 10 to the minus 11, but giving the operator the temptation to open the isolation valves for the containment to measure the hydrogen because he's suspects that it's hydrogen, it's counterproductive. I mean, you should never under any circumstance open the isolation valves if you suspect that the containment is contaminated.

So, in my opinion, we have two options. We can just not have a hydrogen system or connect the hydrogen system that works. Because connecting the system to the CES and the third system, which I don't know what it is, which none of them are seismically-qualified, you are asking for trouble.

MR. OSBORN: So, I understand that they're not seismically-qualified, they're not Seismic Cat 1, they're not safety-related. That's because the NRC relaxed the regulatory requirements on this system based on its risk significance. So, NuScale did not do this on their own. They did this in response to the NRC regulations.

MEMBER MARCH-LEUBA: Okay. So, we will talk to the staff here in person.

Sorry, can you relay for the public?

MR. MILTON: Oh, sure. The answer is it's we understand that our system was designed because the

NRC relaxed the requirements based on the risk significance. We understand your point, but we feel our design was justified, per the regulations.

MEMBER MARCH-LEUBA: Yes. In my opinion -- and I will take care of this with the staff -- the hydrogen system the way it's designed is producing more problems than it solves. Because if you ever need it, you are going to de-isolate the containment.

MEMBER PETTI: So, let's ask the question and the staff may know. Current PWRs, is the hydrogen monitoring system safety-grade or non-safety-grade? We can wait for the answer until staff speaks.

MS. FOSAAEN: Okay. I was going to say Reg Guide 1.7 provides the requirements for hydrogen monitoring systems, and our system followed Reg Guide 1.7, and it does specify that it does not need to be safety-related.

MEMBER PETTI: Okay. Thank you for that information.

MR. MILTON: So, to repeat, our design, per Reg Guide 1.7, does not require the system to be safety-related, and we followed the design per the Reg Guide, to repeat that.

MEMBER MARCH-LEUBA: I'll follow up with

the staff when we can actually communicate.

MR. MILTON: I understand. Thanks.

MEMBER PETTI: Any more questions?

MEMBER REMPE: Okay. So, I have a question that stems from what was discussed at the Subcommittee meeting, that comes from the source term evaluation. And it was discussed in the open session. And I'd like to bring it up again to NuScale because I think we dismissed something I was trying to raise last meeting prematurely. Okay? So, I want to give them the opportunity to respond.

When you did your source term, you looked at small break LOCAs; you looked at rod ejection accidents. And as the release is coming from the vessel, you know, the depressurization occurs, I mentioned some concerns about some aerosols that might be going out into the containment that would interfere with that wonderful radar-based sensor for water level detection.

And NuScale came back and said, hey, we won't have degradation; we're only worried about design basis events here, and the iodine spike came from that.

But there is something called fuel fragmentation and dispersal that we've been talking about. And that occurs before you have core degradation. And so, I'd like to bring that up again, that there is the potential that there could be some aerosols released into the containment, and the ECCS is triggered when the water level gets to a certain height, and that could interfere with the triggering of the ECCS.

And so, I'd like to hear NuScale's response back again on that question.

MEMBER BROWN: Would the aerosols be any different than the normal foaming you get from the boiling in the upper area? Because that's the pressurizer. So, you've got a steam-water interface there that gives you the same issues relative to whatever detector you're worried about, which I'm aware of, as any injected or introduced aerosols would be due to something else. I mean, they've got to make the system work at this steam-water interface where all these bubbles -- and you've got to compensate for that. I mean, everybody that builds these things has to compensate for it, like 30 percent. It's not a half-a-percent error thing.

MEMBER REMPE: The staff has defined an ITAAC that talks about pressure conditions, radiation conditions, et cetera. There's nothing in there about dispersed aerosols. And so, yes, it would be different than just a foamy thing. You could have depressurization occurring and they could be elevated. There's uncertainty on what those aerosols would be like, but it's something that the staff has raised for PWRs.

MEMBER BROWN: But if they're in the upper part, as opposed to part of the surface steam interface --

MEMBER REMPE: They don't have to be in there.

MEMBER BROWN: -- that would be a different issue --

MEMBER REMPE: Yes.

MEMBER BROWN: -- relative to the disturbing of the thing.

MEMBER REMPE: Absolutely.

MEMBER PETTI: Just to be clear, the fuel aerosols, this is pieces of fuel, right?

MEMBER REMPE: Right.

MEMBER PETTI: These would be fairly large.

MEMBER REMPE: No, not necessarily. If you looked at some of the pictures of fuel fragmentation and dispersant from the tests --

MEMBER PETTI: When you call them "aerosol," it sounds like they're pieces of metal.

MEMBER REMPE: It's not pieces of metal. They're fine fragments.

MEMBER PETTI: Fines of -- fine micron?

MEMBER REMPE: I'd have to go back and look at some of the reports, but they looked pretty small. And they could be elevated just like the sediment, or whatever they talked about that they artificially --

MEMBER PETTI: They're really particulate dust?

MEMBER REMPE: It could be, yes, particulates that are elevated.

MEMBER PETTI: Not aerosol necessarily?

MEMBER REMPE: Yes. And so, again, it could be particulates.

So, anyway, I'm waiting for NuScale to respond back to the question again.

MR. MILTON: So, this is Mike. I have to repeat back the responses. So, just kind of break up a little bit and give me a moment to be able to relay the information because of the phone line issue going on.

Back to you guys, Jim, Carrie.

MEMBER REMPE: Has the connection been lost?

MEMBER BROWN: NuScale?

MR. OSBORN: Could you give us a second, please?

MEMBER BROWN: Oh, okay.

MR. MILTON: Yes, let me know. Let me know.

MR. OSBORN: All right. Just a moment.

(Pause.)

MEMBER REMPE: You know, they don't have to answer like right now because I'd like the staff to also weigh-in on it, and they could perhaps answer later, instead of just waiting here.

MR. MILTON: That's fine.

MEMBER REMPE: Is that okay with you?

CHAIRMAN RICCARDELLA: Hey, guys, can we have one meeting, please?

MEMBER KIRCHNER: So, Joy, for clarification, are you asking whether the particulate, whatever comes out of the core, is going to actually deposit upon the sensor and interfere with its performance, or it's dispersed in the atmosphere and it's going to impact the performance of the radar?

MEMBER REMPE: It's the latter. It's the

fact that radar -- and we can say it's a radar-based sensor. That's in the open now, but --

MEMBER KIRCHNER: Then, the issue isn't fuel aerosol particulate; it's how it performs in the fog and steam.

MEMBER REMPE: Right. Well, but fog with particulates.

MEMBER KIRCHNER: Okay, but you're looking for a hard interface and a water level, and it's not likely that -- I'm not going to answer the question for NuScale. But, based on my experience with radar systems, fog and steam is not going to impact its ability to find a hard object or an interface.

MEMBER REMPE: But this is not just fog and steam. It could be particulates. You've seen pictures of what happens --

MEMBER KIRCHNER: Yes, but it's still my understanding that --

MEMBER REMPE: -- with the fuel that way. It's oxidized cladding.

MEMBER KIRCHNER: Yes, but you're not going to have that much fuel dispersed.

MEMBER REMPE: We don't know that.

MEMBER BROWN: If you'll go look at some of the designs of radar-type detectors for this, they talk about a frothy steam-water interface --

MEMBER KIRCHNER: Right. Yes, that's more of an issue, but that's independent of having any particulate.

MEMBER BROWN: Put aside the particulate, okay?

MEMBER KIRCHNER: Yes.

MEMBER BROWN: That you have to have compensation for.

MEMBER KIRCHNER: Right.

MEMBER BROWN: And I think they're advertising a fairly decent accuracy for it, like 1 percent or a half a percent or 2 percent. I don't remember the number. I read it at one time. So, Joy's concern about that, basically the steam-water interface, and then, the particulate thing comes in as a secondary relative to the --

MEMBER REMPE: But the staff has taken great pains to have ITAACs that identify the characteristics that have to be validated.

MEMBER BROWN: No, I understand that.

MR. OSBORN: So this is NuScale if you guys are ready.

Right. So we've taken a look. And we don't have this level of detail yet because it hasn't been defined yet in the EQ program, but the EQ program does require that we identify the specific environment in which the instruments are required to operate in. And there are a lot of variables here including whether or not it's a plausible scenario to have significant core melt at the time that the instrument would be required to operate and then evaluate whether or not the equipment would operate in that environment, if required.

So the program has to define those attributes and then determine whether or not the equipment is qualified to operate then. And that's where we are. So we don't have the answer to your specific question.

MEMBER REMPE: Let me be real clear. This is before you get core melt. This is something that -- that's how you deterred me a couple of weeks ago and I thought about it some more and it's like no, it's operations. Some of the cladding becomes oxidized and that's something that's been discussed in the LWRs and now we are trying to deal with what happens with a design basis accident and I'm not talking about core melt. And the staff has been very specific about what you've got to qualify that since before and I'm probing about maybe the staff did add another piece to that. Okay?

MR. OSBORN: Right, so the fundamental tenants of my answer still apply, that the environment in which they're qualified to operate and has to be defined at that level of detail and it hasn't yet been done.

MEMBER REMPE: But the staff has radiation levels. They've got humidity levels. They've got a bunch of temperatures. They've got a bunch of requirements.

Yes, so I'll probe with the staff, but anyway, I appreciate us discussing it now. Thank you.

CHAIRMAN RICCARDELLA: Let me just do a check now.

Steve Schultz, are you hearing the full conversation now?

DR. SCHULTZ: Yes, we are. It seems as if it's fixed.

CHAIRMAN RICCARDELLA: Okay. Thank you.

MEMBER PETTI: Any other questions for NuScale? Okay. Thank you. Time goes fast. Thank you.

(Pause.)

MR. TESFAYE: Are you ready for us?

MEMBER PETTI: Yes, go ahead.

MR. TESFAYE: Thank you. Good afternoon, everyone. My name is Getachew Tesfaye. I am the NRC Project Manager for NuScale's topical report on accident source term and the TR as you know describes a general methodology for developing accident source terms and performance corresponding design base accidents and other required accident radiological consequence analysis to be referenced for NuScale's Small Modular Reactor and other applications are referenced in NuScale's SMR.

The NRC staff submitted an advanced topical report evaluation to this committee on October 18 and presented its finding to the NuScale Subcommittee on November 20 of this year.

Today, we will present the high-level summary of the staff's findings with a focus on a couple of items we took from the subcommittee meeting.

Jason, here to my right, and I will be making presentation. The rest of the staff are sitting in the audience and will be ready to answer any question you have.

So topical report positions to NuScale and NuScale requested a profile of 15 specific positions listed in Section 1.2 of the report. And NRC staff has determined that subject to the conditions and limitations specified in Section 6 of the SER, the methods described in the topical report are acceptable for developing accident source terms and performing accident radiological consequence analysis to be referenced by the NuScale SMR design.

The staff approved positions 2 through 15 requested in topical report. The staff did not make any finding of position 1 where NuScale categorizes a core melt accident as beyond design basis event. And the applicable NRC regulations do not require classification of source terms of design basis or beyond design basis to demonstrate compliance as a requirement.

Therefore, the staff has determined that the classification of a core melt accident as a beyond design event for the NuScale design is not material with staff's findings under this regulation. Therefore, the staff did not make a finding on position 1.

With that, I'll go to Jason to present one takeaway from the subcommittee meeting, that is the staff's independent analysis.

MR. SCHAPEROW: So one thing that the staff did as part of its evaluation of NuScale's topical report methodology was to perform an independent analysis. This was to evaluate NuScale's core damage event and their analysis of that and the off-site consequences resulting from that.

Our approach was to use MELCOR. We used MELCOR to simulate two scenarios, two core damage scenarios. One was a CVCS line break inside containment and the other was a failed open reactor vent valve. In both of these scenarios, we assumed that the ECCS failed to function properly.

So we used MELCOR. We calculated the fission prior release into the environment for the two scenarios and we took each of the two MELCOR results and we put them into RADTRAD to turn them into a dose. We predicted EAB, LPZ, and controlling doses and we used this independent evaluation to compare against what the applicant had come up with. And the doses were comparable and also they were below the regulatory dose criteria.

So this is -- again, this is one thing that we did as part of our evaluation.

Next slide, please?

So the documentation is a little bit complicated and in case the committee would like to go into a little more detail on this. So the MELCOR calculations themselves that the staff did are documented in the report from April of this year. It's listed at the top of slide 7.

We actually did calculations for three scenarios in this report. The third scenario was a bypass accident which wasn't used for the topical report review. The reason we did these three scenarios was to help the staff understand the behavior of the NuScale reactor under severe accident conditions and we also did a number of comparisons against NuScale results for severe accident simulations.

The second report listed here is -- we took the MELCOR output from the two scenarios that were in containment, had in containment releases, not to bypass accident, and again, we turned those into doses using standard -- using our RADTRAD model. So the second report documents in further detail the MELCOR results, MELCOR releases to the environment, release two scenarios, and it also explains how the releases were used in RADTRAD to calculate doses.

MEMBER PETTI: Just to be clear, you only took two of them for the dose stage.

MR. SCHAPEROW: That's correct. The third one was a bypass accident. We didn't take that through the dose stage.

MEMBER REMPE: So during our subcommittee meeting, there was some confusion, but I was looking at the correct report and although ACRS had looked at it previously, but looked for a different reason to support the PRA. And if I look at that report from MELCOR, there are a lot of postulated reasons on why there were differences in the result, whether it was nodalization, where you assume the break was.

Do you have any -- now that you've had since I think it was April when it was issued and you had more time to think about it, do you have any strong feelings on why there was so many differences? Because I think the actual doses were a factor of 2 to 3 off. They were low, like by this earlier latter stage that you probably applied, but there were some significant differences in the report.

MR. SCHAPEROW: Yes, so I've thought about -- so there's no -- because it's an integrated calculation, there's not really -- it's very difficult, it's very, very difficult to tease out exactly what factors are dominating, driving the differences.

There's a couple, in my mind, there's a couple of obvious differences. If we could explore just a little bit. One was the assumption of five percent of the iodine was vapor that NuScale had made. Another one dealt with a containment leak rate and the NRC was calculating higher leak rates. So this is some of -- maybe the bigger differences. There were some differences between the NuScale calculations and the staff calculations.

MEMBER REMPE: And even the code was different is what I had heard in the past, that you had different versions of MELCOR --

MR. SCHAPEROW: Yes --

MEMBER REMPE: -- being made.

MR. SCHAPEROW: So our comparison of our MELCOR severe accident simulation against NuScale severe accident simulation in the first document you see on the slide, there were some differences, but standing back the staff -- we don't feel the differences were significant enough to affect these kinds of calculations.

MEMBER PETTI: But in terms of the leak rate, as I recall, NuScale just assumed a leak rate. They didn't let the pressure determine the leak rate. You guys used the actual pressure of the --

MR. SCHAPEROW: So NuScale had a technical specification leak rate that they used in their analysis. We did -- I think it was done classically for many, many years which is to take the tech spec leak rate and convert that into a hole size, a very, very small hole, but convert that into a hole size and then use that for the MELCOR simulation.

MEMBER PETTI: If the pressure goes higher, then you get greater leak.

MR. SCHAPEROW: That's correct. And also, if the materials -- if the gases in the containment are different, you're going to get a different leak.

So NuScale's tech spec leak rate was based on pressurizing the containment to err at a thousand pounds. So we did that with MELCOR. We pressurized NuScale's containment to a thousand pounds and we set the hole size so that we got the 2 percent per day leak rate, I'm sorry, .2 percent per day leak rate. And then -- but that was it. We set the leak rate and then we ran our MELCOR severe accident simulations. And we ended up getting time variant leak rates, exactly.

Actually, at one point the leak rate went the other way, actually started going into the NuScale containment because in a NuScale accident before you start the heat up and generate hydrogen, you've got a vacuum in there. So you actually -- actually, at one point you draw a vacuum just before you get to the core damage stages.

MEMBER PETTI: For the benefit of the other members, it was not presented at the subcommittee, but in the report, NuScale compared STARNAUA the aerosol code they used in the containment against MAEROS which is the subroutine inside MELCOR. And they were on top of each other. So I think the aerosol physics is the same in the two codes and it has something to do with bounding conditions and initial conditions in terms of the differences.

MR. SCHAPEROW: In my mind, two of the big differences again was in one case NuScale had -- I would characterize that as a conservative approach for the amount of iodine vapor that's going to be sitting in containment hour after hour after hour. But on the other hand, we also were calculating a time dependent leak rate which in some cases went above the .2 percent per day per leak rate that NuScale had assumed.

So again, the calculations were different. We did an independent calculation and to the best of our ability to predict what would be leaving the containment and we said fed that into RADTRAD.

MEMBER REMPE: I have one question that I'd love to ask you just now and get it over with and not close the meeting, but there's been a sentence in this report that I tried to ask at the subcommittee meeting, but we were in an open and I was told it was the wrong version.

I would like to close it because it doesn't make sense and it may be a typo. But I am curious on what the sentence is and I'm afraid to say it aloud in the open session, so let's -- if we can have a brief closed session, if you don't mind.

MEMBER PETTI: If it's only on the staff's confirmatory, that can't be in the open session?

MEMBER REMPE: Well, there's some numbers in it. I sure would love to, but I'm afraid I'll get in trouble, so I don't know what to do.

MR. SCHAPEROW: Are you referring to the second report here?

MEMBER REMPE: No, the very first report, there were some hours that are cited and I don't know, the document is marked proprietary, so I don't know. I have been curious about it for the last month or so and I'd like to have my curiosity satisfied.

MR. SCHAPEROW: There is a public version of the first document. I don't know if you --

MEMBER REMPE: I did not have that. I was only given the proprietary one. I could try and read it without the numbers, but I don't think it will make much sense to you, so if you don't mind, just close and ask you the question. Thank you. Go ahead.

MR. TESFAYE: Any additional questions to Jason?

MEMBER BLEY: Joy, were you saving your question about --

MEMBER REMPE: I think I have to until we close the --

MEMBER BLEY: No, I mean the hydrogen one you were asking --

MEMBER REMPE: Oh, you mean about the aerosols and the seal crack mutation one?

MEMBER BLEY: Yes, and I was a little surprised NuScale said what they did. It sounds like they're saying you have to give the environmental conditions under which it has to work, but it would seem to me they should have set that up and should have addressed the issue you raised about particulates out there.

In any case, you heard the discussion. Is there anything you guys can say about that issue? Either the issue itself or whether that might -- somehow you're setting the environmental conditions under which the detector has to work.

MEMBER REMPE: There's a later slide that I was going to ask that part on and I don't know if Jason is the right person or not.

MR. SCHAPEROW: I have not been involved at all.

MEMBER REMPE: Slide 8 is a good place to ask it.

MR. TESFAYE: So I'm going to go over some of the high level conclusions we made in terms of all of the chapters that were impacted by the accident source term in the topical report.

One of the things the environmental qualifications the staff finds acceptable to use iodine spike source term methodology and the environment has qualification dose methodology described in Appendix B of the topical report for calculating one of that qualification, the doses inside containment and under the bioshield.

We also give a detailed discussion of the equipment survivability when core damage was not assessed for EQ. Certain equipment associated with the containment integrity and combustible gas monitoring is designed to function to withstand core damage events. Qualitative assessment testing and all additional analysis may need to be performed to ensure equipment survivability. And this evaluation is performed in Chapter 19 of the SER.

MEMBER REMPE: So here's where I was going to ask, the discussion about my concerns about fuel fragmentation and dispersal. I know this was not the primary focus, the radar base sensor wasn't the primary focus of this chapter. But do you have any thoughts about maybe that somebody needs to add something to that list of environmental conditions for this --

MR. TESFAYE: I don't know if we have the right people here in the audience.

MEMBER REMPE: I kind of expected what happened.

MS. GRADY: This is Anne-Marie Grady with NRR. And aerosols and fuel fragments are not specified under the conditions of equipment survivability neither in SECY 90-016 or 93-087. NuScale didn't provide that information and we didn't ask a question about it.

MEMBER REMPE: So again, you understand my concern and I'm sure that the guidance didn't think about this because it's a different design. The guidance wasn't written for it. So I just think it's another -- we've raised issues about this since or before in our letters and it's another thing that came to light during this discussion.

MR. TESFAYE: Thank you. The other related topic that we discussed in the subcommittee was the post-accident sampling exemption request and it's related to the topical report.

The regulation requires that applicants provide the capability to promptly obtain normalized post-accident samples from the reactor coolant system and containment atmosphere.

Since equivalent information to that provided by the sampling is provided by other means such as radiation monitors, under the bioshield, core exit thermal couplers, and hydrogen and oxy monitors.

The staff determined that a post-accident sampling need not be required. Therefore, the staff approved the exemption request for post-accident sampling for the NuScale design.

MEMBER MARCH-LEUBA: Wait, let's clarify.

MR. TESFAYE: Okay.

MEMBER MARCH-LEUBA: First, why do you say need not be required? Do you mean it's not required?

MR. TESFAYE: Yes, that's probably it. It's not required. We have other means to gather the same information as we could get from that --

MEMBER MARCH-LEUBA: Did NuScale ask for an exception not to have a hydrogen system?

MR. TESFAYE: No, they did not. In fact, they used the hydrogen monitoring to justify this exemption request.

MEMBER MARCH-LEUBA: This goes back to what I was trying to explain before. If you connect your hydrogen monitoring system downstream from the safety isolation valves of the CES, in order to operate the hydrogen system, you need to open up the containment.

MR. TESFAYE: Yes.

MEMBER MARCH-LEUBA: To a whole bunch of non-safety related components. If the equipment -- I mean in operating plans you have a hydrogen monitoring system which is non-safety related, but is connected to the safety-related containment. I mean what the design levels as defined is equivalent to opening of the containment to the turbine building and then measuring the hydrogen inside the building which would be completely crazy.

By connecting the hydrogen monitoring system to a CES and whatever the second component is, you are telling the operator, if you suspect there is a severe accident, the isolated containment and send all that contamination to these three non-safety related systems.

If we don't think a hydrogen monitoring system is needed, they should not have one. The operator should not be tempted to open the containment.

MR. TESFAYE: Okay. I'll defer that to my colleague, Anne-Marie.

MS. GRADY: This is Anne-Marie Grady again from NRR.

The means of hydrogen and oxygen post-accident monitoring is established by a closed loop. Containment atmospheric sample is taken by opening the CIV in the containment evacuation system, sending it past the two-line monitor for both hydrogen and oxygen back through the containment flooding and drain system back to the containment. So unless it leaks, it's not released to any other environment. It's a closed loop.

Severe accident mitigation is the reason why we needed to have hydrogen and oxygen monitoring and for severe accident mitigation, none of this has to be safety related.

MEMBER MARCH-LEUBA: The hydrogen loop, hydrogen monitoring loop doesn't need to be safety related.

MS. GRADY: Does not.

MEMBER MARCH-LEUBA: But you're connecting it to the containment operation system which is a two-inch -- or four-inch pipe with a valve, with a pump that goes all the way outside to the support building and comes back and all of it is non-safety qualified.

I think that by opening the isolation valves to the containment into that CES system, you are creating more problems than you're solving.

MS. GRADY: So you're concerned that they're leaving the system by some other means.

MEMBER MARCH-LEUBA: The only reason you can have a severe accident if you have a really bad day.

MS. GRADY: Yes, sir.

MEMBER MARCH-LEUBA: And most of these are seismic and that CES is going to be broken. I mean you're worried about leaking from the high-level leakage just a one-eighth inch line which is probably -- and you have this four-inch line with a big pump with seals. You are venting -- the containment bounding becomes the CES bounding.

MS. GRADY: Because it's not safety related, required to be safety related and in fact, it was not safety related. We don't have to postulate a further accident.

MEMBER MARCH-LEUBA: In operating the reactor the rule was written for, you connect the high-level system to a safety-related containment, so you're only ever creating hydrogen through your hydrogen monitoring system.

In the design proposal of NuScale, you are -- the isolated containment surrounding the CES which is a lot of a system with pumps, seals, vents and you're flooding that with all of the contamination from the containment in order to sample hydrogen. You're making the problem worse. I really don't know.

MS. GRADY: I don't follow that scenario as to how it makes --

MEMBER MARCH-LEUBA: CES has a valve, has a vacuum pump.

MS. GRADY: Right.

MEMBER MARCH-LEUBA: And seals with components when it reaches, safety goes up.

MS. GRADY: The CES --

MEMBER MARCH-LEUBA: You are dumping all the containment environment, the containment atmosphere with all those aerosols and iodine, you're putting it on your vacuum pump which is up there on the operating floor in order to measure hydrogen.

MS. GRADY: When we un-isolate the CES portion of that closed loop, the CES system isn't operating.

MEMBER MARCH-LEUBA: But it's open.

MS. GRADY: Once we open the containment isolation valve it is, yes.

MEMBER MARCH-LEUBA: Yes, so you're dumping all of the containment environment --

MS. GRADY: It's flowing through a closed loop flow path.

MEMBER MARCH-LEUBA: No, into all of it, it's in vacuum. It will fill it up with iodine and astringent.

MS. GRADY: The containment atmosphere will be in that closed loop, I agree.

MEMBER MARCH-LEUBA: Not the closed loop, the CES.

MS. GRADY: I don't think the CES system is open to any --

MEMBER MARCH-LEUBA: You just opened it.

MS. GRADY: -- any open path from the CES portion of the line we're using. I don't believe it is. I'll double check on that.

MS. BRADFORD: This is Anna Bradford from the staff. Is this a question that really should be directed to NuScale and the design in terms of why it's designed this way?

MEMBER MARCH-LEUBA: I think if we have reached the conclusion that the hydrogen system is not needed --

MS. BRADFORD: We have not reached that, no.

MR. STUTZCAGE: This is Ed Stutzcage. I'll try to clarify it for the NRC.

So the exemption that NuScale has is an exemption from physically taking grab samples, taking them to a lab to analyze it. And as part of their exemption to not need to take grab samples, they credited the hydrogen and oxygen monitors, so the monitors, you know --

MEMBER MARCH-LEUBA: It's a different exemption.

MR. STUTZCAGE: It's a different exemption. The exemption is just physically grabbing the material and analyzing it in a lab. They still have the requirement to monitor, have the monitor -- had it monitored.

MEMBER MARCH-LEUBA: But do you understand what I'm saying that in order to operate the hydrogen sampling system, you need to open up the containment isolation valve and then the containment radiation aggregate into a CES system which is a vacuum pump, it's a HEPA filter. It's a tower. You're putting all the junk, the containment, in the containment, you're sampling and you're putting it on the floor of your -- that is not reasonable.

MR. STUTZCAGE: Right, and NuScale hasn't requested an exemption from the 5044 hydrogen and oxygen monitoring requirement and that's where our concerns in radiologic rates protection comes from where you're doing this, you're operating the system and they haven't demonstrated an ability to re-isolate the system and they haven't analyzed leakage --

MEMBER MARCH-LEUBA: They can re-isolate the isolation valves to take a sample, but all the iodine and the strontium and it's already in the pump and the HEPA filter.

MR. STUTZCAGE: Right. To us, they never provided us any assurance that they could re-isolate the system.

MEMBER MARCH-LEUBA: You have the valves in there.

MR. STUTZCAGE: Go ahead, Ron.

MR. LAVERA: So the way the system works

is that they can open some of the valves from the control room. They may have to go out to the skids which are located in the 100-foot, 126-foot elevation and manually open some of the other valves.

When they go to isolate the system, it's the same thing. They can isolate some of them from the control room, one pair of them, but not the other pair. So they have to physically go out to the skids and push the buttons.

Where the staff has some concern is that the amount of leakage that you use to get from the system to cause a problem for people trying to access those valves is not the pipes falling off. The analysis that the staff did was using .3 CFM -- I think it was -- I'm going off of memory here so it's close to 30 rem to the control room operator. So it was a significant dose.

So that led us to believe that there would be issues for personnel trying to access this area even under the exposure -- elevated exposure authorization.

MEMBER MARCH-LEUBA: My claim is whatever leak rate you assume from operating this closed loop hydrogen monitoring system, multiply times a hundred because all of the leakage from the CES system.

MR. LAVERA: We were counting on the activity in containment from the core damage event going out into all these three systems and the total leakage from all these three systems being .3 CFM. So that's how we did our analysis of the --

MEMBER MARCH-LEUBA: -- so we're assuming this is a normal system --

MR. LAVERA: We were looking at that.

MEMBER MARCH-LEUBA: -- is still intact.

MR. LAVERA: So now the -- yes, and we agree with you that there's going to be seals and stuff, valves, interfacing valves that are going to leak, so we understand this. So we don't agree with the characterization that you would have to have a pipe break causing those problems.

We believe that if you do have a leak from the system that you may not even be able to isolate the system under the plan's special exposure provision to Part 20, never mind the 5 rem limits of Part 20. We believe that if you do have a leak from the system from leakage rates on the order of .3 CFM that you do present a challenge to the public health and safety. And this also impacts the LPZ zone is what we call it. And then it also impacts the control room operator dose.

So in part of our review that started on this question back on March of 2018, we were asking NuScale to tell us, hey, what is the maximum and allowable leakage rate that you could tolerate from this system and not challenge the dose to the control room and the offsite? Can you isolate the system by doing this manual actuation? Can you safely send somebody in to the area? What's the maximum dose that you can get from that?

We have not been able to get an allowable leakage value from NuScale. They don't have the ability to isolate this from the control room without sending somebody out to the field. So this is the reason the staff has concerns about this.

MEMBER MARCH-LEUBA: We were also told that to open those valves, the containment has to be below 200 psi in order to bypass.

MR. LAVERA: And they have to go out to the skid to do it. Now you wouldn't have a vacuum. After a couple of days, you will not have a vacuum in containment. You will be at 60 pounds, I think, just from the normal stuff going on. And over the course of the accident, it can go up to 160 pounds.

MEMBER MARCH-LEUBA: And all those 160 pounds of dirt are going to move into the CES system the moment you open those valves.

MR. LAVERA: So my understanding is and Anne-Marie, you're welcome to correct me if I misspeak. The people responsible for containment integrity in the hydrogen monitoring have determined that hydrogen monitoring is required.

NuScale has put in for an exemption request from that. So from a radiation protection perspective which leads us how do you know a valve leak will result in having this activity in this system. That's weighted against leakage criteria that represent a potential challenge to the control room operator and members of the public and anybody that would have to go in there and manually shut the system. So if it's the only way you have to go in there and shut the system, if you do determine that you have enough leakage that's causing problems to the control room or offsite dose, send somebody out there to push a button.

MEMBER MARCH-LEUBA: I made my point. The last time I will interrupt. Either the hydrogen system is required or it is not, but if it is required, how we need from a non-safety grade large system full of valves, seals, pumps, HEPA filters, connected to the exhaust power, all non-safety grade.

And that's my opinion.

Is it required, is it not required? If it is required, collect it like the operating plants do to the containment.

MS. GRADY: Dr. March-Leuba, it is required. It is part of NuScale's design and we've accepted it.

MEMBER MARCH-LEUBA: But the argument I'm making is if it is required, the design is defective.

MS. GRADY: The guidance --

MEMBER MARCH-LEUBA: They're making the --

CHAIRMAN RICCARDELLA: Excuse me just a second. Everyone needs to speak up louder because Corvallis can't hear what we're saying. Okay? Get closer to the mic and speak up.

MEMBER MARCH-LEUBA: Okay, this is equivalent to an operating plant, so it wants to sample the high-level in containment and you still are sampling the containment, they put the sample in the turbine building. And to sample the hydrogen, they open up the valves so the containment was in the turbine building and then they measure the hydrogen in the turbine building. I mean you would consider that ludicrous, right?

MS. GRADY: Yes.

MEMBER MARCH-LEUBA: But this is what we're doing here. We are opening up to the CES system which is a non-safety grade, vacuum pumps, HEPA filters, connected to the tower, which may or may not have isolated and you dump all your dirt into there and then you sample the CES. It's the same thing as dumping it in the turbine building. It's crazy.

MEMBER DIMITRIJEVIC: Well, Jose, we can address this also as a part of PRA because it's a matter of containment isolation during the accident, and if this actually means guaranteed containment bypass. With some accidents, that shouldn't be the case. So we should really -- I mean I made the note for myself to look into this because it seems like you will have an accident and you're going to bypass containment which is against the plan and the additional containment probability failure is less than one because it definitely in seismic cases is going to be point something. So the thing is that we have to look what does that mean from the containment condition of failure probability the safety plan.

MEMBER REMPE: So Jose, you keep bringing this up to the staff and what can they do? If somebody comes in to a design, what regulatory hook could they use?

MEMBER MARCH-LEUBA: They can tell them that this is not good enough.

MEMBER REMPE: What regulation are they breaking is where I'm kind of going? I know you tried to get the NuScale folks to do something about it and they didn't want to, so what do you do with the regulator?

MEMBER MARCH-LEUBA: The thing is the operator is more than 5 rem, you push a button, so therefore this design doesn't work. That's what I'm getting at.

MS. BRADFORD: This is Anna Bradford from the NRC. I think what you're saying is you think it's not a good idea for those systems to all be connected. That's what I'm hearing you say, right?

NuScale came in with this design. We evaluate it. They were able to meet our regulations except for where they requested exemptions and it was fine. Like you said, it's not our job to say you know, we don't think this is the best design. It would be better if we designed it this way and I don't know if that's even true, but that's really not our responsibility.

MEMBER REMPE: It's why they've got this carve out which may be difficult to meet, but they've got a carve out. I mean I get what you're saying. It's kind of like I don't what I'd do if I was in staff's position.

MEMBER MARCH-LEUBA: What I would do is have them give me a probability of failure of the CES system in a severe accident event. The CES system --

MEMBER DIMITRIJEVIC: I mean they will not meet safety goal if this is an inability to fail for an accident, definitely. So that's why they would call that.

MEMBER BLEY: Well, it's also not so much what can the staff do about it. We advise the Commission. If we really think this is a problem and the regulations don't cover it, then it's up to us to raise it to the Commission and say for this new kind of design it ought to be there. I'm not saying I'm of that opinion, but that is a way for us to proceed.

MEMBER KIRCHNER: Can I recap where we might be? And that is the applicant has asked for an exemption from post-accident sampling. Is your granting that because they can provide equivalent information by sampling by other means? So one is radiation monitored under the bioshield. That will tell you something. Core exit thermocouples. And then hydrogen and oxygen monitors.

Now specifically then this system would be necessary --

MR. TESFAYE: Absolutely --

MEMBER KIRCHNER: -- to support your exemption.

MR. TESFAYE: Absolutely.

MEMBER KIRCHNER: And then the issue is what dose would be at risk for the operators to operate the system and then to isolate it?

MR. TESFAYE: Yes, to open the containment, I think we have evaluated that.

MEMBER KIRCHNER: Notwithstanding the .3 CFM leak rate and the containment evacuation system, what's the dose just in the pipe from the piping when it's filled with all of the containment atmosphere? Do you have a ballpark number for that?

MR. STUTZCAGE: I don't think we have that. We only reviewed the dose to un-isolate the system and --

MEMBER KIRCHNER: Yes, I think that's what was presented by Anne-Marie and the staff. You proposed a leak rate and then there's a dose associated with that. If the system doesn't leak, what is the dose? There will be dose.

MR. LAVERA: There will be dose, so it

won't be -- first of all, NuScale's proposal is if the system doesn't leak you don't change anything, you just let it go, and we're okay with that. There's no need to go out there and re-isolate the system.

If you have a leak, it's most likely that the airborne cloud around the area is going to be the major dose driver. We didn't do that because NuScale didn't specify a maximum allowable leakage rate, so we didn't do the dose calculation for that specific area and there's other issues that were keeping us from trying to do that calculation.

We were able to do the calculation for the control room dose and the LPZ and those calculations led us to believe that it could be a significant problem for public health and safety.

MEMBER KIRCHNER: Well, I think Jose has eloquently stated the design concerns that we have, that you open up -- you bypass containment, open up a large, I believe that line is four inches to penetration. And that is a concern from the design standpoint. Although we're not here to re-design the system. We stated that in our subcommittee meeting.

MEMBER BLEY: We must have written a letter on the SER with open items on Chapter 9. Did we raise this back then? Is it in our letter?

MEMBER MARCH-LEUBA: Probably not.

MEMBER KIRCHNER: I don't know that that detail was available then. It may have been and we just didn't cover it.

DR. CORRADINI: Can I ask a question? I want to make sure that the two line requirements are both short-term monitoring and long-term monitoring or just short term?

MS. GRADY: Continuous, long-term.

DR. CORRADINI: And so long term is defined within 30 days. So short term is of no consequence to the staff. It's the long-term monitoring that's --

MS. GRADY: For this particular change, Dr. Corradini, the hydrogen and oxygen monitoring has to be established by 72 hours. Before then, the containment integrity is not challenged, even if there is combustion in the containment.

Long term, we looked at and NuScale looked at up to 60 days and there's a potential challenge again due to the fact that there's radiolysis around 45 to 54 days, but that's long term.

DR. CORRADINI: Okay, just so -- let me repeat. I want to make sure I'm clear about the regulatory requirement. The regulatory requirement is they must establish hydrogen monitoring before 72 hours.

MS. GRADY: They must establish it and they have shown us that they don't need to do it before 72. Seventy-two is not in the regulation.

DR. CORRADINI: Okay, excuse me. I'm sorry. Thank you. Thank you for clarifying my point.

And then once established, then according to regulation, it must be maintained continuously after that --

MS. GRADY: Yes.

DR. CORRADINI: Or intermittently?

MS. GRADY: No, continuously after that. Practically speaking, it could be intermittent if that were an operationable decision, but the regulation is continuous.

DR. CORRADINI: Okay. Thank you.

MS. GRADY: You're welcome.

DR. CORRADINI: Thank you, Anne-Marie.

MEMBER PETTI: So my question is the source term is where at 72 hours in these calculations? These calculations of source term is weighed out. All the aerosols have settled. The steam is condensed. So what source term did you use in your analysis? Because your big peak, I'm with you, but I think there's a timing offset here that might be important.

MR. SCHAPEROW: So just to maybe throw out a fact idea, so NuScale's assumptions for their source term topical is that 5 percent remains airborne forever, at least for 30 days.

So that might be the source of -- I can't speak for Michelle Hart. Unfortunately, she's not here today. There is an assumption, a conservative assumption in NuScale's topical report in the area of iodine vapor.

MEMBER REMPE: And Dennis, because it may come up later this week with respect to the letter on Chapter 9, one of our conclusions was there were potentially risk-significant items in NuScale's design that are not yet fully developed. So these items, requirements to be included in the DCA to ensure that the licensee's plant will perform as credited.

So we didn't call out this particular item, but we acknowledged that we were uncertain about a lot of aspects in the plant design.

MEMBER BLEY: And there's a lot of parts to Chapter 9.

MR. TESFAYE: Okay. Thank you.

DR. CORRADINI: There's silence again. May I get another clarification point just to be clear?

So it's NuScale's contention that they don't -- that their design will meet the requirement if they can be exempt from long-term monitoring? I want to make sure I understand what the exemption is that is being requested. I'm sorry that I'm going over old ground.

MEMBER PETTI: No, I think to be clear there's an exemption from physical sampling. They actually need the hydrogen and oxygen monitoring to support the exemption. Have I got it?

MS. GRADY: That's my understanding of it.

DR. CORRADINI: And then NuScale has gone further to say that they can go in an un-isolate and re-isolate if necessary with operator action. Am I understanding that correctly?

MR. STUTZCAGE: This is Ed Stutzcage at the NRC. They provided information to show that they can un-isolate the system. They have not provided information to the NRC to demonstrate that they can re-isolate the system.

They have indicated that that's something that will be handled as part of their emergency action, if necessary. They didn't say -- respond, they don't have to provide that information to the staff at this time.

DR. CORRADINI: Okay, thank you. Thank you for the clarification, I appreciate it.

MR. TESFAYE: Okay. Thank you. I think we have discussed this, this slide --

PARTICIPANT: Just a little.

MR. TESFAYE: -- the last 15, 20 minutes, so I'm not going to go over that. So I will jump straight to what the subcommittee requested us to present at this meeting, which is the proposed recommendation to the rulemaking.

I am not going to read this. This is out of the Chapter 12 SER. I am just going to highlight the areas where we are going to focus. Specifically, 10 CFR Part 52, Appendix 2, which is not there yet, that will be the NuScale SMR appendix.

Under issue resolution we will state the design and evaluation of leakage from combustible gas monitoring loop is not considered but it was in the meaning of 52.63 which is with respect to the finality of the standard design.

And then in Section 14, Additional Requirements, it will be stated a COL applicant is responsible for providing sufficient design information demonstrating that the requirements of 10 CFR 50.34(f)(2)(xxv)(8) are met with respect to potential radiation release under accident conditions from systems used for post-accident hydrogen and oxygen monitoring.

So this is what we are recommending, and, again, I note this is not the proposed rule language. This is what is in the SER. The rule language has not yet been developed yet.

So as an example on the next slide I give you two carve outs, as we call, carve outs of recommendation. This is from the design specification rule for ESBWR design an applicant for COL include as part of its application.

One of them is for the hurricane loads in excess of total tornado loads and hurricane- generated missile loads, so on the structures this was not part of -- It was in the design specification a scope, but it was not done so they carved out or they included this in the rulemaking.

And the other one is similar to what we are doing here, that's the spent fuel pool level instrumentation was not fully developed in the design specification rule.

Another way to handle this is to include this kind of information in 2-1 Chapter 4 under interface requirements with an ITAC and that would have an easier way to go but the applicant chose not to include this language in the Tier 1 interface requirement.

So the staff is kind of forced to do this rule carve out in the design specification rulemaking. So this is, again, the takeaway from the subcommittee meeting.

There was other items that was requested of us. Chapter 12 which had all this recommended rulemaking language, we gave you the draft of that and when we issued the final there was some change to the draft and we have provided the compare and contrast between the draft and what the final one.

The major difference is the ventilation system fire dampers, which is the second item here. Obviously we didn't have enough information. The ventilation dampers were not closing on high radiation monitor.

The staff looked at the risk and they said the primer is to operator or equivalence of operability involves core damage event with a failure of the ventilation's exhaust fans as well as an open bay exhaust damper, so all these three things have to happen.

And before we issue the final SER for Chapter 12 we made a finding on this although the design did need something in order to be fully responsive to the staff's questions.

It wasn't, but the staff took the risk approach and whatever to make a finding on this. So we have two rule carve outs, one is the penetration shielding design, which is the first bullet, and we have discussed that at subcommittee, and the other one is the leakage issue that we discussed earlier.

MEMBER BLEY: And up on Slide 11 where you started this rulemaking discussion the rule would state that the COL applicant is responsible for --

MR. TESFAYE: Providing the information --

MEMBER BLEY: -- providing the information.

(Simultaneous speaking.)

MR. TESFAYE: -- information, and making sure the regulations are met in terms of those.

MEMBER BLEY: Okay.

MR. TESFAYE: Or, you know, design a means to re-isolate the containment. So if you don't have any questions on this, I think we've discussed this at length, we'll go to the conclusion.

Staff found acceptable the methods for

developing accident source terms and performing accident radiological consequence analysis to be referenced by NuScale SMR design.

All phase three SER open items related to the accident source term methodology have been closed except those involving penetration shielding and the leakage from hydrogen/oxygen monitoring system.

They are not considered resolved and must be addressed by the COL applicant. And that's all we have.

MEMBER KIRCHNER: May I go to the first one then. When you push of, pardon my phraseology, the responsibility for the radiation shield wall design to the COL, I'm trying to think through the implications of that.

The applicant has a nominal design for the shield blocks and so on. If it turns out, and I'll just do this rhetorically, that twice as much shielding is needed to meet whatever the dose criteria are that has implications that ripple through the design, simple things like the building, the main, the reactor building crane operations, et cetera, and potential dose during refueling operations, et cetera.

I am wondering what the ramifications are of making that a COL applicant responsibility. Can you talk through that?

MR. LAVERA: So this is part of the reason that we went down this path is we wanted to make sure that this got designed appropriately.

We recognized that the potential interactions of the shielding that they would have to install, it's equivalent to five feet of concrete over what appears to us to be a fairly large area, so we, too, are concerned about that.

We tried to work with NuScale to determine several ways of addressing it within the scope of the application without having physical design information there.

The only way we could reach a safety finding on this was to do a carve out, so that's why we went down that path.

MEMBER BLEY: Well I said this at the subcommittee meeting, but putting this off on the COL -- Well, I'm not NuScale, but if I were this would make it a lot harder to deal with potential customers when they look at this and say, hey, I got to make this work after I commit to this design. It just seems a bad place to leave things.

MEMBER KIRCHNER: Yes. I am thinking through the ramifications, because, pardon the digression, but if I remember right the initial lifting mechanism for moving the modules was to kind of strap on to two solid anchors, take it up.

Then I believe that changed so that the upper frame then connected to the module and became the lifting point and the interactions of that design, which may be in FLEX, I'm not sure where that design came out, and the shielding are, there is important ramifications there as they change that in terms of, as you labeled this, large penetrations in the shield wall and others.

So have you looked at that at the latest iterations on that upper lifting design and the ramifications for radiation protection?

MR. LAVERA: Okay, so, yes, we have been looking at that shield block on the top of the module bay. This shielding is not anywhere near that. It won't interact with that particular issue, particular thing.

MEMBER KIRCHNER: Right.

MR. LAVERA: So I understand where you are coming from, but there is absolutely no interaction between those two.

There are other interactions, potential interactions for equipment, locations, weight, structural loading, structural analysis for components in structures outside of the module bay wall on the 100-foot and out.

So it's not the module shield that's on top that you lift with a crane and move it around.

MEMBER KIRCHNER: So it's more the penetrations into the reactor building?

MR. TESFAYE: Yes.

MR. LAVERA: So, yes, that's a closer approximation to it.

MEMBER KIRCHNER: Okay.

MR. LAVERA: It's between the power module bay and the rest of the reactor building.

MEMBER KIRCHNER: Thank you.

MEMBER PETTI: Any other questions?

MEMBER REMPE: Well I wanted to --

MEMBER PETTI: I know that though. Do we ask for public comment around?

PARTICIPANT: Yes.

PARTICIPANT: Yes, we do, and we have some --

(Simultaneous speaking.)

MS. FOSAAEN: This is NuScale Corvallis if I could just make a quick statement with regard to the shielding. I just want to clarify that the shielding that we have not provided is just the shielding around piping penetrations in the ventilation.

The rest of the information and what we have provided is consistent with the level of information provided by previous applicants.

So, you know, we're talking about what material goes around the piping equivalent and we did provide a COL item that said the shielding that would be provided in those penetrations around the piping would be equivalent to the dose rate maps that were provided as part of the DCD.

So we had provided, in fact, with that COL item more than previous applications.

MR. LAVERA: So this is Ron Lavera. You know, I have been involved in the previous reviews and when you're talking about having a small gap around a pipe or a small pipe, yes, the NuScale application is consistent with that.

We are looking at penetration for main steam, main feedwater lines, these are big penetrations.

The ventilation ducts, which are feet in size, and you're not talking about a little bit of shielding, you're talking five feet of concrete shielding that they are crediting both for occupational, EQ, and beyond design basis equipment survivability considerations.

So in our -- The way we look at things that is not an inconsequential something that you should be able to just squirt a little goop in there and move on your way.

MEMBER KIRCHNER: Yes. Yes, but they are the concretes tight fit around one pipe probably.

MR. LAVERA: And if you were to try to do shadow shielding it would be a significant way to interfere with the equipment that is there. Like I said you have main steam lines and other things there, so we have concerns about physically being able to fit the equipment in there, the shielding in there when the other equipment is present.

MEMBER PETTI: Okay. Let's try to take public comment. Anybody in the room?

(No response.)

MEMBER PETTI: Seeing no one, anybody on the public line want to make a comment?

(No response.)

MEMBER PETTI: Okay. Then we'll adjourn this part of the meeting and go into closed session.

(Whereupon, the above-entitled matter went off the record at 4:10 p.m.)

NuScale Design Certification Application Accident Source Term Methodology Topical Report and Related Topics

Presentation to the ACRS Full Committee

December 4, 2019

Staff Review Team

- Technical Staff
  - Michelle Hart, NRR
  - Jason White, NRR
  - Jason Schaperow, NRR
  - Tony Gardner, NRR
  - Ed Stutzcage, NRR
  - Ron LaVera, NRR
  - Shawn Campbell, RES
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  - Anne-Marie Grady, NRR
  - Boyce Travis , NRR
  - Ryan Nolan, NRR
  - Zach Gran, NRR
  - Amanda Marshall, NSIR

- Project Managers
  - Getachew Tesfaye – Chapter PM
  - Greg Cranston – Lead PM

Contents

- NuScale SMR Accident Source Term Methodology

- Staff Independent Analysis

- Accident Source Term Related Major Topics
  - Environmental Qualification and Equipment Survivability
  - Post Accident Sampling (PAS) Exemption
  - Hydrogen and Oxygen Monitoring Radiological Review

NuScale SMR Accident Source Term Methodology

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- In Topical Report TR-0915-17565, Revision 3, NuScale requested approval of 15 specific positions listed in Section 1.2 of the report.

- The NRC staff has determined that, subject to the conditions and limitations specified in Section 6.0 of this SER, the methods described in the topical report are acceptable for developing accident source terms and performing accident radiological consequence analyses to be referenced by the NuScale SMR design.

NuScale SMR Accident Source Term Methodology

- The staff approves Positions 2 through 15 requested in the topical report.

- The staff does not make a finding on Position 1 where NuScale categorizes a core melt accident as a beyond-design-basis event. The applicable NRC regulations do not require classification of source terms as “design basis” or “beyond design basis” to demonstrate compliance with the requirements. Therefore, the staff has determined that the classification of a core melt accident as a beyond-design-basis event for the NuScale design is not material to the staff's findings under these regulations. Therefore, the staff does not make a finding on Position 1.

Staff Independent Analysis

- Objective: Evaluate NuScale’s methodology for core-damage-event offsite radiological consequence assessment

- Approach:

  - Use MELCOR to predict releases to the environment for 2 scenarios

  - Input MELCOR-predicted releases to the environment into RADTRAD to predict EAB, LPZ, and control room doses

- Conclusion: Staff’s predicted doses were comparable to applicant’s predicted doses and were below regulatory dose criteria

Staff independent analysis - reports

- "Independent MELCOR Confirmatory Analysis of NuScale Small Modular Reactor," RES/FSCB 2019-01, April 2019 (ML19205A016)
  - Documents staff's MELCOR calculations for 3 scenarios (LEC-06T, LCC-05T, LCU-03T)
  - Helps understand behavior of NuScale under severe accident conditions
  - Compares the staff's severe accident predictions with NuScale's
- "Independent Confirmatory Analysis for NuScale Offsite Radiological Consequence Assessment," RES/FSCB 2019-03, August 2019 (ML19240A046)
  - Documents the fission product releases to the environment from the staff's MELCOR calculations for LEC-06T, LCC-05T
  - Explains how the releases were input into the staff's RADTRAD analysis

Accident Source Term Related Topics

Environmental Qualification and Equipment Survivability

- The staff finds it acceptable to use the iodine spike source term methodology and the environmental qualification dose methodology described in Appendix B of the topical report for calculating environmental qualification (EQ) doses inside containment and under the bioshield.

- While core damage was not assessed for EQ, certain equipment associated with containment integrity and combustible gas monitoring is designed to function to withstand core damage events. Qualitative assessments, testing, and/or additional analyses may need to be performed to assure equipment survivability. This evaluation is performed in Chapter 19 of the SER.

Accident Source Term Related Topics

U.S.NRC
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Post Accident Sampling (PAS) Exemption

- 10 CFR 50.34(f)(2)(viii) requires that applicants provide the capability to promptly obtain and analyze post-accident samples from the reactor coolant system and containment atmosphere.
- Since equivalent information to that provided by sampling is provided by other means, such as radiation monitors under the bio-shield, core exit thermocouples, and hydrogen and oxygen monitors, the staff determined that post-accident sampling need not be required. Therefore, the staff approves the exemption from post-accident sampling for the NuScale design.

Accident Source Term Related Topics

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Hydrogen and Oxygen Monitoring Radiological Review

- Post-accident hydrogen and oxygen monitoring can be safely established.
- NuScale did not specify an acceptable amount of leakage and did not assess the leakage from the Hydrogen and Oxygen monitoring systems in the main control room or offsite dose assessment.
- Staff calculations using the limited amount of available information indicates the potential for leakage from these system to be a significant contributor to offsite and MCR dose limits and could potentially result in exceeding dose limits.
- The applicant has not demonstrated a capability to re-isolate the systems, so it is unclear if unacceptable leakage can be mitigated.

Accident Source Term Related Topics

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Hydrogen and Oxygen Monitoring Radiological Review – Recommended wording for Rule making:

- Therefore, the NRC staff recommends that the Commission include language in the proposed rule stating that the NRC is not making a finding on the design of components to minimize and control leakage from systems outside containment. This includes potential leakage from these systems that could impact the offsite dose analyses, the dose analyses for the MCR, and if necessary, the ability to safely re-isolate these systems after monitoring has been initiated. Specifically, 10 CFR Part 52, Appendix G for the DC for the NuScale SMR, Section VI, "Issue Resolution," will state that the design and evaluation of the leakage from the combustible gas monitoring loop is not considered resolved within the meaning of § 52.63(a)(5) and Section IV, "Additional Requirements and Restrictions," will state that the COL applicant is responsible for providing sufficient design information demonstrating that the requirements of 10 CFR 50.34(f)(2)(xxviii) are met with respect to potential radiation releases under accident conditions from the systems used for post-accident hydrogen and oxygen monitoring. The COL applicant is to provide assurance that post-accident leakage from these systems does not result in the total MCR dose exceeding the dose criteria (i.e. 5 rem) for the surrogate event with significant core damage and/or include design features in accordance with 10 CFR 50.34(f)(2)(xxvi) and 10 CFR 50.34(f)(2)(xxviii) to provide assurance that the dose criteria are not exceeded. The COL applicant will also provide information to verify, as appropriate, that post-accident leakage from these systems does not result in the total dose for the surrogate event with significant core damage exceeding the offsite dose criteria, as required by 10 CFR 52.47(a)(2)(iv). In addition, if manual actuation is required to re-isolate the system in order to contain potential leakage, the COL applicant will demonstrate that this can be done safely and within the requirements of 10 CFR 50.34(f)(2)(vii).

Accident Source Term Related Topics

U.S.NRC
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Examples of Rule Language from Previously Certified Design:

- Appendix E to Part 52—Design Certification Rule for the ESBWR Design

  An applicant for a COL … Include, as part of its application:

  IV(g). Information demonstrating that hurricane loads on those structures, systems, and components described in Section 3.3.2 of the generic DCD are either bounded by the total tornado loads analyzed in Section 3.3.2 of the generic DCD or will meet applicable NRC requirements with consideration of hurricane loads in excess of the total tornado loads; and hurricane-generated missile loads on those structures, systems, and components described in Section 3.5.2 of the generic DCD are either bounded by tornado-generated missile loads analyzed in Section 3.5.1.4 of the generic DCD or will meet applicable NRC requirements with consideration of hurricane-generated missile loads in excess of the tornado-generated missile loads.

  IV(h). Information demonstrating that the spent fuel pool level instrumentation is designed to allow the connection of an independent power source, and that the instrumentation will maintain its design accuracy following a power interruption or change in power source without requiring recalibration.

Accident Source Term Related Topics

U.S.NRC
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Other related areas where NRC is not making a finding on design finality:

- Large Penetrations in the Radiation Shield Wall:

  The penetrations and penetrations shielding design were not finalized at the design certification stage. NuScale has stated that it would be completed in a future phase of the design, that will be the responsibility of the COL applicant. Therefore the staff recommends that the Commission include language in the proposed rule stating that the NRC is not making a finding on the adequacy of the necessary shielding.

- Ventilation System Fire Damper:

  NuScale application neither describes the instruments and controls for closing the dampers on a signal other than smoke or fire (e.g., high radiation) nor states that the operators will perform a manual action to shut the fire dampers following an accident. However, using a risk informed approach the staff is not recommending a rule language to include a means to close the dampers on high radiation. The primary risk to operators or equipment survivability involves a core damage event with a failure of the RBVS exhaust fans as well as an open NPM bay exhaust damper. The NRC staff concludes that there is a low risk of these events occurring concurrently.

Conclusion

- Staff found acceptable the methods for developing accident source terms and performing accident radiological consequence analyses to be referenced by the NuScale SMR design.
- All Phase 2 SER open items related to accident source term methodology have been closed except those involving the penetration shielding and the leakage from the Hydrogen and Oxygen monitoring systems that are not considered resolved and must be addressed by the COL applicant.

Abbreviations

CDE	core damage event
CDST	core damage source term
COL	combined operating license
CRHS	control room habitability system
CRVS	normal control room HVAC system
CVCS	chemical and volume control system
DBST	design basis source term
DCA	design certification application
DF	decontamination factor
EQ	environmental qualification
FHA	fuel handing accident
HVAC	heating ventilation and air conditioning
LWR	light water reactor
MHA	maximum hypothetical accident
MSLB	main steam line break
pH_T	temperature dependent pH
PWR	pressurized water reactor
REA	rod ejection accident
rem	Roentgen equivalent man
RG	regulatory guide
RVV	reactor vent valve
SECY	Commission paper
SGTF	steam generator tube failure
SMR	small modular reactor
SSCs	structures, systems and components
TEDE	total effective dose equivalent
TR	topical report

December 3, 2019 Docket No. PROJ0769

U.S. Nuclear Regulatory Commission
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One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Presentation Materials Titled “ACRS Full Committee Presentation: Accident Source Term Phase 5 Implementation,” PM-1219-68131, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) NuScale Full Committee Meeting on December 4, 2019. The materials support NuScale’s presentation of Topical Report, “Accident Source Term Phase 5 Implementation.”

The enclosure to this letter is the presentation titled “ACRS Full Committee Presentation: Accident Source Term Phase 5 Implementation,” PM-1219-68131, Revision 0.

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

If you have any questions, please contact Carrie Fosaaen at 541-452-4126 or at cfosaaen@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
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Enclosure: “ACRS Full Committee Presentation: Accident Source Term Phase 5 Implementation,” PM-1219-68131, Revision 0

Enclosure:

"ACRS Full Committee Presentation: Accident Source Term Methodology Phase 5 Implementation," PM-1219-68131, Revision 0

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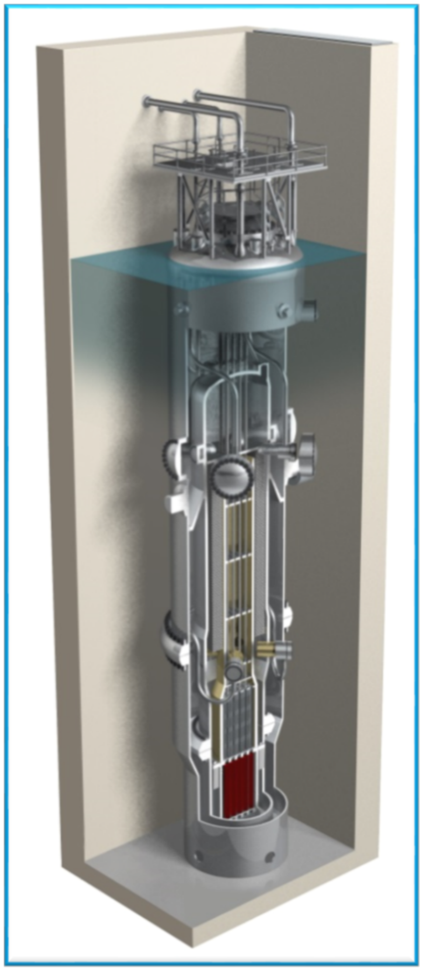
NuScale Nonproprietary

ACRS Full Committee Presentation



Accident Source Term Phase 5 Implementation

December 4, 2019

PM-1219-68131
Revision: 0

Template #: 0000-21727-F01 R5

Risk Significance

- Because of the very low frequency of core damage events, the sequences in which the hydrogen monitoring system could be in operation are negligible
  - Risk = Frequency x Consequence
- Sequences that contribute to the core damage frequency for an operating module with intact containment are on the order of <3E-11/mcyr (Table 19.1-18, FSAR)
- If leakage were to increase the dose (consequence) by a factor of two, there would NOT be an appreciable change to risk. Even if the dose increased by an order of magnitude, the risk would still be insignificant
- "In any licensing review or other regulatory decision, the staff should apply risk-informed principles when strict, prescriptive application of deterministic criteria … is unnecessary to provide for reasonable assurance of adequate protection of public health and safety." SRM for SECY-19-0036, July 2, 2019.

Hydrogen Monitoring System Leakage

- The hydrogen monitoring system is included in the Leakage Monitoring Program, required by post-TMI action item III.D.1.1
- Therefore the only way there would be an increase in leakage during a severe accident is if it induced a concurrent pipe break in the monitoring system
  - The most probable initiating event that could induce a concurrent pipe break in the monitoring systems is a very large seismic event, which is assumed to result in a containment bypass, and hydrogen monitoring is therefore irrelevant.

Hydrogen Monitoring System Leakage

- What if – the hydrogen monitoring system leaks excessively? The operators have the ability to isolate the leak.
  - Because this is an unplanned and unanticipated emergency response action, there are no explicit regulatory dose acceptance criteria.
  - In the Brunswick SER for Hardened Vents, dated 11/21/2019, the NRC states, “there are no explicit regulatory dose acceptance criteria for personnel performing emergency response actions during a beyond-design-basis severe accident.”
  - Therefore, the 5 rem limit of 10 CFR 50.34(f)(2)(vii) does not apply to emergency response actions during a beyond design basis event.

Hydrogen Monitoring System Leakage

- The hydrogen monitoring system is only used for severe accidents and can therefore be classified as non-safety related.
  - Regarding 10 CFR 50.44, 68 FR 54123 “Combustible Gas Control in Containment” states, “The final rule … relaxes the requirements for hydrogen and oxygen monitoring equipment to make them commensurate with their risk significance.”
- It is not appropriate to relax the requirements based on risk significance, and then penalize the design by presuming it will leak because it is non-safety related.
- Per RG 1.183, offsite dose consequence evaluations are not required for containment venting/purging, if only used for severe accidents.

Acronyms

FR	Federal Register
Mcyr	module critical year
SER	Safety Evaluation Report
SRM	Staff Requirements Memo
TMI	Three Mile Island

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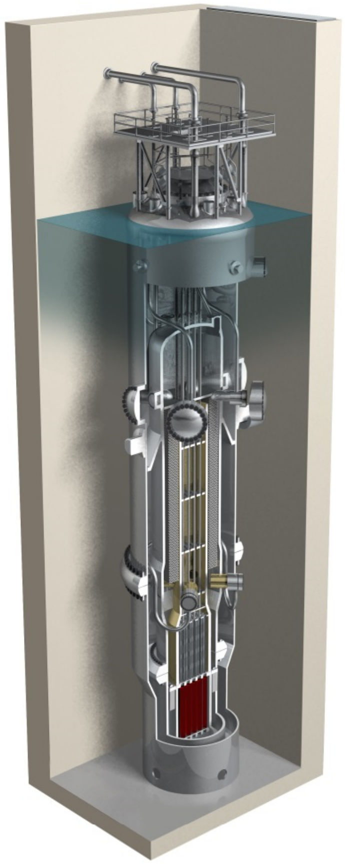

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United States Nuclear Regulatory Commission

Protecting People and the Environment

Advisory Committee on Reactor Safeguards

Peach Bottom Atomic Power Station Units 2 and 3

Subsequent License Renewal

December 4, 2019

Bennett Brady, Senior Project Manager

Division of New and Renewed Licenses (DNRL)

Office of Nuclear Reactor Regulation

Presentation Outline

- Overview of Safety Review of Peach Bottom SLRA
- SER Section 2, Scoping and Screening Review
- SER Section 3, Aging Management Review
- SER Section 4, Time-Limited Aging Analyses
- Closure of Confirmatory Item
- SLRA Review Conclusion
- Region I Initial License Renewal Inspection and Plant Material Conditions and Conclusion
- Summary Conclusion

Overview of Safety Review of Peach Bottom SLRA

Unit	Initial License	Initial License Renewal Application	Renewed License	Expiration Date	Subsequent License Renewal Application
2	10/25/1973	07/02/2001	05/07/2003	08/08/2033	07/10/2018
3	07/02/1974	07/02/2001	05/07/2003	07/02/2034	07/10/2018

- Application Submitted – July 10, 2018
- Acceptance Determination – September 6, 2018
- Safety Evaluation Report with Confirmatory Item – October 7, 2019
- Safety Evaluation Report – November 19, 2019

SLRA Audits and Inspections

	Dates	Location
Operating Experience Audit	September 17-27, 2018	Rockville, MD
In-office Audit	November 13, 2018 - April 29, 2019	Rockville, MD

SER Overview

- **SER with Confirmatory Item Issued October 7, 2019**
  - Confirmatory Item 3.0.3.2.3-1 on BWR Vessel Internals
- **Safety Evaluation Report issued November 19, 2019**
  - Confirmatory Item 3.0.3.2.3-1 closed
- **Requests for Additional Information (RAIs)**
  - 48 RAIs issued, 4 of which were follow-up RAIs

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SER Section 2

Structures and Components Subject to Aging Management Review (AMR)

- Section 2.1 Scoping and Screening Methodology
- Section 2.2 Plant Level Scoping Results
- Sections 2.3, 2.4, and 2.5 Scoping and Screening Results

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SER Section 3

Aging Management Review (AMR)

- Section 3.0 Use of the Generic Aging Lessons Learned Report
- Section 3.1 Reactor Vessel, Internals, and Reactor Coolant System
- Section 3.2 Engineered Safety Features
- Section 3.3 Auxiliary Systems
- Section 3.4 Steam and Power Conversion Systems
- Section 3.5 Containment, Structures, and Component Supports
- Section 3.6 Electrical and Instrumentation and Control Commodities

SER Section 3

3.0.3 - Aging Management Programs (AMPs)

SLRA - Original Disposition of AMPs	SER - Final Disposition of AMPs
• 11 new GALL programs – 8 consistent – 3 consistent with exceptions • 35 existing GALL programs - 8 consistent - 27 consistent with enhancements/exceptions • 1 plant specific with enhancement	• 11 new GALL programs – 8 consistent – 3 consistent with exceptions • 35 existing GALL programs - 8 consistent - 27 consistent with enhancements/exceptions • 1 plant specific with enhancement

SER Section 4

Time-Limited Aging Analyses (TLAAs)

- 4.1 Identification of TLAAs
- 4.2 Reactor Vessel and Internals Neutron Embrittlement Analyses
- 4.3 Metal Fatigue Analyses
- 4.4 Environmental Qualification of Electric Equipment
- 4.5 Concrete Containment Tendon Prestress Analysis
- 4.6 Primary Containment Fatigue Analysis
- 4.7 Other Plant-Specific TLAAs

Closure of Confirmatory Item 3.0.3.2.3-1 BWR Vessel Internals

<u>Issue</u> SLRA, AMP B.2.1.7 “BWR Vessel Internals” proposed and enhancement to either:

- install core plate wedges <u>or</u>
- submit for NRC approval an inspection plan for the core plate rim hold-down bolts to mitigate stress corrosion cracking.

<u>Resolution</u> Applicant revised the AMP B.2.1.7 enhancement to be in accordance with BWRVIP-25, Revision 1 to:

- install wedges <u>or</u>
- inspect core plate rim hold-down bolts, <u>or</u>
- demonstrate instead via analysis that the installation of wedges and inspections of the core plate rim hold-down bolts are not required.

SLRA Review Conclusion

On the basis of its review of the SLRA and the resolution of the confirmatory item, the staff determined that the requirements of 10 CFR 54.29(a) have been met for the subsequent license renewal of Peach Bottom Atomic Power Station Units 2 and 3.

Region I Initial License Renewal Inspections

- Five to ten years following the entry into the period of extended operation the Region conducts one additional license renewal team inspection—IP 71003 Phase 4.

- The team examines a sample of AMPs to verify the effects of aging were being managed effectively to ensure structures, systems, and components in the scope of these programs maintained the ability to perform their intended functions.

Region I AMP Inspections

The Peach Bottom IP 71003 Phase 4 <u>initial</u> license renewal inspection was performed in November 2018 on both Units 2 and 3.

- Flow Accelerated Corrosion Program (existing)
- Maintenance Rule Structural Monitoring Program (existing)
- Ventilation System Inspection and Testing Activities (enhanced)
- Outdoor, Buried and Submerged Component Inspection Activities (enhanced)
- Fire Protection Activities (enhanced)
- In-accessible Medium Voltage Cables not subject to 10 CFR 50.49 Environmental Qualification Requirements (New)

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Inspection of Plant Material Condition

- Reactor Oversight Process performance indicators and findings indicate plant material condition meets regulatory requirements.
- Resident Inspector routine plant walkdowns support this conclusion.
- Resident and Region based inspectors continue to inspect and assess the licensee performance to manage the effects of aging through the baseline inspection program.

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NRC Inspection Results

The inspectors found the licensee's aging management programs were being effectively implemented in accordance with the facility's renewed license. The NRC will continue to monitor AMPs using the baseline Reactor Oversight Process.

Summary Conclusion

- The staff has completed its presentation and conclusions on the safety review of the Peach Bottom SLRA and the Region I conclusions on inspections and plant material conditions.

- Additional questions

Peach Bottom Atomic Power Station, Units 2 and 3 Subsequent License Renewal Application

ACRS Full Committee Presentation
December 4, 2019

Exelon

Introductions

• Mike Gallagher	VP, License Renewal
• Anna Krause	PB Sr. Mgr. Design Engineering
• Paul Weyhmuller	LR Technical Manager
• Julian Laverde	PB Mechanical Design Manager
• Dave Distel	LR Licensing Engineer

Agenda

• Introductions	Mike Gallagher
• Station Description and Overview	Anna Krause
• GALL Consistency and Commitments	Paul Weyhmuller
• Confirmatory Item	Julian Laverde
• Technical Topics	Julian Laverde
• Closing Remarks	Mike Gallagher

Peach Bottom Station

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Peach Bottom Current Performance

- Plant operates on 24 month refueling cycle

- Plant Capacity Factor:
  - 2018 94.2%
  - 2019 96.2% (as of 10/31)

- Regulatory Status
  - ROP Action Matrix Column 1 (Licensee Response/Baseline Inspection)
  - All ROP Indicators are Green

Exelon.

Station Overview

Peach Bottom	Unit 2	Unit 3
Full Power License - 3293 MW_t	10/25/1973	7/02/1974
5% Power Uprate to 3458 MW_t	1994	1995
Independent Spent Fuel Storage Installation (ISFSI)	2000	
First License Renewal Approval	2003	2003
15% EPU to 3951 MW_t	2014	2014
1.66% MUR to 4016 MW_t	2017	2017
Current License Expiration	8/08/2033	7/02/2034

Significant Plant Modifications

Peach Bottom	Unit 2	Unit 3
Main Condenser Upgrades (titanium tubes)	1991	1991
Hydrogen Water Chemistry	1997	1997
Noble Metal Chemical Addition	1998	1999
Main Power Transformers	2010	2009
RPV Core Spray Piping Upgrade	Not Required	2013
Torus Recoat	2012	2013
RHR Cross-tie Modification (EPU)	2014	2015
Steam Dryer Replacement (EPU)	2014	2015
Turbine/Generator Set Upgrade (EPU)	2014	2015
Digital Control Systems (EHC and Feedwater)	2018	2017
Fuel Pool Cooling Heat Exchangers	2017	2017
ISFSI Pad Expansion	2020	

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GALL-SLR Consistency and Commitments

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SLR Application Development

- Scoping and Screening
  - ✓ Updated for plant modifications
  - ✓ Updated to NEI 17-01 guidance
- Aging Management Reviews
  - ✓ PB FLR was pre-GALL, additional aging effects required assessment based on NUREG-2191 GALL-SLR
- Aging Management Programs (AMPs)
  - ✓ Total of 47 AMPs per GALL-SLR guidance
- Time-Limited Aging Analyses (TLAAs)
  - ✓ Existing TLAAs re-assessed
  - ✓ New TLAAs for SLR due to component repair/replacement
    - ✓ Jet Pump repair components for Loss of Preload
    - ✓ Replacement Steam Dryer Stress Report and Fatigue Evaluations
    - ✓ Replacement Core Plate Plugs for Stress Relaxation Analysis
    - ✓ U/3 Core Spray Replacement Piping for Fatigue and Loss of Preload
  - ✓ Total of 35 TLAA analyses per GALL-SLR guidance

GALL Consistency

- Submittal based on GALL-SLR
- High AMR consistency (98.6% Notes A thru E)
- 50 License Renewal Commitments
  - ✓ 47 Aging Management Programs
  - ✓ 3 Additional Commitments
    - ✓ OPEX Review, EPU OPEX Review, FERC Inspection of Conowingo Dam
  - ✓ UFSAR Supplement (Appendix A of the SLRA)
  - ✓ Managed by Exelon Commitment Tracking program based on NEI 99-04, “Guidelines for Managing NRC Commitment Changes”

		AMPs Consistent with GALL	AMPs Consistent with Enhancement	AMPs with Exception without Enhancement	AMPs with Exception and Enhancement	Plant Specific AMPs
Existing	**36**	8	19	2	6	1
New	**11**	8	0	3	0	0
Total AMPs	**47**					

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FLR Aging Management Effectiveness Reviews

- Program effectiveness reviews included:
  - ✓ Detailed review of inspection schedules, results, and data
  - ✓ Review of relevant operating experience within the Corrective Action Program

- All first LR Programs were effectively implemented

- Summary of each review is found in Element 10, “Operating Experience” of each AMP and in the SLRA in Appendix B

- In November 2018, the NRC staff conducted a 71003 Phase 4 inspection at PBAPS, to assess aging management program effectiveness, and identified no issues

Confirmatory Item

- **Confirmatory Item**
  - CI 3.0.3.2.3-1: BWR Vessel Internals Program
    - NRC Staff review of Enhancement 1 identified that additional information was required for core plate rim holddown bolts
    - A revision to Enhancement 1 was made to include the guidance of BWRVIP-25, Revision 1
    - Response to this Confirmatory Item was submitted to the NRC Staff in a supplement October 9, 2019
    - Closed by NRC Staff in the Updated SER dated November 19, 2019

Technical Topics

RPV Embrittlement

IASCC of Reactor Vessel Internals

Peach Bottom will manage aging consistent with recommendations in GALL-SLR

Concrete and Containment Degradation

Electrical Cable EQ and Condition Assessment

Peach Bottom Atomic Power Station, Units 2 and 3 Subsequent License Renewal Application

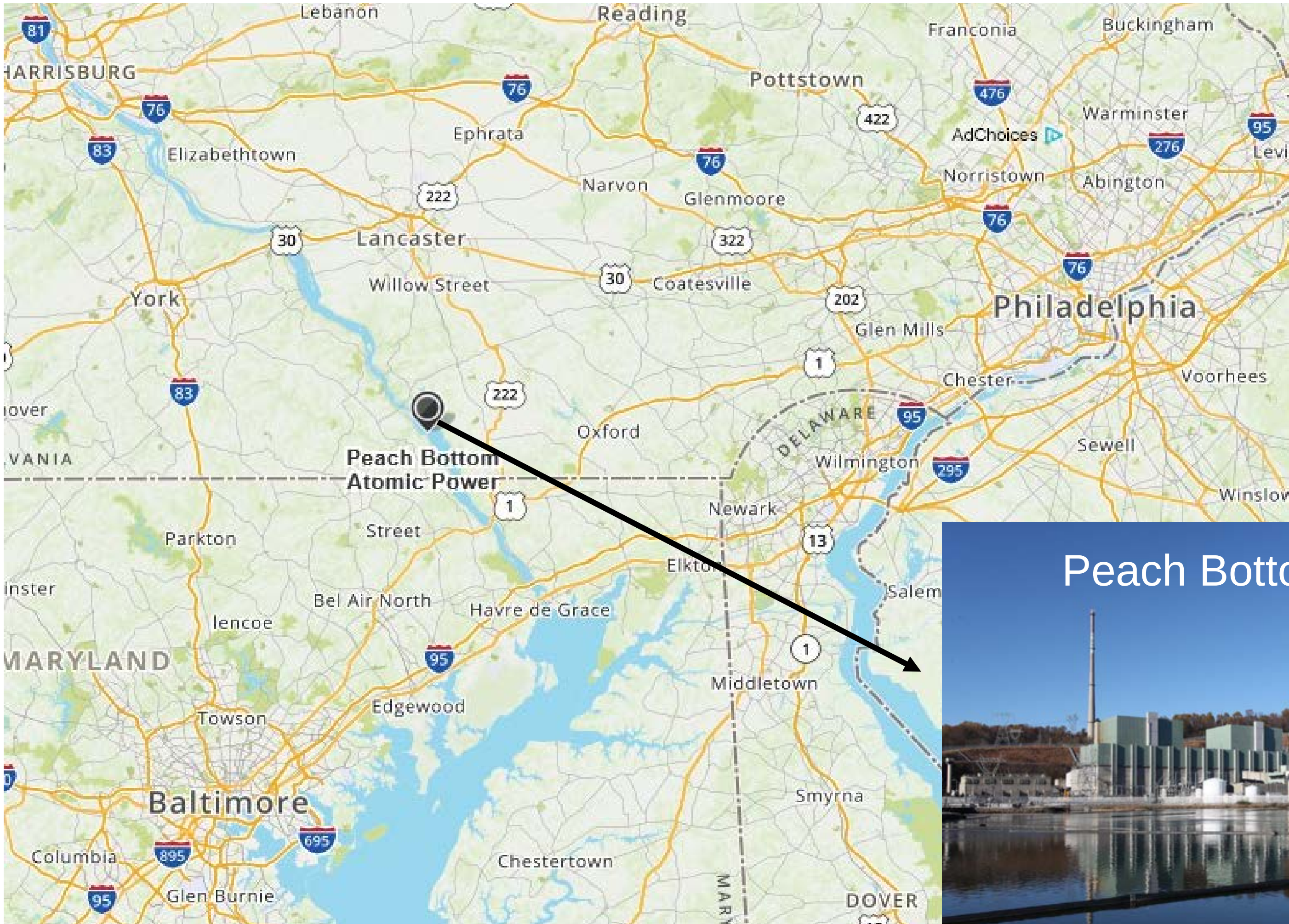
ACRS Full Committee Presentation
December 4, 2019

Peach Bottom Atomic Power Station,
Units 2 and 3
Subsequent License Renewal Application

ACRS Full Committee Presentation
December 4, 2019
Back-up Slides

Exelon

Peach Bottom Station Location



GALL Consistency - AMP Exceptions

Program	Exception	Justification
Water Chemistry	Using this AMP to manage Auxiliary Boiler water chemistry.	Scope addition, while not part of BWRVIP-190, standards exist for monitoring water parameter (ISBN-0-7918-1204-9).
Bolting Integrity	Using this AMP to manage submerged mechanical bolting on intake structure traveling screens.	Scope addition, while this AMP is used to manage closure bolting for pressure retaining components, inspection requirements will be adequate to manage loss of preload.
Closed Treated Water	NUREG-2191 recommends EPRI document “Closed Cooling Water Chemistry Guideline” Rev. 1. Peach Bottom uses Rev.2 of this guideline.	Revised guideline incorporates latest industry OPEX. No changes to monitoring criteria.
Reactor Head Closure Stud Bolting	NUREG-2191 requires the use of material with ultimate tensile strength of less than 170 ksi for in-service studs. Both units have studs installed with studs over 170 ksi.	Test reports show some test values over limit. Studs are inspected for cracking.
	NUREG-2191 requires the use of material with yield strength of less than 150 ksi for replacement studs. Replacement stud has test results over 150 ksi.	Test reports show some test values over limit. Stud was inspected for cracking and will be re-inspected if utilized.
BWR Vessel Internals	Steam Dryer will not be inspected per BWRVIP-139-A	BWRVIP-139-A is for GE designed steam dryer assemblies. PB has installed Westinghouse steam dryers and has submitted an inspection plan to the NRC.

GALL Consistency - AMP Exceptions

Program	Exception	Justification
Fire Water System	NUREG-2191 requires foam system discharge test annually to confirm spray patterns. When not possible, visual inspection of nozzles and air testing is performed.	Single nozzle which sprays across down the inside of the tank. Nozzle has a vapor seal. One time visual inspection to assure proper orientation as it is within the fuel tank.
Internal Coatings	NUREG-2191 requires an internal inspection of portions of concrete lined pipe. Opportunistic inspections will be performed.	Fire header piping is buried. Various periodic flow tests will assure coating has not degraded impacting performance. 2014 inspections found concrete lining in good condition. When made available, visual inspection will be performed.
	NUREG-2191 requires coating found not meeting acceptance criteria are repaired, replaced, or removed. HPCI lube oil reservoir coating will not be repaired.	NMAC's Terry Turbine User's Group provides recommendations that degraded coatings not be replaced. Only remove portions that show poor adhesion.
ASME Section XI-IWE	NUREG-2191 requires pressure retaining components subject to cyclic loading that have no fatigue analysis are inspected for cracking. Peach Bottom will only inspect high temperature mechanical penetrations.	Peach Bottom, had it been constructed to a later code, would have met requirements of ASME Code for fatigue waivers for low temperature penetrations. High temperature penetration accessible surfaces will be inspected for cracking.
	Program will manage flow blockage due to fouling for the Core Spray System, High Pressure Coolant Injection System, Reactor Core Isolation Cooling System, and Residual Heat Removal System pump suction strainers.	No existing GALL line items exist for the management of flow blockage due to fouling for these components and as a result the IWE Program was selected because the station Containment ISI program plan and procedures will perform the required aging management actions.

GALL Consistency - AMP Exceptions

Program	Exception	Justification
E3A - Medium Voltage Cables E3B - I&C Cables E3C - Low Voltage Cables	NUREG-2191 recommends, inspections for water accumulation and manhole condition annually. Additionally, inspections for water accumulation are also to be performed after event driven occurrences, such as heavy rain. Manholes with level monitoring and alarms that result in consistent, subsequent pump out of accumulated water prior to wetting or submergence of cables will be inspected at least once every five years with additional inspections following event driven occurrences, such as heavy rain, rapid thawing of ice and snow, or flooding, when level monitoring indicates water is accumulating.	Level monitoring instrumentation, with alarms monitored by Operations Personnel, provide for detection of water level on an on-going basis. Corrective actions are taken when an alarm is received which includes manual pumping of the manhole as needed. In cases where it can be determined that cables have not been subjected to significant moisture, manhole inspections will be performed on a five-year frequency when structural inspections are performed. Following event driven occurrences, inspections and subsequent pump outs, as needed, will be performed when level instrumentation has detected increasing water levels.

RPV Embrittlement

	SLRA Sections Addressing GALL-SLR Recommendations
Reactor pressure vessel neutron embrittlement at high fluence	3.1.2.2.3 Loss of Fracture Toughness Due to Neutron Irradiation Embrittlement 3.1.2.2.13 Loss of Fracture Toughness due to Neutron Irradiation or Thermal Aging Embrittlement 4.2 Reactor Vessel and Internals Neutron Embrittlement Analyses A.2.1.20 Reactor Vessel Material Surveillance A.3.1.2 Neutron Fluence Monitoring

- Fluence projections through SPEO (70 EFPY) were performed for neutron embrittlement analyses
- Analysis for USE, ART, Axial/Circ Weld Failure Probability, and Reflood Thermal Shock for beltline materials have been satisfactorily evaluated using the 70 EFPY fluence projections
- PBAPS will manage fluence projections consistent with GALL-SLR AMP X.M2, Neutron Fluence Monitoring Program
- PBAPS will manage embrittlement consistent with GALL-SLR AMP XI.M31, Reactor Vessel Material Surveillance Program.
  - ✓ One capsule will be withdrawn from each unit during SPEO at 60-62 EFPY

IASCC of Reactor Vessel Internals (RVI)

	SLRA Sections Addressing GALL-SLR Recommendations
IASCC of reactor internals and primary system components	3.1.2.2.12 Cracking Due to Irradiation-Assisted Stress Corrosion Cracking 4.2.1.2 Reactor Vessel Internals Neutron Fluence Analyses 4.2.14 First License Renewal Application Core Shroud IASCC and Embrittlement Analysis A.2.1.7 BWR Vessel Internals A.3.1.2 Neutron Fluence Monitoring

- IASCC is addressed in accordance with BWRVIP guidelines through:
  - ✓ periodic inspection using techniques capable of detecting cracking due to SCC
  - ✓ flaw tolerance guidance that considers the effect of neutron fluence on material properties and SCC growth rates.
- BWRVIP guidelines are adequate for use to determine the proper re-inspection interval and are not time dependent, rather are based on neutron fluence values.
- PBAPS Rx vessel internals have been assessed using governing BWRVIP inspection guidelines and existing program requirements were found acceptable
- PBAPS will manage RVI components and welds that are susceptible to IASCC consistent with GALL-SLR AMP XI.M9

Concrete and Containment Degradation

	SLRA Sections Addressing GALL-SLR Recommendations
Concrete and containment degradation	3.5.2.2.1 Pressurized Water Reactor and Boiling Water Reactor Containments 3.5.2.2.2 Safety-Related and Other Structures and Component Supports 4.6 Primary Containment Fatigue Analyses A.2.1.30 ASME Section XI, Subsection IWE A.2.1.32 10 CFR Part 50, Appendix J A.2.1.34 Structures Monitoring A.2.1.35 Inspection of Water-Control Structures Associated with Nuclear Power Plants

- Concrete overall is in good condition
  - ✓ No effects of ASR have been identified for PBAPS concrete structures
  - ✓ PBAPS will manage concrete structures consistent with GALL-SLR AMPs XI.S6, “Structures Monitoring” and XI.S7, “Inspection of Water-Control Structures Associated with Nuclear Power Plants”
- The Peach Bottom Mark I steel containments are in good condition
  - ✓ The Sand Pocket Region has been observed to be free of water leakage, each refueling outage
  - ✓ Reactor Vessel Shield Wall gamma and neutron irradiation remains within conservative radiation exposure levels, through SPEO, consistent with GALL-SLR
  - ✓ PBAPS will manage each containment consistent with GALL-SLR AMPs XI.S1, “ASME Section XI, Subsection IWE” and XI.S4, “10CFR 50, Appendix J”

Electrical Cable EQ and Condition Assessment

	SLRA Sections Addressing GALL-SLR Recommendations
Electrical cable qualification and condition assessment	3.6.2.2.1/4.4.1 Environmental Qualification of Electric Equipment A.2.1.37 through 41 Cable and Connection Insulation Programs A.3.1.3 Environmental Qualification of Electric Equipment

- Environmental Qualification of Electrical Equipment
  - ✓ EQ cable analyses have been updated for 80 years of operation
  - ✓ EQ cables have been evaluated to have a qualified life > 80 years
  - ✓ Cable analysis and EQ program are consistent with GALL-SLR

- Electrical cable condition assessment
  - ✓ Added new or enhanced programs to be consistent with GALL-SLR
    - o E1 Accessible Non-EQ Cables and Connections (enhanced)
    - o E2 Non-EQ Instrument Cables and Connections (enhanced)
    - o E3A for Medium Voltage Cables (enhanced)
    - o E3B for Instrument & Control Cables (new)
    - o E3C for Low Voltage Cables (new)