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(202) 234-4433

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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708TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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WEDNESDAY

SEPTEMBER 6, 2023

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The Advisory Committee met via teleconference at 8:30 a.m., Joy L. Rempe, Chairman, presiding.

COMMITTEE MEMBERS:

JOY L. REMPE, Chairman

WALTER L. KIRCHNER, Vice Chairman

DAVID A. PETTI, Member-at-Large

RONALD G. BALLINGER, Member

VICKI M. BIER, Member

CHARLES H. BROWN, JR., Member

VESNA B. DIMITRIJEVIC, Member

GREGORY H. HALNON, Member

JOSE A. MARCH-LEUBA, Member

ROBERT MARTIN, Member

THOMAS ROBERTS, Member

MATTHEW W. SUNSERI, Member

ACRS CONSULTANTS:

DENNIS BLEY

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

WEIDONG WANG

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

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

8:30 a.m.

CHAIRMAN REMPE: This is the first day of the 708th meeting of the Advisory Committee on Reactor Safeguards.

I'm Joy Rempe, Chairman of the ACRS. Other members in attendance are Ron Ballinger, Vicki Bier, Charles Brown, Vesna Dimitrijevic, Greg Halnon, Walt Kirchner, Jose March-Leuba, Robert Martin, Dave Petti, Thomas Roberts, and Matt Sunseri.

I note we do have a quorum. Today, the Committee is meeting in-person and virtually.

The ACRS was established by the Atomic Energy Act and is governed by the Federal Advisory Committee Act. The ACRS section of the U.S. NRC public website provides information about the history of this Committee and documents, such as our charter, bylaws, Federal Register Notices for meetings, letter reports, and transcripts of all full and subcommittee meetings, including all slides presented at the meetings.

The Committee provides it advice on safety matters to the Commission through its publically available letter reports.

The Federal Register Notice announcing

this meeting was published on August 17, 2023. This announce provided a meeting agenda as well as instructions for interested parties to submit written documents or request opportunities to address the Committee.

The Designated Federal Officer for today's meeting is Mr. Weidong Wang.

A communications channel has been opened to allow members of the public to monitor the open portions of the meeting. The ACRS invites members of the public to use the MS Teams link to view slides and other discussion materials during these open sessions. The MS Teams link information was placed in the Federal Register Notice and agenda on the ACRS public website.

We've been notified that representatives of Integrated Nuclear Solutions, LLC, and NEI have requested to make oral statements during one of today's sessions.

Periodically, the meeting will also be opened to accepted comments from participants listening to our meetings. Written comments may be forwarded to Mr. Weidong Wang, today's CFO, or Designated Federal Official -- Officer.

During today's meeting, the Committee will

consider the following topics. Revision of Reg Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents. St. Lucie Subsequent License Renewal Application.

Although the agenda states that the topic, Pressurized Water Reactors Owners Group Topical Report on Hydrogen-Based Transient Clad Strain Limit will be discussed at 4:00 p.m., ACRS will continue with report writing at this time.

At the end of the August 24th Subcommittee on the PW Owners Group Topical Report, it was recommended that the Committee should consider whether a letter is needed.

Stakeholders interested in hearing a discussing regarding this topic may listen to the planning procedures portion of our meeting on Friday morning when this topic will be addressed.

Note the portions of the sessions for the ACRS meeting was on the agenda, there is a transcript of the open portions of the meetings is being kept and is requested that speakers identify themselves and speak with sufficient clarity and volume so they can be readily heard.

Additionally, participants should mute themselves when not speaking.

So, at this time, I'd like to ask other members if they have any opening remarks.

Not seeing anyone, I'd then like to ask Dave Petti to lead us in our first topic for today's meeting.

MEMBER PETTI: Okay -- yes, I've unmuted myself so hopefully everybody can hear.

Today, we're going to talk about Reg Guide 1.183 which, if you recall, we met on this quite a while ago, back in 2022, I think, and maybe even in 2021. It's getting kind of long to remember the history. But, given everything we have, the tight schedule, and our audio difficulties, let's just get into it and start with Mike Franovich to provide some opening remarks.

Mike?

MR. FRANOVICH: Good morning, I see you all have the same lights and tones as we have in the main office over on 100 White Flint. Hopefully, people online can hear me.

I do want to thank Chairman Petti and the Committee and Chairman Rempe to make the time for us today.

I do have some good news for you. We will not be presenting on Part 53. So, you do get a break

from us. But I'd also like to say on behalf of the NRR executive leadership, I want to acknowledge the tremendous flexibility the ACRS has shown to accommodate shifting priorities among NRC rule makings and licensing activities.

Specifically, many of these activities center on rapidly enabling the use of advanced technologies. Strong congressional bipartisan interest, industry, and may other stakeholders have made it abundantly clear that the NRC must meet this challenge. And such circumstances pose signification enterprise risk for our Agency while maintaining safety.

An important step forward to meet the moment is finalizing an important revision to the Agency's regulatory guidance pertinent to advanced fuels, what we'll say in short is Revision 1 or Reg Guide 1.183 today.

The guide and public comment disposition document were made publically available on the 31st of August in support of today's meeting.

As a reminder, the staff had planned on a phased approach to update guidance with a second revision in the works.

The staff has been actively working on a

future, quote, Revision 2 to address fuel burn up levels up to 80 gigawatt tons -- gigawatt days per ton in support of the Commission's high priority increased enrichment rule making and other matters such as guidance on BWR suppression pool scrubbing credit.

Today's discussions will center around the content of the Reg Guide and changes since issuance of Rev 0 in the year 2000.

A nearly quarter-century journey has been enriched by licensing experiences, technical advancements, and modeling severe accident progression, development of updated containment source terms, and use of modern consequence analysis and techniques.

It is a history ripe with strong opinions and strong views on how to best achieve reasonable assurance of safety without creating unnecessary conservatism. I am grateful and very proud of our multidisciplinary team delivering -- diligently delivering a guide that is of very high quality.

It is important to note that this guide provides regulatory certainty to licensees of methods the NRC staff considers acceptable for design basis accident dose consequence analysis using what's called the alternative source term.

During the development and revision alone, we have embraced a regulatory transformation. Transformation is an unending process, not a once in a lifetime experience, although sometimes it may feel that way.

As such, we are a learning organization, adaptively and creatively improving our efficiency and reliability as regulators.

The key theme under riding our work is applying the Commission's direction and use integrated decision making principles.

Those principles include defense in depth safety margins, risk insights, and use of engineering insights as well. But also, have a disciplined adherence to the back fit and forward fit requirements of the Agency.

Rev 1 and the guide which was termed DC-1389 was issued for public comment in August -- in April of 2022. So, it's been a little over a year.

Today, we will go over the comment disposition from the public comments we received, which is approximately 168 public comments.

We'll focus on the more substantive comments.

I'd also like to mention that you will be

hearing from two staff members who have differing views on the potential implications of the decision on a plant-specific licensing action associated with MSIV leakage.

This is a complex subject and it is not unexpected to have differing views.

The EDO, Executive Director for Operations, rendered a decision on the differing professional opinion, the DPO, appeal on August 25th of 2023.

This discussion will occur after the main discussion regarding Reg Guide 1.183 Revision 1.

Of note, the ACRS members with a need to know where provided with a copy of the DPO case file last week following established Agency processes.

The staff are currently conducting appropriate redactions of the case file and expect to make the case file publically available in the near term.

That said, I will turn the presentation over to Mark Blumberg, Senior Reactor Engineer and Irradiation Protection and Consequence Branch in my division, the Division of Risk Assessment in the Office of Nuclear Reactor Regulations.

Mark, I turn it over to you.

MR. BLUMBERG: Thank you very much for that introduction, Mike.

I'd like to tell you the Reg Guide Working Group is pleased to present to you a summary of our work on Reg Guide 1.183 today.

We are grateful for the time we have today to provide additional information for the ACRS review of the draft final Reg Guide.

Joseph Messina and I will lead this presentation and we will call upon other working group members as the need may arise do discuss specific issues.

I'd like to start with the discussion of our agenda. So, please go to slide two.

We'll start the presentation with some key messages for our presentation.

Then, review the purpose of Reg Guide 1.183 and the associated regulatory requirements.

Next, we will discuss the changes to the guidance since the last time the ACRS was asked to review it.

Lastly, we will conclude by providing a summary of our presentation.

We will also provide a look forward regarding our intentions for further revisions to Reg

Guide 1.183.

Please go to slide three.

Before we get into the heart of our presentation, we'd like to acknowledge the 17 working group and steering committee members.

We coordinated the proposed changes across the management and the staff of multiple organizations within the NRC.

Our steering committee is comprised of four division directors from the Office of Nuclear Reactor Regulation and the Office of Nuclear Reactor Research.

Similarly, the working group is comprised of staff from within these two offices and the Office of General Counsel.

This steering committee and the working group provided diversity, coordination, and checks and balances of the regulatory guide revision among subject matter experts within the NRC.

Please go to slide four that introduces our key messages.

Slide four please? And then, to slide five?

We have four key messages for this presentation.

First, DG-1389 was issued as a draft for public comment in April of 2022.

Second, the ACRS and stakeholder involvement on the revision to Reg Guide 1.138 began prior to 2009, when an initial draft was issued for public comments.

Recent development of the guidance involved numerous public workshops and a meeting with the ACRS Subcommittee prior to issuing DG -1389.

The third key message is that the working group, in coordination with the steering committee, had prepared responses to the public comments and updated the Reg Guide based upon these comments.

We obtained internal concurrences, and the OGC, no legal objection of these -- on these documents and is providing the ACRS an opportunity to review the draft final version of the guidance before we publish it.

Lastly, although the disposition of 163 public comments required significant and detailed responses, the changes to DG-1389 in response to these comments were minimal because the DG-1389 guidance continues to be appropriate and defensible.

Please go to slide six. That introduces our discussion on Reg Guide 1.183's purpose and

associated regulatory requirements.

And then, go to slide five.

I'm sorry, this is slide seven.

Reg Guide 1.183 Revision 0 was issued --

MEMBER SUNSERI: Mark, this is ACRS, can we interrupt for a second?

MR. BLUMBERG: Yes, sir.

MEMBER SUNSERI: I have a question.

CHAIRMAN REMPE: Okay, can you hear me, Mark, or do I need to get closer to --

MR. BLUMBERG: Yes, Joy, I can hear you.

CHAIRMAN REMPE: Okay.

I had a question about how the interactions with some of the stakeholders went.

When I looked at some of the PWR Owners Group questions and the responses back about implementation, and in particular, partial implementation when Rev 0 and Rev 1 were coexisting.

I thought when I read the question, and I saw the response back, it wouldn't have answered my question.

Did the staff interact with some of these public meetings saying, hey, this is how I plan to -- or the staff plans to address this question?

And was there a response back from the PWR

Owners Group that says, yes, that takes care of it and we know how to move forward?

MR. BLUMBERG: No, there were no meetings with the public after the public comments were resolved or during the comment resolution.

The meetings that occurred were all before it was issued for public comment.

I will say that during those meetings, this came up several times as well as during the last ACRS meeting. I think Steve Schultz had brought up this issue as well.

And the thing that is difficult for the staff is that this method provides that -- this regulatory guidance provides a method that we find acceptable. It's not the only method.

But we can't -- the difficult thing is that, everyone's licensing basis is different. And as such, it's very difficult for the staff to come up with all the if, then, else situations that might arise.

So, our plain response to this is that each of these methods current provide acceptable methods to the staff.

Deviations from those methods are acceptable, but they need to provide justification as

to why they would be acceptable.

CHAIRMAN REMPE: Okay.

So, we get to implementation and the request for examples, I'll have more questions. Because it just seems like some general guidance or some examples that give people an idea of what to do when one starts to load in a partial core of high burn up or -- excuse me, higher enriched fuel might be useful.

Okay?

Thank you, thank you.

MR. BLUMBERG: Understood. Thank you, Joy.

Okay, are there any more questions before I move on?

Okay, hearing none, I'll go on to slide seven. Reg Guide 1.183 Revision 0 was issued over 20 years ago to support a landmark change that uses a more realistic source term.

This source term was developed from decades of severe accident research and stimulated by the accident at Three Mile Island.

The Reg Guide was created to support the implementation of 10 CFR 50.67, known as the Alternative Source Term Rule.

This rule established design basis dose limits for the control room, exclusion area boundary and low population zone for the purposes of determining the design requirements for mitigating safety systems.

Reg Guide 1.183 provides an acceptable method for modeling the design basis accidents used to demonstrate compliance with 10 CFR 50.67.

At the time it was developed, it was recognized that the source term derived from NUREG-1465 might need to evolve. So, the source term was not specified in 10 CFR 50.67, but put into Regulatory Guide 1.183.

One of the main reasons we are proposing this revision to Reg Guide 1.183 is to support the inclusion of a revised source term needed for increased enrichments, burn ups, and near-term accident tolerant fuel designs.

Please go to slide eight.

Slide eight provides a table summarizing the source terms used in various versions of the regulatory guidance. It is provided to give you an overall view of the various ranges of applicability for the various versions of the regulatory guidance.

Basically, as the guidance evolved, it

evolved to provide guidance for expected burn ups and enrichment ranges.

Please go to slide nine.

Slide nine lists the regulations directly tied to Reg Guide 1.183. They are grouped based on the safety assessments or systems that they relate to.

These include site safety assessments, control room, and technical support habitability.

Although Reg Guide 1.183 was originally created to support compliance with 10 CFR 50.67, and General Design Criteria 19, the proposed regulatory guidance expands its possible use to provide methods that could be used to demonstrate compliance with dose criteria in Part 52.

Please go to slide ten to introduce the next topic, then slide 11.

Slide 11 provides a representative bar graph that breaks down the comments into categories of interest.

These categories are listed with the category with the greatest number of comments first, and those with the least last.

As an example, general and editorial comments were provided in approximately 17 percent of the comments received.

Comments about equipment qualification were about 1 percent of the comments.

This graph is provided to give you an idea where the greatest interest exists based upon public comments received.

Please go to slide 12 to introduce the highlighted changes, then slide 13.

Slide 13 lists the categories of significant changes due to public comments.

We will be going over each of these in more details in the slides that follow.

The categories will be discussed in an order that highlights key areas of interest first and issues of less significance last.

The slides that follow are rather detailed, so we will not have time to go through every detail. Rather, I will summarize them rather than going into those details on each slide.

Please go to slide 14.

In DG-1389, staff did not propose any changes to regulatory positions on suppression pool scrubbing.

The two comments provided were to re-evaluate BWR release fractions based upon new accident sequences including suppression pool scrubbing.

And to allow for this scrubbing based upon a contractor's NUREG report.

And the state of the ARC reactor consequences analysis.

Our responses point out that suppression pool scrubbing guidance is already provided in the guidance and a revision of the LOCA release fractions is beyond the scope of the regulatory guidance revision.

Changes were made to DG-1389 guidance so that it does not discourage credit for suppression pool scrubbing.

We were requested to include an example of when we credited suppression pool scrubbing for the economic simplified boiling water reactor. So, we added a footnote that provides information on this example.

Please go to slide 15.

In DG-1389, staff introduced the term transit dose. It was added to demonstrate compliance with the General Design Criteria 19 requirements.

GDC-19 requires adequate protection to permit operators access and occupancy of the control room under accident conditions.

The footnote was also added providing

guidance for calculating the impact of SHINE dose.

Previously, guidance was silent on the issue of control room access transit dose, except for providing an end leakage value for ingress and egress through the control room operating doors.

Eleven public comments on the control room were received. These comments requested that the added guidance on transit dose and footnote be removed, acknowledgment that a correction factor is already included in the dose codes like RADTRAD, requested allowances for the alternative geometry correction factors, and questioned whether sufficient safety margin exists for the control room dose calculations.

The staff agreed with the request to remove the guidance on transit dose, removed footnote 15, acknowledged the finite geometry correction factor, and addressed the concern regarding safety margin.

Regulator Position 4.2 which provides guidance on calculating control room dose was revised to clarify the guidance.

Please go to slide 16.

DG-1389 revised the alternative source terms and methods for demonstrating compliance with 10

CFR 50.67 which stated that Revision 0 continue to be available for use by licensees and applicants.

Eleven comments requested clarifying statements, examples of selective implementation, and stated that Revision 1 should supersede Revision 0 because of errors.

There was also request for the NRC to evaluate the impact of Rev 1 on a sampling of plants.

We agreed with adding clarifying statements and an example of selective implementation. The background section and Regulatory Position 1.1.1 and 1.1.5 were revised in response to these public comments.

Please go to slide 17.

CHAIRMAN REMPE: Mark, this is Joy again, can you hear me, Mark?

MR. BLUMBERG: Yes, I hear you, Joy.

CHAIRMAN REMPE: Sorry, we were -- this is an unusual situation today with our audio.

But anyway, I -- when I looked at those sections, what I saw as the clarification or example that really go to Rev 1 if you go above a certain enrichment or burn up.

Is that -- am I correctly reading what the clarifying statements an example for?

MR. BLUMBERG: Yes.

My recollection of the statements that were made revolved around the fact that each particular sets of guidance have applicabilities and scope for which they are -- they can be used.

If you're --

CHAIRMAN REMPE: So, then --

MR. BLUMBERG: If you're trying to use one outside of the scope of the other -- I mean, combining the two, it potentially caused problems if you're going outside the scope of one and then you're applying to something that is higher burn up and enrichment for which it was developed.

CHAIRMAN REMPE: So, I had a question. If I wanted to load and mix and half of its core, then would I need to go and use Rev 1 for the source term for the entire core or would I use Rev 1 for the higher enriched fuel and Rev 0 for the lower?

And is that just obvious to everybody?

And then, I was curious why we were doing that in light of the Sandia 2023 report which showed that source term really wasn't that highly affected by enrichment and burn up?

And so, I just was puzzled why that was the only example or clarification made?

Because that didn't seem to answer what I thought the BWR Owners Group was asking.

MR. BLUMBERG: I can give you my response, maybe some others in the group can chime up.

But that was the clearest example in our minds to provide them and probably the most likely example that, you know, in this situation, it might be encountered.

At least from the standpoint that many people want to use this for higher burn up and enrichment, we focused on that example.

Is there anyone else --

CHAIRMAN REMPE: Designed --

MR. BLUMBERG: -- from the working --

CHAIRMAN REMPE: I just want to understand.

Is that -- does everybody in industry know then you should divide the core up and you should have a source term based on where the fuel is with respect to enrichment and burn up, despite the fact that the Sandia report showed that that wasn't a big factor?

I just was curious, but again, the PWR Owners Group says, you know, this is costly. Do we have to do the whole licensing basis over? What exactly are we supposed to do?

And the staff believes that answers the PWR Owners Group question ? And if that's what the staff believes, is my first question, and then, I really hope NEI answers that, too, today.

But anyway, it's clear to the licensees with that clarification what they need to do? And just tell me what they should do?

MR. BLUMBERG: So, it provides a method for meeting the regulations. It doesn't mean that it's the only method.

So, if licensees think that there's some other way of doing this, then they can propose that and provide justification for doing such.

The guidance doesn't prevent -- prohibit alternatives, it just provides a method.

CHAIRMAN REMPE: Anyone else?

(No response.)

CHAIRMAN REMPE: Okay, go ahead. Thank you.

MR. BLUMBERG: Okay, thank you.

Slide 17, please?

MEMBER MARCH-LEUBA: If you're using the microphone and the speakers, did you get that?

MR. BLUMBERG: I'm sorry, I didn't catch that.

MEMBER SUNSERI: You're okay.

MR. BLUMBERG: Okay, thank you.

staff proposed minor wording changes in DG-1389 and added that the reduction of airborne radioactivity by both sprays and gravitational setting that are competing processes should be evaluated on an individual case basis.

Four public comments questioned whether applying the gravitational settling model in NUREG-14 -- or CR 61.89 continues to be applicable when considering the added main steam line deposition models and with an MHA source term?

Our response was that the main steam line models did not consider removal by both sprays and natural deposition simultaneously.

And in NUREG CR 61.89 can be used if adjusted to incorporate the revised MHA source term.

Regulatory Position A-5 was clarified to state the main steam line models are not valid when crediting other aerosol removal mechanisms.

It revised Regulatory Position A-2.2 to state that reductions in NUREG CR 61.89 are not accepted, but that methods, when adjusted for the MHA source term, could be credited on a case by case basis.

Please go to slide 18.

DG-1389 introduced the modifier maximum hypothetical accident, or MHA, to the loss of cooling accident.

This was done to differentiate the accident described in regulations and its associated dose acceptance criteria from the mechanistic evaluation that's used to demonstrate compliance with 10 CFR 50.46 on the ECCS.

We received seven public comments on the MHA.

We were questioned whether the terminology and reduce in new accident asserted that the source term did not represent the worst case accident.

And asked whether specific pipe breaks are assumed in the MHA LOCA evaluation.

We disagreed with these comments and responded to the questions regarding the MHA.

Our overall responses are based upon the fact that the MHA is not a new accident and it's described in regulations.

We also stated that no specific pipe break is assumed for the MHA LOCA evaluation.

No changes were made to the DG-1389 language on the MHA due to these comments.

Please go to slide 19.

In DG-1389, we added three main steam line deposition models and new guidance for crediting alternative drain pathways to the condenser.

We also clarified what parts of the release path can be credited.

Thirteen public comments were received on the MSIV leakage pathway.

These comments are on the use of non-safety related equipment, source term scaling factors, in board deposition credit, clarifying the acceptability of proposed deposition models, the assumed sizes of containment aerosols, and the removal mechanisms for the re-evaluated steam line deposition models.

We agreed with most of the public comments, but disagreed with revising containment aerosol sizes and to limiting credit to only safety related components in alternative leakage pathway.

We added clarifications to Reg Position A-5 regarding scaling factors and removed the need for case by case evaluations when using the revised steam line deposition models.

Please go to slide 20.

In DG-1389, staff expanded Reg Guide

1.183's scope to use new light water reactor -- to include new light water reactor applications.

Five public comments related to the meaning of advanced LWRs and the applicability of the guidance to new LWR applications were received.

The staff responded to public comments by clarifying that the guidance is applicable to any new LWR application under Parts 50 or 52.

And by listing all relevant regulatory requirements.

Changes were made to the reasons for the revision, background, and regulatory position sections of the guidance.

Now, I'll turn the presentation over to Joseph Messina for the next few slides.

MR. MESSINA: Thank you, Mark.

MEMBER MARCH-LEUBA: Wait, wait, let's check the microphone.

MR. MESSINA: Hello? Okay, good morning, my name is --

(Audio interference.)

MEMBER MARCH-LEUBA: Pull the speakers while he's speaking.

MR. MESSINA: How about now?

Good morning, my name is Joe Messina. I'm

a Reactor Systems Engineer in the Methods and Fuel Analysis Branch of the Division of Safety Systems. When Paul Clifford left the Agency, he transferred the lead of the non-LOCA gap fractions portion of this Reg Guide to me.

So, I'll be going over a couple of slides. In Tables 3 and 4 of the Reg Guide, we provide non-LOCA gap release fractions for BWRs and PWRs respectively to go along -- and to go along with these tables, we provided Figure 1 which specifies a power history applicability envelope that should be met in order to use the gap fractions in the tables.

So, operation, in terms of LHGR versus burn up should remain below that envelope which I'll show in the slide.

I'll note that the non-LOCA gap fractions are highly sensitive to fuel design and power history.

Now, we understand that these power envelopes may not work for maybe a 100 percent of the fleet, so we provided a detailed analytical procedure on how gap fractions can be calculated independently by industry with different power histories or different fuel designs.

This is especially important as we get to more of these ATF designs which are proprietary and we

can't really provide generic gap fractions for those.

We received a few comments on the non-LOCA gap fraction portions and Figure 1.

Most of the comments, we simply added clarification to the Reg Guide such as that, uncertainty should be calculated -- accounted for when comparing to the curve.

But the biggest change, which I would still say is not too significant, was the treatment of partial length rods.

We were asked about the applicability of the figure to partial length rods. Since we gave the figure in terms of rod average power, partial length rods at the bottom of a BWR likely would not be able to meet the curve, even if at the same peak power since they won't have the regions of lower power that a full length rod would have.

So, we added the peak power curve to the Figure 1. And that peak power curve can be met for the partial length rods, then the gap fractions in the tables can be used.

And I will note that we still state that, for partial length rods, they should use the -- assume that it's a full length rod for inventory purposes since the design changes would not necessarily be a

100 percent accounted for.

But that -- assuming it's a full rod should be conservative.

And then, we also received a few comments about non-LOCA and non-control rod drop and control rod ejection accident, fragmentation induced transit fission gas release.

But ultimately, we maintained our position, just expanding on it, adding clarification.

I won't talk much about that for the sake of time, but if there are questions on that paragraph of the Reg Guide, I can speak more to that.

Next slide, please?

So, here's the slide depicting the applicability limits for non-LOCA gap fractions for Rev 0, which is the gray box in -- on the left side, on the left curve.

And DG-1389 and then, the new curves in Rev 1.

Now, I will say that the curves in Rev 1 are the same and the peak power curves were applied based on the statement boxed in red at the top of the curve. But we added that for clarify for the partial length rods.

And you'll see that the Rev 0 limits, they

only had -- Rev only had limits from 54 to 62 gigawatt days, which is not really justifiable because anyone can operate at any LHGR before that and, as long as they're inside that gray box, they can use the gap fractions.

And we know that gap fractions would not be equal with different -- drastically different power histories.

MEMBER PETTI: So, Joe, I have a question.

MR. MESSINA: Yes?

MEMBER PETTI: This shows --

MEMBER MARCH-LEUBA: Wait, wait. We only need one microphone.

(Off-microphone comments.)

MEMBER PETTI: But the Sandia 2023 report shows no benefit.

I interpreted that, the Sandia report of 2023 LOCA but in the non-LOCA because of they were not --

MR. MESSINA: Correct, yes.

So, the Sandia report only deals with MHA LOCA. Non-LOCA should be dealt with -- seen as separate.

MEMBER PETTI: Okay, thanks.

MR. MESSINA: Okay. Any more questions?

And if not, I'll hand it back over to Mark.

MR. BLUMBERG: Thank you.

MR. MESSINA: Next slide, please.

MR. BLUMBERG: Thank you, Joey. In DG-1389, staff removed the environmental qualification guidance contained in Reg Guide 1.183 Revision 0.

This guidance was --

MEMBER SUNSERI: Hold on a second, we've got -- we need to get Joe on the audio here, just stand by.

(Pause.)

MR. BLUMBERG: Are we ready?

MEMBER SUNSERI: Yes, we're good now. Thank you.

MR. BLUMBERG: Okay. Would you like me to start over?

MEMBER SUNSERI: Yes, please.

MR. BLUMBERG: Yes, sure, great.

In DG-1389, staff removed the environmental qualification guidance contained in Reg Guide 1.183 Revision 0.

This guidance was provided in Appendix I of DG-1389 entitled, Assumptions for Evaluating Radiation Doses for Equipment Qualification.

This change was intended to put the EQ

guidance where it belonged, back into Reg Guide 1.89 on equipment qualification.

staff received two public comments related to the continued use of EQ guidance in Reg Guide 1.183 Revision 0, including questions regarding the continued use of the TID-14833 source term for EQ, as was specified in Revision 0 of Reg Guide 1.183.

In our response, we stated that EQ guidance in Revision 0, including the use of TID-14833 guidance can continue to be used, provided that the plant design and licensing basis is in accordance with the applicability and limitations of Reg Guide 1.183 Revision 0.

We've provided language on the applicability and limitations of Rev 0, the Rev 0 source term, and stated That if it was continued to be used, that it needed to be technically justified.

The background section was updated to clarify the continued applicability of Rev 0, for Reg Guide 1.183 consistent with our response to the public comments.

And please go to slide 24.

In DG-1389, staff provided a revised fuel handling accident model, allowing for the modeling of retention and re-evolution of iodine from the spent

fuel pool.

This model replaced the non-mechanistic releases previously assumed for the fuel handling accident and allowed for hold up of radioactivity within the spent fuel pool.

Twelve public comments were received by the working group.

They included requesting clarifications of the iodine species assumed, revisions to the water depths in the model, a method for calculating doses to workers around the spent fuel pool, and requests for information to confirm the applicability of the release fractions in Tables 3 and 4 to the fuel handling accident.

The requested clarifications were provided and information on the limitations of data.

We also clarified that the method for calculating control room dose and off site doses and not for those working near the spent fuel pool.

Please go to slide 25. DG-1389 added guidance that states that a modified version of the control room methodology in Reg Guide 1.194 may be used to estimate the off site atmospheric dispersion factors out to distances of 1,200 meters and to align the most unfavorable dispersion coincident with the

most adverse releases.

Three public comments on meteorology were provided.

These comments requested the basis for the modified control room methodology and requested us to state the codes to be used for atmospheric dispersion, acceptability of the prior guidance, and questioned the need to align the most unfavorable dispersion coincident with the most adverse release.

We stated that we agreed that a basis needed to be provided but when we transmitted DG-1389 to the ACRS, the responses to public comments -- and the responses to public comments to the ACRS, Reg Guide 1.249 had not been issued.

It was recently issued so we plan on revising the Regulatory Position 5.3 to include Reg Guide 1.249 and update our responses to public comments to reflect that update.

We also pointed out that codes for atmospheric dispersion --

MEMBER PETTI: Just a question.

MR. BLUMBERG: Yes?

MEMBER PETTI: So, before you go with Rev 1, you'll make those changes?

MR. BLUMBERG: That's correct.

It's a very simple change. It's just putting that reference in that statement.

MEMBER PETTI: Thank you.

MR. BLUMBERG: You're welcome.

We also pointed out that codes for atmospheric dispersion are contained in references for Reg Guide 1.183 and met the new guidance on aligning -- excuse me for a second, let me get a drink of water.

We also pointed out that the codes for atmospheric dispersion are contained in references for Reg Guide 1.183 and that the new guidance on aligning atmospheric dispersion factors with releases was to align the guidance with that currently in Reg Guide 1.194.

Please go to slide 26 to introduce our conclusions and a look forward on future changes to Reg Guide 1.183. And then, go to slide 27.

Now, we will discuss our conclusions for this presentation.

The NRC has developed updated design basis dose analysis guidance based upon experience since issuing Reg Guide 1.183, Rev 0, research data, new analyses, and significant stakeholder involvement.

These changes represent significant

advances for modeling accident tolerant fuel, high burn up, and increased enrichment source terms.

They also include advancements from modeling BWR MSIV leakage, including guidance for crediting hold up and deposition of MSIV leakage within the main steam lines and the condenser for BWRs, non-LOCA source terms, and revised transport and decontamination models for the fuel handling accident design basis analyses.

Lastly, we want to make you aware that we are currently investigating further improvements to accommodate even higher enrichments and burn ups.

This work seeks to create guidance for up to ten weight percent enriched uranium U-235, and burn ups up to 80 gigawatt days for metric ton uranium.

We are looking to expand the scope of the guidance for the near-term accident tolerant fuel designs and potential for future rule making that might affect the dose analyses.

So, we are planning on re-engaging the industry on these potential changes and expect to be back before the ACRS to discuss these issues when we revise Reg Guide 1.183 to Rev 2.

DR. BLEY: Mark?

MR. BLUMBERG: Yes?

DR. BLEY: It's Dennis Bley.

Could you go back to slide 27? There was one thing -- I'm sorry, 25?

MR. BLUMBERG: Twenty-five?

DR. BLEY: Yes, up in your first bullet you talk about the issue raised is aligning the most unfavorable dispersion coincident with the most adverse releases.

The extent to which that assumption is conservative is really site specific and depends on kind of how often a particular site's in an unfavorable dispersion condition.

You didn't -- I don't believe you said how you dealt with that comment down in your responses below.

Can you refresh that one?

MR. BLUMBERG: So, one of the things I believe -- and I don't have that response in front of me, so I'm going by memory.

My recollection was that when we responded to it, we just pointed out that this was to align it with guidance -- other guidance for other dose points that's contained in the Reg Guide 1.194.

And also, it seems to me that in the response to the public comment, there were some

concerns about how to go about doing this.

And I think we responded with a method for doing so. I'd have to double check on that, though.

DR. BLEY: Okay, thank you. And, clearly, it is a conservative step and I'm just -- I was wondering what they were pushing on and I guess I still think it's really dependent on the site, how conservative that assumption is.

But thanks, that's enough for me.

MR. BLUMBERG: You're welcome, thank you.

So, that concludes our presentation. We're now open for questions from the ACRS.

MEMBER PETTI: Yes, I have a couple that I want to make sure I understand.

I heard other members raise some issues related to mixed core guidance, which I'll let Joy, you know, talk about. But the red zone Rev 1 coexistence, it struck me as odd because it's not something that's commonly done. I know it's been done in the past.

Does it allow them, on a case by case basis, to mix and match, if you will, something from Rev 0 and something from Rev 1? Or are there some guardrails that, you know, prevent that? How does that actually work from an implementation perspective?

MR. BLUMBERG: So, each method by themselves provides a method acceptable for meeting the regulation. Combining those methods creates a new method. And when combining, the licensee or applicant would need to provide additional justification for doing so.

MEMBER PETTI: Okay, that helps.

The second is this understanding of this new report Sandia 2023 that has LOCA source terms that show, apparently, higher releases but no explicit burn up of enrichment dependence.

So, I'm sitting here trying to understand that there's information on the street that says there are higher source terms, potentially, than what are in Table 1 and 2 in Rev 1.

And how do I justify that in my head that we know that the data from the Tables 1 and 2 may not be conservative because we know these new results from Sandia?

And I worry about how long it'll take to get that into Rev 2 -- gets into Rev 2 because all these uncertainties out there in the future. And I'm struggling.

I'm sure there's us on this committee, this is information we need when we get to our

discussions.

So, any perspective on that, Mark?

MR. BLUMBERG: So, I think the perspective that I have is based upon timing and the development of the regulatory guidance and the issuance of this revised source term and its findings.

When we started development of DG-1389, the source term that we had that was vetted was the 2011 source term.

The 2023 source term, let's see, I'm sorry, let me make sure I've got the right number on this, one second, yes, the 2023-01313 source term was under development at that time.

And we're currently in 2023. I'm trying to recall about whether or not it has gone through the vetting process. It seems like it has.

Could I have Office of Research to provide some perspective on that?

CHAIRMAN REMPE: So, this is Joy. They did a peer review and the peer review has been completed in a transmittal letter that Kim Webber sent to Mike Franovich. Right?

And so, I don't know NRR Research wants to answer it, but what else is required besides a peer review for vetting?

MR. BLUMBERG: So, what we had done in the past, not necessarily with 2011, but with NUREG-1465, we did issue that for public comment in NUREG-1465. And the public was involved in commenting on that.

We had talked about doing that. Those discussions, we haven't gotten that far with it.

So, as of today, what we have in this is the 2011 source term. That's just the status of where we're at.

MEMBER PETTI: I understand all that, I just, I guess I'm surprised that the train's still moving on the track when there's potentially a body on the track up ahead.

Terrible analogy, sorry. But there's something that could derail Rev 1 and you know it's out there, but it hasn't gone through and got all the I's dotted and the T's crossed, yet, here we keep going.

Help me here, I'm just struggling. I can see Mike has his hand up.

MR. FRANOVICH: I don't know if you can hear me now?

MEMBER MARCH-LEUBA: Okay, yes we can.

MR. FRANOVICH: I'm on the phone version. Can you hear me?

MEMBER MARCH-LEUBA: Yes, now we can.

MR. BLUMBERG: Yes, I can.

MR. FRANOVICH: Okay.

MEMBER MARCH-LEUBA: If you can get close to this microphone. Yes, that's clear.

MR. FRANOVICH: Okay. I think the short answer is here the state of balance keeps shifting. We've been seeing this for the last 25 years.

Some of the severe accident modeling, I don't want to overstate what our colleagues have done in research, but it is a better knowledge of what the timing of certain releases and pressures and the systems, whether they're PWRs or BWRs.

And that is more of the driver of these changes in terms of a containment source code.

This report that came over from research from Ken Roberts and myself was done in a manner to expedite visibility on the work. But we haven't done the piece -- the protocol piece to engage all stakeholders on that. That was envisioned to be done in Revision 2. We don't have a broader viewpoint.

Yes, there was a peer review done, but we don't have broader views on it. And so, that's really the short answer of it. And it's kind of a moving target. I recognize that. But for 25 years, things

have been a moving target in this area.

So, at what point do you make a decision and say, we move forward knowing there are analytical margins available and physical margins in these plants and dealing with it with a more fulsome set of issues in Revision 2?

And I'll talk about that more a little bit later. But that's the short perspective on it.

MEMBER MARCH-LEUBA: Let's test -- say something.

MEMBER BIER: Can people hear me?

MEMBER MARCH-LEUBA: Yes.

MEMBER BIER: Okay, I talk pretty loud. I have a question similar in the way of Dennis's questions. I mean, clearly, it's possible mathematically to get a bad source term in combination with bad dispersion conditions.

And so, as a PRA person, I would think, well, how likely is that combination and how much worse is it than other combinations?

And so, the question that I have is if people want to get away from the level of conservatism of assuming both at their worst, could they do a probabilistic analysis with it and do a more risk informed kind of process, how would that be taken into

account?

MR. BLUMBERG: So, I'll just point out that the source term that was developed was developed using a risk informed process. 1465 looked at the most important sequences to inform that.

So, the process that we do have, does incorporate the use of risk in its development.

MEMBER BIER: It's okay, we can move on. I'm not sure I fully understand the answer, but I'll think about it.

MR. BLUMBERG: Thank you.

MEMBER PETTI: Members, other questions?

MEMBER ROBERTS: Yes. It's Tom Roberts. You probably need to mute the speaker.

MEMBER MARCH-LEUBA: You can use this one. It really affects --

MEMBER ROBERTS: Yeah, we're -- yeah, the mic is on.

MEMBER MARCH-LEUBA: No, no, no. You have to mute yours.

MEMBER ROBERTS: Mute mine? Can you hear me now?

MEMBER MARCH-LEUBA: Yes.

MEMBER ROBERTS: Yeah, Mark, can you hear me now?

MR. BLUMBERG: Yes, Mr. Roberts.

MEMBER ROBERTS: We can't hear him. I can't hear him.

MR. BLUMBERG: Yes, sir. I can.

MEMBER ROBERTS: Yeah. It's Tom Roberts, following up on what Vicki just said. I think one of the areas that I know confuses me, and I think from the public comment the fact is it confuses a lot of people.

Is the role of the Reg Guide 1.183 analysis to show compliance with regulations in 10 CFR 50.67 in a risk informed context that it really is a deterministic analysis that makes some, you know, stylized assumptions to get you through to a result to show you meet regulation.

And it -- the stylized substances aren't necessarily, maybe the word rational is too strong of a word. But, not necessarily, you know, mechanistic.

You combine things like a somewhat arbitrary assumption that really stops at the really end vessel, release phase.

And so, there's just this somewhat, you know, arbitrary stop to the transient that says, okay, that's good enough, that's enough, you know, source term release to containment.

And then, it also has deterministic assumption that containment, well containment will meet its leak rate assumptions regardless of what the reaction and phenomena might show.

And so, you've got what's a mix of very conservative assumptions with some conservative -- some assumptions that may or may not be conservative depending on the sequence you look at.

So, defining the role of this analysis, how it fits into the overall regulatory structure, seems like something that's of value. I noted in the DPR I think you're going to talk about next, there was a recommendation to step back and write down what the purpose of this analysis is.

And then, you know, presumably then, determine what would change in the prescription and what would change in its use depending on what that written down view of what this analysis intended to show would lead to.

And I would like to get your comment on that. And maybe one specific question, is one of the changes in this revision is to lump in the technical and support center dose, the habitability of the TSC into the Reg Guide.

And it wasn't there before. Near as I can

tell, that was standard practice. But, it wasn't in the Reg Guide before. That's now added.

And so, it would seem like the technical support center, its role is emergency management. And so, why a prescription that uses all the deterministic stylized assumptions going in for a technical support center that may or may not, you know, need to function or be designed to function in that environment is an example of where it maybe the, you know, when you write down the reason why the assumptions are made, maybe you start to see maybe some different set of assumptions are needed.

So, for example, maybe there's analysis somewhere, which I couldn't find, that said that a Reg Guide 1.183 analysis would be conservative for a severe accident at the TSC. I don't know, there may be analysis like that.

But, maybe there's some PRA analysis that says the TSC, it doesn't need to be credited in any event more significant in more likely sequences. I don't know. I'd have to guess why it's reasonable to use the Reg Guide 1.183 for the TSC capability.

So, maybe you can comment on the overall need to write down the rationale for this prescription. And then specifically on the TSC.

Thanks.

MR. BLUMBERG: Okay. There's a lot there to unpack, Mr. Roberts. So, I'll try to do my best.

So, first of all, I agree with you that especially in the current environment of risk informing that we need to be clear about the purposes of this guidance.

The -- and, I'm not sure that that's done in this guidance. The two, of course the deterministic analysis came first before we had PRA. And now that we've got PRA, how these things are combined is a topic of many, many discussions.

And, as we'll see later when I put my other hat on in the DPO, there were some concerns with how that was done with some license amendments.

So, I agree with you. I think that in the guidance we need too clearly, as best we can, define that.

With respect to the TSC, I came from a facility that basically because of the 0737 actions, had done analysis consistent with GDC 19 for their TSCs. And not all plants had those particular requirements or analysis in place.

Some plants did. Some plant's don't. It is my understanding if we go to, let's see here, which

slide was it? I think it is the slide on regulatory requirements. One moment, please. It's slide nine, if you could go to that.

Within -- the basis for including it in this guidance, is provided in 10 CFR Part 50, Appendix E. And if Ed Stutzcage is on the line, if you would like to talk about that, please speak up, Ed.

MR. STUTZCAGE: I'm not sure that I'm following you on this one.

MR. BLUMBERG: That's not good.

MR. DICKSON: Hey Mark, this is Elijah Dickson. Do you mind?

MR. BLUMBERG: Sure. Thank you, Elijah.

MR. DICKSON: Yeah. I'll help supplement. You're spot on the problem.

MR. BLUMBERG: Yeah, I can't hear you.

MR. DICKSON: Yeah. This is Elijah Dickson with the staff, a Senior Reliability Risk Analyst with the Division of Risk Assessment.

Mark, you are spot on those TSC requirements. They were a spinoff of Three Mile Island actions, right.

MR. BLUMBERG: Um-hum.

MR. DICKSON: And so, if you go back and you look at NUREG 03-0737, there's a variety of items

there, of course, just like a list of them. But, one of those is to go assess TSC.

And that was put into guidance eventually with I think Rev 0, there was at least a point or two with this NUREG. And then, we got at, you know, a little bit further into this update.

MEMBER ROBERTS: Sure. I looked at that. And 0737 uses the term accident conditions.

MR. DICKSON: Um-hum.

MEMBER ROBERTS: Which is somewhat vague, because you could interpret that to be design basis, or MHA conditions.

MR. DICKSON: Right.

MEMBER ROBERTS: Or, you can interpret it at least severe accident conditions, depending on how you choose to read it.

MR. DICKSON: Um-hum.

MEMBER ROBERTS: So, the, I guess, the case law is that's been largely interpreted to be the design basis or MHA accident where the containment bypass scenarios are basically, you know, rolled out.

MR. DICKSON: Right. The source term itself is this design basis source term that doesn't affect kind of a severe accident type of source term for the purposes of assessing, you know, the safety-

related SSCs for containment, you know, contain the leakage and mitigation, and then, of course, release out to the environment.

And then, like when it comes to like boots on the ground, you go and you assess, you know, where the TSC is and your chi over q meteorological data. And make sure that it's below those criteria, those dose related criteria in 0737.

MEMBER ROBERTS: Yeah. And I would think a severe accident assessment or a level three, level two, PRA would look at the TSC and if it's credited in one of the sequences, then the height ability would be assessed for that scenario not just for the, you know, the stylized assessment per the Reg Guide.

MR. DICKSON: Right. Yeah.

MEMBER ROBERTS: And I would think that, you know, it would be interesting, does designing it through the regulatory guide get it robust enough to be a unique facility for their atmosphere accidents?

And that's the major question, I think.

MR. DICKSON: Right. It's a hard question. Because when you're doing the PRA work, right, you're looking at core damage frequency and LERF-type events, right. Those are the figures of merit that you're assessing for PRAs too.

And that's kind of two steps behind when you're actually assessing these figures of merit of dose to these operators.

MEMBER ROBERTS: Um-hum.

MR. DICKSON: It's -- especially when you're using this deterministic source term as well that already includes core damage, right. You assume that the core melts.

Per the regulation there's a footnote in the regulation that said that you need to assume that there's a melt to assess these safety-related systems. I don't think there's very many operator actions and design basis space from the TSC that would help conditions for a melted core, right, in these design basis accidents in the PRAs, right.

So, you're not supposed to credit operator actions in these design basis analyses. But, in the PRAs you could.

I'm not sure if I've answered your question or not.

MEMBER ROBERTS: Yeah. I think it's -- well, that method is an example of where the, probably good exercise to write down what the purpose analysis using the stylized mix of assumptions are. And then, see if here are gaps.

MR. DICKSON: Understood.

MEMBER ROBERTS: I don't know if there's a gap here, because as you point out, you know, operator action is clearly not credited for these kinds of things. Or, there may be scenarios where it is.

MR. DICKSON: Um-hum.

MEMBER ROBERTS: In which case, presumably the PRA would have looked at that.

MR. DICKSON: Right. Okay. Thank you.

MEMBER ROBERTS: No, it's just, you know, trying to be clear.

MR. DICKSON: Got it.

MEMBER ROBERTS: Okay. Thank you.

MEMBER PETTI: Other questions, Members?

MEMBER MARCH-LEUBA: If you speak loudly, I think I can hear you.

(Off-microphone comments.)

MR. FRANOVICH: There's a whole set of questions here on the role of PRA. I understand the welcome to the world of very deterministic regs.

But, it is risk informed in the terms of part of the source term in the original 1465. It was an outgrowth of NUREG 1150 and the five plants that were studied in the late '80s, early '90s.

We have made measures to try to bring in more engineering insights with risk perspectives. And, one of those issues, really this is probably a good time to talk about it. And we'll talk about it more in the next session.

But, in terms of actual physical features of the plant and their robustness, we're dealing with seismic events and other sources or pathways where those could actually be reduced.

As you recall, we did have a development of a draft interim staff guidance document. It was focused at the time really on matters when licensees were not crediting pathways like the condenser.

Since that time, we have learned a lot more. We were attempting to update those really antiquated experiences from methods in the 1990s to use more of the risk insights in engineering, insights that look at the robustness of these pathways.

And what we've done, is in this particular Reg Guide, there are elements when licensees would like to try to credit such pathways.

There is elements that we lifted out of a reference document. It was a draft document, a technical basis document. We have some of the experts in the room today that could speak to that for cases

where licensees wish to credit it.

For cases where we want to apply insights with a greater level of effort, we will not pursue this ISG. I know the Committee had a question about that.

We will find a more durable place to place such guidance if we use it, in terms of our internal reviews and leveled effort for cases when licensees are not crediting such pathways. Which is a subject that will come up in the next session.

So, I would say we're incrementally moving toward a more risk informed approach. We're not quite there fully.

Revision Two, does provide us opportunities of one key rulemaking, the increased enrichment rulemaking, is affording some optionality in there, depending on how the Commission goes with the rulemaking itself, to re-look at 5067 and what under currents that rule which was created at the time from the 1990s.

So, I'm going to share that perspective with the Committee that it is lightly risk informed right now in terms of guidance. But, we'd like to more -- move more in that direction if possible.

CHAIRMAN REMPE: Mike, while you're up,

could I ask you a question?

MR. FRANOVICH: Absolutely.

CHAIRMAN REMPE: When do you anticipate Rev 2 will be issued, the first draft?

And, secondly, in your opinion, will you have then Rev 0, Rev 1, and Rev 2 available? Or, will you combine those?

MR. FRANOVICH: We have a recommendation out of a DPO panel, I don't want to get too far ahead, to look at potentially sunsetting Rev 0.

State of practices have moved along. We have been dealing with some older methods and looking at some offsets as you know, which has been challenging for both licensees and regulators.

What is the proper way to sunset Revision Zero and also look at, there's a regulatory information summary that establishes positions, it's called 2006-1, I forget the number exactly, but, it's 2006.

That too also needs to be looked at from our regulatory process on how to -- how to either withdraw that risk or modify it.

In regards to the schedule on increased enrichment rulemaking, that is a high priority rulemaking for the Commission. I believe the draft of

the rule is due the end of next year.

There is a Reg basis that should be released here within about a week or two. And so, in tandem with rulemaking, one needs to provide guidance for stakeholders.

And so, that is why we feel a lot of these matters that are being raised and that rulemaking is sort of the central point enforcing function for us to get on with Revision Two. Which is kind of a good thing.

CHAIRMAN REMPE: So, Zero will be eliminated and Rev 1 will also be eliminated. And we'll just have Rev 2 then?

MR. FRANOVICH: Well, I can't say that definitively until we go through a corporate evaluation.

CHAIRMAN REMPE: Okay.

MR. FRANOVICH: There are regulatory processes we must follow to see are we backfitting, are we forward fitting? The reason we're living with the two stages of coexistence of Rev 0 and Rev 1 right now when it goes final, is, it's a forward fit consideration and backfit consideration.

But, I know it's very sophisticated and detailed. And I don't want to spend the Committee's

time on that.

But, there are control measures that we have to go through and evaluate for these processes.

CHAIRMAN REMPE: Thank you.

MR. FRANOVICH: Sure.

DR. SCHULTZ: Mark, this is Steve Schultz.

MR. BLUMBERG: Hi, Steve.

DR. SCHULTZ: Good morning.

MR. BLUMBERG: Good morning.

DR. SCHULTZ: I still have a concern related to your comment that not only could Rev 0 be used for those licensees that currently have implemented it, but then one could move to Revision One.

But, there's the opportunity to mix and match it seems, that could be presented to the staff. And I guess I'm most concerned because I think the staff has enough on their plate with what will become submittals associated with Revision One as well as the development of Revision Two.

So, are you really just providing that middle approach as an option that you don't expect and don't want licensees to choose, and you're going to hold to the implementation of Revision One?

That's the first question. And then,

secondly, with regard to Revision Two, what's the plan for the public interaction associated with the developments in Revision Two?

Have you got a program planned for public meetings associated with that that's ongoing? And if not, when will it start?

MR. BLUMBERG: Okay. Today I'm wearing two hats. I'm going to be talking about these issues and wearing my second hat with respect to the DPO.

So, the decisions that were made with respect too not withdrawing Rev 0, as well as the second question on Rev 2, could I please defer to my management to respond to, please?

DR. SCHULTZ: Okay.

MEMBER MARCH-LEUBA: No, say your name. They can't hear you.

MR. HSUEH: Hi, this is Kevin Hsueh.

MEMBER PETTI: Please use the microphone if too, if you can.

MR. HSUEH: Okay.

MEMBER MARCH-LEUBA: And speak loudly and clearly.

MR. HSUEH: Okay. This is Kevin Hsueh in the Office of (audio interference.)

DR. SCHULTZ: Somebody has their mic open.

MEMBER MARCH-LEUBA: Can you mute him? He's muted now. Okay, yeah.

MR. HSUEH: Yeah. We speak to the second question about on the proper interactions about Rev 2. I think what we plan to do is to add current. Right now, our plan is to add current to Rev 1 on this issue.

Then, we will use the same approaches that we have done in the Rev 1. We have -- for the Rev 1, we have three or four public comment meetings and to interact with the external stakeholders.

And then, also including those, there probably meet and greet, and kind of shore up proposed changes and then to receive feedback from the public, from the external stakeholders.

And so, what we envisioned to do is for the Rev 2, we kind of envision using the same process. And that we'd like to have a lot of this early interactions.

Basically, I know that there's a lot of high interest. There's a great force for spotting issues. So, that's one of the areas where we don't like to kind of engage with the most stakeholders and before we proceed further.

That is kind of our current plan.

DR. SCHULTZ: Thank you, Kevin. Do you have comments on the implementations of licensees that pick and choose between Rev 0 and Rev 1 elements that they might want to promote or propose?

MR. HSUEH: I think that Mark mentioned earlier and clearly, is that basically we understand the situation that the licensee could pick some elements in Rev 0 and some elements in Rev 1.

And I think that the -- Mark mentioned earlier, when it is a combination, it becomes a new measure. And then, work with NRC staff, we need to evaluate those proposals on a case by case basis.

And so, it's just one measure that the Rev Guide just kind of proposes, one measure when there's a combination of those, we anticipate. That's why we plan to do this kind of measure depends on the case specific.

DR. SCHULTZ: It just seems that that's very generous for the applicants that it would be much easier for the staff to implement a review of Rev 1. And stick to the advanced guidance that's provided there.

But, I appreciate your comment. Thank you.

MR. HSUEH: You're welcome.

MEMBER PETTI: Okay. Seeing nobody else wanting to make a comment, let's move onto the differing views.

We're just a few minutes behind schedule. We're doing actually pretty well, I think.

MEMBER MARCH-LEUBA: This is Jose. May I suggest a short going off the record, so I can tell you what's going on with the sound for everybody in the room?

MEMBER PETTI: Okay.

MEMBER MARCH-LEUBA: Yeah. We're off the record.

(Whereupon, the above-entitled matter went off the record at 9:52 a.m. and resumed at 9:53 a.m.)

MEMBER PETTI: Okay. Mark, continue.

MR. BLUMBERG: Okay. So, we've concluded our presentation on the Regulatory Guidance from the staff. And we're now moving onto the Differing Professional Opinion presentation.

So, Mike and I would like to thank you for this opportunity to provide our differing views on this Regulatory Guide.

(Off-microphone comment.)

MR. BLUMBERG: I'm sorry, is there a problem?

MR. MARKLEY: Nope. I'm getting seated. Thanks.

MR. BLUMBERG: Great.

MEMBER MARCH-LEUBA: You have to talk when you are seated here.

MR. MARKLEY: Yeah. I am seated now. Thank you.

MR. BLUMBERG: Great. Once again, we appreciate this opportunity to provide this, these views to you. The remarks we are about to make represent our views and do not represent the positions of the NRC staff or our management.

As we mentioned before in this previous discussion, your first introduction to our DPO was, which is called DPO 2021-001, was provided in our presentation to the ACRS during the staff's presentation on an interim staff guidance document that was referred to as DRA ISG 2021-01.

During that presentation, because the DPO results were not finalized and because the DPO process procedure at that time limited what information could be discussed, we were unable to provide you details regarding that DPO.

Today, the DPO case file has been provided to you for your review. However, we understand that

it's unlike that you, like us, have had much time to digest the information in this extensive case file.

Today, Mike and I will try to provide a short view of our concerns that we had with the publishing of Reg Guide 1.183, Rev 1 in its current form.

We will discuss a summary of the report, the DPO report and its recommendations, examples of specific issues, observations and conclusions identified in the EDO's Appeal Panel Report, and as these issues pertain to impacting nuclear safety.

We will provide key points for revising Reg Guide 1.183. And, lastly, we will provide a pertinent lessons learned from this DPO that we feel impact safety. Could you please go to slide two?

This slide summarizes the directed actions from the Executive Director for Operations, some selected recommendations from the DPO Panel Analysis Report, and where we stand with those recommendations and actions.

On August 25, 2023, the EDO issued a memo and a DPO Panel Report. The memo directed actions were to one, take actions to ensure compliance with 10 CFR 56(c)(7) for the subject plan and resolve the licensing basis clarity issues for the license

amendment, including the impact of MSIV packing leakage, the basis for the limiting break location, and the aerosol deposition credit for the main condenser.

And secondly, and as it pertains to this Reg Guide, develop an implementation plan for recommendations in the DPO Appeal Panel Analysis Report. Recommendations in that report stated in part, in light of the issues identified in this report, in the near term, revise and consolidate the staff's updated guidance, and it pointed to the guidance, which was DG-1389 and the ISG.

The DPO Panel Report believes any update to Reg Guide 1.183 should be consolidated into a single revision of the regulatory guidance and not include a companion interim staff guidance.

And lastly, this is the subject we brought up in the previous presentation, enhance focus on the overall intent of regulations related to DBA analysis. For example, focus on assessing the acceptability of engineering safety features rather than over reliance on non-safety related features such as deposition in the power conversation systems.

The report also provides specific issues, observations and conclusions that should be addressed.

The NRC staff is evaluating the Report recommendations and possible revisions to Rev 1.

But, despite the above direction and recommendations, no changes have been made to the version of Reg Guide 1.183 that you have for your review. Please go to slide three.

Slide three provides a very limited set of examples of specific issues, observations, and conclusions that impact nuclear safety. And are directly tied to the guidance that needs to be in Reg Guide 1.183 to ensure safety is maintained.

The Appeal Panel Report states that the staff should revise language in the Guidance relative to the MSIV leakage pathway so that the limiting pathway to the environment would be considered. Guidance in Reg Guide 1.183, Rev 1, does not specify this pathway.

Regulatory Position A-5.5 would allow licensees who have prior staff approval for crediting deposition and steam lines that could be bypassed by stem leakage. To continue, it allows these licensees to continue to credit that piping.

Ignoring this pathway significantly underestimates the doses calculated and does not perform an adequate test of the needed safety systems

to protect the operators who are needed to protect the health and safety of the public.

The second issue has to do with staff assuming conservativisms that are not reflected in the licensing basis.

The guidance needs to be more comprehensive to prevent this from happening in the future and to align with the guidance in 10 CFR -- with the rule in 10 CFR 5067 that requires the staff to issue the amendment only if the applicant's analysis demonstrate compliance with the dose limits in the regulations.

From a safety perspective, this is important, since conservativisms not credited in the licensing basis can be modified without the recognition that they are important to safety.

The third issue is that the removal coefficients for aerosol settling are non-conservative. While the staff attempted to resolve this issue in the guidance, the perspective was that what was done for the license amendments, that is the subject of this DPO, is appropriate.

The model in the Reg Guidance states specifically that they are only applicable when sprays are not credited. It needs to be more specific and

state that both should not be credited at the same time.

Updating this guidance would resolve the DPO Panel Report finding that stated that the NRC had not acknowledged these non-conservatisms and revised guidance.

Revision 1 should state that Rev 0 guidance that was previously issued and enabled the use of this non-conservative removal coefficient should be withdrawn.

The fourth bullet pertains to the limited location for the break. It points out that the recirculation line break does not represent a bounding condition.

The working group removed language to address this issue in a previous draft revision that should be restored.

Lastly, the DPO Panel identified issues with the method for determining what non-safety related pathways to the condenser should be credited.

The DPO Panel stated that they disagreed with the inherent seismic robustness of the pathway. That it provides high confidence that a pathway to the condenser will be available in and of itself.

Past seismic walk downs related to this

type of application for some licensees have identified the need for physical modifications to ensure the pathway is not failed by seismic related failure mechanisms such as system piping interactions and anchor issues. This finding needs to be added to the guidance to reflect this operating experience.

And now, I'd like to turn the presentation over to Mike Markley.

MR. MARKLEY: So, the key points as far as revising Reg Guide 1.183, these issues are sufficiently technical that it could have warranted a subcommittee meeting. It's very complex, the details and the issues, the mappings, source term tables just like they're talking about here today.

And it would have been, I think, much better had we gone that route. But, they're proposing a final now. And they're -- you know, Revision 1 to Reg Guide 1.183 should incorporate the issues in the DPO Appeal Panel and other things that need to be fixed.

I mean, fix things that need fixing now. Because what's going to happen is, is you're going to have a whole population of plants propagating these same errors again, or errors that weren't fixed.

But, once you have that, then essentially

where these DPOs spawn from was essentially that plants use that ISV or the context of that ISV and submit its submittals that basically they don't meet the systems six and seven.

And, you know, the canned rating of conservatism that's not docketed, or it's not part of the licensee's case is not really where we want to be fundamentally.

So, you know, for that, I think, you know, the DPO needs to be in this revision. Plus, once you have a population of plants that have adopted a particular revision, they're under no obligation to adopt the next revision just because you issue it. That's their correct licensing basis.

We have a lot of plants right now that are using Reg Guide 1.174, Rev 2 rather than Rev 3, because that's their licensing basis they had. That tested five of five approved that way. They have the NOP-05 that way. They haven't figured the PRAs sufficiently to want to take a next step.

And so, that's where we would end up, with a large population of plants replicating some of the same errors. And any future revision could be years away.

And, I mean, look how long it took to get

to Rev 1. These things don't happen quickly. And so, I expect it would be a long, long time before, you know, Rev 2 would become available.

And what you heard today was a normal process. And, once you have an expedited process, the DPO lessons learned and implementation plan won't be incorporated if it goes that route. As will your concern about the Sandia questions that you asked today.

So, you know, to me you should fix what you fix today. I don't think we should put enterprise risk over radiological risk. This is about dose of the control operators.

To the point of one of your questions earlier today, although it's not part of this presentation, we're reviewing licensees' proposals right now to move the TSC away from the control room to other parts of the plant.

They have the same dose criteria for the TSC. They have to have the same filtration systems and so forth. So, this is a separate regulation, it's the emergency planning regulations.

But, we have other plants that are doing that now. And they've got mitigating pathways of getting into the plant and so forth to do what they

need.

But, we prefer to see a Reg Guide 1.183 right now include the DPO. And that's basically it. Is if you wait until Rev 2, it will be beyond my working career at the NRC.

That's all I have. Mark, back to you.

MR. BLUMBERG: All right. Thank you, Mike. Could you just go to slide five, please?

The EDO report included recommendations on revising and consolidating the staff's updated guidance in DG-1389 in the ISG. It also provided specific issues that should be addressed and recommendations and authorizations regarding the methods to use in the DBA analysis.

Of note, DG-1199, which is a previous draft revision to Reg Guide 1.183, was not issued in part because the NRC wanted to inform Reg Guide 1.183, Rev 1, with experience gained by four license amendments.

The subject of the DPO is one of those four amendments. So, DG-1389 which superseded 1199, and even the draft final Reg Guide 1.183, Rev 1, were written to propagate the methods that the NRC found acceptable in those amendments.

However, the DPO shows that the methods

used in the amendments did not demonstrate compliance with 10 CFR 5067, which is used to ensure nuclear safety. These methods erroneously allowed a safety system to be removed and containment isolation valve leakage to be relaxed.

Approving the current version without incorporating the lessons learned and recommendations, will continue to propagate these methods that do not ensure nuclear safety.

So, in conclusion, we believe that it is clear that issuing Reg Guide 1.183, Rev 1, without incorporating the lessons learned from our DPO would be a mistake. The DPO provides a real life example of how methods in the proposed revision do not provide adequate guidance to ensure nuclear safety.

That concludes the presentation that we have on the DPO.

MEMBER PETTI: Questions? Thank you. Members, questions or comments again on anything before we go to public comment?

MEMBER BROWN: This is Charlie Brown.

MEMBER MARCH-LEUBA: Loudly, Charlie.

MEMBER BROWN: Okay. I'll speak louder. Can you hear me?

MEMBER MARCH-LEUBA: Yes, thanks.

MEMBER BROWN: As one that's not steeped in the language of this Reg Guide and the DPO, and, if I take the spin from it, I'm trying to understand from somebody that's knowledgeable, it's plain it's not me.

Main steam valve leakage from piping and other places that's obvious you shouldn't have that. What I get is that leakage that is a mess and plant experience will, this creates a problem with a source, to come up with a source term type determination.

Did you hear me okay?

MR. BLUMBERG: So, from my perspective, it causes a problem because it bypasses the pathways that we're crediting for removal of that radioactivity.

And that's exactly what was done in this example. Was that particular pathway was not considered and the downstream piping was credited.

MEMBER BROWN: But, is the volume of that, is the volume of that extension to create a problem? Or, is that just because we removed the downstream pathway?

In other words, is that a major contributor to overall source term usability or viability?

That was my question. I mean, it seems to me you can't operate because steam is pouring out of

all the packing breaks. So, how does -- whether you have this capturing system or not, it seems to me to be just another -- it maybe that -- it may not be that bad. I don't know how to say that any other way.

MR. BLUMBERG: So, there are several examples in the DPO Panel Report. I don't know if we have a DPO Appeal Panel member on this call. So, I don't want to speak for them.

But, there are several examples if you look at the report as to how that can occur. And why they felt that it was something that needed to be considered.

MEMBER BROWN: Okay.

MEMBER MARCH-LEUBA: The staff wants to make some comments?

MEMBER PETTI: Do you want to call it?

MEMBER MARCH-LEUBA: It's up to you.

MEMBER PETTI: We've got to be careful here, because this is a unique situation. So, I just want to make sure that the Members have adequately asked enough questions of the folks that have issued the DPO.

We can then move when that's done to what the staff feels. I just don't want to get into a back and forth on this.

MEMBER BROWN: That's fine. I just happened -- I tried to, when I read what this -- a lot of this was all I had.

MEMBER PETTI: Absolutely.

MEMBER BROWN: And I saw the issue and I'm just relating it to past experience. And how big of a source -- how big of a conflict is that?

And that's why I couldn't find an answer. I -- there was no way I was going to find it in the voluminous number of pages.

So, I figured I'd try to listen to the presentation and then see what the facts the operating plants have to say. But, I'm done. That was it for me.

MEMBER PETTI: Okay.

CHAIRMAN REMPE: So, the question really is a staff question rather than a question for the DPO presenters.

MEMBER BROWN: Well, I guess, I don't know. I don't know if maybe it's a mix. I don't know. I'll ask the staff about those.

CHAIRMAN REMPE: So, that's finished with the DPO presenters. And I suppose it's allowed to go back to a staff question before we go to public comment?

MR. BURKHART: It's up to you all. This is Larry Burkhart. I just want to make clear that this is the DPO presentation. And they are available for your questions.

Of course, whenever you want to talk to the staff, they'll answer your questions with that.

MEMBER BROWN: I'm just -- I'm just looking for somebody to tell me what the answer to that question is. That's all.

But, the staff sounds like the right place then.

MR. MARKLEY: The staff can explain it to you. But, I mean, the simple answer is, is this is how you don't meet 5067 for controlling dose.

MEMBER BROWN: So, it's about controlling dose. Not about dose levels.

MR. MARKLEY: This is Mike Markley. I apologize.

MEMBER BROWN: Okay. So, there is not a contributor to the general population. This is strictly a local issue relative to the dose.

MR. BLUMBERG: So, it would contribute to both offsite dose as well as in the control room. The source room contributes to both.

MEMBER PETTI: Right. But, it doesn't

challenge -- right, the public dose limits, it challenges the control room, dose limits imposed by the control room.

MR. BLUMBERG: So, in the example in the DPO, it did not challenge the offsite dose limits.

MEMBER BROWN: Okay. I understand that now. Okay.

MEMBER PETTI: Yeah. But, obviously it's higher, but yeah. Okay.

MEMBER BROWN: All right. I'm done and I'll let you all finish this off. That was my question.

MEMBER PETTI: Okay. How about the staff? Are you happy with the staff? Do you want -- are you happy with that?

MEMBER BROWN: I'm done. Yes. Thank you for that.

MEMBER PETTI: Okay.

MEMBER ROBERTS: I have one quick question. What is the downside of holding off on Rev 1? That means Rev 0 would be in effect for some time while you resolve the Rev 1 issues.

Is there some mitigation plan to get rid of Rev 0, or is Rev 0 just fine for the near-term?

MR. MARKLEY: This is Mike Markley.

Certainly the DPO panel report describes what the EDO's office would like to see done with it. To say, I don't know if you want to call it sunsetting it or retiring it or however they characterize it, but the challenge you have, and it was described earlier, is that a plant that has that as their licensing basis is kind of an, the regulator is in a hard place to try to encourage them to want to adopt a new version. They are capable of maintaining it at version, it's just it wouldn't be available for future adoption as a Rev 0.

MEMBER ROBERTS: Okay. Maybe I wasn't clear on the question. If you go to Slide 4 your last bullet says you prefer to resolve Reg Guide 1.183 Rev 1 now. It would presumably take, I don't know, I'm guess, it's not going to happen tomorrow, it takes time to work through those issues. Is there any consequence to delaying the existing version of Rev 1 and continue to rely on Rev 0 for however long it takes to resolve these issues?

MR. MARKLEY: I think Mark's Slide 3 spoke to some of the deficiencies. And I'll let Mark answer some of that. But I think the challenge from my perspective, this is Mike Markley again, that it's not going to solve the errors, you're going to still propagate more errors in the outcomes. I mean, it is

an improvement over Rev 0, no doubt.

MR. BLUMBERG: So I think the answer, this is Mark Blumberg again. I think the answer to your question relies with management. Maybe they can answer that. But my understanding is that it imposes an enterprise risk. And I'll let them expand upon that.

CHAIRMAN REMPE: We're going to have a problem --

PARTICIPANT: Yes.

CHAIRMAN REMPE: -- here again --

PARTICIPANT: Yes.

CHAIRMAN REMPE: -- with, this is the DPO Q&A part.

PARTICIPANT: Correct.

CHAIRMAN REMPE: And we have some questions to the staff after we get through this.

MEMBER ROBERTS: I understand that. And of course my question was, there is some downside to the recommendation.

(Simultaneous speaking.)

MEMBER ROBERTS: -- professional representation to delay to Rev 1. I'm trying to understand what that risk is.

MR. MARKLEY: So this is Mike Markley

again. And I apologize, but the last slide we have on, I mean, the last bullet on Slide 4, really characterizes the problem it puts us in. We feel like, as the DPO submitters, we'll never get out of the submittal of different views. We may end up having to submit a different view on the Reg Guide, as well as the first one (audio interference) precedence to come in the door because it's not being fixed.

MEMBER ROBERTS: Okay, so you're saying that it is, you think it's more of a risk proceeding with Rev 1 then delaying Rev 1?

MR. BLUMBERG: I would fully agree with that. You know, I'd just like to be given the opportunity to fix these issues.

MEMBER BROWN: Tom, could you repeat that, what you said again?

MEMBER ROBERTS: Yes. I think what Mark just agreed with is there is more of a risk with proceeding with Rev 1 than delaying it. But there is some risk either way. Right? There is some risk with continuing to proceed with Revision 0 for another, I don't want to say a time, a year or two years. I don't know what it will be to cut through all these DPO issues and come up with a revised Rev 1.

They said it will continue Rev 0 for some

time. I was trying to understand what risk that presents. That's the question I meant.

CHAIRMAN REMPE: So I guess I have a question. And I wasn't going to nitpick it, but I will I guess.

On Slide 2, the last bullet, you have the slide is evaluating the DPO appeals panel report recommendations and possible revisions to Rev 1 of Reg Guide 1.183. My understanding is that at some point the staff must evaluate and implement something to address those recommendations. It's a question of when, which is sort of what Tom is going into to. And so Tom hit the bottom line.

But I think if I were going to nitpick I would say that they less, and it's more of a, a win to implement the recommendations. Is how I look through to that last bullet --

MR. MARKLEY: Right.

CHAIRMAN REMPE: -- because they know they have to.

MR. MARKLEY: Yes. But what we have been told is it's going to kick down to Rev 2.

CHAIRMAN REMPE: Okay.

MR. MARKLEY: Yes.

MEMBER PETTI: At this point, given where

we are on time, I think we should give public a couple comments and then we'll come back to our discussion.

PARTICIPANT: Hold on.

MEMBER PETTI: Okay. Okay. Okay. So before we close it -- yes. Mike, I'd figured we'd address this after public comments, but where are you in the appeal process?

Tell us what the lay of the land is. And really, with your best guess of schedules really I think (audio interference) in this whole discussion.

MR. BLUMBERG: So the question is where we are in the appeal process?

MEMBER PETTI: Yes. It's not --

CHAIRMAN REMPE: This --

MEMBER PETTI: It's not for you; this is for the staff.

MR. BLUMBERG: Thank you.

MR. FRANOVICH: I don't want to take too much time, and we didn't come here to debate point-for-point, respecting the Committee's time.

There is probably one or two technical matters that I, like you chose just to address, that's already captured in the guidance. We also have some guidance already in this current rev regarding seismic lock downs. But I'd like Steve to address one or two

things, and then I have an overall perspective I'll share with you on time, and time frames and other matters that have been over relayed to you.

MR. JONES: So Steve Jones. Currently with Division of Advanced Reactors, but previously with the Division of Safety Systems.

With respect to packing leakage I'd just like to briefly indicate why it's considered. As it's been discussed in this meeting it's a style, you know, a generalized stylized assumptions of regarding how to calculate the dose on operators. It's a potential unfiltered release path.

When most of the leakage paths are contained within the secondary containment and there's a standby gas treatment system that would gather that leakage and filter it and release it through a stack which, you know, removes it from the control room.

In the case of a, the way the leakage is assessed on the MSIVs, there is two valves. One is inside primary containment, the other is outside primary containment and secondary containment in the stream tunnel. And their pressurized in-between and measure the overall leakage.

They cannot really detect what leakage goes where, but there is generally an assumption in

the calculations. But the inboard valve fails open and the outboard valve thus sees full containment pressure. And if you look at what's the most adverse lead path it would be when crediting potentially downstream, that position and other factors.

The leakage out the packing would be then released into the steam tunnel, which is a seismic structure. But it doesn't have a sealed confinement. And it's assumed to release directly to the atmosphere at that point. Generally as a very conservative assumption.

So there is a lot of conservatism stacked out there with the leakage, you know, the inboard valve being open, all the measured leakage being assumed to go out one path, and then that path being an unfiltered direct release to the atmosphere, where that location is, with respect to the control room intake. Or in the case of the offsite dose consequences, the dose boundary.

MEMBER MARCH-LEUBA: So you're -- this is Jose.

MR. JONES: Yes.

MEMBER MARCH-LEUBA: In summary what you're saying is the calculated dose from the leakage of the MSIV includes a large number of conservatisms?

MR. JONES: That's correct. Right.

MEMBER MARCH-LEUBA: Thank you.

MR. JONES: And then with respect to the Rev 0 versus Rev 1 of the Reg Guide, Rev 0 has in place endorse, or essentially endorsement of a BWR owner's group topical report that allows credit for the main condenser to collect leakage. And in that approach, there generally the packing leakage is not necessarily assessed. It's assumed all the leakage goes to the main condenser.

And we're essentially maintaining that same approach. We've both, in Rev 1 we've relaxed the detail requirements for low seismic hazard plants in assessing how robust the steamed piping is and the condenser is for that function.

MEMBER MARCH-LEUBA: For my back of the envelope thinking type calculations, what's the fraction of the source term that gets described by the condenser, is it 98 percent or is it 50 percent? Do you know?

MR. JONES: That's really outside my area of expertise. What I would say is that I guess the reason we're looking here is, this is a small fraction of overall primary containment leakage. What makes it significant is that it's unfiltered. And the filters

give you a, you know, 95 percent. So 20, a factor of 20 in reduction in dose. Plus it's released at a point far away.

In this case it's unfiltered --

MEMBER MARCH-LEUBA: Right.

MR. JONES: -- so you get 20 times the value. And it's near the, near the point that you're measuring, so you care about it more.

MEMBER MARCH-LEUBA: So again, in your mind it's conservatism that you can calculate the number simply because you know what the real answer is, but you know it's too large by design?

MR. JONES: Right.

MEMBER MARCH-LEUBA: Thank you.

MR. BLUMBERG: Could I as a staff Member, not a DPO presenter answer the technical question that's at hand?

MEMBER PETTI: Go ahead.

MR. BLUMBERG: So the amount of scrubbing that is performed in the condenser is highly dependent upon the aerosol deposition upstream. In general there is very little scrubbing in the condenser, but the condenser provides a hold up volume that is quite large that reduces the dose substantially. And it can be anywhere from, you know, ten to 100. Factor of ten

to 100.

It really is very site dependent in how much leakage is coming in and going out of that hold up volume. Does that help?

MEMBER MARCH-LEUBA: Yes. Thank you. So it could be as high as a hundred, a factor of a hundred, which would make it significant. But there are many assumptions like failure of the MSIV inside containment, which I think is the main number if you were going to PRA. Okay, thank you very much.

MR. MARKLEY: Mike please? Thank you. I'll be respectful of your time. I really do appreciate the views that Mark and Mike have shared with their colleagues and with the ACRS. I think we're better off organizationally to have those conversations. They're not easy conversations, but they are necessary.

We are a learning organization. We have lessons learned from our experience of the four LARs. In particular, the one plant right now where we have to clarify the licensing basis, working with the licensee.

And so, I want to note that this is part of our journey here in trying to advance our practices. Remind you of many of the specifics that

were raised today have been directly or indirectly considered in the DPO appeal process. Which essentially is now a settled matter.

Some of the raised concerns were supported and some were not. It's kind of an unprecedented situation. You know, I don't think we've seen that historically before.

A lot of the issues were very plant-specific also. So, we need to recognize that the discussion points were focused on a particular licensing action while there are some extended condition issues we have to deal.

The recommendations to the guide are opportunities to increase clarity in the future. But in our opinion, they didn't rise to the level of nuclear safety concerns.

Further, the DPO decision will be available to all NRC staff so the decision and expectations will be made clear for licensing actions that are reviewed in the interim period between Revision 1 and 2. So it will be widely known. It's already widely know that we have some lessons learned here.

Any new issues that were presented today are due appropriate consideration during the

development of Revision 2 and thus athwart all stakeholders opportunities for review and comment. Once publicly released, external stakeholders will also have the benefit of the EDO's decision to better inform their perspectives.

The concerns raised today need to be properly weighed against other changes that may obviate or support new positions. Along with any potential changes based on the EDO's direction from the appeal.

We have noted today that there are many future updates that need to be done. I won't repeat all those for sake of time. They are urgent matters. They do present enterprise risk.

If you haven't watched some of the hearings that our Commission has been faced with on Capitol Hill, you can watch those certainly on YouTube. I think there is a very high stakeholder interest in advancing technologies. And this is one small link, but an important link, in that overall mosaic of activities in our country.

The management team and staff have taken time to look at the EDO's decision on the appeal. Based on those reviews we have not identified any safety significant concerns on the impact of Rev 1.

We further believe that the issues raised or concerns raised are those that we can address, add clarity if needed in Rev 2. They are significant enhancements identified for in Rev 1 that we see an importance of issuing this guide now in the near-term.

In short, a balanced, integrated and timely effort is necessary to meet the Commissions expectations for the high priority increased enrichment rulemaking. We discussed that earlier.

And regulatory guide 1.183 is an important link in the effort. Regarding the appeal, the staff will develop an implementation plan, we're required to, we were directed by the EDO, and execute that plan in accordance with requirements, policies and other processes of the Commission.

In furtherance of the way forward I want to note the following. The EDO concluded there is no immediate safety concern. And as noted in the appeal panel's report, the issues did not warrant assessment under NRR's process for emergent safety issues, better known as the LIC-504 process.

I'm the executive sponsor for that process. I'm probably one of the few people left in the agency who actually worked on it. Not having to invoke this process does give you a perspective on the

level of safety concerns.

Shaping this outcome is the recognition of U.S. plants and performance and having greatly strengthened over the 25 years since 50.67 was actually issued. That was a long list of accomplishments, but here is a few of key moments.

Post-9/11 mitigation measures, post-Fukushima flex mitigation strategies, BWR reactor accident and capable events, reevaluation of both seismic and flood hazards, and other physical plant modifications ought to reduce plant risks.

That said I'd like to convey on behalf my comprehensive review of the EDO's decision of the appeal. My position, and that of the NRR executive team, is to issue the Reg Guide Revision 1 without undue delay. It's time to take the first step in advance regulatory positions and methods reflected in the state of the practice, and accident consequence dose assessments.

This step forward enables development and employment of the myriad of advance fuel technologies. The staff is requesting a letter report from the Committee for issuance of Revision 1. Thank you very much. And I'll take any questions if you have any (audio interference).

MEMBER PETTI: Okay.

MR. MARKLEY: Very good. Thank you.

MEMBER PETTI: Thank you. Let's now turn to public comment, and then we'll come back to our next steps to go over. So, let's see. First on the agenda is Paul Clifford from Integrated Nuclear Solutions.

MR. CLIFFORD: Okay, good morning. And thanks for the opportunity to speak. My name is Paul Clifford, and I am here representing myself. I am not here representing the NRC staff, like I did for 20 years. And I'm not here representing the Nuclear Industry.

Today my focus is on the MHA LOCA releases that are in Tables 1 and 2 of the proposed revision to the Reg Guide. I provided a set of slides, which I believe will be put in the transcripts. That was sent last week to each of the members. It might help to walk through those slides with me. And there is a lot of good background material there that may be useful to you during future deliberations on this topic.

Let me start with some conclusions and recommendations, and then I'll walk back through some of the examples that really illustrate the regulatory uncertainty that's introduced by the issuance of Rev

1.

First of all, the first main conclusion is that the Sandia reports document that, both the 2011 and the 2023 reports, that there is no burn-up dependents of the released fractions and that the dramatic differences are due solely to the improvements and enhancements to the MELCOR model that then have downstream effects on the accident progression, the timing and the magnitude of releases. So therefore these release fractions are applicable to the current operating fleet because they're not just increased as you go up higher and burn-up, they're applicable today.

That being said, these represent significant research findings. And they suggest that the plant's licensing basis, which is based on the Rev 0 regulatory basis, may be challenging. The NRC's response to these research findings is inconsistent past, present and regulatory policy.

The second major point is that the risk attributes and safety significance associated with these dramatic changes need to be evaluated to determine what, to determine if immediate actions are needed to ensure adequate protection of public health and safety.

I don't believe this has been done. The very first thing that's usually done in response to new research findings is to assess individual plant safety.

Moving on to Point Number 3. As I will show by these examples, issuing Rev 1 in its current form does not provide regulatory stability or predictability. A good example is that the 2011, if you recognize that there is no burn-up dependents, that means the 2011 release fractions are no longer relevant and they've been superseded by the 2023 release fractions. So therefore issuing Rev 1, by the time it's published, it will already be outdated.

The only way to provide regulatory stability and predictability is for the staff to follow regulatory requirements with respect to back fit and to follow their own management directives. A back fit determination to document a reasoned justified risk-informed decision on how, or if, these changes should be implemented on the exited fleet needs to be completed.

Second, after a revision to the guidance is made, a forward fit determination needs to document a reasoned justified risk-informed decision on when, or if, any future guidance will be applied to future

licensing actions.

As I mentioned, there is no burn-up dependence. But if you look at the revisions, Rev 0 is clearly listed, the range of applicability up to 62. Rev 1, range of applicability to 68. And the future proposed Revision 2, which will be based on the Sandia 2023, would be applicable up to 80 gigawatt days.

But the existing pathway that the staff has chosen is that all three Reg Guides are going to coexist. So if you recognize that there is overlapping applicability, you can understand the introduction of regulatory uncertainty with having three sets of guidance that are applicable to all plants.

So in my slides I provided some interesting text extracted directly from the Sandia reports. Which backs up these claims of no burn-up dependence.

Now, with respect to Commission policy, all of the Sandia reports are applicable to the existing fleet. Implementing new release fractions is a change to a regulatory position and must be evaluated in accordance with 10 CFR 51.09, back fit, and in accordance with Commission policy as dictated

by management directive 8.4.

Now let me just read just an excerpt of what the Commission policy is as stated in management directive 8.4. Now back fit and forward fit evaluation and analysis requirements provide the benefits of, A, regulatory stability, by ensuring that the changes the plant makes are necessary or provide a substantial safety enhancement. B, provide reasoned and informed NRC decision making by requiring the proposed actions be properly justified. And C, provide transparency of NRC decision making by requiring that the NRC document and make publicly available its analysis and its evaluations. To date I don't believe this has been done.

Now, my proposed pathway --

MEMBER PETTI: Paul how much longer?

MR. CLIFFORD: I believe one minute.

MEMBER PETTI: Okay.

MR. CLIFFORD: My proposed path forward would be to just recognize that the 2023 report supersedes the 2011 and therefore 2011 is irrelevant. And so the staff should evaluate the safety implications of 2023 using risk and safety significance in accordance with procedure, identify whether immediate actions are necessary and then

document a back fit determination on how to proceed with this new information. At that point Rev 1 should be issued based upon the 2023 report, and a forward fit determination should be documented.

An alternative, to avoid further delays, would be to issue Rev 1 without any changes to Tables 1 and 2 with just the other improvements that we heard about today. Get that on the street and then provide the staff an opportunity and time, schedule, to do a back fit, forward fit determination.

Now the three examples I'm providing here are, so Plant Y requests approval for a power uprate. No change to burn-up, so they're going to maintain their existing 62 meter watt days. All three revisions would be applicable to that current plant because it's staying at 62. And even if you follow the advertised range of applicability they're all applicable to 62.

So, is the licensee able to maintain Rev 0? I don't know. Will the staff expect that the latest revision be followed? Maybe. We don't know.

That's the regulatory uncertainty. We don't want licensees to spend a lot of time updating dose counts, submit them, then all of a sudden they're not accepted. So maintaining the three active

versions is a problem.

The second example would be Plant C, which would request a high burn-up time management reload cause up to 75. Well, if you just read the range of applicability, only Rev 2, the future Rev 2 would be applicable.

However, the Sandia reports would clearly demonstrate that the releases are not sensitive to high burn-up, so therefore would the plant be able to maintain Rev 0 which is in its current license basis, along with the argument that there is no burn-up dependents. I don't know if that would be accepted. And I'm not sure how the staff would response. That's regulatory uncertainty.

The final example would be, if we continue down this path and say six months from now a new plant requests a modest increase of burn-up, say 68 gigawatt days, they follow Rev 1, which has the 2011 release fractions, and they update all their dose calculations. Well during the LAR review, a individual staff reviewer may insist that I've got new information here with the Sandia 2023, and based upon adequate protection I require you to implement the latest review. The latest release fractions.

So therefore having guidance does not

equate to regulatory stability and predictability because we don't know how individual staff reviewers are going to address the fact that there is this known information out on the street left over.

MEMBER PETTI: Thank you. Next we will hear from Al Csontos representing NEI. Come on up.

MR. CSONTOS: Okay.

PARTICIPANT: Speak loudly.

MR. CSONTOS: Okay. Thank you. My name is Aladar Csontos from Nuclear Energy Institute presenting on the Industry's feedback and perspectives on the Reg Guide.

So we have a position paper that we will provide for the record. And we'll email it to Larry after this meeting is over. I will not go over the three pages, I will give you the highlights.

The highlight is, is that the Industry supports the publication of Rev 1, okay? We know it's not perfect, okay?

We have multiple utilities with multiple PWRs, and also BWRs, who are interested in exercising Rev 1 to support going to 68 burn off, but also put into your cycles. And I understand some of the concerns we have here but we need to walk before we run, okay?

And in this case we're walking, okay? And moving forward to a place where we can go to ATF higher burn-up. First and foremost going to the smaller portion here where we're going to multiple utilities and multiple sites that are going to need this to go to two year cycles. Okay?

I have one person from Southern Nuclear. Let me give some background. We have been waiting for this Reg Guide for a while. Okay? It was supposed to be out last September, I believe was the initial date that this given when this was first created. We have had many of meeting. Okay?

And the last RIC, NRC RIC meeting that was out there, we had Southern Nuclear provide an update and a request to have this done by no later than June of this year to support their business decisions. Okay. And that's just one utility.

So I'd like to have Tom Kindred from Southern Nuclear. I don't want you to just hear from me, I want you to hear from one of the utilities. I didn't want to get all five or six of the utilities to come in and tell you this, so I just got one for right now. So, Tom, can you go ahead and speak to your interests?

MR. KINDRED: Sure thing, Al. Can

everyone hear me?

MEMBER KIRCHNER: Yes.

MR. KINDRED: Hey, this --

MEMBER PETTI: Yes.

MR. KINDRED: All right, thank you. This is Tom Kindred. I'm a consulting engineer in the nuclear of fuel and analysis department at Southern Nuclear. I'd like to thank the Committee for hearing Southern Nuclear's position on this important topic.

In regards to Regulatory Guide 1.183 Revision 1, Southern Nuclear would like to affirm our support and alignment with the remarks delivered by the Nuclear Engineer Institute. Our fleet, the Southern Nuclear fleet, has a pressing need this year to see final publication of this Reg Guide as it would enable, whereas approval of the Reg Guide would enable streamline alternative source term methodologies and mid-increased burn-ups in the broader adoption of accident tolerant fuel load pellets and pressurized water applications.

We believe at Southern Nuclear that Revision 1 provides a stable and predictable licensing approach that is prudent to support our initial investments to move forward with ATF features that improve safety and lower the cost of electricity for

our customers. Thank you.

MR. CSONTOS: Thank you. And so that's the key point here is that, you know, we need Rev 2 as well. And we need Rev 2 as expeditiously as possible. We want to have engagements with the staff early, often and frequent, to address many of the issues that we're hearing articulated here.

We've identified a lot of the technical issues that we've identified in the Rev 1. We placed them into here. And also the discussion topics for Rev 2. Okay?

I don't want to go into, I don't want to belabor the point, but ultimately we do need durable, predictable guidance in this area for the future. Especially when it comes to going to two year cycles and up to 75 gigawatt days ran to you. And the ATF features. Okay?

That's a tall order in a short period of time, but we need it as soon as possible, but no later than 2026. Okay? That's really a hard, yes. We need, that's a hard goal date. So that's going to be tough, but I think that we can get there if we have these engagements, workshops.

And we have our counts and we can share these information back and forth. In an open forum.

And that's ultimately all I wanted to say. If there is anything else?

MEMBER PETTI: Thank you. Members, let's hope we can finish by the top of the hour here with where we want to go. Not that we'll have it all locked in. We are --

MEMBER MARCH-LEUBA: No, no. The court reporter --

(Simultaneously speaking.)

CHAIRMAN REMPE: The court reporter is to be on for the rest of this or is it time to take a break, release the court reporter --

(Simultaneous speaking.)

MEMBER PETTI: Oh, yes, true. Okay. Yes, are there any other public comments? Not hearing any, then, yes, I think we should release the court reporter.

CHAIRMAN REMPE: Okay. So at this time, Jim, we'd like you to go off -- we would like to go off the record, and we'd like you to come back at 1:00 p.m. today.

(Whereupon, the above-entitled matter went off the record at 10:46 a.m. and resumed at 12:59 p.m.)

CHAIRMAN REMPE: It is 1:00 p.m. on the

East Coast. And I'd like to ask our Member, Matt Sunseri, to lead us through the next topic for today.

MEMBER SUNSERI: Thank you, Chair Rempe. Good afternoon, everyone. My name is Matt Sunseri and I chair the Subcommittee that reviews plant license renewals.

This afternoon we are reviewing a subsequent license renewal applicant for plant St. Lucie. We will hear from the applicant, Florida Power & Light, and staff on the technical and safety aspects of the application.

The Subcommittee received the SLR application, relevant inspection reports and staff review around the 1st of August. We know that leading into our review there were no open or unresolved issues, and no confirmatory items. This status, by our process, allows the Subcommittee to perform our review virtually and to bring the results directly to the full Committee.

We have prepared a draft report based on our preliminary review, and we will be ready to finalize our report following the applicant and staff presentations and the full Committee deliberations.

Due to the work that I do outside of the ACRS, I am recusing myself from portions of this

review related to the metal and environmental fatigue or radiation embrittlement of the reactor pressure vessel. At this time I now turn the presentation over to Bernie Thomson, deputy director, new and renewed reactor licensing. Or NRC opening remarks. Bernie.

MS. THOMSON: No --

MEMBER SUNSERI: That's not going to work. You're going to have to speak louder.

MS. THOMSON: Thank you.

MEMBER SUNSERI: Yes.

MS. THOMSON: We are experiencing IT issues throughout all of the following results not just you, it's everyone. But thank you for the introduction. And thank you, Chair Rempe, and Members of the ACRS. My name is Bernie Thomson and I'm the Deputy Director or (audio interference) --

MEMBER SUNSERI: Okay. All right.

MS. THOMSON: Thank you again. Can you hear me now?

MEMBER MARCH-LEUBA: Yes.

MS. THOMSON: Okay. So, thank you again. And good afternoon. My name is Bernie Thomson. I'm the Deputy Director of the Division of New and Renewed Licenses in the Office of Nuclear Reactor Regulation.

We appreciate the opportunity today to

present to the ACRS the results of the staff's review of the seventh application for subsequent license renewal. This application was submitted by Florida Power & Light for the St. Lucie Plant Units 1 and 2 located in Jensen Beach, Florida.

For background, St. Lucie Units 1 and 2 received approval for their initial license renewal on October 2nd, 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 initial GALL Report was issued in 2001. The NRC guidance for license renewal has evolved over the years through enhancements and improvements based on the lessons learned from the NRC application reviews. And from consideration of both domestic and international industry operating experience.

The initial GALL Report for license renewal went through two revisions with additional interim staff guidance following Revision 2. GALL report Revision 2, along with these ISGs, were used to develop the guidance for subsequent license renewal that's contained in the GALL SLR report.

In addition to the previous license

renewal guidance, the GALL SLR Report included additional focus on the aging management and (audio interference) time-limited aging analysis focused on the operations in the 60- to 80-year time period.

NRC project managers, the project manager for the St. Lucie subsequent license renewal application review is Vaughn Thomas. Vaughn will introduce the staff who will be presenting and addressing questions regarding the safety review.

Part of the management team here with me today is Lauren Gibson, chief of the license renewal projects branch. As well as branch chiefs of the staff involved in the technical review.

Our senior technical advisory for aging management, Dr. John Wise, will also be available to answer questions from the Committee.

Paula Cooper, Region II, senior reactor inspector, will discuss the regional inspection activities.

John Hickman, senior resident inspector at St. Lucie, is also attending virtually and will support today's presentations.

I'd like to note that the staff completed its review with no confirmatory or open items in the safety evaluation. Finally, we will address any

questions you may have on the staff's presentation. And we look forward to a productive discussion today with the ACRS.

At this time I will turn the presentation over to Mr. Michael Davis, lead licensing project manager at Florida Power & Light to introduce his team and commence the presentation. Thank you.

MEMBER MARCH-LEUBA: Speak loud. I want to make sure I can hear you.

MR. DAVIS: All right. Good afternoon. And hopefully you can hear. Great.

Thank you for the opportunity for us to speak with you all today. We look forward to presenting the overview of the St. Lucie subsequent license renewal application.

And you can go ahead and go to Slide 2. R is introduction. On, before I say that, please feel free to stop us at any time if you do have a question as we go through the presentations.

My name is Mike Davis, licensing projects director for NextEra Energy and FPL. Presenting with me today will be Rob Craven, the St. Lucie site Vice President who's joining us virtually. And Steve Hale from ENERCON, our partner in developing the subsequent license renewal application.

Steve is one of the technical leads for the St. Lucie SLR project. And had the same role for the Point Beach and Turkey Point SLR efforts. Previous to that he worked for NextEra Energy for over 46 years. And held many positions throughout this career, including engineering director at Turkey Point. He was directly involved with the original license renewal for the Turkey Point and St. Lucie sites, as well as extended power uprates for Point Beach, Turkey Point and St. Lucie.

I would like now to turn the presentation over to the St. Lucie Vice President, Rob Craven, who has some opening remarks. And you can go to Slide 3. And hopefully Rob is able to join us. Rob, can you hear us?

PARTICIPANT: He's still showing muted.

MEMBER MARCH-LEUBA: He needs to unmute himself.

MR. DAVIS: Oh.

MEMBER SUNSERI: Rob, unmute yourself if you're trying to talk.

MR. CRAVEN: Okay, good. So, well, first of all, thanks for having us today. There is a couple of things I just want to touch on.

So, on the slide that we see on the screen

there, just one thing about our mission. And that's we produce energy in a safe, reliable and cost effective way while caring for employees, communities and the environment.

And the way we do this is really through some of our core values. And I want to touch on committee to excellence. And what that really refers to is we have good programs processes in place.

One of the things we are focusing on specifically is around aging management. And also our summer capital improvements we're doing at the site. We're spending tens of million dollars a year on improving the site overall. And then focusing on our aging management programs.

At the bottom you see some of our focused areas. One of the things we'll touch on is generation reliability. And those things don't happen without a very well run plant. Which is a focus for us.

The next slide for overview, not going to do our capacities, but just highlight that we are ROP Column 1. And all of our indicators are currently green. If there are no questions I'll turn it back over to Mike.

MR. DAVIS: Thank you, Rob. We can go to Slide, the next slide. All right, this slide, as well

as the next few slides will cover, contain some facts, figures and photos, which will provide a brief overview of the site itself.

St. Lucie Units 1 and 2 are four loop combustion engineering PWRs that are located on the Hutchinson Island just north of Jensen Beach, Florida. Cooling water for the units is pulled from, and discharged to, the Atlantic Ocean.

Both units initial licensed core power was 2560 megawatt thermal. And as a result of a power uprate in the 1980s, and an extended power uprate in 2012, both units are now licensed for 3020 megawatts thermal.

CHAIRMAN REMPE: Excuse me?

MEMBER MARCH-LEUBA: You're good.

CHAIRMAN REMPE: Okay. Could you talk a little bit about, if there are any differences in Units 1 and Units 2 because it's been a lot of years you've been operating, and are they still identical or have changes made as you go --

MR. DAVIS: Oh. Well there are differences between the units because of the time they were constructed. You know, around seven or eight years difference. And --

MR. HALE: Mike, let me. You know, Unit

1 was before Browns Ferry modeling. Unit 2 was after. So when you walk around the sites you will see some differences.

But basically NSSS is the same and a lot of the auxiliary systems are the same. But what you see more is some of the licensing criteria. Like missile detection for example. You'll see a part side wall on (audio interference) Unit 1, whereas you see a total enclosure on Unit 2. So the missile criteria changed.

And St. Lucie 2 is a cool shutdown plant, St. Lucie was a hot shutdown plant in terms of the design basis addition. So you see some differences like that, but overall the plants are very close to each other in terms of a secondary plant (audio interference).

CHAIRMAN REMPE: I can remember that your steam generators are made from a different vendor --

MR. HALE: Yes.

CHAIRMAN REMPE: So that's going to affect some of the subsequent discussion you had.

MR. HALE: Yes.

CHAIRMAN REMPE: And that's another --

MR. HALE: Yes. Unit 1 was B&W Canada. And Unit 2 was (audio interference) --

MEMBER SUNSERI: You just have to talk louder.

MR. HALE: Okay.

MEMBER SUNSERI: And maybe direct your voice this direction.

(Simultaneous speaking.)

CHAIRMAN REMPE: I just was curious, knowing the differences with the steam generators, and actually your response about the licensing criteria being different is interesting too because I don't remember that from before when we talked about St. Lucie.

MR. HALE: Well we did original license from (audio interference). We had to go, you know, the scoping and screening the first time we went through that. We spent a lot of time with the staff walking through the differences between the units on the licensing side.

CHAIRMAN REMPE: Okay. Thank you.

MR. DAVIS: Okay, we can go to the next slide.

MEMBER HALNON: Yes, before we get off that line of questioning, this is Greg Halnon, can you just briefly discuss the difference in the station blackout?

I noticed there is some scoping differences based on station blackout. Like the Unit 1 air compressors and whatnot. Is there a licensing difference or is there a physical interconnection to that?

MR. HALE: Yes.

MEMBER HALNON: Or --

MR. HALE: Yes. No, there is a licensing difference. Unit 1 has an alternate power supply. So Unit 1 relies on the interconnections with the Unit 2 diesel generators for station blackout. They call it alternate power supply.

Unit 2 is a DC coping plant. So its station blackout is based on DC power and restoration to the diesels after a certain time period.

MEMBER HALNON: Okay. That answers a lot of my questions. Appreciate it. Thank you.

MR. HALE: Yes, no problem.

MR. DAVIS: All right. So the current slide you see is a map of the site and its location. St. Lucie's Unit 1 and 2 are located on the Florida Atlantic Coast on Hutchinson Island, as I mentioned before. Just north of Jensen Beach.

On the map, the blue star in the center of the circle marks the location of the site. And then

the black dashed circle is a six mile radius, just to give perspective of the distances at the site in relation to the mainland there.

We can go to the next slide. Our current license expiration dates are March 2036 for Unit 1, and April of 2043 for Unit 2. The original license renewal application was approved in October of 2003. It was based on the draft Rev 0 of NUREG-1801. As part of that renewal ten programs were updated to the NUREG-1801.

Inspector procedure 71003, Phase 2 inspections were completed for Unit 1 in 2015. And for Unit 2 in 2017. Unit 1 entered the period of extended operation in 2016. And Unit 2 entered the period of extended operation this year in April.

And NEI 14-12 aging management plan effectiveness review was completed in January of 2021. And our subsequent license renewal application was submitted in October of 2021.

Go to the next slide. So this is an aerial view of the St. Lucie site. As I mentioned earlier, the site pulls it cooling water from the Atlantic Ocean. The intake canal is at the bottom of the picture, or the south, south of the plant. And continues around to the west side of the plant for the

cooling water. The discharge canal is the, returns water to the Atlantic Ocean. And it's the northern canal. The one at the top of the picture.

Looking at the inset picture you can see the Unit 1 containment building is to the north. And Unit 2 containment building to the south. The turbine buildings are on the west side of the containment buildings. And each auxiliary building is to the south of its containment building. Hopefully you can see that.

We can go to the next slide. So the next two slides are a listing of modifications and upgrades that the plant has completed since the first license renewal. Although not a comprehensive list, we have included some of the major modifications to the units. These modifications include those associated with the extended power uprate I mentioned that occurred in 2012.

The second bullet lists replacement of the Unit 2 steam generators. And we know that there is interest in discussing that. We have a slide coming up to talk about that in more detail.

And we can go ahead and go to the next slide.

MEMBER SUNSERI: Just a quick question.

When you have like, replace heater drain pumps, does that include the motors or just the pumps?

MR. HALE: This is Steve Hale again. Yes, the heater drain pumps, we did replace the motors as well with the pumps themselves.

MEMBER SUNSERI: Okay, thank you.

MR. HALE: In some cases you may be able to use the existing motor, but --

MEMBER SUNSERI: Right. Right.

MR. HALE: -- in those cases we had them replaced. Both.

MEMBER SUNSERI: And the feedwater pumps I presume are turbine driven or --

MR. HALE: No. They're --

MEMBER SUNSERI: Oh --

MR. HALE: -- motor driven.

MEMBER SUNSERI: And did you do the motors on them as well?

MR. HALE: -- FPL motor drive feed pumps. Yes.

MEMBER SUNSERI: Okay.

MEMBER MARCH-LEUBA: You're too far aware.

MEMBER SUNSERI: Yes. You'll just have to talk louder or get closer.

MR. HALE: Okay, I'm going to --

MEMBER SUNSERI: Okay. Thank you.

DR. SCHULTZ: Okay, this is, excuse me. This is Steve Schultz. I have a question related to the previous slide, as well as this one.

You mentioned that some of the modifications were done in support of the 2012 uprate. Could you just go through these pieces and let us know what the timing was for the, when these refurbishments were done so we get an idea of what was done in 2012?

And you mentioned how much you're doing on a annual basis, but give us an appreciation for what you've done recently and what was associated with the 2012 upgrade.

MR. DAVIS: Steve, do you want to come over?

MR. HALE: Let me get a little closer.

MR. DAVIS: Yes.

MEMBER MARCH-LEUBA: Steve, why don't you sit in my place.

MR. HALE: Most of the modifications we have listed here were implemented in the 2011 to 2012 time frame in order to prepare for the extended power uprates. Some that weren't necessarily were the reactor vessel heads. I don't have the specific dates, but they were replaced kind of the mid-2000s I

believe. In that general time frame.

Unit 1 steam generators, which aren't listed here, were replaced prior to our original license renewal. And that's why they're not listed here.

And the Unit 2 steam generators were replaced probably, you know, we had the one cycle of operation prior to going to EPU conditions. So they were towards the 2012 time frame as well.

But of course the high and low pressure kick steam pass, replacement of the MSRs, MSIVs and pre-steam bypass control capacity, feed pumps, feeder drain pumps, you know, all of those are all related to the extended power uprate. We did replace the main transformers. Those were replaced in advance of the EPU due to aging.

But generally most, to answer your question, generally most of these modifications were implemented to support the extended power uprate.

DR. SCHULTZ: That helps. Thank you very much.

MR. DAVIS: Okay.

MR. HALE: Okay.

MEMBER SUNSERI: You can stay there if you want.

(Laughter.)

MR. DAVIS: You can go probably, yes, to the next slide now. Oh, I'm sorry, yes, it's still me. We are on the next slide. Catching up.

Here we go. So as you can see, NextEra Energy has made significant investments in various systems and components. Steve Hale, who was just talking, one of our ENERCON technical leads here with us today, was directly involved with the St. Lucie extended power uprate project. And as you can tell by his discussion, he is very familiar with that.

And any other questions about any of these upgrades or modifications? All right, we can proceed to the next slide.

I want to talk a little bit about the SLR project team we assembled. They have many years of experience, both in Florida Power Light, and St. Lucie, specifically in license renewal. It is a multiple-discipline team that consists of ENERCON as the lead preparer for the submittal. And as well as Westinghouse, Framatome and Structural Integrity Associates in supporting those. Of course the project team was also supported by NextEra and FPL personnel, both at the fleet level and at the site itself.

Every aging management program, or SLR,

was assigned a program owner to support a portion of the application preparation and NRC review. The few technical leads for our project partner, ENERCON, one of them being Steve here today, and Jack Hoffman, who is with us virtually, have combined of almost 80 years of experience in FPL. Both developed the original license renewal application for both Turkey Point and St. Lucie.

This project team generated over 100 reports, which supported the application. These reports not only provided the next level of detail for the various aging management programs, and other parts of the application, but going forward they will also provide a way to ensure that the knowledge for SLR will be passed on to the personnel who will be implementing SLR.

In addition, as part of our implementation process, these reports will be incorporated into the plant control document system. As part of our implementation plan, we have also established an SLR coordinator position in order to ensure we have a successful transition to subsequent, I'm sorry, subsequent period of extended operation.

So now I am going to turn the presentation over to Steve Hale, who has already had a good part of

it, from ENERCON.

MR. HALE: Okay, thanks, Mike. For the subsequent license renewal application we follow the guidance of NEI 17-01, which was developed specifically for subsequent license renewal.

We also reviewed RAIs and responses associated with the first three application that went through SLR review. Which was Turkey Point Surry and Peach Bottom.

And when you address the results of those resolutions in our technical documents, and in the application. Other activities we implemented to ensure quality SLRA, included extensive interviews with the AMP owners, both at the site and the fleet level, as well as an AMP effectiveness review that Mike mentioned, which was done in January of 2021.

We also had several pre-application meetings with the NRC staff to ensure that we address everything we needed to in the application itself. Our approach going in was to comply with NUREG-21-91, our generic aging lessons learned SLR, and 21-92, the standard review plan SLR, to the greatest extent possible. And we feel that we have been able to accomplish that.

We, along with the NRC, work diligently to

ensure we met the SLR review schedule. Which culminated in the issue of the SER in July this year.

Go to Slide 13. Having been involved with both original, and now subsequent license renewal, I have somewhat of a unique perspective as to what was involved in the integrative plant assessment for both efforts. We thought the best way to describe our integrated plant assessment methodology was discuss the differences we found as we went through the process.

For scoping and screening there were really minimal differences because the scoping criteria really hasn't changed. There has been some evolution in the guidance documents associated with what's called the (a)(2) scoping criteria, which has to do with non-safety, which can affect safety. And that's because through various revisions to NEI 95-10 and other reviews that were performed on the license, original license applications, we had to address some additional criteria there.

Once you start moving into aging management reviews you start seeing the differences. When we did the original license renewal we addressed aging effects that were identified in industry documents at the time. And once you move into the

changes associated with GALL, you know, NUREG-1801, Revs 1 and 2, as well as the ISGs, and then with issue of GALL SLR, there were a number of aging effects we had to address expanded somewhat.

But the real differences you see are in the aging management programs. If you'll see, for initial license renewal, St. Lucie had 27 aging management programs whereas for subsequent license renewal they'll be 47 AMPs.

If you go to the next slide I'll provide a little more detail. The, you know, our commitment to try and comply with GALL SLR to the greatest extent possible, I think is confirmed by looking at the consistency. If you notice in the license renewal application there is just pages and pages of tables where you have to address the individual aging effects associated with all the various systems, structures and electrical systems. And we had over a 98 percent consistency with those AMR line items which originally goes to our goals when we establish the project.

If you look at the aging management programs, we had 47. As I mentioned previously, 11 of those are new, 36 are existing. And all of these were evaluated against the GALL AMP guidance documents. The differences were addressed either with

enhancements to those aging management programs, or exceptions where necessary.

There was one plant-specific program, which was a carry-on from original license renewal, which was management of fatigue, the pressurizer surge line.

As I mentioned previously, we did address RAIs, but in the individual AMPs we actually had a section in the technical documents that talked about the RAIs, the resolutions and how we addressed them in the aging management programs. In addition to that, to facilitate the NRC review we provided them a matrix of the Turkey Point, Surry and Peach Bottom RAIs and where they were addressed, both in the technical documents as well as the application. And they were able to use that on the ePortal.

Next slide.

MEMBER KIRCHNER: I got a question.

MR. HALE: Yes.

MEMBER KIRCHNER: Walt Kirchner speaking. When you went from the experience at, previous experience to this SLR application, did you find any new areas that you didn't previously cover in those prior programs?

MR. HALE: Well, there are some areas

where we had established new programs. I think certainly in the electrical area. Not that we weren't doing the activities. I'd like to distinguish between --

MEMBER KIRCHNER: Yes.

MR. HALE: -- performing activities and then something under the blessing of a license renewal AMP.

You know, we have some examples of that. Like a compressed air program. We don't have a, you know, we didn't have a license renewal compressed air program but we were committed to the regulatory guidance documents and things of that sort. And so we just basically took that, compared it to what was in GALL and identified enhancements if we needed to.

MEMBER KIRCHNER: Right. Any new areas or any surprises?

MR. HALE: No, I don't believe so. You know, when you operate the plant, as we've operated these for as long as we have --

MEMBER KIRCHNER: Yes.

MR. HALE: -- you know, there is very little that's going to surprise you. The staff is going to cover one item, which was the selective leeching they identified on the diesel generator

radiators. And they'll speak to that.

MEMBER KIRCHNER: Okay. Thank you.

MR. HALE: Yes. These are a list of the programs we had exceptions to. I don't have to go into these in detail, but if you have a specific question we can. These exceptions were reviewed and evaluated by the staff and accepted.

Most have to do with specific design features at the site that made it difficult or impossible to comply with the GALL requirements. And we took exceptions to that. Any questions there?

MEMBER KIRCHNER: Just, if I could continue?

MR. HALE: Sure.

MEMBER KIRCHNER: Given that you're an ocean front site, do you have any saltwater issues, like exterior tanks and such? Do those present special challenges for you?

MR. HALE: Yes, I think being on a saltwater site and salt latent atmosphere you will have more corrosion to manage. But typically, you know, those are already picked up by the programs identified with nothing really unique associated with it. Okay.

MEMBER KIRCHNER: Thank you.

MR. HALE: Yes. So any other questions here? Well now I want to move on to, I guess, Joy, you mentioned replacement steam generators at St. Lucie.

Next slide. I'm sorry. Having been involved with those discussions with ACRS originally, I remember the challenges we had at time. It was kind of like the perfect storm. We saw indications on the Unit 2 steam generators after one cycle, and then SONGS was going on at the same time. So we had quite a bit of dialogue considering we were increasing the power level by 12 percent on these steam generators.

Well, it's a good news story. We've done five inspections since EPU was implemented. We've seen a steady decrease in the wear rates on the steam generator tubes. And we've seen no evidence of tube-to-tube wear, which was the primary failure mechanism at SONGS. And the inspection technic we utilize specifically looks for that when we do our inspections.

The, I'm sorry, I got ahead of myself. The increase in tube wear rate after implementation of EPU was less than the conservative forecast that we had made. So it really shows that we had made the right calls in terms of what we assumed the wear rates

would be. And it actually turned out to be less with the first inspection after EPU. The inspections that we have performed have demonstrated acceptable margin against tube structural integrity with no tube-to-tube wear observed.

Comparing the pre and post-EPU values, overall benchmark steam generator tube wear rates at the anti-vibration bars had fallen significantly. So we've really seen what we anticipated to happen, happen.

CHAIRMAN REMPE: I have a question.

MR. HALE: Sure.

CHAIRMAN REMPE: Okay. So I actually got curious enough I went back and looked at some --

MR. HALE: Yes.

CHAIRMAN REMPE: -- of my old notes --

MR. HALE: Yes.

CHAIRMAN REMPE: -- and it was around 9,000 tubes I believe in each --

MR. HALE: Yes.

CHAIRMAN REMPE: -- of those generators.

MR. DAVIS: Yes.

CHAIRMAN REMPE: And at the time there were like, it seemed like there were like 2,000 indicators and around --

MR. HALE: Yes.

CHAIRMAN REMPE: -- a 1,000 tubes.

MR. HALE: Yes.

CHAIRMAN REMPE: What kind of indicators are you getting now?

MR. HALE: The specific numbers, I think we have Kester Thompson, you can respond to the specific numbers of indications.

CHAIRMAN REMPE: Okay. So, I mean --

MR. HALE: Yes.

CHAIRMAN REMPE: -- you've said significantly down.

MR. HALE: Yes.

CHAIRMAN REMPE: I like numbers, so could you --

MR. HALE: All right.

CHAIRMAN REMPE: -- tell me kind of what you're seeing nowadays, and what happened right after the EPU too?

MR. HALE: We did see in an increase in the wear rate, but was not as much as we anticipated.

CHAIRMAN REMPE: Yes. Okay.

MR. HALE: Okay. And, Kester, are you on?

MR. THOMPSON: Yes, I am.

MR. HALE: Why don't you give your name

and your affiliation. And then, I don't know if you heard the question?

MR. THOMPSON: Yes, I've heard the question.

MR. HALE: Okay.

MR. THOMPSON: Yes. This is Kester Thompson. I'm the steam generator program owner for the plant.

Yes, so like said before, right after EPU, yes, we did see a slight uptick in the wear rate. However, we've seen a dramatic drop in the wear rate since then. Very, very significant.

And right now the number of tubes that we're seeing exhibiting wear at the AVB locations, are about 2,600 tubes. 2,631 from the last inspection in the Alpha generator and 2,105 in the Bravo generator. Overall, there are more indications in the generator, but the area of concern in the U-band, where the anti-vibration bars are located, those are the numbers.

CHAIRMAN REMPE: So it's in the 2,000's for the indicators. We used to quote two different numbers, how many tubes as well as indicators. Is it -- I assume it's a fewer number of tubes with multiple indicators per tube. Do you have those numbers?

MR. THOMPSON: What I have is the total

number. Total number, not just for anti-vibration bars, would be the -- and the alpha generator would be 8,618, and the bravo generator is 6,267. The number of tubes I don't have immediately, but I can pull that by the time we get to the end of the presentation.

CHAIRMAN REMPE: Okay. So I'm a little confused. I thought you gave something that was in the 2,000s for the number of indicators that you're seeing, and it varied. It was like 2,300 and 2,105 for A and B. And then I was asking how many tubes give those indicators, and I would have expected you to come back with 1,000. But you came back with something like 8,000 or something, and I'm a little puzzled.

But if you could kind of fine-tune those numbers and let me know. And then I guess it doesn't matter as long as we can see, hey, it went down. But I am curious how much it went up right after the EPU. But anyway, it would be good to see actual numbers. It's nice to see the words. I'm not sure any of this goes in the letter, but I just would like to confirm, yeah, it is going down with actual numbers.

MR. THOMPSON: Okay. I understand the question, and we'll get an answer to you.

MEMBER SUNSERI: So, just for

clarification, let me add when you have an indication, it never goes away. It's always there. So they just accumulate.

CHAIRMAN REMPE: Oh. So that would be reasonable to have --

(Simultaneous speaking.)

MEMBER SUNSERI: And they grow, if you will, until they reach a tube plugging limit, which then the tube gets plugged and taken out of service. Now, a tube can have multiple indications on it, as you pointed out. Your tube plugging criteria is something around 40 percent, something like that? Yeah.

So you could have an indication that's 10 percent, and they just keep running. And then they monitor how much it wears, if it grows any. And so, when they're tracking that, they say their growth rates or their wear rates are declining; that's because they're not growing as fast as they had predicted.

MR. HALE: To give you an example, the number of tubes plugged in the 2A steam generator is 436, and the 2 Bravo steam generator, which doesn't seem to have quite the same number as the 2 Alpha, is 52.

MEMBER BALLINGER: This is Ron Ballinger. How much margin do you have?

MR. HALE: Well, the steam generators were originally built with 20 percent margin based on LOCA analysis. When we went through the extended power-up rate because I don't know if you all recall the thermal conductivity degradation issue with --

(Simultaneous speaking.)

MR. HALE: Yeah, with the best-estimate LOCA analysis, we had to restrict that as EEQ to 10 percent. But if needed, you know, there are evaluations that can be accomplished to expand that.

CHAIRMAN REMPE: So (audio interference.) about 27 percent or 16 tubes were plugged in 2A, and 12 percent or five were plugged in 2B. But that was back in 2011, and you said 426 --

MR. HALE: In 2 Alpha. Yeah.

CHAIRMAN REMPE: In 2 Alpha. So that's cumulative over all of the years.

MR. HALE: Cumulative. Yes.

CHAIRMAN REMPE: I guess if we don't want to talk about indicators, because it's a cumulative number and you're just getting more all the time, how many are plugged and --

MR. HALE: Yeah. That's really the bottom

line, is the tube's plugged, because that's where you've reached the limit that needs to be taken care of. And secondly, that can have a direct impact, depending on how many are plugged in your safety analysis.

(Simultaneous speaking.)

CHAIRMAN REMPE: -- give us some numbers on the last couple of inspections, how many you're seeing that now need to be plugged to get that 426, whatever the number is for each of them.

MR. HALE: Do you have that handy, Kester?

MR. THOMPSON: Yes. During the last inspection, we plugged 65 tubes. That was -- yes. Yeah. We plugged 65 tubes, and that was 59 in Alpha and six in Bravo. That was in the fall of 2021. For the previous outage -- get you the one here for the previous outage. These are in our reports that we submitted on the docket here.

MEMBER SUNSERI: But -- so let me ask this. I mean, from an overall safety and aging management program perspective, you're following every guideline. You're --

MR. HALE: Yes, sir.

MEMBER SUNSERI: -- doing the material management program. You're monitoring the performance

of the tubes using the latest inspection technology. You have plug-in criteria. You have thermal limits on, you know, the plant. And it just becomes an asset management --

(Simultaneous speaking.)

MR. HALE: Exactly. Yeah. And that's why you --

MEMBER BALLINGER: And there's no imbalance between steam generators. From the way I read it, it's pretty much uniform. You don't have X percent in one steam generator and 4X percent in the other steam generator. They're consistent throughout.

CHAIRMAN REMPE: They said 59 and --

MEMBER BALLINGER: And 60.

CHAIRMAN REMPE: I thought 65 total, 59 and six.

MR. HALE: That is correct.

(Simultaneous speaking.)

MEMBER BALLINGER: Well, I mean looking at the total. So you'd only have a mismatch in the margin.

CHAIRMAN REMPE: The total of --

MEMBER SUNSERI: Of 8,000, 9,000 tubes.

CHAIRMAN REMPE: Yeah. So there's 400-and-something-or-other total, and I'm not sure, I can

take another look.

MR. THOMPSON: On the previous outage, I do have those numbers. On the previous outage, it was a total of 85 tubes plugged, and 79 of the 85 were in the Alpha generator.

MEMBER BALLINGER: Oh.

CHAIRMAN REMPE: So it is -- you are aware of it, and you're tracking it. And eventually, again, you're at 400 and -- what was the number out of 8,000?

MR. THOMPSON: Four hundred and thirty-six.

CHAIRMAN REMPE: That's in one steam generator?

MR. THOMPSON: Yeah. That's in 2 Alpha.

CHAIRMAN REMPE: Out of 9,000 --

MR. THOMPSON: Right.

CHAIRMAN REMPE: -- which is good to know. As I recall -- and help me if I misremember -- the problem wasn't really the design. It was how they installed this --

MR. THOMPSON: It was how they were fabricated.

CHAIRMAN REMPE: Oh. I thought that was that they weren't supported --

MR. THOMPSON: Supported while they were

fabricated.

CHAIRMAN REMPE: Okay.

MR. THOMPSON: They were actually fabricated horizontally, and it had to do with the way they were fabricated where they had tubes actually resting on the AVBs and causing some deformation there. But I think from our perspective, bottom line, we wanted to ensure that we weren't seeing what happened at SONGS. And while we would prefer not to have that many indications in the new steam generator, I think it has been banished by the steam generator program. And that's the bottom line.

CHAIRMAN REMPE: Thank you. I just was curious --

(Simultaneous speaking.)

CHAIRMAN REMPE: -- appreciate the additional information.

MR. THOMPSON: I do have that additional information you asked earlier about the number of tubes, also. And can I respond?

CHAIRMAN REMPE: Mm-hmm.

MR. THOMPSON: So, right now, for wear and anti-vibration bars, the number of indications, 8,618 in Alpha and 6,267 in Bravo. The corresponding number of tubes that were affected were, in the Alpha

generator, 2,631, and in the Bravo generator, 2,105.

CHAIRMAN REMPE: Sounds (audio interference). Thank you.

MR. THOMPSON: Okay.

MR. HALE: Any other questions?

DR. SCHULTZ: Steve, this is Steve Schultz calling. Do you have any particular rationale associated with the consistent attenuation as we go forward in time? Is it just the wear rates perhaps were -- are different as you go forward, or any operational rationale associated with it?

(Simultaneous speaking.)

MR. HALE: Go ahead, Kester.

MR. THOMPSON: Oh, no. I was just about to say that the thing that you expect is to see something that's well behaved. You expect to see that attenuation and that our reduction in the wearing taking place -- so the fact that it came down tells you that you don't have an unstable environment taking place there, which is good. So attenuation is what we expect to see there.

So that's something that we're tracking, and as we manage the program and we conduct ready-current testing, we're at the same time looking to see, do we have any tube-to-tube we're developing? So

we have special techniques that we're using for that.

So we're doing a lot on the preventive front and just trying to stay ahead of things as we manage the aging of the equipment.

DR. SCHULTZ: The inspection technique sounds very well done. I really appreciate that you're able to do more than just look inside the tubes and see what's happening. Thank you.

MR. HALE: Okay. If we could move to --

(Simultaneous speaking.)

MR. HALE: -- other questions. Okay, sir.

If we move to slide 18, with regard to commitments, there are 52 for St. Lucie 1 and 51 for St. Lucie 2. And these are mostly on an AMP-by-AMP basis. And they also include all the pre-SPO inspections. These will be maintained separately for clarity and to avoid confusion with commitments for current license renewal.

There will be a new Chapter 19 in each of the St. Lucie UFSARs specifically dedicated to subsequent license renewal. The new chapter will include a complete table of the SLR commitments in each UFSAR. Both NextEra, FPL, and ENERCON -- we have extensive experience with license renewal and subsequent license renewal commitment management and

implementation. And this will ensure all actions will be completed per the schedule.

And as Mike mentioned, we've also established a lead position on the site, a highly qualified and experienced person to be the primary lead for subsequent license renewal.

If we move to slide 19, this just provides a summary of a time-limited aging analysis for St. Lucie. We perform the same level of effort and looking at all the CLB documents to identify TLAAs.

With comparison to original license renewal, we saw some of the dispositions change from the analytical resolutions, which are I and double I, to the aging management resolutions, or triple I. And that's primarily due to the fact that there are TLAA aging management programs at GALL now, which didn't exist when we went through this originally at St. Lucie.

We've also updated all of the environmentally assisted fatigue calculations. And this is because there were new guidance documents, like NUREG-6909, that provided, I would say, more evolved and developed guidance for performing those calculations.

If there are no more questions, I can turn

it back over to Mike.

MR. DAVIS: Okay. Thank you, Steve.

So, in summary, even though St. Lucie is a draft GALL Rev 0 plant for the first round of license renewal, we have adopted the GALL for subsequent license renewal with minimal exceptions, as Steve pointed out. In keeping with our sustainability focus, the goal now is to build and maintain margin to achieve 80 years of operation.

And you were asking about the upgrades that we have done, and we gave you a list of completed upgrades. But there is another upgrade, a significant one that's ongoing now, and that is the intake cooling water piping replacement. And that is an example of something currently in progress to maintain and build that margin for long-term operation.

Again, we would like to thank all the members of ACRS for the opportunity to present today. And if Rob Craven is still able to join us or to come off mute, I think he has some closing remarks.

Rob?

MR. CRAVEN: Thanks. Yeah, I just want to thank the team and the cooperation from the NRC and working together on this project for us.

MR. DAVIS: Thank you.

MEMBER SUNSERI: Okay. Thanks, Rob. Thanks, team members.

Any other questions or comments for this group before we switch chairs here?

MEMBER KIRCHNER: I have one.

MEMBER SUNSERI: Yeah, Walt?

MEMBER KIRCHNER: Just on a lighter note, the dates are pretty far out. You talked about having the senior person to oversee the SLR activities. Is that person going to make it to the --

(Simultaneous speaking.)

MEMBER KIRCHNER: -- program to get that person's --

MR. DAVIS: It's Greg. I think Greg is on the call. Greg Summers is that person. If he is willing to divulge his age --

MEMBER SUNSERI: He doesn't have to do that, but yeah, people need to be able to change their jobs and all that stuff like that. The fact that you're doing this thoughtfully is the important part.

MR. BRUNSON: This is Matt Brunson. So Mike alluded to it in one of the slides. We've --

(Simultaneous speaking.)

MEMBER SUNSERI: Just hold on. Stand by.

MR. BRUNSON: My apologies. This is Matt

Brunson. I'm the ENERCON project management at SLR. Just to answer that continuity concern, it's a fair one. I'm sure Greg will outlive us all, but he is a senior person, Greg Summers, the SLR site coordinator.

Those aging management programs are, of course, procedural documents. Underneath those live the procedures that execute those aging management programs. All of the bases reports that we developed the SLR from, 100 or so reports, have been updated through all of the NRC audit process. So they are living documents, those aging management programs.

And the procedures and the preventative maintenance requirements, sampling plans and work orders -- that's all written down. So that ensures the ability of continuity for the engineers that will be implementing this, especially for Unit 2, which is quite similar. Yeah. So does that answer your question okay, sir?

MEMBER KIRCHNER: Yes. Thank you.

MEMBER SUNSERI: Any others?

DR. SCHULTZ: Steve Schultz. Just one question. You mentioned early on that when you did the interviews of the site program SLR leads or owners -- and I was curious to know what were the one or two major things that you learned from that interview

process. Is that an extensive -- you follow the EPRI guidelines in doing the interviews? What was the takeaways, the major ones?

MR. HALE: Well, having participated in performing those interviews, my first take was the aging management program owners were very knowledgeable of their systems and some of the issues that may be involved with them. And nothing really -- I'll call it significant -- came out of it. But I was impressed with the level of knowledge and the commitment the aging management program owners had to their programs.

DR. SCHULTZ: Good. Thank you. Appreciate that.

MR. HALE: You're welcome.

MEMBER SUNSERI: All right. With that, we'll wrap up this part of the session, and we'll move into the next. So, if we could transition -- and is it Vaughn, or Vaughn Thomas, and Paula Cooper coming up? How's your outside voice? Is it strong? Does someone need to sit here?

SPEAKER: Let's do a sound check. Say your name.

MR. THOMAS: My name is Vaughn Thomas.

SPEAKER: You're good.

MS. COOPER: My name is Paula Cooper.

SPEAKER: Say again.

MS. COOPER: Paula Cooper.

SPEAKER: Yeah.

MR. WISE: My name is John Wise.

(Simultaneous speaking.)

SPEAKER: Okay. Let's make Vaughn talk more.

MR. THOMAS: Yeah. Yeah. No problem. Let me know when you're ready to start.

MEMBER SUNSERI: We're ready when you are.

MR. THOMAS: All right. Great. Thank you, Mike.

And good afternoon, Chairman and members of the ACRS. My name is Vaughn Thomas, and I'm the Licensing Renewal Project Manager for the safety review of the St. Lucie Units 1 and 2 subsequent licensing renewal application, or the SLRA.

As you heard from Bernie earlier today, we're here to discuss the staff's safety review of the St. Lucie SLRA as documented in the safety evaluation report, or the SER, which was issued on July 21st, 2023. Joining me today at the table is Lauren Gibson, Branch Chief for the Licensing Renewal Branch; Dr. John Wise, senior technical advisor for aging

management; and Paula Cooper, Region II senior reactor inspector.

Also joining us today, both in the audience and virtually, are some of the original staff, along with members of the ENERCON staff who participate in the video of the SLRA and conducted the audits.

Next slide, please.

We'll begin today's presentation with an overview of the St. Lucie licensing history before we move on to the St. Lucie aging management programs. We'll then discuss selected technical areas that we believe are of interest to the ACRS and hear from Region II inspections and plant material conditions before sharing the conclusion of the staff's safety review.

Next slide, please.

St. Lucie Units 1 and 2 were initially licensed on March 1st, 1976, and April 6th, 1983, respectively. In November 2001, the applicant submitted the initial license renewal application. Initial renewed license were issued October 2003, extending the expiration dates by 20 years to the max 2036, and for Unit 1 and April 2043 for Unit 2.

On August 3rd, 2021, Florida Power & Light

submitted an SLRA for St. Lucie Units 1 and 2, which was accepted for review on September 29th, 2021. And the safety evaluation was issued on July 21st, 2023, we have no open items or other items.

Next slide, please.

The St. Lucie SLRA described a total of 47 aging management programs, or AMPs, consisting of 36 existing programs and 11 new programs. This slide identifies applicant's original disposition of this as initially submitted in the application in the left column, and the final disposition as documented in the SE in the right column.

Other AMPs, except one, were evaluated for consistency with their own SLR report. And ultimately, all the AMPs were found to be consistent with suitable enhancements exceptions. applicant included one plant-specific (audio interference) program, which is reviewed in accordance with our standard plan, and a subsequent license renewal was found to be acceptable.

PARTICIPANT: Keep your voice high.

MR. THOMAS: Oh. Thank you. I would like to talk a little bit about what we did to review aging management activities and other technical information in the application. As part of our review, the staff

are conducting an aging management audit to review operating experience, or AMPs, and (audio interference) TLAAs.

This audit spanned 21 weeks from October 2021 to February 2022 and included both on-site and virtual activities and leveraged the portal and breakout sessions between the staff and the applicant. We were able to issue approximately 53 RAIs and two wrong RAIs, secondary RAIs, from this review.

The applicant submitted six SLRA supplements. Several clarification calls were conducted to discuss a variety of responses to RAIs that were issued by the NRC staff, including (audio interference) to the external surface of the emergency diesel generator radiator tubes, which I will discuss on the next slide.

Based on the review of the SLRA that resolves the audits and additional information provided by the applicant. The staff included that the applicant's aging management program activities were consistent with the criteria of the standard review plan, or SLRA, and of the requirements of 10 CFR Part 54.

Next slide, please. This slide represents a sample of several target areas of the SLRA review.

The first area relates to (audio interference) operating experience at St. Lucie (audio interference) generator radiator tube leak that occurred in June 2022.

Because of the leak -- was selective leaching on the external surfaces of the Unit 1 EVG radiator tubes that were exposed to an aggressive indoor air environment into salt-laden air. St. Lucie's failure analysis noted that inaccessibility of the radiator tube surface made visual and mechanical detection methods ineffective to detect selective leaching on the external surfaces of the radiator tubes.

A subsequent review of the historical operating experience by the NRC staff identified EVG radiator tube links from May and June 2021 -- June 2001, sorry -- and November of 2007. The discovery of these previous radiator tube failures and the ongoing periodic replacements of the Unit 1 EVG reactor radiators prompted the staff to request additional information.

In the applicant's response to staff RAI, St. Lucie proposed additional one-time inspection to confirm that the aggressive air environment was limited to energy radiators and to ensure selective

leaching of other (audio interference) components similar to the EVG radiator tube (audio interference) was not occurring.

St. Lucie also proposed to perform one-time volumetric inspections of the Unit 2 EVG radiator tubes to confirm that the selective leaching on the radiator tubes was unique in the Unit 1 EVG radiator tubes. St. Lucie also clarified that because of the Unit 1 EVG radiator tubes currently being replaced, they are not considered long-term components and do not require (audio interference).

The staff found St. Lucie's proposed changes acceptable to manage detected leaching (audio interference) components exposed to the aggressive air environment. The staff review of buried and underground piping and tanks program focused on external puttings, used for the subject components.

(Audio interference) the need for additional information with respect to (audio interference) and unlevel packing and tanks are externally quoted in accordance with the SLR report recommendations. The staff reviewed the applicant's response from the applicant demonstrates that the buried metallic piping is either concrete encased or externally (audio interference) and that underground

steel piping and portions of the buried fire protection system pipings are externally coated.

In addition, the applicant provided a (audio interference) exception related to their steel piping, which has not been confirmed as externally reported. The staff reviewed this exception a found it acceptable based on its review of solid proximity testing there -- which demonstrated a nonaggressive environment. And the fact that the staff did not identify instances of age-related degradation (audio interference) during its audit.

If there are any other questions or further information related to those areas, we have (audio interference) and are prepared to respond.

MEMBER KIRCHNER: Yes, I'll ask a question.

MR. THOMAS: Go right ahead.

MEMBER KIRCHNER: This is Walt Kirchner. So this is directed more to Paula, your team. How would you summarize the condition of the plant, particularly with respect to salt corrosion (audio interference) issue that's likely to (audio interference)

MS. COOPER: Yeah. As you can expect, St. Lucie's fall on both sides. Having the Indian River

on one side and having the Atlantic Ocean on the other, it's a very salty environment, for a lack of better word.

I would say last year -- I believe it was last year -- I was deployed from the region to do problem identification and resolution samples at the plant because there was concerns about the corrosion associated with the supports within the intake structure.

So it's -- I will say that the great thing about saltwater corrosion is it's very predictable. It's not a hidden mechanism. It's very easy to see, St. Lucie recognizes the condition, they're actively now, I guess, upgrading their intake cooling water piping to a more corrosion-resistant carbon-sealed pipes with stainless steel supports within the structure.

But it is a known mechanism. Is a managed mechanism. They do preemptively try to coat -- they're even doing some advanced needling techniques on the coating of some of these external surfaces. So they're definitely doing what they can in order to mitigate the corrosion and replace it when necessary.

But it does exist. They are managing that. And at this point, there's no safety concerns

with how they're managing plant. It's just (audio interference)

MEMBER KIRCHNER: How would you describe the overall condition of the external parts of the plant? How would you -- this is Walt Kirchner asking the staff. How would you describe to the public the general condition of the external exposed parts of the plant?

We heard earlier about differences between Unit 1 and Unit 2 in terms of missile protections (audio interference) water storage tanks. I would also ask about diesel storage tanks and such. What's the general condition? What's your assessment?

MS. COOPER: I'll also leverage -- John, if you'd like to also chime in on this question. Overall, from a public standpoint, we can't really just say it meets all regulatory requirements; it's meeting its design basis. Those words don't necessarily make the public feel good, right? They want to know more --

(Simultaneous speaking.)

MEMBER KIRCHNER: Condition assessment.

MS. COOPER: Yeah. So I would say that, I mean, especially given the environment, they're doing exceptionally well keeping up with the

degradation that they know, that they see the containment. You can see where they have the pulled forms. They've got a little bit of corrosion kind of coming through those old form concrete layers.

But they're aware of it. They're following all the ASME codes. I would say, at the end of the day, the conditions, especially given the environment, are being exceptionally maintained. They really are keeping up with it, and they do -- it's very apparent that they are doing what they can in order to get as long of a life as they can for this plant.

MR. HICKMAN: And if I could add to that -- this is John Hickman, the senior resident inspector here at St. Lucie. Can you hear me?

MEMBER KIRCHNER: Yes.

MR. HICKMAN: Just adding to what Paula just said, you know, I concur with that. And part of our baseline inspection program, which I manage here at St. Lucie, addresses and looks for those age management issues that might arise.

And one thing I can say with the relationship of the NRC with the licensee, they're very receptive to our observations and very equipped to correct these deficiencies that we bring up to

them. So, once again, they are meeting all regulatory requirements and maintaining the health and safety of the public.

MEMBER KIRCHNER: Thank you.

MR. THOMAS: Are there any other questions?

If there are no other questions, I'll turn it over to Paula Cooper, senior reactor inspector in Region II, who's going to discuss the inspections and the plant material conditions.

Paula?

MS. COOPER: Thanks, Vaughn.

Good afternoon, everyone. I'm Paula Cooper, senior reactor inspector from the Region II. Also joining me virtually, whom you've just heard, is John Hickman, senior resident inspector at the St. Lucie plant.

My role here today is to present the inspectors' perspective on the material condition of the plant and the adequacy of the site's performance on managing the effects of aging. These insights are gained from both region-based inspections and those performed by the residents.

This table represents the inspections that were performed through the license renewal inspection

program, specifically by the 71003 inspection procedure, which is a series of inspections that are performed after the initial license was renewed.

Each of the two units received a Phase 1 inspection. This phase occurs prior to the period of extended operation during an outage where the inspectors can walk down normally inaccessible areas, such as containment, to observe the implementation of aging management programs.

For the Unit 1 Phase 1 inspection, the inspector observed the phased array ultrasonic exam on safety injection lines for the indications of thermal embrittlement, traditional ultrasonic on the main feed water line for float-accelerated corrosion, and conducted walk-downs of safety-related structures, including containment.

The Phase 2 was a five-week inspection with two weeks on-site performed by a team of five inspectors prior to entering the PEO to verify the license renewal activities were completed. The inspectors reviewed 21 commitments and four aging management programs.

The inspectors determined that based on the samples selected, that the licensee completed or was on track to complete the necessary tasks to meet

the license renewal commitments, license conditions, and regulatory requirements associated with the issuance of the renewed operating license for Unit 1.

Unit 2, however, had several activities pending where it was decided to perform a Phase 2 for Unit 2. Prior to the Phase 2, a Phase 1 for Unit 2 was performed. During this inspection, the inspector observed ultrasonic exams on small-bore piping on the safety injection lines, phased array on the steam generator cold legs, and conducted walk-downs of safety-related structures, including containment.

The Phase 2 for Unit 2 was intended to be a five-week inspection with two weeks on-site. However, the second on-site week was canceled due to Hurricane Irma. I was able to travel back to the site a month later to complete the walk-downs and scoping reviews.

Despite the sudden change in the inspection schedule, the six inspectors on the team were still able to complete the review of 18 aging management programs and four TLAs and reached the same conclusion we reached for Unit 1.

The Phase 4 is the last remaining phase associated with the initial renewed license, and it's scheduled to be performed in 2027. Additionally, if

the license is renewed for the subsequent license renewal period, the inspection program has been recently revised to include two additional phases. These phases are equivalents of a Phase 2 and a Phase 4, shown on the slide as Phase 5 and 6, but for the SLR application.

In addition to the inspections mandated by the license renewal inspection program, inspectors used several ROP baseline inspections procedures to evaluate the implementation of the aging management activities. I'd also like to point out that this is not an exhaustive list. Most of the programs within the ROP have been updated to include aging management programs that are not represented on this slide.

The first example is a baseline inspection of the in-service inspection program. This inspection is performed each refueling outage and provides the inspectors the opportunity to review and assess inspections credited for aging management. This also includes the steam generate Eddy current inspection if you're interested in that particular one.

The second example is the heat sink inspection. That has now been incorporated under the comprehensive engineering team inspection. But this provides the inspectors an opportunity to review the

service water system as well as the ultimate heat sink.

Lastly, the problem identification and resolution inspection provides the opportunity as needed to address age-related degradation identified in the plant. One of the elements we focus on when reviewing the aging management programs under the license renewal inspection program is the ability for the site to react or adjust to newly identified aging mechanisms.

Last year, as Vaughn described earlier, the emergency diesel generator radiator tubes experienced a leak which in turn activated the licensee's corrective action program. The licensee performed additional examinations and testing which later concluded that the mechanism that led to the leak was selective leaching.

The material environment, in this case, air indoor uncontrolled, in the GALL did not identify selective leaching as an aging mechanism requiring management. Through the problem identification and resolution inspection, the residents, with consultation from regional and headquarters staff, were able to review and verify that the licensee was able to adequately address and implement appropriate

corrective actions to address this new aging mechanism.

I will now speak to the material condition of the St. Lucie plant from a regional inspector viewpoint. As a senior reactor inspector, I am tasked with performing several region-based inspections at sites generally located in the Southeast.

As it relates to St. Lucie, historically, I've performed a heat sink inspection, license renewal inspection, in-service inspections, ISFSI inspections, and multiple samples in the area of problem identification and resolution. As a result, there are few areas of the St. Lucie plant that I haven't seen.

Overall, I have no concern with the overall material condition of the plant that needs to be addressed outside of the baseline reactor oversight process, and the licensee has been successful at completing large capital improvement projects that maintain or improve the material condition of its structure, systems, and components. And the inspector will continue to inspect and assess the licensee's ability to manage the effects of aging through those baseline inspections.

At this time, I'll turn it back over to Vaughn Thomas, provided there are no questions.

MEMBER SUNSERI: Well, I would just pause here for a second, Vaughn, if that's okay. This Committee has over the years truly appreciated the eyes-on-site perspectives that the resident inspectors provide. So I would just want to pause here and make sure that this Committee has ample time to question Lauren and -- or Paula and --

MS. COOPER: John.

MEMBER SUNSERI: -- John about anything that we can't see but they can.

MEMBER KIRCHNER: So, Paula, this is Walt Kirchner again. How would you describe the containment condition? You mentioned that as a specific object of some of your inspections.

MS. COOPER: Yes. So, when I go into containment, it's during an outage. So, as you would expect, the one thing that you will routinely see in containment are booties and gloves. And that's the kind of, quote unquote, garbage that you could see inside a containment, and that's very common in every site across plant.

But what John will speak to and what he can follow up on it up before they even start, they have to do a full scrub of containment to get every little bit of material out of there (audio

interference) design basis accidents.

So John, as part of his role as a resident, is to do a containment close-down inspection and to make sure it is clean because there are some conditions that you would expect -- you get a lot of trays in, and you go in with double booties because a containment, one, easily breaks or cracks or -- they're not exactly very robust materials that you're using when you go into containment.

So that's very common. Outside of that, the plant is always maintained very clean. You don't have combustibles. You don't have garbage or Big Gulp cups or anything along those lines anywhere along the plant. It is very exceptionally well maintained.

MEMBER KIRCHNER: Well, that's good to hear. That's housekeeping. I should have been more specific. How about the actual containment structure? How (audio interference) the reactor vessel and other places where the Agency has targeted their inspections?

MS. COOPER: Sure. Yeah. In terms of your containment liner, your floor, your moisture barriers, they have an exceptional record with keeping up. They did have, as a fleet, a concern with a containment liner, but it was at a different plant.

So, as in response to that corrective action, they do have a more focused view on making sure that things like your moisture barriers are intact so you don't get moisture going into your inaccessible portions of your liner.

In terms of primary piping, it's generally stainless steel, so you don't see a lot of any kind of corrosion on a lot of that level of piping. So that's generally really well maintained. Their numbers are in good condition in terms of -- like they follow the IST program.

So, in terms of a lot of the internal components, especially when we're dealing with safety-related, they're all in exceptional condition. They are very well maintained, or they're in a material that is in need of maintenance, so it's just by default.

(Simultaneous speaking.)

MEMBER SUNSERI: Hold on a second. Jose first, and then whoever's on the line next.

(Simultaneous speaking.)

MR. HICKMAN: This is John Hickman.

MEMBER MARCH-LEUBA: Yeah. John, can you hold on a minute?

MR. HICKMAN: Sure.

MEMBER MARCH-LEUBA: (audio interference) age, there is some settlement on the foundation, and you will go down two inches in a corner. Any issues with -- I mean, obviously this plant is built over sand, right? So any issues with settlement?

MS. COOPER: Not that I'm aware of. There are sites in Region II that do have settlements, and they carry a license condition to monitor for settlement. So they're required to survey and verify that the differential settlement won't adversely impact the structure.

For St. Lucie, they do not have any of those concerns. Why their licensing base for their lost heat sink is the assumption that their bridge will fail due to liquefaction. But in terms of safety-related structures, those soils have been mitigated to eliminate those mechanisms. So you just don't see that concern.

MEMBER MARCH-LEUBA: Okay. Thank you.

MEMBER SUNSERI: Okay, John. Your turn.

MR. HICKMAN: Yeah, just to add to what Paula said and aspects of the responsibilities of the resident inspectors when containment is open of our pre-inspections at the beginning of outages and post inspections at the end of outage -- the things that

you guys just discussed is the conditions that we look for and address. And the licensee has to answer those questions prior to closing out those containments.

Another thing that we do and the aspect that really tests containment's ability to contain is the local leak rate test that we observe and we look at to ensure that if you have a radiological accident, that is maintained inside the containment, which is another way to assess the containment's performance for its design performing duties.

MEMBER SUNSERI: All right. Thank you.

Yeah, I don't want to go by without acknowledging the housekeeping comments, though, because my experience has been that's a good leading indicator of the attitudes of the staff. I mean, people are willing to pick up the trash and put it in the bins, or not put it on the ground, anyway. They're interested in taking care of the facility.

And that attitude permeates through the organization. So hearing the housekeeping results and then hearing the technical results, I'm not surprised that they're maintaining it well.

Any other questions?

MEMBER HALNON: Matt, this is Greg. I had a question about what Walt was talking about on the

containment.

A lot of the containments rooms -- I say a lot; I mean a couple of containments that had been on saltwater have had some cosmetic problems with spalling and other things. And plus your brother to the north, of course, are dealing with ASR. But is the containment outside structure from the standpoint of -- I realize it's probably sound from a structural perspective, but is the saltwater environment causing any unique problems or issues that the plant's dealing with from a -- presently is cosmetic, but maybe 20 years down the road may not be?

MS. COOPER: Yeah, at this point, I would say that their cold forms in between the layers of containment are getting saltwater intrusion where you can see some level of corrosion kind of on the edge surrounding containment. They don't have spalling, at least nothing that I would consider significant.

But in terms of corrosion at this point, it's qualitatively managed because it's not at a level that anyone would really consider long term would be a concern. But you can see it. It is occurring. Same thing with a lot of feed mechanisms on-site. It's a predictable mechanism. So they can see the level of corrosion just clearly from that leaching or

that kind of staining aspect on those layers.

But that's managed by ASME on their IWE and IWF. So, in terms of the level of inspection, it's a high-level inspection considering the safety significance of that structure. But I can't say other than that that I'm aware of any other structural deficiencies on their containment structure.

MEMBER HALNON: Okay.

MEMBER SUNSERI: John, do you want to add anything to -- you're good?

MR. HICKMAN: No. I'm good with what Paula said. And as she said, there's some staining, but nothing that rises to the level of concern.

MEMBER SUNSERI: All right. Okay. Good.

MEMBER HALNON: Trying to get back to the steam generator replace -- was that -- did they have to cut a bigger hole in containment to get the steam generators in or out, or could they use the equipment hatch as it was?

MS. COOPER: I don't know if it was for the reactor vessel --

(Simultaneous speaking.)

MR. HALE: Yeah. Yeah. Fortunately for St. Lucie, we did create a hatch that was big enough to pull the steam generators in and out.

MEMBER HALNON: Okay. So you don't know the condition, necessarily, of the outer layer or rebar, then, inside containment?

MR. HALE: Well, you've got to -- the St. Lucie containment is a free-standing steel containment with a three-foot annulus and a rebar containment, what they call a shield building.

MEMBER HALNON: Okay. So --

(Simultaneous speaking.)

MEMBER HALNON: Okay. I got it.

MR. HALE: Yeah. Okay?

MEMBER HALNON: Well, and -- so that's similar to the Davis-Besse shield building issue, and of course, they have the cold weather event that causes their problems. But it's still got to be part of the licensing basis for the area to be freestanding, and they went through a tremendous amount of calculations to determine that it wasn't going to fall down.

And I realize that's kind of far out there, but I just wanted to make sure that you don't have any question about the condition of the rebar inside the shield building.

MR. HALE: Yeah. No, we didn't see that when we did the steam generator replacement.

MEMBER HALNON: Okay. Thank you. That's all I got.

MEMBER SUNSERI: That was Steve Hale speaking.

MR. HALE: Oh, I'm sorry. Steve Hale.

MEMBER SUNSERI: No, I got a question.

Vesna, are you -- have your hand up, Vesna?

MEMBER DEIMITRIJEVIC: Yes. Sorry. I could not unmute myself.

MEMBER SUNSERI: No, you're good.

MEMBER DEIMITRIJEVIC: Short question. Does plant have a risk-informed ISI program for the section 11 piping?

MS. COOPER: I'm not sure. I don't know if John Hickman can respond. But that's something that's only been recently transitioning in the last five or so years. I don't know if (audio interference) are transitioned yet.

MR. HICKMAN: Yeah, I'm not sure if I can answer that. I've only been at St. Lucie for about two months, so --

MR. HALE: Yeah. They are certainly considering a risk-based ISI program. But based on my knowledge, I don't think they've implemented that yet.

MEMBER DEIMITRIJEVIC: Yeah. I have a lot of information, but I'm not at home all my regular computer, so I cannot access it. But my question was related because these risk-informed programs are based on the degradation mechanism, and I was wondering if -- I think if St. Lucie hasn't, it will probably be done ten years ago or something. I'm not sure.

And they may need upgrade after these degradation mechanism studies you have done. So that was what my question was related to. So okay. Thanks.

MR. HALE: All right. Thank you.

MEMBER SUNSERI: All right. So let's go back Vaughn now.

You have some closing remarks, I think.

MR. THOMAS: Yes.

Next slide.

Okay. Thank you, Paula. In conclusion, for the SLRA safety review, the staff finds the requirements of 10 CFR 54.29(a) have been met (audio interference).

This concludes our presentation, and the (audio interference) will answer any questions you actually have.

MEMBER SUNSERI: All right. Members, any

final questions for the staff?

Okay. We're going to open up the phone lines now for any comments from the members of the public. So, if you're a member of the public listening in, you can unmute your phone line by star-six. Or if you're on the Teams, just unmute and state your name and your statement.

All right. So we will transition away from the public comments, and one last chance for ACS members.

All right. I do want to think Florida Power & Light, your team. Obviously, you put a lot of effort into this application, good showing of support or to answer our questions, and likewise the staff. And in particular, we always love that the resident inspectors -- they especially make an effort to come here and visit with us in person. It makes it so much more meaningful.

Chair Rempe, at this point in time, that concludes this part of the presentation. I would recommend that if you're willing to entertain this, that we take a ten-minute break, put the draft report up, and I can at least read it into the record, and then see where you want to go from there.

CHAIRMAN REMPE: So, at this time, I'd

like to have us go off the record.

(Whereupon, the above-entitled matter went off the record at 2:27 p.m.)

St. Lucie Units 1 and 2
Subsequent License Renewal Application

ACRS Committee Meeting
September 6, 2023

Agenda

- Introductions (Mike Davis, Licensing Projects Director)
- Performance Philosophy (Rob Craven, PSL Site Vice-President)
- St. Lucie Site Overview (Mike Davis)
- Subsequent License Renewal (SLR) Project (Steve Hale, ENERCON)
  - Project Team
  - Regulatory Guidance
  - Integrated Plant Assessment
  - Aging Management Programs (AMPs)
  - Time-Limited Aging Analyses (TLAAs)
- Closing Remarks (Mike Davis)

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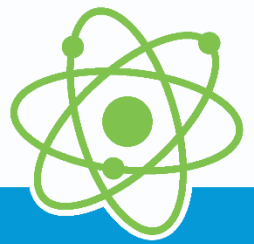
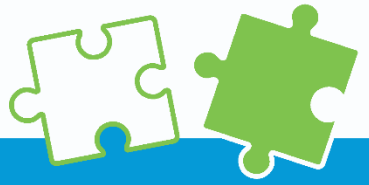
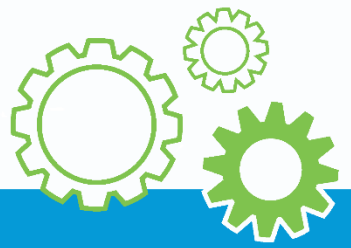
OUR VALUES	Committed to Excellence	Do the Right Thing	Treat People with Respect	Live Our Safety Guiding Principles	Use Our PDC Model
OUR CORE PRINCIPLES	Put Safety First	Committed to Excellence	Take Ownership	Build Trust	Act with Integrity
	Show Respect for All Individuals	Build a Diverse and Inclusive Team	Communicate Effectively	Promote Teamwork	Value Employee Development
OUR STRATEGIC FOCUS AREAS	Operational Excellence 	Organizational Effectiveness 	Generation Reliability 	Effective Business and Financial Performance 	

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St. Lucie Site Overview

Plant Performance

- Plant Capability Factor[1]:
  - Jun 2023: U1 – 93.35% U2 – 94.49%
  - Dec 2022: U1 – 93.02% U2 – 91.07%
  - Dec 2021: U1 – 92.35% U2 – 92.95%
  - Dec 2020: U1 – 90.22% U2 – 94.08%

- Regulatory status
  - ROP Actions Matrix Column 1
  - All ROP Indicators are Green

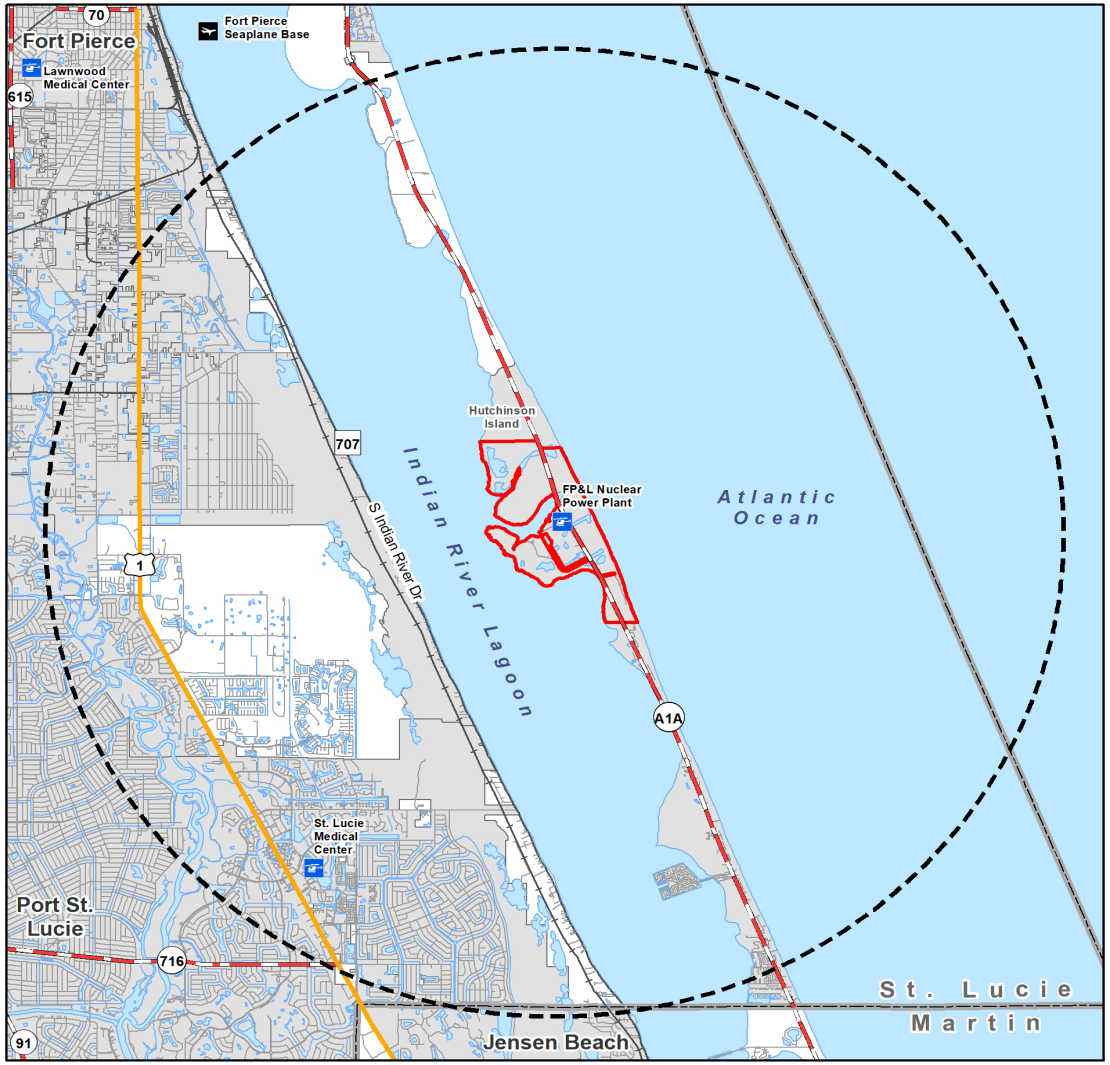
1) cumulative mean for previous 18 months

St. Lucie Site Overview

- Plant Description
  - Four-Loop Combustion Engineering (now Westinghouse) PWR
  - Cooling Water pulled from the Atlantic Ocean
  - Located on Hutchinson Island just North of Jensen Beach, Florida

- Licensed core power history, Units 1 and 2
  - 2560 MWt, initial license
  - 2700 MWt, Stretch Power uprate (U1-1981 and U2-1985)
  - 3020 MWt, 11% Extended Power Uprate (EPU) (2012)

St. Lucie Site Overview



St. Lucie Site Overview

- Current license expiration dates, 3/1/2036 (Unit 1), 4/6/2043 (Unit 2)
- Original license renewal application (LRA) approved on October 2, 2003
  - Based on draft Rev. 0 of NUREG-1801, Generic Aging Lessons Learned (GALL)
  - 10 programs were updated to NUREG-1801, GALL Rev. 0, as part of the application review process
- Unit 1 Inspection Procedure (IP) 71003 inspection completed 11/20/2015 and for Unit 2 on 10/20/2017
- Unit 1 entered PEO 3/1/2016 and Unit 2 entered PEO 4/6/2023
- NEI 14-12 AMP effectiveness review completed 1/25/2021
- Submitted Subsequent License Renewal Application (SLRA) on 10/12/2021

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St. Lucie Site Overview

Legend

- Plant Area (Restricted)
- PSL Site Boundary
- Exclusion Area Boundary
- County Park

Feet
0 1,000 2,000

St. Lucie Site Overview

Significant Plant Modifications Since Initial License Renewal

- Replaced reactor vessel heads
- Replaced Unit 2 Steam Generators
- Replaced High and Low Pressure Turbine Steam Paths
- Replaced Moisture Separator/Reheaters
- Upgraded Main Steam Isolation Valves
- Increased Main Steam Bypass Control System Capacity
- Replaced Main Feedwater Pumps and Modified Steam Generator Flow Control Valves
- Replaced Heater Drain Pumps

St. Lucie Site Overview

Significant Plant Modifications Since Initial License Renewal

- Replaced No. 5 Feedwater Heaters
- Replaced Main Generator Rotor and Rewind Stator
- Replaced Main Generator Hydrogen Coolers
- Replaced Turbine Plant Cooling Water Heat Exchangers
- Replaced Main Transformers

SLR Project

St. Lucie/ENERCON Team

- Multi-discipline team with significant nuclear experience, both on site and corporate
- Extensive license renewal experience, both licensing and implementation, including original LR efforts for Turkey Point and St. Lucie
- Extensive St. Lucie (PSL) specific engineering and licensing experience including the extended power uprate effort
- SLR Coordinator position staffed at site by senior, experienced person

SLR Project

Regulatory and Industry Guidance

- Used NEI 17-01 guidance
- Incorporated lessons learned from previous LRA/SLRAs and RAIs from the SLRA review of Turkey Point, Peach Bottom and Surry
- Conducted interviews with aging management program (AMP) owners on site November 2020 following the guidance of EPRI TR-110089, “Experience-based Interview Process for Power Plant Management”
- NEI 14-12 AMP effectiveness review completed 1/25/2021
- Followed NUREG-2191 (GALL-SLR) and NUREG-2192 (SRP-SLR) to the greatest extent possible
- Incorporated SLR ISGs

Current Status

- Safety Evaluation Report (SER) issued July 2023

SLR Project

Integrated Plant Assessment – Overall approach similar to that of original LR

- Differences between LR and SLR
  - Scoping and screening
    - -- Minimal differences
    - -- Some updates required to address 10 CFR 54.4(a)(2)
  - Aging management reviews (AMRs)
    - -- PSL initial LR per Draft GALL – additional aging effects required disposition based on GALL-SLR
  - Aging management programs (AMPs)
    - -- Significant differences
    - -- PSL initial LR, 27 AMPs
    - -- PSL SLR, 47 AMPs

SLR Project – AMPs

Consistency with NUREG-2191

- AMRs (SLRA Section 3)
  - Very consistent, >98% A through E notes (~2500-line items)
- AMPs (Appendix B)
  - Goal is to maximize consistency
  - Includes aging management effectiveness review of current LR AMPs

AMP Category		AMPs Consistent with GALL	AMPs Consistent with Enhancement	AMPs with Exception	AMPs with Exception and Enhancement	Plant Specific AMPs
Existing	36	6	23	1	5	1
New	11	11	0	0	0	0
Total AMPs	47					

Turkey Point, Surry, Peach Bottom RAIs addressed

- Separate section in each AMP basis document summarizes how the RAIs were addressed
- RAI matrix provided in ePortal

SLR Project – AMPs

AMPs with exceptions to GALL

- XI.M3, Reactor Head Closure Stud Bolting
  - Current bolting is high strength
- XI.M29, Outdoor and Large Atmospheric Metallic Storage Tanks
  - Unit 1 Refueling Water Tank bottoms inspection
- XI.M30, Fuel Oil Chemistry
  - Some fuel oil tanks do not allow for internal inspection, complete draining and/or cleaning
- XI.M31, Reactor Vessel Material Surveillance
  - Incremental adjustment to the current capsule removal schedule required
- XI.S3, ASME Section XI, Subsection IWF
  - High strength bolting is utilized in some applications
- XI.S6, Structures Monitoring
  - Ground water/Soil sampling

SLR Project – AMPs

PSL AMP effectiveness review, evaluated all AMPs – Completed

- Performed in accordance with NEI 14-12, Aging Management Program Effectiveness, in January 2021
- Review concluded that all AMPs continue to be effective with no failed elements

Steam Generator AMP

St. Lucie 2 Steam Generator performance after EPU

- Five RSG tube eddy current testing (ECT) completed post EPU – includes specific technique to identify tube-to-tube wear (TTW)
- Inspections of the St. Lucie 2 RSGs have confirmed that the primary tube degradation mode reported at San Onofre - RSG TTW has not been observed at St. Lucie 2

Wear at AVBs observed at St. Lucie 2 has consistently attenuated after EPU. TTW is not a credible degradation mode for the St. Lucie 2 RSGs.

SLR Project – AMPs

Commitments

- 52 for Unit 1, 51 for Unit 2
- Will be maintained separate from commitments for current LR

PSL Units 1 and 2 UFSARs

- New Chapter 19 in each UFSAR, maintained separate from current LR
- SLR commitments included in table in Chapter 19 in each UFSAR

Project Team has extensive experience with LR commitment management and implementation

SLR Project – TLAAs

- PSL TLAAs
  - RV Embrittlement
  - Metal Fatigue
  - Environmental Qualification
  - Metal Containment and Penetration Fatigue
  - Plant Specific TLAAs
- Based on GALL-SLR AMPs for TLAAs, some TLAA dispositions shifted from (i) or (ii) to (iii)
- Updates required to environmentally assisted fatigue calculations due to changes in guidance documents

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Closing Remarks

- Manage aging effects to ensure intended functions are maintained
- Evaluated TLAAs with acceptable results
- Satisfied requirements for subsequent license renewal
- Retain gains and build margin for the future
- Site VP Closing Comments

Advisory Committee on Reactor Safeguards
St. Lucie Plant, Units 1 and 2
Subsequent License Renewal Application (SLRA)
Safety Evaluation (SE)

September 6, 2023

Vaughn Thomas, Project Manager
Paula Cooper, Region II Senior Reactor Inspector

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Presentation Outline

- St. Lucie Plant (St. Lucie) Licensing History
- St. Lucie Aging Management Programs
- Specific Technical Areas of Review
- Inspections and Plant Material Conditions
- Conclusion on St. Lucie SLRA Review

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St. Lucie, Units 1 and 2: Licensing History

Initial License Renewal

Unit	Initial License	Initial License Renewal Application	Renewed License	Expiration Date
1	3/1/1976	11/30/2001	10/2/2003	3/1/2036
2	4/6/1983	11/30/2001	10/2/2003	4/6/2043

Subsequent License Renewal

Application Submitted	8/3/2021
Acceptance Determination	9/29/2021
Safety Evaluation	7/21/2023

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St. Lucie Units 1 and 2 Aging Management Programs

SLRA - Original Disposition of AMPs	SE - Final Disposition of AMPs
○ 47 AMPs in total ○ 35 existing programs • 4 consistent with GALL-SLR • 30 consistent with enhancements and/or exceptions • 1 plant-specific ○ 12 new programs • All consistent	○ 47 AMPs in total ○ 36 existing programs • 7 consistent with GALL-SLR • 28 consistent with enhancements and/or exceptions • 1 plant-specific ○ 11 new programs • All consistent

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Specific Areas of SLRA Review

- Selective Leaching & External Surface Monitoring
  - Selective Leaching in Air Environment
  - Aggressive Air Environment from Salt Laden Air
  - Inability to Externally Inspect Radiator Tube Surfaces
  - Prior Failures Discovered by the NRC Staff
- Buried and Underground Piping and Tanks

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Region II: AMP Inspections

License Renewal Inspection Program for Initial Period of Extended Operations

Inspection	Dates	Results
U1 IP 71003 Phase 1	May 22, 2015 ML15142A614	No Findings
U1 &2 IP 71003 Phase 2	January 4, 2016 ML16004A248	No Findings
U2 IP 71003 Phase 1	April 12, 2017 ML17102B262	No Findings
U2 IP 71003 Phase 2	November 30, 2017 ML17334A308	No Findings
U1 & U2 IP 71003 Phase 4	TBD: 2027	------

License Renewal Inspection Program for Subsequent License Renewal

Inspection	Dates	Results
U1 & U2 IP 71003 Phase 5	TBD: 2035	------
U1 & U2 71003 Phase 6	TBD: 2046	------

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Region II: AMP Inspections

ROP Baseline Inspections

Inspection	Date	Aging Management Program
IP71111.08 ISI	18-month RFO 2022 U1 2023 U2	Augmented Inspection Activities Boric Acid Corrosion Surveillance ISI Program – Component and Component Support Inspections ISI Program – Containment Inspections ISI Program – Reactor Vessel Reactor Vessel Internals Inspection Steam Generator Inspections
IP71111.21M Comprehensive Engineering Team Inspection (IP71111.07T Heat Sink)	Triennial TBD: 2025 (2022 U1&U2)	Service Water System and Inspection of Water Control Structures
IP71152 PI&R	Annual & Biennial 2023	Ensure activities in the licensee's aging management program are adequate to identify the aging effect prior to loss of SSC intended function, and whether the licensee's corrective actions address the adequacy of the aging management program.
IP71111.21N.04 Age Related Degradation	TBD: 2024	ALL

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Region II: Plant Material Condition and Conclusion

- Plant material condition meets regulatory requirements for systems, structures, and components.
- The inspectors found that the AMPs were being implemented in accordance with the license condition.
- The NRC will continue to monitor AMPs using the baseline Reactor Oversight Process and License Renewal Inspection Program.

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SLRA Review Conclusion

On the basis of its review of the SLRA, the staff determined that the requirements of 10 CFR 54.29(a) have been met for the subsequent license renewal of St. Lucie Plant, Units 1 and 2.

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United States Nuclear Regulatory Commission
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Revision of Regulatory Guide 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

Mark Blumberg, Senior Reactor Engineer (Technical Lead)
NRR/DRA/ARCB
mark.blumberg@nrc.gov

Joseph Messina, General Engineer (Technical Contact)
NRR/DSS/SFNB
joseph.messina@nrc.gov

September 6, 2023 - ACRS Full Committee Meeting

Agenda

1. Key Messages
2. Purpose and Regulatory Requirements
3. Revised Guidance
    a. Summary of Public Comments
    b. Highlighted Changes to Draft Rev. 1 in Response to Public Comments
4. Conclusions and Looking Forward (Rev. 2)

NRC Management and Staff Coordination

Steering Committee
Mike Franovich, NRR/DRA
Joseph Donoghue, NRR/DSS
Mohamed Shams, NRR/DANU
Michele Sampson, RES/DE

Working Group
NRR/DRA:
Mark Blumberg (Technical Lead)
John Parillo
Jerry Dozier
Shilp Vasavada
Elijah Dickson
Sean Meighan (Project Lead)
Ed Stutzcage (Acting Project Lead)

NRR/DSS:
Joseph Messina (Technical Lead)
Steve Jones

NRR/DANU:
Michelle Hart

RES:
Michael Eudy

OGC:
Mary Frances Woods

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Key Messages

- The NRC staff issued draft Regulatory Guide (RG) 1.183 Rev. 1 (DG-1389), "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" in April 2022 for public comment.

- ACRS and stakeholder involvement on the revision to RG 1.183 began prior to 2009 when an initial draft was issued for public comments. Recent development of the guidance involved numerous public workshops (4) and a meeting with the ACRS subcommittee prior to issuing DG-1389.

- The NRC staff prepared responses to the public comments on DG-1389, obtained internal concurrence, an Office of General Counsel no legal objection and provided the draft final version of the guidance for the ACRS review.

- Although the disposition of the 163 public comments required significant and detailed responses, the changes to DG-1389 in response to these comments were minimal because the DG-1389 guidance continues to be appropriate and defensible.

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Purpose and Regulatory Requirements

- NRC staff developed RG 1.183 Rev. 0 (July 2000) to support implementation of 10 CFR 50.67, “Accident source term”

- 10 CFR 50.67 establishes design basis dose limits for the control room, exclusion area boundary and low population zone for the purposes of determining the design requirements for mitigating safety systems

- RG 1.183 provides an acceptable method for modeling the design basis accidents used to demonstrate compliance with 10 CFR 50.67

- 10 CFR 50.67 allows for the use of alternative source terms other than the source term derived from NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants” used in RG 1.183, Rev. 0.

Purpose and Regulatory Requirements (cont.)

Regulatory Guide	Max. Rod Average Burnup GWd/MTU	Enrichment w/o U-235	Source Term
1.3, 1.4, 1.195	18-25	3-3.5	TID-14844
1.183, Rev. 0	Up to 60, With NUREG/CR-5009 up to 62	5	NUREG-1465
1.183, Rev. 1	Up to 68	8	SAND-2011-0128
1.183, Rev. 2 (proposed)	Up to 80	10	SAND-2023-01313

Purpose and Regulatory Requirements (cont.)

- Site Safety Assessment
  - 10 CFR 50.67, 10 CFR 50.34, 10 CFR 52.17, 10 CFR 52.47, 10 CFR 52.79, 10 CFR 52.137, 10 CFR 52.157
- Control Room Habitability
  - 10 CFR 50.67, 10 CFR 50, Appendix A, General Design Criteria (GDC) 19, 10 CFR 50.34
- Technical Support Center Habitability
  - 10 CFR 50, Appendix E

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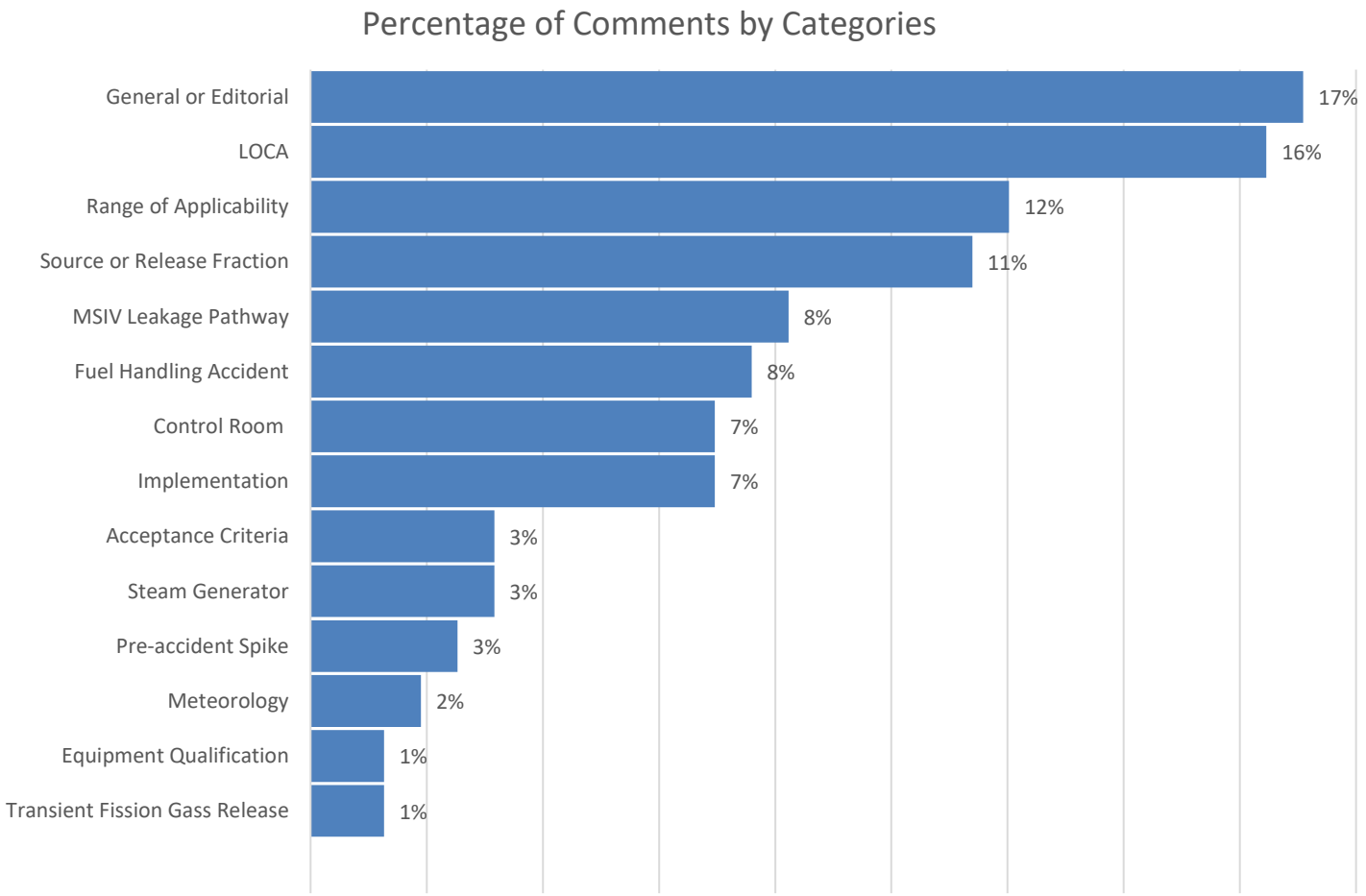
Summary of Public Comments



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Highlighted Changes to Draft Rev. 1 in Response to Public Comments

- Suppression Pool Scrubbing
- Control Room
- Implementation
- Natural Deposition in Containment
- Maximum Hypothetical Accident (MHA) loss of coolant accident (LOCA) Source Term
- Main steam isolation valve (MSIV) Leakage Pathway
- New Reactor Applications and Regulations
- Non-LOCA Gap Fractions, Figure 1
- Equipment Qualification
- Fuel Handling Accident
- Meteorology and Atmospheric Dispersion

Suppression Pool Scrubbing
2 Public Comments

- **Overview:** In DG-1389, staff did not propose any changes to the regulatory positions (RPs) on suppression pool scrubbing
- **Public Comments:** 1) re-evaluate boiling-water-reactor (BWR) release fractions based upon new accident sequences including suppression pool scrubbing, 2) allow for suppression pool scrubbing based upon a contractor's report on suppression pool decontamination (NUREG/CR-6153) and the State-of-the-Art-Reactor Consequence Analyses
- **Response to Public Comments:** 1) staff position on suppression pool scrubbing is already contained in RP A-2.5, and a revision to the LOCA source term release fractions is beyond scope of the RG revision, 2) revised RP A-2.5 to state that scrubbing has historically not been credited, rather than stating it should not generally be credited. Added an example where scrubbing has been credited in the primary containment cooling system for the Economic Simplified BWR
- **Changes made to DG-1389:** As described above and added the following footnote 1 to Appendix A.

  For an example of the modeling of radionuclide transport in containment with scrubbing credit in the primary containment cooling system (PCCS) of a new BWR reactor application, see Section 15.4.5 of NUREG-1966, "Final Safety Evaluation Report, Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design," (ADAMS Accession No. ML14100A304 (package)) (Ref. A-7).

Control Room
11 Public Comments

- **Overview:** DG-1389 introduced 1) the term transit dose to address the issue of control room access as stated in the GDC 19: *Adequate radiation protection shall be provided to permit* ***access and occupancy*** *of the control room under accident conditions…* Previously guidance was silent on the issue of control room access "transit" dose with the exception of ingress/egress, and 2) footnote 15 for radiation shine calculations
- **Public Comments:** 1) Requested the proposed guidance regarding transit dose be eliminated or a regulatory basis provide since most licensees have not included this evaluation in their licensing basis and that the GDC could be interpreted to apply only to the control room structure itself, 2) Footnote 15 be clarified or removed, 3) acknowledge that the finite geometry factor is incorporated computer codes such as RADTRAD, 4) add allowances for alternative conversion factors, and 5) whether sufficient safety margins exist for control room doses
- **Response to Public Comments:** Agreed with the need to revise transit dose, remove footnote 15, acknowledge the finite geometry factor and address the concern regarding safety margins.
- **Changes made to DG-1389:** The staff has removed the language associated with transit dose in RP 4.2, restored the RG 1.183, Revision 0 language and removed footnote 15. Staff is evaluating this issue and will determine whether to address this issue in a future revision to this RG.

Implementation

11 Public Comments

- **Overview:** DG-1389 revised the alternative source terms and methods for demonstrating compliance with 10 CFR 50.67, but stated that Revision 0 will continue to be available for use by licensees and applicants
- **Public Comments:** Requested clarifying statements, evaluations of implementation of Revision 1 for sample plants, examples of selective implementation and stated that Revision 1 should supersede Revision 0 because of *errors* in Revision 0
- **Response to Public Comments:** Agreed that clarifying statements and an example of selective implementation should be added
- **Changes made to DG-1389:** Clarifying statements regarding implementation were made to the *Background*, RP 1.1.1, *Safety Margins*, and RP 1.1.5, *Applicability to Light-Water Reactor Applications, Including, Advanced Evolutionary and Passive Designs*

Natural Deposition in Containment

4 Public Comments

- **Overview:** Staff proposed minor wording changes in DG-1389 and added that the reduction of airborne radioactivity by both sprays and gravitational settling (competing processes) should be evaluated on an individual case basis
- **Public Comments:** Questioned whether applying the gravitational settling model in NUREG/CR-6189 continues to be applicable when considering the added main steam line deposition models and the continued applicability of NUREG/CR-6189 with the MHA source term
- **Response to Public Comments:** Main steam line models did not consider removal by sprays and natural deposition in containment and NUREG/CR-6189 can be used if adjusted to incorporate the revised MHA source term
- **Changes made to DG-1389:** RP A-5 clarified to state the main steam line models are not valid when crediting other aerosols removal mechanisms, revised RP A-2.2 to state that reductions in NUREG/CR-6189 are not accepted, but the methods, when adjusted for the MHA source term, could be credited on a case-by-case basis

MHA LOCA Source Term
9 Public Comments

- **Overview:** Staff included text in DG-1389 to clarify many aspects of the ranges of applicability of the source term and its use
- **Public Comments:** applicability of/to 1) accident sequences and reactor designs, 2) chromium coated cladding and chromium-doped fuel, 3) impact of burnup/use of earlier source term, 4) Impact of burnup/use of earlier source term, 5) applicability to mixed oxide fuel (MOX), 6) applicability to accident tolerant fuel (ATF), 7) crediting of multiple removal mechanisms, 8) applicability of physical models to other designs, and 9) proposed restrictions to applicability in burnup, fuel type (including MOX), and clad type
- **Response to Public Comments:** 1) clarified ranges of applicability in text, 2) revised the applicability range of chromium limits for chromia-doped and chromia-coated fuel, 3) no change, 4) added text to *Background* of the draft final regulatory guidance regarding applicability of Rev 0, 5) clarified that the source term is not endorsed for MOX, 6) revised to state that the source term is not applicable to iron-chromium-aluminum (FeCrAl) and other long-term ATF concepts, 7) clarified text to address the use of different removal mechanisms, 8) revised text to state that the models would be considered for other designs on a case-by-case basis, and 9) defined applicability of source term before first RP (Section C) of RG
- **Changes made to DG-1389:** 1 & 2) Modified RP 3.2, *Release Fractions* , 3) no change in RG, 4) Modified *Background* section, 5) Modified *Introduction* to clarify the RG is not endorsed for MOX fuels 6) Revised footnote 10, 7) Modified R.P. A-5, 8) Modified R.P. A-5.6, 9) Modified beginning of Section C

MSIV Leakage Pathway

13 Public Comments

- **Overview:** In DG-1389 three main steam line deposition models and revised alternative drain path seismic guidance were added and the parts of the release pathway that can be credited were clarified
- **Public Comments:** On the use of non-safety related equipment, source term scaling factors, inboard deposition credit, clarifying the acceptability of proposed deposition models, assumed sizes of containment aerosols, and removal mechanisms for re-evaluated the Office of Nuclear Regulatory Research, Accident Evaluation Branch (AEB) 98-03 models
- **Response to Public Comments:** Agreed with most public comments, but disagreed with revising containment aerosol sizes and limiting credit to safety related components in the alternative leakage pathway
- **Changes made to DG-1389:** Added clarifications to RP A-5 regarding: scaling factors, removed the need for case-by-case evaluations when using the revised AEB 98-03 method in RP A-5.6.2, and added the technical basis for the crediting the condenser and main steam line piping

New Reactor Applications and Regulations

5 Public Comments

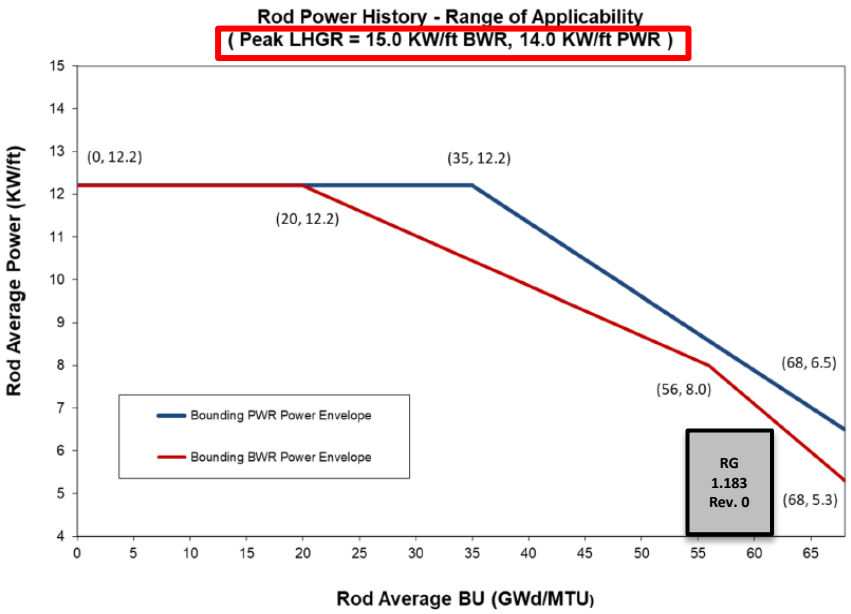
- **Overview:** In DG-1389, staff expanded the scope of use of RG 1.183, Revision 0 for use with new reactor applications
- **Public Comments:** Related to the meaning of *advanced LWR* [light-water reactor] and applicability of the guidance to new LWR applications
- **Response to Public Comments:** The staff responded to the public comments by clarifying that the guidance is applicable to any new LWR application under Parts 50 or 52 and listing all relevant regulatory requirements
- **Changes made to DG-1389:** Staff provided the following changes to RG
  - Section A, *Applicable Regulations*, Section B, *Reason for Revision* and *Background*, Section C, Subsection 1.1.5, *Applicability to New Light-Water Reactor Applications, Including Advanced Evolutionary and Passive Designs* by clarifying RG applicable to new LWR applications, including advanced evolutionary and passive LWR designs, and
  - Section A, *Applicable Regulations*, Section C, Subsection 1.3.1 *Design Basis Radiological Analyses* by providing the applicable regulations for design basis radiological analyses for new reactor applications (i.e., safety analysis report requirements, include all subparts of Part 52, and include 10 CFR 100.21 for siting)

Non-LOCA: Gap Fractions, Fig. 1

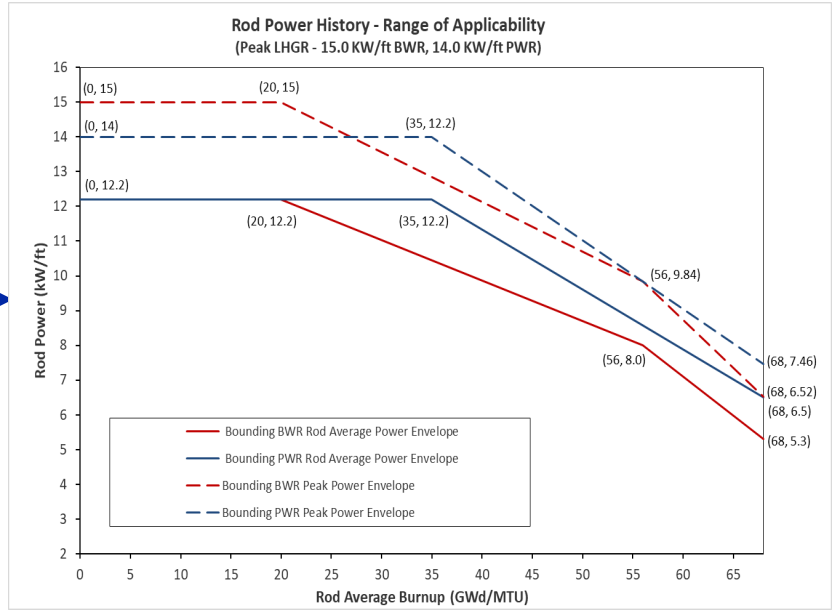
4 Public Comments

- **Overview**: In DG-1389, staff added a power history curve (LHGR (Linear heat generation rate) vs burn-up - see Figure 1) that must bound operation in order for the added non-LOCA gap fractions in Tables 3 (BWR) and 4 (pressurized-water reactor (PWR)) to be applicable and replaced the Rev. 0 Footnote 11 LHGR and burn-up limits
- **Public Comments**: Staff received public comments related to Figure 1 regarding the treatment of power uncertainties, the peak LHGR, and the applicability of Figure 1 to part-length rods (PLRs)
- **Response to public comments**:
  - Staff specified that power uncertainties should be accounted for when demonstrating adherence to the Fig. 1 power envelope, as stated in the Clifford tech. bases document referenced in DG-1389
  - Staff stated that the peak LHGR is not meant to be exceeded. If it cannot be met, the licensee can follow the procedure outlined in Appendix I
  - Staff stated that the non-LOCA gap fractions can be applied to PLRs if they meet the peak LHGR curve that was added to Fig. 1
- **Changes made to DG-1389**: staff updated Figure 1 to include peak LHGR power envelopes and added clarification on treatment of PLRs and power uncertainties to RP 3.2

Non-LOCA Gap Fractions, Fig. 1 (Cont'd)



DG-1389 Figure 1



New RG 1.183 Rev. 1 Figure 1

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Environmental Qualification
2 Public Comments

- **Overview:** In DG-1389 staff removed the Environmental Qualification (EQ) guidance from RG 1.183, Revision 0 (Appendix I, *Assumptions for Evaluating Radiation Doses for Equipment Qualification*), since updated EQ guidance is being incorporated into RG 1.89, Revision 2 (DG-1361)
- **Public Comments:** Staff received public comments related to the continued use of EQ guidance in RG 1.183, Revision 0, including the continued use of TID-14844 for EQ, as specified in RG 1.183, Revision 0
- **Response to Public Comments:** EQ guidance in RG 1.183, Revision 0 can continue to be used (including the use of the TID-14844 source term), provided that the plant design and licensing basis is in accordance with the applicability and limitations of RG 1.183, Revision 0
  - Other facilities, such as those that increase enrichment above 62 gigawatt-days per metric ton uranium (GWd/MTU) of 5 weight percentage Uranium-235 enrichment (bounds specified in RG 1.183, Revision 0), need to ensure that the guidance used is technically justified
- **Changes made to DG-1389:** Staff updated the *Background*, to clarify the continued applicability of RG 1.183, Revision 0, consistent with the public comment response, removed the reference to RG 1.183, Revision 0, Appendix I, and referenced RG 1.89, Revision 2 for EQ guidance. Also, since the revised RG 1.89 (Rev. 2) guidance on EQ was issued, it is now referenced in RG 1.183.

Fuel Handling Accident
12 Public Comments

- **Overview:** In DG-1389 staff provided a revised Fuel Handling Accident model allowing for modeling of retention and re-evolution of iodine from the spent fuel pool
- **Public Comments:** Requested: 1) clarification on the acceptable iodine species assumptions and a constant from a reference document, 2) revision of the method for water depths outside the applicability specified, 3) a method that calculated doses for fuel handlers and spent fuel pool workers, and 4) confirmation regarding the applicability of the release fractions labeled *other halogens* from Tables 3 and 4.
- **Response to Public Comments:** 1) The requested clarifications were provided, 2) revisions to the method are outside the applicability of the experiments used as the basis for the method, 3) the guidance is not intended for calculating doses to fuel handlers and workers near the spent fuel pool, and 4) the requested confirmation was provided
- **Changes made to DG-1389:** Modifications were made to include the requested clarifications into RPs B-1.3 and B-2

Meteorology and Atmospheric Dispersion

3 Public Comments

- **Overview:** DG-1389, added guidance that: 1) states that a modified version of the control room methodology in RG 1.194 may be used to estimate the offsite atmospheric dispersion factors out to distances of 1,200 meters, 2) align most unfavorable dispersion coincident with most adverse releases
- **Public Comments:** 1) Stated that the basis for the added guidance appears to be DG-4030, however, the associated Reg Guide 1.249 had not been issued, 2) clarify codes for dispersion and acceptability of prior guidance, 3) questioned need to include above "Overview" item 2
- **Response to Public Comments:** 1) The NRC staff agrees with the comment that a basis needs to be provided so a reference to RG 1.249 will be added, 2) codes are contained in referenced guidance, 3) aligned guidance with RG 1.194
- **Changes made to DG-1389:** The staff will revise RP 5.3 to add the reference to RG 1.249 (for estimating the offsite atmospheric dispersion factors using a modified version of the control room methodology in RG 1.194), and the staff clarified guidance to ensure a conservative dose is calculated

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Conclusions and Looking Forward

- The NRC staff has developed updated design basis dose analysis guidance, based upon experience since issuing RG 1.183, Rev. 0, research data, new analyses, and significant stakeholder involvement
  - These changes represent significant advancements in guidance for:
    - ATF, high-burnup fuel, and increased enrichment source term analyses (68 GWd/MTU rod average and enrichments up to 8 weight percent Uranium-235 for certain near-term ATF designs (chromium-coated cladding and chromia-doped fuel)
    - modeling BWR MSIV leakage including guidance for crediting holdup and deposition of MSIV leakage within the main steam lines and condenser for BWRs
    - non-LOCA source terms
    - revised transport and decontamination models for the fuel-handling design-basis accidents
- Looking Forward (Rev. 2)
  - Increased Enrichment (10 weight percent Uranium-235) and burnup (80 GWd/MTU)
  - Expanded scope for near-term ATFs (chromium-coated, FeCrAl –iron-chromium-aluminum)
  - Additional items under consideration – e.g., Increased Enrichment rulemaking, suppression pool scrubbing

Questions/Comments?

Mark Blumberg, Senior Reactor Engineer (Technical Lead)
NRR/DRA/ARCB
mark.blumberg@nrc.gov

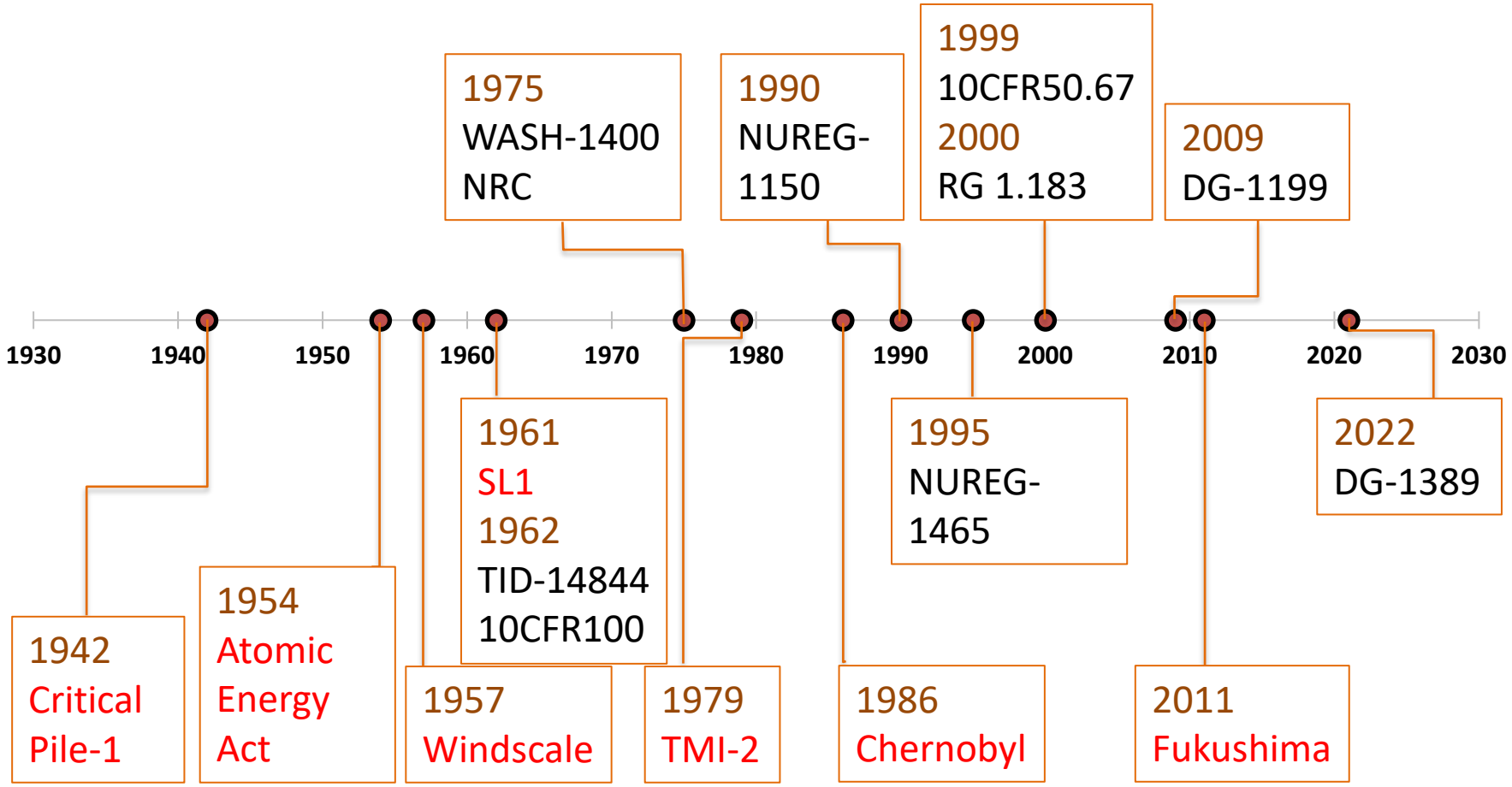
Joseph Messina, General Engineer (Technical Contact)
NRR/DSS/SFNB
joseph.messina@nrc.gov

SELECTED ACRONYMS

AEB	Accident Evaluation Branch
ATF	accident tolerant fuel
BWR	boiling-water reactor
DPO	differing profession opinion
EQ	equipment qualification
GDC	general design criteria
GWd	Gigawatt-days
LHGR	linear heat generation rate
LOCA	loss-of-coolant accident
LWR	light-water reactor
MHA	maximum hypothetical accident
MOX	mixed-oxide fuel
MSIV	main steam isolation valve
MTU	metric ton of uranium
PCCS	primary containment cooling system
PLR	part-length rods
PWR	pressurized-water reactor
RADTRAD	RADionuclide Transport, Removal, and Dose Estimation
RG	regulatory guide
RP	regulatory position

Backup Slides

LWR Source Term Timeline



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Differing Professional Opinion (DPO-2021-001)
Executive Director for Operations Appeal Panel Report Recommendations and Observations Impacting Draft Final Regulatory Guide (RG) 1.183, Revision 1

Mark Blumberg, NRR/ARCB

Michael Markley, NRR/LPL2-1

September 6, 2023

Differing Professional Opinion (DPO-2021-001) Panel Report[1]

- EDO issued the DPO Appeal Panel Analysis Report on August 25, 2023, and a memo that includes the following staff directed actions:
  - Take appropriate actions to ensure compliance with 10 CFR 50.67 for the subject plant and resolve the licensing basis clarity issues for the license amendment including the impact of MSIV packing leakage, the basis for limiting break location, and the aerosol deposition credit for the main condenser.
  - Develop an implementation plan for recommendations in the DPO Appeal Panel Analysis Report.
- Report recommendations stated, in part, that:
  - *In light of the issues identified in this report, in the near term, revise and consolidate the staff's updated guidance (DG-1389 and DRA-ISG-2021-01).*
  - *The DPO Appeal Panel believes any update to RG 1.183 should be consolidated into a single revision to the regulatory guide and not include companion interim staff guidance.*
  - *Enhanced focus on the overall intent of regulations related to the DBA analysis (e.g., focus on "assessing the acceptability of engineered safety features" rather than overreliance on non-safety-related features (e.g., deposition in power conversion systems)).*
- The report also provided specific issues, observations and conclusions that should be addressed.
- The NRC staff is evaluating Report recommendations and possible revision to Revision 1 of RG 1.183.

[1] DPO Case File for DPO-2021-001 (ADAMS No. ML23240A717, not yet publicly available).

Examples of Specific Issues, Observations and Conclusions Impacting Nuclear Safety

- *Revise language… relative to MSIV packing leakage, to better reflect actual operating experience (i.e., events indicate that packing leaks may not be a small contributor to overall main steam line leak rate)* and consider these limiting pathways to the environment.
- Compliance with 10 CFR 50.67 must be based upon the licensee's submittal that becomes part of the licensing basis. *The DPO Appeal Panel notes that care must be used when determining what assumptions constitute actual conservatisms vice basic assumptions underpinning a deterministic and somewhat stylized evaluation such as the DBA analysis.*
- *Removal coefficients for aerosol settling used in the ... AST LAR [license amendment request] are nonconservative and do not appear to reflect the state of knowledge that has been developed since issuance of AEB-98-03. … non conservatisms in AEB 98-03 … and other issues such as crediting spray[s] simultaneously with main steam line deposition the NRC has not issued timely updates, revised guidance or generic communication to … address these issues.*
- The *selection of the recirculation line break does not represent a bounding condition*.
- The DPO Panel disagreed *that the inherent seismic robustness of the pathway provides "high confidence" that a pathway to the condense[r] will be available in and of itself. Past seismic walkdowns related to this type of application for some licensees have identified the need* for *physical modifications to ensure the pathway is not failed by seismic-related failure mechanisms, such as system piping interactions and anchor issues.*

Key Points for Revising Regulatory Guide 1.183

- The technical issues are sufficiently complex to have warranted an ACRS Subcommittee meeting,
- Revision 1 to RG 1.183 should incorporate issues from DPO Appeal Decision,
- If not incorporated, plant-specific precedents using Revision 1 would be broadly replicated,
- Any future Revision 2 may be years in development and never be adopted,
- Prefer to resolve RG 1.183, Revision 1, now, including the DPO issues, and not submit differing views on RG 1.183, Revision 1, or plant-specific licensing actions.

DPO Lessons Learned

- The subject of this DPO provides a real-life example of why safety is not assured with the current draft revision of RG 1.183, Revision 1
- The AST license amendment, subject to the DPO, was one of four license amendments used to inform the development of RG 1.183, Revision 1 whose issuance was delayed so that it could be informed by the methods used.
- The DPO results show that these methods, used to remove and relax safety systems, were in error or need clarity.
- This version of the RG will continue to propagate methods that do not ensure nuclear safety.

Integrated Nuclear Solutions LLC

Regulatory Uncertainty Associated With RG 1.183

708th ACRS Meeting
September 6, 2023

Presenter: Paul Clifford

Conclusions and Recommendations

- SANDIA 2011 and 2023 MHA release fractions are **not burnup dependent** and are applicable to the current fleet

- These release fractions are **significant research findings** which suggest that existing plant licensing bases may be challenged

- NRC staff's response to these research findings are **inconsistent** with past precedent and regulatory policy

- **Risk attributes and safety significance** associated with these **changes** needs to be evaluated to determine if **immediate actions** are needed to ensure reasonable assurance of adequate protection of public health and safety

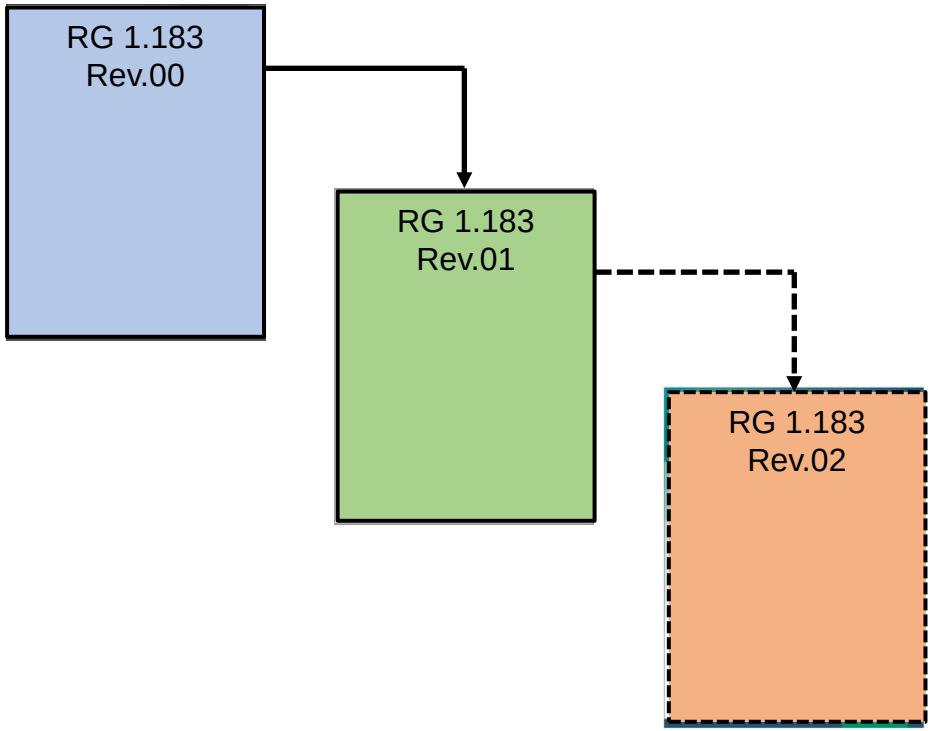
Conclusions and Recommendations (cont.)

- As shown in the examples, issuing RG 1.183 Rev.01 in its current form does not provide regulatory stability or predictability

- SANDIA 2011 is no longer relevant, being superseded by 2023

- To provide regulatory stability and predictability, the following actions should be completed:
  - **Backfit Determination** to document a reasoned, justified, risk-informed decision for how, **or if**, these changes should be implemented on existing fleet
  - **Forward Fit Determination** to document a reasoned, justified, risk-informed decision for when, **or if**, these changes should be implemented going forward

Proposed Path Forward - Range of Applicability



- Up to 62 GWd/MTU, 5.0 w/o enrichment
- Up to 68 GWd/MTU, 8.0 w/o enrichment
- Up to 80 GWd/MTU, 8.0 w/o enrichment

All versions of RG 1.183 will coexist

Misleading Range of Applicability

SAND2011-0128

In terms of fractional releases, source terms developed for high burnup fuel and for MOX do not differ markedly from source terms developed by similar means for lower burnup fuel or for low-enrichment uranium dioxide fuel. The source terms do differ from those described in NUREG-1465. These differences can be attributed to improved understanding of reactor accident phenomenology and modeling since publication of NUREG-1465.

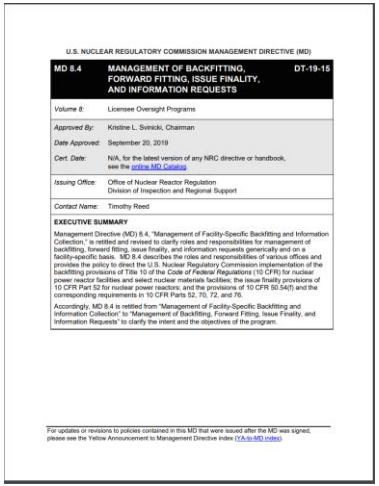
SAND2023-01313

Finally, this analysis demonstrates that in-containment source terms are essentially unchanged by increased burnup or elevated enrichment and that the most significant variation in source term continues to arise from differences between accident scenarios.

No Burnup Dependence

Commission Policy

- Both SANDIA reports are applicable to existing fleet (although 2023 supersedes 2011)
- Implementing new release fractions is a **change to a regulatory position** and should be evaluated in accordance with Commission Policy



U.S. NUCLEAR REGULATORY COMMISSION MANAGEMENT DIRECTIVE (MD)

MD 8.4	MANAGEMENT OF BACKFITTING, FORWARD FITTING, ISSUE FINALITY, AND INFORMATION REQUESTS	DT-19-15
Volume 8:	Licensee Oversight Programs	
Approved By:	Kristine L. Svinicki, Chairman	
Date Approved:	September 20, 2019	
Cert. Date:	N/A, for the latest version of any NRC directive or handbook, see the online MD Catalog.	
Issuing Office:	Office of Nuclear Reactor Regulation Division of Inspection and Regional Support	
Contact Name:	Timothy Reed	

EXECUTIVE SUMMARY

Management Directive (MD) 8.4, "Management of Facility-Specific Backfitting and Information Collection," is retitled and revised to clarify roles and responsibilities for management of backfitting, forward fitting, issue finality, and information requests generically and on a facility-specific basis. MD 8.4 describes the roles and responsibilities of various offices and provides the policy to direct the U.S. Nuclear Regulatory Commission implementation of the backfitting provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) for nuclear power reactor facilities and select nuclear materials facilities; the issue finality provisions of 10 CFR Part 52 for nuclear power reactors; and the provisions of 10 CFR 50.54(f) and the corresponding requirements in 10 CFR Parts 52, 70, 72, and 76.

Accordingly, MD 8.4 is retitled from "Management of Facility-Specific Backfitting and Information Collection" to "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests" to clarify the intent and the objectives of the program.

For updates or revisions to policies contained in this MD that were issued after the MD was signed, please see the Yellow Announcement to Management Directive Index (YA-to-MD Index).

MD 8.4 Section I, Policy

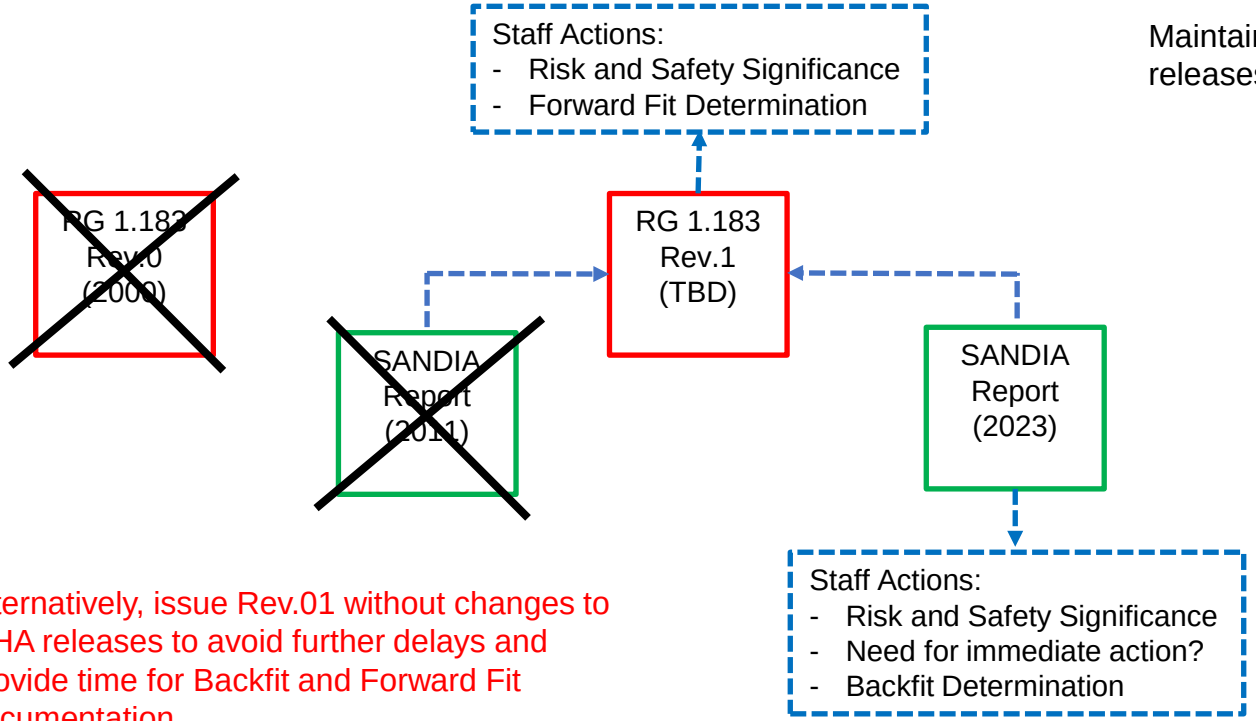
The backfit and forward fit evaluation and analysis requirements ensure that the NRC demonstrates, before implementing the backfit or forward fit, that the action would meet certain standards that vary based on the type of facility. In accordance with the NRC Principles of Good Regulation and the Administrative Procedure Act, the backfitting rules and policies and the forward fitting policies provide the following:

(a) Regulatory stability, by ensuring that the changes the NRC makes are necessary or provide a substantial safety enhancement;

(b) Reasoned and informed NRC decisionmaking, by requiring the proposed action be properly justified; and

(c) Transparency of NRC decisionmaking, by requiring that the NRC document and make publicly available its analyses and evaluations.

Path Forward – Recommended



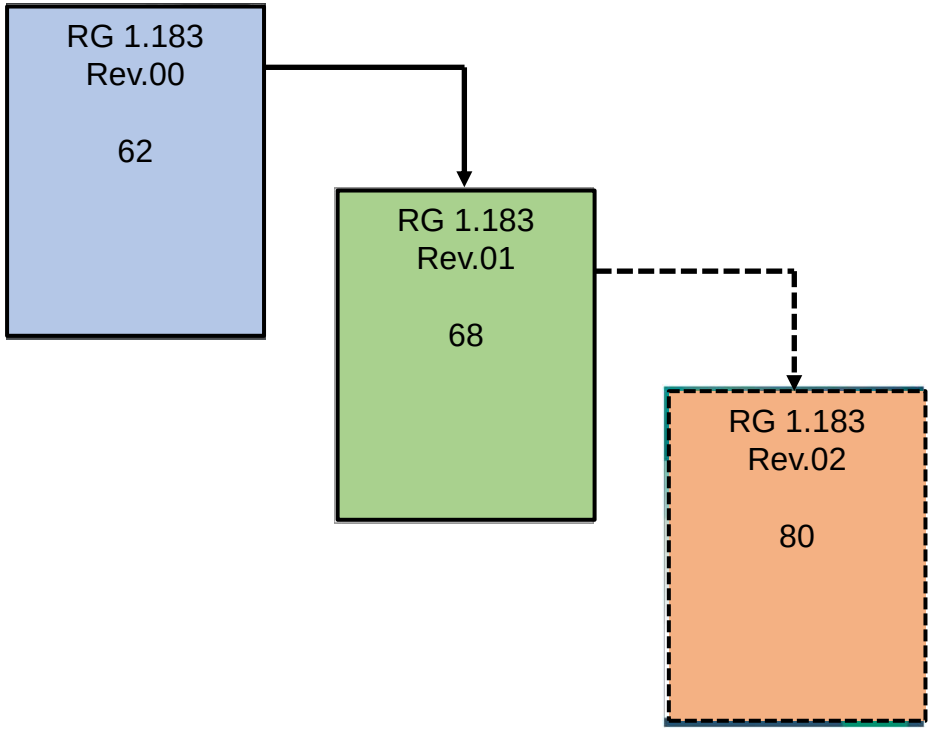
Maintaining multiple versions of MHA releases promotes regulatory uncertainty

Retire Rev.0

Alternatively, issue Rev.01 without changes to MHA releases to avoid further delays and provide time for Backfit and Forward Fit documentation

Regulatory Uncertainty – Examples

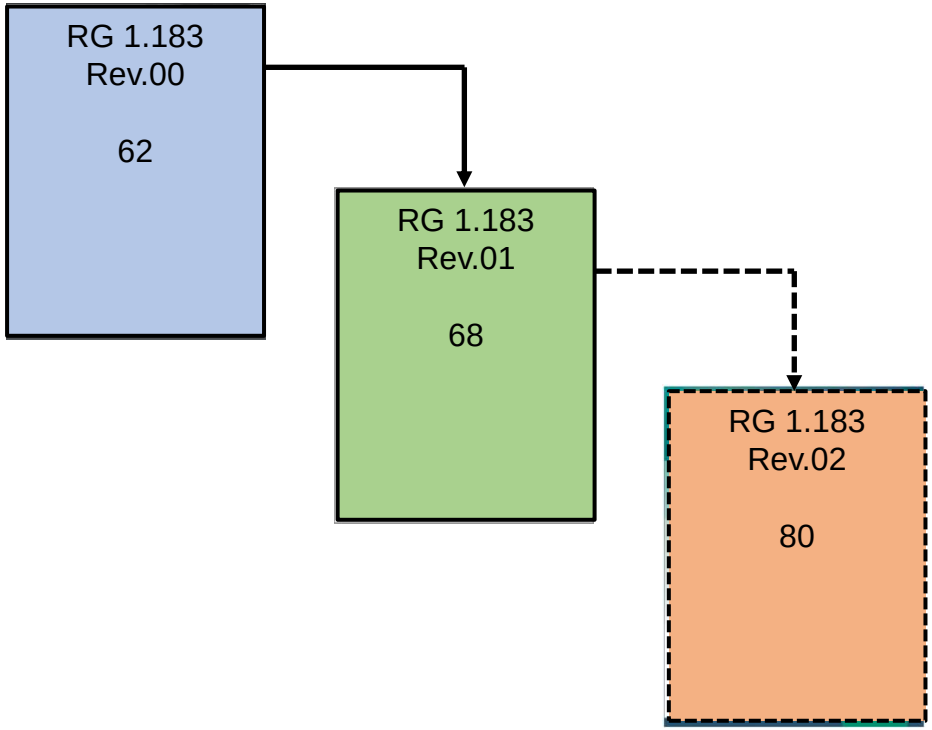
Regulatory Uncertainty – Example 1



Plant Y requests approval for a power uprate, but will maintain their existing 62 GWd/MTU burnup and 5.0 w/o enrichment limits

- All 3 revisions are applicable at current BU limit
  - Licensee able to maintain Rev.00?
  - Staff expects latest revision?
- Maintaining 3 active versions of the same guidance creates regulatory uncertainty

Regulatory Uncertainty – Example 2



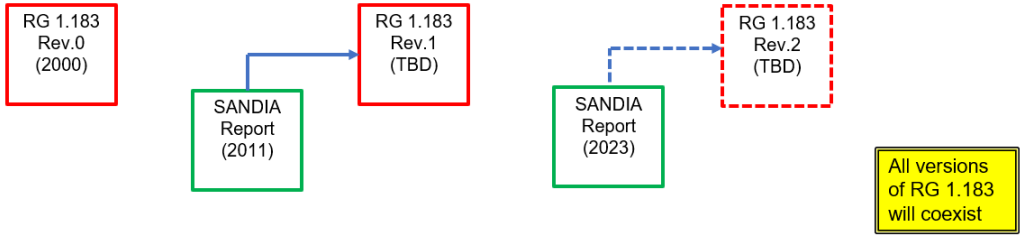
Plant Z requests approval for HBU/IE reload cores, up to 75 GWd/MTU burnup and 7.0 w/o enrichment limits

- Based upon stated range of applicability, only Rev.02 is applicable at HBU
- However, SANDIA reports demonstrate that releases are not sensitive to HBU or enrichment
- Does plant's license basis Rev.00 remains applicable?
- Maintaining 3 active versions of the same guidance creates regulatory uncertainty

Regulatory Uncertainty – Example 3

- Six months from now, Plant Z requests approval for a modest increase in allowable fuel burnup
  - Licensee applies Rev.1 (2011) release fractions within updated dose calculations

- During LAR review, staff insist that the latest MHA releases (SANDIA 2023) need be incorporated due to adequate protection



- Following guidance does not equate to regulatory stability and predictability

Backup Slides

Voluntary Guidance - Implementation

D. IMPLEMENTATION

The NRC staff may use this RG as a reference in its regulatory processes, such as licensing, inspection, or enforcement. However, the NRC staff does not intend to use the guidance in this RG to support NRC staff actions in a manner that would constitute backfitting as that term is defined in 10 CFR 50.109, "Backfitting," and as described in NRC Management Directive (MD) 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests" (Ref. 39), nor does the NRC staff intend to use the guidance to affect the issue finality of an approval under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." The staff also does not intend to use the guidance to support NRC staff actions in a manner that constitutes forward fitting as that term is defined and described in MD 8.4. If a licensee believes that the NRC is using this RG in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfitting or forward fitting appeal with the NRC in accordance with the process in MD 8.4.

One Acceptable Means to Demonstrate Compliance

Alternative Non-LOCA Releases

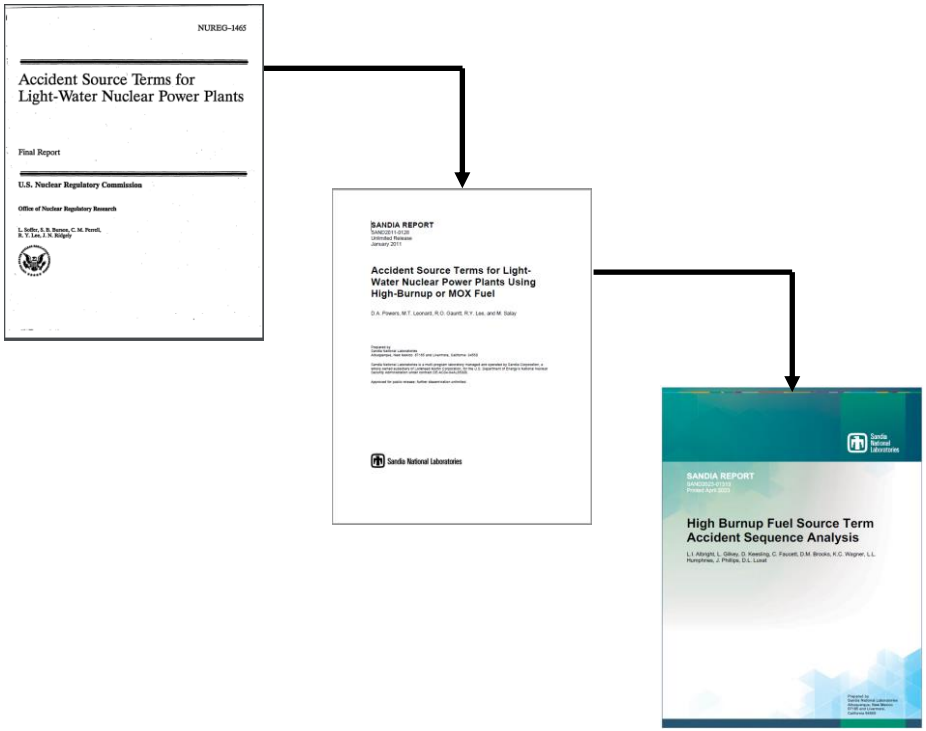
- RG 1.183 provides generic, bounding radionuclide release fractions for Non-LOCA radiological consequence assessments
  - Unlike MHA core average releases, Non-LOCA release fractions' range of applicability limited due to sensitivity to burnup, enrichment, fuel design, and operating history

- RG 1.183 also provides an acceptable analytical procedure for calculating alternative Non-LOCA radionuclide release fractions

- In the past, many licensees have successfully licensed alternative "gap fractions" to accommodate more economical fuel utilization

Minimal hurdle to develop and license alternative releases

MHA Releases



- MHA releases derived using nonparametric order statistics to develop distributions for the timing of radionuclide release during four accident phases and for release fractions of nine chemical classes of radionuclides as calculated with the MELCOR accident analysis computer code.
- Numerous MELCOR simulations involving multiple severe accident scenarios using several NSSS designs
- Extensive validation and peer review

Alternative MHA Releases

- RG 1.183 Section 2 defines attributes of an acceptable alternative AST, along with the caveat:

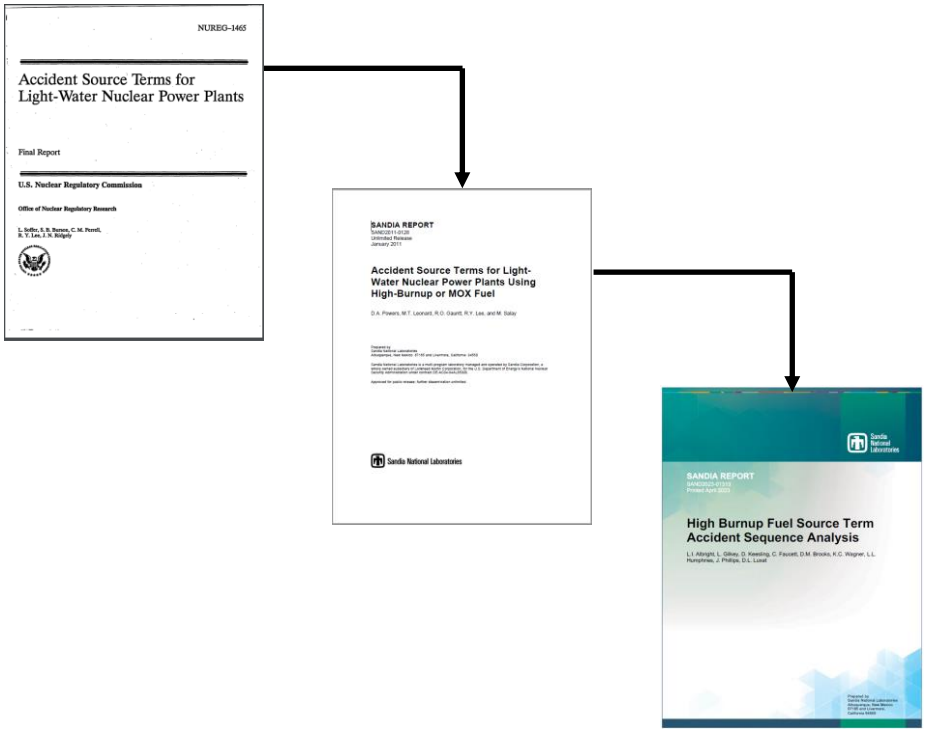
  The NRC, its contractors, various national laboratories, peer reviewers, and others expended substantial effort in performing severe accident research and in developing the source terms in Sandia National Laboratories technical reports….The NRC staff will consider applications for an AST different from that identified in this guide, although the staff does not expect to approve any MHA LOCA source term that is not of the same quality as the source terms in NUREG-1465 and SAND-2011-0128.

- Significant hurdle to develop and license alternative releases

Effectively, RG 1.183 is not <u>one</u> acceptable means, but the <u>only</u> acceptable means to satisfy regulatory requirements

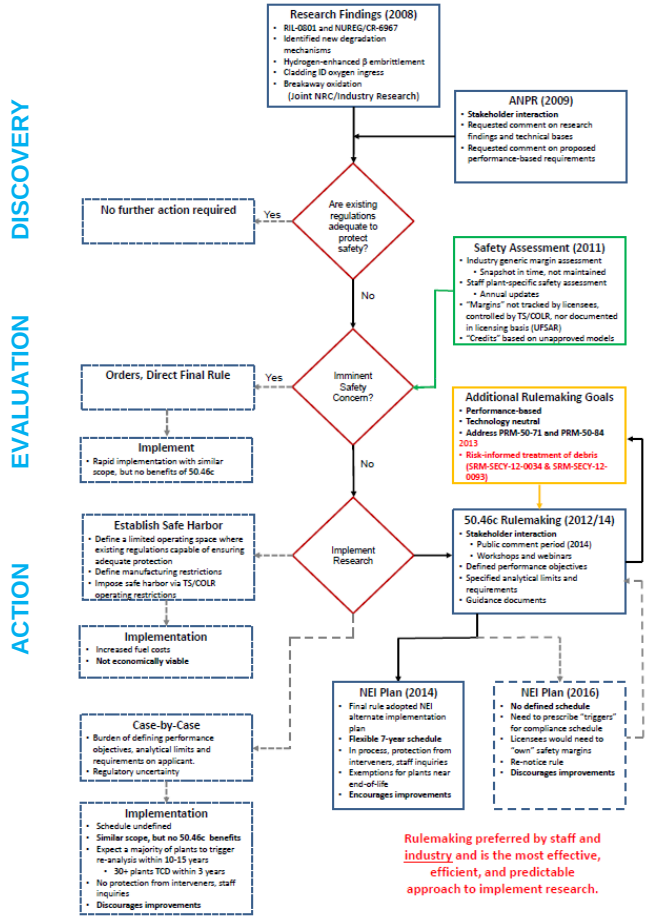
NRC Staff Response to SANDIA Research Findings

NRC's Response to New MELCOR Predictions



- At the time of their publication, the SANDIA reports represent the state-of-the-art
  - Most accurate representation of severe accident progression and releases
- Significant differences from plants' license bases
- NRC's response to these research findings should have mirrored their response to other safety-significant research findings (e.g., RIL 0401 (RIA) and RIL 0801 (LOCA))

Logical and Systematic Response



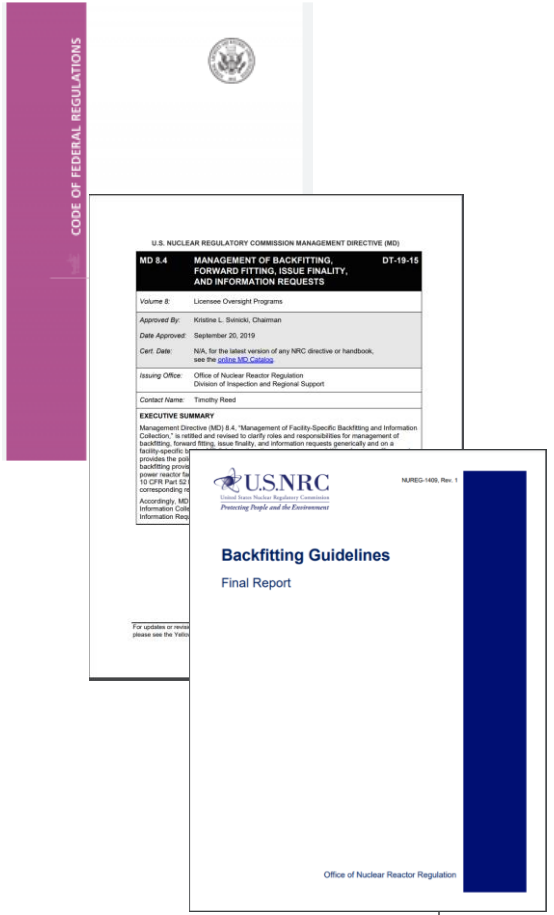
The NRC staff's response to RIL 0801 followed a logical progression:

Discovery → Evaluation → Action

The staff should follow the same approach, combined with MD 8.4 Backfit and Forward Fit requirements, in response to the SANDIA findings

Backfit and Forward Fit Policy

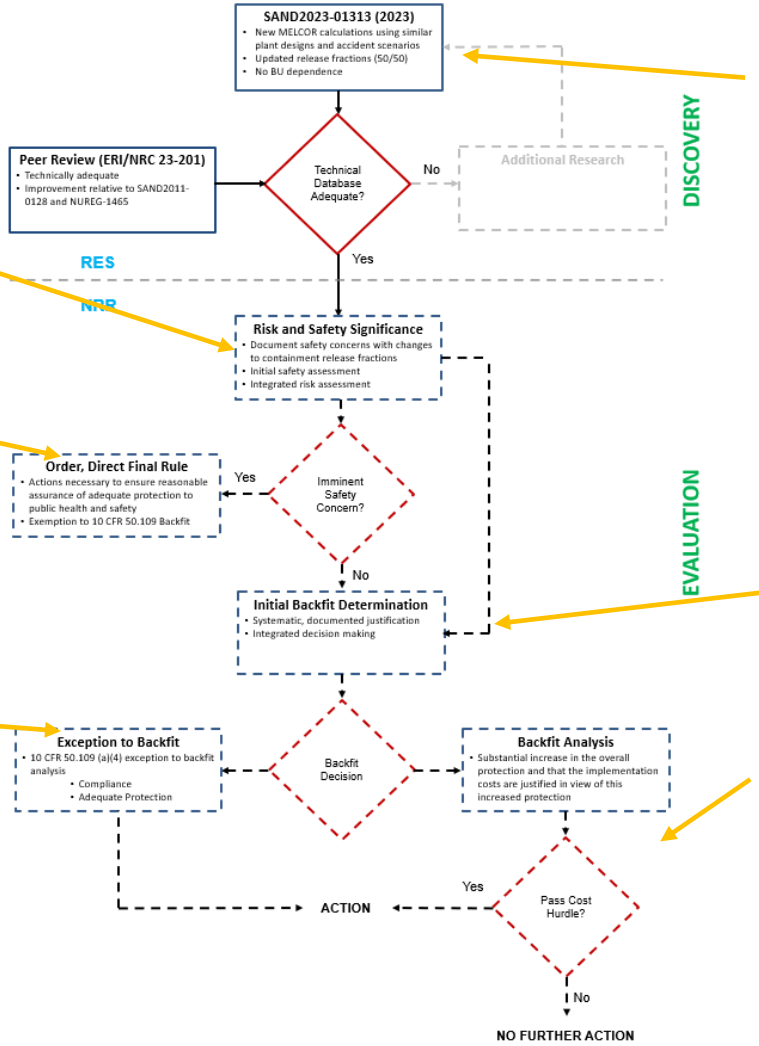
- 10 CFR 50.109, *Backfitting*, provides a codified process for evaluating the imposition of new or modified regulations, interpretations, or staff positions
- Management Directive (MD) 8.4, *Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests*, provides direction and expectations for staff
- Detailed staff guidance provided in NUREG-1409



Where is risk and safety significance assessment?

Is action needed today to ensure adequate protection?

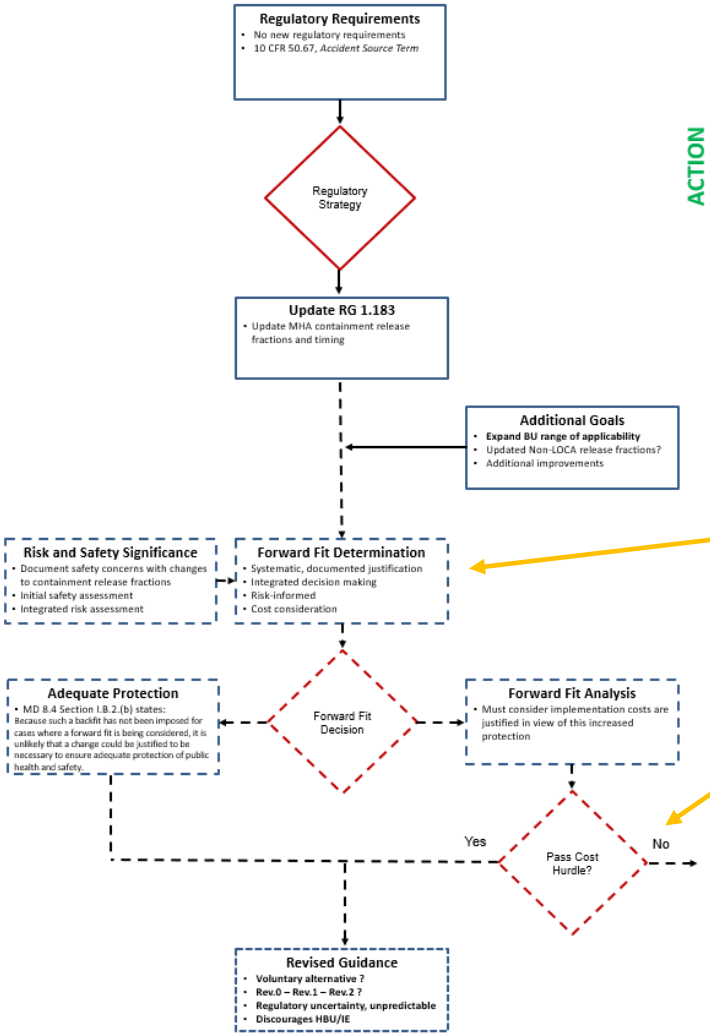
Compliance or adequate protection exception from Backfit Analysis?



Change accident sequences based on modern risk-insights?

Where is Backfit Determination?

Cost justified substantial increase in protection?



No change to regulatory requirements

Decision to update RG 1.183

Where is risk and safety significance assessment?

Where is Forward Fit Determination?

Adequate protection?

Cost justification?

Why Maintain Multiple Versions of Same Guidance?

Retiring Inactive or Revising Regulatory Guides

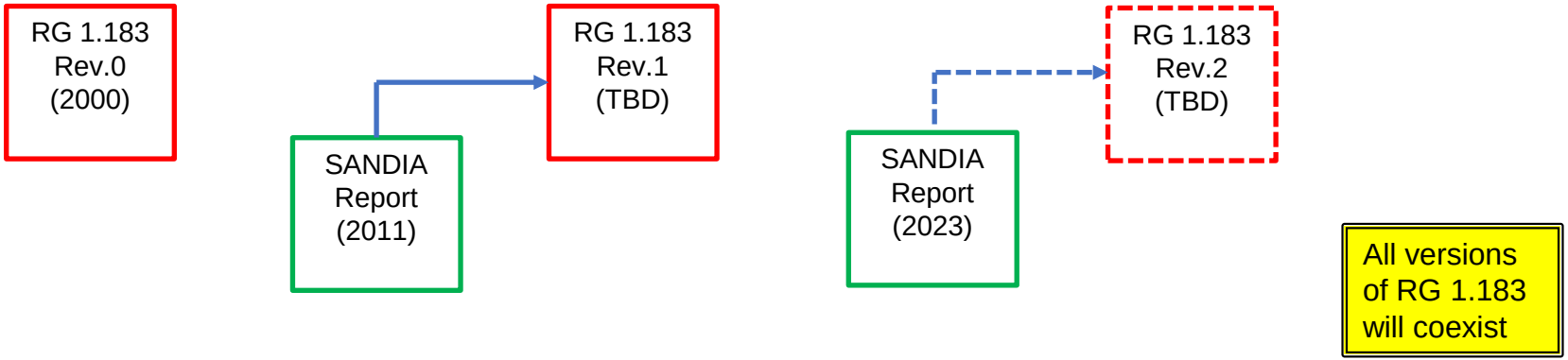
Withdrawal of a Regulatory Guide

Withdrawal of a Regulatory Guide should be thought of as the final revision of the guide. Guides are revised for a variety of reasons including changes in technology and methodology. Although a RG is withdrawn, current licensees may continue to use it, and withdrawal does not affect any existing licensees or agreements….

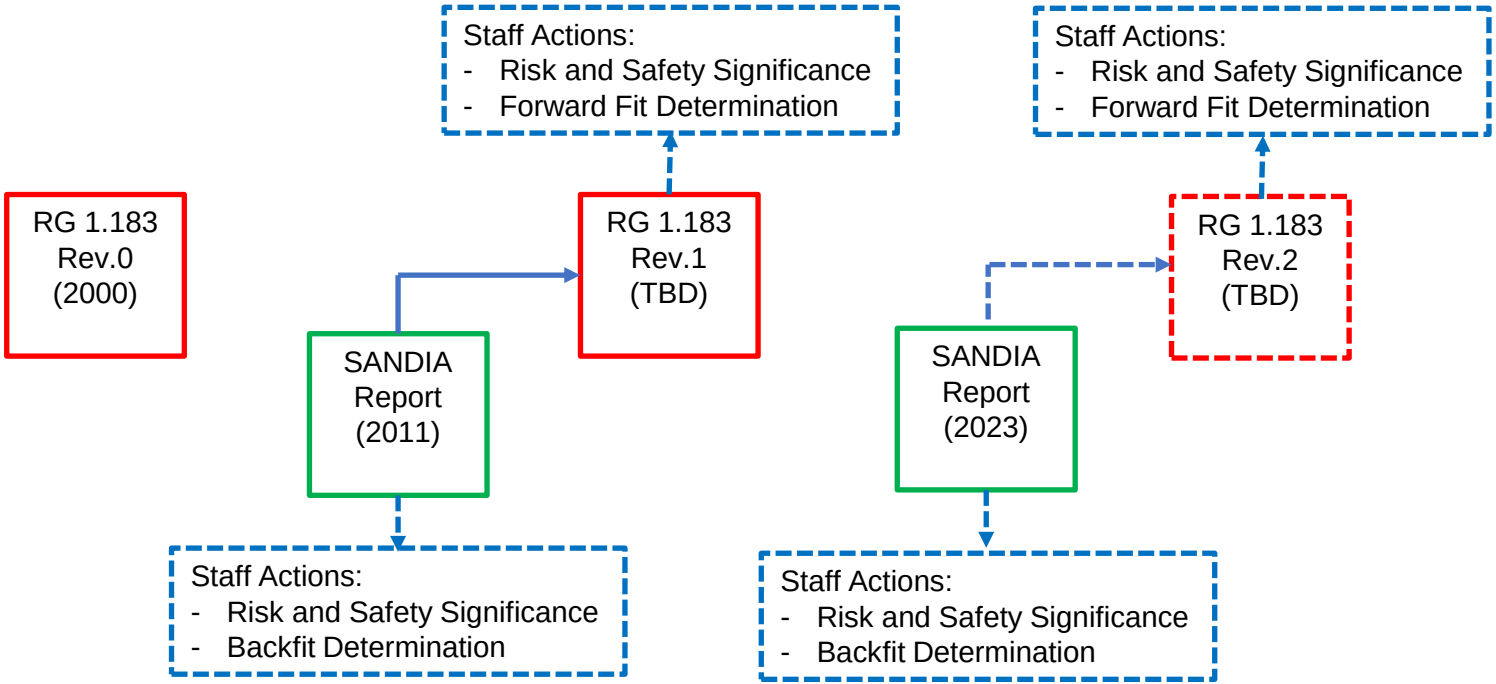
https://www.nrc.gov/reading-rm/doc-collections/reg-guides/index.html

- As described on NRC website, withdrawal is part of the RG process
- Same is true for RG revisions
- Not uncommon to retire legacy RGs
  - In 2020, RG 1.77 was retired when RG 1.236 was issued

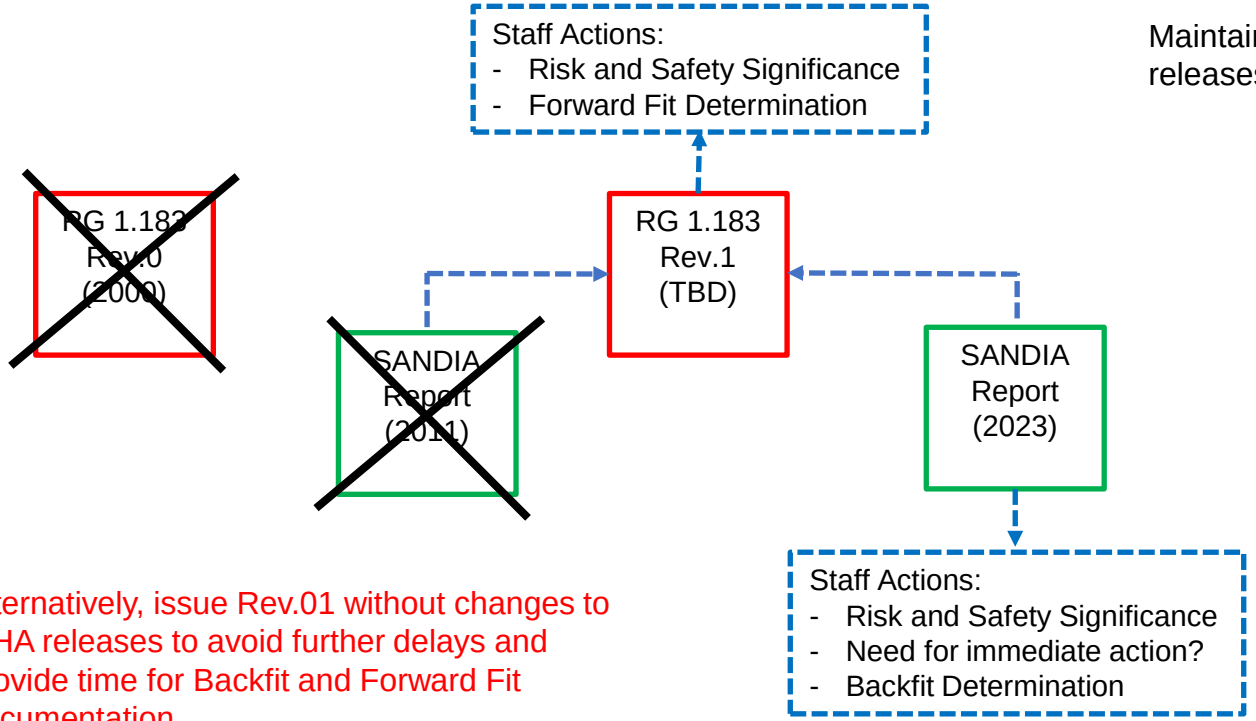
Path Forward - Status Quo



Path Forward – Required by MD 8.4



Path Forward – Recommended



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NEI Position for RG 1 .183 FC ACRS meeting:

Regulatory uncertainty surrounding the radiological source terms in RG 1.183 is a major hurdle for early adoption of ATF features. This issue needs the regulatory stability and predictability offered through RGs. Issuing RG 1.183 R1/2 is CRITICAL path to the widespread adoption of ATF/LEU+/HBU fuels in a cost-effective manner.

We recognize that this revision has been in development for over 15 years; and the NRC will soon issue RG 1.183 Rev. 1 based on SAND2011-0128. The industry supports many aspects of the proposed revision which provide updates in the understanding of accident modeling and/or relaxing constraints on inputs. These include 1) the updated model for the fuel handling accident which demonstrates that this accident actually results in much lower doses and can allow for earlier fuel movement during outages, 2) non-LOCA gap fractions which are not constrained to the burnup and linear power limits which were defined in Revision 0, and 3) a more realistic consideration of the post-accident integrity of the BWR steam line and condenser which provides a simpler approach to this credit for those BWRs that have not yet adopted it.

Other changes in RG 1.183 Revision 1 are more problematic for the industry such as the large increase in halogen releases for BWRs. These higher release fractions may require plant modifications, offsetting the benefits of ATF designs in BWRs. While many plants may struggle to meet these release fractions, the latest research in SAND2023-01313 (ML23097A087), performed for even higher burnups and increased enrichments, indicates that release fractions are only going higher in RG 1.183 Revision 2. Further, the SANDIA reports show that burnup and enrichment do not significantly influence the release fractions and therefore do not increase the safety consequences of transitioning to higher burnups. Increases in release fractions are based on changes to the MELCOR code.

Therefore, while industry supports the near-term issuance of RG 1.183 Revision 1, we want to be sure the need for an expeditious Revision 2 is also acknowledged. It is imperative that Revision 2 addresses some key issues identified in the latest SANDIA report to support BWR implementation. Results of the updated in-containment source term analysis documented in SAND2023-01313 (ML23097A087), indicate that suppression pool scrubbing is prevalent in the BWR severe accident progression and that it significantly decreases the non-noble gas airborne activity because the vast majority of the activity would be released into the suppression pool. This phenomenon, inherent to the BWR severe accident progression, needs to be incorporated in the RG 1.183 guidance either by the release fractions directly or by including an acceptable method to account for it downstream from the release fractions. Also, where appropriate, multiple layers of conservatism in assumptions need to be removed from the guidance. For example, acceptable methods for calculating aerosol deposition introduced in the proposed Revision 1 do not allow credit for main steam line deposition along with credit for calculating aerosol removal from drywell sprays. Multiple BWRs currently have credit for aerosol removal from drywell sprays as well as aerosol deposition within the main steam lines in their licensing basis. In addition, credit for aerosol impaction in the BWR MSIV leakage path should be generically approved as currently applied in some BWRs. Considering the number of BWRs currently modeling these removal mechanisms, RG 1.183 R2 should include guidance for crediting these important, mitigative features.

Without additional changes to RG 1.183, many BWR and PWR plants may not be able to implement HBU and IE due to higher source terms resulting from the updated in-containment source term analysis for High Burnup/High-Assay Low Enriched Uranium fuel (HBU/HALEU) documented in SAND2023-01313

(ML23097A087). The sequence of events considered in SAND2023-01313 have not changed since the Individual Plant Examinations (IPEs) documented in NUREG-1560 (1996) and ignore improvements in plant safety, such as B5B and FLEX implementations and updated risk insights over the last 30 years because of Fukushima learnings and risk-informed applications. The disconnect between reality and the regulation, and the disconnect between different regulatory requirements within the applicable guidance documents, lead to the incorrect determination of significant risk contributors that overly estimate conservative source terms. As a result, industry's effectiveness is severely challenged when trying to enable changes such as HBU and IE.

We also want to acknowledge the significant amount of time industry spent to provide detailed comments on this DG and that we were not provided the staff responses to our comments with enough time prior to this committee meeting for our review and understanding (provided only three days prior to this ACRS meeting). Also, attempts at collaboration during the revision process proved ineffective, resulting in no significant changes or effective engagement to resolve issues identified by the industry.

Going forward, the industry remains ready and willing to collaborate more closely with the NRC in the development of future revisions to this regulatory guide, such that it is useful and can be readily implemented to reach anticipated future operating goals. Durable guidance is needed today to facilitate the widespread implementation of ATF/LEU+/HBU and we expect early and frequent engagement with the NRC to accomplish this.