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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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RADIATION PROTECTION AND NUCLEAR MATERIALS

SUBCOMMITTEE

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WEDNESDAY

MARCH 16, 2022

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The Subcommittee met via Teleconference, at 9:30 a.m. EDT, David A. Petti, Chair, presiding.

COMMITTEE MEMBERS:

DAVID A. PETTI, Chair

RONALD G. BALLINGER, Member

VICKI M. BIER, Member

CHARLES H. BROWN, JR. Member

VESNA B. DIMITRIJEVIC, Member

GREGORY H. HALNON, Member

JOSE MARCH-LEUBA, Chairman

JOY L. REMPE, Member

ACRS CONSULTANTS:

DENNIS BLEY

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

WEIDONG WANG

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P R O C E E D I N G S

8:30 a.m.

CHAIR PETTI: (presiding) The meeting will now come to order.

This is a meeting of the Radiation Protection and Nuclear Materials Subcommittee of the Advisory Committee on Reactor Safeguards.

I'm Dave Petti, Chairman of the Subcommittee. Members in attendance are Charles Brown, Jose March-Leuba, Joy Rempe, Ron Ballinger, Vicki Bier, and Greg Halnon.

Vesna, are you online?

MEMBER DIMITRIJEVIC: Yes, I am.

CHAIR PETTI: Okay, Vesna is here. We also have our consultants, Steve Schultz, who's here with us, and I saw Dennis is online remotely.

Weidong Wang is the Designated Federal Officer for this meeting.

As posted in the agenda on the ACRS website, the topic for today is to review Draft Reg Guide DG-1389; Proposed Revision 1 to Reg Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

The Subcommittee will hear presentations by and hold discussions with the NRC staff and other

interested persons regarding this matter.

Rules for participation in all ACRS meetings, including today's, were announced in The Federal Register on June 12th, 2019.

The ACRS section of the U.S. NRC public website provides our Charter, Bylaws, agendas, Letter Reports, and full transcripts of all full and subcommittee meetings, including slides presented. The meeting notice and agenda for this meeting were posted there.

We have received a request to make an oral statement from a public member. So, that will happen at the end of our discussions here.

The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

The phone lines have been opened to all members of the public to listen in on the presentations and the committee discussion. Additionally, we have an MS Teams link available for the public.

There will be an opportunity for public comment, and we have set aside time at the conclusion of the prepared presentations and discussion for

comments from members of the public attending or listening to our meetings.

A transcript of the meeting is being kept. And it is requested that the speakers identify themselves and speak with sufficient clarity and volume, so that they can be readily heard.

Additionally, participants should mute themselves when not speaking. To mute or unmute, please push *6.

So, to start off the staff's presentation, I call on NRC management to make opening remarks.

MR. FRANOVICH: Sound check. I hope you can hear me.

CHAIR PETTI: We can.

MR. FRANOVICH: Great.

Good morning, Chairman Petti and ACRS Members.

I am Mike Franovich, and I serve as the Director of the Division of Risk Assessment in NRR.

I want to thank you for the opportunity today for the staff to share advances in updating our Regulatory Guidance, Regulatory Guide 1.183, on alternative source terms.

Based on the February 17th, 2022, ACRS meeting discussion on the integration of source term

activities, we recognize that there has been a great interest in learning more about the RG update. Since like 2020, NRR has refocused its efforts on Reg Guide 1.183 revision, or what we now call DG-1389, including forming a steering committee consisting of Division Directors in NRR and the Office of Nuclear Regulatory Research, and a staff working group.

Some of the working group members also made presentations or interacted with the Committee during the February 17th meeting on the source-term-related topics.

During the development of DG-1389, we have been fully embracing regulatory transformations and the performance-shaping factors to improve our efficiency and reliability as regulators. A key theme is that our licensing activities, other regulatory decisions, and backfit/forward-fit actions must be risk-informed. The Commission's 2019 direction reminded the staff that we are, and have been, able to use risk-informed, performance-based approaches in our work. This direction also serves as an accelerant for transformation to become a more modern and risk-informed regulator.

The 2019 Staff Requirements Memorandum, SRM, commonly referred to as the NuScale block valve

SRM, draws upon longstanding practices, in particular, a 1999 Commission paper that states succinctly, "a risk-informed, performance-based approach is an approach in which risk insights, engineering analysis, and judgment, including the principles of defense-in-depth and incorporation of safety margins and performance history, are used in decisions."

In 2019, we received additional clarifications from the Commission regarding treatment of backfits and proper treatment of forward-fits. This Commission direction serves as a regulatory stabilizer, applying the reliable principles of good regulation in our licensing activities.

Secondly, I would like to point out improved realism evaluation techniques and additional information are applied to our decisionmaking activities. The NRC's application of risk-informed decisionmaking continues to evolve, as improved realism, evaluation techniques, and additional information are applied to our decisions.

What that means to the staff, when it comes to update of our guidance for alternative source terms, is that there are tremendous opportunities for applying engineering and risk insights. This vast amount of information includes plant operating

experience and 21st century research, as well as our experiences from post-Fukushima activities. All of this effort is focused to make more realistic and, ultimately, better decisions while abiding by the Commission's backfit and forward-fit expectations.

Today, you will hear from the collective strength of a diverse team with deep experience in radiological accident analysis, reactor safety and systems, seismic engineering, accident-tolerant fuel, high burnup fuel, and risk assessment techniques and practices as well.

I do want to express my deepest appreciation to all of the staff who have contributed to this Regulatory Guide. It's quite comprehensive. There are many updates that we will be going through today.

There has been a significant amount of work done by the staff, which you will hear about in more depth -- in great depth, actually -- today in our session.

We look forward to hearing the ACRS members' views as we continue our track to collect stakeholder perspectives on this draft guidance.

At this time, I'm going to turn the meeting over to Mark. So, thank you very much.

CHAIR PETTI: Thank you.

Go ahead, Mark.

MR. BLUMBERG: Thank you for the introduction.

We are pleased to be here today to provide a presentation on a Draft Revision to Reg Guide 1.183, known as DG-1389. The names on the cover slide are the names of the presenters for today's discussion. These individuals represent the many individuals in the steering committee and the working group that developed DG-1389.

Let's get started on the presentation by moving to slide 2.

Are the slides up?

MEMBER BIER: No, we are not seeing any slides in this. There we go. Thanks.

MR. BLUMBERG: Okay. Can you go to slide 2, please?

This slide provides the --

CHAIR PETTI: Mark, hold on.

Please put it in Presentation Mode. It's awful tiny on our very big screen. Thank you.

MR. BLUMBERG: You're welcome.

This slide provides the agenda for our presentation.

First, we'll give you introductory information containing key messages for our presentation and some background information related to the development of the Reg Guide 1.183 methods.

Next, we will discuss several of the key changes proposed in DG-1389. In the presentation, we will go into more details on several of these changes. Those detailed discussions will be on proposed methods to model the MSIV leakage pathway and on a new source term for the maximum hypothetical loss of coolant accident. We will also discuss proposed new source terms for the non-LOCA accidents.

Next, we will discuss a change to the fuel-handling accident methods.

Lastly, we'll summarize some of the key takeaways from our presentation and discuss what needs to be done to complete the revise guidance, and then, provide some time for discussion and any feedback that you may have.

Please go to slide 3 now.

Before we get into our key messages, we would like to acknowledge the 17 individuals involved in the DG-1389 development. As Mike said, we coordinated the proposed changes across management and 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 those offices and the Office of General Counsel.

The steering committee and working group provide diversity and coordination among several subject matter experts within the NRC.

Let's now review our key messages for this presentation. Please go to slide 4.

The first key message is that the NRC has restarted in an effort to update Reg Guide 1.183, Revision 0. The purpose of Reg Guide 1.183 is to provide methods and assumptions acceptable to the NRC staff for showing compliance with the regulation known as 10 CFR 50.67.

As a result of a change to our Management Directive in Handbook 8.4 on Backfitting, we are proposing that Rev. 0 of Reg Guide 1.183 and Rev. 1 coexist, rather than have Rev. 1 supersede Rev. 0.

Rather than finalizing a draft revision to Reg Guide 1.183, known as DG-1199, the staff is issuing a new draft revision known as DG-1389. 1389

considers and addresses technical issues and public comments related to the issuance of DG-1199. It also incorporates new guidance based upon current issues and lessons learned during the review of license amendment requests using Reg Guide 1.183.

This restarted effort involved early and frequent outreach to external stakeholders. The dates and slides for three of these public meetings are provided in bullet 3 on this slide.

Reg Guide 1.183, Rev. 1, provides a preapproved method for modeling MSIV leakage. Use of individual regulatory positions in isolation will need additional justification.

Please go to slide 5.

Slide 5 discusses some of the objectives of the proposed revision to Reg Guide 1.183. They include the following:

Making the guidance more useful.

Incorporating relevant operating experience.

Responding to changes in the regulatory environment.

Incorporating lessons learned from previous reviews.

Providing guidance for changes in fuel

utilization in accident-tolerant designs.

And to support licensing of advanced light water designs.

And lastly, to incorporate insights from new research activities.

Please go to slide 6 now.

CHAIR PETTI: Mark?

MR. BLUMBERG: Yes?

CHAIR PETTI: Before you go on, go back.

What is it about the NuScale SRM that impacted this draft guide?

MR. BLUMBERG: If you can wait just a little while, we'll be getting --

CHAIR PETTI: No problem.

MR. BLUMBERG: -- into detail.

CHAIR PETTI: Okay. No problem.

MR. BLUMBERG: And if we don't answer your question, then we can bring it up at the end, and I'll be happy to get into those details.

DR. BLEY: And, Mark, this is Dennis Bley. Could you back up one more?

This is not a substantive issue, but in Reg Guides, until a few years ago, there used to be a section called "Regulatory Positions." You guys changed that and it has a new name. And I think,

increasingly, referring to regulatory positions within your own document that doesn't have a section named "Regulatory Positions" will become more and more confusing to people.

The first time I ran across it, it confused me because I didn't see anything labeled "Regulatory Positions." So, just a comment: instead of "positions," if you can come up with another word that fits what you're calling the section now? That's all.

MR. BLUMBERG: Okay. Thank you.

MEMBER HALNON: Yes, Mark, this is Greg Halnon.

One other question, and it's more probably editorial. The coexistence of Rev. 0/Rev. 1, is it a little unusual? Is that an artifact of the document control system, or did you consider just making it a 1.1a3a, or some other number?

MR. BLUMBERG: No, it's more of an issue that relates back to the SRM that we were talking about before, the SECY-18-0049. It's a Management Directive and Handbook on Backfitting and Finality of Information Collection.

And when we went through this revised change to the backfitting, we knew that we needed an

effective program that would ensure backfitting and forward-fitting for these facilities. And as a result of looking at the previous positions, what we're trying to do is we're saying that, rather than making major changes that would override those that are previously existing, in this particular instance, we're proposing a completely revised method.

Does that make sense?

MEMBER HALNON: Not entirely, but I'm going to let it go. I just felt like it could lead to some other confusion down the road, but I'm sure you've thought through that. But if I come up with a more coherent question that can make it clearer to me, I will later on.

MR. BLUMBERG: And we can revisit it.

MEMBER HALNON: Thank you.

MR. BLUMBERG: Thank you.

MEMBER BROWN: This is Charlie Brown.

I wanted to ask, springboarding off of Greg's comment, how do they decide what they're going to use? Somewhere you've got to be telling people which one you all accept more than one of the others.

MR. BLUMBERG: So, both provide different methods. In the instance of Rev. 1, as we'll go through in this presentation, we'll be discussing the

benefits of using Rev. 1. And so, based upon their needs -- so, say, for example, they needed a source term for expanded burnup or enrichment, they may choose to use Rev. 1 over Rev. 0 because it contains a source term that's applicable to those changes.

MEMBER BROWN: Okay. Let me just make one observation relative to how you phrased that. When you go through this, a lot of the -- I've forgotten what they're called -- the tables that show percentages of byproducts on whatever, they change. You introduced a new accident called, I guess -- what is it? -- maximum hypothetical accident. And then, you threw in -- I can't remember the paragraph that it's in, but you literally said, whatever you do when you do your analysis, do not do that. Now, you've added on and tacked on a separate part of it where it says, but if you decide to, then go ahead and do it, and you maybe can justify, whatever.

So, that's at least three somewhat major changes between the earlier version, Rev. 0, and this one. So, I'm just an electrical engineer. So, I understand parts of -- physics seems to dictate source terms, and it just seems we've got conflicting directions on what percentages of source terms and how they should be incorporated into the overall analysis.

So, I'm just echoing Greg and trying to provide a little bit more meat from my perspective, or from one Committee member's perspective, on the differences between the two. So, I just throw that out there. It's a confusion to me. And that's all I wanted to make sure you understood.

MR. BLUMBERG: Okay. Thank you.

We'll be going through more details, and hopefully, we can clear up the majority of that confusion.

DR. BLEY: Hey, Mark, Dennis Bley again.

Back to my editorial point. At places during the document you refer to Regulatory Position 3, or whatever. The Table of Contents, Section C is "Regulatory positions," the old language; Section C itself is labeled "Regulatory Guidance." So, maybe I'm not the only one confused. That's all.

MR. BLUMBERG: I can't remember why that particular change was made. It seems like there were changes in the formatting to Reg Guides.

DR. BLEY: Yes. Yes, I think that's it, because we've had several other Reg Guides come through, and that language has been changed from "Regulatory Positions" to "Regulatory Guidance." So, somewhere you've got guidance telling you how to do

that, but it's not consistently carried out.

MR. BLUMBERG: Okay. We'll take that note back to our Reg Guide people and discuss the potential for confusion with that.

DR. SCHULTZ: Mark? Mark?

MR. BLUMBERG: Yes?

DR. SCHULTZ: This is Steve Schultz.

Are you saying that, with regard to 1.183, Rev. 0, that an applicant, a licensee can utilize that to support extended burnup, increased enrichments, new fuel designs, and that would be suitable for the staff to receive such application under Rev. 0?

MR. BLUMBERG: So, Steve, the applicants are always able to propose new methods to support compliance with the regulations. Rev. 1 and Rev. 0 are standalone documents. They provide those preapproved methods. If one is to try to change those methods, then additional justification is needed.

So, with respect to Rev. 0, currently, Rev. 0 does not apply to those extended burnups. It only goes to -- there's a footnote within the regulatory guidance that goes out to, I think, 62,000 gigawatt days per metric ton uranium. The extended burnups that are being sought are beyond that. What Rev. 1 does is provides a source term that's

applicable to that. If an individual were to want to use Rev. 0 for extended burnup, then that would involve more justification, rather than just a preapproved method.

Does that make sense?

DR. SCHULTZ: That's a good clarification. Thank you.

MR. BLUMBERG: You're welcome.

Okay. So, this part of the presentation will discuss some background information.

If we can go to slide 7, please?

First, we would like to provide background on the accident analysis at the NRC. The NRC exists to protect the health and safety of the public from accidental releases of fission products. One of the ways the NRC and licensees determine what measures and barriers are needed is to perform accident dose analysis. Specifically, these analyses address the situations where we are wrong about the success of the facility's response to events or accidents.

Dose analyses provide an effective way to account for uncertainties in equipment and human performance. In particular, these analyses account for the unlikely events that involve unknown and unforeseen failure mechanisms or phenomena which are

not reflected in either the PRA or the traditional engineering analysis.

The origin of the radiological consequence analysis is based upon siting criteria contained in 10 CFR Part 100. Part 100 criteria were used to evaluate the defense-in-depth provided by the containment. The source term for these analyses were provided in a document known as Technical Information Document 14844, or TID-14844, which was incorporated into the Part 100 regulation.

As a result of the TMI accident, the NRC initiated severe accident research to learn from the accident. This research led to significant advances in the understanding of the timing, the magnitude, and the chemical form of the fission products released from severe nuclear power plant accidents.

When the revised source term was developed using that research, and the sequences from NUREG-1150, this revised source term was put into a document known as NUREG-1465. That document provides estimates of accident source terms that are more physically-based and could be applied to the design of future light water reactors, as well as current reactors. Reg Guide 1.183, Version 0, adopted the NUREG-1465 early in vessel fuel melt source term and

provided guidance for operating power reactors.

Can you go to slide 8 now, please?

Reg Guide 1.183, Revision 0, was issued 21 years ago. It provides assumptions and methods that are acceptable for performing design basis radiological analyses using an alternative source term that is an alternative to the TID-14844 source term specified in Part 100.

Because future research and changes in the plant operation could occur, it was recognized that there could be a need for a source term other than 1465. So, within Reg Guide 1.183, we also included guidance on acceptable attributes for source terms developed in the future.

Could you go to slide 9, please?

In October of 2009, the staff incorporated lessons learned from the use of Reg Guide 1.183 into a draft revision known as DG-1199. Examples of other reasons for revising Reg Guide 1.183 are also included in this slide. They'll be discussed in more detail later in our presentation.

We received 150 public comments on DG-1199. These comments were used by the staff to create the new revision known as DG-1389.

Please go to slide 10.

The next two slides will discuss several license amendments that were used to inform DG-1389.

Since the issuance of DG-1199 for public comment, we have continued to process license amendments tied to Reg Guide 1.183. Several of these license amendment requests are directly tied to the non-LOCA gap fractions in Table 3 of Reg Guide 1.183. Table 3 has limitations that are exceeded by modern fuel utilization. The license amendments request listed here reinforced the need for expanding the applicability and for providing new non-LOCA gap fractions based upon modern fuel utilization. This will be discussed in more detail later in the presentation by Paul Clifford.

If you'll go to slide 11 now, please?

In 2019, the NRC received four alternative source term amendments requesting increased allowable MSIV leakage limits. One of these license amendment requests also requested removal of their main steam leakage control system. As a result of this work, the finalization of Reg Guide 1.183 was postponed, allowing the NRC staff time to develop lessons learned from these reviews.

We have incorporated the lessons learned from these reviews into DG-1389. This slide lists

those four license amendment requests to increase their MSIV leakage.

Now I'd like to turn the presentation over to John Parillo to discuss our next slide.

MR. PARILLO: Thank you, Mark.

The thesis of the referenced differing professional opinion is that licensees should be able to credit safety-related systems to distribute the deterministic substantial fuel melt source term within the containment atmosphere at the times for which these systems are designed to achieve operability. Previous assumptions associated with the alternative source term specified that mixing of the deterministic source term should be delayed for two hours to allow for the deterministic fuel source term described in NUREG-1465 to develop.

The DPO suggests that any mechanistic explanations for how the source terms occurs, including the delayed activation of safety-related systems, or the specification of main coolant system pipe breaks, should be avoided.

The DPO recommends that the concept of the maximum hypothetical accident be reinstated. I just wanted to amplify that because I heard a comment that we have added a new accident. Actually, something old

is new again. The term "maximum hypothetical accident" in this Reg Guide is meant to describe how we will analyze this accident. It is not a new accident.

The maximum hypothetical accident, historically, is that accident whose consequences, as measured by the radiation exposure of the surrounding public would not be exceeded by any other accident whose occurrence during the lifetime of the facility would appear to be credible.

The MHA is described in several footnotes in regulation as "a major hypothetical accident resulting in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products."

The DPO recommends that there should be a clear distinction between the mechanistic design basis loss of coolant accident used to show compliance with Section 50.46 and the dose consequence analysis used to demonstrate compliance with the regulatory dose acceptance criteria. This recommendation has been incorporated into the proposed guidance by referring to the regulatory dose consequence analysis described in Appendix A of Draft Guide 1389 as the MHA LOCA.

The adoption of DG-1389 of the concepts in

this DPO resulted in the elimination of the Draft Guide 1199 assumption that the initial source of BWR MSIV leakage should be the reactor vessel's steam dome. Therefore, the source term multiplication factors endorsed in Draft Guide 1199 to ratio containment atmospheric concentrations to simulate reactor vessel steam dome concentrations have been eliminated in the proposed guidance being discussed today.

Now, if there are no questions, I will turn the presentation back to Mark to discuss our next slide.

MR. BLUMBERG: Thank you, John.

We'll now discuss the updates we've proposed in DG-1389.

If you'll go to slide 14, please?

Slide 14 provides a list of some proposed updates to Reg Guide 1.183. These are changes that were originally proposed in DG-1199. Where applicable, these changes were informed by public comments previously provided on DG-1199. I'll now go through some of these changes.

Our proposed revision updates Reg Guide 1.183 to add new guidance and remove withdrawn guidance. For example, since Reg Guide 1.183 was

issued, Reg Guide 1.194 for Meteorology was developed. Since meteorology is critical to the dose calculations, the Reg Guide 1.194 method was added in DG-1389.

As previously discussed, we will include a source term which includes guidance for modern fuel utilization. We also incorporated changes due to two regulatory issue summaries. For example RIS 2006.04 discusses 12 issues regarding experience the NRC has had with the review of ASTs. These experiences include those involving the MSIV leakage pathway and decontamination factors for the fuel-handling accident. The proposed update to Reg Guide 1.183 includes revised methods to address some of these issues.

We will also include changes in terminology that have occurred since the issuance of Reg Guide 1.183, Rev. 0. For example, the terminology "EDEX," or Effective Dose Equivalent for External Exposures, is now used, rather than "EDE," which was known as the Effective Dose Equivalent.

Next is the removal of guidance on equipment qualification from Reg Guide 1.183. This guidance is being transferred to a revision of Reg Guide 1.189 which is for the qualification of Class 1E

equipment for nuclear power plants. This change is contained in the proposed revision of Reg Guide 1.189, which is known as DG-1361. Currently, the NRC staff is addressing public comments on that draft guidance.

Lastly, there is an update that's not listed that I must mention. The NRC originally created Reg Guide 1.183 for use by existing nuclear-powered reactors to satisfy regulations under 10 CFR 50.34 and 50.67. Reg Guide 1.183, Revision 1, extends the applicability of the proposed guidance for use by advanced and passive LWR designs in satisfying the requirements for dose analysis contained in 10 CFR Parts 50 and 52 for safety and siting analysis.

Would you please now go to slide 15?

Reg Guide 1.183, Revision 0, did not include an acceptable method for aerosol deposition in the main steam lines. Some licensees proposed the use of a calculation developed by the NRC staff to review the Perry Alternative Source Term Pilot Project submitted. This document is known as AEB-98-03. Other licensees used proprietary methods.

The NRC staff saw some issues with some calculations that used AEB-87-03. As a result of these issues, the NRC staff wrote Regulatory Information Summary 2006-04 to convey what the staff

saw and needed to approve amendments that utilized AEB-89-03.

In response to the issues we saw with AEB-98-03, the NRC now has developed three deposition models for the steam line.

The first is a holistic, state-of-the-art model for modeling aerosol deposition in the steam line and in the condenser. That model is provided in the Sandia National Laboratory's Document known as SAND-2008-6601. That model was also included in DG-1199.

Two additional models were developed to address the issues the staff saw with AEB-98-03. These models are known as the Multi-Group Method and the Numerical Integration Method. These models will be discussed in more detail later in the presentation.

If you'll now go to slide 16, please?

In addition to the three new steam line deposition models, the staff is proposing clarifications to Regulatory Positions for modeling the MSIV leakage pathway. These clarifications include:

Modeling the steam lines as being intact.

Deleting the Regulatory Positions crediting the containment sprays when also crediting

aerosol removal in the steam lines.

And thirdly, no longer crediting holdup in aerosol deposition within the inboard steam line.

Will you please go to slide 17 now?

I will now discuss one of the three methods for the MSIV leakage deposition in the steam lines. That method is known as the SAND-2008-6601 Modified Method.

Please go to slide 18 now.

A state-of-the-art, well-benchmarked computer code called MELCOR was used to perform an uncertainty analysis. Within MELCOR, a code module called MAEROS, which is an acronym for Multi-Component Aerosol Time Evolution, is used to model the deposition of the aerosols throughout the MSIV leakage pathway. MAEROS models the growth in particle sizes due to agglomeration or combining of aerosols and due to condensation. It also models other removal processes such as thermophoresis, diffusiophoresis, and Brownian Motion.

The MELCOR uncertainty analysis was very similar to those currently endorsed in Reg Guide 1.183, Revision 0, for modeling containment aerosol removal by natural processes and by sprays. The Modified SAND-2008-6601 Method does not use the steam

dome concentrations as the source of the MSIV leakage for the first two hours of the accident.

And now, I'll turn the presentation to Elijah Dickson, who will discuss the two other methods for modeling MSIV leakage deposition.

MR. DICKSON: Thank you very much, Mark.

Please, the next slide. Slide 20, please.

This slide highlights our work incorporating lessons learned over the last 21 years reviewing alternative source term boiling water reactor main steam isolation valve leakage pathways. This effort, in updating 1.183, we decided to revisit the AEB-98-03 analyses, which Mark has mentioned supported one of the first approved ASTs for the Perry Pilot Plant.

The work in regards to reevaluating the AED models and methods and the Multi-Group and Numerical Integration Methods are presented in detail in the first referenced memo at the bottom of this slide.

As Mark discussed, Rev. 0 does not contain an aerosol deposition model for the main steam lines. That is a model that you could use to compute removal coefficients for the dose calculations. And what many licensees have done is to follow the initial analyses

of AEB-98-03, in conjunction with the 2006 RIS.

To address issues that were addressed in the 2006 RIS, what many licensees would do is apply more conservative settling velocity than what was done for the Perry Pilot Plant. And some licensees also developed what's termed as the 20 Group Method, which is a Monte Carlo method to integrate the sudden velocity distribution in that particular steam line.

So, to date, we've performed a cursory review of BWR SOAR. Around 25 out of 30 of these licensees have adopted some form of AED-98-03, and four licensees have adopted or utilized this 20 Group Method.

In our reevaluation of AED-98-03, we reached out to the Office of Research, looking for recommendations in how to correct the issues brought up in the 2006 RIS, as well as to assess this 20 Group Method. They performed a literature review for us and updated the technical bases for AEB-98-03, applying updated physics parameters.

We settled on recommendations that are discussed or provided in the European State-of-the Art Report on Nuclear Aerosols, or the SOAR report, which evaluated results from the PHEBUS test experiments, giving recommendations for our aerodynamic size

distributions ranging from 1 to 4 microns with a log-normal distribution.

In regards to assessing how to integrate the settling velocity distribution to compute an effective removal constant that can be used in the dose analyses, we assessed two integration methods, one being the 20 Group Method as a Monte Carlo integration, and the other being the Numerical Method.

When assessing the industry-produced 20 Group Method, we were unsure if 20 groups was enough to converge the results, and later found that you need about 2,000 groups. You need to discretize the dataset into about 2,000 groups to properly integrate the settling velocity distribution in the main steam line. And so, we're now referring to it as the Multi-Group Method.

To verify the Multi-Group Method, we developed the second method, the Numerical Integration Method, which is integration by parts, which effectively normalizes the distribution, and then, allows you to compute an effective removal constant.

There's many benefits to these updated analyses. We expect that the analyses themselves should be simplified, and we should see consistency between licensees in submitting license amendment

requests.

We are using state-of-the-art recommendations from the SOAR report. And the models themselves are really rather quite flexible. They utilize plant-specific information and could be used for small modular reactor, BWR-type designs, notwithstanding the source term distributions intended for light water reactors. The SOAR report really focused on large light water reactors, and we're uncertain if the distribution of aerosols being 1 to 4 microns, as discussed in the SOAR report, would be appropriate for BWRs of small modular design.

And in the memo referenced here at the bottom of this page, we also have some example cases in which an individual can go and rerun the calculations using the Multi-Group or this Numerical Integration Method.

And I'd like to note that the referenced values used for the example calculations themselves are not specifically being established as Regulatory Positions. They are merely example calculations. For the purpose of this Reg Guide, Rev. 0, we've established that an appropriate aerodynamic diameter particle distribution being 2 microns.

And with that, I will turn the

presentation over to --

CHAIR PETTI: So, I have a question, Ed, before you go on.

MR. DICKSON: Yes?

CHAIR PETTI: So, the Multi-Group Method or the integration by paths --

MR. DICKSON: Yes.

CHAIR PETTI: -- gives you the most detailed result. But you seem to imply at the top of this slide that there's a 20 Group Method that's still acceptable, even though it may not converge adequately?

MR. DICKSON: We've never endorsed the 20 Group Method. We've reviewed them over the years. And in updating this Reg Guide, we reviewed the 20 Group Method, and the, I would say that we made modifications and enhancements to it to make the Multi-Group Method an acceptable model for the Regulatory Guide.

CHAIR PETTI: But there have been LARs that have used the 20 Group Method.

MR. DICKSON: There have been, yes.

CHAIR PETTI: Okay. Thank you.

MR. BLUMBERG: Just I'd like to point out that, within those LARs, typically, the staff required

additional conservatisms to try to offset the issues we saw with the 20 Group Method.

CHAIR PETTI: That was my next question. Great. Thanks.

MR. BLUMBERG: Yes.

MR. DICKSON: Yes, thank you, Mark. I appreciate that.

And with that, I will turn the presentation over to Steve Jones.

MR. JONES: Yes. Good morning. I am Steve Jones from the Containment and Plant Systems Branch of NRR.

I'll be describing proposed changes to guidance for use of the power conversion system to reduce dose consequences from BWR main steam isolation valve leakage following a postulated accident.

Next slide, please.

The approach establishes a low resistance flow path to the main condenser from the main steam lines just downstream of the MSIVs. The condenser has sufficient volume to delay the leakage flow by the main steam isolation valve, or MSIV, seats to provide for holdup of radionuclides. In addition, the models provide for radionuclide deposition within the main steam lines, as Elijah and Mark discussed previously,

downstream of the MSIVs, and a specified release rate from the main condenser, once the leakage reaches that point.

This pathway was originally defined in an industry Topical Report and has been included in the existing Rev. 0 version of Reg Guide 1.183 to provide a quantitative reduction in dose consequences. Approximately half of the operating BWR fleet has adopted that BWR Owners' Group methodology.

Next slide, please.

MEMBER BROWN: Can I ask a question before you go on?

MR. JONES: Oh, sure.

MEMBER BROWN: Go backwards with the slides.

Has that always been there as a passive path, that you're just now allowing it --

MR. JONES: Originally --

MEMBER BROWN: Was it used, is what I'm trying to say, before to evaluate the accidents?

MR. JONES: Prior to the early 1990s, it was not used. The BWRs were originally designed with leakage control systems, which involved active valves to redirect the downstream portion of the main steam lines, the portion downstream of the MSIVs, to a

suction, to a fan that pushes that release onto the standby gas treatment system and out the stack.

However, those systems were found to be problematic in their ability to both handle postulated leakage amounts, and, you know, you can see, going from a 1,000-pound system to a very low pressure system like the standby gas treatment system, there can be issues with the controls and having that system function correctly. So, most plants have gone to a more passive reliance on this use of the power conversion system itself to hold up the radionuclide leakage.

And as I indicated, I guess the staff accepted the BWR Owners' group report in 1999 with a Safety Evaluation Report. I'll be getting to discussion of the conditions and limitations of that in one of the future slides.

MEMBER BROWN: Okay.

DR. SCHULTZ: Steve, this is Steve Schultz.

The earlier slide on clarifications for the leakage pathway, slide 16 says, "No credit for holdup or deposition within the inboard main steam line." Is that a new condition or does the BWR Owners' Group methodology already not take credit for

that?

MR. JONES: Well, the BWR Owners' Group methodology just deals with the leakage that has gotten past the MSIVs. A deposition inboard of the main steam isolation valves is actually inside containment, and I would have to refer to Mark or Elijah to address it, if you have any more questions about that.

MR. BLUMBERG: Sure. I can try to address that.

As far as enrollments to the same DM method, when we did the Sandia 2008-6601 Method, we found that there were expectations that there would be turbulence in that section of the piping due to the mixing that's credited up to the MSIV leakage isolation valves. And so, the methods that, at least with respect to utilization of the Stokes equation that's used in AEB-98-03, in that volume, we didn't see that it would be appropriate to credit deposition in that volume using that Stokes equation because you had forces other than buoyancy and gravity that would be present. The turbulence would potentially change the conditions by which that Stokes equation is applicable.

And then, secondly, that volume up to the

MSIVs is also part of the source term volume for containment. From a regulatory perspective, up to those MSIVs are considered part of containment, an extension of containment, because of the break that occurs in the accident. So, modeling it separately, we didn't see it would be appropriate.

So, for those two reasons, this is a potential change for some licensees, if they were to adopt this method in Rev. 1.

DR. SCHULTZ: So, that would have been a change, if you will, between the initial application and the results from the Sandia 2008 study?

MR. BLUMBERG: It's really site-specific. You would have to go back and look at the individual licensees because there were differences between applications. So, I can't really answer that for everyone, but I do know that up to those MSIVs are part of containment, and that source term is a releasant to containment. So, crediting deposition in that volume, in addition to other deposition methods played out or sprays, might not be appropriate. So, that's one of those --

DR. SCHULTZ: This may be --

MR. BLUMBERG: Go ahead.

DR. SCHULTZ: This may be a change for

some licensees?

MR. BLUMBERG: Yes, that's correct.

DR. SCHULTZ: Thank you.

MR. JONES: Okay. Slide 23, please.

Okay. As we discussed, Reg Guide 1.183, Rev. 0, actually endorses the BWR Owners' Group methodology. And a part of that includes identification of the pathway, that is, what valves would need to be operated to establish a low resistance flow path to the main condenser from downstream of the MSIVs.

And this change in Rev. 1 does not affect the identification and establishment of the pathway and its ability to direct that leakage flow to the main condenser. However, the revised guidance includes streamlined justification of seismic capacity as an alternative to the methods defined in the Safety Evaluation Report in the Limitations and Conditions section for the BWR Owners' Group Topical Report.

And this slide attempts to identify two flow paths, really, with respect to not physical flow paths, but decision points with regard to the approach. Information on the seismic capacity could still use the Reg Guide 1.183, Rev. 0, method, or leveraging risk information, the Staff's Technical

Assessment.

Next slide, please.

MEMBER BROWN: What do you mean by "seismic capacity," for the uninitiated?

MR. JONES: Okay.

MEMBER BROWN: Does that mean this other alternate path provides --

MR. JONES: Right. For the majority of BWRs, the main steam system is not safety-related downstream of the MSIVs beyond the outboard seismic restraints. It's typically augmented quality. So, better than typical power piping, but, in other words, it's been subject to generally volumetric NDE to verify that the welds are correct and that the supports and things have been installed correctly. And, in addition, it's typically designed to ASME B31.1 power piping code.

In some cases, some BWRs have upgraded piping designs that use ASME boiler and pressure vessel code for that piping. And the latest designs often include stop valves that are intermediate between the MSIVs and the turbine stop valves.

MEMBER BROWN: Thank you.

MR. JONES: Okay. Next slide, please.

Okay. The BWR Owners' Group report relied

predominantly on post-seismic event damage assessments at non-nuclear sites to assess the seismic fragility of piping and piping systems and develop a generic seismic hazard curve applicable to those systems.

The associated NRC staff Safety Evaluation, as I mentioned, included several limitations and conditions on the use of that generic information that may be unnecessarily burdensome to provide the reasonable assurance of the pathway's seismic capability. These conditions included relatively detailed walkdowns to confirm configuration and information on the piping supports.

Next slide, please.

The NRC staff has developed a Technical Assessment of seismic capability that considered engineering insights and updated probabilistic assessment of reliability following seismic events and operating experience in sites from nuclear power plant walkdowns following seismic events.

The staff has identified a median fragility for components in the pathway to the condenser of at least 0.4 g. That value exceeds the peak ground acceleration values for the safe shutdown earthquakes at most U.S. BWR sites.

The staff intends to release that

Technical Assessment as part of the Reg Guide package and solicit public comments regarding that assessment.

The use of this Regulatory Position to establish seismic capacity would involve provisions of the results of more limited assessments that could be verified through an audit or inspection, rather than providing complete evaluations as part of the application submittal.

MEMBER HALNON: Steve, this is Greg Halnon.

MR. JONES: Yes?

MEMBER HALNON: I guess the seismic aspect where, if the piping's not there, it can't retain it. But how do you account for all the myriad of potential failures a seismic event could cause, such as stream trap problems that will bypass directly to the condenser and bypass a lot of the piping that we're probably trying to take credit for to have deposition? That's just an example, but there's probably many others, in addition to the myriad of different designs on the secondary side, based on who constructed it. How do you account for all that?

MR. JONES: Well, that is part of the identification of the pathway. And actually, the predominant or the normal pathway selected, and that

is identified actually, in the BWR Owners' Group Topical, is use of the drain pathway from the main steam lines. And usually, there's a bypass valve around the drain, the moisture traps, to direct that leakage to the main condenser.

MEMBER HALNON: Okay. So, the piping that we're crediting is the earliest, quickest path to the condenser relative to the steam flow. Is that because many of the preheaters are bypassed and other things are happening on a turbine trip that prevent water induction into the turbine? So, much of that is shut down.

MR. JONES: Right.

MEMBER HALNON: So, are you telling me that it's a minimalist-type approach?

MR. JONES: Right, it's a minimalist approach, just attempting to establish a low resistance flow path to the main condenser. However, we still expect that there would be some holdup within the main steam lines themselves. It's a very limited driving head. We're talking on the order of a couple hundred cubic feet per hour of leakage flow. And there would be some cooling and possibly condensing of that leakage within the main steam lines after some period of time.

Again, I mean, looking at the release assumptions, there would be some time delay to establish this leakage path and provide that flow path directly to the main condenser.

MEMBER HALNON: Okay. The next question: how do we account for maintenance issues on the condenser itself with high-oct gases and potential small tube leaks, and stuff? Is that assumed to be at the maximum tech spec allowables, and all that stuff?

MR. JONES: The Reg Guide 1.183 provides a defined leakage rate from the main condenser. It's not intended to be leak-tight following the event.

MEMBER HALNON: Okay.

MR. BLUMBERG: This is Mark Blumberg of the staff.

In general, the way that is modeled for this particular accident is what goes in comes out.

MEMBER HALNON: Okay. So, eventually, it does get out?

MR. BLUMBERG: Right.

MEMBER HALNON: Okay. I got it.

MR. JONES: Okay. Thank you, Mark.

DR. SCHULTZ: Steve, this is Steve Schultz.

From your previous slide, note regarding

the original requirement for the investigation of the components with outside contractors, and so forth. I presume that, on this slide, the docketed information that licensees will need to provide does not include that, based upon what was presented to the Subcommittee last July and the overall evaluation of experience bases on seismic. Is that true?

MR. JONES: Right, that's correct. I'll be getting to that in a little more detail in the next slide.

DR. SCHULTZ: Oh, thank you.

MR. JONES: Thank you. Thanks.

MR. HSUEH: This is Kevin Hsueh, NRR.

So, I guess this slide, I saw there was an earlier question about using the NuScale block valve SRM. And so, this is the Technical Assessment kind of used in terms of applying the risk-informed principles. And so, this is how we are kind of following up with the design, how we apply risk-informed principles. So, this is one example.

MR. JONES: Okay. All right. Thank you, Kevin.

Getting back to this slide, the regulatory guidance establishes two paths for confirmation of seismic capacity: a deterministic dynamic analysis of

seismic response to the safe shutdown earthquake, using the code of record, which would typically be, as I mentioned, ASME B31.1.

This is not necessarily a safety-related assessment, but a dynamic assessment that the licensee references and makes available, either through audit or inspection, to verify the integrity of the piping.

Alternatively, the licensee could follow a more risk-informed evaluation method with gradation of the elements of that method, based on the site ground motion response spectra.

For the risk-informed method, the evaluation would confirm margin in the existing evaluation to the code of record. That would be more of a static evaluation. Considering insights from the individual plant examination for external events; also, verification of flexible joints at major branch connections and terminal ends to verify that there's no likely locations of rupture that would be inconsistent with the staff's probabilistic assessment of seismic fragility. And variable degrees of sampling of the pathway during walkdowns to establish construction consistent with design and adequate supports.

When the site seismic hazard exceeds the

median fragility, the guidance would call for more extensive walkdown sampling and confirmatory calculations to be completed to verify seismic integrity commensurate with the increased seismic hazard and considering the extent of quality assurance applied to the piping design and fabrication during initial construction. The intent is to provide a more streamlined and risk-informed assessment consistent with the safety importance of the holdup function.

This approach differs significantly from the Draft Interim Staff Guidance previously presented to the Subcommittee last fall. The ISG did not provide methods to change the radiological dose consequence analysis of record. However, for the change to Reg Guide 1.183 in Revision 1, the staff is seeking confidence in the reliability of the pathway to the main condenser commensurate with the dose reduction credibly provided by these components.

If there are no further questions, I will now turn the presentation over to John Parillo to address MSIV leakage rates.

CHAIR PETTI: Yes, so hold on.

MR. JONES: Okay.

CHAIR PETTI: I just had a question --

MR. JONES: Sure.

CHAIR PETTI: -- on the ISG. This is the ISG that we, as a full Committee, reviewed here a few months ago.

MR. JONES: Yes.

CHAIR PETTI: And now, you're saying there's a change to that.

MR. JONES: No, there's no -- I'm just differentiating what we're doing with Reg Guide 1.183 compared to the ISG. The ISG is more of an aid to the reviewer in assessing what level of assurance they need for some of the assumptions going into the dose consequence analysis, rather than any change to the actual analysis of record and the assumptions that go into that.

However, the Reg Guide 1.183, this revision is looking for detailed information about the design and pathway established. And as a result of having that information, actually modified the dose consequence analysis to take credit for that pathway as a means of reducing the release to offsite or control room personnel.

CHAIR PETTI: And I'm not sure I fully understood. As I recall, when we talked about the ISG, I thought we were told that that was going completely into Reg Guide 1.183 as is. But now,

you've said that there's additional information and an additional option?

MR. JONES: No, the --

CHAIR PETTI: Is that the way to think about it?

MR. JONES: The ISG is not intended to enter the Reg Guide at all. It would be eventually adopted into the Standard Review Plan, because it's really, essentially, staff guidance. But it does not affect the method that the licensees use to perform their analysis of record for the dose consequences, which is the purpose of the -- you know, alternatively, which is the purpose of the Reg Guide, is to provide guidance to the licensees when they're developing their analysis.

CHAIR PETTI: But they're not inconsistent, I guess is a better way to say it?

MR. JONES: They're different, I guess. The ISG is intended to indicate that we have some confidence that, even if a pathway is not specifically identified, the power conversion system is still going to be there. It doesn't vaporize. And therefore, it would provide some level of holdup, but it's unquantified and it doesn't -- it just is intended, as I mentioned, for staff use in assessing their

information needs regarding assumptions and other inputs to the analysis of record.

CHAIR PETTI: I guess, you know, I'm probably obtuse here. I'm just confused. We will be writing a letter on an ISG. I don't want to write a letter on an ISG that is somehow inconsistent with what the underlying philosophy and technical basis is on 1.183. That's what I'm trying to get --

MR. JONES: I guess it's the commonality is, for both, we're relying on the probabilistic seismic assessment, and that we have a lot of confidence that the power conversion system, including the steam lines, the drain piping, other equipment, will actually provide holdup volume for actual changes to the analysis of record that, in fact, credit that holdup in the main condenser -- in other words, the delay of the release, allowing more fission products to decay. If that credit is actually taken, then the licensee needs to provide sufficient information, based on the seismic hazard, to justify that they will establish a flow path, and that that would be a reliable means of directing leakage flow to the main condenser.

From the ISG perspective, what we're telling the staff is that, even though the power

conversion system is not safety-related and not seismic, there is substantial confidence that there will be surfaces available for deposition and some lower confidence pathways that eventually get to the main condenser, where other volumes that may allow for holdup of the -- that will most likely allow for holdup of a release.

And therefore, the quality or -- I'm sorry, I'm having trouble finding the right word. But, you know, their confidence in the accuracy of the assumptions need not be quite as great, since the assumption in the actual analysis of record is that there is a direct release at the MSIV.

So, does that -- is that helpful in clarifying the distinction between the ISG and the Reg Guide?

CHAIR PETTI: Yes, I understand the purpose of the two documents. I just want to assure myself that what you're telling applicants in the Reg Guide and what you're telling the staff to look for in evaluation are not inconsistent. That's all I'm asking.

MR. JONES: They are not inconsistent, but, for the Reg Guide, there's a lot of work that the applicant or a licensee would need to do to establish

that this is a viable path and that it should be credited. Our intent here is just to relax a little bit from the degree of effort that's required to establish that path compared to the BWR Owners' Group Topical Report and our associated Safety Evaluation.

CHAIR PETTI: Okay. I see a hand up. Mike?

MR. FRANOVICH: This is Mike Franovich.

I think you got the gist of what the purposes of the ISG vis the Reg Guide, and it really comes down to, essentially, if the licensee is crediting these pathways in their analysis of record to stay within the limits of the regulations, they're going to have to do a little bit more demonstration to provide us assurance that that's appropriate for crediting.

The ISG is more focused on, well, the licensees -- and I'm generalizing a little bit here -- if they are not going to rely on these pathways, they are going to do an analysis without the pathways, essentially, and we're looking at where they are in terms of dose estimates. As Steve said, physically, the condenser and piping will be there. How much more information or how should we be looking at that, when we're making an alternate decision that the analysis

that they're providing, which doesn't rely on those pathways, is within the 5-rem limit of the regulation?

So, that's the kind of gist of it. So, there's more work the licensee has to do. They wouldn't really be using the ISG for an analysis of record. They're going to be using the Reg Guide.

But I understand your point about, are these two documents, from a technical standpoint, are they compatible? I would say, yes, they are compatible. There's just some more rigor that's needed if a licensee's own analysis of record is going to depend on the pathways.

CHAIR PETTI: Okay. So, let me ask a different question. Does the Reg Guide require greater rigor than the LARs that have been accepted that have credited the condenser to date?

MR. JONES: No, the intent is to reduce the level of rigor with respect to the seismic analysis.

CHAIR PETTI: Right. Okay.

MR. JONES: There are still walkdowns involved, but they're more limited and focused on where the high risks are. And one of those is, as I pointed out, verification of flexible joints at the major branch connections, as any constrained joints

might be more subject to cracking during a seismic event to an extent, that could cause bypass of what we're really trying to accomplish here.

And then, of course, actually looking at the flow, walking down parts of at least the flow path to the main condenser to establish that it's adequately supported.

CHAIR PETTI: Thank you.

MEMBER BROWN: Yes, this is Charlie Brown again.

And the basis for allowing less rigor is?

MR. JONES: It's predominantly based on our assessment of the seismic fragility of the median pipe systems; our operating experience from a number of seismic events that have directly impacted nuclear power plants. Between North Anna, the KK -

MEMBER BROWN: Let me --

MR. JONES: I'm sorry.

MEMBER BROWN: Let me phrase that. So, you're fundamentally basing it on actual end results of seismic events where the systems have maintained their integrity?

MR. JONES: Right.

MEMBER BROWN: Did I say that right?

MR. JONES: That's correct. That's part

of it, but another part of it is, as I mentioned, the code of record used to design the piping and augmented QA applied to the fabrication and construction. There's a lot of elements that address that. But, in total, we're predominantly looking at a passive system that has demonstrated a high degree of reliability over the life of these units.

MEMBER BROWN: But you're expanding it beyond the exact plants that actually experienced the seismic events, based on an assessment that their design bases are similar, and therefore, it should be acceptable?

MR. JONES: Right. That's correct.

MEMBER HALNON: So, just carrying on with what Charlie -- something struck me. This is Greg.

Looking at a post-trip situation with a lot of steam flow going to the condenser, there have been some events where flow-accelerated corrosion has reared its ugly head and has had significant leaks. How do things like that, where there's potential degradation on the secondary side that's not necessarily picked up -- is that in any way looked at it? Or is that part of the walkdown process? In programmatic, I guess, from the standpoint of you're looking at their programs, making sure they're up to

snuff.

MR. JONES: Well, I guess, from the standpoint of flow-accelerated erosion, we don't expect that in these drain lines that flow directly to the main condenser.

MEMBER HALNON: But, again, that's where they have manifested themselves in time. I mean, some of them have done it on the reheat system, but there have been -- I believe the FAC programs cover those volumes, don't they?

MR. JONES: Yes, I believe the erosion/corrosion program does apply, but I'm not actually familiar with any that have actually occurred in these small -- we're talking very small-diameter drain lines, like 1-inch or so.

MEMBER HALNON: Yes, I agree. I think it's larger lines that the erosion/corrosion volumes look at, but they do look at areas.

MR. JONES: Right.

MEMBER HALNON: I was thinking more of the margin. I mean, we talk about walkdowns. You can be walking down a secondary system and everything looks good. But if there's internal erosion or corrosion, that you're not going to see it, and a seismic event would normally not affect the high-integrity pipe, it

may cause a leak. I'm just wondering, is that even discussed in any of the review?

MR. JONES: No, we have not addressed that specifically. We are looking for margin in the piping design. These pipelines are, you know, generally, as I mentioned, small diameter.

My familiarity with the operating experience is that erosion/corrosion is generally are in larger-diameter pipes that experience two-phase flow. We don't expect any flow, you know, any phase change to be occurring in this piping. And, in fact, normally, they're isolated on a number of these plants.

MEMBER HALNON: Well, there will be a phase change somewhere, whether it's at the very --

MR. JONES: Well, I mean, we're looking at establishing this an hour or so after the event. So, it's not an --

MEMBER HALNON: Much lower pressure.

MR. JONES: Right.

MEMBER HALNON: Okay.

MR. JONES: And it should be closed. The piping should have experienced an hour or so of cooldown, and those would be the conditions. Now we're more cooled-down and we'll be looking at that

1 particular plant condition.

2 MEMBER HALNON: Okay. Well, weakened pipe

3 would occur when it's online, and then, manifest

4 itself in a seismic event later on. But, yes, I get

5 the lower pressure. And for some reason, the 3-inch

6 line hits my mind as relative to the erosion/corrosion

7 program. So, we may be talking different lines, but

8 I'll look it up, and if I have another question, I'll

9 get back to you.

10 MR. JONES: Okay. Thank you.

11 MEMBER BROWN: It's Charlie Brown again.

12 Is the basis for this, the basis for the

13 less rigor, is that documented in either the Reg Guide

14 or the ISG, or some other supporting document?

15 MR. JONES: Yes. There's a --

16 MEMBER BROWN: There's written

17 documentation?

18 MR. JONES: There's a Technical

19 Assessment. There is a Technical Assessment that will

20 be issued for public comment in conjunction with the

21 regulatory guide.

22 MEMBER BROWN: Which explains why less

23 vigor is satisfactory?

24 MR. JONES: Yes.

25 MEMBER BROWN: Okay.

MR. JONES: And it predominantly relates to those three factors, or three major factors really: the margin and the code of record applied to the main steam system; the seismic fragility evaluation that's been done, and operating experience from other plants. I'm sorry, I should have said four. And the walkdowns that help establish the integrity of the system.

MEMBER BROWN: Okay. Thank you.

MR. JONES: Thanks.

MEMBER BIER: I have some followup questions or comments. Vicki here.

First, I think part of the confusion or angst about this is just the choice of words; that if you say, "less burden," everybody might say, okay, fine. If you say, "less rigor," everybody gets worried about, is this going to be less accurate as well? So, that's just kind of a suggestion.

But the other thing that I would worry about, again, without having seen all the analyses, and whatever, is evidence from a small number of earthquakes from a Bayesian perspective may be very weak evidence about whether you could expect those successes to be replicated in future earthquakes.

It's one thing if you go in with an assumption that an earthquake is guaranteed to cause

problem, and then, you see a couple of events where nothing terrible happened. You could say, okay, it's obviously not guaranteed; the probability is a lot less.

But if you go with in a perception that, oh, maybe half the earthquakes are going to have big problems, and then, you see two or three where it didn't happen, it's not clear that, just from a statistical perspective, you learn much.

Again, you know, there may be a lot of other analysis that supports that, but, you know, I mean, I have not seen it and may not be qualified to comment on it. But, statistically, two or three successes seems like pretty weak evidence to me.

MR. JONES: Okay. Dr. Vasavada can address your questions. I believe he has his hand up.

DR. VASAVADA: Yes, thanks, Steve.

This is Shilp Vasavada. I'm at the divisional risk assessment in NRR.

To address that question, I'll say that the operating experience and the walkdowns for the three plants that Steve mentioned, or especially two, North Anna and Kashiwazaki Kariwa, they were one of the important, not the only places, for whatever is being proposed in the new provision.

We also had a, and the Technical Assessment is that the fragility evaluation that we looked at, actually, fragility data for bolted pipes and other confidence, which will be part of this -- but from the 1990s all the way to the current contemporary seismic PRAs that were developed post-Fukushima, that is also a central piece of this particular evaluation. So, it's not just about dumped; it's also a seismic fragility evaluation.

And based on that, what was determined was a lower bound in comparison to multiple failure modes. It's even less, if you can think about it, than the offsite power facility, which is expected to fail, offsite power that's expected to fail for seismic.

So, it's not just one piece, the walkdowns. It supports the fragility assessment, which is the central piece, and the Technical Assessment carries all of that information.

Thank you.

MEMBER BIER: Thank you.

MR. JONES: Okay. If there are no further questions, I'll pass it on to John Parillo to address the MSIV leakage rates.

MR. PARILLO: Thank you, Steve.

As discussed, the Draft Guide contains

three BWR MSIV leakage deposition models, as well as a streamlined procedure to allow credit for an intact power conversion system proving a pathway to the condenser.

The staff's intention is that the revised guidance will provide licensees with a methodology that will have a more predictable regulatory outcome. Currently, the two licensees with the highest total MSIV leakage values in the BWR fleet, 400 and 350 standard cubic feet per hour, credit a pathway to the condenser in their dose consequence analysis of record. However, there are several licensees with total MSIV leakage values exceeding 200 standard cubic feet per hour that do not credit a pathway to the condenser.

The NRC staff is concerned that providing a streamlined procedure to allow credit for a pathway to the condenser may result in licensees with relatively high MSIV leakage values submitting license amendments requesting to credit a pathway to the condenser, in conjunction with proposed MSIV leakage values exceeding the current fleet maximum values.

The NRC's staff's position is that it is not unreasonable to encourage BWR licensees to limit their maximum allowable MSIV leakage, since these

valves are, in essence, containment isolation valves. Leakage values more than 200 standard cubic feet per hour per valve may be an indication of a substantial valve defect, which should be evaluated and corrected.

A recent review of the current maximum total allowable MSIV leakage values relative to the maximum-per-valve values indicates a fleetwide average ratio approximately two. That is, typically, the total allowable MSIV leakage is a factor of two higher than the maximum-per-valve value.

In addition, the total MSIV leakage value of 400 standard cubic feet per hour approximates the total containment leakage for all other containment isolation valves.

Finally, these proposed maximum MSIV leakage, MSIV limitations, should be viewed in relationship to the original per-valve limit of 11.5 standard cubic feet per hour.

If there are no questions, I'll turn the presentation over --

CHAIR PETTI: Sorry. John, just one question.

MR. PARILLO: Sure.

CHAIR PETTI: Does any of this -- as I recall, when we heard about the topic through the ISG,

there was also this discussion of, you know, the worker dose that gets received in trying to make sure you meet these leakage valves, and this idea that workers were taking dose when there was sort of a safety margin because you could trade off of the two.

Is that still part of the thinking here? Because I haven't heard anybody talk about that aspect.

MR. PARILLO: Yes, that has always been a consideration in balancing the risks of a hypothetical accident versus the actual dose that is incurred in maintenance of these valves. Yes, we didn't mention that, but that is always a consideration for any of the license amendment requests for increasing MSIV leakage.

CHAIR PETTI: Okay. Thank you.

DR. SCHULTZ: John, this is Steve Schultz.

The second bullet that you have talks about the maximum of 400 standard cubic feet per hour or below. But, then, the next sentence says, "Higher values will be considered on a case-by-case basis."

You seem in your presentation to have suggested that, you know, we're really going to try to hold to 400. Where does this come out? Is it, in fact, true that higher values will be considered on a

case-by-case basis?

MR. PARILLO: Well, Steve, you know, we kind of struggle with this. We don't want to make this a hard-and-fast rule. We don't have that capability. But we just wanted to convey to licensees that, should they propose leakages higher than what is currently the maximum in the fleet, that they should expect to receive a high level of scrutiny. I'm not sure how else I could explain that, but --

DR. SCHULTZ: I would have used the same words.

The next question I had -- thank you, John -- the next question is, and it goes back to our original discussion on Rev. 0 versus Rev. 1, this seems like an area where a BWR licensee might want to use this particular segment of Rev. 1 to perform their evaluation. Is that something that could be carved out of Rev. 1 and applied to a licensee who's using Rev. 0 for their other evaluations?

MR. PARILLO: Well, I would probably want to defer that to Mark.

But my sense of it is that that would be an acceptable approach, but Mark may have some additional concerns on that. That's getting back to this idea of having both Rev. 0 and Rev. 1 coexisting.

DR. SCHULTZ: Yes, that's why I asked the question. Thank you.

MR. BLUMBERG: I'm sorry, would you mind repeating the question again? I apologize, I didn't catch it all.

DR. SCHULTZ: Thank you, Mark.

The question is, could a licensee who's using Rev. 0 take advantage of BWR MSIV leakage value reevaluation under the guidance of Rev. 1 without changing their overall evaluations of other non-LOCA events?

MR. BLUMBERG: So, once again, all the different permutations that could occur, we have not gone through all those scenarios. What we have gone through is we've created a guidance that includes a methodology that we find acceptable for meeting the regulations. Like with any methodology, if a licensee proposes differences to that, then they have to provide additional justification why they think that that continues to meet the regulations.

So, I don't know that on the fly I can give you an answer, but I can just tell you that, if you go outside the method, then you've got more work to do.

DR. SCHULTZ: Okay. Thank you.

MR. BLUMBERG: You're welcome.

So, I don't know; on the schedule right now, it shows 10 o'clock is a break. We are at the slides that we expected to get to at that point. I don't know if you would consider a break or you want us to continue on.

CHAIR PETTI: Yes. No, I had the same thought because we're going to transition a little to talk about other aspects of the Reg Guide.

So, why don't we take a break for 15 minutes and be back at 10:25 Eastern time?

(Whereupon, at 10:07 a.m., the foregoing matter went off the record and went back on the record at 10:25 a.m.)

CHAIR PETTI: Okay. Hopefully, everyone is back from our break.

Mark, let's continue.

MR. BLUMBERG: Okay. We're going to be starting with Dr. Elijah Dickson.

Elijah?

MR. DICKSON: Thank you, Mark.

I'd like to preface with slide 28 and 29.

We're talking about the MHA LOCA source term here for ATF high burnup and increased enrichments. That is the source term to containment.

Can you please move on to slide 29, please?

This slide highlights our work in updating Reg Guide 1.183 to meet, in part, the Nuclear Energy Innovation and Modernization Act, or NEIMA, that requires the NRC to be prepared to license ATF-type designs. To do so, the staff has identified several improvements to Rev. 0 that can benefit from off-the-shelf research and models which expand and encompass near-term AFT designs. Those would be chromium-built and chromium-cladded fuel designs with burnups up to 68 gigawatt days per metric ton uranium peak rod average and enrichments up to 8 percent.

Since NUREG-1465, there have been substantial improvements in the computational resources and data available to predict radionuclide release and behavior during severe accidents. And the intention of Rev. 1 is to cite the source term recommendations from the Sandia National Lab report, SAND-2011-0128, which is patterned after NUREG-1465, which was published in 1995, and forms the technical bases for Rev. 0.

This is a well-documented analysis with two additional supporting technical documents for BWRs and PWRs, respectively. It has been thoroughly peer-

reviewed. In short, it's a well-analyzed and vetted update to the accident source term.

We find that, for the MHA LOCA source term, fission product inventory releases to the containment. They are essentially unchanged due to increases in burnup and enrichment. However, an important distinction to make is that there are variations between the two source terms -- that is, NUREG-1465 and the 2011 Sandia work -- in regards to fission product fractional releases and the timing. These changes are due to differences in the accident scenarios, as modeled in the MELCOR code, as well as modeling improvements.

For instance, we have increased model fidelity. That is, we can model the core geometry much more finer. There's state-of-the-art severe accident modeling practices that we've learned since NUREG-1465; specifically, the work that was done in SOARCA. And we have updated test data from the PHEBUS fission product terminal program, as well as separate effects tests. All of this leads to a better understanding of severe accident progression, which is captured in this update to Reg Guide 1.183.

In addition to the work done by Sandia National Lab, we did some research with the Office of

Research in regards to assessing fragmentation, relocation, and dispersal for this in the containment source term, focusing on the particular aspects of radionuclide released fractions and timings. It is expected that, for FFRD, it's expected to have limited impact on the MHA LOCA source term, which is a result of a substantial fuel melt of the core. This work can be found in the memo describing this research, the second memo at the bottom of this page.

We do, also, have ongoing research being executed right now with Sandia National Lab to update this 2011 Sandia accident source term analysis to accommodate higher burnups and increased enrichments. And this work is ongoing right now, and it's not expected to be fully completed before the issuance of Reg Guide 1.183, Rev. 1. So, this would initiate some work in regards to updating it again to a possible Rev. 2 to incorporate this work probably several years down the line.

And with that, if there's no questions, I'd move the presentation on to Dr. Paul Clifford.

DR. SCHULTZ: Elijah, this is just a question about terminology -- well, not terminology, but what you've chosen in the first sub-bullet. You've indicated a fuel burnup of 50 megawatt metric

tons, gigawatt days per metric ton, for a maximum assembly. Traditionally, the focus is on the peak rod average. And I'm just curious as to why you've incorporated a maximum assembly value. Is that coming out of some of the research or something like that? Because I'm not sure it's generically applicable.

MR. DICKSON: Yes, I would have to pass that question on to the Office of Research. James Corson or Hossein Esmaili are on the line. Perhaps they can answer your question.

DR. SCHULTZ: I just think it raises the issue of what limit should we be thinking about? Assembly average now, if we get assembly average, or a rod average?

MR. DICKSON: With my experience with the vendors, they use these interchangeably. Some use peak rod average; some use max assembly.

DR. SCHULTZ: In their licensing documents?

MR. DICKSON: Yes, discussions with the vendors who develop the fuel, yes.

MR. ESMAILI: James, are you on the line?

MR. BLUMBERG: So, I can maybe help some. This is Mark Blumberg from the staff.

In the regulatory guidance, we cited as

peak average.

DR. SCHULTZ: Peak rod average rod --

MR. BLUMBERG: Sixty-eight gigawatt days per metric ton uranium peak rod average.

MR. CORSON: Yes, this is James Corson from the Office of Research.

I think this might just be a typo on the slide. Because 59 gigawatt days max assembly average burnup, that was the assembly average burnup from the Sandia SAND-2011-0128 calculations.

DR. SCHULTZ: Okay.

MR. CORSON: And that actually corresponded to a peak rod average of 62 in those particular calculations, but --

DR. SCHULTZ: That's what I would have thought.

MR. CORSON: Yes, yes. So, the memo, though, that is referenced here does discuss why like 68 peak rod average is appropriate, even though the calculations only went to 62.

DR. SCHULTZ: Okay. Thank you.

MR. CLIFFORD: Okay. If there's no further questions, my name is Paul Clifford. I am the Senior Technical Advisor for Reactor Fuels in the Division of Safety Systems.

And we're kind of changing gears now. And now, we will talk about radiological release fractions to be used in predicting doses for the non-LOCA design basis accidents. And these consist of two components. There is a steady-state component, which is the amount of each radionuclide released from the pellet into the void space of the rod during normal operation as a result of temperature-driven diffusion. And then, there is, for certain types of accidents, there is a transient release, almost a burst release, as a result of fuel fragmentation.

So, in my presentation, I'll be talking about several updates to the guidance relative to Rev. 0, which was, I believe, released in the year 2000.

First, we're expanding the applicability of the guidance to support modern fuel utilization and fuel rod burnups up to 68 gigawatt days per metric ton uranium rod average burnup.

We're providing separate BWR and PWR steady-state releases, and you'll see why that's important later on the presentation.

We're introducing a new component, and that is the transient fission gas release, which is based upon, as I mentioned earlier, not like the slow diffusion base release, but, really, a burst release

as a result of fuel fragmentation under prompt power increasing accidents, like control rod ejection or control blade drop.

And finally, we will talk about an acceptable analytical procedure that's provided in the guidance which could be used by the licensees to calculate plant-specific steady-state release fractions.

So, moving on to slide 32, this figure depicts the expanded range of applicability. The Y-axis is rod average power for the peak rod, and rod average burnup on the X-axis. You'll see the red line is indicative of what you would find, if you looked at, say, a BWR corroborating limits report for the TMOL. We've taken the general trends, where you see a knee relative early, towards the end of the first cycle, followed by a second inflection point latter on in rod life. We've expanded those both out in burnup and up in power to provide flexibility for fuel managers.

The same with the PWR, you'll see the blue line here, which goes out to about 35 gigawatt days at 12.2 kilowatts a foot, and then, drops down to 68, which is our target burnup here.

Now, how does this compare to Rev. 0? So,

in Rev. 0, there was a footnote, footnote 11, I believe, which provided a limitation on the applicability of the Table 3 non-LOCA steady-state release fractions. And it stated that they were acceptable, provided your rod power was below 6.3 kilowatts a foot beyond 54 gigawatt days. So, that limitation is kind of shown in this blue box. You have to be below this power burnup envelope when you're in just this limited range of rod life.

As we'll get into in the next slides, you'll see that the releases for different radionuclides are driven at different times in life. So, you know, providing a small window like this is not adequate to really, truly define the range of applicability.

And what was mentioned earlier by Mark, one of the first slides in this overall presentation, was that we've been busy issuing or reviewing and approving license amendment requests by various utilities which find themselves outside of this blue box. In other words, they're operating above the 6.3 at this high burnup region.

So, we've issued -- they've come in with license amendment requests with unique methods for calculating release fractions, and we've reviewed and

approved these. So, there was definitely a need to expand both upwards and outwards of these lines to provide flexibility to support modern fuel utilization.

So, I ran this code called FAST. FAST is the NRC's confirmatory fuel rod thermal-mechanical model. It superseded FRAPCON and FRAPTRAN that many of you may be familiar with. It represents the state-of-the-art modeling of the fuel's response during normal operation and has transic capabilities built into it also. And it's well-documented and well-validated against the latest and greatest empirical database.

The FAST code was also updated to include the ANS-5.4 standard release model. The ANS model is used to predict releases of the short-lived radionuclides. And we'll get into that in the next slide.

So, using this updated version of FAST, we ran sensitivity cases to determine what's an appropriate set of tables and release fractions to provide the industry for use.

Unlike what we talked about earlier, which was kind of a deterministic fuel melt-driven source term that's used for MHI, these steady-state releases

are very sensitive to rod power and fuel rod design. So, changes in either of these inputs makes a significant difference on how much of each of the radionuclides are released at different times in life.

And so, in order to provide bounding values, we had to identify which fuel designs were the most limiting, and then, separately, a develop a bounding power history that would, then, be useful to the industry. In other words, we don't want to provide a rod power envelope which licensees couldn't use because they're outside of it. So, we tried to bound both, but with being sensitive to the fact that we can't be overly conservative in our rod power history; otherwise, we're going to put very large release fractions. And if we do the opposite, then they're not as useful.

So, the most limiting fuel designs are the BWR 10x10 lattice-type and the PWR 14x14 lattice design. In general, though, the larger-diameter fuel rods at a given power level have higher operating temperatures. Higher operating temperatures means more efficient gas release from the pellet into the rod plenum.

So, there were several competing effects. As I mentioned, by expanding the operating domain to

provide flexibility for fuel management, that promotes higher releases.

Also, we applied modeling uncertainties, because there is a large scatter in data. When we validate these models, we also quantify the uncertainties in the models' capabilities of predicting things like fuel temperature and fission gas release. So, we want to provide something -- when you're doing a safety analysis, especially a dose calculation to the public, you want to have something that accounts for those modeling uncertainties. So, when you apply the modeling uncertainties, that promotes even higher releases.

But, to offset that, the application of the new ANS Standard Release Model provided a lot of benefit relative to the model that was used for the Rev. 0 of the Reg Guide 1.183, which goes back to the 1982 standard. Application of this latest and greatest model is also a significant decrease in iodine-131 and other short-lived radionuclides.

So, let's get into some results of the calculations.

DR. SCHULTZ: Excuse me, Paul.

MR. CLIFFORD: Yes.

DR. SCHULTZ: Before you leave that slide,

on the previous slide, I believe slide 32, you presented the power versus burnup for those limiting curves.

MR. CLIFFORD: Uh-hum.

DR. SCHULTZ: And do those correspond to what you would expect to see for these limiting fuel designs that you've evaluated? Do you see what I'm saying?

MR. CLIFFORD: Yes, yes. No, no, I understand. Right.

So, generally, the larger fuel rods, there's less linear feet of fuel, because they're a larger diameter, and you generally have higher operating kilowatts per foot. Of course, it depends on the power density of the reactor and how many assemblies there are, but, generally, that's what happens.

So, when you see a plant go from a 10x10 to a configuration to, say, an 11x11, BWR, you've added 21 percent more linear feet of fuel. So, your linear heat rate goes down. So, you actually would move away from these. You would be closer to -- you know, you'd be further down here.

So, yes, these power operating histories are representative of those two fuel designs.

DR. SCHULTZ: Okay. Thank you.

MR. CLIFFORD: Yes.

That's a problem with the animation. Okay.

BWR, let's talk about some results. So, there's different trends in the release of different radionuclides which are important. So, here in this figure on the top left -- we're at slide 34 now -- here, we show this red line is what we just talked about. This is the bounding rod power history with the two knees in the curve for the BWRs. And that's shown here.

And then, on this axis, we have the predicted best-estimate release fraction for krypton-85. Krypton-85 is a long-lived noble; half life of just over 10 years.

Now what's imported about that is, if you're generating fission gas back here at the beginning of life, and it's working its way through the matrix to the grain boundaries, from the grain boundaries out into the void space, by the time you get further on in life, up here, that's still active krypton-85. It hasn't gone anywhere. It hasn't decayed. So, as a result, you're continuously accumulating more and more fission gas as it gets

released.

So, as you would expect, later on in life, is when you get your maximum concentration of long-lived radionuclides within the void volume. And so, that's the trend you're seeing here.

Now, you may ask, why is it diminishing? It's slightly diminishing as you get further on because there is such a dramatic decrease in rod power. With this decrease in rod power, there's a significant decrease in fuel temperature. As such, you're not really releasing anything. And since this is a ratio of what's produced versus what's released, the release isn't changing, but the production continues to go up. So, you'll start seeing a slight decrease here. So, that's the trend you would see in the long-lived.

And then, when you get to the short-lived, it has a different trend. So, the same power operating limits; the same fuel rod. And here, you're calculating the release rate, which is shown in these yellow squares, for iodine-131. Now iodine-131 has a half-life of eight days. So, the active iodine that's produced at the beginning of life, it's not active iodine by the time you get further on in life.

So, the population or amount of active

iodine-131 in the gap is much more dependent on the instantaneous flux; in other words, the production rate. It becomes very dependent on both fuel temperatures, which govern the release, and then, production rate, which keeps the amount of active iodine available for release.

So, what you're seeing here is the maximum combination of that occurs at the maximum production rate, which is at this first knee of the curve. And so, it's building up. It's building up here, and you may ask, why is it different here than here? That's because your burnup has increased; your diffusion coefficients have changed, and also, your fuel temperature has changed, with the degradation of thermal conductivity. So, as that decreases, fuel temperatures are actually going up, even though you're at the same part.

And there's other competing effects, like irradiation swelling, closure of the gap, things like that. But this turns out to be the maximum point with respect to the production over the release to calculate the amount of active iodine that's in the gap.

And so, combining the highest release fractions for each of their points in life, you end up

with a table for the BWRs.

Now, how does this compare to what's in Rev. 0? For iodine-131, Rev. 0 had a fraction release of 0.08. So, even though we pushed this limit up, this knee of this curve was pushed up relative to Rev. 0, we were able to compensate for that by using the new ANS standard. So, there actually ended up being a decrease in the active iodine, which is the most important of the radionuclides when it comes to calculating the effects of radiation on the human body because of the appetite of your thyroid for iodine.

Krypton-85 was 0.10 in Rev. 0, and that has gone up to 0.32. And that is, again, because we pushed this envelope way up and we pushed it out in burnup. So, the combination of those two, since it accumulates continuously, is you're getting a much larger value. But the nobles don't have as a big of an impact.

And we'll get into some more discussions about the alkalis, which is like cesium-134. This was at 0.12, and now it's at 0.16.

So, that's the releases for the BWRs. Now let's go --

DR. SCHULTZ: Paul?

MR. CLIFFORD: Yes?

DR. SCHULTZ: Before you leave this slide, you mentioned that the curves are the past best-estimate values, as you've shown them.

MR. CLIFFORD: Right, this value right here.

DR. SCHULTZ: In the table, the table is derived with some focus on conservatism and other information, as you mentioned, from ANS-5.4?

MR. CLIFFORD: Correct. So, to get an upper tolerance value for the long-lived, you're using the empirical database for which the models of FAST were tuned. So, it's a model-specific fission gas release uncertainty that's being applied for the long term. For the short-lived, there is uncertainty terms dictated by ANS-5.4.

DR. SCHULTZ: Got you. I figured there was a transition there. That explains it. Thank you.

MR. CLIFFORD: Okay. So, let's move on to the PWRs. So, on the top left, it's the same trends. With respect to long-lived, they continuously accumulate. When we get to the short-lived, once again, you're seeing the values peak at the first knee of the curve for the PWRs, noting that the PWR has a knee that curves a lot further out. We used the same power level, but, instead of dropping and starting at

20 gigawatt days, it goes out to 35. So, you'll see that that has a first order effect on the results.

So, the iodine-131 went from 0.13 for the BWRs to the 0.07 for the PWRs. This was really the driving factor, why we wanted to have separate tables for the Bs and the Ps, because, intrinsically, they have different operating regimes, and there's no reason why we should, essentially, the BWRs for a different regime than a PWR. That was why that decision.

So, we're up at 0.07, but we're still below the Rev. 0 values, once again, due mostly to the introduction of the new ANS standard. The krypton-85 has just gotten even a little higher because of the shape and the slope of these lines, and the same with the alkalis.

So, in the next section, I wanted to talk about our treatment of cesium. The ANS-5.4 standard, either the earlier standard, the '82 standard, or the 2011 standard, they provide a means to infer the release fractions of cesium, based upon the predicted release fractions of krypton-85. And essentially, they recommend that, since the diffusion coefficient is considered to be twice that for cesium versus the noble metals, that you increase the predicted release

fraction of krypton-85 by the square root of the diffusion coefficient. So, that would be a multiplier of 1.414. So, in other words, whatever you predict for krypton-85, you multiply that by 1.414, and that's your estimated releases for cesium-134 and cesium-137.

Now, as you saw from the previous values, the value of krypton went way up because of we were pushing the fuel envelope so aggressively. And as a result, we ended up with values of cesium that just seemed completely unreasonable and unrealistic.

So, when the ANS model was developed, there was not a lot of data; there was not a lot of like post-irradiation examination, destructive examinations of the chemical compositions of what was in the gap region, you know, targeting cesium versus nobles. But there have been recent research programs that have done this.

So, looking at these various research programs, we identified there were notable differences between the physical properties of the chemical effects between nobles and cesium. Most important, the cesium tends to react with other constituents to form relatively immobile compounds, which, then, accumulate along the grain boundaries or can get trapped within the porous, high burnup structure that

we refer to as the rim.

And below 1200 C, 2200 F, the cesium is in a liquid form, and generally, forms a film along the grain boundaries. And that's the temperature you would see in higher burnup fuel, and that's the temperature you would see towards the outer region of the fuel pellet, you know, like at the rim region.

And when it is released, the studies show that the cesium actually can react with the I.D. zirconium cladding. And that can form, once again, less mobile, stable, non-gaseous compounds. So, that, again, hinders the release.

Because, for krypton-85, what you're doing is you're taking these chemically inert noble gases, which can readily diffuse, you know, from the fuel matrix onto the grain boundaries, and then, get released into the plenum region in there. They're just, essentially, hanging out in the plenum region of the fuel rod, and if you get any perforation in the cladding, there's essentially a puff release. So, they're available to release because they're relatively chemically inert; whereas, the cesium generally gets trapped along the way. And even if it gets out of the pellet, it kind of gets trapped again. So, it's not as readily released.

And so, from looking at all these studies, there were recommendations that were made of the various studies, looking at the ratio of release between the nobles and the cesium. So, we saw between 2.2-to-1 release to a 10-to-1 release from the different studies. So, conservatively, we chose a lower bound of a 2-to-1 release ratio between cesium and the nobles.

So, what you end up with is this relationship. So, you would take your predicted krypton-85 releases and you would multiply them by .5. That's, obviously, a dramatic change compared to multiplying it by 1.414. So, this is, it's more realistic using better engineering, and this is generally a release that we identified.

CHAIR PETTI: So, Paul?

MR. CLIFFORD: Yes?

CHAIR PETTI: A question: given this potential of the clad to hold up fission products if they're cesium, other clads, non-Zircaloy clads then, there would be some more work required to get a database on those, I would think.

MR. CLIFFORD: Absolutely. That is absolutely correct.

As I mentioned, the steady-state releases

are extremely sensitive to rod design and power history. Any changes that are introduced really would invalidate these. When you're predicting the releases from fuel as a result of a core-wide melt, it really doesn't matter, the operating history, how you got to where you were. You're melting the fuel and releasing everything in the pellet.

CHAIR PETTI: Right.

MR. CLIFFORD: Whereas, here, we're just talking about kind of finds its way out of the pellet. And so, it's very sensitive to fuel temperatures and operating history, and then, pellet diameters and other constituents which would cause the noble, cause the various radionuclides to get trapped could all change.

A good example would be a dopant. If you added certain dopants into the fuel, those dopants could interact differently with the different radionuclides, which could cause differences in release. Maybe sometimes it increases the diffusion coefficient; maybe sometimes it traps it along the grain boundaries. So, it's very sensitive to that.

And so, as a result, the limits of applicability are kind of set at the UO2, and we're kind of in an unusual place. Because in the past,

everybody was using UO2, and it was all roughly the same within the same ASTM specifications. And they were all using zirconium tubing, roughly, within the same ASTM specifications, except for some minor tweaks here and there.

And so, the fuel behavior was very similar. Now we have ATF coming, where we're going to have a bunch of deviance, and especially when you start talking dopants and different theoretical densities and grain sizes of the different fuel and different materials for cladding. They're all going to behave differently.

And I'm not saying we can't model that, and I'm not saying we can't collect the data and validate models and do that. But the problem is, all that data is proprietary.

So, I think we're moving to a space in non-LOCA where the NRC's not going to be able to provide a one-size-fits-all because I don't have 10 proprietary fission gas release models that I can, then, exercise, and then, publish the results. I mean, I could do it internally to check what the licensees are doing, but I can't publish those results because it's based on proprietary information that I don't own.

And so, that will get us to really one of the next sections, and I think it will share inputs of the analytical procedure we provided.

So, next, we want to move away from the diffusion-based, steady-state release, you know, from the pellet into the void volume of the rod. And we'll kind of get into kind of a transient burst fission gas release, which is not dictated by diffusion. It's not dictated by, hey, you have a transient; you're running at a temperature for a couple of seconds, and that's causing more fission gas. This is really fission gas that's released as a result of grain boundary separation and fragmentation of the pellet, and the high burnup structure.

So, there have been many tests which have been conducted, prompt-pile excursion-type testing. And they have collected and reported fission gas, which was released during the accident itself.

And I know this figure is a little busy, but there's a lot of good information. Let me just run through it real quick.

So, what we've chosen to do is plot the data as a function of measured transient fission gas release. So, this is the amount of fission gas that was released during the prompt-pulse test.

And on the X-axis is the change or peak enthalpy increase. So, it's the change in the fuel enthalpy. So, it's the amount of energy that was deposited during that brief power excursion.

So, you have different symbols. You have tests that were conducted at the CABRI reactor in France, the NSRR reactor in Japan, the BIGR and IGR reactors in the Russian Federation. You also have designations for PWR rods and BWR rods, and finally, you have a designation for MOX. So, everything that's not labeled MOX would be standard UO2.

And, in addition, we've color-coordinated the results -- green being very low burnup fuel, below 30, all the way up to purple, which would be greater than 70. So, you've got all this data, and there is a large spread in the data, but that's not unexpected. You actually see a pretty good spread in the data for steady-state fission gas release. So, it's not surprising to see a spread in the data for transient fission gas release.

And what the general trends are is, as you deposit more energy, you, basically, have more fragmentation of the pellet, which results in more fission gas release. So, as you move in this direction, you're seeing more transient fission gas

release.

And if you look kind of going straight up vertically, and if you pick up 100 delta calories per gram, you kind of see a general trend where the higher burnup rods for the same energy deposition are releasing more fission gas.

As I mentioned, there is some spread in the data, but there are general trends. And we've chosen to use engineering judgment and kind of try to capture the leading edge of this correlation, which we feel is a function of both the increased or the deposited energy, which is delta H and burnup, and we came up with these correlations here.

Okay. Moving on to --

CHAIR PETTI: So, Paul, just for clarity in terms of how this would actually be applied, the whole closedown, seeing these increases, and in these events there's a certain number of rods around whatever rod ejection event is being postulated. Is that --

MR. CLIFFORD: Yes, it's a very localized event. It generally only occurs in a relatively short axial region of the core, and it's a very localized radially region of the core. So, you could see only a few assemblies, and then, only a few feet of each

assembly, and then, based on proximity to the actual rod that gets ejected, you're going to see a distribution of delta Hs.

So, the modern fuel, the modern core physics parameters -- the modern core physics models are certainly capable of calculating changes in fuel enthalpy across several different bundles, as a result of an ejected rod; whereas, you're seeing relatively large quantities -- like you're saying, well, 20 percent fission gas release, that's a lot. But you're generally not -- it's only being applied to a few axial segments of the rod. It's not over the entire length of the rod. So, it's not, when you actually apply it on a rod segment basis, based upon individual delta Hs across the different rods, I don't see it as being a major impact.

MEMBER HALNON: So, Paul, this is Greg.

Hey, how do these trendlines get drawn? Just if I put my thumb up to it, I would have made a much flatter line for the red line.

MR. CLIFFORD: Yes. Yes, I mean, to try to look at this from numerical analysis and do a data reduction, and try to come up with a two sigma, it gets very complicated. It was really just engineering judgment trying to capture the leading edge. Yes, I

agree, you could draw these lines slightly differently, but that's just the way we chose to draw these lines.

MEMBER HALNON: Could you do a sensitivity analysis to see if you drew it drastically different, but the results would be at least reasonably the same?

MR. CLIFFORD: So, we did not. We did not. And really, it's difficult. It's difficult to do a sensitivity analysis on the impact of these fission gas release correlations because the transients are very plant-specific and very cycle-specific. And in the margins to the actual offsite and control room doses, it varies widely throughout the industry. So, if you're saying, well, if I added an additional 3 percent fission gas release as a result of incorporating these various lines to the different rods in the core, how does that impact the offsite doses? It's impossible to do that.

MEMBER HALNON: Okay. So, a steeper line is more conservative, then, to the end result?

MR. CLIFFORD: Correct. A steeper line is more conservative. So, that's what we tended to go for.

MEMBER HALNON: Okay. That's good. Thanks.

MR. CLIFFORD: Okay. So, the transient fission gas release, as we mentioned, it's an additional component. So, you have the fissile -- I'm sorry -- you have the radionuclides which are in the fuel rod void and ready for release when the cladding fails. And now, you have this additional component which you need to add, add to the overall source term when doing your dose consequence assessments. As I mentioned previously, it's sensitive, we feel, to both fuel burnup and deposit of energy, and these are the numerical values of the equation that are in the Reg Guide.

Now the releases are based upon measurements of the noble gases. So, there needs to be some adjustment made for kind for the short-lived, more volatile radionuclides, like iodine-131. Actually, PNNL had done a sensitivity calculation many years ago, when we went out with DG-1199, and they came up with a relationship between the nobles and the short-lived half-lives radionuclides, like iodine-131. So, there was a correction factor of multiplying it by 0.333.

DR. SCHULTZ: Paul, just to go back to what you said on the previous slide, that is, if one looks at the accidents, as Dave mentioned, that have

a relatively localized effect of a transient power increase, that your judgment is that the additional dose associated with that event is likely to be small, a small effect on the overall evaluation?

MR. CLIFFORD: That would be my assessment, that's correct.

DR. SCHULTZ: And then, the other statement you made -- and I didn't interrupt you at that point -- but you said that, I think you used the term that there's wild differences amongst the evaluations of doses that are calculated across the licensees' reactors.

MR. CLIFFORD: Uh-hum.

DR. SCHULTZ: That kind of surprises me, that that would be the case.

MR. CLIFFORD: So, the licensing basis of each plant is considerably different.

DR. SCHULTZ: Yes.

MR. CLIFFORD: It really depends on how old the plant is; when they were licensed; what they've done since then; how many licenses they request.

DR. SCHULTZ: Okay.

MR. CLIFFORD: But they have altered a source term. Maybe they don't alter a source term.

DR. SCHULTZ: Okay.

MR. CLIFFORD: But they may have different wind patterns. You know, there's different atmospheric dispersion factors in their dose calculation.

DR. SCHULTZ: That's okay. I understand what you've said now. It's over the decades, and so forth?

MR. CLIFFORD: Yes.

DR. SCHULTZ: Over the methodologies that have been applied.

Thank you.

MR. CLIFFORD: Okay. So, the next topic is the analytical procedure. As I mentioned earlier, and as Mark mentioned at the beginning, we've actually reviewed and approved several applications that have come in, most notably, from Duke Energy. They submitted almost their entire fleet, where they could not reach that range of applicability that's in Rev. 0, that 6.3 kilowatts per foot, about 54 gigawatt days. They just couldn't meet it.

So, they had to come in and say, "I want to calculate some new values, but I don't even know where to start." Because like how am I going to replicate something that the NRC did 20 years ago that

wasn't even well-documented?

So, we've looked at different -- and it wasn't just Duke. There were other utilities which have also run into that limit.

And as a result, we felt, hey, why not publish an acceptable analytical procedure for calculating these values? And this significantly reduces regulatory uncertainty because now the licensees can say, okay, well, maybe I can't live with what the Reg Guide proposes, or maybe I don't want to live with them; it's not like I can't, but maybe I want to seek some margins in my dose calcs.

I mean, there's a big difference between predicted steady-state releases for the same power level between a 14x14 CE fuel rod and a 17x17 Westinghouse fuel rod. So, if I'm a licensee and I've got a 17x17, I don't want to use the 14x14. So, I'm going to go off and calculate my own.

And maybe my operating domain looks differently. Maybe my fuel management schemes are significantly different. So, the knee of that curve moves dramatically, and I can capture some of that margin by using something more realistic.

And as the NRC, we can't publish every variant because there's so many variables here between

fuel rod design and operating history. We're providing a generic bounding assessment that someone can use, if they don't want to go off and do their own calculation. Like it's a default.

But if people want to do their own calculations, which I believe most of the industry will, then we're providing this means for them to do that, and it needs for us to review it.

So, each of the vendors have approved fuel thermal-mechanical codes which are well-validated and verified and have quantified uncertainties. And they can use those methods, incorporating the 5.4 standard, the ANS-5.4 standard to calculate the short-lived ones. And that provides them with a tool.

And this provides them with a lot of flexibility, but it also supports things like high burnup. If a vendor has an approved code that's approved after 72 gigawatt days, then they can use that code to calculate release fractions that are good up to 72.

So, you know, as they get more information, and they quantify their models and validate their models, they can use that to, then, predict better release fractions, and use that in their dose calculations.

And as I mentioned, the benefits are the licensees can use plant-specific values, recover some margin. They can use that margin for high burnup, or wherever else they need that margin. And once again, it really reduces regulatory uncertainties. So, we don't have people doing it all different ways that we wouldn't find acceptable, once it's finally submitted to us. So, that's the basis for what we issued the analytical procedure.

And here's my last slide here. I just kind of wanted to talk briefly about high burnup and increased enrichment in ATF.

So, DG-1389 provides the revised steady-state fractions that are applicable to an expanded operating domain, which helps support high burnup applications and increased enrichment, because you're moving the power levels both up and out.

There's a significant relaxation in the short-term iodine-131, in the long-term cesium-137 radionuclides which were achieved through more rigorous engineering, meaning applying the latest and greatest ANS releases and, also, going off and looking at some available research data, and making some reasonable assumptions on holdup of cesium.

The Reg Guide provides an acceptable

analytical procedure for calculating application-specific release fractions, and that definitely supports high burnup in ATF applications.

And once again, we're going to end up with a lot of different proprietary fission gas release models out there. And it's difficult for the NRC to use that information and publish results. It's really going to be, as we deviate from UO2 and we start adding a lot of things to the fuel or changing materials for the cladding, the vendors, it's going to be their responsibility to generate proprietary -- let me rephrase that. They're going to have to publish results using their proprietary models. And certainly, there's a lot of flexibility that's provided to the industry by providing that guidance.

And finally, combined with the Reg Guide 1.236 coolability limits, the transient fission gas release components provide a means to address FFRD for the postulated PWR control rod ejection and BWR control rod drop access. And that, again, supports high burnup and increased enrichment applications.

So, that's the conclusion of my slide presentation. Are there any questions before we move on to Elijah and fuel-handling accident?

DR. SCHULTZ: Paul, this is Steve Schultz.

On the second bullet, the ATF applications, I mean, there's several different types of applications, ATF applications, that have been proposed. But the Reg Guide, the Rev. 1 is focusing on a couple of those in terms of the overall -- the current evaluations that the staff has done. Is that correct?

MR. CLIFFORD: Not entirely. So, the fuel melt MHA source terms have deemed appropriate for high enrichment and different fuel designs such as chromia-doped and chromium-coated.

DR. SCHULTZ: So, it's "such as." So, the other designs that have been proposed aren't included at this point?

MR. CLIFFORD: Correct. And neither of those designs are approved for the non-LOCA. In other words, the Table 3 and 4, I believe, which is the non-LOCA steady-state releases, are not applicable to doped fuel or coated cladding right now.

DR. SCHULTZ: And your second sub-bullet, the relaxation and flexibility may be achieved, how is the staff going to review and validate the submittals of the licensees?

MR. CLIFFORD: So, well, the licensees would need to submit an LAR because they're making a

change, a fundamental change, to the assumptions in their dose analyses. So, we would expect that they would use their approved codes, such as GALILEO or PRIME or PAD5, and along with the ANS-5.4 model, follow the procedure that's in the appendix and calculate design-specific or plant-specific release fractions. And then, we would review and approve those.

DR. SCHULTZ: And you'd use that with the results from FAST?

MR. CLIFFORD: I mean, we could. We certainly could run some confirmatory calculations with FAST. I'm not sure it would be needed all the time because we've done so many cases already with FAST. So, we understand the sensitivity.

So, if someone came in and said, look, I'm moving that knee of the curve back and down a little, and, oh, by the way, instead of 7 percent iodine releases, I'm now getting 6 percent, or 5 percent, we'd be like, well, that's obviously in the right direction.

DR. SCHULTZ: Understood. That's good. Thank you.

MR. CLIFFORD: Okay.

CHAIR PETTI: I just want to note that

we're running a little behind.

MR. CLIFFORD: Don't blame me.

CHAIR PETTI: We still have industry, and we have significant public comment.

Keep going.

MR. DICKSON: Okay. I'll make this short.

With the interest of updating Reg Guide 1.183, we decided to revisit the Appendix B fuel-handling analyses. This slide highlights our work that incorporates lessons learned for the last 22 years, or 21 years, of reviewing the fuel-handling accident analysis for the alternative source term.

To give you a little bit of history, the FHA can require significant staff resources, as well as from the licenses, and in some cases become a limiting accident, despite a low safety risk significance.

What we decided to do is revisit the old studies supporting the current FHA analysis, much of which stems back from work that was done back in the early '70s.

The old model retained, Rev. 0 retained much of the old models from Reg Guide 1.125, which is a very, very old Reg Guide, and assumed that the release to the environment is a puff release, where 5

percent of the gas gap for iodine is available for release.

With the new model, we developed a new model that's based off of improvements in our current understanding of the reactor fuel in physics, iodine chemistry, and the environmental conditions under which fuel-handling operations are taking place.

Also, this would lend to changing the chemical speciations from the two different models. So, the current Rev. 0 model assumes that iodine is in a gaseous form. The new model recognizes that, under fuel-handling operations in which the pool water is between 100 and 180 degrees Fahrenheit, that cesium iodide is actually -- the predominant form of iodine is in cesium iodide form. Under these conditions, it's in a solid form.

So, the revised model assumes that the entire fuel pin gap inventory of iodine is available for release. Ninety-five percent of it -- I'm sorry, can we go to the next slide, slide 43? There we go.

So, the revised model assumes that the entire fuel pin gap inventory, 100 percent of that iodine is available for release to the pool within two different phases. The first is the initial puff phase. The second would be the re-evolution phase of

cesium iodine, given the low pH of the spent fuel pool or the reactor cavity.

In general, the alternative fuel-handling accident models compute estimated radiological consequences that are around 90 to 95 percent lower than that of the current models. So, we did identify that there is considerable margin due to the scrubbing effects of iodine in the spent fuel pool, as well as considering the chemical speciation of iodine being in a solid form under these conditions.

And with that, I will pass the presentation on to Mr. Mark Blumberg.

MR. BLUMBERG: Thank you, Elijah.

Can we go to slide 44, please?

So, since we provided you the DG, a couple of issues have come to our attention that we're considering addressing. This slide shows those potential changes that we are considering

The first of these is in response to an issue that was recently identified. It is being considered because the currently proposed Regulatory Position for release fractions might benefit from additional clarification to directly state that the applicability of these particular release fractions do not pertain to fuel with iron, chromium, aluminum

alloy cladding.

The second is with respect to atmospheric dispersion factor modeling. We are considering adding guidance to address facilities that need to determine offsite atmospheric dispersion factors for distances less than 1200 meters. This change provides some guidance to address those situations.

I want to emphasize that these changes at this point have not been added to DG-1389, but we wanted to make the ACRS and the public involved in this meeting aware that we are considering these two additional changes.

Can we go to slide --

MEMBER HALNON: Mark, I'm sorry, this is Greg.

Would that include the new 1.249 Reg Guide for ARCON?

MR. WHITE: Yes.

MR. BLUMBERG: Yes.

MR. WHITE: That's what it's referring to.

Sorry, this is Jason White.

MEMBER HALNON: Okay. I didn't see the number up there. That's why I was wondering.

MR. WHITE: Yes, we didn't reference the number because it hasn't been issued yet. It hasn't

been approved. So, it's still --

MEMBER HALNON: It's supposed to be this month, I think.

MR. BLUMBERG: It's imminent, I believe.

MEMBER HALNON: Right.

MR. BLUMBERG: Okay. Can we go now to slide 45?

So, I wanted to summarize for you some of the key takeaways from our presentation.

Currently, Reg Guide 1.183 does not have an acceptable main steam line aerosol deposition level provided in it.

To address future changes for this particular pathway, DG-1389 provides guidance for three acceptable means to align aerosol deposition levels.

DG-1389 also provides guidance to support a streamlined approach for licensees to credit an alternative pathway for BWR MSIV leakage to the main condenser.

The staff are prepared to review near-term ATF designs and burnups up to 68 gigawatt days per metric ton uranium peak rod average and U2-35 enrichments up to 8 percent. Future updates to the guide are expected to accommodate higher burnups and

enrichments.

The range of applicability has been expanded to support modern fuel utilization. The impact of this expanded operating domain on radiological source terms has been minimized by applying more rigorous engineering judgment.

Lastly, an acceptable analytical procedure for predicting plant-specific radiological or radionuclide release fractions has been included and provides flexibility in margin recovery.

I'll now turn the presentation over to Sean for our last slide.

MR. MEIGHAN: Hi. My name is Sean Meighan. I'm the Project Manage for the update to Regulatory Guide 1.183, and I'll be quickly reviewing the next steps for the update to the Regulatory Guide.

So, looking forward, the next step would be to issue the Draft Regulatory Guide 1389 for 60-day public comment, and that's going to occur fairly soon, in April of 2022, this year, so next month.

The next step after that is staff review and disposition of the public comments.

And then, we will update the Draft Regulatory Guide, as necessary, in response to the public comments.

The next step after that would be the Office of General Counsel review of the final draft, followed an ACRS full Committee meeting.

And then, in the November-December timeframe of 2022, we expect the issuance of Reg Guide 1.183, Revision 1.

And that's the next steps, looking forward.

So, I turn it back to Research, or ACRS, for the rest of the meeting.

Thank you.

MEMBER HALNON: Hey, Sean, this is Greg Halnon.

When we get to the full Committee, and it may be a "no, never mind," but I'd like to hear it at least the effect of the flow accelerated corrosion programs on the seismic rigidity and other conditions that may be latent in the secondary system that we're taking credit for for holdup. Like I said, just consider it, and it may be a "no, never mind," but when we get to full Committee, I'd like to at least hear that piece of it.

MR. MEIGHAN: Okay. So, I'll with Steve and Shilp to ensure that that aspect is addressed in the full Committee meeting.

Go ahead. I heard somebody hopping in.

(No response.)

CHAIR PETTI: Members, any other questions?

(No response.)

Thank you, Staff.

And now, let's turn -- I don't know who it is. NEI is going to speaking.

MS. PIMENTEL: Hi. This is Frankie Pimentel from NEI. Can everyone hear me?

CHAIR PETTI: Yes, we can.

MS. PIMENTEL: Okay. I want to be able to share my screen also. Okay.

MEMBER REMPE: Is it Michael that needs to quit sharing his screen, in order to enable Frankie to share hers?

MR. EUDY: Frankie, this is Mike. I've got your slides. If you're having trouble, I can probably just display yours for you.

MS. PIMENTEL: Okay. That would be great. Thank you.

MR. EUDY: All right. Bear with me for a second.

(Pause.)

MS. PIMENTEL: So, while we wait, Greg

Broadbent will be presenting our slides. So, I'll just provide our opening remarks.

So, good morning to everyone, and thank you, Subcommittee Chair David Petti and other Committee Members, for allowing us to participate in today's meeting, and provide comments on the proposed draft of Reg Guide 1.183.

The proposed revision to this Reg Guide is important to the industry to support ATF objectives. Dose analyses are a long lead-time item for the analyses and the NRC review. So, clarity as timely as possible on this proposed guidance is important to us. We don't want unnecessary delays, but we recognize that some changes may be needed to be able to use the guidance provided in this revision.

Some fuel suppliers and utilities have described the dose consequences resulting from the Draft Reg Guide 1.183 as challenging. Based on the proposed draft, select sites have performed dose calculations using the revised release fractions, timing, and elemental groupings, assuming no increase in burnup. For these select sites, dose rates increased, significantly so for BWR units. While this isn't a systematic evaluation of the fleetwide impact, these results indicate that the current licensing

bases source terms will require revision.

While industry desires how the proposed Revision 1 of Reg Guide 1.183 establishes methods for achieving 68 gigawatts days per MTU on peak rod average burnup, industry views the proposed revision as introducing unnecessary conservatism to the dose consequences for PWRs and BWRs in doing so.

The intent of our presentation is to provide additional insights from our initial concerns, from our review of the proposed revision. Greg Broadbent, Senior Staff Engineer in Nuclear Analysis for Entergy will be providing our presentation.

Thank you.

MR. BROADBENT: All right. Can you hear me?

CHAIR PETTI: Yes.

MR. BROADBENT: Okay. So, you can back it up a slide. That's the slide we'll start with.

But, first of all, I'll say, good morning. I'm Greg Broadbent. I work for Entergy as a Senior Staff Engineer in the Corporate Nuclear Analysis Group in Jackson, Mississippi.

And I'll say that the industry recognizes that the staff has expended significant resources on developing this Draft Reg Guide, and our goal is to

ensure that this effort results in a product that all currently licensed plants and applicable new plants can apply well into the future.

So, starting with slide 2, which you're on, the proposed changes in the Draft Reg Guide are significant. The impact of a transition to the proposed guidance on the plant licensing bases can be wide-ranging, including offsite and control room doses, environmental qualification, containment or drywell spray timing, and suppression pool chemistry.

These proposed changes are considered as significant as those introduced with the original transition from TID-14844 source term to the AST. As such, the intended and unintended impacts of these changes and the release fractions and timing should be evaluated and shared with the industry before the Reg Guide is issued, similar to those re-baselining analyses performed for the AST introduction some 20 years ago.

As I'll discuss later, the conservatisms and changes in the current version of this proposed Reg Guide may preclude its use by many plants that need to implement ATF design concepts with extended fuel burnups and enrichments.

All right. Slide 3.

A primary concern for the industry is the significant change in the release fractions for the maximum hypothetical accident. The BWR release fractions for the MHA nearly doubled for the iodines, which currently comprise a large fraction of the calculated offsite and control room doses.

Contributing to this is the eightfold increase in tellurium, which eventually decay in the iodine. The bases for these large increases are not well-understood, but appear to be driven by the scenario selection and pool scrubbing assumptions in Sandia's evaluation and report SAND-2011-0128.

The core damage scenarios applied in the development of the proposed release fractions are more heavily weighted to station blackout sequences compared to those previously used in NUREG-1465, in spite of the industry's recent efforts in minimizing SBO risk with post-Fukushima modifications.

It's also unclear from the Sandia report whether the beneficial effects of suppression pool scrubbing have been credited to reduce the airborne release fractions for these SBO scenarios.

In addition, the proposed models for BWR steam line deposition appear to be more conservative than those currently approved for many BWRs. And I

think Mark had mentioned that in his presentation.

As a result of these conservatisms, BWRs may find it difficult to meet the proposed Reg Guide without significant licensing basis changes or plant modifications. These changes could include reductions in MSIV or containment leakage rates, or tighter control room boundaries that may result in higher occupational exposures to meet these more stringent limits with little or no safety benefit to show for it.

All right. Next slide, please.

Considering the potential difficulties for some plants in implementing the proposed guidance, the industry wants to ensure there's clear acceptance of remaining with the current guidance. Specifically, what analyses or inputs would need additional justification or modification for applications submitted under the current Reg Guide revision? Are the currently approved non-LOCA gap fractions and main steam line deposition models still acceptable, or would a plant be forward-fit to the updated guidance, particularly with regard to steam line models?

For those plants remaining with the current guidance, we would also like to ensure there's a clear path for selective implementation of certain

aspects of Revision 1, such as the revised fuel-handling accident. We recognize that Section 1.2.2 of the proposed guidance describes this approach.

All right. Next slide, please.

Appendix B documents a revised approach to analyzing the radiological consequences of a fuel-handling accident. This model was based on a thorough review of decades of historical information and the current understanding of aqueous iodine chemistry. This model was not developed based on an industry proposed application, but, instead, was internally developed by the NRC.

The industry wants to acknowledge the staff's effort, and based on our review of the proposed model, we agree with the staff that this approach is a more accurate characterization of the low consequences expected from this event.

Slide 6.

And this is just with regard to the DPO that was mentioned back in November on ISG-2021-01. There was some discussion of a DPO regarding potential inadequacies associated with credit for steam line deposition. And the industry wants to be proactive and work to ensure that any valid NRC concerns are addressed. However, there's been no communication

with the industry regarding the details or status of this DPO. And we would like to know if this DPO will affect the revised seismic analysis in the proposed guidance or the schedule for releasing the proposed Reg Guide.

Any questions for me?

Oh, and I'd like to just mention, with regard FAC, the Flow Accelerated Corrosion issue that came up recently, the BOP systems do have models for flow accelerated corrosion. We have a FAC program at all our sites. We have models that identify areas where FAC is more likely to occur. These are typically areas where there's two-phase flow and the droplets tend to generate the FAC on the inside of the piping.

And we do inspect at some certain frequency, as recommended by our models, that piping to ensure that the minimum wall thickness is maintained. And obviously, if that piping does fail during operation, it will be something that we would, obviously, recognize, and we would potentially have to shut down to fix.

So, I just to, at least from the industry perspective, address that FAC issue.

So, any questions for me?

MEMBER BALLINGER: This is Ron Ballinger.

I guess the question is, the FAC program identifies in principal areas where you've got decreased wall thickness, not necessarily wall thickness reduction that would threaten the sort of operational integrity. But do these results get factored into the analysis that you're doing here? In other words, you're now starting out with a pipe which has a different wall thickness.

MR. BROADBENT: Yes, I'm not sure what the answer to that is. You know, we ensure that we do maintain that minimum wall thickness, and I would think that the requirements would be to only maintain that wall thickness.

MEMBER HALNON: This is Greg.

That's why I thought we should have the staff consider it and bring us back the answer to it. It's small piping, typically, but, still, think about it, and then, we'll go forward.

Greg, I guess this is just a lingering question in my mind. NEI, is Entergy a member of NEI? Are you speaking for the industry? Or are you speaking for Entergy?

MR. BROADBENT: I'm in the NEI Working Group that's addressing this, and I'm also the

Chairman of the BWR Owners' Group Committee that is associated with this Reg Guide.

MEMBER HALNON: Okay.

MR. BROADBENT: So, I'm speaking for the industry.

MEMBER HALNON: Okay. Thank you.

CHAIR PETTI: Any other comments, Members?

MEMBER BROWN: Yes. I just went off and looked at the implementation part of the Reg Guide. Normally, that's a little bit shorter and there's no backfit and no forward. You know, it's very clear statements. And this is a very long statement that is somewhat fuzzy, like the staff does not intend or it's not a declarative statement, is all I'm saying. So, I'm not so sure whether this supports the industry's concern or inflames the industry's concern. I'm not trying to say that in a inflammatory manner. It's just that it's a very vague, long implementation, based on what I've seen in other Reg Guides, where we've so no backfit is there. And I understand the industry's concerns.

So, I think somehow this needs to be defined a little bit more conclusively, if you look at the NEI's statement, or the industry's statement on the earlier slide, if I go back to it, where the

overall impact should be evaluated prior to issuing it.

It seems to me the backfit considerations are going to be open to the public, open to some comments. So, I'm just raising that now, so that you can address that, and stuff, in the future. This is for the staff.

MR. EUDY: Did the staff understand the question? This is Mike Eudy. I'm just trying to take notes.

MEMBER BROWN: It was really a comment, based on reading the implementation paragraph on page 34.

MR. EUDY: Yes, Section D?

MEMBER BROWN: Yes, Section D.

MR. EUDY: I can pull it up, if you'd like.

MEMBER BROWN: If you can put it on the screen, that would be fine. That way, at least somebody can see what it says. It's pretty long, though. It's most of the backfit.

MR. EUDY: Yes. Just for perspective, it's pretty much our boilerplate for all Part 50.52 Reg Guides, this section. It's much shorter than it used to be. But this is up-to-date with the current

MD 8.4 and our considerations of backfit and forward-fit.

MEMBER BROWN: Well, do you think this satisfies the industry concern, that they don't have to abide by that and make plant changes, based on the issuance of this Reg Guide, and go back and reassess all the statements that they made during their presentation?

MR. EUDY: Well, the main point is that, you know, Reg Guides used to be just forward-fit. And now, it's not. It's, basically any applicant can use any regulatory guidance that's out there, even if it predates this one. They would just need to justify it.

So, we're pretty clear. We're trying to be pretty clear on that. And if they believe that there's an issue, like if we bring up this Reg Guide during a review and say, "Well, you're not consistent with Reg Guide, Revision 1," they have remedies for that, based on the language in MD 8.4. So, really, you know, unless a previous revision of a Reg Guide is withdrawn for maybe safety reasons, you know, they can still use previous revisions.

Now this Reg Guide, as discussed earlier -- someone can jump in, if they'd like -- how they're

going to coexist together, it's a little complicated with this particular Reg Guide.

MR. BLUMBERG: Yes, this is Mark Blumberg of the staff.

Every plant has their own design and licensing basis. So, making general statements about what can be and what can't be done is very, very difficult.

But there are differences between the two sets of guidances and what they're applicable for. And so, in those instances, a plant that wants to use Rev. 0, but they want to apply it to fuel that has higher burnups than the applicability for Rev. 0, then they need to justify why that's appropriate. It's not just a slam dunk, you know, I've got new fuel, and now, I want to use Rev. 0, because Rev. 0 was not designed for those higher burnups.

MEMBER BROWN: Has Rev. 0 been used for any plants that want to use it for higher burnups? Has that been through staff review at any point yet for BWRs? Or PWRs, if that's applicable?

MR. BLUMBERG: Paul, would you like to address that for the non-LOCA?

MR. CLIFFORD: Well, we have not received any applications for burnup extension.

MEMBER BROWN: So, it's an open issue? Is that a correct assumption or statement by me?

MR. BLUMBERG: What do you mean by "open issue"?

MEMBER BROWN: Well, I mean, if it hasn't been applied, then the staff has not had to address the use of Rev. 0 for higher burnups for existing plants.

MR. BLUMBERG: That's a correct statement for higher burnups.

MEMBER BROWN: So, industry's considered thoughts are that this could be a problem for them. Because it sounds like there's a lot of work for them to adapt this on a forward-fit basis, the higher burnups. Is that --

MR. BLUMBERG: The intent of this revision to the Regulatory Guide was to provide a methodology to address that, to, basically, create less work. That is the intent of the guidance, is for efficiency and effectiveness of regulation.

MEMBER BROWN: Okay. All right. So, it's going to go out for public comment, and we'll see what falls out. I guess that's the conclusion I'd draw from that.

If anybody on the Committee disagrees with

me, let me know. I'm just kind of looking at it from a non-source-term development person.

I'm done.

MR. BLUMBERG: Okay.

MEMBER HALNON: So, Greg, back to Greg from Entergy.

MR. BROADBENT: Yes, I've got it.

MEMBER HALNON: You mentioned earlier that this was going to discourage the BWRs from trying to get their MSIV leakage. Do you think that or do you have a feel for whether or not, if Rev. 1 existed on its own and there was no Rev. 0, that we would not see any more ASTs coming in for leakage?

MR. BROADBENT: That would be likely. Yes, if they play with the new release fractions for BWRs, you know, doubling the iodine release -- I mean, iodine is the main contributor to doses and most of our analyses. So, if you double that release, you're significantly increasing the outside and control room calculated doses.

And then, you throw in changes to the steam line deposition models, as I think previously mentioned, some of the earlier revisions are not considered acceptable to the staff anymore. That would make it very difficult for BWRs to come in with

any changes to MSIV leakage rates. In fact, they probably would have to be decreased.

MEMBER HALNON: So, given the fact that the numbers are higher or the release fractions are higher, are the numbers reasonably correct? Or are they just overly conservative, and you can identify where that conservative point is that caused this to happen, and can justify why it's not necessary?

MR. BROADBENT: So, the report, the Sandia report that developed them, has very little detail regarding how they were developed. It doesn't really describe the model or anything that was used.

But what it does list, for example, the events that went into developing them, and they're significantly different for BWRs than for previously applied in NUREG-1465, and it's different than even what was applied for PWRs. If you look at that Sandia report -- 2011-0128, I believe is the number -- most of the core damage sequences are associated with station blackout for BWRs. And we've done a lot of work in the last 20 years to reduce our station blackout risk. So, if that's scenario selection was risk-informed, it's going the wrong way.

And station blackout releases occur through the SRVs and through the pool. So, there

would be a significant amount of scrubbing associated with those releases before the source terms would become airborne. And it doesn't appear as though that scrubbing effect was considered in the analyses that were performed.

So, we think those are the two impacts that are driving the increase in BWR releases. We don't think it's the increase of burnup itself.

MEMBER HALNON: Thank you.

MEMBER BROWN: Can I ask one more question? I'm not much of an expert on the station blackout. I mean, I know what it is, but that's assuming you have no power. And could that be mitigated by plant modification which would decrease the risk of having the station blackout?

MEMBER HALNON: No, all SBOs are not created equal.

MEMBER BROWN: I figured that.

MEMBER HALNON: It depends on what feeds into it, diesel generators configuration, and --

MR. BROADBENT: Yes, I think what these scenarios that were assumed, you know, the NRC needs to generate some core damage. And so, we have makeup capability for station blackouts through our RCIC reactor cooling, reactor core isolation cooling

system, RCIC, R-C-I-C. And I would assume that those analyses probably assume that that trips at some point in time and you get a late-term-type core uncovery, and therefore, fuel failure.

But what I was pointing out was the selection of core damage sequences was heavily weighted towards station blackout. In fact, nearly all the scenarios that went into developing the release fractions for boiling water reactors were station-blackout-related sequences.

MEMBER BROWN: Okay. Thank you.

CHAIR PETTI: Okay. Let's turn now, then, to public comments.

Since we have received a request to speak, we'll start with Brian. Are you on the line?

MR. MAGNUSON: Yes.

CHAIR PETTI: Go ahead.

Unmute yourself. Unmute yourself.

MR. MAGNUSON: Hello. This is Brian Magnuson. Can you hear me?

CHAIR PETTI: Yes.

MR. MAGNUSON: Okay. As I stated, my name is Brian Magnuson. I'm a Lead Emergency Management Specialist at Constellation, formerly Exelon Generation. I'm a former NRC-licensed Senior Reactor

Operator and Operation Shift Manager at Quad Cities nuclear power plant. I am acting expressly as a member of the public.

First of all, I appreciate the opportunity and the invitation to attend this public meeting.

I submit my prior public comments that are posted in ADAMS and understand that documented responses from the NRC are now forthcoming.

I also submit my petition for rulemaking regarding Reg Guide 1.183, Rev. 0. It's PRM-50-122.

In my prior public comments, I referenced SAND-2008-6601, which determined that the BWR MSIV source term methodologies provided in Reg Guide 1.183, Rev. 0, are, quote, "non-conservative and conceptually inaccurate."

My prior comments expounded on this Sandia document and identified other examples in which Reg Guide 1.183 methodologies violate the laws of physics.

It appears that Draft Guide 1389 may correct some of the errors. However, if the NRC allows Rev. 0 to coexist with Rev. 1, then the NRC will still allow nuclear plants to ignore the laws of physics in accident dose calculations. The practice of Rev. 1 coexisting with Rev. 0 seems imprudent.

I submitted documented pre-meeting public

comments to the NRC and the ACRS on March 15th, 2022.

My initial comments for this meeting focus on the Reg Guide 1.183 and the Draft Guide 1389 assumption that primary containments will not fail in a maximum credible accident or, as the NRC is redefining, maximum hypothetical accident.

This is contrary to the intent of TID-14844 and the purpose of assessing worst-case credible accidents in source term dose calculations.

In some of the severe accident studies referenced in today's presentation, the NRC clearly recognizes that containments will fail in credible nuclear accidents. Many of us watched the containment at Fukushima catastrophically fail.

Therefore, the use of lesser accidents to determine the regulatory capability of nuclear power plants to prevent or mitigate radiological releases seems imprudent and illogical.

There was just some discussion on station blackout or long-term station blackout accidents. And I'll add to that the floods capabilities and the use of hardened containment events that were required by the NRC in Order EA-13109; that those source terms are considerably greater than the DBA LOCA used in the current Reg Guide and the proposed Draft Guide.

Additionally, I believe the NRC's use of risk-informed insights and regulations conflict with the regulatory premise of the maximum credible accident source term that's used in the design basis of each nuclear power plant.

It appears the NRC's risk-informed insight regulations are sometimes used to allow nuclear powers to evade deterministic regulations, particularly General Criterion 19, because the NRC recognizes, and I quote, "For certain plants, the margin for meeting acceptable dose limits in the control room or certain DBAs is very small." The concern is the implementation of the new ICRP 103 values because these would exceed regulatory dose limits. A subsequent evaluation confirmed this concern was accurate. Therefore, the NRC rejected ICRP 103.

This motivation epitomizes my concerns with Reg Guide 1.183 and Draft Guide 1389. I am concerned that the NRC's efforts are overly focused on mitigating known nuclear power plant design basis efficiencies, providing additional, sometimes questionable, allowance to ensure nuclear power plants can comply with deterministic regulations, circumvent deterministic regulations, while bringing them into compliance. This does not comport with the NRC's duty

to protect people and the environment.

That's the end of my comments for now, unless anybody has any questions.

CHAIR PETTI: Thank you for your comment.

Anybody else? Identify yourself, unmute yourself, and make your comment.

(No response.)

Okay. Not hearing anything, I think we are at the end of our meeting. So, I would like to adjourn our Subcommittee meeting.

Thank you all.

(Whereupon, at 11:56 a.m., the meeting was adjourned.)

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

Revision of Regulatory Guide 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

ACRS Subcommittee Meeting

Mark Blumberg, Senior Reactor Engineer, NRR/DRA (Technical Lead)
Elijah Dickson, Reactor Scientist, NRR/DRA
Steve Jones, Senior Safety and Plant Systems Engineer, NRR/DSS
Shilp Vasavada, Senior Reliability and Risk Analyst, NRR/DRA
John Parillo, Senior Reactor Engineer, NRR/DRA
Paul Clifford, Senior Technical Advisor, NRR/DSS
Sean Meighan, Reactor Scientist, NRR/DRA (Project Lead)

March 16, 2022

Agenda

- Key Messages
- Background
- RG 1.183 Updates Proposed in DG-1389
- Modified SAND 2008-6601 Method
- Re-evaluation of AEB-98-03 with Multi-group/Numerical Integration Method
- Technical Assessment on Seismic Analysis for Alternative Drain Pathway
- ATF, HBU, Increased Enrichment (LOCA)
- Steady-State and Transient Release Fractions for Non-LOCA Analyses
- Revised Fuel Handling Accident (FHA)
- Draft Revisions to Current DG-1389
- Key Takeaways
- Looking Forward
- Discussion/Feedback

NRC Management and Staff Coordination

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Mike Franovich, NRR/DRA
Joseph Donoghue, NRR/DSS
Mohamed Shams, NRR/DANU
Louise Lund, RES/DE

Working Group
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NRR/DSS:
Paul Clifford
Steve Jones

NRR/DANU:
Michelle Hart

RES/DE:
Michael Eudy

OGC:
Mary Frances Woods

Key Messages

- The NRC staff is revising RG 1.183, Rev 0 “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

- RG 1.183 Rev. 0 and Rev. 1 will co-exist as a result of SRM-SECY-18-0049, “Management Directive and Handbook 8.4, ‘Management of Backfitting, Issue Finality, and Information Collection.’”

- The NRC staff has elected not to finalize a draft revision to RG 1.183, known as DG-1199, and is issuing DG-1389 as a replacement. DG-1389 considers and addresses technical issues and public comments related to the issuance of DG-1199.

- NRC staff made efforts for early outreach to external stakeholders by holding 3 public meetings focused on the revision of RG 1.183.
  - November 19, 2020 (Slides ML20296A425)
  - March 5, 2021 (Slides ML21056A058)
  - May 13, 2021 (Slides ML21124A121)

- RG 1.183 Rev. 1 provides an acceptable method of modeling BWR MSIV leakage. Use of individual regulatory positions in isolation will need additional justification.

Key Messages (Cont'd)

- The objectives of the revision are to:
  - make the guidance more useful by considering feedback and comments from licensees;
  - incorporate relevant operating experience as well as recent post-Fukushima seismic risk insights and walkdowns;
  - respond to change of regulatory environment (e.g., backfit guidance SRM-SECY-18-0049 & NuScale SRM-SECY-19-0036);
  - incorporate lessons learned from recent NRC staff reviews of Alternative Source Term (AST) and BWR Main Steam Line Isolation Valve (MSIV) leakage LARs;
  - ensure sufficient guidance is in place for licensing advanced light-water reactors (LWRs), accident tolerant fuel (ATF), high-burnup, and increased enrichment fuel; and,
  - incorporate insights from new research activities.

Background

Background

- Origin: Footnote to 10 CFR 100.11(a) is a performance-based rule to evaluate the defense-in-depth provided by the containment.
  - TID-14844 Source term provided guidance which assumed the source term is instantaneously available in the containment.

- Radionuclide behavior observed during the TMI accident did not appear at all similar to the TID-14844 source term.
  - NRC initiated research efforts in the area of severe accidents which culminate in publication of NUREG-1150.
  - NUREG-1465 source term was derived from the sequences in NUREG-1150.
    - RG 1.183 Rev. 0 adopted the NUREG-1465 early in-vessel fuel melt source term.

Background (Cont'd)

- NRC staff developed RG 1.183 Rev. 0 (July 2000) to support implementation of 10 CFR 50.67, "Accident Source Term."
  - Applicable to nuclear power reactor applicants and licensees adopting 10 CFR 50.67.
    - Limited range of applicability on Non-LOCA release fractions.
  - Identified the significant attributes of an acceptable accident AST based on NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (1995).
  - Provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an AST.

DG-1199

- In October 2009, the NRC issued for public comment DG-1199 as a proposed Rev. 1 of RG 1.183.
- Staff received 150 public comments
- The reasons for revision of RG 1.183 in DG-1199 were:
  - Providing additional guidance for modeling BWR MSIV leakage,
  - Expand applicability of Non-LOCA release fractions to support modern fuel utilization,
  - Extending the applicability of the proposed RG for use in satisfying the radiological dose analysis requirements contained in 10 CFR Part 52 for advanced LWR design and siting,
  - Providing additional meteorological assumption guidance.

Modern Fuel Utilization

- Since DG-1199 was issued for public comment, NRC issued several license amendments to support modern fuel utilization.
  - Oconee Units 1, 2, and 3 (2019)
  - Shearon Harris (2018)
  - H.B. Robinson (2017)
  - Catawba Units 1 and 2, McGuire Units 1 and 2, Oconee Units 1, 2, and 3 (2016)
  - Diablo Canyon Units 1 and 2 (2015)
- Reinforced need for expanded Non-LOCA release fractions

2019 License Amendment Requests

- In 2019, NRC received several AST LARs requesting increased BWR MSIV leakage
- As a result, work on DG-1199 was postponed to allow NRC staff to incorporate lessons learned, from evaluation of the LARs, into the revised RG 1.183:
  - James A. FitzPatrick Amendment No. 338 for AST, July 21, 2020 (ML20140A070)
  - Quad Cities Nuclear Power Station, Units 1 & 2 – Amendment Nos. 281 and 277 to increase allowable MSIV leakage, June 26, 2020 (ML20150A328)
  - Nine Mile Point Nuclear Station, Unit 2 – Amendment No. 182 to change allowable MSIV leak rates, October 20, 2020 (ML20241A190)
  - Dresden Nuclear Power Station, Units 2 & 3 – Amendments Nos. 272 and 265 to increase allowable MSIV leakage, October 23, 2020 (ML20265A240)

Changes from Disposition of Recent Differing Professional Opinion (DPO-2020-002) Panel Report[1]

- A significant change in how radiological consequences are performed.
- Crediting Safety Related Systems with a deterministic fuel melt source term.
- Reinstatement of the Maximum Hypothetical Accident (MHA).
- Removing mechanistic explanations for the deterministic source term.
- The LOCA will be defined as an event resulting in the loss of the ability to cool the core.
- The MHA LOCA will be defined as an unspecified event resulting in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.
- Appendix A will describe acceptable assumptions for the evaluation of the MHA LOCA.
- Elimination of the assumption of a 2-hour delay in the distribution of the fuel melt source term into the containment and the elimination of the R-Factors in the Sandia Method.

[1] For more information see the DPO Case File for DPO-2020-002 (ML21067A645).

RG 1.183 Updates Proposed in DG-1389

General Updates

- Staff plans to include changes proposed in DG-1199 as modified by public comments.
  - Incorporate new or remove withdrawn regulatory guidance (e.g., RG 1.194 atmospheric dispersion modeling)
  - Guidance for modern fuel utilization (non-LOCA gap fractions).
  - Changes due to Regulatory Information Summaries (i.e., 06-04, 01-19).
  - Lessons learned from license reviews (i.e., clarify DFs and containment isolation as used in the FHA).
  - Clarify TEDE calculation terminology (i.e., EDEX vs. EDE).
  - Remove environmental qualification guidance from RG and refer to RG 1.89.

BWR Main Steam Line Aerosol Deposition Models

- RG 1.183 Rev. 0 does not provide an aerosol main steam line deposition model for the LOCA main steam line leakage pathway.
- DG-1389 provides guidance for the three deposition methods as acceptable in regulatory positions to be used in conjunction with revised seismic analysis.
- The first model utilizes select recommendations from the report, SAND 2008-6601, "Analysis of Main Steam Isolation Valve Leakage in Design Basis Accident Using MELCOR 1.8.6 and RADTRAD."
- Two additional models were developed as the staff addressed issues identified in RIS 2006-04, "Experience with Implementation of Alternative Source Terms."
  - Reconstitution of the AEB-98-03 settling velocity modeling parameters and developed the: (1) "multigroup method," and (2) numerical integration method, to address changing settling velocity distributions in downstream piping sections.

BWR Main Steam Line Aerosol Deposition Models (Cont'd)

- Clarifications for BWR MSIV leakage pathway
  - All steam lines are intact, assuming no break.
  - Delete regulatory position crediting containment sprays when crediting aerosol removal within the main steam lines.
  - No credit for holdup or aerosol deposition within the inboard main steam line.

Modified SAND 2008-6601 Method

Modified SAND 2008-6601 Method

- A MELCOR uncertainty analysis of the main steam line model was developed to calculate the aerosol deposition in the steam line and condenser.
- MELCOR is a well vetted, benchmarked state-of-the-art computer code that holistically models the pertinent phenomena for the LOCA.
- The MELCOR models considered the range of physical parameters (steam line configurations and designs) and aerosol physics parameters.
- The model considers following deposition mechanisms: gravitational aerosol settling, diffusiophoresis, thermophoresis, Brownian motion.
- The model also considers the effects of agglomeration, condensation and hygroscopicity.
- The uncertainty analysis used is very similar to those currently endorsed in RG 1.183 for modeling containment aerosol removal by natural processes and by sprays (See Regulatory Positions A-3.2 and A-3.3).
- The Modified SAND 2008-6601 method eliminates the use of steam dome concentrations as the source of BWR MSIV leakage for the first 2 hours of the accident evaluation.

Re-Evaluation of AEB 98-03 with Multi-group/Numerical Integration Method

Re-Evaluation of AEB 98-03 with Multi-group/Numerical Integration Method

- AEB 98-03 is utilized by many licensees in conjunction with RIS 2006-04, "Experience with Implementation of Alternative Source Term" to address staff concerns.
  1. Applied more conservative AEB-98-03 settling velocities.
  2. Developed the "20-Group Method" to account for the entire settling velocity distribution which the NRC has never endorsed.

- Re-evaluate AEB 98-03 in coordination with the Office of Research.[1]
  - Simplified AEB-98-03's model to just one known distribution (e.g., aerodynamic mass median diameter (AMMD)).
  - European State of the Art Report on Nuclear Aerosols (SOAR) .[2]
- Assessed methods for integrating the settling velocity distribution. [1]
  - (1) "Multi-group" (Monte Carlo Integration) and (2) Integration by parts.

1 - See ADAMS Accession No. ML21141A006. Note that while referenced in the draft RG, this document does not establish regulatory positions. For example, input parameters such as the AMMD, assumed in example calculations and statements regarding the validity of the existing 20-group method are not endorsed in the RG.

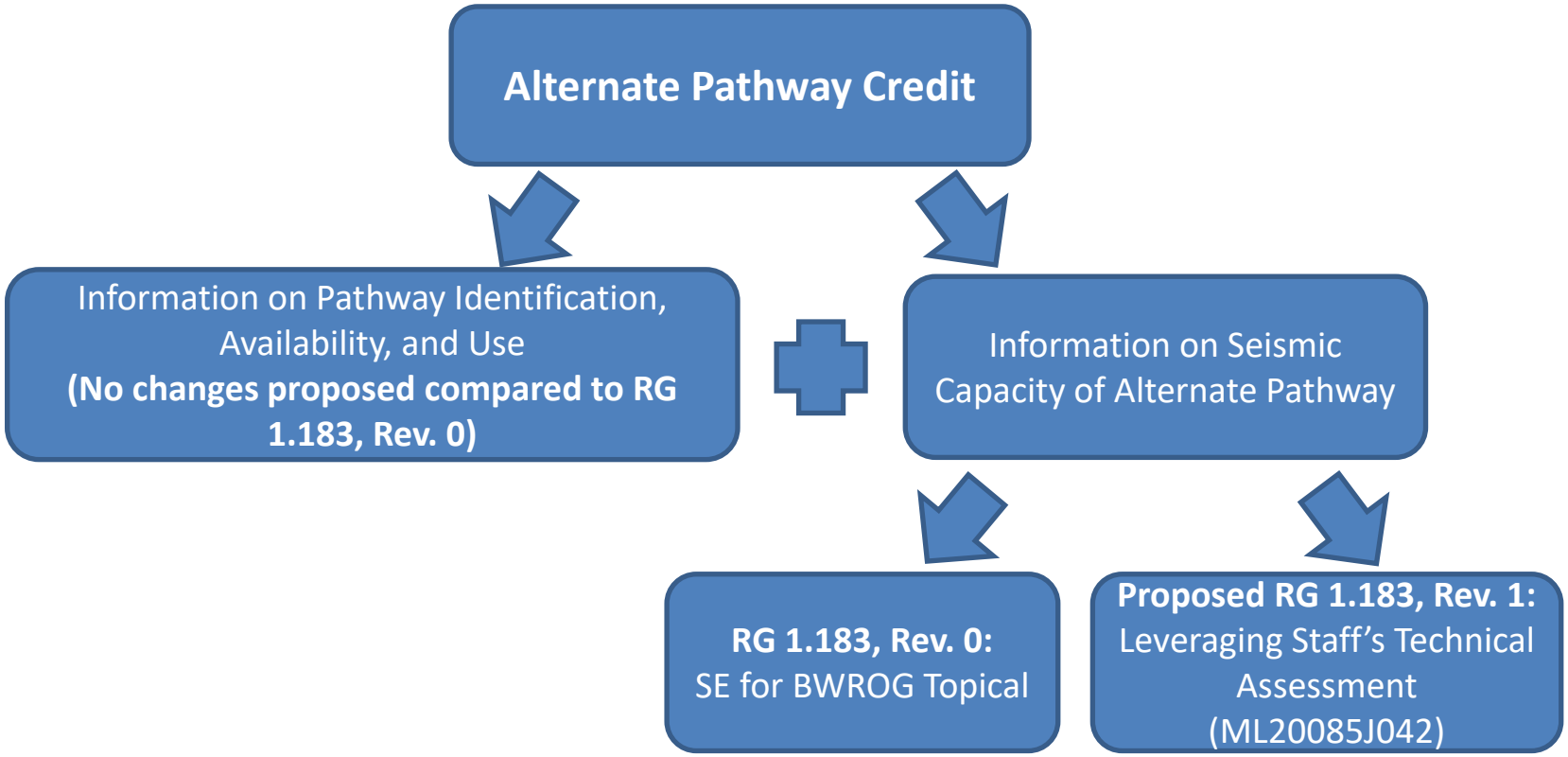
2 - Allelein, et. al., "State-of-the-Art Report on Nuclear Aerosols," NEA/CSNI/R(2009)5

Technical Assessment on Seismic Analysis for Alternate Drain Pathway

What is the Alternate Pathway in RG 1.183?

- Pathway for BWR MSIV leakage to main condenser via drain lines
- Holdup in main condenser -> fission product decay -> lower dose
  - RG 1.183 defines release rate from condenser (not leak tight)
- Credit for pathway results in quantitative change in calculated dose
  - Licensees request credit with docketed supporting information

Information for Alternate Pathway Credit



BWROG Topical Report Approach in Rev. 0

- BWROG NEDC-31858P (1993)
  - Non-nuclear seismic experience information
  - Generic seismic hazard curves

- NRC staff's safety evaluation issued in 1999 (ML010640286)
  - Included 9 limitations and conditions on use including:
    - Applicability of generic seismic hazard curves
    - Walkdowns of main steam piping by independent contractors

Proposed Approach in Rev. 1

- Based on NRC staff's technical assessment (ML20085J042)
- Presented to ACRS Subcommittees on July 23, 2021 (ML21223A034)
  - Based on operational insights, seismic capacity data, and risk insights over past 20 years
  - Identifies lower bound median fragility to encompass potential failure modes of alternate pathway SSCs
- FRN will solicit public comments on technical assessment concurrent with DG
- Docketed information to justify that plant is within the bounds of the technical assessment

Proposed Approach in Rev. 1 (Cont'd)

- Available dynamic analyses against plant-specific safe shutdown earthquake (deterministic) **OR**

- Risk-informed evaluation
  - Margin in code of record of alternate pathway SSCs
  - Sample confirmatory calculations for flexibility at major branch connections
  - Walkdowns sample alternate pathway SSCs
  - Gradation based on latest site-specific ground motion response spectrum

- NRC staff considers proposed approach to be streamlined

- Credit for pathway impacts licensee's analysis of record (AOR)
  - Licensee requests quantitative credit and includes it in its dose assessment
  - Draft Interim Staff Guidance (DRA-ISG-2021-01, ML21278A372) does not change the proposed AOR
  - NRC staff needs high confidence for accepting inputs that impact the AOR

Consideration of BWR MSIV Leakage Values

- RG 1.183 Rev. 1 provides 3 different BWR MSIV leakage models which can be used in conjunction with a streamlined strategy to credit a pathway to the condenser.

- NRC has approved BWR MSIV leakage of 200 scfh or below per MSIV with a total MSIV leakage of 400 scfh or below. Higher values will be considered on a case-by-case basis with sufficient justification.

- Maintaining BWR MSIV leakage at or below certain values is based on the following considerations:
  - Leakage in excess of 200 scfh per MSIV could be indicative of substantial valve defects
  - These values represent maximum values in existing fleet
  - 400 scfh is on the order of total containment leakage
  - Comparison to original design value of 11.5 scfh per MSIV

ATF, HBU, Increased Enrichment (LOCA)

ATF, HBU, Increased Enrichment (LOCA)

- Approach: for MHA-LOCA, utilize readily available information to support near-term licensing activities.
- Updating RG 1.183 Rev. 0 Tables 1, 2, and 4 (MHA-LOCA) with hybridized accident source term tables from SAND 2011-0128, "Accident Source Terms for Light Water Nuclear Power Plants Using High-Burnup of MOX Fuel."
  - Expanded to encompass near-term ATF design concepts[1] fuel burnup extension to 59 GWD/MTU max assembly-averaged discharge burnup (~68 GWd/MTU peak rod-average) and ^{235}U enrichments up to 8.0 wt%.
  - Considered impact of FFRD for the Appendix A assumptions[2].
  - Provide conditions and limitations of the report applicability for regulatory purposes.

- On going research efforts are underway to update the SAND 2011-0128 accident source term to accommodate higher burnup and increased enrichments for LOCA releases. However, completion of the updated analyses may not be finished before the update to the regulatory guide.

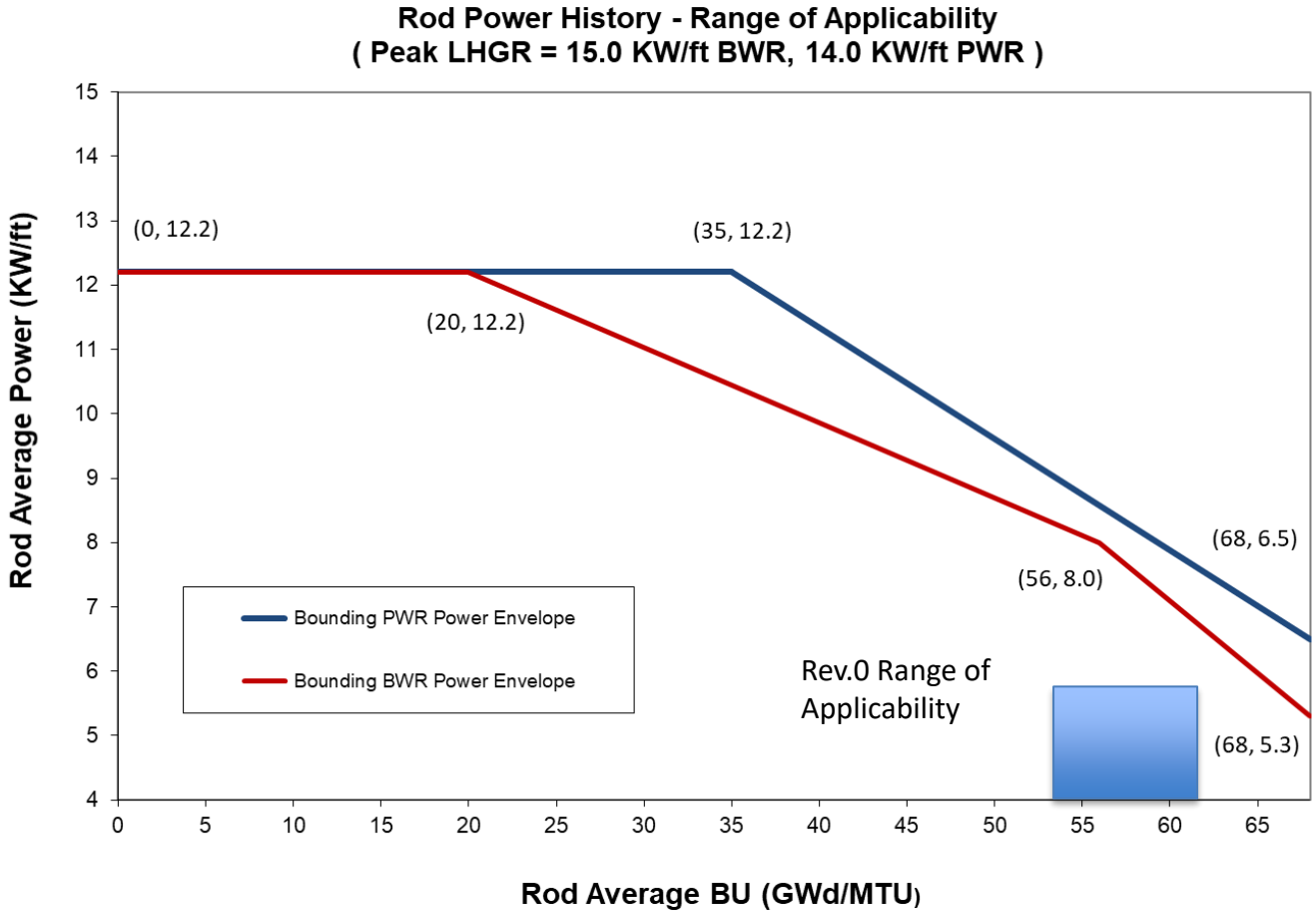
1- NRC Memorandum, "Applicability of Source Term for Accident Tolerant Fuel, High Burn Up and Extended Enrichment," dated May 13, 2020, ADAMS Accession Number ML20126G376
2- NRC Memorandum, "Letter Report on Evaluation of the Impact of Fuel Fragmentation, Relocation, and Dispersal for the Radiological Design Basis Accidents in Regulatory Guide 1.183 (ADAMS Accession No. ML21197A067)

Steady-State and Transient Release Fractions for Non-LOCA Analyses

Updates to Guidance

- Expand applicability of guidance
  - Modern fuel utilization
  - High burnup (68 GWd/MTU rod average burnup)
- Separate BWR and PWR steady-state releases
- Burnup-dependent transient fission gas release (FGR) correlations for prompt power increase accidents
- Analytical procedure for calculating revised steady-state release fractions

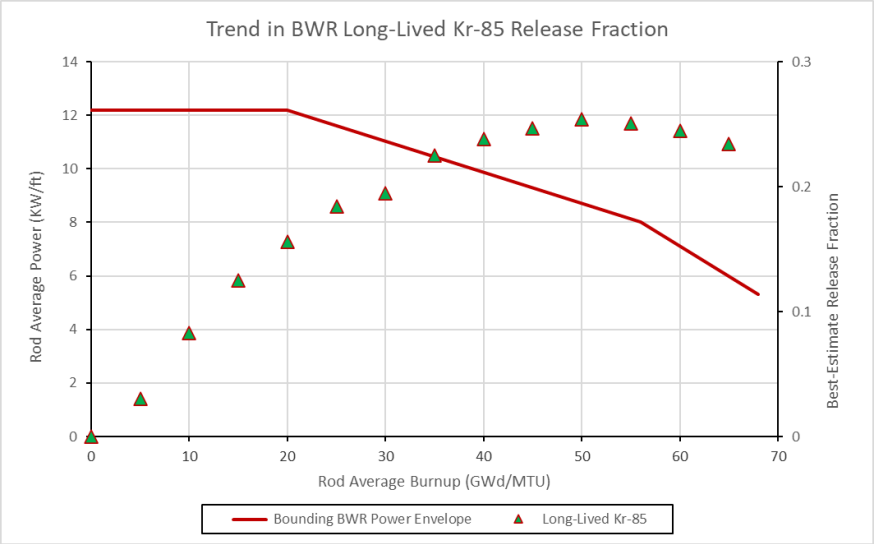
Expanded Range of Applicability

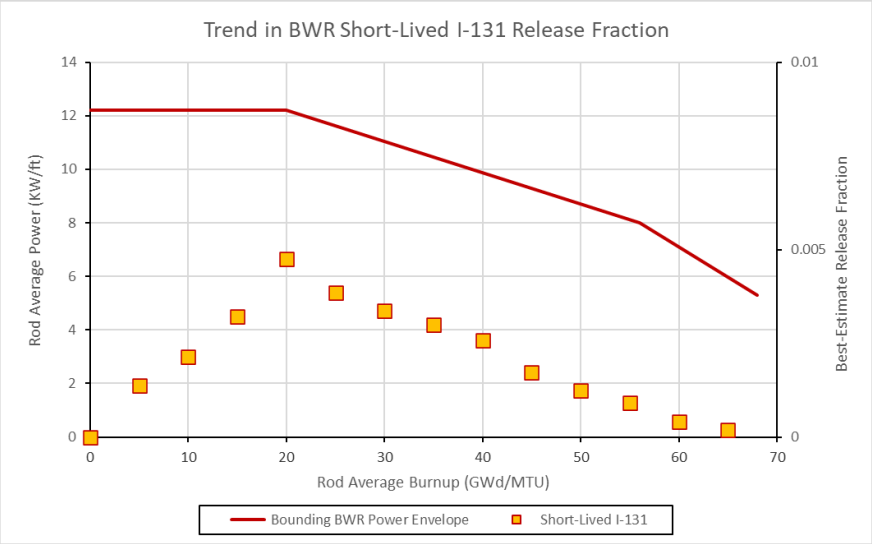


FAST Steady-State Release Fractions

- FAST-1.0.1 along with ANS-5.4 (2011) standard release model used to calculate bounding steady-state release fractions
  - Results sensitive to fuel rod design and operating history
    - BWR 10x10 and PWR 14x14 designs
  - Competing effects:
    - Expanded power operating domain promotes higher releases
    - Application of model uncertainties promote higher releases
    - ANS-5.4 (2011) standard promotes lower I-131 releases relative to ANS-5.4 (1982)

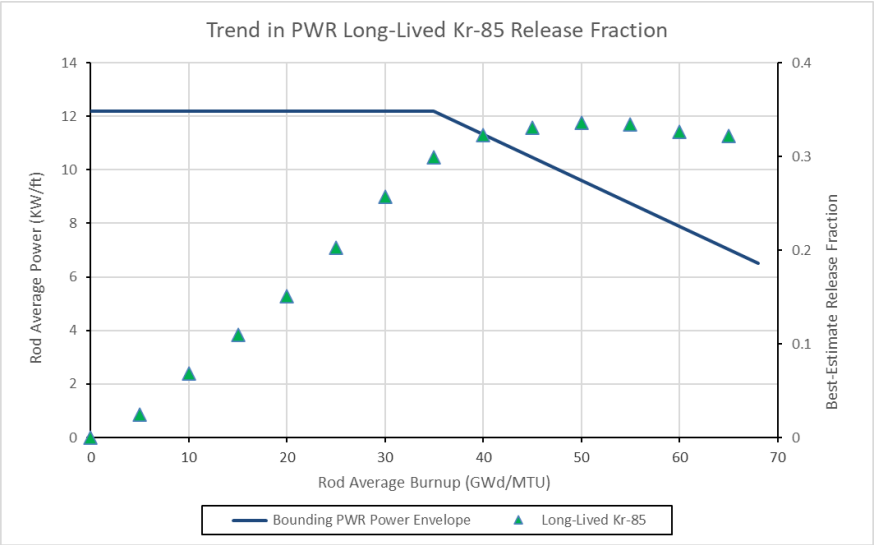
BWR Steady-State Release Fractions

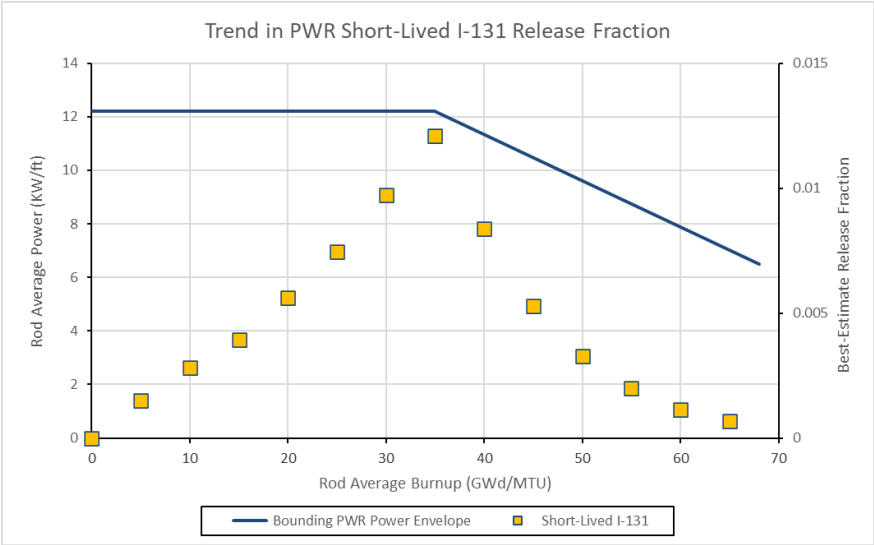




Group	Release Fraction
I-131	0.03
I-132	0.03
Kr-85	0.32
Other Noble Gases	0.03
Other Halogens	0.02
Alkali Metals	0.16

PWR Steady-State Release Fractions





Group	Release Fraction
I-131	0.07
I-132	0.07
Kr-85	0.40
Other Noble Gases	0.06
Other Halogens	0.04
Alkali Metals	0.20

Alternate Treatment for Cesium Release

- ANS-5.4 standard release model recommends a diffusion coefficient for cesium equivalent to twice that of Nobles

$$(\text{Release Fraction})_{\text{Cs-134, Cs-137}} = (\text{Release Fraction})_{\text{Kr-85}} * (2.0)^{0.5}$$

- Key observations from various research programs:
  - Notable differences in physical properties and chemical effects
    - Cesium reacts with other constituents to form relatively immobile compounds which accumulate along grain boundary and rim structure
    - Below 1200°C, cesium is a liquid and forms a film along grain
    - When released, cesium may react with the zirconium cladding to form more stable (i.e., non-gaseous) compounds
    - Nobles are chemically inert gas which readily diffuse to grain boundary, form interconnected bubbles, and then released

Alternate Treatment for Cesium Release (Cont'd)

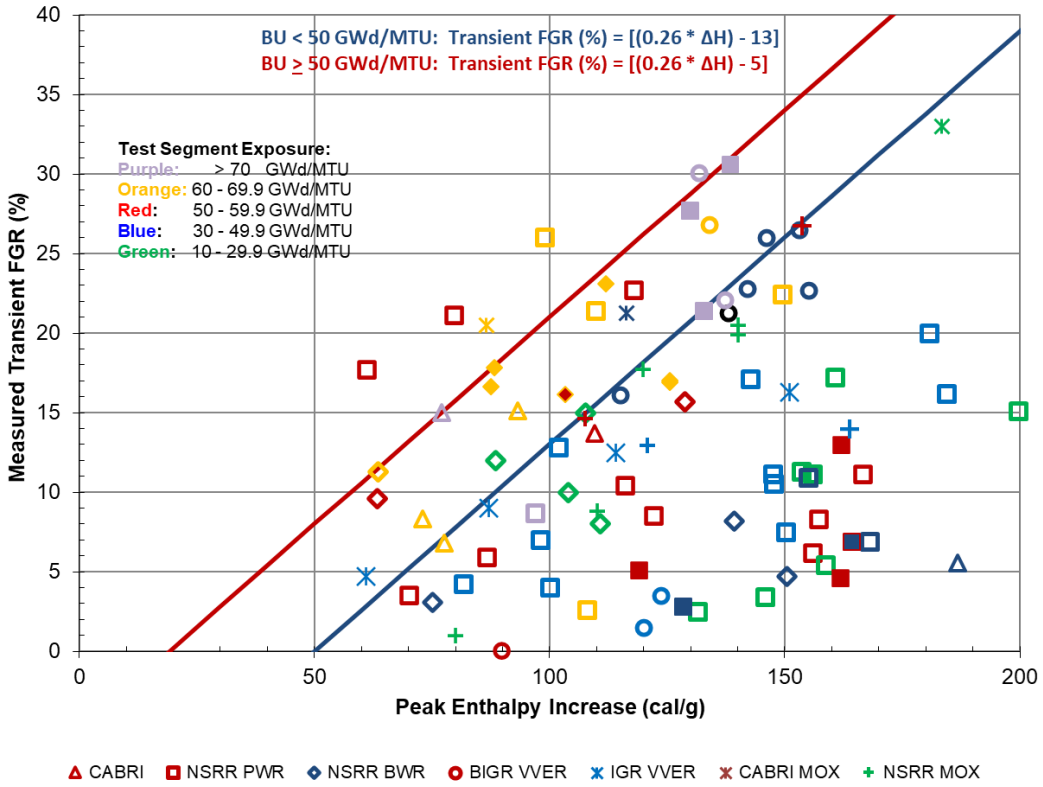
- Ratio of measured noble gas and cesium releases varies between 2.2/1 to 10/1 from the different studies

- Based on these key observations, a conservative lower bound of 2/1 was selected

$$(\text{Release Fraction})_{\text{Cs-134, Cs-137}} = (\text{Release Fraction})_{\text{Kr-85}} * (0.5)$$

Transient Fission Gas Release

- Fission gas released during prompt power excursion testing on irradiated fuel rod segments has been reported



Transient Fission Gas Release (Cont'd)

- TFGR is an **additional component** of radiological source term and should be considered in dose consequences
- TFGR is sensitive to local fuel burnup and deposited energy

pellet burnup < 50 GWd/MTU

$$TFGR = \text{maximum}\ [\ (0.26 * \Delta H) - 13) / 100,\ 0\ ]$$

pellet burnup > 50 GWd/MTU

$$TFGR = \text{maximum}\ [\ (0.26 * \Delta H) - 5) / 100,\ 0\ ]$$

where:

TFGR = transient fission gas release, fraction, and

ΔH = increase in radial average fuel enthalpy, Δ calories per gram.

- Correlation adjustment to account for differences in diffusion coefficients and half-life (i.e., 1/3 for short-lived nuclides)

Analytical Procedure

- An acceptable analytical procedure provided for calculating plant-specific steady-state release fractions
  - Significant reductions in steady-state release fractions achievable using plant-specific fuel rod designs and power operating domains
  - Fuel vendors may use their approved fuel rod thermal-mechanical codes
- Provides flexibility and dose margin to accommodate more aggressive fuel utilization, HBU, etc…
- Reduces regulatory uncertainty

HBU, IE, and ATF Applications (Non-LOCA DBAs)

- DG-1389 provides revised steady-state release fractions applicable to an expanded operating domain
  - Supports HBU/IE applications
  - Relaxation to both short-term (e.g., I-131) and long-term (e.g., Cs-137) radionuclide releases achieved through more rigorous engineering
- DG-1389 provides an acceptable analytical procedure for calculating application-specific release fractions
  - Supports HBU/IE and ATF applications
  - Relaxation and flexibility may be achieved
- DG-1389 TFGR correlations + RG 1.236 coolability limits provide a means to address FFRD for postulated CRE/CRD accidents
  - Supports HBU/IE applications

Revised Fuel Handling Accident (FHA)

Revised Fuel Handling Accident

- FHA can require significant staff resources, and in some cases has become a limiting accident, despite its low safety and risk significance.

- Revisited the original studies from the early 1970's forming the technical basis for the FHA and ACRS recommendations[1].

- Model improvements established from the current understanding of reactor fuel pin physics and iodine chemistry under the environmental conditions in which fuel handling operations are taking place.[2]
  - Two-phase model:
    1. Initial gas gap release – Elemental, I_2, and Organic Iodine, CH_3.
    2. Re-evolution – Assumes CsI dissociates into the pool water and re-evolves as I_2 over the accident duration.

- Concluded considerable margin exists regarding the scrubbing effects of iodine in the spent fuel or reactor pool and that the current staff DBA FHA fission product transport model can be refined while still maintaining conservatism.

1 - 473rd ACRS Full-committee meeting, held June 7, 2000, the committee members expressed concern with the staff's treatment of iodine speciation and modeling the release as a puff release.

2 - Memo from RES to NRR, "Closeout to Research Assistance Request for Independent Review of Regulatory and Technical Basis for Revising the Design-basis Accident Fuel Handling Accident," November 23, 2019 (ML19270E335)

Draft Revisions to Current DG-1389

- **3.2 <u>Release Fractions</u>**
  - "Applicability of Source Term for Accident Tolerant Fuel, High-Burnup and Extended Enrichment," (Ref. 25) in part, documents the applicability of SAND 2011-0128 for burnups up to 68 gigawatt-days per metric ton of uranium (GWd/MTU) peak rod average and enrichments up to 8 percent for certain near-term ATF designs which include chromium-coated cladding and chromia-doped fuel, but do not include Iron-Chromium-Aluminum alloy cladding.

- **5.3 <u>Atmospheric Dispersion Modeling and Meteorology Assumptions</u>**
  - RG 1.145 and RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," issued June 2003 (Ref. 38), should be used if the FSAR χ/Q values are to be revised or if values are to be determined for new release points or receptor distances or release characteristics that affect plume rise. In addition to calculating control room χ/Q values, the modeling methodology outlined in RG 1.194 may be modified to estimate offsite χ/Q values at offsite boundaries out to distances of 1,200 m as long as the procedures are consistent with the Regulatory Positions in Part C of RG 1.145.

Note: These draft revisions have been prepared and are being released to support ongoing public discussions. These revisions have not been subject to NRC legal reviews and approvals, and its contents are subject to change and should not be interpreted as official agency positions.

Key Takeaways

- DG-1389 provides guidance for three acceptable main steam line aerosol deposition models.
- DG-1389 provides guidance to support a streamlined approach for licensees to credit an alternate pathway for BWR MSIV leakage to the main condenser.
- The staff are prepared to review near-term ATF designs and burnups up to 68 GWd/MTU peak rod-average and U-235 enrichments up to 8.0 wt%. Future updates to the guide are expected to accommodate higher burnups and enrichments.
- The range of applicability has been expanded to support modern fuel utilization. The impact of this expanded operating domain on radiological source terms has been minimized by applying more rigorous engineering judgement.
- An acceptable analytical procedure for predicting plant-specific radionuclide release fractions has been included and provides flexibility and margin recovery.

Looking Forward

- Issue DG-1389 for 60-day public comment – April 2022
- Staff review and disposition of public comments
- Update draft RG 1.183 Rev. 1, as necessary
- OGC review of final draft
- ACRS Full Committee Meeting
- Issuance of RG 1.183 Rev. 1 – Nov/Dec 2022

Discussion/Feedback

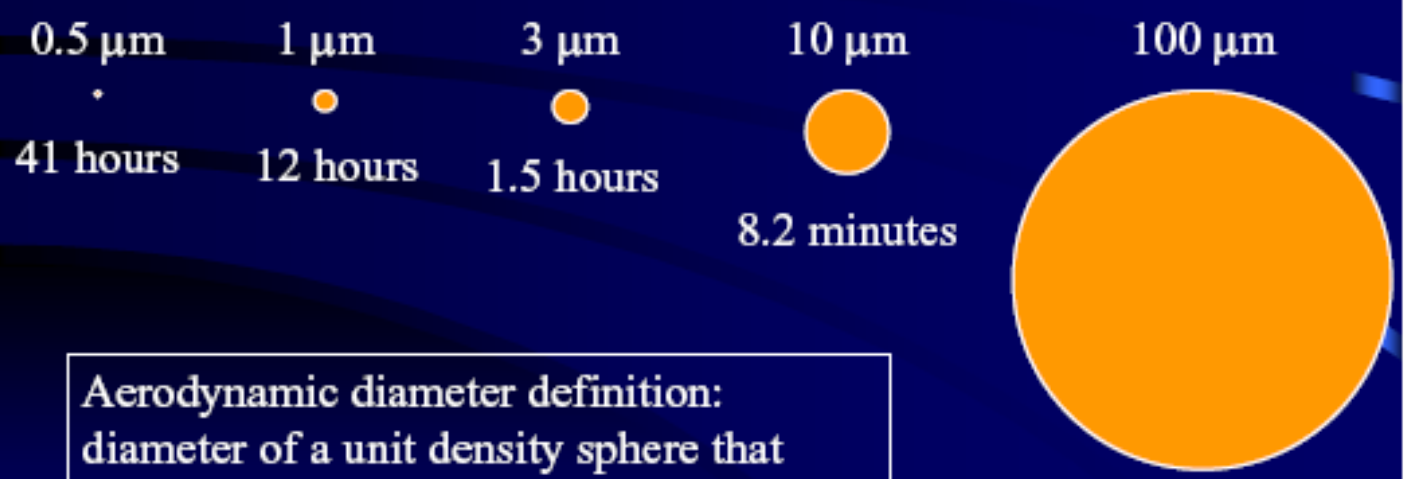
Acronyms

ACRS – Advisory Committee on Reactor Safeguards
AEB – accident evaluation report
AMMD - aerodynamic mass median diameter
AOO – anticipated operational occurrence
AOR – analysis of record
ATF – accident tolerant fuel
AST – alternative source term
BWR – boiling water reactor
BWROG – BWR Owner's Group
CFR – Code of Federal Regulations
Cs – cesium
DANU – Division of Advanced Reactors and Non-power Production and Utilization Facilities
DBA – design basis accident
DE – Division of Engineering
DF – decontamination factor
DG – draft guide
DPO – Differing Professional Opinion
DRA – Division of Risk Assessment
DSS – Division of Safety Systems
EDE – effective dose equivalent
EDEX – effective dose equivalent (for external exposures)
EOL – end of life
FAST – Fuel Analysis under Steady-State and Transients
FFRD – fuel fragmentation, relocation, and dispersal
FGR – fission gas release
FHA – fuel handling accident
FRN – Federal Register Notice
FSAR – final safety analysis report
GMRS – ground motion response system
GSD – geometric standard deviation
GWD/MTU – gigawatt day per metric ton uranium
HBU – high burnup
IE – Increased Enrichment
Kr - krypton
KW/ft – kilowatt per foot
LAR – license amendment request
LHGR – linear heat generation rate
LOCA – loss of coolant accident
LWR – light water reactor
MHA – maximum hypothetical accident
MOX – mixed oxide
MSIV – main steam isolation valve
MSL – main steam line
NRR – Office of Nuclear Reactor Regulation
OGC – Office of General Counsel
PNNL – Pacific Northwest National Laboratory
PWR – pressurized water reactor
RADTRAD – RADionuclide, Transport, Removal, and Dose Estimation
RCS – reactor coolant system
RES – Office of Research
RG – regulatory guide
RIS – Regulatory Issue Summary
Scfh – standard cubic feet per hour
SE – safety evaluation
SSC – systems, structures, and components
TEDE – total effective dose equivalent
TFGR – transient fission gas release
TID – technical information document
TMI – Three Mile Island

Backup Slides for
Modified SAND 2008-6601 Method

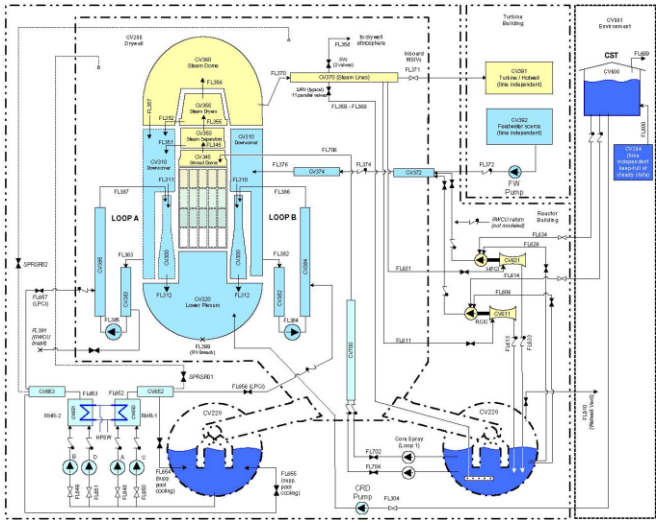
Particle Settling in Still Air

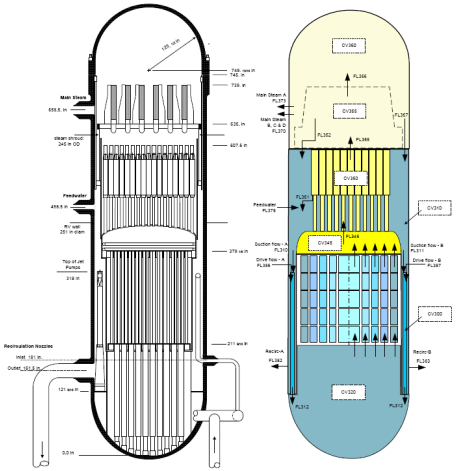
Time to settle 5 feet by unit density spheres

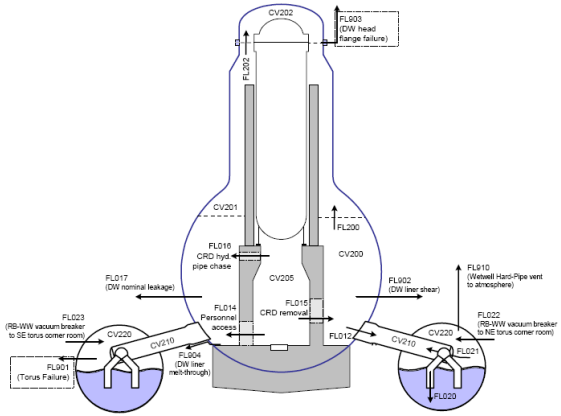


Aerodynamic diameter definition: diameter of a unit density sphere that settles at the same velocity as the particle in question

MELCOR Analysis of Full Plant DBA

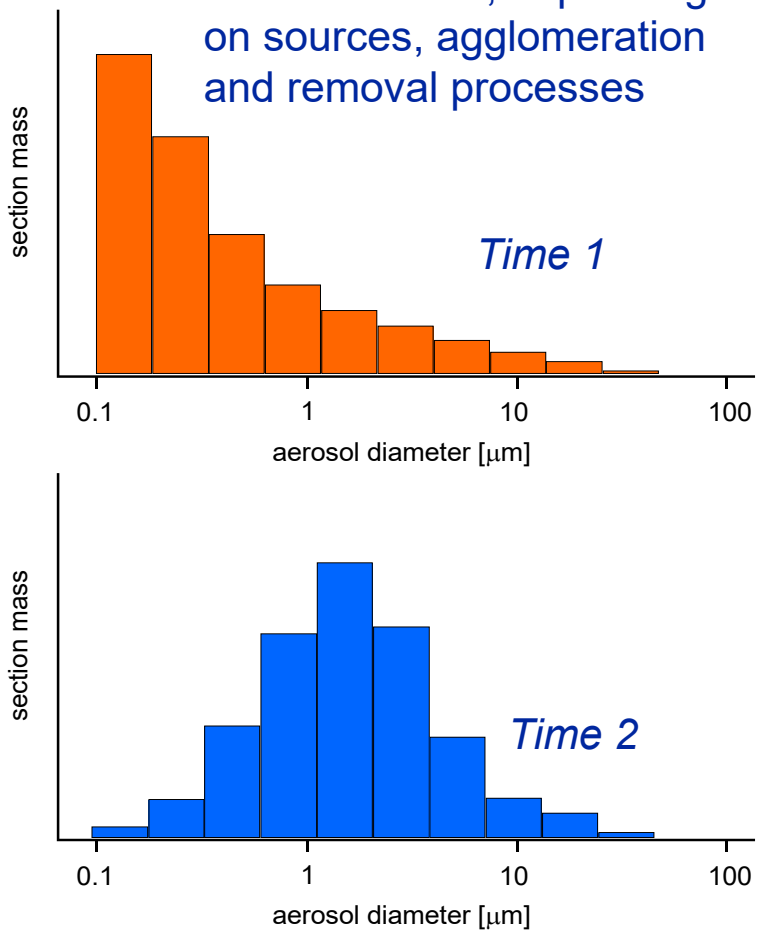






SAND 2008-6601 Method (cont'd)

Aerosol size distribution evolves in time, depending on sources, agglomeration and removal processes

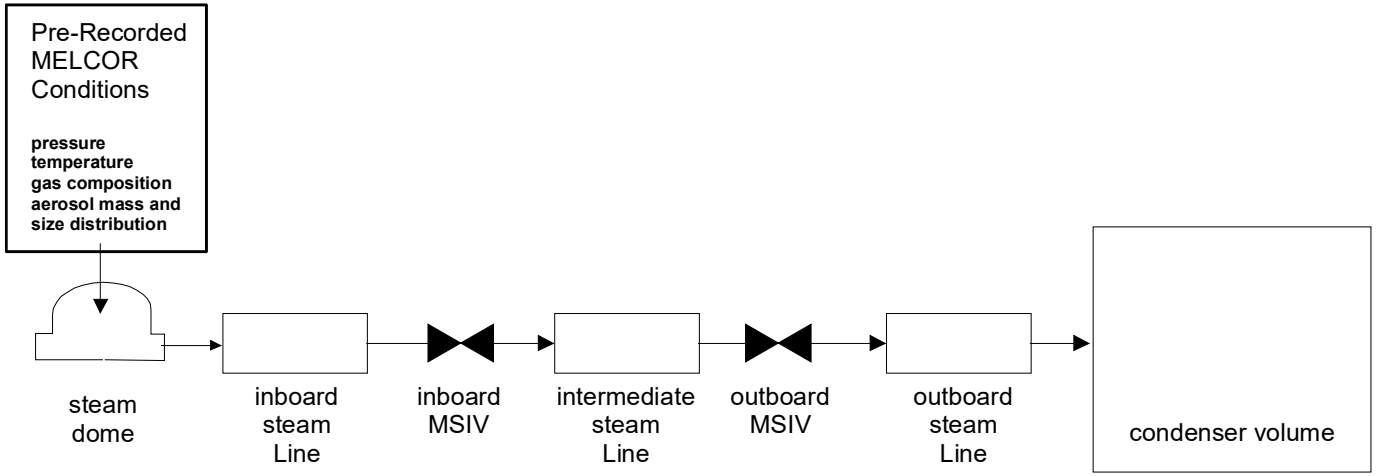


- MAEROS sectional model of Gelbard
- Particles grow in size
  - Agglomeration
  - Water condensation
- Particle fallout by gravitational settling
- Particle deposition processes
  - Thermophoresis
  - Diffusiophoresis
  - Brownian motion
- Cs chemisorption in RCS modeled
  - Iodine from CsI re-volatilizes when reheated

Validation Experiments

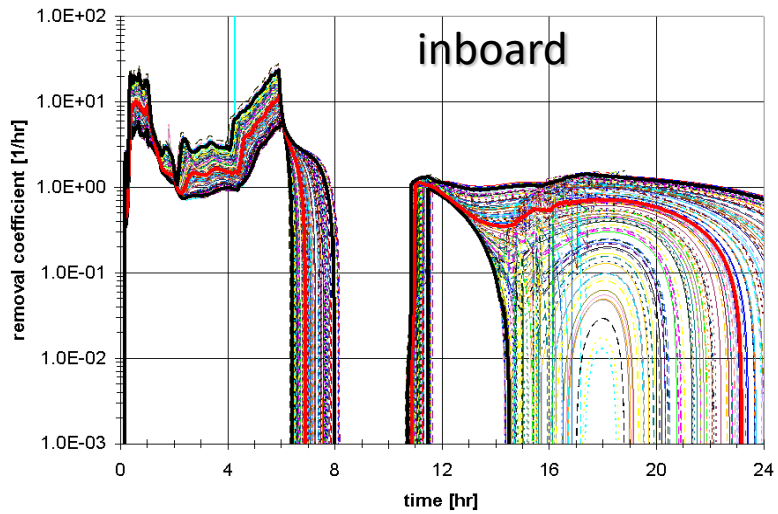
PANDA, DEHBI, CVTR, FALCON, LACE-LA4, ABCOVE, Wisconsin flat plate

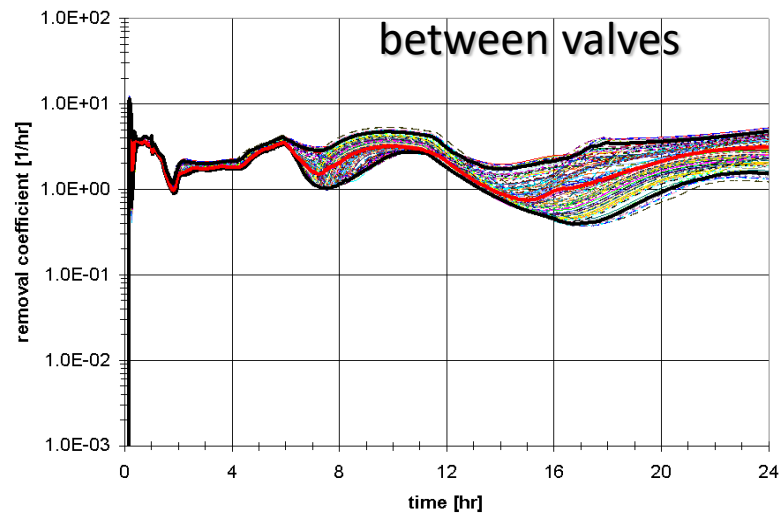
SAND 2008-6601 Method (cont'd)

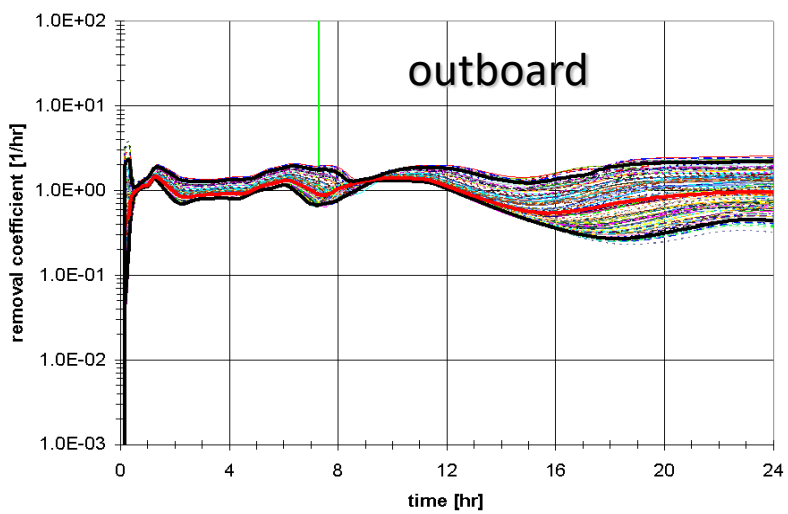


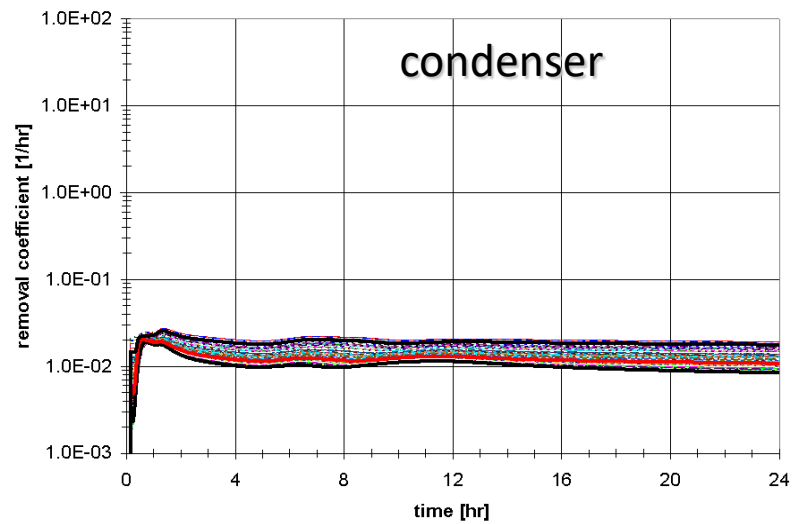
- Reduced nodalization
- Boundary conditions from full-plant MELCOR analysis
- Monte Carlo sampling used to derive distributions

Illustrative Example of Results from Deposition Analysis









SAND 2008-6601 Results
MSL and Condenser Removal Coefficients

MSL section	0 - 10 (hr)	10^+ (hr)
in-board	0*	0*
between MSIVs	1.8	1.0
out-board	1.0	0.7
condenser	0.015	0.012

Note. removal coefficients are given in 1/hr

* Not credited because of thermal bi-directional flow, turbulence, containment boundary

- Based upon Recommendation 6.4 of SAND 2008-6601 for post-reflood conditions (taken from Table 6-1)
- Lambdas decrease as remaining aerosols becomes smaller

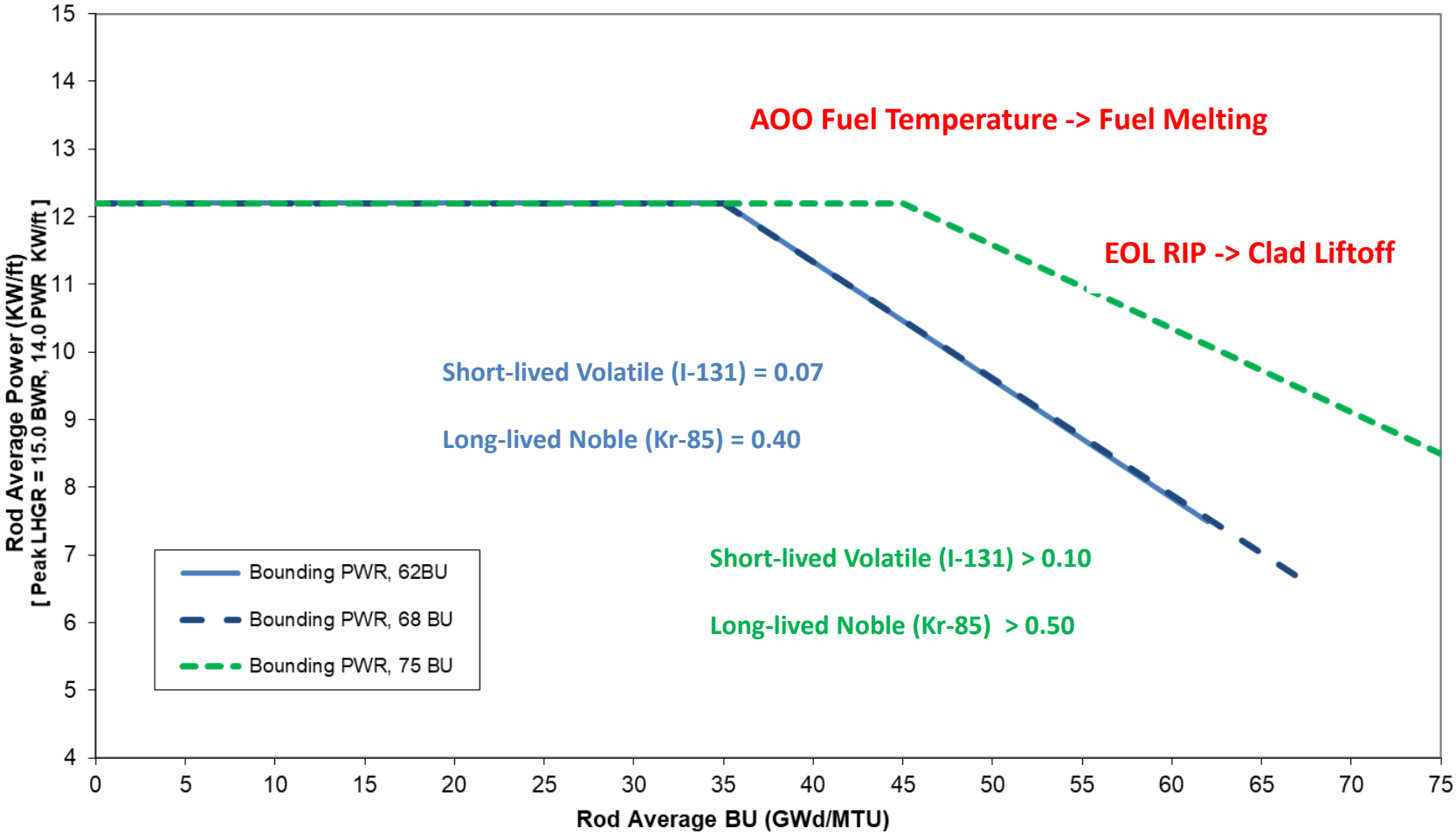
Backup Slides for Steady-State and Transient Release Fractions for Non-LOCA Analyses

Challenges to 75 GWd/MTU

- Approved fuel rod thermal-mechanical performance model validated up to 75 GWd/MTU
  - Quantified model uncertainties

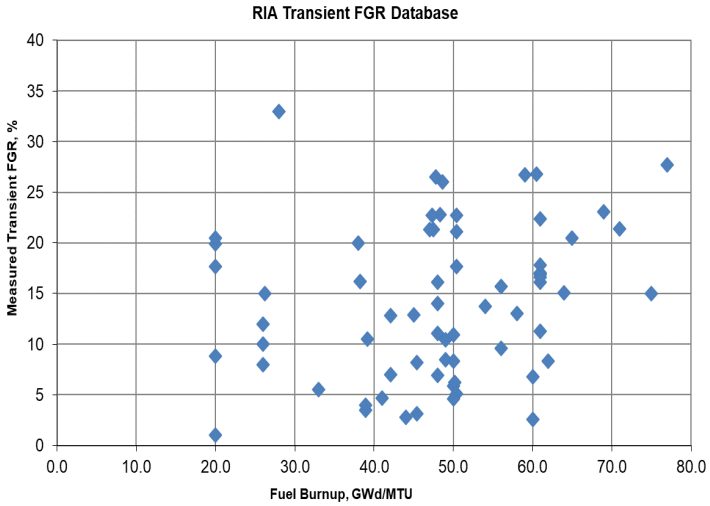
- Defined fuel rod power operating domain
  - PWR and BWR fuel management scoping studies needed to define realistic fuel rod power history
  - Predicted released sensitive to power history

Impact of Burnup Extension



Extent of Empirical Database

- PNNL completed a critical assessment of the various models within the FAST fuel performance code.
  - Most models validated up to 75 GWd/MTU
- ANS-5.4 Release Model limited to 70 GWd/MTU
- TFGR database limited to ~68 GWd/MTU



Use by Non-LWRs

- Guidance which is not LWR-specific could provide useful information on radiological consequence analyses and development of source terms
  - Section 2: “Attributes of an Acceptable Accident Source Term”
  - Section 4: “Dose Calculation Methodology”
  - Section 5: “General Considerations”

ACRS Meeting on Proposed Draft RG 1.183

Frankie Pimentel, Sr. Project Manager - Engineering & Risk, NEI

Greg Broadbent, Senior Staff Engineer - Corporate Nuclear Analysis, Entergy

March 16, 2022

NUCLEAR ENERGY INSTITUTE

Initial Concerns from Review of Proposed Draft RG 1.183

The overall impact on sample BWR/PWR plants should be evaluated prior to issuing

- Industry strongly recommends that NRC staff provide information indicating that aspects of the updated guidance, such as release fractions and timing, have been considered for impacts to resulting analyzed doses
- Proposed changes are as significant as when the guidance transitioned from the TID Source Term to the Alternative Source Term
- As proposed, the conservatisms and changes incorporated in this revision precludes its use by many plants that are interested in implementing near-term ATF design concepts, fuel burnup extension to 68 GWd/MTU (peak rod average), and 235U enrichments up to 8.0 wt%

Initial Concerns from Review of Proposed Draft RG 1.183

Majority of BWR plants may not be able to adopt Rev. 1 without significant licensing basis changes or plant modifications

- The new BWR/PWR core inventory release fractions and timing may have considerable impact on the margin to the Control Room exposure criteria
  - BWR iodine release fraction nearly doubles for LOCA/MHA
  - BWR tellurium (parent of iodine) release fraction increases significantly
  - Basis of the changes in the BWR release fractions is not understood
    - Core damage scenarios heavily weighted by SBO
    - Suppression pool scrubbing apparently not credited
- New BWR main steam line model applies more conservative assumptions than previous models
- Lowered CMT and MSIV leakage rates
- Higher occupational doses to support lower leakage requirements
- Reduced control room in-leakage allowances

Initial Concerns from Review of Proposed Draft RG 1.183

Option to maintain compliance to Rev. 0

- Clear validation regarding continued acceptance of Rev. 0
- Any additional justification needed
  - non-LOCA gap fractions
  - MSL deposition model (Rev. 0 is silent on acceptable models)

Option for selective implementation of Rev. 1

- Describe acceptable paths for selective implementation
- Acceptable on an accident basis (e.g., fuel handling accident only)

Feedback from Review of Proposed Draft RG 1.183

Revised Fuel Handling Accident

- Industry recognizes the new FHA model developed by the Staff
  - Based on thorough review of decades of historical information and current understanding of iodine chemistry
  - Internally motivated – not industry initiated
- Resulting model is a more accurate characterization of the low consequences of this accident

Other

Communication Regarding DPO Pertaining to MSIV Leakage

- No updates have been provided on the status of the DPO discussed during the November 2021 ACRS meeting on ISG-2021-01
- Will the resolution of the DPO affect the changes regarding the Revised Seismic Analysis of the Alternative Drain Pathway introduced in the proposed revision to RG 1.183
- Will the resolution of the DPO affect the schedule for issuing the revision to RG 1.183

RE: ACRS Subcommittee Meeting on Revision of Regulatory Guide 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

March 15, 2022

In preparation for tomorrow's ACRS Subcommittee Meeting, I submit the following pre-meeting public comments:

While I appreciate the invitation and opportunity to attend another public meeting, I do not believe the NRC has been responsive to my prior RG 1.183/DG-1199 public comments. The NRC posted my public comments (and questions) in ADAMS (ML20343A064 and ML20351A321), but has not provided documented responses. In effect, this negates my efforts and the intent of NRC public meetings.

As recently informed, I understand that documented responses are now forthcoming.

My initial comments on "*DRAFT REGULATORY GUIDE DG-1389, Proposed Revision 1 to Regulatory Guide 1.183"* are posted (bulleted) below relevant sections.

Sincerely,
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1020 Station Blvd. #212
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Lead Emergency Management Specialist—Constellation (formerly Exelon Generation)
Former NRC Licensed Senior Reactor Operator/Operations Shift Manager
—Acting expressly as a member of the public.

DRAFT REGULATORY GUIDE DG-1389, Proposed Revision 1 to Regulatory Guide 1.183

B. Discussion

This revision of the guide (Revision 1) addresses new issues identified since the guide was originally issued. These include (1) using the term maximum hypothetical accident (MHA)[1] loss-of-coolant accident (LOCA) to define the accident described in the applicable regulations in Section A above . . .

Footnote 1: *The maximum hypothetical accident (MHA) (also referred to as the <u>maximum credible accident</u>) is that accident whose consequences, as measured by the radiation exposure of the surrounding public, would not be exceeded by any other accident whose occurrence during the lifetime of the facility would appear to be credible. As used in this guide, the term "LOCA" refers to any accident that causes a loss of core cooling. The MHA LOCA refers to a loss of core cooling resulting in substantial meltdown of the core with subsequent release into containment of appreciable quantities of fission products. <u>These evaluations assume containment integrity with offsite hazards evaluated based on design basis containment leakage</u>.*

- The term "*maximum hypothetical accident (MHA)"* appears misleading. Because Regulatory Guides are not regulations, it seems inappropriate to use an NRC Regulatory Guide to bound applicable regulations to the DG-1389 definition of the MHA. Otherwise stated, the NRC does not have the authority to redefine the applicability of federal nuclear safety regulations (or the current licensing basis (CLB) of nuclear power plants) by using Regulatory Guide 1.183. For example, the requirements of 10 CFR 50, Appendix A, General Design Criterion—19 are not limited to a LOCA or the MHA described in DG-1389.

- The term "*maximum credible accident* (MCA)" was developed by the Atomic Energy Commission (TID-14844, *Calculation of Distance Factors for Power and Test Reactor Sites*) to quantify the threat of nuclear power plants in terms of the worst-case possible accident release of radiological material to the environment. In theory, by quantifying the worse-case radiological release, nuclear power plants would be built safe distances from populations. Subsequently, plant designers used the "maximum credible accident" to design safety systems, such that they would prevent or mitigate the quantified threat, thereby, ensuring compliance with 'source term' regulations. The AEC authors of TID-14844 developed, what they thought was, a conservative example to describe the "*maximum credible accident,"* but they intentionally did not define it, stating:

    - *"Consideration of these as well as other aspects of* [radiological] *hazards evaluation involves so many different situations and such complex technological problems that it would be quite impossible to anticipate and answer all questions that will arise."*

    - *"Designers of reactors are expected to examine all significant aspects of the hazards and safety problem they believe are appropriate to the particular situation with which they are dealing. In any case, the designer and/or, applicant bears the responsibility for justifying all the assumptions and methods of calculation used in a* [radiological] *hazards evaluation. The fact that aspects of the problem are not considered in the example set forth here, does not in any way relieve the designer and/or applicant of the responsibility for carefully examining, in his particular case, every significant facet of the hazards and safety problem."*

- As stated in RG 1.183 Revision 0, *"Since the publication of TID-14844* [1962]*, significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents."* These advances were made as the result of severe accident studies conducted by or for the NRC. Conclusions from studies include the following:

    - "*As in the DBA-LOCA class, the doses from "melt-through" releases (involving thousands of curies) generally would not exceed even the most restrictive PAG beyond about 10 miles from a power plant. The upper range of the core-melt accidents is categorized by those in which the* <u>*containment catastrophically fails*</u>

and releases large quantities of radioactive materials directly to the atmosphere because of over-pressurization or a steam explosion. These accidents have the potential to release very large quantities (hundreds of millions of curies) of radioactive materials. There is a full spectrum of releases between the lower and upper range with all of these releases involving some combination of atmospheric and melt-through accidents. These very severe accidents have the potential for causing serious injuries and deaths." [(1978) NUREG-0396 (EPA 520/1-78-016)]

- *"The accident at TMI demonstrated the reality of the risk, previously only theoretically assessed, of accidents that result in substantial degradation and melting of the core. This risk arises from the fact that core-degradation accidents can lead to containment failure and the eventual release of large amounts of radioactivity to the environment."* [(1980) NUREG-0660 (Vol. 1), "NRC Action Plan Developed as a Result of the TMI-2 Accident"]

- *"In examining the source terms . . . one must assess the probability of escape from the containment under routine use conditions or in any postulated accident situation.* [(1983) NUREG/CR-3332, *"A Textbook on Environmental Dose Analysis"*]

- *"Conclusion 9: Containment performance (survival, failure, or bypass), which is described by input parameters in all current source term codes, is a major factor affecting source terms."* [(1986) NUREG-0956, *"Reassessment of the Technical Bases for Estimating Source Terms"*]

- *"The consequences of severe reactor accidents depend greatly on containment safety features and containment performance in retaining radioactive material. The early failure of the containment structures at the Chernobyl power plant contributed to the size of the environmental release of radioactive material in that accident. Maintaining the integrity of the containment can affect the source term by orders of magnitude. The NRC's 1986 reassessment of source term issues reaffirmed that containment performance "is a major factor affecting source terms.""* [(1990) NUREG-1150, "*Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*"]

- *"Shortly after the accident at Three Mile Island, the NRC initiated a program to review the adequacy of the methods available for predicting the magnitude of source terms for severe reactor accidents. After considerable effort and extensive peer review, the NRC published a report entitled "Reassessment of the Technical Bases for Estimating Source Terms," NUREG-0956. As expected, the magnitude of the source term varies between different accident progression bins depending on whether or not containment fails, when it fails, and the effectiveness of engineered safety features (e.g., BWR suppression pool) in mitigating the release. However, within an accident progression bin, which represents a specific set of accident progression events, the uncertainty in predicting severe accident*

phenomena is great." [(1990) NUREG-1150, "*Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*"]

- *"With regard to the lack of treatment of some severe accident phenomena and containment failure modes, it would not be appropriate to attempt to identify or analyze all containment failure modes or scenario pathways. It is the intent of the latest sequence selection, reported in NUREG/CR-4624, to analyze at least the most risk significant of these pathways*." [(1990) NUREG-1150, "*Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*"]

- These NRC studies conclude that containments will fail in credible nuclear accidents. Regretfully, we no longer need to rely on theoretical studies. We watched the containments at Fukushima Daiichi catastrophically fail.

  - *Despite the efforts of the operators at the Fukushima Daiichi nuclear power plant to maintain control, the reactor cores in Units 1–3 overheated, the nuclear fuel melted and the three containment vessels were breached. Hydrogen was released from the reactor pressure vessels, leading to explosions inside the reactor buildings in Units 1, 3 and 4 that damaged structures and equipment and injured personnel. Radionuclides were released from the plant to the atmosphere and were deposited on land and on the ocean. There were also direct releases into the sea.* [*The Fukushima Daiichi Accident, IAEA Report by the Director General*]

  - *The common cause failures of multiple safety systems resulted in plant conditions that were not envisaged in the design. Consequently, the means of protection intended to provide the fourth level of defense in depth, that is, Prevention of the progression of severe accidents and mitigation of their consequences, were not available to restore the reactor cooling and to maintain the integrity of the containment.* [*The Fukushima Daiichi Accident, IAEA Report by the Director General*]

  - *The confinement* [containment] *function was lost as a result of the loss of AC and DC power, which rendered the cooling systems unavailable and made it difficult for the operators to use the containment venting system. Venting of the containment was necessary to relieve pressure and prevent its failure. The operators were able to vent Units 1 and 3 to reduce the pressure in the primary containment vessels. However, this resulted in radioactive releases to the environment. Even though the containment vents for Units 1 and 3 were opened, the primary containment vessels for Units 1 and 3 eventually failed. Containment venting for Unit 2 was not successful, and the containment failed, resulting in radioactive releases.* [*The Fukushima Daiichi Accident, IAEA Report by the Director General*]

- Now that we know containments have and will fail in credible nuclear accidents, it is important to understand why they may fail again.

1.1.3 Integrity of Facility Design Basis

The DBA source term used for dose consequence analyses is a fundamental assumption and the basis for much of the facility design. Additionally, many aspects of an operating reactor facility are derived from the radiological design analyses that incorporated the TID-14844 accident source term.

- Otherwise stated, nuclear power plants were constructed with physical barriers and safety systems that were specifically designed to prevent and mitigate the TID-14844 "*maximum credible accident*" source term release example. As we now know that is no fault of TID-14844, its example of the "*maximum credible accident*" was far from conservative; it incorrectly assumed that containments would not fail. Unfortunatley, the designers of nuclear power plants did not fulfil their "*responsibility for carefully examining, in his* [each] *particular case, every significant facet of the* [radiological] *hazards and safety problem."* Instead, they designed nuclear plants, and their safety systems, assuming containments would never fail. This is, in part, the fault of the AEC and now the NRC.

Background

An accident source term is intended to represent a major accident involving significant core damage not exceeded by that from any other credible accident. NRC staff experience in reviewing license applications has indicated the need to consider other accident sequences of lesser consequence but higher probability of occurrence. Facility-analyzed DBAs are not intended to be actual event sequences; rather, they are intended to be surrogates to enable deterministic evaluation of the response of engineered safety features (ESFs). These accident analyses are intentionally conservative to compensate for known uncertainties in accident progression, fission product transport, and atmospheric dispersion.

- After studying containment failures for the past 40 years and watching the containments at Fukushima Daiichi catastrophic fail, the NRC required that similar U.S. designed nuclear power plants install hardened containment vents to reduce the risks posed to the public from credible containment failure accidents.

- Now that we know containments will fail, it is important to understand why venting (intentionally releasing) large quantities of highly radioactive material directly to the environment would be required to reduce the risk to the public.

  - Nuclear power plants were designed and built based on the defense in depth philosophy. They were designed with four (4) physical barriers, such that, each barrier must fail before there would be a significant radiological release to the environment. The first barrier is the uranium-oxide fuel pellet. The second barrier is the fuel pellet cladding. The reactor coolant system provides a third barrier to fission product release. The final and ultimate barrier to fission product release is the reactor containment.

(1) The first barrier proved to be ineffective when the NRC discovered that "*high burnup fuel pellets could fragment, relocate axially and possibly disperse outside of the fuel rod during postulated design-basis accidents including, but not limited to, LOCA*." This means that lesser (clad failure) accidents may have larger radiological release source terms than more severe (core melt) accidents. [SECY-15-0148, *Evaluation of Fuel Fragmentation, Relocation and Dispersal Under Loss-Of-Coolant Accident (LOCA) Conditions Relative to The Draft Final Rule on Emergency Core Cooling System Performance During A LOCA* (November 30, 2015)]

(2) The second barrier, the fuel cladding, is probably the costliest 'engineering mistake.' This design error, not the tsunami, created the high concentrations of hydrogen that exploded, causing the catastrophic containment failures at Fukushima Dai-ichi.

"During accident conditions when the core materials are inadequately cooled, the fuel cladding (zirconium alloy) can overheat which promotes and accelerates a corrosion reaction commonly referred to as the Zirc-Water Reaction. When the core is no longer submerged in water, cladding surface temperature heats up with the uranium fuel being the heat source. At cladding surface temperatures in excess of 2000 ^{0}F the reaction rate is significant. At cladding surface temperatures approaching 2500 ^{0}F the resultant heat is enough to maintain a high reaction rate (exothermic) regardless of fuel temperature."

When needed the most, the fuel clad barrier will essentially self-destruct, adding heat to an overheated core while creating large concentrations of explosive gas. The second barrier will contribute to, or directly, cause the failures of barriers three and four (reactor coolant system and containment).

It is also important to recognize that the fuel cladding is the only physical barrier between the spent fuel and the environment. Spent fuel pool Zirc-fires are also credible accidents that have may have the largest source terms.

(3) During core melt accidents, the third barrier (reactor coolant system) will fail from mechanical damage, core-melt creep rupture or high pressure melt ejection.

(4) According to NRC studies (referenced below), the fourth barrier, containment, is expected to fail "*as a result of direct attack by molten core debris." "Drywell [containment] rupture due to pedestal failure or rapid over pressurization is also an important contributor to early containment failure." "Late failure of containment is also most likely to occur in the drywell* [containment] *but in the form of prolonged leakage past the drywell head.*" And, "*early containment failure in station blackout is dominated by hydrogen deflagrations."*

Because of the "*relatively high probabilities that those* [BWR Mark I and II] *containments would fail should an accident progress to melting the core*," the NRC

issued Order EA-13-109, requiring hardened containment vents to be installed at specified plants. This NRC order, essentially, requires nuclear plant operators to intentionally vent (release) large amounts of highly radiological material to the environment, because the containment barriers were not designed to survive credible accidents, which include hydrogen explosions.

Consideration of International Standards

The International Atomic Energy Agency (IAEA) works with member states and other partners to promote the safe, secure, and peaceful use of nuclear technologies. The IAEA develops Safety Requirements and Safety Guides for protecting people and the environment from harmful effects of ionizing radiation. This system of safety fundamentals, safety requirements, safety guides, and other relevant reports reflects an international perspective on what constitutes a high level of safety. To inform its development of this RG, the NRC considered IAEA Safety Requirements and Safety Guides pursuant to the Commissions International Policy Statement (Ref. 13) and Management Directive and Handbook 6.6, "Regulatory Guides" (Ref. 14).

The NRC staff did not identify any IAEA Safety Requirements or Guides with enough detailed information relevant for use by Part 50 and 52 licensees and applicants as related to the topic of this RG.

4.1 Offsite Dose Consequences

The licensee should use the following assumptions in determining the TEDE for persons located at or beyond the EAB:

b. The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in International Commission on Radiological Protection (ICRP) Publication 30, "Limits for Intakes of Radionuclides by Workers," issued in 1979 (Ref. 28).

- Since 1978, the ICRP has twice updated their recommendations. Nevertheless, the NRC continues to resist the adoption of more recent international standards that are "*based on the latest available scientific information of the biology and physics of radiation exposure.*" The NRC evaluated ICRP-103 in 2017:
    - *"The reason for performing this evaluation is that for certain plants, <u>the margin for meeting acceptable dose limits in the control room for certain DBAs is very small. The concern is that the implementation of the new ICRP 103 values could cause these plants to exceed the regulatory dose limits</u>."*
    - *"In <u>design basis</u> radiological consequence analyses* [<u>in which containments do not fail</u>]*, iodine isotopes are a major contributor to the estimated thyroid dose. ICRP Publication 26 assigns a thyroid tissue weighting fact or of 0.03. ICRP Publication 103 assigns a thyroid tissue weighting factor of 0.04. It is seen therefore that the estimated fatal cancer for the thyroid is about 33 percent higher in the ICRP Publication 103 recommendations compared to ICRP Publication 26 recommendations. <u>The evaluation of the application of the ICRP 103 iodine DCFs as compared to the ICRP 26 DCFs results in an increase in control room TEDE of approximately 23 to 25 percent</u>."*

- The ICRP-103 evaluation confirmed the NRC's overriding concern "*that the implementation of the new ICRP 103 values could cause these plants to exceed the regulatory dose limits."* Therefore, the NRC concluded: *"The NRC staff's decision to discontinuing the rulemaking activities associated with potential changes to the radiation protection and reactor effluents regulations was* *based on the knowledge that the current NRC regulatory framework continues to provide adequate protection of the health and safety of workers, the public, and the environment."*

The NRC's reason for rejecting ICRP-103 is my overring concern with RG 1.183 (Revision 0) and DG-1389.

Sincerely,
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Former NRC Licensed Senior Reactor Operator/Operations Shift Manager
—Acting expressly as a member of the public.

References:

1988 - NUREG 1228, Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents

3.2 Barriers to Fission Product Release

For the fission products generated within the core to reach the environment, they must pass through four fission product barriers. These barriers are Illustrated in Figure 3.1 for a pressurized-water reactor (PWR). The first barrier is the fuel pellet often referred to as the fuel matrix. The second barrier is the fuel pellet cladding. The reactor coolant system provides a third barrier to fission product release. The final and ultimate barrier to fission product release is the reactor containment.

(1962) TID-14844

It is the intent that this document to provide reference information and guidance on procedures and basic assumptions whereby certain factors pertinent to reactor siting as set forth in Title 10 Code of Federal Regulations Part 100[(1)] (10 CFR 100) can be used to calculate distance requirements for reactor sites which are generally consistent with current siting practices.

For any proposed reactor: the performance experience accumulated elsewhere; the engineering safeguards; the inherent stability and safety features; and the quality of design, materials, construction, management and operation are all important factors that must be included in the evaluation of the suitability of a site.

For a particular site; size, topography, meteorology, hydrology, ease of warning and removing people in times of emergency, and thoroughness of plans and arrangements for minimizing injuries and interference with offsite activities, all enter an evaluation.

Consideration of these as well as other aspects of hazards evaluation involves so many different situations and such complex technological problems that it would be quite impossible to anticipate and answer all questions that will arise.

This technical document sets forth one method of computing distances and exposures, for one general class of reactors. In developing this example conservative assumptions have been intentionally selected.

Designers of reactors are expected to examine all significant aspects of the hazards and safety problem they believe are appropriate to the particular situation with which they are dealing. In any case, the designer and/or applicant bears the responsibility for justifying all the assumptions and methods of calculation used in a hazards evaluation. The fact that aspects of the problem are not considered in the example set forth here, does not in any way relieve the designer and/or applicant of the responsibility for carefully examining, in his particular case, every significant facet of the hazards and safety problem.

(1978) NUREG-0396 (EPA 520/1-78-016) PLANNING BASIS FOR THE DEVELOPMENT OF STATE AND LOCAL GOVERNMENT RADIOLOGICAL EMERGENCY RESPONSE PLANS IN SUPPORT OF LIGHT WATER NUCLEAR POWER PLANTS

This report introduces the concept of generic Emergency Planning Zones as a basis for the planning of response actions which would result in dose savings in the environs of nuclear facilities in the event of a serious power reactor accident.

Design Basis Accidents

Under NRC Regulations, the site/reactor design combination must be such that the consequences of design basis accidents are below the plume exposure guidelines of 10 CFR Part 100.

Class 9 Accidents

Class 9 accidents cover a full spectrum of releases which range from those accidents which are of the same order as the DBA-LOCA type of releases; i.e., doses on the order of PAGs within 10 miles; to those accidents which release significant fractions of the available radioactive materials in the reactor to the atmosphere, thus having potential for life-threatening doses. The lower range of the spectrum would include accidents in which a core "melt-through" of the containment would occur.

As in the DBA-LOCA class, the doses from "melt-through" releases (involving thousands of curies) generally would not exceed even the most restrictive PAG beyond about 10 miles from a

power plant. The upper range of the core-melt accidents is categorized by those in which the containment catastrophically fails and releases large quantities of radioactive materials directly to the atmosphere because of over-pressurization or a steam explosion. These accidents have the potential to release very large quantities (**hundreds of millions of curies**) of radioactive materials. There is a full spectrum of releases between the lower and upper range with all of these releases involving some combination of atmospheric and melt- through accidents. These very severe accidents have the potential for causing serious injuries and deaths. Therefore, emergency response for these conditions must have as its first priority the reduction of early severe health effects. Studies have been performed which indicate that if emergency actions such as sheltering or evacuation were taken within about 10 miles of a power plant, there would be significant savings of early injuries and deaths from even the most severe atmospheric releases.

As discussed in Appendix Ill, the Task Force has concluded that both the design basis accidents and less severe core-melt accidents should be considered when selecting a basis for planning predetermined protective actions and that certain features of the more severe core-melt accidents should be considered in planning to assure that some capability exists to reduce the consequences of even the most severe accidents.

(May 1980) NUREG-0660 (Vol. 1), NRC Action Plan Developed as a Result of the TMI-2 Accident

The actions to improve operational safety described in Chapter I are the more important responses to the accident at Three Mile Island. However, possible weaknesses in siting and plant design revealed by the accident should be evaluated and corrected, where necessary.

The NRC had been critically examining siting policy prior to the accident at Three Mile Island. This examination was concluded in August 1979 (**NUREG-0625**). Recommended changes in siting policy from that study and other changes are soon to be issued in an Advanced Noticed of Rulemaking, as recently directed by the Commission. The new rule will be applicable to the siting of newly proposed nuclear plants; however, the treatment of existing nuclear plants, either operating or under construction, also will be considered.

The accident at TMI demonstrated the reality of the risk, previously only theoretically assessed, of accidents that result in substantial degradation and melting of the core. This risk arises from the fact that core-degradation accidents can lead to **containment** failure and the eventual release of large amounts of radioactivity to the environment.

*

(1983) NUREG/CR-3332, “A Textbook on Environmental Dose Analysis”

In examining the source terms, therefore, one must assess the probability of escape from the containment under routine use conditions or in any postulated accident situation.

(1986) NUREG-0956, "Reassessment of the Technical Bases for Estimating Source Terms"

Conclusion 9. Containment performance (survival, failure, or bypass), which is described by input parameters in all current source term codes, is a major factor affecting source terms.

[“Containment” is used 748 times in NUREG-0956.]

(December 1990) NUREG-1150, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants

The consequences of severe reactor accidents depend greatly on containment safety features and containment performance in retaining radioactive material. The early failure of the containment structures at the Chernobyl power plant contributed to the size of the environmental release of radioactive material in that accident. In contrast, the radiological consequences of the Three Mile Island Unit 2 (TMI-2) accident were minor because overall containment integrity was maintained and bypass was small.

Normally three barriers (the fuel rod cladding, the reactor coolant system pressure boundary, and the containment pressure boundary) protect the public from the release of radioactive material generated in nuclear fuel. In most core meltdown scenarios, the first two barriers would be progressively breached, and the containment boundary represents the final barrier to release of radioactivity to the environment.

Maintaining the integrity of the containment can affect the source term by orders of magnitude. The NRC's 1986 reassessment of source term issues reaffirmed that containment performance "is a major factor affecting source terms" (Ref. 9.1). [“*Reassessment of the Technical Bases for Estimating Source Terms*," United States Nuclear Regulatory Commission (USNRC) Report NUREG-0956, July 1986]

> 9.3 A. C. Payne, Jr., et al., "*Evaluation of Severe Accident Risks: Peach Bottom Unit 2*," Sandia National Laboratories, NUREG/CR-4551, Vol. 4, Draft Revision 1, SAND86-1309, to be published.

Shortly after the accident at Three Mile Island, the NRC initiated a program to review the adequacy of the methods available for predicting the magnitude of source terms for severe reactor accidents. After considerable effort and extensive peer review, the NRC published a report entitled "Reassessment of the Technical Bases for Estimating Source Terms," NUREG-0956 (Ref. 10.1).

As expected, the magnitude of the source term varies between different accident progression bins depending on whether or not containment fails, when it fails, and the effectiveness of engineered safety features (e.g., BWR suppression pool) in mitigating the release. However, within an accident progression bin, which represents a specific set of accident progression events, the uncertainty in predicting severe accident phenomena is great.

With regard to the lack of treatment of some severe accident phenomena and containment failure modes, it would not be appropriate to attempt to identify or analyze all containment failure modes or scenario pathways. It is the intent of the latest sequence selection, reported in NUREG/CR-4624, to analyze at least the most risk significant of these pathways.

> 11. Unfortunately, the necessary calculations cannot be made based on data and scientifically accepted principles and methodologies. Because of major inadequacies in the data base, because of the vast complexity of nuclear plants, because a tremendous

number of assumptions must be made, because all contributors to risk cannot be quantified, and because core meltdown phenomena are poorly understood, no one calculation of risk yields a remotely meaningful value of risk (Ref. C.5, p. 38109).

(1989, 1992) NUREG/CR-5247, RASCAL Version 2.1 User's Guide

The Radiological Assessment System for Consequence Analysis (RASCAL) (Athey et al. 1989, 1992) is a set of personal computer-based tools. RASCAL Version 2.1 contains tools to estimate source term, atmospheric transport, and dose from a radiological accident (ST-DOSE), to estimate dose from field measurements of radionuclide concentrations (FM-DOSE), and to compute decay of radionuclides (DECAY). RASCAL was developed for use by U.S. Nuclear Regulatory Commission (NRC) personnel who report to the site of a nuclear accident to conduct an independent assessment of dose projections.

Predicting [nuclear accident] *doses or consequences . . . requires several steps: (1) predicting the quantity and timing of the release from the plant (source term), (2) predicting the movement of the plume (transport), and (3) predicting the dose from the plume and predicting the health effects from the dose. Each of these steps requires collection of appropriate data, and data collection and the subsequent computations are subject to uncertainties.*

The largest single component of uncertainty is expected in the estimate of the source term. Unanticipated catastrophic containment failure is a case in which the source term could be underestimated by a factor of 1,000,000 if monitor readings are used to estimate the source term. For lesser (non-core damage) accidents in which the total release is through a monitored pathway and consists mostly of noble gases, the source term uncertainty can be reduced. However, the transport and dose uncertainties would remain unchanged.

(1989) NRC Generic Letter 89-16, "*Installation of a Hardened Wetwell Vent*"

(2001) NUREG-1738, Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants

(2007) The 2007 Recommendations of the International Commission on Radiological Protection. ICRP Publication 103. Ann. ICRP 37 (2-4).

Abstract - These revised Recommendations for a System of Radiological Protection formally replace the Commission's previous, 1990, Recommendations; and update, consolidate, and develop the additional guidance on the control of exposure from radiation sources issued since 1990.

Thus, the present Recommendations update the radiation and tissue weighting factors in the quantities equivalent and effective dose and update the radiation detriment, based on the latest available scientific information of the biology and physics of radiation exposure. They maintain the Commission's three fundamental principles of radiological protection, namely justification, optimisation, and the application of dose limits, clarifying how they apply to radiation sources delivering exposure and to individuals receiving exposure.

The Recommendations evolve from the previous process-based protection approach using practices and interventions by moving to an approach based on the exposure situation. They

recognise planned, emergency, and existing exposure situations, and apply the fundamental principles of justification and optimisation of protection to all of these situations. They maintain the Commission's current individual dose limits for effective dose and equivalent dose from all regulated sources in planned exposure situations. They reinforce the principle of optimisation of protection, which should be applicable in a similar way to all exposure situations, subject to the following restrictions on individual doses and risks; dose and risk constraints for planned exposure situations, and reference levels for emergency and existing exposure situations. The Recommendations also include an approach for developing a framework to demonstrate radiological protection of the environment.

(2013) EA-13-109, NRC Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions

The events at the Fukushima Dai-ichi nuclear power plant following the March 2011 earthquake and tsunami highlight the possibility that events such as rare natural phenomena could challenge the traditional defense-in-depth protections related to preventing accidents, mitigating accidents to prevent the release of radioactive materials, and taking actions to protect the public should a release occur. At Fukushima Dai-ichi, limitations in time and unpredictable conditions associated with the accident significantly hindered attempts by the operators to prevent core damage and containment failure. In particular, the operators were unable to successfully operate the containment venting system. These problems, with venting the containments under the challenging conditions following the tsunami, contributed to the progression of the accident from inadequate cooling of the core leading to core damage, to compromising containment functions from overpressure and over-temperature conditions, and to the hydrogen explosions that destroyed the reactor buildings (secondary containments) of three of the Fukushima Dai-ichi units. The loss of the various barriers led to the release of radioactive materials, which further hampered operator efforts to arrest the accidents and ultimately led to the contamination of large areas surrounding the plant.

The events at Fukushima reinforced the importance of reliable operation of hardened containment vents during emergency conditions, particularly, for small containments such as the Mark I and Mark II designs. On March 12, 2012, the NRC issued Order EA-12-0501 requiring the Licensees identified in Attachment 1 to this Order to implement requirements for a reliable hardened containment venting system (HCVS) for Mark I and Mark II containments. Order EA-12-050 required licensees of BWR facilities with Mark I and Mark II containments to install a reliable HCVS to support strategies for controlling containment pressure and preventing core damage following an event that causes a loss of heat removal systems (e.g., an extended loss of electrical power). The NRC determined that the issuance of EA-12-050 and implementation of the requirements of that Order were necessary to provide reasonable assurance of adequate protection of the public health and safety.

While developing the requirements for a reliable HCVS in EA-12-050, the NRC acknowledged that questions remained about maintaining containment integrity and limiting the release of radioactive materials if the venting systems were used during severe accident conditions.

The requirements in this Order, in addition to providing a reliable HCVS to assist in preventing core damage when heat removal capability is lost (the purpose of EA-12-050), will ensure that venting functions are also available during severe accident conditions. Severe accident conditions include the elevated temperatures, pressures, radiation levels, and combustible gas

concentrations, such as hydrogen and carbon monoxide, associated with accidents involving extensive core damage, including accidents involving a breach of the reactor vessel by molten core debris.

As discussed in SECY-12-0157, the NRC's determination that a venting system should be available during severe accident conditions considered both quantitative assessments of costs and benefits, as well as, various qualitative factors. Among the qualitative factors, one of the more important is enhancing the defense-in-depth characteristics of Mark I and Mark II containments by addressing the relatively high probabilities that those containments would fail should an accident progress to melting the core. Other qualitative factors supporting installation of severe accident capable vents include addressing uncertainties in the understanding of severe accident events, supporting severe accident management and response, improving the control of hydrogen generated during severe accidents, improving readiness for external and multi-unit events, and reducing uncertainties about radiological releases and thereby improving emergency planning and response. The installation of a reliable, severe accident capable containment venting system, in combination with other actions such as ensuring drywell flooding capabilities, reduces the likelihood of containment failures and thereby enhances the defense-in-depth protections for plants with Mark I and Mark II containments.

APRIL 2015 - IAEA REPORT ON ASSESSMENT AND PROGNOSIS IN RESPONSE TO A NUCLEAR OR RADIOLOGICAL EMERGENCY - INTERNATIONAL EXPERTS MEETING VIENNA

Harmonization of approaches and international guidance:

IAEA safety standards and guidance documents (for example, IAEA Safety Standards Series No. GSR Part 7) takes into account the ICRP recommendations (specifically **ICRP 103**) but are not always identical. International, regional and national level documents (for example, the Nordic Flagbook and EPA-400 Protective Action Guidelines in the United States of America) can also have differences. Different standards or dosimetry techniques used either in different areas or by different agencies for the same areas, could make public safety messaging inconsistent and difficult. This underscores the importance of the IAEA's efforts to coordinate with the 'Accident State' prior to release of information in an emergency. Similarly, neighboring countries may want to have pre-established arrangements to coordinate emergency management messaging.

It was highlighted that public protection strategies implemented in response to an emergency will always be a national level responsibility. However, internationally focused efforts to support the harmonization of response actions would be useful to avoid or explain situations where neighboring countries (or even countries far from the 'Accident State') are recommending actions that conflict with the response actions of the 'Accident State'.

(2017) TRANSMITTAL OF THE CONTROL ROOM DOSE EVALUATION USING ICRP 103 DOSE CONVERSION FACTORS LETTER REPORT FOR USER NEED REQUEST NRR-2009-002

In design basis radiological consequence analyses, iodine isotopes are a major contributor to the estimated thyroid dose. ICRP Publication 26 assigns a thyroid tissue weighting fact or of 0.03. ICRP Publication 103 assigns a thyroid tissue weighting factor of 0.04. It is seen therefore that the estimated fatal cancer for the thyroid is about 33 percent higher in the ICRP Publication 103 recommendations compared to ICRP Publication 26 recommendations. The evaluation of the application of the ICRP 103 iodine DCFs as compared to the ICRP 26 DCFs results in an increase in control room TEDE of approximately 23 to 25 percent.

On January 31, 2016, in SECY-16-0009, "Recommendations Resulting from the Integrated Prioritization and Re-Baselining of Agency Activities," (ADAMS Accession No. ML16028A208), the NRC staff requested Commission approval to implement recommendations on work to be shed, de-prioritized, or performed with fewer resources. Two of the items listed to be shed (i.e., discontinued) were the rulemakings that would have amended the radiation protection regulations in 10 CFR Part 20, and the reactor effluents regulations in 10 CFR Part 50, Appendix I.

The NRC staff's decision to discontinuing the rulemaking activities associated with potential changes to the radiation protection and reactor effluents regulations was based on the knowledge that the current NRC regulatory framework continues to provide adequate protection of the health and safety of workers, the public, and the environment. In the SRM for SECY-16-0009, dated April 13, 2016 (ADAMS Accession No. ML16104A158), the Commission approved discontinuing the two rulemaking activities. However, since all of the technical work for this task had been completed prior to the Commission's decision above, RES and NRR mutually decided to finish the report.

(2017) CONTROL ROOM DOSE EVALUATION USING ICRP 103 DOSE CONVERSION FACTORS (Information Systems Laboratories, Inc. Contract No. NRC-HQ-13-P-04-0099 - GS23F0060L)

The reason for performing this evaluation is that for certain plants, the margin for meeting acceptable dose limits in the control room for certain DBAs is very small. The concern is that the implementation of the new ICRP 103 values could cause these plants to exceed the regulatory dose limits.

As noted earlier, the Peach Bottom BWR and the Wolf Creek and STP PWR were selected by NRR for this analysis. The DBAs that were studied were the FHA and the LOCA. NRR staff selected these accidents as they are bounding accidents for the control room dose. Note that only the control room TEDE dose is reported in this study.

Application of the ICRP 103 DCFs will result in an increase in the range of 23 to 25% in the TEDE doses for the control room. Note that only iodine radionuclides are included in these results. Including the ICRP 103 DCFs for all nuclides may (or may not) result in a smaller difference in the TEDE results.

Other References

Appendix A to Part 50—General Design Criteria for Nuclear Power Plants

Introduction

Under the provisions of § 50.34, an application for a construction permit must include the principal design criteria for a proposed facility. Under the provisions of 10 CFR 52.47, 52.79, 52.137, and 52.157, an application for a design certification, combined license, design approval, or manufacturing license, respectively, must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.

The development of these General Design Criteria is not yet complete. For example, some of the definitions need further amplification. Also, some of the specific design requirements for structures, systems, and components important to safety have not as yet been suitably defined. Their omission does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary safety requirements. These matters include:

(1) Consideration of the need to design against single failures of passive components in fluid systems important to safety. (See Definition of Single Failure.)

(2) Consideration of redundancy and diversity requirements for fluid systems important to safety. A "system" could consist of a number of subsystems each of which is separately capable of performing the specified system safety function. The minimum acceptable redundancy and diversity of subsystems and components within a subsystem, and the required interconnection and independence of the subsystems have not yet been developed or defined. (See Criteria 34, 35, 38, 41, and 44.)

(3) Consideration of the type, size, and orientation of possible breaks in components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss-of-coolant accidents. (See Definition of Loss of Coolant Accidents.)

(4) Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of protection systems and reactivity control systems. (See Criteria 22, 24, 26, and 29.)

It is expected that the criteria will be augmented and changed from time to time as important new requirements for these and other features are developed.

There will be some water-cooled nuclear power plants for which the General Design Criteria are not sufficient and for which additional criteria must be identified and satisfied in the interest of public safety. In particular, it is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power

units of advanced design. Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

Criterion 2—Design bases for protection against natural phenomena. Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Criterion 4—Environmental and dynamic effects design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

Criterion 16—Containment design. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Criterion 50—Containment design basis. The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

This margin shall reflect consideration of

> (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning,

(2) the limited experience and experimental data available for defining accident phenomena and containment responses, and

(3) the conservatism of the calculational model and input parameters.

Criterion 19—Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.

[36 FR 3256, Feb. 20, 1971, as amended at 36 FR 12733, July 7, 1971; 41 FR 6258, Feb. 12, 1976; 43 FR 50163, Oct. 27, 1978; 51 FR 12505, Apr. 11, 1986; 52 FR 41294, Oct. 27, 1987; 64 FR 72002, Dec. 23, 1999; 72 FR 49505, Aug. 28, 2007]

Page Last Reviewed/Updated Wednesday, March 24, 2021

Appendix B to Part 50—Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

III. Design Control

Measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in § 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and

that deviations from such standards are controlled. Measures shall also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components.

Measures shall be established for the identification and control of design interfaces and for coordination among participating design organizations. These measures shall include the establishment of procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces.

The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. The verifying or checking process shall be performed by individuals or groups other than those who performed the original design, but who may be from the same organization. Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, it shall include suitable qualifications testing of a prototype unit under the most adverse design conditions. Design control measures shall be applied to items such as the following: reactor physics, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for inservice inspection, maintenance, and repair; and delineation of acceptance criteria for inspections and tests.

Design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design and be approved by the organization that performed the original design unless the applicant designates another responsible organization.

[35 FR 10499, June 27, 1970, as amended at 36 FR 18301, Sept. 11, 1971; 40 FR 3210D, Jan. 20, 1975; 72 FR 49505, Aug. 28, 2007; 84 FR 63568, Nov. 18, 2019]

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10 CFR 50.2 Definitions

Design bases means that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state of the art" practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals.