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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 & NUCLEAR MATERIALS

SUBCOMMITTEE

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FRIDAY

JULY 23, 2021

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

COMMITTEE MEMBERS:

DAVID A. PETTI, Chairman

RONALD G. BALLINGER, Member

DENNIS BLEY, Member

CHARLES H. BROWN, JR. Member

VESNA B. DIMITRIJEVIC, Member

GREG HALNON, Member

WALTER L. KIRCHNER, Member

JOSE MARCH-LEUBA, Member

JOY L. REMPE, Member

MATTHEW W. SUNSERI, Member

ACRS CONSULTANT:

MIKE CORRADINI

STEVE SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

MICHAEL SNODDERLY

ALSO PRESENT:

GREG BROADBENT, Entergy

JERRY DOZIER, NRR

MIKE FRANOVICH, NRR

KEVIN HSUEH, NRR

STEVE JONES, NRR

SCOTT MOORE, Executive Director, ACRS

JOHN PARILLO, NRR

FRANCES PIMENTEL, NEI

SHILP VASAVADA, NRR

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P R O C E E D I N G S

9:30 a.m.

CHAIR PETTI: Okay, so I have 30 minutes after the hour so let's have the meeting come to order.

This is a meeting of the Advisory Committee on Reactor Safeguards, Radiation Protection and Nuclear Materials Subcommittee. I'm Dave Petti, chairman of today's subcommittee meeting.

Members with us today are Charlie Brown, Dennis Bley, Greg Halnon, Jose March-Leuba, Walt Kirchner, Consultant Mike Corradini.

MEMBER REMPE: Dave, this is Joy. I'm also here.

CHAIR PETTI: Yes, Member Joy Rempe, Consultant Steve Schultz, Member Ron Ballinger, Member Vesna Dimitrijevic, and Member Matt Sunseri.

Mike Snodderly is the Designated Federal Official for this meeting. The subcommittee will review the staff's draft interim staff guidance entitled Supplemental Source Guidance for Radiological Consequence Analysis Using Alternative Source Term. We also have members of the NRC staff and NEI to brief the subcommittee.

The ACRS was established by statute and

it's governed by the Federal Advisory Committee Act, FACA. The NRC implements FACA in accordance with its regulations found in Title 10 of the Code of Federal Regulations, Part 7. The committee can only speak to its published letter reports. We hold meetings to gather information, perform preparatory work to support our full deliberations at a full committee meeting.

The rules for participation in all ACRS meetings were announced in the Federal Register on June 13, 2019. The ACRS section of the U.S. NRC public website provides our charter, bylaws, agendas, letters, of course, and full transcripts of all full subcommittee meetings including slides presented there. The agenda for this meeting was also posted there.

As stated in the Federal Register notice and in the public meeting notice posted to the website, members of the public who desire to provide written or oral input to the subcommittee may do so. You should contact the Designated Federal Official five days prior to the meeting as practicable.

We've set aside 15 minutes for comments from members of the public, attending or listening to our meetings. We have not received written comments

or requests for time to make oral statements for members of the public regarding today's meeting.

A transcript of the meeting is being kept and will be made available on the ACRS section of the NRC public website.

It is requested that speakers identify themselves and speak with sufficient clarity and volume so they can be readily heard. Additionally, participants should mute themselves when not speaking. A telephone bridge line has been established for the public to listen to the meeting. To minimize disturbance, the public line will be kept in a listen in only mode.

With that, we will now proceed with the meeting. I call upon Mike Franovich, Director of the Division of Risk Assessment in NRR to begin today's presentations. Mike?

MR. FRANOVICH: Good morning, Chairman Petti. If we could have Slide 2, please.

Good morning, Chairman Petti and good morning, ACRS Subcommittee members. I am Mike Franovich and I serve as the Director of the Division of Risk Assessment in NRR. Thank you for the opportunity today for the staff to share advances in our regulatory reviews of radiological consequence

analysis using the alternate source term.

To set the stage, let me highlight a few of the more recent regulatory transformations and the performance shaping factors improving our efficiency and reliability as regulators.

A key theme is that our licensing, other regulatory decisions, and backfit/forward fit actions must be risk informed and there are two particular staff requirement memoranda that have been directing those items here noted on the slide.

The Commission's recent direction reminded the staff that we are enabled to use risk-informed, performance-based approaches in our work. This direction in 2019 also serves as an accelerant for transformation become a more modern risk-informed regulator.

The SRM, commonly referred to as the NuScale Block Valve SRM, draws upon long standing practices and in particular, a 1999 Commission paper that states succinctly a risk-informed, performance-based approach is one of risk insights, engineering analysis, and judgment including the principle of defense in depth and the incorporation of safety margins including performance history are used in decisions.

In 2018, we also received additional clarifications from the Commission regarding regulatory backfits and proper treatment of forward fits. This Commission direction serves as a regulatory stabilizing applying the reliability principles of good regulations in our license amendment reviews.

Secondly, improved realism evaluation techniques and additional information are applied to improve risk-informed decision making.

As noted in a 2019 memo to the Executive Director for Operations on applying risk-informed principles, the NRC's application of risk-informed decision making continues to evolve, as improved realism, evaluation techniques, and additional information are applied to improve our decisions. What that means to the staff when it comes to review of applications to allow for possible increases in leakage from BWR main steam isolation valves is that there are tremendous opportunities to apply engineering and risk insights. This mosaic of information includes plant operating experience, as well as our experiences from post-Fukushima activities to make more realistic and ultimately better decisions while abiding by the Commission's backfit and forward fit expectations.

And third, cultural realignment is needed to ensure that we identify and resolve challenges and roadblocks for the appropriate and consistent integration of risk insights. Two important agency activities address NRC internal cultural realignment needs that are detailed in the previously mentioned memo to the EDO. Most recently, the Be riskSMART Initiative is driving agency-wide practices for a uniform risk and reward mindset and use of graded approaches in our safety, security, corporate support and other agency business.

This concept is also applicable as we assess licensees' request to allow for increased leakage while satisfying overall plant performance objectives, limiting potential consequences during hypothetical accidents. For BWR MSIV leakage, here is once again an opportunity to be mindful of radiation dose ALARA objectives for workers who maintain these MSIVs in terms of their performance.

Lastly, NRR continues to implement the after actions of our 2018 risk-informed decision making action plan and a key insight out of that plan is promoting greater use of integrated review teams.

Today, you will hear from a diverse team consisting of management, systems and component

experts, risk analysts, seismic experts, and accident dose analysts supporting this Draft ISG.

And with that said, I will turn it over to Kevin.

MR. HSUEH: Thanks, Mike. Good morning, everyone. I am Kevin Hsueh, Branch Chief, Radiation Protection and Consequence Branch in NRR, Division of Risk Assessment.

I'll cover Slides 3 and 4 and we are currently on Slide 3. In 2019, we received four license amendment requests to increase MSIV leakage allowed by tech specs for BWR. Traditionally, this type of amendment requests were reviewed using deterministic review methods.

In the SRM that might mention the NuScale Block Valve SRM, the Commission directed the staff to apply risk-informed principles in any licensing review or other regulatory decision when strict, prescriptive application of deterministic criteria is unnecessary to provide for reasonable assurance of adequate protection of public health and safety.

In response to this and other previous risk-informed related SRMs and soon after we received this amendment request, we started to look for ways where we can increase use of risk insights to perform

this review.

After several months of efforts and a number of meetings among NRR staff for more support (phonetic) the regions staff developed a technical assessment and followed NRR office instructions LIC-206 process to integrate this insight with these types of traditional decommissioning reviews.

CHAIR PETTI: It looks like Member Rempe has her hand up. She has a question, I think.

MR. HSUEH: Oh, okay. All right. Go ahead.

MEMBER REMPE: Sorry, Kevin. I have a question that is a bit off topic and it's probably due to me not being fully informed on what the staff did with these reviews, but again, we only saw the Fitzpatrick and the ISG, but if the staff is going to be using risk insights for design basis actions source terms, I'm wondering if other risk insights were also considered such as the impact on operator actions if they allow increased leakage from the MSIVs.

As I think about how the operators would know if the MSIV closed, there is probably differences in temperature or radiation, monitor readings or something like that, but again, they're used to seeing increased leakage. And so I'm wondering if their

ability to detect that it really closed or if it failed to close would be impacted.

Did the staff consider that as they risk informed this process? Because I didn't see anything about operator actions in what the staff sent back on the Fitzpatrick request.

MR. HSUEH: Thank you for the question and please hold that thought and maybe later on we can touch base on that and we have the staff can answer that question if it's okay with you.

MEMBER REMPE: Sure. It was something that crossed my mind when I was reading this and again, perhaps the staff did something and it just wasn't in the documentation you were given. But I am curious about that. To meet holistically, consider risk insights, not just pick and choose things that are going to address the consequences is what my thought was.

MR. HSUEH: Okay. Thank you. So we will answer -- respond to that at the staff's presentation and I think that the staff is prepared to respond to that question.

MEMBER REMPE: Great.

MR. HSUEH: All right, so I continue the Slide 3. So during our review, we overcame many

changes and completed all four safety evaluations with risk and engineering insights to support our reasonable assurance finding. And to document our reasonable assurance finding each of the four safety evaluations includes new risk and engineering insight section summarizing the finding and conclusion of a technical assessment.

All four safety evaluations received OGC's no legal decision (phonetic) individually prior to staff approval of this amendment request.

Mike mentioned the November 2019 NRC memo, agency's efforts in implementing the NuScale Block Valve SRM. The memo highlights the staff's efforts and staff's challenges and continuous efforts in applying risk-informed principles in our decision making and making the progress one decision at a time.

Consistent with the implementation of that SRM and to memorialize our practice and experiences, we developed this interim staff guidance or ISG. This ISG serves as an example of our continuous effort in working toward being a more modern and risk-informed regulators.

Next slide.

We are now on Slide 4. In addition to the ISG developments, there has been a separate on-going

effort to revise Reg Guide 1.183 in parallel. Specifically, staff has restarted its efforts to revise Reg Guide 1.183 since late last year to update the current Rev. 0 which was issued in 2000.

A working group and a steering committee were established with a step-by-step project plan to keep the project moving efficiently and effectively. So far, we have held three public meetings to seek stakeholders' input and feedback on a variety of proposed changes to Rev. 0 and completed a threat revision.

The threat revision is currently being processed by the Office of Research and the subcommittee meeting on the threat revision is scheduled in fall of this year.

So for today's meeting, we're trying to focus our discussion on the threat ISG and how we use the recent engineering insight to support our reasonable assurance findings. With that, I'll turn it over to Jerry to start that presentation.

MR. DOZIER: Hello. My name is Jerry Dozier. I'm a Senior Risk and Reliability Analyst from the Radiation Protection and Consequences Branch in the Division of Risk Assessment.

In this presentation today that we'll

have, as Mike said, diverse skilled members of staff will provide a presentation which will basically give a background of the ISG, the basis for the ISG. We were requested to provide about the difference between this ISG and the Reg Guide as Kevin just mentioned, and also how we use this ISG in the LARs. And then we'll finish with a few takeaways.

MEMBER KIRCHNER: Jerry, this is Walt Kirchner. May I just ask you for a little background or context from you and your team about the BWR LARs? The way the viewgraphs are written might lead the public to -- might mislead the public into thinking there's an issue with the performance of the main steam isolation valves.

So could you just give us some context for the record as to what the issues are? Again, the way the viewgraphs actually read, kind of just on the surface, is that you're allowing increased leakage from valves. And it suggests that that might be actually a problem. And of course, you've used your risk-informed approach to determine that it is not.

But could you just provide a little more context about the LARs and the issues with the main steam isolation valves in BWR?

MR. DOZIER: Sure. Sure, I can. And also

basically for -- we have one of the reviewers of the LARs toward the end of the presentation actually talking about some of the challenges that were in the LARs, how this was used to resolve that. However, that will come a little bit later.

MEMBER KIRCHNER: Okay, I'll wait until then. I just thought up front it might provide more context for the public.

MR. DOZIER: Okay, and what I would like to say that as far as the leakage, what the theme of these LARs were, was basically it was the licensees asking for an increase in allowed leakage in their technical specifications. So it was -- that was the whole purpose of the LAR was to ask for this increase in the leakage. And of course, the reviewer has to use our guidance to review this. And there was some challenges even within our guidance that we resolved. We'll talk about it in detail, talk about the slides, if that's okay.

MEMBER BROWN: Did we lose the presenter?

MR. DOZIER: I'm still here. I was just saying if that's okay with you.

MEMBER KIRCHNER: I'll wait.

MR. DOZIER: Okay.

MR. FRANOVICH: Thank you. This is Mike

Franovich, Director of Risk Assessment, NRR. I think it's important to note here, as the staff will explain further in the presentation. I'm sorry, I hear a little bit of feedback on the line. I'm not sure all the lines are muted.

The proposed amendments that came in doesn't suggest or at least we're not aware of any suggestion that they are some type of industry trend issue with the performance of MSIVs. We've had other plants that have requested increased allowables for leakage. And so this is not necessarily a reflection of issue with the components as an industry wide type of issue in terms of trends. I just want to set that out there for members of the public.

MEMBER KIRCHNER: That's what I was thinking, Mike. You know, because just like I said, a superficial reading of the viewgraphs would suggest there's problems with the valves and now they want to have a more allowance for leakage, et cetera, so thank you.

MR. FRANOVICH: And if I can also add -- no, that's very fair. I appreciate you giving us the opportunity to explain that because the other tempering factor that the licensees are challenged with is managing ALARA. And these are not small, low-

dose type of activities to overhaul our maintenance on these types of valves. So there is a balancing there between that of the needs for radiation protection for the rad workers versus what would be an acceptable increase in allowables for these hypothetical type scenarios that we look at for consequence analysis.

MEMBER KIRCHNER: So just thank you. Just one recommendation, when this comes before the full committee, I think it would be important for that -- that context and background to be up front in the presentation. Thank you.

MR. DOZIER: Slide 6. So for the overview of the ISG, basically, this ISG was published in the Federal Register on June 21, 2021. We did get comments from NEI, as well as also some anonymous comments. There was 13 from NEI, 20 anonymous comments.

We'll have an ACRS full committee briefing that's scheduled for November 2021. OMB approval would be after that and we expect final FRN for this to be February of 2021.

Slide 7.

MEMBER REMPE: I think you meant 2022, right?

MR. DOZIER: Yes, as the slide --

MEMBER REMPE: It's what's on the slide, but, yeah, I can't do it in --

(Simultaneous speaking.)

MR. DOZIER: Thank you. Slide 7.

This ISG is expected to be transitioning into SRP Section 15.0.1. Section 15.0.1 will include a reference to the revised Reg Guide 1.183 that we are working on as a separate project as was explained earlier. Then the ISG will be closed after transition to this section.

So this is kind of the high level primary insights. We had many insights, but this is really the primary insight that we're taking from this look and that is that there's a high probability that doses will be lower than those estimated strictly using traditional deterministic methods and by using this we're using all of the accepted assumptions that's already in the guidelines. We're not changing those. And what we're saying is that do not credit hold-up and retention of the Main Steam Isolation Valve leakage within the power conversation system.

So big picture. Piping and components downstream of the MSIVs sees significant pressure, temperatures, and vibrations 7 days a week, 24 hours a day while the reactor is operating. If the plant

does not ask for formal credit for hold-up in the condenser, Reg Guide 1.183 assumes that the piping and components disappear with a safe shutdown earthquake and the MSIV leakage goes directly to the environment. This is all that stuff after the second MSIV.

For plants requesting an increase in the tech spec allowable MSIV leakage and they do not credit the condenser, the staff may recognize that there's a high likelihood that this robust, high pressure pipe, and components is available for hold-up instead of it being rubble on the floor to support the staff's reasonable assurance.

And that was kind of the simple way of maybe I could say these formal words.

MEMBER HALNON: So Jerry, this is Greg Halnon. When you credit and make that assumption, what is the condition of the piping downstream of the second MSIV to a condenser that you're assuming that it's all intact and then there's no additional leakage or that there's a -- so that's one thing we don't do in BWRs. We don't necessarily measure the leakage, but we do have leakage.

You can just look at the amount of leak repairs that are done on line through the power conversion systems, MSRs, and other extraction, steam,

steam trap and other things that are releasing it, not realizing BWRs that's a lot more evident because you'll have the radiological measurement of those types of things, but there are quite a lot of leaks that the industry deals with as they operate these plants in those U.S. conditions.

So how did you assume that piping was intact enough even in normal operation to get that leakage to the condenser to have that hold up?

MR. DOZIER: Well, actually, you know, if those valves -- okay, so those valves are the money makers for the utility, okay? You know, so they're very important valves for the plant.

We're going to go into detail on this later in the presentation, but even if there is no hold up, I mean especially if those valves close, you know, there's really, as far as leakage to the control room, you know, it's not any leakage, because everything held up in that large PCF volume that we'll get into.

MEMBER HALNON: Okay, well, if you're going to get into more detail, we can hold the question.

MR. DOZIER: Yes, I'll let the expert on that one answer it.

MEMBER HALNON: Okay. I can hold the question, so we'd be -- start looking at full diagrams and what not.

MR. DOZIER: Now I'm on Slide 9. In this, as I said, this ISG is to make a formal footprint. That's really what the objective is. The staff will use this to offset uncertainties and input parameters for deterministic calculations and supports -- and the main thing is it supports the staff's reasonable assurance finding during reviews and it will be transitioned.

Now the most -- a very important caveat to this is it does not change the licensee's responsibility to demonstrate compliance within 10 CFR 50.67.

It also -- since we're using something that probably 50 percent of the plants formally credited earlier, and now we're doing for these LARs that did not ask for this credit, we are not changing acceptable methods for demonstrating compliance with 10 CFR 5067. So we're not changing things upstream from these MSIVs or any of those assumptions.

As even in the opening remarks, now I'm on Slide 10, given in the opening remarks, we're basically trying to be a modern, risk-informed

regulator as we're directed. And the documents we spoke about before, we had those four license amendments submitted to us and they were asking for increased MSIV leakage in 2019. We had some challenge that we'll talk about later and since we had those challenges, we invoked LIC-206 which is basically for risk informed decision making that we do in reviews. So this was a new process that we would follow.

We got an integrated review team --

MR. CORRADINI: Excuse me. This is Corradini. Can you just remind people of what LIC-206 is? Maybe I'm the only one that doesn't remember.

MR. DOZIER: It's risk-informed decision making and reviews, license amendment reviews.

MR. CORRADINI: Thank you.

MR. DOZIER: So it's specific to license amendment reviews.

MEMBER BLEY: This is Dennis Bley. For everybody else, maybe who hasn't been around a long time on this, LIC-206 is worth reading. It's really good background and you understand how people are implementing some of the risk-informed activities.

MR. DOZIER: Dr. Vasavada, who will be speaking later, was one of the contributors, significant contributors to that document.

Okay, it identified that risk insights support consideration of holdup into PCS and ability to -- okay, so we had that group and we looked at many risk insights. There was many things we looked at. We focused on just the condenser because we wouldn't have to change up a lot of the very conservative assumptions that was in Reg Guide 1.183. So we focused on this holdup, okay? And we (unintelligible) those insights and that technical assessment which you basically see in the basis portion of the ISG. It was talked about internally, multi-division, multi-people. It was a team approach. It was lots of insight.

So we --

MEMBER REMPE: LIC-206 does say you need to have a holistic approach, so are you the right person to ask my question about did you consider the impact on what the operators would see and if there's some penalties associated with allowing increased leakage. And I'll mention that at TMI, they changed the tech spec for the core and they did see increased leakage and that may be one of the reasons the operators didn't detect that they had a small break LOCA. So my question -- are you the right person or is that going to come up later?

MR. DOZIER: I'm kind of waiting for the

team to see if someone in the team wants to jump in on that.

MR. JONES: This is Steve Jones. I'll be speaking next and I can try to address that issue with how testing and tech spec limits on the testing relate to what operators would see and conditions that they would respond to.

MEMBER REMPE: Great. Thank you. Again, sorry, to keep bringing it up, but I am curious about it.

MR. DOZIER: Okay, with that, we'll get right into Steve's presentation to hopefully address those, Dr. Rempe.

MR. JONES: Good morning. I'm Steve Jones, a Senior Plant and Safety Systems Engineer from the Containment and Plant Systems Branch in the NRR Division of Safety Systems.

As Jerry went over the current guidance in Reg Guide 1.183, specified the assumption of a direct ground level release at the downstream MSIV when no seismically qualified main steam piping downstream of that MSIV is present.

In part, to address those issues, the BWR owner's group developed a topical report and the staff approved that in 1999 to allow computational credit to

consider holdup and deposition within the main steam piping and the main condenser as kind of indicated by the dash line in the diagram to the right.

Roughly half of the operating boiling water reactors have adopted this methodology which significantly reduces the effect of mainstream isolation valve leakage on the calculated dose consequences as evaluated for the control room and the site boundaries.

However, again, as Jerry mentioned, even without a thorough evaluation of the seismic robustness of the main steam lines and the remainder of the power conversion system, the staff determined that there's significant evidence supporting the main steam system and other parts of the power conversion system would contribute to holdup and potential deposition of fission products when not formally credited in the dose calculation.

Slide 13, please.

Okay, this diagram shows the configuration of the main steam system in a typical boiling water reactor. The curved gray wall represents the dry well or primary containment and the second straight gray wall represents secondary containment boundary.

So the downstream main steam isolation

valves are the ones to the -- between the two gray barriers, the air operated valves with the mushroom-shaped actuator depicted in this diagram. And then this diagram indicates that there are several paths once there's flow beyond the main steam isolation valves and these paths include main steam piping drains and turbine bypass valves that lead directly -- both of which lead directly to the main condenser.

In addition, there are other flow paths through the main turbines, potentially steam driven main feedwater turbines that could provide additional holdup volumes for any main steam isolation valve leakage.

Slide 14, please.

Going into just a little bit more detail on our approach, the staff developed this overall assessment considering the risk triplet for releases beyond the main steam isolation valves. We considered operating experience related to the ruggedness of piping systems and other components and that operating experience included the events at Fukushima, North Anna earthquakes, and other events that demonstrated the robustness of secondary plant system components through earthquakes.

We also considered the pathways available

to the leakage without operator efforts or operator actions to redirect flow to any specific location and assessments of seismic capacity that Dr. Vasavada will address in the later slides.

MR. DOZIER: Dr. Rempe had a question.

MEMBER REMPE: It's a different question this time.

MR. JONES: Okay.

MEMBER REMPE: I'm curious about anchorage of the piping and the condenser. And again, the documentation we were given was very limited in that area. But what assurance do you have that the anchorage is similar, because that was one question that I've seen raised in other forums about what can we learn from the events at Daiichi and if the way that the components were anchored were similar. And did you investigate that?

And then I noticed you mentioned Onagawa in your documentation. But you didn't mention what you saw at Daiichi or Daini. And was there a reason for that?

MR. JONES: No. I'm sorry. We were just generally considering the operating experience developed from the Great Tohoku Earthquake. And we did consider the experience at some of the other sites

where we found documented information about, for instance, the seismic damage that resulted from plants that did not experience core damage events to the --

MEMBER REMPE: But --

MR. JONES: -- secondary system.

MEMBER REMPE: But Daini had no core damage and yet it might've had a more -- did you see any damage where the earthquake occurred too in the condenser? Why just Onagawa?

MR. JONES: I guess I did not note any documentation of specific damage states at that site.

MEMBER REMPE: Okay. And then also, do we know at Anchorage at all of the plants for the condensers and the piping is similar, not only in Japan versus a U.S. plant but also even at North Anna versus other U.S. plants?

MR. JONES: I guess from my perspective, we're not really worried about quantitative credit. We're just looking at, is there a volume for this to go to and will there be some level of delay in the release of the radioactive material that may propagate through the system? Looking at these assessments, the design-basis leakage rate would be on the order of a couple -- a few hundred standard cubic feet per hour or just a few cubic feet per minute. And even if the

condenser loses structural integrity, it would still provide an area or a volume that would retain the fission products for a while until they find whatever openings may have developed as a result of loss of integrity. And Shilp -- Dr. Vasavada will be going over a little bit more about the fragilities and how a generic determination of structural stability was established for the condensers in the next part of the presentation.

MEMBER REMPE: Okay, thanks.

MR. HSUEH: Dr. Vasavada raised his hand. I just wonder if you want to respond at this time.

MR. VASAVADA: Thanks, Kevin. This is Shilp. I think Steve covered it, and I'll go in more detail. And I can answer additional questions.

Steve's overarching point that you're not giving quantitative credit and you're not trying to draw exact comparisons but get just insights from earthquake experience where plants have exceeded their safe shutdown earthquakes and what that means for the seismic capacity of the ECS components. That's what we are trying to draw other than exact comparisons or numerical credit. And I can talk in further detail or answer any further questions when I go over that.

MR. HSUEH: Thank you.

MR. JONES: Okay. Slide 15, please. Okay. This slide gets into a little bit more detail of the technical assessment. And it primarily looks at the undesirable outcomes considered in the staff's evaluation which reflect the default assumption of a direct ground level release at the downstream main steam isolation valve. Absent these undesirable outcomes, the release would be maintained within the main steam system and other attached piping systems and components.

And these systems and components establish a boundary for fission product holdup under conditions where there's a very low differential pressure to drive any release out. So just having the volume present certainly delays the release and provides the opportunity for additional deposition. Sorry. Slide 16, please. From the operational insights, the staff determined that the typical steam system design used the Power Piping standard and would be designed and fabricated to augmented quality standards which include consideration of a seismic load, design verification, and establish our use of volumetric non-destructive examination techniques to verify the fabrication and construction of the system.

The latest boiling water reactors have

fully safety-related main steam systems designed to withstand the safe shutdown earthquake and design to the ASME boiler and pressure vessel code out to the turbine stop valves or, in some plants, the intermediate stop valves which are located within the turbine building. The main steam isolation values themselves are angled globe valves that are designed to seat more firmly with pressure from the reactor side. The inboard valve may be tested in the opposite direction of its normal seating design.

That is pressure could be applied between the two MSIVs. And therefore, you might see higher than actual leakage during the test. But you still need to maintain test values within the technical specifications.

Another testing methodology may rely on steam line plugs where you test the inboard valves and the outboard valves separately. But again, you're subject to increased leakage because the plugs themselves may be the source of leakage. So I just wanted to address that from the standpoint of conservatism with respect to the overall testing program relative to the technical specification leakage limits.

We also considered the potential -- sorry,

I'm still on Slide 16 -- the potential for stem leakage. But that is considered small and believe would be addressed promptly by operators. Again, we're not looking at specific credit for this particular control, but we don't believe that would be a significant contribution to offsite dose as it would be a release to the steam tunnel in the boiling water reactor. The main steam -- therefore, we concluded the --

(Simultaneous speaking.)

MR. JONES: Yes.

MR. VASAVADA: So this is Shilp again. I think what Steve provided for the ACRS members is considered an answer to the question by Member Hanlon about -- sorry, Halnon, about the leak tightness. Essentially, the ISG, and as Steve pointed out, there is no assumption that it is being bottled up.

We are considering the fact that there can be leakage. It's a comparison against the situation where the downstream piping and the PCS is not considered at all and how that factors into the staff decision. Just wanted to make that point. Thanks.

MR. JONES: Right. And I was going to continue that the main steam and attached systems are there, therefore available to collect this leakage.

There would be leakage potentially directly to the main condenser through the turbine bypass, active drain lines through an orifice, or drain lines where the isolation valves leak. In addition, there's potential for leakage through other paths to the remainder of the power conversion system.

(Simultaneous speaking.)

MR. JONES: But I'll touch on that the next slide.

MEMBER HALNON: Yeah, this is Greg Halnon. Just so I understand, you're saying that the -- it was assumed that the -- any rate of leakage that's typically not measured which is other valves, other steam traps, other areas in this main steam system that maybe have leaked by that that is going to be a small contribution and it was negligible to the overall calcs. Is that essentially what you're saying?

MR. JONES: I guess what I'm saying is that we're only looking for maybe a reduction by a factor of two or three of the release of the full volume of the tech spec limit of leakage from these values to be held up or delayed in such -- well, what I should say, it's not a fraction of the amount. But the effect of a delay, the holdup in these volumes,

and potential for some deposition -- some limited deposition on some surfaces, particularly if it gets to the main condenser, would result in maybe a factor of two or three reduction in what is actually represented in the calculations. And that is intended. One of the other presenters, John Parillo, will be getting into exactly how that was used and what specific assumptions and uncertainties he was trying to address in his evaluation of those consequences.

MEMBER HALNON: Okay. I'll take it all in. And if I don't understand it at the end, I'll ask the question again. But I think I'm getting it. So continue on. Thank you.

MEMBER KIRCHNER: Greg, this is Walt. It seems to me, Steve, with the cubic feet per minute leakage assumption, what effectively is the pressure in the power conversion system? With a condenser, is the condenser just -- is it slightly above atmospheric? In the case where the MSIVs shut, you have some leakage as the tech spec's amount of leakage. Then the pressure in the power conversion system is what, pretty low, isn't it? I mean, so --

MR. JONES: Right, yes.

MEMBER KIRCHNER: -- where I'm going with

this, Greg, is that the leakage compared to normal leakage at operating conditions would be much less if the pressure of the system is significantly reduced which I expect.

MR. JONES: Right. I should probably get into that. The whole dose evaluation process occurs at a delayed time and not at the instant of the accident. But Reg Guide 1.183, I don't have the exact timing in my mind. I believe it's two hours.

But the -- so there's a delayed release. By that time, the containment is at accident pressure. And that's what's acting against the MSIVs. You have an assumption of one failed open MSIV. So the remaining valve is leaking at its tech spec limit for that steam line --

MEMBER KIRCHNER: Okay.

MR. JONES: -- which is on the order of a few cubic feet per minute in these cases. And then the other MSIV lines are also leaking at some reduced rate.

MEMBER KIRCHNER: Okay.

MR. JONES: So that rate does not maintain the main steam system or anything at any significant pressure. It's very close to atmospheric --

MEMBER KIRCHNER: Okay.

MR. JONES: -- by that time because you've had the condensation cool down.

MEMBER HALNON: That helps. Thanks, Walt. I guess my background is biasing me thinking, how do I maintain a tech spec value given a leakage in one cart given the fact I know I got leakage throughout in the system that I'm crediting for some kind of holdup? So I understand now that they're apples and oranges at this point. So you can continue on. Appreciate it.

MR. DOZIER: This is Jerry Dozier. On that -- and I'm only talking from an operations standpoint. But you was asking about the pressure downstream of the second MSIV.

If you look at what the leakage is and compare the leakage as being requested, as one of the team members indicated, that leakage is about what a kitchen fan -- that the flow rate, which is in a cubic feet per hour standpoint, is about the level of a bathroom fan. So picture that bathroom fan kind of going into that downstream piping. And so that's why you would see such a low leakage. So it's low pressure down there that's beyond a second MSIV.

MR. PARILLO: Jerry, this is John Parillo. I just want to mention that a bathroom fan is usually about one to two hundred cubic feet per minute, the

allowable. It may seem an isolation valve limits that we're talking about here are on the order of 200 standard cubic feet per hour, per hour. So I just wanted to make that clarification.

MR. DOZIER: Thank you for that.

MR. VASAVADA: This is Shilp Vasavada from the staff. I just wanted to also, I don't know, clarify one item. I think we have been talking about calcs and credit. And I think Steve mentioned factors of two and four. So I just wanted to make it very clear that as you will see later on, especially in John's presentation, what this ISG and the work that was done that I think was -- that was used by the staff simply as a decision making input to achieve confidence to reach a reasonable assurance finding.

It did not, in any way, shape, or form, change the licensee's calculation. So no factor was applied. No calculations were changed. No number was put into a calculation, no quantitative credit was taken. So I just wanted to make that fact clear. Thanks.

MEMBER MARCH-LEUBA: Sorry. This is Jose. You just confused me. Were calculations performed on the holdup on what pressure? Or are you saying there was no calculation, no deterministic number

calculated?

MR. VASAVADA: So -- go ahead, Steve.

MR. JONES: Okay. Yes, that's correct. There is no deterministic evaluation of this holdup or deposition. It's just considered as a factor in addressing uncertainties with respect to other inputs to those calculations. There are several points where removal of the fission products is modeled by different physical mechanisms at different points in the system. And this is just a consideration in resolving uncertainties with those values. And again, John Parillo will be getting to that later in the presentation. I did want to --

MEMBER MARCH-LEUBA: I would have -- sorry. Keep going.

(Simultaneous speaking.)

MR. JONES: So --

MEMBER MARCH-LEUBA: So this is Jose. I honestly would have liked to see a deterministic calculation because you get surprises when you start modeling things. Clearly if this PCS piping is not on the floor and is still intact, it's good for leakage. I mean, I can't deny that.

But what the pressure is doing there as you're losing cooling in the condenser and everything

is -- whatever atmosphere you had in the PCS before the leak started is going to heat up as you lose condenser cooling. And then you're adding this cubic feet per hour, however much, you get surprises. That's why we do the calculations with good codes and figure out what the output is. I would have liked to see the calculations. Thank you.

MR. JONES: I see. Okay. I did want to touch back on issues that Dr. Rempe raised and Mike Franovich discussed, I guess, with respect to the testing and indication of the main steam isolation valves and the issue with, I guess, balancing their safety performance post-accident with the maintenance and operational dose consequence issues with maintaining these valves that are very high leak type condition at low pressure. So as I mentioned that the valves are intended for -- to seat with pressure at very high normal operating pressure for the boiling water reactors, near 1,000 psi.

During the accident, we're addressing conditions closer to 40 to 60 psi inside the main containment. And evaluating leakage, I guess the operators can -- or the operating companies that maintain these valves have provided a suitable basis to demonstrate that leakage on the order of a few

hundred cubic feet per hour is commensurate with very good operation of these valves. So we have very little concern with the valves not operating properly or not seating.

And we expect that this does not impact the operation of the valves with respect to their performance as a primary reactor coolant pressure boundary valve or a containment isolation valve. And again, we were not really modifying any operating procedures or any inputs to the dose analysis with this evaluation, just considering what the real world impacts of the downstream power conversion system would be on the dose consequences and how that could be used to address uncertainty. All right. Our last slide is Slide -- the last slide I'm discussing --

MEMBER REMPE: This is Joy. Then if that's the answer to my question, maybe I didn't make my question clear enough. And again, I'm not an expert on operator response during a BWR event. But if the -- again, valves sometimes just fail to close. That's why they have reliability numbers.

And if the valve failed to close, how would the operators detect the leakage that's occurring versus the higher leakage rate allowed by the revised tech spec? And are there any actions that

they have to take? And if there's no actions they take, then the answer to my question is it doesn't matter. But typically, you detect whether a valve is open or closed by temperature or radiation readings usually is my understanding.

And are there any actions that suddenly they don't detect that the thing failed to close because it leaks more is my point. And are there any actions that should be considered, because we're going to use some more realistic risk-informed insights to have a lower release to the environment. And I'm wondering if there's other concurrent risk insights that we're forgetting about. Does that make sense what I'm trying to get to?

MR. JONES: Yes, but I'd just point out, I guess, the MSIV leakage detection -- I mean, the leakage detection systems and things in a boiling water reactor are designed to detect ruptures in the main steam system so that the high temperature would be outside in the steam tunnel. And we wouldn't expect any of that to result when these are operating per the assumptions. I mean, this is a very stylistic calculation design to test primary containment performance. It's not really reflective of the most likely outcome of -- or likely configuration that

would exist following a core damage event. So I mean --

MEMBER REMPE: But I mean, if it fails to close, don't you really have a containment bypass? It's open, right? And --

(Simultaneous speaking.)

MR. JONES: You would. And for that reason, I mean, these valves are designed to be redundant and very reliability in their closing. And there is an assumption that one of them does not close. In fact, does not close --

(Simultaneous speaking.)

CHAIR PETTI: Right. Isn't that part of the calculation is to assume that one does not --

MEMBER KIRCHNER: Yeah, single failure.

CHAIR PETTI: And so I think I'm still struggling understanding. There's leakage from all the others. But then there's one that didn't close. It seems like the one that didn't close is going to dominate what's going on.

MR. JONES: Well, there's --

(Simultaneous speaking.)

MR. JONES: -- because there's two valves in a series, the overall leakage is still limited. It's just that line would have maybe higher pressure

acting on the one remaining valve. So you would see -- there's an assumption that there's higher leakage down one particular steam line than the other three. But the other three still contribute some factor to the dose consequence. And again, I think John Parillo is the best one to address that when he gets --

MR. DOZIER: Dr. Vasavada wanted to contribute, I think.

MR. JONES: Okay.

MR. VASAVADA: No, I think -- this is Shilp from the staff -- Steve covered it. But I think the point I was going to make is as Steve said. This is a postulated scenario with certain stylized -- in some cases in my personal opinion -- unphysical assumptions about, like, for example, how long you have choke flow, et cetera.

And those inputs remain unchanged by all the work for the ISG. So the fact -- I mean, it's already assumed that multiple redundant protection systems have failed and you have achieved core damage. And then there is a single failure of one of the MSIVs that's already part of the analysis.

And then the other one leaks at a particular rate which is being requested. And the dose consequence analysis is considered in that rate.

So just wanted to give that background. Nothing about that is being changed by the ISG. Those postulated assumptions remain as they are.

MEMBER KIRCHNER: Thank you. I think also I would've answered Joy -- Joy, this is Walt. On the highly low -- I'm not saying correctly. In the very low probability of two isolation valves on the same steam line failing to close, you would see pressure on the secondary system. So I think the answer to your question, how would you know that the valves didn't seat, it would be a high pressure would build up in the -- to whatever the containment pressure is in the secondary system. So indirectly, you could check on how well the valves seated by monitoring the pressure in the power conversion system.

MEMBER REMPE: Okay. You're getting to my point that, yes, there are ways they could detect this. And again, there's this unusual situation because we did design-basis calculations with a stylized calculation. But then there are the real world.

And in the real world, you're going to allow higher leakage. The tech spec is going to be changed. And so my point is, is that when you allow that, it seems like somebody ought to be thinking

about the real world and a scenario that's not just a design-basis calculation. But should you be considering about the instrumentation?

Now again, Vesna has been sending me emails and she said, I don't think there's anything the operators. There's no actions. And that would've been also a good response to my question. But I'm not hearing that anybody has thought about, is there something that should be done or not?

And again, I'm just curious because I think if we're going to do risk-informed stuff, we ought to -- and a make a change with the tech spec, we ought to think about what the operator should or should not do. And there's no actions they take, oh, well, I guess that's it. But if there are other instrumentations that should be giving them insights, then we ought to think about the difference between a leaky valve versus an open valve. Maybe that ought to be done. Does that kind of explain where I'm coming from a bit more?

MR. JONES: Yes, I think I understand. We're getting a lot more into the detail of how the MSIVs operate. They are spring powered to close, and they do have a pilot system that increases the pressure to close the valves once they're actuated.

And there would be an indication that the valve is fully closed. So the operators would be looking for that. If they did not fully close, they could verify that the air was released or maybe taken action to release air.

But other than the actual -- I'm sorry, an indication of the valve position, there wouldn't be anything for the operators to indicate that there's a problem with the valve because, I mean, under normal operating conditions, you have obviously full steam flow going through these valves and it's very high temperature. So where we're seeing the temperature just gradually falling off in the system and as one of the other members mentioned, there would be potentially secondary system pressures holding up at higher than expected values. But other than that, I wouldn't expect to see anything from excessive leakage beyond the --

MEMBER REMPE: Is there an operator action that would be done? Or would they just say, hey, the thing didn't close? And they'd see it more slowly because they have to figure out whether there's pressure, there's increased leakage that they're allowing because it -- from a much lower leakage. And I just am curious about it. And did somebody think

about this when they were saying, yeah, we're going to let them increase the tech spec?

MR. DOZIER: Dr. Rempe, what I could propose, okay, I was an STA at Grand Gulf, okay? And if you visualize this stylized accident that is done basically for calculation purposes, you would have -- you basically have core damage here -- I mean, an assumed core damage. You have an assumed drywell pressure at that maximum level.

I mean, at the point of especially the initial leakage or whatever, the operator is focused on getting water in the core. They're focused on protecting other containment, things like that. So it's hard. It's very difficult to put this into a realistic situation. But we have done that with some of our risk studies, with SOARCA, and also the original report that was one on this to talk about the releases to the environment.

MR. JONES: I guess what I can say is that we've considered the effect of the increased allowable leakage on valve operation and we see no effect. With respect to operator action, it would be driven again by position indication on these valves. And that would be unaffected by this action as well. If the valve does not close, I'm certain there are actions in

the EOPs, although I can't confirm right now, to address that condition.

MEMBER REMPE: Yeah, if the EOPs take care of it, those are the kind of things I wanted to hear. Yeah, we thought about it and the increased leakage. And it doesn't matter or something, that it's been taken care of in the EOPs. And I just didn't see it. But I may not have seen all the information that's available. I only saw one document from FitzPatrick.

MR. JONES: Right. Okay.

MR. VASAVADA: This is Shilp from the staff. I just also wanted to point out that all of what Steve said and the EOPs, et cetera, that is true for MSIV leakage, increased LARs and reviews and decisions whether the ISG exists or not. So the question about valve closure and indications, the ISG doesn't either change it, improve it, or reduce it. It's the same. It does not change anything from whether the ISG is used or not.

MEMBER REMPE: That's also true. I was more concerned when I was seeing tech specs change for the actual plants. And I only, again, saw one of them. But it's a related topic, and that's why I wanted to bring it up because maybe Daiichi should mention that you are changing a tech spec.

I don't know. Maybe it's a separate topic. And I did start off my discussion that this is a bit off topic. But I was curious about what was going on.

MR. JONES: Okay. I guess we can move on to Slide 17, just again discussing a realistic transport pathway. I did want to clarify a distinction from the BWR topical report. In this -- for this ISG, the staff does not assume any operator action to align a specific path to direct main steam isolation valve leakage to a particular location like the main condenser.

Drain lines and turbine bypass lines lead directly to the main condenser. And if they leak, that would be one way that any leakage release would get to the main condenser. Other leakage paths primarily through the stop and governor valves on the high pressure turbine would go to the high pressure turbine area.

There is definitely less holdup and deposition in the main condenser. But there's still a volume there. And the additional -- those valves are designed again to be fairly reliable and leak tight.

We do not expect with a high likelihood

there would be an easy flow path even to the steam chest and the -- of the main turbine. But if they did get there, then there would be a release path via the high resistance path through the shaft seals -- the main turbine shaft seals since there would not be steam pressure to provide the ceiling steam. There would still be kind of a torturous path for the release to follow to get to the turbine building at that point.

But I just want to point out that we're not necessarily considering complete holdup or that all the flow gets to the main condenser. But there are places that would delay and otherwise reduce the dose consequences from the event when realistically considering the transport pathways through the power converting system. That's all I have.

(Simultaneous speaking.)

MR. VASAVADA: Yeah, this is Shilp. I just wanted to also point out to what Steve said and support that. They're not considering, like, bottling up that if you are to compare that -- again, this is no credit was given. This is a decision making input.

If we are to compare that with the actual quantitative credit for the condenser in Reg Guide, the disparity there because the quantitative credit in

the condenser does not mean that the holdup -- I mean, the deficient part of the water in the condenser. There is a defined, you can say, leakage rate from the condenser. So the condenser is considered, quote, open, end quote. So the disparity there to restart as if the condenser is leak tight if quantitative credit using Reg Guide is used.

MR. JONES: Okay. And with that, that was the end of my section with this presentation. And I'd like to hand it over to Dr. Vasavada to go over the seismic considerations.

CHAIR PETTI: Yeah, so maybe this is a good point. We usually take a break around 9:00 o'clock before we get into the seismic stuff. Is that okay with you guys?

And we take about a 15-minute break and then reconvene at the top of the hour to start the seismic.

(Whereupon, the above-entitled matter went off the record at 10:46 a.m. and resumed at 11:00 a.m.)

CHAIR PETTI: Okay. I have top of the hour, so let's reconvene and start with the seismic slides. Thank you.

MR. VASAVADA: Okay. Thank you. This is

Shilp Vasavada. I hope you can hear me.

CHAIR PETTI: Yes, we can.

MR. VASAVADA: I'll start. Good morning to everyone. As I said, my name is Shilp Vasavada. I'm in the Division of Risk Assessment in NRR, and I'll be discussing the seismic capacity evaluation for the SSCs in the power conversion system or PCS that is documented in the technical assessment for the ISG.

As many of you may know, the SSCs in the PCS do not need to be seismically qualified primarily because they are not safety related. Therefore, the context for the seismic capacity evaluation, if you think about the risk triplet for the ISG is to understand the risk of gross failure of the SSCs in the PCS, especially the safe shutdown earthquake of the plants. The intent is not to, again, provide a factor for reduction of the dose.

It is to see whether the SSCs in the PCS have a high confidence of surviving the safe shutdown earthquake in a realistic scenario or considering realism, provide holdup volume that the staff can use that's realism in its decision making. It will -- overcoming any challenges with uncertainties in other parameters. Before I provide an overview of the evaluation, I'll just go over a primer on some seismic

capacity-related terms that you will be hearing.

They may be new to you. But they are meant to help with common understanding and the communication. I'm on Slide 18, and the terms over here, I'll give a caveat. They're not textbook definitions, please don't hold me to that.

Firstly, fragility, it's the conditional probability of failure of an SSC as a function of seismic acceleration. And one of the common ways of expressing fragility is what's known as median fragility, also known as A sub m. This is a seismic acceleration at which there is a 50 percent probability of failure.

And along with A sub m, there are two uncertainty parameters, beta r and beta u as they are called, to which together define the median fragility of an SSC. These uncertainty parameters characterize the, again, uncertainty in the median fragility or the fragility of an SSC. The seismic acceleration is the measure of the strength of an earthquake.

It is usually expressed in terms of multiples of the gravitational acceleration. So you'll hear terms like 0.1g, 0.2g where g is the gravitational acceleration. And peak ground acceleration is a commonly used term -- or commonly

used acceleration level for seismic analysis.

It corresponds to the acceleration of a 100 Hertz oscillator. A lot of times, a lot of analyses are so to say anchored to the peak ground acceleration to provide a common language for comparison and use. Next slide, please. So I'm on Slide 19.

And this slide, basically, the figure over here brings all the terms that we discussed in the previous slide together and pictorially. You can see the curves which are the cumulative fragility curves, the 95th percentile, the 50th percentile, and the 5th percentile. As previously explained, the fragility is a function of the seismic acceleration and peak ground acceleration in this case which is the commonly used acceleration value.

In this example, so the median fragility is 0.8g -- 0.87g. So you have a 50 percent probability of failure of this example, SSC at that particular acceleration. One concept that I wanted to also share is that a higher median fragility value implies a more robust or a higher seismic capacity SSC.

The reason being that a higher median fragility value shifts all those curves to the right.

So you have -- for the same seismic acceleration, you'll have a lower failure probability. Next slide, please. So I'm on Slide 20.

To evaluate the seismic capacity of the PCS components, we looked at several diverse sources of information that compiled fragility data to get an idea of where the fragility is of different components that may exist in the PCS line. We also looked at insights from both earthquake walkdowns. And we performed representative risk calculations to estimate the risk of gross failure of the SSCs in the PCS.

For the fragility data, we looked at the various NUREGs and industry reports, including EPRI reports as well as information we recently submitted, a seismic probabilistic risk assessments or PRAs, that were submitted in response to the Agency's post-Fukushima's actions. As many of you may know, seismic PRAs usually don't model PCS components or balance of plans components as they are called. However, the seismic PRAs do carry information related to the fragility of several other components which exist in the PCS like welded and bolted piping, valves.

And they also provide information about the extent of seismic risk accelerations up to the plant safe shutdown earthquake. All of that is

relevant to this evaluation. In terms -- Jerry, can you go to the next? I'm still on Slide 19.

We also reviewed the post-earthquake walkdown experience for North Anna in the U.S. and Kashiwazaki-Kariwa in Japan and Onagawa also in Japan following the Great Tohoku Earthquake of 2011. All these plants experienced earthquakes that exceeded their respective safe shutdown earthquakes and equivalent in Japan. And we focused our observations over there for the PCS components and the impacts -- post-earthquake impacts that were observed or not on those components.

As I think in response to a question, I'll just mention it over here. And I can additional detail if that's necessary. The ISG nearly mentions that the purpose of the reviewing the walkdowns was not to draw one is to one comparisons.

We recognize it's plant-specific, it's design-specific, and location-specific and also maybe operating practices-specific. The reason for doing the evaluation of the post-earthquake walkdowns was to identify insights related to the behavior of PCS components in general: were there any gross failures that were observed, were there any issues that we need to consider in our evaluation, and to take it in

conjunction with the data as a body of evidence which then was used to determine and lower bound median frigidity value which we believe encompasses the seismic failure modes for the SSCs in the PCS and then use the representative risk to determine what would be the representative risk of gross failure of the SSCs in the PCS. Next slide, please. I'm on Slide 21.

So the insights that we gathered from all of that evaluation was that welded piping, bolted piping, as well as valves have high median fragility values which as we talked about implies that they have high seismic capacity. In addition, the main condenser is usually -- I mean, as all of you may know -- a huge structure which is bolted to the floor of the turbine building. It's usually a seismic Category II structure, so the anchorage is designed to avoid failure at design-basis loads to prevent what's called a Seismic II or I interaction.

In addition, all the post-earthquake walkdowns of plants in the U.S. and Japan demonstrated that the PCS components have high seismic capacity. No gross failures or major issues were identified for even the nonsafety-related PCS components at any of those post-earthquake walkdowns. And finally, the seismic PRAs demonstrated that the seismic risk from

acceleration at accelerations below -- at and below plant safe shutdown earthquake is small.

The contribution to the overall plant risk -- seismic risk from those acceleration levels is a very small fraction. Next slide, please. So based on all of that, the information, we determined a lower bound median fragility parameters for the use in our evaluation. These parameters as displayed over here are the median fragility value of 0.4g and the beta r and beta u of 0.22.

They are based on the fragility of an expansion joint connecting the circulating water piping to the condenser. And based on, again, the evaluation and the survey of data and the walkdown information, we believe that it encompasses all the failure modes for the relevant SSCs in the PCS piping, et cetera, and valves. And it supports the low likelihood that the gross failure of the SSCs in the PCS would not occur.

For context, as I mentioned, the --- but for parity, when the actual quantitative credit for holdup in the condenser is taken, the condenser is considered code open with a specified leakage rate from the condenser. So using this lower bound is also actually conservative because it is essentially

treating the entirety of the PCS at a very lower fragility to see what would be the failure and release if this fragility were to be exceeded as compared to just considering a small level of leakage which is anyways allowed or considered when the credit is taken -- quantitative credit is taken. Next slide, please. So I'm on Slide 23.

Using that lower bound fragility value, what we did was we, it was called a range of seismic hazard curves with that condition failure probability from the median fragility values to determine what is the -- you can say frequency of release due to the gross failure of the SSCs in the PCS. And we used a range of recently developed seismic hazard curves which are developed in response to the Agency's post-Fukushima actions. And our results -- estimates showed that the risk is low from -- the risk of release from a gross failure of the SSCs in the PCS is low.

And even if you consider the entire seismic hazard curve, if you were to consider only up to the safe shutdown earthquake as is necessary for the purposes of this -- I mean, this entire MSIV dose calculation, the risk would be even lower. So next slide, please. So Slide 24 talks about the

uncertainty, how we considered the uncertainty for the seismic capacity evaluation. In terms of the uncertainty, the --- in the selected parameter, that, I mentioned, is already included explicitly to the beta r and beta u parameters and used in the calculation of the representative risk estimates. In order to address the conservatism in the selection of the median fragility parameter, that is, is 0.4 sufficiently lower bound? Or it should be 0.3 or should be 0.5?

We looked at the conservatisms that already exist brought in our evaluation as well as in the overall MSIV calculation to address that uncertainty. So we are using a lower bound region fragility as we mentioned to kind of encompass all potential failure modes in spite of the fact that several of the SSCs show much, much higher median fragility values. And it is not necessary to have a leak tight approach for the decision making that the ISG provides.

As we have noted, there'll be low pressure conditions because of the postulated scenario. So the piping and the SSC in the PCS are designed for high pressure and high temperature. So the margin that is in that design is not explicitly being accounted for,

especially with the choice of the lower bound fragility parameters.

We are considering the safe shutdown earthquake occurring concurrent with a postulated accident that results in complete core damage. So that's an additional level of conservatism. And as we have said multiple times, the important assumptions, parameters, guidance, boundary conditions that go into the actual dose calculation remain unchanged by this ISG. So that's additional conservatisms in there which we have not changed or taken advantage of.

MEMBER KIRCHNER: Shilp, this is Walt Kirchner. On a previous slide, you identified the weak link in your analyses -- seismic analyses as the expansion joint for circulating water to the main condenser. But that's not a leak path unless you have massive tube rupture in the condenser. So was there -- what was the second most fragile component in the PCS systems that actually are forming the holdup volume?

MR. VASAVADA: All right. So we didn't -- or at least I didn't go line by line through all that exists in the PCS in order to determine the next -- sorry, Steve, did you want to say something?

MR. JONES: Yes, sorry to interrupt. But

I mean, just to clarify, the expansion joint is the boot between the bottom of the low pressure turbine and the condenser, not -- I mean --

MEMBER KIRCHNER: And it wasn't explained --

MR. JONES: -- there's still an expansion --

MEMBER KIRCHNER: And it wasn't explained --

MR. JONES: -- joint there.

MEMBER KIRCHNER: -- correctly in the viewgraph.

MR. JONES: Okay.

MR. VASAVADA: Sorry about that, yeah. And again, to --

(Simultaneous speaking.)

MEMBER KIRCHNER: Okay.

MR. VASAVADA: -- that question, if --

MEMBER KIRCHNER: That's part of a leak path then? Okay. Thank you. That answers my question.

MR. VASAVADA: Okay. Thank you. All right. So that was basically an overview -- I'm still on Slide 24 just closing up -- overview of the seismic capacity evaluation that is included in the technical

assessment. And in summary, in the context of the ISG, the seismic capacity evaluation supported the case that the SSCs in the PCS have a high seismic capacity. And the risk of release, because of the gross failure, especially at the safe shutdown earthquake level, is low.

And taken together with the other elements of the assessment that were described by Steve and Jerry, it supports the insights and the recommendation that the ISG provides to the staff to consider this realism in their decision making if they are challenged because of any uncertainty in input parameters in the rest of the dose calculation. Again, I'd like to reiterate the seismic capacity evaluation or the ISG evaluation does not change the dose calculations. They remain the same.

As submitted by the licensee, the margin to the acceptance guidelines remains the same. It's just a decision making input to provide staff the confidence to reach reasonable assurance if there are challenges with uncertainty in a parameter or the other. Next slide, please. I'm on Slide 25.

And I wanted to, again, put this ISG in the context -- a different holistic context of the entirety of the MSIV evaluation to kind of represent

the fact that it's a small sliver in the entirety if you consider the whole dose calculation approach. So first of all, the licensee's calculations will start with the assumption of core damage arising from a postulated scenario with failure of multiple redundant protection systems. In addition to that, there is a single failure of the inboard MSIV which is assumed.

In addition to that, there are conservatisms in the analysis -- in the guidance which include acceptable assumptions and parameters. I'll give an example that, for example, choked flow is assumed for 24 hours so that the leakage flow can be at the tech spec limit for 24 hours as part of the analysis. That conservatism remains in the guidance. It remains unchanged.

As I mentioned, we assume that a safe shutdown earthquake concurrent with this postulated scenario. And we use a lower bound median fragility value to kind of see what is the risk of gross failure. So in the context of the entirety of the evaluation, this ISG forms what I would call a small sliver of realism in a universe of conservatism.

And again, the purpose was not to change the dose calculations. The purpose was to provide the staff with additional confidence to reach the

reasonable assurance if there was talk about whether X acceptable value is okay for a particular parameter or Y is okay. Next slide, please. I'm on Slide 26 now.

So before I pass it on to John for more details, I wanted to provide a kind of comparison between what the ISG intends to do and what the proposed or planned revision to Reg Guide 1.183 is intended to achieve. And this slide attempts to clearly differentiate between the two. The ISG obviously is directed to the staff to support their decision making whereas the Reg Guide will be directed to the licensees and provides acceptable means of showing compliance with regulations.

The ISG, again, as I've been repeating it again and again, provides staff additional confidence to reach its reasonable assurance finding. It is not the only reason that the staff would reach the finding. It is providing additional confidence.

The Reg Guide provides acceptable methods, including it would provide the method and the guidance for getting quantitative credit for holdup in the main condenser which the ISG does not provide any quantitative credit. So the ISG, because of that reason, does not change the dose calculations of the

analysis of record that the licensee has submitted and that the staff uses for its reasonable assurance finding. The Reg Guide revision, if the licensee were to take -- follow that guidance for quantitative credit, does change the dose calculations.

It does introduce if -- I believe the term is decontamination factor which does reduce the dose. The ISG does not do that. So for that purpose, the ISG does not need information from the licensee. It's a decision making tool for the staff. It is expected that the revision to Reg Guide 1.183 for quantitative credit in the condenser would be requesting docketed information from the licensee for that purpose. I see Member Corradini's hand up for a question.

MR. CORRADINI: Just for clarification, so the licensing basis calculation -- dose calculation would not be -- would be changed. But this is an internal document the staff would use for any subsequent LAR or similar LARs? Am I -- I'm still not clear about the use of the ISG. That's where my question is coming from.

MR. VASAVADA: Sure. So I'll give, you can say, an answer. And it will be maybe fleshed out in more detail by John as he goes through the details. I'll give -- let's start with an example.

MR. DOZIER: Mike Franovich has his hand up.

MR. VASAVADA: Okay.

MR. FRANOVICH: I don't think I did, Jerry, but --

MR. DOZIER: Oh, I'm sorry. Another Michael, I think, had their hand up. I'm sorry. Or Dr. Kirchner?

MEMBER KIRCHNER: Yes, thank you. Just to be precise here, I'm presume that the using in the spirit a holistic approach provides guidance for quantitative credit for holdup in the power conversion system, not just the main condenser, albeit the main condenser is probably the largest volume. The turbine is probably the second largest volume. The piping, I don't know where that all -- how much volume that adds up to. But it would be for the full PCS, right?

MR. VASAVADA: So I'll address that one first and then go back to Member Corradini's question. In that context of Reg Guide 1.183 for quantitative credit, it is the condenser. It is not the PCS because over there, the way it works is that a pathway to the condenser has to be opened up.

And that pathway is given credit with, you can say, numerical decontamination factor which

decreases the dose because of the decay in the condenser. So it was just the condenser. And obviously, yeah, the lines that need to be opened up to -- for example, the drain lines that need to be opened up to get to the condenser. Does that help?

MEMBER KIRCHNER: Yeah. Well, unless the -- well, I'm trying to think through a BWR response to an accident. I guess the main turbine stop valve probably closes so that -- and doesn't have -- usually have a direct path into the turbine itself which is a significant volume, although the leakage paths from turbine are minimal. But okay.

MR. VASAVADA: So that is the --

MEMBER KIRCHNER: I assume you would just let the applicant assess his or her particular design and the line -- what would be the configuration of the PCS under the accident condition that's assumed?

MR. VASAVADA: That's correct. And what you just said about the configuration you're thinking about is the thought process we used for the ISG. But it does not provide quantitative credit. It is a realistic kind of decision making input using insights, operation and seismic insights, to support the staff that in reality this is what will happen.

So if you are challenged with some input

assumptions, you use this to get confidence in your decision that using accepted assumptions that doses are just calculated by the licensee would likely be lower if you were to think about it realistically. And you can still go out and make your conclusions. For the context of taking quantitative credit by the licensee to actually make a numerical change in their dose calculations, the guidance would be in 1.183.

And that just talks about opening a path to the condenser and credit for the condenser. And that would be a licensee's decision if they want to take credit or not. So I'll go back to Member Corradini's question.

MR. CORRADINI: You know what? I don't think you have to. The way you answered Walt helped me out. So I'm fine. Thank you.

MR. VASAVADA: Okay, thanks. And I'll continue on this slide. Again, to repeat, the ISG does not change the licensee's responsibilities to show compliance or to -- it does not change the acceptable methods to demonstrate compliance for the guidance. I think that ends my portion of the presentation. I'll turn it over to John Parillo to discuss how the content of the ISG was actually used to support the staff's reviews of recent LARs. John?

MR. PARILLO: Thank you, Shilp. This is John Parillo speaking. I am a member of the Radiation Protection and Consequence Branch in the Division of Risk Assessment. I was a contributor to the review of the James A. FitzPatrick license amendment requesting an increase in their allowable main steam isolation valve leakage limits.

The FitzPatrick plan had the lowest MSIV leakage limits in the USBWR fleet and requested leakage limits more in line with the rest of the operating fleet. Slide 27 includes excerpts from Section 50.67, accident source term, and highlights that the rule states, the NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that specific dose acceptance criteria will be met. Slide 28. Slide 28 contains information pertaining to challenges that the NRC staff has encountered when reviewing applications, requesting increases in main steam isolation valve leakage limits.

Regulatory Guide 1.183 does not contain an aerosol deposition model suitable for the evaluation of the dose consequences from main steam isolation valve leakage. In addition, the staff has concerns with the settling velocities used and researches

accident evaluation branch AEB-98-03. Due to the lack of a main steam line aerosol deposition model in Reg Guide 1.183 and issues with some of the assumptions in AEB-98-03, the NRC staff has issued many requests for additional information, questioning the aerosol deposition models submitted for NRC review.

Many licensees have incorporated concepts from AEB-98-03 with additional conservatisms in their licensing basis. Slide 29. The 2006 regulatory information summary included general concepts that licensees should consider when modeling main steam line deposition. However, the RIS did not provide an acceptable model or a reference to one that licensees could follow to provide some level of confidence in gaining staff acceptable of their license amendment requests. Slide 30.

Shortly after publishing of AEB-98-03, the staff identified concerns with the methodology used in this document. In spite of these concerns, the RIS stated that it is acceptable for licensees to continue to reference AEB-98-03 but that their deposition model needs to reflect individual plant characteristics. As evidenced by the continuing numerous requests for additional information pertaining to main steam line deposition modeling, the RIS did not resolve the

ongoing issues encountered when licensees submitted license amendment requests for increasing MSIV leakage limits. In 2009, the --

CHAIR PETTI: This is Dave. I have a question, just a clarification. As I recall reading in FitzPatrick, is it the aerosol modeling in the steam line or in the drywell or both that have been the concern?

MR. PARILLO: Well, primarily in the main steam line. And specifically, we will discuss the -- there's a particular uncertainty we had in trying to evaluate the combination of the aerosol deposition in the drywell due to drywell sprays and then that subsequent deposition in the main steam line which was the primary uncertainty in this review. And in fact, it was the primary uncertainty in all of the four recent MSIV leakage license amendment requests.

CHAIR PETTI: Okay. Thank you.

MR. PARILLO: Thank you. Let's see. Where was I? In 2009, the NRC published Draft Guide DG-1199 for public comment as a proposed revision to Reg Guide 1.183. Draft Guide 1199 provided a model for assessing MSIV leakage. However, the approach described in this Draft Guide provided very challenging for licensees and has never been

implemented in a license amendment request.

The proposed model in DG-1199 did not credit safety-related equipment to distribute the deterministic source term for the first two hours of the evaluation. This assumption has been determined to be inappropriate for design-basis assessments as described in the case file for differing professional opinion, DPO 2020-2, and is being eliminated in the proposed revision to Reg Guide 1.183. Slide 31. Slide 31 summarizes a significant uncertainty encountered in the review of the recent MSIV leakage license amendment request.

The interaction between aerosol removal by drywell sprays and the subsequent aerosol removal due to main steam line deposition as modeled by the licensees was questioned. Sensitive analyses were submitted that indicated that if the power conversion system is assumed to be intact providing a pathway to the condenser, the dose reduction would be substantial. The effectiveness of an intact power conversion system providing a pathway to the condenser is acknowledged by the NRC staff as evidenced by assumptions in Appendix C to Reg Guide 1.183 for the evaluation of the boiling water reactor rod drop accident.

Due to model limitations and for additional conservatism, licensees only credit a deposition in two of the four main steam lines. This conservatism alone accounted for an approximate 30 percent increase in the calculated dose consequences. And in addition, licensees did not credit mixing between the drywell and the wetwell air space for the first two hours of the accident. This assumption has also been determined to be inappropriate for design-basis assessments as described in the previously mentioned DPO case file. Slide 32.

Slide 32 describes the licensees' accident analyses of record and how the NRC staff use the insights described in the ISG to reach its conclusion of reasonable assurance. As stated, the key points are that, one, the licensees' sensitivity analyses are not part of their licensing basis, two, a pathway to the condenser was not credited in the analyses of record, three, the licensees provided analyses which met the acceptance criteria in 10 CFR 50.67, and most important for this discussion is that the staff's determination of reasonable assurance was supported by the recognition that there is a high probability that doses will be significantly lower than those estimated by the licensees using deterministic methods that do

not credit holdup and retention of the main steam isolation valve leakage within the power conversion system.

This essential concept explains the relationship between the review of the four license amendment requests and the concepts presented in the ISG. I'm looking to see if there are any questions. Oh, well. No hands raised. So now Jerry will review the key takeaways from today's presentation.

MR. DOZIER: Okay. So as we've demonstrated today, the ISG will result in consideration of large holdup volume in future MSIV leakage LARs. It offset some of the certainty and input parameters for deterministic calculations, supports the reasonable assurance findings during reviews, and is only applicable if quantitative credit is not included in the licensee's calculations. As we've emphasized, the ISG does not change the licensee's responsibility to demonstrate compliance with 2 CFR 50.67.

Acceptable methods for demonstrating compliance remain unchanged. ISG is expected to be transitioned to the standard review plan. And we've got more work to do because formal condenser holdup credit for licensees is being considered in the

revision to Reg Guide 1.183. That concludes our presentation.

CHAIR PETTI: So question, it's being considered or it will be incorporated?

MR. VASAVADA: This is Shilp. If I can answer, I think -- this is Shilp Vasavada from the staff. First of all, I mean, being considered may not be the right words because it's already there in Revision 0 of Reg Guide 1.183. What I think we are trying to say over here is the insights that we gained from the development of the technical assessment for the ISG will be leveraged to see how we can streamline the credit -- getting the credit in the revision to 1.183.

CHAIR PETTI: Okay. Thank you.

MR. DOZIER: And in other words, it might change it because realize that if you look at that old topical report, it actually goes to looking at fossil plans. With this new approach that we're looking at, we're looking at a wide range of more modern information, for example, the Fukushima operating experience, all these lessons learned that we've gotten from Fukushima. So we're looking at in the update, maybe we could use these insights to change, I want to say, our archaic way that we had done

before.

CHAIR PETTI: Okay. Thank you.

MEMBER REMPE: Dave, this is Joy. Can I ask a question again?

(Simultaneous speaking.)

MEMBER REMPE: I just want to confirm what I heard from Shilp earlier. Shilp, did you tell me that all the BWRs do have positive indication of closure for the MSIVs?

MR. VASAVADA: I think -- I believe it was maybe Steve. But Steve, can you --

MR. JONES: Yes, this is Steve. The indication would be from the stem position of MSIVs. But in that sense, there's indication of closure.

MEMBER REMPE: So my question is irrelevant because it would not -- you can immediately detect closure. So the fact that you have higher leakage does not matter. Is that the answer to my question then?

MR. JONES: Well, I think the higher leakage is really independent. We're talking about very small leakage still. And the indication of valve closure is really just greater than 90 percent closed. And you have the valve being spring actuated would drive it to full closure. So --

MEMBER REMPE: Okay.

MR. JONES: -- in that sense, I don't see a lot of follow-up. If you get the indication of closure, there would not be a need for operator action.

MEMBER REMPE: So there would not be any change? Okay. I just wanted to make sure I fully understood it because it is just something that was concerning me. So thank you.

MR. JONES: Okay. Thank you.

MEMBER HALNON: This is Greg. I get a little bit confused. First of all, Joy, there's other indications, steam tunnel temperatures and other things that the operators would expect to see upon closure versus not. So it's not just based on indication. But back on Slide 26, we talk about that the licensee from Reg Guide 1.183 provides guidance for quantitative credit for holdup in main condenser. And then this last slide was that there's no credit for holdup and condenser.

And I guess I lost the bubble here somewhere. What are licensees taking credit for in their deterministic calculation, just the general PCS deposition rates? Or was it always predetermined that they could just redo their -- take out some

conservatisms or add some depositions such that they could get a higher leakage rate anyway without any other conservatisms being taken off?

I kind of just lost that last statement that said you don't take credit for the holdup. I think I lost the overall concept. So help me with that. Just reconcile the two statements, the one on the last slide and the one on Slide 26.

MR. VASAVADA: John, do you want to take that? Or do you want me to start?

MR. PARILLO: Yeah. If I understood your question, we have a bit of a discontinuity in the existing Reg Guide. Appendix C which is for the control rod drop accident for boiling water reactors, that provides assumptions that licensees use which implicitly assumes that a pathway to the condenser and that allows for a certain deposition in the condenser. But more importantly, it allows you to release the effluent one percent per day.

Now that accident is only evaluated for 24 hours. But that, for a long-term release, would be very, very significant reduction. But for Appendix A which deals with -- well, currently, it's referred to as the loss of coolant accident. But we prefer to use the term, the maximum hypothetical accident, or, the

accident that's described in the regulation in a footnote.

And for that evaluation, Reg Guide 1.183 allows you to credit systems downstream of the second MSIV leakage -- MSIV valve, excuse me, main steam valve -- isolation valve, providing that they -- words to the effect that they could survive a safe shutdown earthquake. So there's a difference in the guide. We're applying a more strict interpretation for the MHA than is applied for whatever you wanted to call lesser accidents.

So the credit now does exist for a pathway to the condenser in the current guide. What we're anticipating -- and of course, this is all pre-decisional. But what we're working on is to provide some guidance that could be followed for the maximum hypothetical accident which could -- we would have reasonable assurance of a pathway to the condenser. And we will also provide certain removal coefficients that could be used for that particular evaluation.

MEMBER HALNON: Okay. That reconnected the dots. Thank you. I appreciate it.

MR. PARILLO: Thank you.

MEMBER REMPE: So Greg, now you're getting back into the, oh, well, yeah, but then they also

confirm it. Well, will the confirmatory information be less distinct is what I'm trying to say. And again, is the flow rate so big that there are no needed changes in the EOPs, because there's no mention of this at all in the FitzPatrick response from the staff.

And so again, were there -- and I heard earlier, well, I think there might've been some changes in the EOPs, I believe from Steve. Were there some changes in the EOPs? And is that something that's always done? Or it's just, no, this does not matter?

MR. PARILLO: Well, maybe I should take it. This is John Parillo again. I think a key to understanding what went on with FitzPatrick, I mean, they -- like I said before, they -- prior to this recent amendment which was granted, FitzPatrick had the lowest MSIV leakage limit in the fleet, 46 standard cubic feet per hour.

Now they -- by virtue of this amendment, now they will have a leakage limit, total leakage of 200 standard cubic feet per hour. Now I'm not an operator. But if you think -- I think what you're asking is what does that change -- what does the impact of that change and is that going to have a

significant impact on operator actions?

And I would say that you're going from an incredibly --- to a very low leakage. And I don't think offhand that there would be any perceptible difference downstream. But I would defer to an operator to verify that. But my gut feeling is that it's not something that would require any kind of a change in an emergency operating procedure. But that's --

MEMBER REMPE: And this was a plant that had very low leakage valves. And again, this is going on with the operating -- some of the operating fleet.

MR. PARILLO: Yeah, well --

MEMBER REMPE: I'm just curious. Would it be too much to ask if you could pull the string and say, yeah, it was considered even if the staff didn't do it in their review of the LAR by the actual plants and they did not need to make any changes to EOPs or they did but they thought about it? That's what I'm trying to ask today.

And it's not -- again, maybe I'm slow. I'm also not an operator. But I just am curious because it's of interest. It could be a safety issue. I just am curious.

MR. DOZIER: For that particular

consideration, of course the things that the licensee would need to do is being considered in the Reg Guide update. So certainly, I think we could -- we would be asking those questions, what would need to be done? So I think the answer to your question, Dr. Rempe, is yes.

MEMBER REMPE: Yes, you are asking that question and so it is being done. That's what I wanted to hear today.

MR. JONES: This is Steve Jones. I think just to clarify, there is an evaluation that's included in the BWR topical report dealing with credit for this type of main steam isolation valve, leakage, holdup, and deposition. And one of the main concepts is that these higher level leakages do not represent any condition that's outside what you would expect from a good or a very well performing main team isolation valve.

In other words, as John said, we're just going from extremely low leakage to very low leakage to get these type of leakage limits. Three cubic feet per minute at 45 psig is still a very tight valve for this size. We're talking about nearly a foot diameter of the seat. So it's just, again, balancing how frequently these valves are reworked and the operator

dose associated with that versus how well they perform in the absolute containment of fission products post-accident.

MEMBER REMPE: So I'll try and paraphrase again. You're saying that there is some sort of BWR topical report where the staff does -- or a Reg Guide that would guide the staff to ask that question. Are there any changes that need to be made to the EOPs if there were a need to verify that the MSIV has closed in light of the increased leakage?

MR. JONES: Well, what there is, is there's a statement from industry. And the staff has accepted that the -- that in the leakage -- in the range of leakage limits we're discussing here, there's very high confidence that the valve would continue to perform as designed. There's no degradation or any change in response that's expected from this change.

MEMBER HALNON: So Joy, this is Greg. I have never put my eyes on it, but I am pretty sure that there's, in EOP, the symptom-based EOPs were increasing, radiation increasing heat in the steam tunnel, other things that would drive you back to verify that the MSIVs are closed.

MEMBER REMPE: I would think so too. And then the question is with the increased leakage, is it

harder for them to detect it, is what I'm kind of asking. And again, I'm kind of -- I'm not getting a real strong answer that I expected to get when I raised the issue. Maybe it's because I don't communication very well.

MR. JONES: The steam tunnel temperature monitoring and things like that is outside the -- I mean, it's in the steam tunnel. But it's not touching the piping or anything like that. So you're not going to see that response at the type of leakage limits we're referring to here. Again, it's very low. And in terms of mass, a very, very low leakage rate.

MEMBER HALNON: Yeah, the point is --

MR. JONES: And they can't transfer much energy.

MEMBER HALNON: The point is the operators are trained on scenarios that MSIVs have either failed to close or have increased leakage through. And I think what Joy is looking for is some validation, one, was that already there, and two, was there any additional actions they may need to take based on the fact that we're increasing the leakage limit? So that's the --

MEMBER REMPE: And actually, it's only the latter point. I'm pretty confident too it's there.

But the question is does it need to change? And that's what I am trying to get to. And --

(Simultaneous speaking.)

MR. JONES: Yeah, the value was just too low to be significant with respect to the valve performance or the effect it would have on the downstream --

(Simultaneous speaking.)

MEMBER REMPE: And I would buy that. If that's the answer, that makes sense to me and so I'd buy that. What I -- I guess I didn't hear that till this last time, and I know I kept berating it. But again, if that's the case and you're confident that that is the case, it's just too low, it would not matter, that would be fine. I just was curious when I saw that.

MEMBER KIRCHNER: Yeah, Joy, this is Walt. I would second Steve's answer. Basically, the inventory of the PCS, for these low leakage rates, you're not going to pressurize the power conversion system with these low rates of leakage. There's so much -- there's so many thermal -- there's so much thermal loss in the system that with this low leakage, it's unlikely you can pressurize the volume that the PCS occupies. So you're not going to see a

significant difference within the bands that the staff has been talking about in terms of increased leakage from a shut main steam isolation valve.

MEMBER REMPE: Again, that answer makes sense, but it took a while to get to that answer. And that's what I -- again, I just wanted to make sure that either it was thought out or it was thought about and dismissed. But I didn't hear that as clearly until this last time. So thank you for your tolerance of me bringing it up.

MEMBER KIRCHNER: I have a question to the staff. On the viewgraph in front of us, you talk about holdup. Does holdup include deposition? Does this allow -- if the licensee, the applicant can demonstrate a credible deposition model, does -- is he or she allowed to take credit for that as well? You seem to use the term, sometimes, interchangeably. But I think holdup means something different to me than deposition.

MR. PARILLO: This is John Parillo again. I could refer you to Appendix C of Reg Guide 1.183. I believe it provides a deposition of -- I believe it's 10 percent for iodine. And then the holdup is addressed by a leakage of one percent per day.

(Simultaneous speaking.)

MEMBER KIRCHNER: And that was my second question, John. That leakage is based on one percent per day. Is that the venting of the air ejector condenser? Or is that just a cumulative estimate of what a PCS would --

(Simultaneous speaking.)

MR. PARILLO: It's just applied to whatever the source term is that reaches the condenser in that particular accident.

MEMBER KIRCHNER: But where did you come up with the number?

MR. PARILLO: The volume -- the leakage goes into the condenser volume, and then it leaks out of that volume at one percent per day.

MEMBER KIRCHNER: And that's based on the venting of the air ejector condenser or just leakage --

MR. PARILLO: No.

MEMBER KIRCHNER: -- in the system?

MR. PARILLO: No, no. I'll let --

MR. JONES: This is Steve Jones. I guess I can address some of the systems aspects of this.

MEMBER KIRCHNER: Yeah, please, Steve.

MR. JONES: I mean, we're talking about a rod drop accident. So you'd have as the source a

localized high power within several rods -- maybe more than several -- but anyway, a high power condition that results in cladding rupture. And then that's diluted throughout the rest of the RCS and comes out in the steam system. The -- so it would be transported through the turbine to the condenser pretty rapidly. Once it's in the condenser, there are high radiation monitors that would close and isolate the vacuum pumps that are connected to the main condenser. So --

MEMBER KIRCHNER: Okay. So then it's just --

(Simultaneous speaking.)

MR. JONES: -- we'd just be talking about leakage.

MEMBER KIRCHNER: Okay. Does that take on then -- do you then require additional quality assurance requirements on that function of isolating the air ejector condenser?

MR. JONES: We --

(Simultaneous speaking.)

MR. JONES: I mean, that is a safety-related instrumentation function to isolate. But the pressure boundary itself is not.

(Simultaneous speaking.)

MR. JONES: I mean, there's a -- there is also a high radiation trip, primary containment isolation that goes with that. So I mean, all these things are happening at once. So you're going to get just a part of that source term is going to get into the power --

MEMBER KIRCHNER: Right.

MR. JONES: -- conversion system. And then it's going to leak out over a period of time. That's really what you're modeling. And so the distinction between holdup and deposition is holdup is there's just a volume there that --

MEMBER KIRCHNER: No, I understand that.

MR. JONES: -- retains --

(Simultaneous speaking.)

MEMBER KIRCHNER: I just wanted to distinguish whether you also allowed credit for deposition because your viewgraph just talks about holdup. Okay. I get it. Thank you.

MEMBER BROWN: Dave, this is Charlie. Can I ask a question?

CHAIR PETTI: Sure.

MEMBER BROWN: This is definitely not my bailiwick, okay, BWRs in particular since I was a PWR person. But correct me if I'm wrong. But it sounds

like the whole reason for this, the ability to get to this alternate source term, is the deposition and the holdup in the condenser.

And all I heard was model, model, model all the way through the presentation. So I'm not objecting to that. But has there ever been any experimental basis for what expected deposition rates would be in these materials or holdup in the condenser based on the conditions that are there? Or is it just analytically developed based on analysis?

CHAIR PETTI: This is Dave. My view is there's tons of data. Billion dollars was spent in severe accident research --

MEMBER BROWN: Okay. I didn't hear that. I'm sorry.

CHAIR PETTI: -- looking at deposition in different systems, the effect of water as a way to wash out aerosols.

MEMBER BROWN: Okay. All right. I'm good with that. I just didn't hear anybody refer to that --

CHAIR PETTI: Yeah.

MEMBER BROWN: -- for the modeling calculations. Okay. Thank you very much. This has been enlightening, for me anyway.

CHAIR PETTI: Any other comments from members? If not, we'll move on to the NEI. I see Steve Schultz had a question. Steve?

MR. SCHULTZ: Yes, this is a question for Mike and then also probably for John Parillo as well because it looks at it from two different ways. The discussions we've had today talks about the opportunity to use a risk-informed approach with the first stab to look at this particular set of license amendment requests and help with the decision making associated with it. And then at the end, John and others have talked about making an application of this or expanding the thinking associated with Reg Guide 1.183 modifications, moving from just to consideration in control rod drop to the design-basis event evaluation. And my question is, is this type of evaluation, the diverse approach, risk-informed approach, can we expect to see this in other areas in the revisions that are being proposed and evaluated for the Reg Guide 1.183? Is the staff adopting this type of an approach for that Reg Guide modification evaluation?

MR. FRANOVICH: This is Mike Franovich. I think Shilp actually might be in a better position to address that. But the extent of the proposal here,

there is a symmetry of sorts between the ISG which is focused on the staff's evaluation, particularly when the licensee has results that may be close to a limit and where you start grading your level of effort, versus the Reg Guide revision which is, again, focused on what the licensee might be taking credit for and perhaps more modernized approaches than relying on trying to use the -- what's been referenced already as the BWR Owners Group topical report that's been -- was reviewed now over 20 years ago. And so that sort of modernization is the extent of the -- the risk-informing part, it's the primary part. But maybe Shilp or John can amplify further.

MR. SCHULTZ: Thank you, Mike. Yeah, for Shilp, one of the things that certainly has been evaluated here which wasn't done 20 years ago was the association of the conservatisms that are in other pieces of the evaluation and analysis. And in other words, the focus was on the dose evaluation and that particular evaluation and not conservatisms that play into that from other areas, like seismic, for example.

MR. VASAVADA: Yeah, so this is Shilp. That's correct. In this case, the -- as I mentioned in one of my slides, we just looked at a sliver of the entirety of the dose calculations in the context of

the challenges that we were facing because of uncertainty in a parameter in the licensee's dose calculations. And what Mike mentioned had more granularity to that. Using what we've done in the technical assessment for the ISG, what we are planning to do or we're trying to do is see if we can streamline some of the information that the licensees have to provide to take formal quantitative credit to approaching the Reg Guide for the condenser.

In the previous case that the BWR topical from 20 years ago, the topical is from '93, I believe, or safety evaluation is from '99 that has nine limitations and conditions which the licensees have to meet to take credit using the latest and greatest information that we have in 20-plus years of operating an earthquake experience. We are looking to see how we can streamline those information needs and kind of order the amount of information that the licensee has to offer, maybe on a tiered approach based on how significant their seismic hazard is at a particular site. So those are some of the thoughts how we are trying to use what we developed for the ISG into the Reg Guide development.

MR. SCHULTZ: Thank you very much. Appreciate that.

MR. VASAVADA: So this is Shilp Vasavada again. If Kevin Petti can allow me, I just wanted to go back to one of the items I had a note from Member March-Leuba's comments. If I can just speak to that.

CHAIR PETTI: Sure.

MR. VASAVADA: All right. So Member March-Leuba mentioned something about needed a calculation or would like to see a calculation. I just wanted to, again, put it in the context of the purpose of the ISG. The purpose was not, again, to give or for the staff to introduce which we cannot, quantitative credit in a licensee's calculation.

The point was to see, okay, if we were to think about this realism in our decision making, can we overcome challenges in the uncertainties and some parameters in the licensee's calculations? So it was hard to come up with a factor. And yes, you're right. Things can heat up and there can be revolatilization of aerosols.

But again, the comparison is against a case where the PCS is not even considered. There's a direct release as Steve pointed out, the ground cover release and then it goes to the control room. Compared to that, if one were to realistically think about the PCS, even with revolatilization -- I can't

ever say that -- there would be in reality holdup volumes which would lead to a decay of even those aerosols and a reduced dose compared to the calculation of the licensee.

And this is on record that the licensee has provided which does not consider any of that. So that was the context and that's the reason why we did not see the need to have a factor developed. So I just wanted to go back to that. Thank you for the time.

CHAIR PETTI: Okay. Unless I hear other questions from members, I think we should turn to NEI at this point.

MR. BROADBENT: So this is Greg. Frankie, did you want to say anything to begin with?

MS. PIMENTEL: No, I mean, other than we appreciate the opportunity to provide industry feedback during this discussion of the ISG. Other than that, we can get started with our presentation.

MR. BROADBENT: All right. Let me go to the start. So can everybody see the presentation?

CHAIR PETTI: Yes.

MR. BROADBENT: All right. I'm Greg Broadbent. I work for Entergy. I'm a senior staff engineer in the corporate office. And we have Frankie

on the call as well from NEI.

I've been working for most of my career at Grand Gulf. So I'm familiar with BWRs a lot more than PWRs. And so this is a radiological analysis as well. So this is a topic that I think I'm prepared to discuss. Go to the next slide.

Talking about in general starting with risk-informed regulation, we certainly support the NRC's efforts to risk inform all regulatory approaches. With regard to this, I'll mention -- with regard to the MSIVs, going off script a little bit here, the MSIVs are some of the biggest valves that we have in our plant. The ones at Grand Gulf, I think the steam lines are, like, 28 inches.

And we've got eight MSIVs that need to close very quickly. And they have some of the lowest leakage requirements of all the valves in the plants. So this is -- you need to take that in context that these are very, very low leakage to begin with and very big valves and that there are ALARA issues associated with this.

Going in and having to rework those valve seats does incur a lot of dose. And that's described in the BWR Owners Group report. And I think they may actually have some numbers in there.

It's also an outage length issue. We could -- we'd have to extend the outage because these valves can't be worked online. So we have to be shut down before we can actually do anything with regard to these valves.

We're only talking about small increases in this already small leakage rate allowance that we have in our tech specs. And by risk informing this, it allows us to spend our resources in areas that are most important to safety. So if it's not important that we incur personnel doses, real doses to people by allowing a little higher leakage rate on these valves, then we feel that's a good allowance to -- a good balance to allow us to increase these leakage rates.

And we've also seen some recent successes with GSI-191 resolution with a risk-informed approach. And 50.46 has recently taken some statistical approaches as well. So we feel like this ISG is the first step in risk informing the radiological analyses.

We feel that there may be other areas. And we'd certainly like to discuss that with the staff. But we're really just talking about the ISG here.

I'll quote Shilp from just about 30

minutes ago where he talked about this ISG is just a small sliver of realism in a universe of conservatism. And I think we would agree with that certainly. And the industry supports including these approaches in Revision 1 of the Reg Guide, and we know that the staff intends to do that.

We also recognize that the power conversion system is likely to remain intact post-accident. It is built to very high standards. And we do have confidence that the SSCs in the power conversion system will provide sufficient volume for holdup and retention of the fission products like the ISG concluded and also recognize that the power conversion system is an important system for plant operation, not just post-accident.

But that is really where we make our money. And we have to keep that system operational and generally leak free in order to get steam to the turbine and turn the generator so that we can stay in operation. Within Entergy, we've had some plants that weren't able to stay in operation because they weren't able to make the financial numbers we needed.

So it is a system that we keep in operation. And we can -- in the event that there are issues with it or any sort of failures, they're

clearly evident in a BWR plant. Steam plumes, high radiation in certain areas would alert us to any leaks, maybe some drain flows, also even things like trouble with people exiting the radiologically controlled area due to contamination of some noble gas isotopes that stick to people's clothes.

So as we mentioned, approximately half the plants have already demonstrated that the structural integrity would be maintained using the BWR Owners Group report that's currently out there. And we feel that this ISG also validates the current regulatory credit that we take for the power conversion system and other accident analyses like the control rod drop. I think that was previously discussed. Going to the next slide.

And we feel that the ISG does incorporate a good amount of operating experience, certainly the post-Fukushima seismic risk insights. The Owners Group report done back in the 1990s used earthquake experience data. Obviously, there's more now than there used to be. So that's incorporated in here.

And we do like the fact that the NRC is including explicit credit for at least these conclusions in the upcoming Reg Guide. And we also recognize that the LARs that were previously approved

really didn't use Rev. 1. They used Rev. 0 of the Reg Guide. So we've also like to have credit for plants that are applying Rev. 0 to be able to use the conclusions in this ISG.

And with regard to steam line deposition credit, there is some parts of the steam line that are credited in the analysis, even if we're not crediting the power conversion system, for example, the volumes between the MSIVs where they're isolated. And we want to make sure that there's realistic modeling associated with a deposition in the steam line. And that's important to us.

That's a release pathway that's directly coupled to the reactor vessel currently modeled to be released directly to the environment. So it can be a very significant release pathway and a contributor to doses due to some of the conservative assumptions that are in that analysis. And I think it was previously discussed from the staff's side that there were -- that there's been a lot of history associated with this.

There was an Owners Group report back in the 1990s. AEB-9803 came out for the Perry Plant. And then there's been some concerns about using AEB-9803 and the applicability to other plants. So from

our perspective, there's been a lot of significant regulatory uncertainty regarding what models are acceptable for steam line deposition.

And as the staff pointed out that there is no specific discussion in Revision 0 of the Reg Guide. And we do like the fact and want to work with the staff to provide an approved model for deposition in Revision 1. And I'm sure that it's flexible enough to apply to all BWRs and even the advanced plants. And I think that was it for our presentation, yes.

MEMBER HALNON: Greg, this is Greg Halnon. Just a quick question. Since we're talking about higher depositions in the secondary side of the plan, is there any impact on control room dose or any other impact with time operator studies as you have to respond to other things in the turbine buildings and areas that the doses may be higher?

MR. BROADBENT: Well, and the answer to that is there's really nothing that we respond to in the turbine building. The turbine building is not a safety-related building. So we can't put anything important in there. So the fact that maybe with these increase the leakage allowances that the doses may increase a little bit on that side of the building. If it's not an area where we have an safety-related

equipment, if it's not an area where the operators have to transit through to get to some safety-related equipment or some action they need to take post-accident, then it's really not an issue for us.

MEMBER HALNON: How about actually changing a shift of operators? I know that some of the plants, you don't go through their turbine buildings or near. But I don't know all the designs.

MR. BROADBENT: Yeah, I'm not changing a shift early, or --

MEMBER HALNON: Yeah, this could go on for a while, though. So anyway, I guess you answered the question. There's no required actions that could be affected by dose.

I'm assuming that the control room doses have been checked and no issue on that. So everything else would be a site-specific issue that might have people transit through an area of higher dose. But I think you answered my question. That's fine.

MR. BROADBENT: Yeah, that's right. And any impact on control room dose, if there's any sort of shine or anything from the turbine would be included in a control room dose if that was really a significant pathway for dose to the operators.

MEMBER HALNON: Okay. Thank you.

MR. BROADBENT: Any more questions for me?

MEMBER MARCH-LEUBA: Yes, this is Jose. I just wanted to make a comment that you triggered my thought process. I found argument on ALARA very convincing. It really makes sense. Unless I'm mistaken, during risk-informed evaluations, we don't consider that.

I'm thinking -- I'm just putting it out on the record -- that maybe we're using the wrong cost functions, thinking like a mathematician, that the reason that we need to minimize is not only the risk to the public but also the risk of the facility workers. It's a combination of the two with proper waiting. Just put it out there that you make a good argument that why am I going to give 10 drams for a couple of operators for something that doesn't really produce a significant benefit.

MR. BROADBENT: And that argument was the Owners Group report back in the 1990s came up with. And that's right. Why take dose that is specifically from -- or 100 percent certain we're going to get because we're going to go out and work those valves and compared to an accident dose that we would receive or the operators would receive and maybe some offsite would receive for a very, very low probability

accident. So that was the Owners Group approach.

CHAIR PETTI: And frankly, Jose, it's one of the reasons I thought we should hear this in subcommittee. This is a very subtle but interesting result. And I just thought that all the members, as we think about risk-informed regulation, would like to understand this tradeoff that we've heard today.

(Simultaneous speaking.)

MEMBER REMPE: -- we've got this other situation where we have some design developers who do not want to consider ALARA in Part 53, but just throwing it out there, right?

CHAIR PETTI: But it's required in Part 20. And they have to live to Part 20.

MEMBER REMPE: Yeah, I know. It's just something to think about where it goes, but anyway.

CHAIR PETTI: Okay.

MEMBER MARCH-LEUBA: And then with respect to -- to comment about control room dose, I missed the plans where you have to run by the feedwater pump to get into the control room. And the path to control room goes by the turbine and all the secondary. So I assume those plants would consider increased leakage to -- we did consider the dose to the control room operators.

But if you have to get there, you have to walk by a contaminated area. It should incorporate analysis. That was just a rhetorical comment.

CHAIR PETTI: Any other comments from members?

Okay. So at this point, Scott and Thomas are going to try to fix the public line. So Scott, before we take public comment, he's going to --

(Simultaneous speaking.)

MR. MOORE: Yes, thank you, Chairman. So for all the members on the phone, we're going to reset the phone line now. And so you're going to drop off as we reset, and you need to call back in immediately.

And this will take just a couple minutes, and then we'll go to public comments. Thanks, everyone. Thomas, you can reset the public line at this point. Thank you.

MR. SNODDERLY: Excuse me, Chairman Petti. This is Mike Snodderly. While we have some time, I just wanted to make one comment.

CHAIR PETTI: Go ahead.

MR. SNODDERLY: So I just wanted to remind the members and the staff that as was mentioned in the slides, the staff has received public comments on the ISG. And the staff is currently -- or has resolved

those comments. But we did not hear about them today because of the very full agenda and making sure that everyone clearly understood the precedent that had been set through the LARs. But the staff does plan to address the resolution of those public comments when we have the full committee meeting right now currently scheduled for November. So I just wanted to remind the staff and the members and also get on the record that we will go over the final resolution of the public comments received in November.

CHAIR PETTI: Thank you, Mike.

MR. MOORE: Thomas or Makeeka, are either of you on?

(Pause.)

MR. DASHIELL: Public bridgeline has been reestablished.

MR. MOORE: Thomas, you said it has been reestablished?

MR. DASHIELL: That is correct, Scott.

MR. MOORE: Okay. Thank you.

CHAIR PETTI: And I assume all those beeps are people calling back in, right?

MR. MOORE: Yes, sir.

CHAIR PETTI: Okay. Let's give them another minute, and then we'll ask for if there's any

public comments.

(Pause.)

CHAIR PETTI: Okay. I'm not hearing any more beeps. So hopefully, everybody is back on. Are there any comments from members of the public? If so, please state your name and your comment.

Again, any comments from the public?

Okay. Not hearing any, we're going to turn to the staff -- I mean to the members. Any other comments on presentations today and the topic at hand?

MEMBER KIRCHNER: Dave, this is Walt. I have one general comment going back to the questions I asked of the staff. When this is presented at full committee, I really strongly feel that more context is needed up front because on the surface, it could be misinterpreted by the public that we're using these risk-informed measures to relax requirements. And that's not really the case.

But it sounds like it on the surface because we're talking about increasing leakage of valves. So that sounds like a problem. And as the NEI presentation made clear, we're not necessarily relaxing protection of the public. We're actually doing a more physically accurate representation of what happens and demonstrating that the public is

probably more protected when you analyze the problem in that manner.

And then obviously, LARs cost money. So the industry does an LAR often because there's an economic gain that offsets the cost of the LAR in terms of the operation of the plant and the safety of the plant. So I just think more context would be useful up front in an abbreviated presentation to the full committee, especially if the NEI doesn't participate in that presentation. Thank you.

CHAIR PETTI: That's a good point, Walt. I'm sure NRR is listening. Thanks.

MEMBER BALLINGER: This is Ron. We often -- we always make a distinction between members of the public and workers at the plant. But as soon as the worker goes home for supper, he or she becomes a member of the public. And so radiation dose is radiation dose. And it's useful to save it, period.

CHAIR PETTI: Anyone else?

Okay. Well, I want to thank the staff and NEI. Very informative presentations today. Members, of course I have a draft letter. I will place it on your NRC emails. And you can hack it at will and add any comments that you feel need to be there.

We've got a little bit of time since full

committee won't be now till November. If I recall, this was going to be an October thing, but things are moving around. But I'll at least get it out there for you to think about it before you forget your thoughts before the November full committee.

And with that, we'll adjourn the meeting. And I guess we come back at 2:00 o'clock Eastern for the afternoon session. Thank you, all.

(Whereupon, the above-entitled matter went off the record at 12:34 p.m.)

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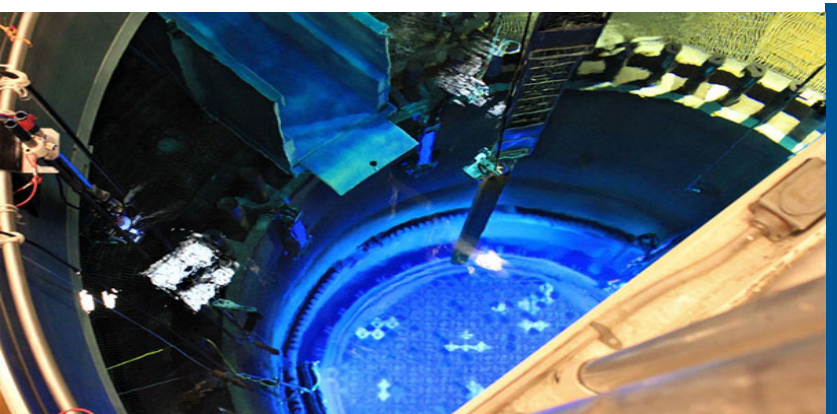
Presentation to ACRS Subcommittee on Draft Interim Staff Guidance for Radiological Consequence Analyses Using Alternate Source Terms

Mike Franovich, Director, NRR/DRA
Kevin Hsueh, Branch Chief, NRR/DRA
Jerry Dozier, Senior Reliability and Risk Analyst, NRR/DRA
Steve Jones, Senior Safety and Plant Systems Engineer, NRR/DSS
Shilp Vasavada, Branch Chief (Acting), NRR/DRA
John Parillo, Senior Reactor Engineer, NRR/DRA



July 23, 2021

Introductory Remarks

A changing regulatory environment:

- Licensing, other regulatory decisions, and backfit/forward fit actions must be risk-informed. (SRM-SECY-19-0036, SRM-SECY-18-0049)
- Improved realism, evaluation techniques, and additional information are applied to improve risk-informed regulatory decision making (ML19319C832)
- Culture re-alignment is needed to ensure that we identify and resolve challenges and roadblocks for the appropriate and consistent integration of risk insights.

Introductory Remarks (Cont'd)

- Integrated Review Team process (LIC-206) was used in the staff's approval for BWR LARs to allow for increased MSIV leakage.
  - All four MSIV reviews were completed using a team approach.
  - Each SE includes a section on risk and engineering insights to support staff's reasonable assurance finding.
- ISG is being developed to memorialize staff's practice.

Introductory Remarks (Cont'd)

- Draft RG 1.183 Rev. 1 (DG-1389)
  - Staff efforts have restarted using an integrated team to revise RG 1.183, "Alternative Radiological Source Terms for Evaluating DBAs at Nuclear Power Reactors."
  - An ACRS meeting is being planned for Fall 2021, prior to issuance of DG-1389 for public comment.

Outline

- Overview of the Interim Staff Guidance (ISG)
- Background of ISG
- Basis for ISG – Technical Assessment
  - Overview
  - Details
- Difference between ISG and Regulatory Guide 1.183 Revision
- Use of ISG approach in LARs
- Takeaways

Overview of ISG - Timeline

- Published in *Federal Register* for public comment
  - 30-day comment period closed on June 21, 2021
- 13 comments received from NEI, 20 anonymous comments
- ACRS full committee briefing scheduled for November 2021(tentative)
- OMB approval - January 2022(tentative)
- Final FRN - February 2022(tentative)

Overview of ISG (Cont'd)

- The ISG is expected to be transitioned into SRP Section 15.0.1 (Radiological Consequence Analyses using AST) in conjunction with the separate RG 1.183 revision effort.
- Section 15.0.1 will include a reference to the revised RG 1.183.
- ISG will be closed after transition to Section15.0.1.

Overview of ISG – Primary Insight

- **High probability** that doses will be **lower** than those estimated strictly using traditional deterministic methods, which **include accepted assumptions**, that **do not credit** hold-up and retention of the Main Steam Isolation Valve (MSIV) leakage within the power conversion system (PCS)

Overview of ISG – Objective and Expectation

- **Objective:** Near-term formal regulatory footprint for staff's use of risk insight
- **Expectations:**
  - Used by staff to offset uncertainty in input parameter(s) for deterministic calculations
  - Supports staff's reasonable assurance finding during reviews
  - Transitioned to Standard Review Plan Section 15.0.1
- **Caveat:**
  - **Does not change the licensee's responsibility** to demonstrate compliance with 10 CFR 50.67
  - **Does not change acceptable methods** for demonstrating compliance with 10 CFR 50.67

Background of ISG - Genesis

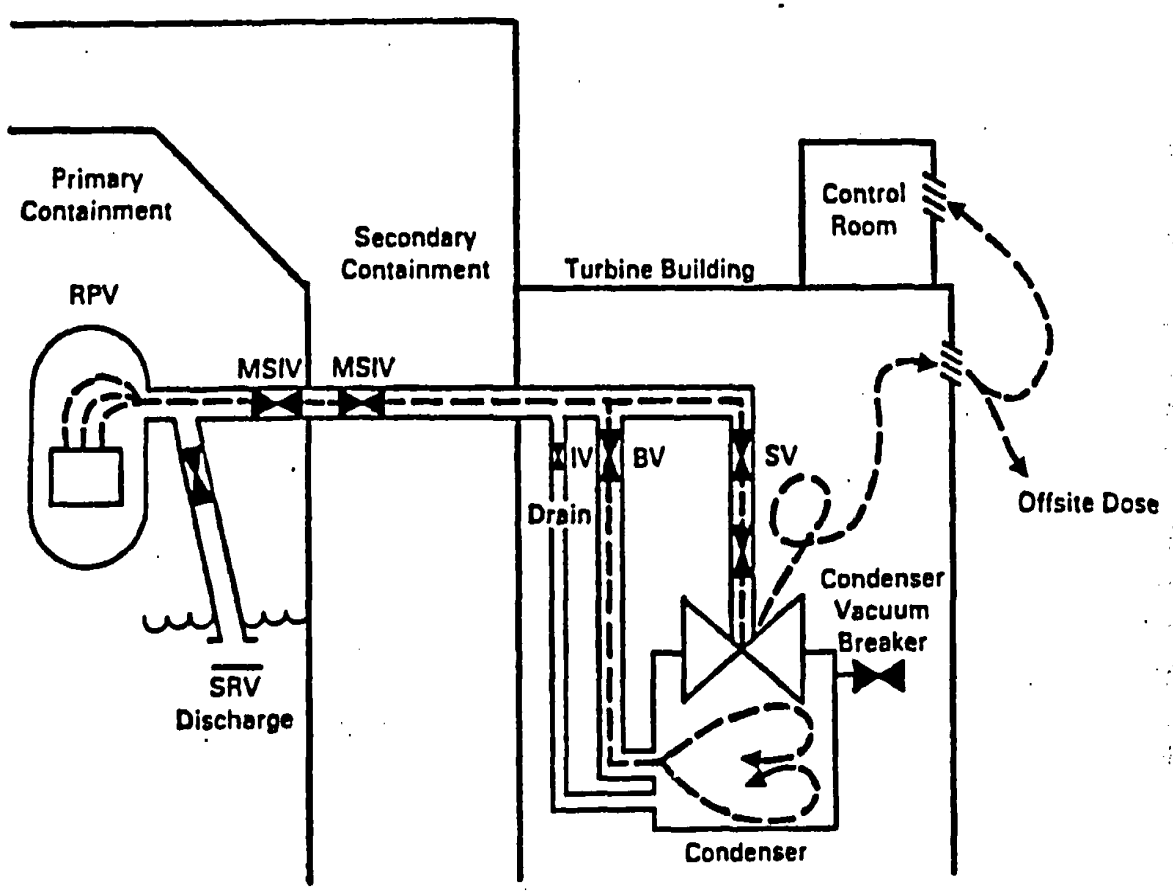
- Commission direction to become a modern, risk-informed regulator (e.g., SRM-SECY-19-0036; ML19183A408)
- Four license amendments were submitted to allow for increased MSIV leakage in 2019
  - Challenges due to uncertainty in input parameter values in dose calculations
  - LIC-206 (ML19031C861) invoked for multi-disciplinary risk insights

Background of ISG – Genesis (Cont'd)

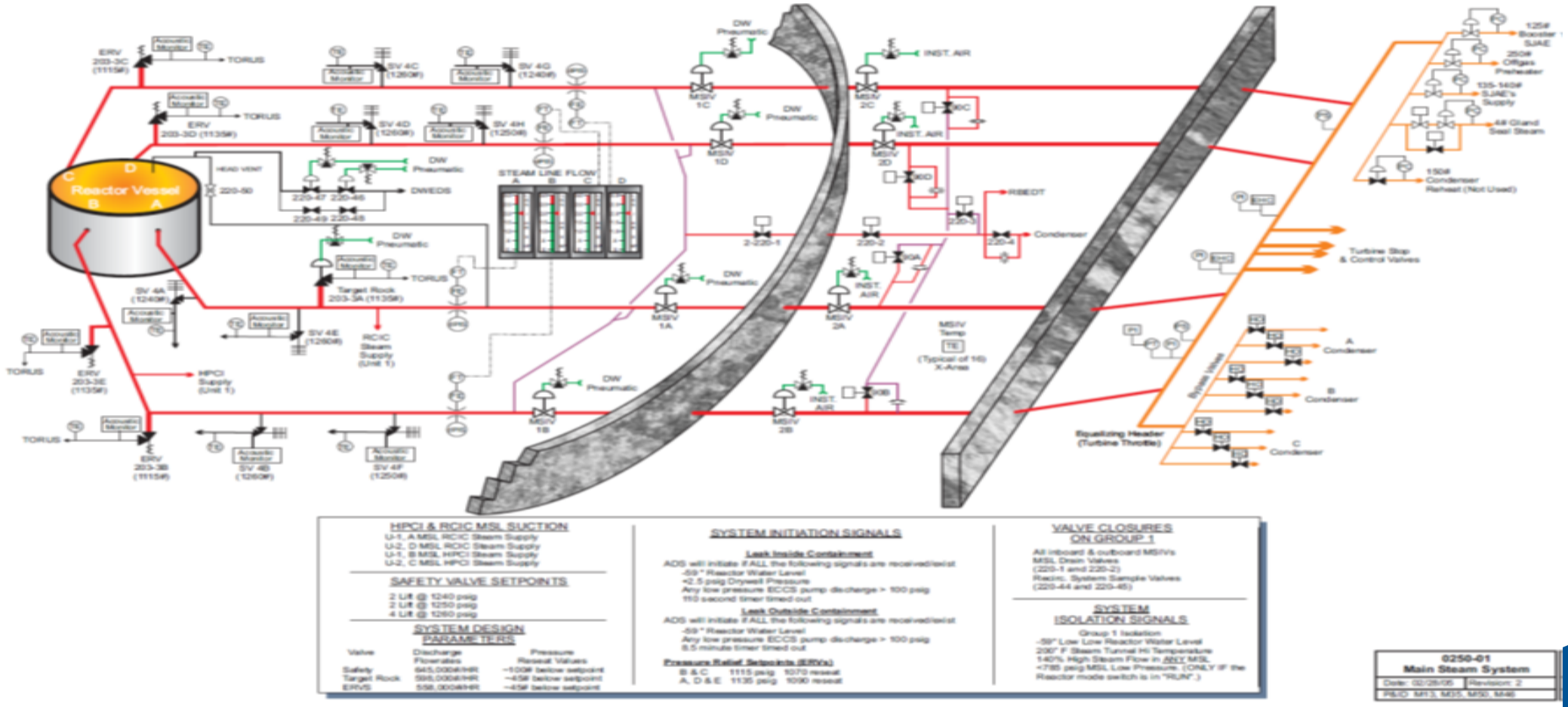
- Integrated review team approach following LIC-206 guidance
- Identified that risk insights support consideration of holdup in PCS
  - Ability to offset challenges without changing calculation methods and assumptions
- Documented insights in technical assessment
  - Internal reviews and deliberations
- Implementation of LIC-206 in deterministic LARs
  - Included in all four safety evaluations for the LARs to allow for increased MSIV leakage (ML20140A070; ML20150A328; ML20241A190; ML20265A240)

Basis for ISG – Technical Assessment

- Dose calculations often do not credit any SSCs beyond outboard MSIVs
- "Formal" credit for condenser through safety evaluation on BWROG Topical Report – approximately half of BWRs have adopted this method.
- Large holdup volume exists in PCS beyond second MSIV



Large Holdup Volume in PCS



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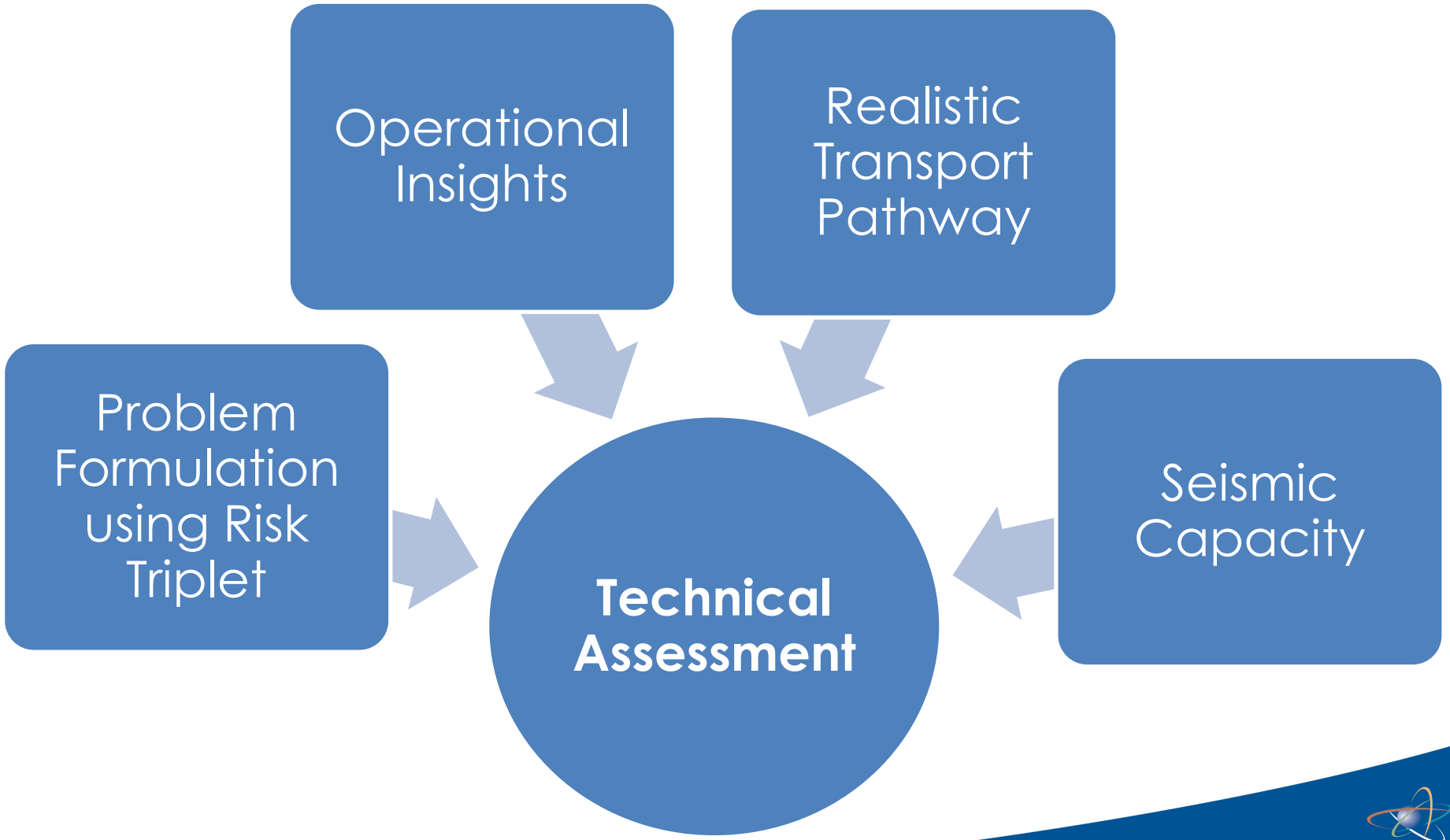
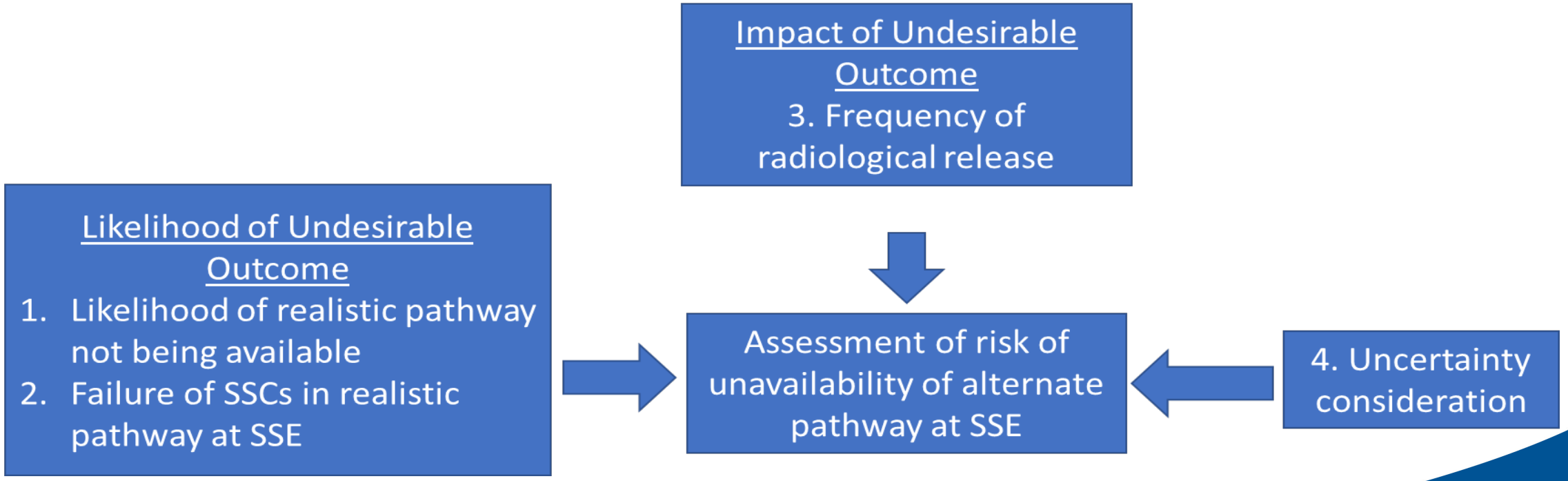
Technical Assessment - Overview



Technical Assessment – Risk Triplet Formulation

Risk = What Can Go Wrong x How Likely Is It x What are the Consequences
= (Likelihood x Impact) of Undesirable Outcome
= (Likelihood x Impact) of Fission Products Not Retained in Power Conversion System



Operational Insights

- Main Steam System Piping:
  - Large internal volume
  - Typically designed to B31.1.0, "Power Piping"
  - Constructed with augmented quality
  - BWR 5 and BWR 6 designed to B&PV Code – safety-related
- Main Steam Isolation Valves:
  - Typically, large globe valves that seat with pressure
  - Stem leakage from outboard valve considered a small fraction of measured seat leakage
- Passive features provide hold-up volume for MSIV seat leakage

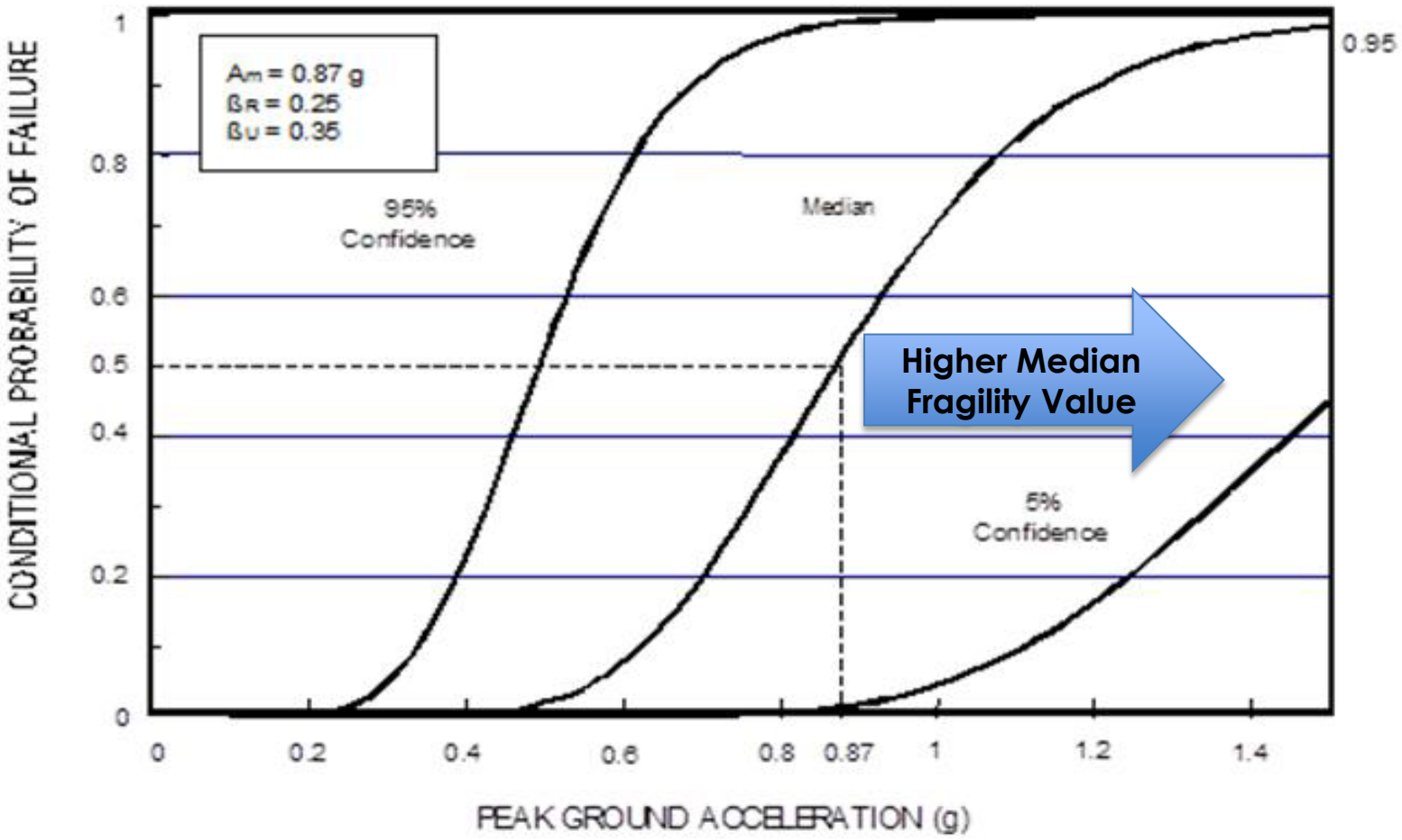
Realistic Transport Pathway

- Consideration of piping attached to steam lines
  - No alignment of specific leakage path
  - Reliability of complete isolation; larger valves leak more
- Functional drain lines flow to main condenser
- Turbine bypass valves also flow to main condenser
- Other leakage, primarily through stop and governor valves to high pressure turbine, provide for less holdup and deposition than main condenser

Seismic Capacity: A Primer

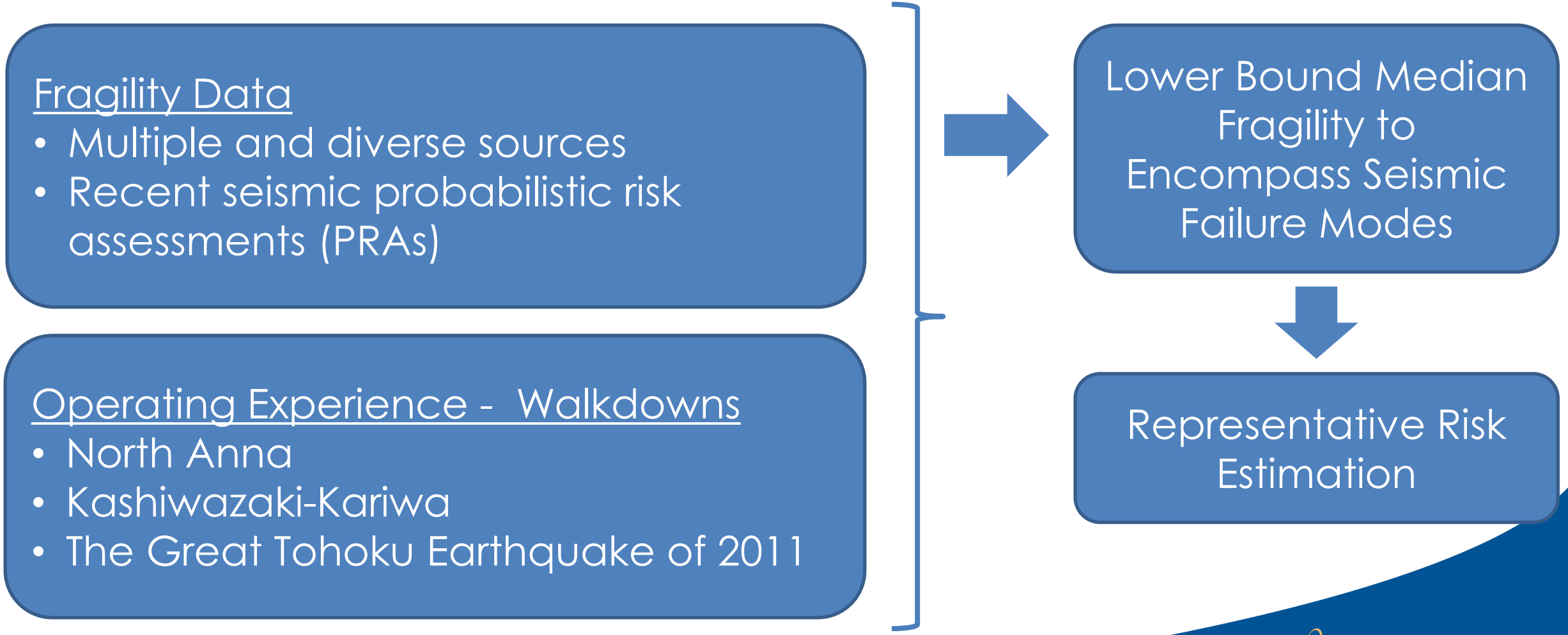
- **Fragility**: Conditional failure probability as a function of seismic acceleration; Analytically determined; Lognormally distributed
- **Median fragility (Am)**: Seismic acceleration at which there is 50% probability of failure
- **Lognormal uncertainty parameters** (β_r for randomness; β_u for uncertainty): Parameters characterizing the uncertainty in the fragility
- **Seismic acceleration**: Measure of strength of earthquake in terms of multiples of gravitational acceleration (e.g., 0.1g, 1g)
- **Peak ground acceleration**: Commonly used acceleration level for seismic analysis; corresponds to acceleration of 100 Hz oscillator

Seismic Capacity: A Primer



Source: Electric Power Research Institute Report 1025287 (also known as SPID; ML123330282)

Approach for Seismic Capacity Evaluation in Assessment



Seismic Capacity Insights

- Welded piping, bolted piping, and valves have high median fragilities
- Main condenser is usually a seismic Category II structure
  - Anchorage designed to avoid failure at design-basis seismic loads
- Post-earthquake walkdowns of plants demonstrate high seismic capacity of SSCs in PCS
- Seismic risk from accelerations at and below plant's safe shutdown earthquake (SSE) is small

Seismic Capacity Insights

- Lower bound median fragility parameters
  - Am = 0.4g; β_r = 0.22; β_U = 0.22
  - Based on fragility of expansion joint connecting circulating water piping to condenser
  - Encompasses failure modes of relevant SSCs
  - Supports low likelihood of gross failure of SSCs in PCS

Representative Risk Estimation

- Convolution of range of hazards with lower bound median fragility parameters
  - Provides estimate of risk of gross failure of SSCs in PCS
  - Uses latest seismic hazard curves
- Estimates demonstrate low risk of gross failure
  - Even lower if contribution only until SSE is considered

Uncertainty Consideration for Seismic Capacity Evaluation

- Uncertainty in median fragility is explicitly included
- Conservatisms exist that address uncertainty in selected median fragility
  - Use of lower bound median fragility
  - Low pressure conditions for high pressure piping
  - Consideration of SSE concurrent with the accident postulated for dose calculations
  - Conservatisms in remainder of dose calculation guidance are unchanged

Holistic Context for ISG

- Requirements in 10 CFR 50.67
  - Initial condition assumes failure of multiple redundant engineered protection systems and core damage
- Single-failure of inboard MSIV
- Conservatism in analysis
  - Use of acceptable assumptions and parameters
- Concurrent SSE
- Lower bound median fragility parameters

Difference between ISG and RG 1.183 Revision

ISG	RG 1.183 Revision
Directed at NRC staff	Directed at licensees
Provides additional support for the staff's reasonable assurance determination	Provides guidance for quantitative credit for holdup in main condenser
Does not change the licensing basis dose calculation	**Quantitative credit changes the licensing basis dose calculation**
Information needs from licensees are not required	Identifies docketed information needs for quantitative credit

Licensee's responsibility to demonstrate compliance unchanged
Acceptable methods for demonstrating compliance unchanged

§ 50.67 Accident source term.

- "b) *Requirements.* (1) A licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under § 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents previously analyzed in the safety analysis report."

- "(2) The NRC may issue the amendment only if the applicant's analysis demonstrates with **reasonable assurance** that……."

RG 1.183 Rev 0 MSL Pathway Challenges

- RG 1.183 Rev 0 does not contain an aerosol deposition model for the main steam lines (MSL).
- Instead, many licensees have utilized the leakage pathway model described in the staff assessment entitled, *Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term,* (AEB-98-03).
- Following multiple BWR license amendment requests to revise their source term to implement an AST under 50.67, the NRC staff published, Regulatory Issues Summary 2006-04, *NRC Regulatory Issue Summary 2006-04, Experience with Implementation of Alternate Source Terms*.

RG 1.183 Rev 0 MSL Pathway Challenges (Cont'd)

- The purpose of RIS 2006-04 was to discuss the more frequent and significant issues encountered by the NRC staff during its review of AST submittals and to provide information for licensees to consider when developing submittals for implementation of an AST.
- A frequent point of contention between licensees and the NRC staff was the deposition of gaseous iodine in the main steam lines.

RG 1.183 Rev 0 MSL Pathway Challenges (Cont'd)

- For calculation of aerosol settling velocity in the main steam line piping of boiling water reactors, the staff reaffirmed the modeling approach in AEB 98-03 [*emphasis added*] but emphasized the report was written based on the parameters of a particular plant and, therefore, the removal rate constant is specific to that plant.
- Any licensee who chooses to reference these AEB 98-03 assumptions would need to provide appropriate justification that the assumptions are applicable to their particular design.
- AEB-98-03 has been utilized for more than 20 years when implementing 10 CFR 50.67.

MSIV Leakage LARs Submitted Using RG 1.183 Rev. 0

- LARs included drywell spray removal and main steam line (MSL) deposition in their MSIV leakage models consistent with RIS 2006-04 and past precedent.
  - Staff questioned how the drywell sprays would impact subsequent MSL aerosol deposition.
  - Licensees provided a sensitivity analysis examining the impact of several parameters.
  - Of the parameters evaluated, a pathway to the condenser provided a substantial dose reduction.

MSIV Leakage LARs Submitted Using RG 1.183 Rev. 0 (Cont'd)

- The licensees' sensitivity analyses are not part of their licensing basis.
- A pathway to the condenser was not credited in the analyses of record (AOR).
- The licensee provided an analysis (the AOR), which met the acceptance criteria in 10 CFR 50.67.
- The staff's determination of reasonable assurance was supported by the recognition that there is a high probability that doses will be significantly lower than those estimated by the licensee using deterministic methods (the AOR) that do not credit holdup and retention of the MSIV leakage within the PCS.

Takeaways

- ISG will result in consideration of large holdup volume in future MSIV leakage LARs
  - Offsets uncertainty in input parameter(s) for deterministic calculations
  - Supports reasonable assurance finding during reviews
  - Applicable if quantitative credit is not included in licensee's calculations
- ISG does not change licensee's responsibility to demonstrate compliance with 10 CFR 50.67
  - Acceptable methods for demonstrating compliance remain unchanged
- ISG is expected to be transitioned to SRP Section 15.0.1
- Formal condenser holdup credit for licensees is being considered in revision to RG 1.183

Backup Slides

Risk Insight References

- U.S. Nuclear Regulatory Commission, NUREG/CR-7110, Volume 1, Peach Bottom SOARCA, May 2013, ML13150A053
- U.S. Nuclear Regulatory Commission, PRAB-02-01, "Assessment of BWR Main Steam Line Release Consequences," October 2002, ML062920249
- General BWR plant design regarding Defense-in-Depth to deter the release of iodine

Iodine Pathway in SOARCA

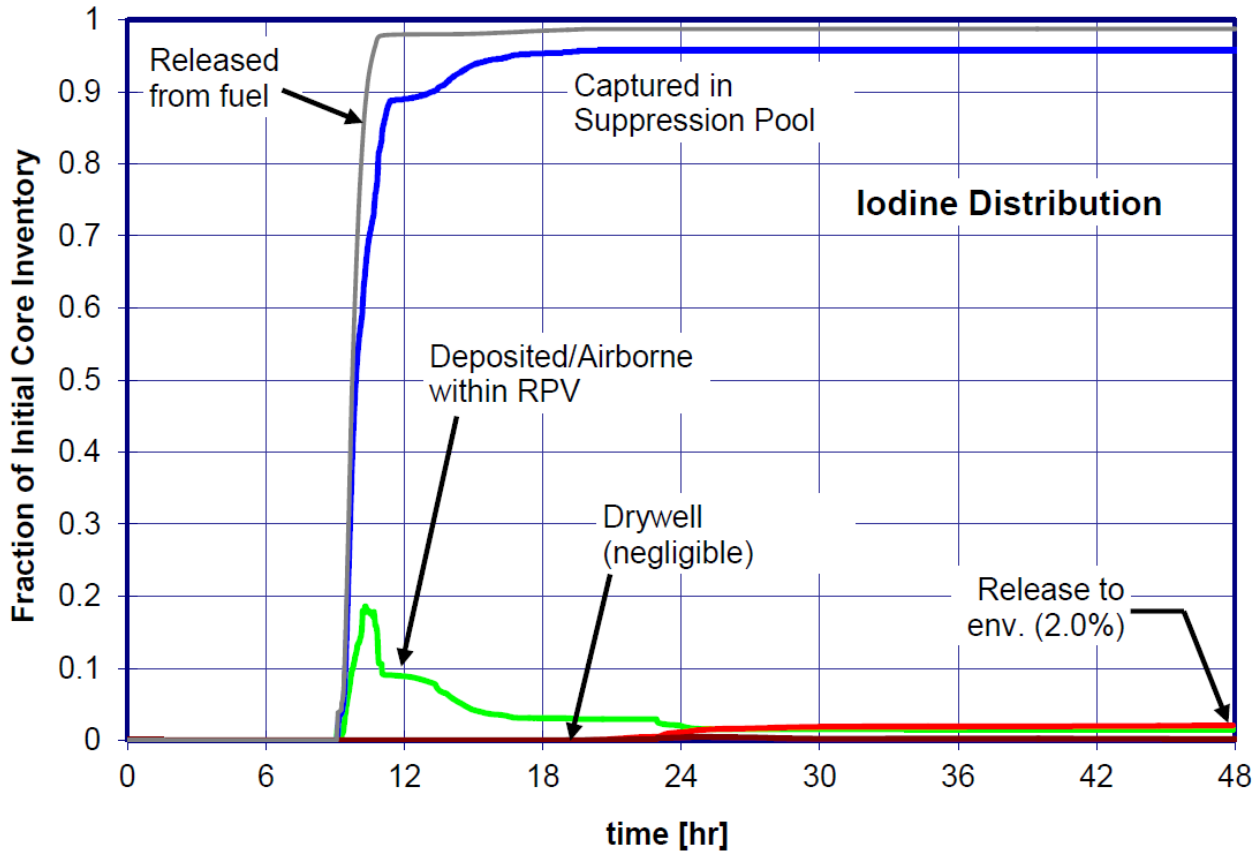


Figure 5-12 LTSBO iodine fission product distribution

Cesium Pathway in SOARCA

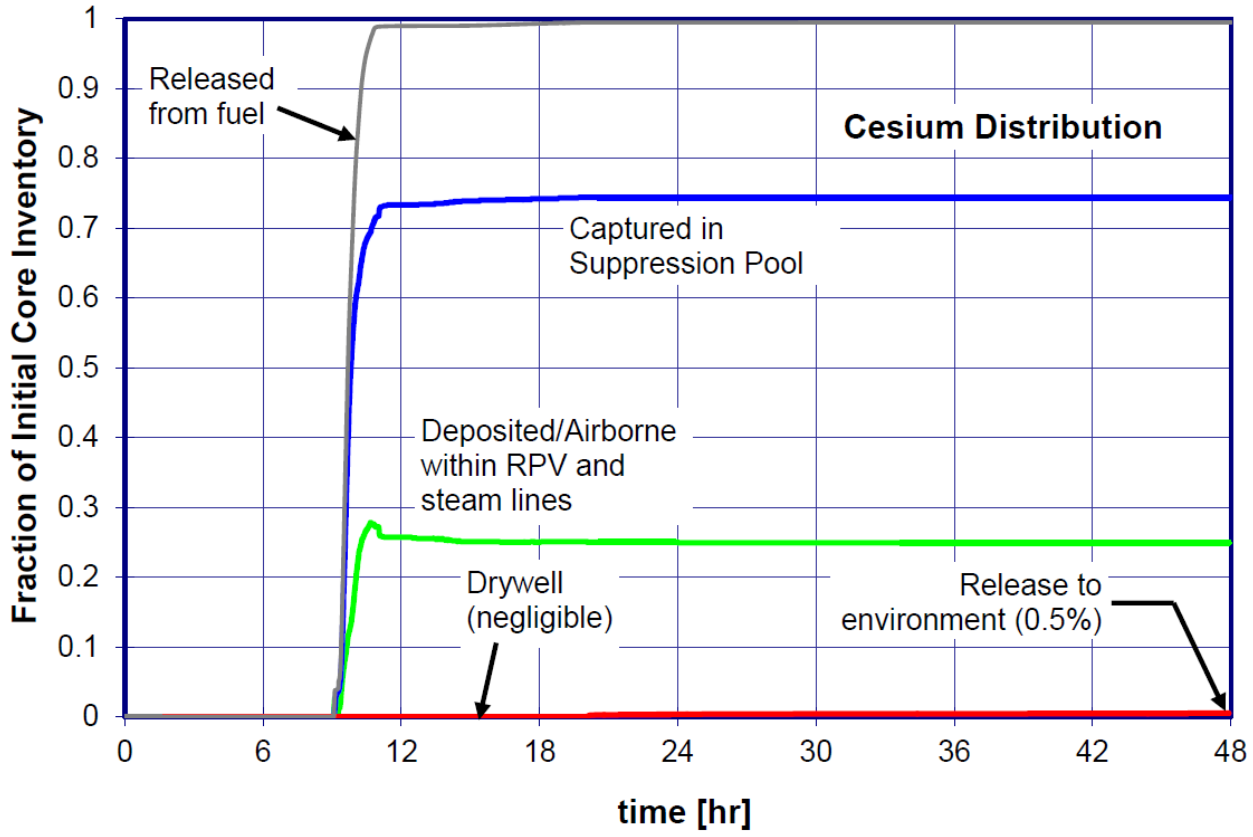


Figure 5-13 LTSBO cesium fission product distribution

RG 1.183 Source Term (Rev. 0)

Table 1
BWR Core Inventory Fraction Released Into Containment

Group	Gap Release Phase	Early In-vessel Phase	Total
Noble Gases	0.05	0.95	1.0
Halogens	0.05	0.25	0.3
Alkali Metals	0.05	0.20	0.25
Tellurium Metals	0.00	0.05	0.05
Ba, Sr	0.00	0.02	0.02
Noble Metals	0.00	0.0025	0.0025
Cerium Group	0.00	0.0005	0.0005
Lanthanides	0.00	0.0002	0.0002

Increased Containment Leakage has Small Impact (SOARCA)

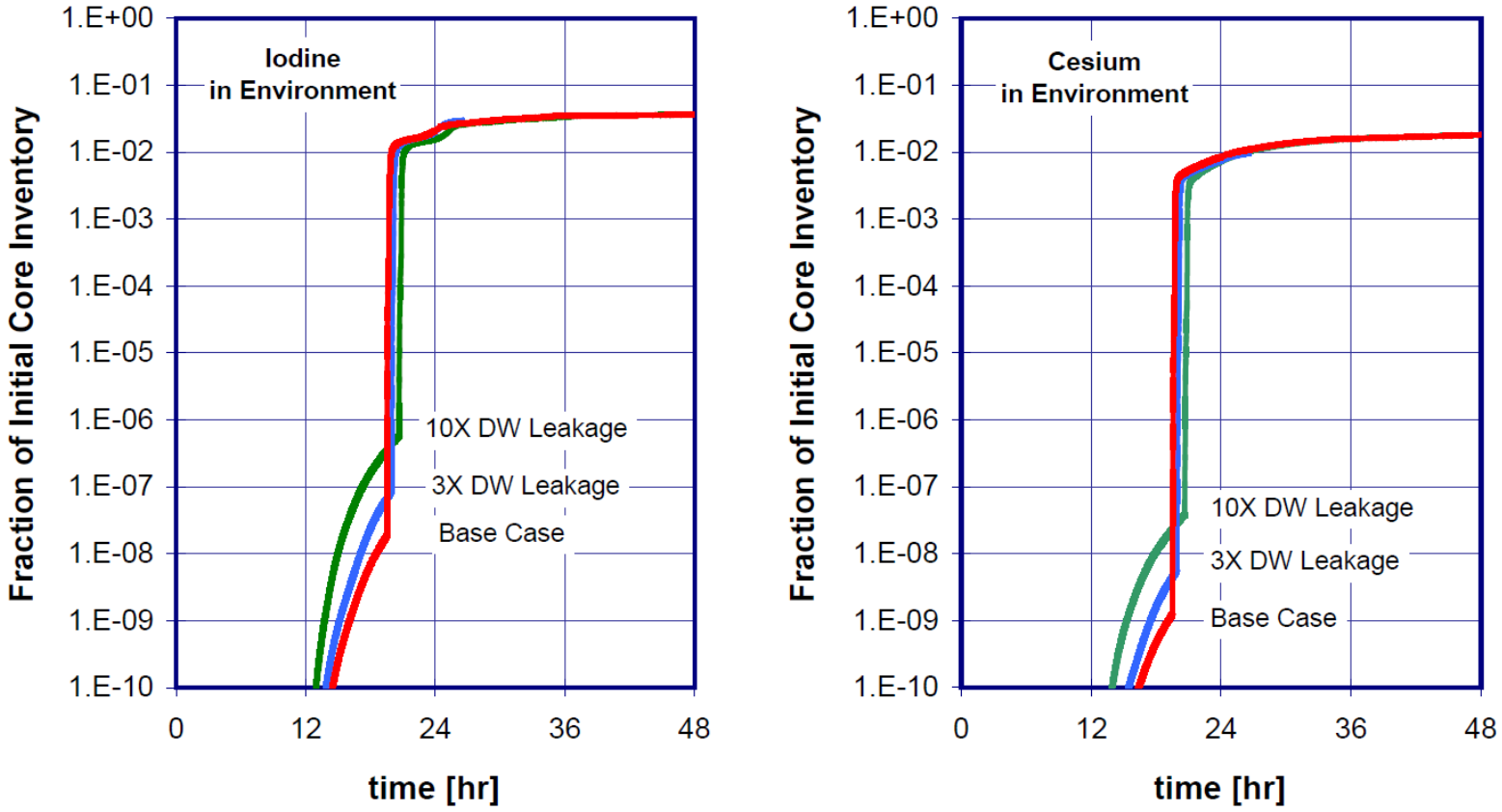


Figure 5-61 Effect of increased containment leakage on the release of iodine to the environment

Realistic SOARCA releases much less than earlier studies

1982 Siting Study SST1 case calculated an iodine release of 45 percent and a cesium release of 67 percent of the core inventory.

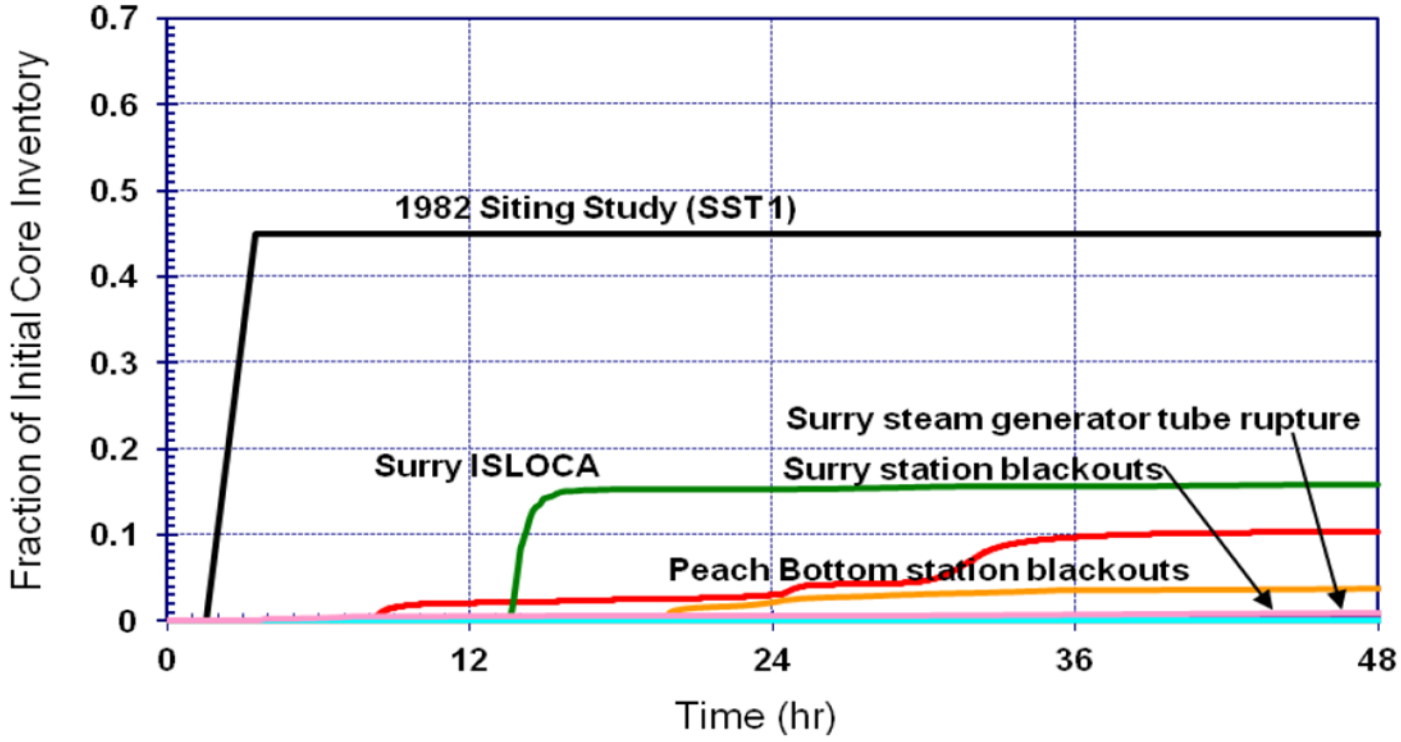


Figure ES-1 Iodine release to the environment for SOARCA unmitigated scenarios and the 1982 Siting Study SST1 case

ACRS Meeting on DRA-ISG-2021-XX

Greg Broadbent, Entergy

Frankie Pimentel, Sr. Project Manager - Engineering and Risk

July 23, 2021

Risk-Informed Regulation

- Industry supports NRC efforts to risk-inform regulatory approaches
  - Risk-informing allows licensees to spend resources in areas that are most important to safety
  - Recent successes include GSI-191 resolution and 50.46 using risk-informed or statistical approaches
- This ISG is the first step to risk-informing deterministic radiological analyses
- Industry supports including these approaches in Reg Guide 1.183, Rev. 1

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ISG Conclusions

- Power Conversion Systems (PCS) highly likely to remain intact post-accident
  - PCS is built to high standards
  - High confidence in the SSCs in the PCS to provide sufficient volume for holdup and retention of fission products
  - Important system for plant operation

- Approximately half of plants have already demonstrated structural integrity with rigorous analyses
  - Validates current regulatory PCS credit for other accident analyses
    - Control Rod Drop Accident

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ISG Conclusions

- ISG incorporates relevant operating experience as well as recent post-Fukushima seismic risk insights and walkdowns
  - NEDC-31858P used earthquake experience data, primarily from nonnuclear facilities
  - Include explicit credit for the conclusions of this ISG approach in Reg Guide 1.183, Rev. 1
    - Allow credit for plants applying Rev. 0

Steamline Deposition Credit

- Realistic modeling of the deposition in the steamline is important for BWRs
  - Directly coupled to the reactor vessel
  - Calculated dose for MSL is significant due to very conservative assumptions

- Significant regulatory uncertainty regarding acceptable models in RG 1.183 Revision 0
  - Revision 0 of RG 1.183 does not contain an aerosol deposition model for the main steam lines
  - Revision 1 should provide an approved model of deposition that is flexible enough to apply to all BWRs including advanced plants

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Dashiell, Thomas	Joined before	7/23/2021, 9:01:24 AM
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Skov, Tammy	Joined	7/23/2021, 9:03:04 AM
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Franovich, Mike	Joined	7/23/2021, 9:05:49 AM
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