UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

September 13, 2012

MEMORANDUM TO: ACRS Members

FROM: John Lai, Senior Staff Engineer /RA/
Technical Support Branch
Advisory Committee on Reactor Safeguards

SUBJECT: CERTIFICATION OF THE MINUTES OF THE MEETING OF THE SUBCOMMITTEE OF RELIABILITY AND PRA ON RISK-INFORMED REGULATORY FRAMEWORK FOR NEW REACTORS ON MARCH 7, 2012, IN ROCKVILLE, MARYLAND

The minutes for the subject meeting were certified on September 6, 2012. Along with the transcripts and presentation material, this is the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachments: As stated

cc w/o Attachments: E. Hackett
C. Santos

cc w/ Attachment: ACRS Members

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

MEMORANDUM TO: John Lai, Senior Staff Engineer
Technical Support Branch
Advisory Committee on Reactor Safeguards

FROM: John W. Stetkar, Chairman /RA/
Subcommittee on Reliability and PRA

SUBJECT: CERTIFICATION OF THE MINUTES OF RELIABILITY AND PRA SUBCOMMITTEE ON RISK-INFORMED REGULATORY FRAMEWORK FOR NEW REACTORS ON MARCH 7, 2012, IN ROCKVILLE, MARYLAND

I hereby certify, to the best of my knowledge and belief, that the minutes of the subject meeting on March 7, 2012, are an accurate record of the proceedings for that meeting.

/RA/ Date 9/6/2012
John W. Stetkar, Chairman
Subcommittee on Reliability and PRA

Certified By: John W. Stetkar
Certified on September 6, 2012

Issued on: September 13, 2012

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF THE ACRS RELIABILITY AND PRA SUBCOMMITTEE MEETING
MARCH 7, 2012

The ACRS Reliability and PRA Subcommittee held a meeting on March 7, 2012 in Room T-2B1, 11545 Rockville Pike, Rockville, Maryland. The meeting convened at 8:30am and adjourned at 4:07pm. The entire meeting was open to the public. No written comments or requests for time to make oral statements were received from members of the public related to this meeting.

ATTENDEES

ACRS Members
John Stetkar, Subcommittee Chairman
Dennis Bley, Member
Said Abdel-Khalik, Member
J. Sam Armijo, Member
William Shack, Member
Michael Corradini, Member
Mike Ryan, Member
Joy Rempe, Member
Gordon Skillman, Member
Jack Sieber, Member
Steve Schultz, member
Harold Ray, member

ACRS Staff
John Lai, Designated Federal Official

NRC Staff
Donald Dube, NRO/DSRA
Charles Ader, NRO/DSRA
Todd Hilsmeier, NRO/DSRA
Ed Fuller, NRO/DSRA
Mark Lombard, NRO/DSRA
Theresa Clark, NRO/DSRA
Rachel Vaucher, NRO/DE
Lynn Mrowca, NRO/DSRA
Rani Franovich, NRR/DIRS
Ron Frahm, NRR/DIRS
Ann Marie Grady, NRO/DSRA

Other Attendees
Masqur Khan, Bechtel Power
Vincent Sorel, UNISTAR Nuclear Energy
Vesna Dimitrijevic, AREVA

Biff Bradley, NEI
Kati Austgen, NEI
Patrick O'Regan, EPRI

SUMMARY

The purpose of the meeting is to review the draft Commission paper in response to the Commission's Staff Requirements Memorandum (SRM) on SECY 10-0121 regarding risk-informed regulatory guidance for new reactors. The staff also presented the results of the tabletop exercises which were part of the SRM. The meeting transcripts are attached and contain an accurate description of each matter discussed during the meeting. The presentation slides and handouts used during the meeting are attached to these transcripts.

Major Issues discussed during the meeting are described in the following Table.

Table 1. Major Issues Discussed During the Meeting

Major Issues Discussed	
Issue	Reference Pages in Transcript
Chairman Stetkar asked if the interpretation of quantitative metrics shall not be changed means that every single number that appears in any regulatory guides shouldn't be changed. Don Dube of NRO replied that some of the key measures, like a change in core damage frequency or in large early release frequency, remain firm according to his interpretation.	10
Members and Don discussed the meaning of enhanced margins and safety of new reactors and how they would be applied to the operational flexibility.	11-16
Members, Don, and meeting participants from industry discussed the tabletop results using NEI 04-00 to illustrate decision-making process for the application of risk-informed categorization (10 CFR 50.69).	25-33
Member Ray asked how SSCs in category 3 of 50.69 regulation are treated since it is now not under appendix B. Don and Biff Bradley from NEI explained that the categorization is governed by specific regulation and is reviewed and approved by the NRC per plant-specific basis prior to its application at the plant. The SSCs of the re-categorization will be monitored by the performance-based program established by the licensee.	33-39
Chairman Stetkar asked why ESBWR uses broader (higher) values of risk-significant parameters (e.g.,Fussell-Vesely, RAW) to determine the risk-significance of the SSC categorization. Todd Hilsmeier of NRO replied that the RAW and Fussell-Vesely threshold criteria should be a function of baseline CDF (Appendix A of Reg Guide 1.174), instead using the same values for every plant. Since the ESBWR has a lower baseline CDF, they used a higher RAW and	49-63

Fussell-Vesely criteria. Members, industry participants, and the staff discussed how this “sliding scale” of relative risk metrics is being applied for the risk-informed performance-based applications.	
Member Skillman asked where the fire protection systems would reside in 50.69. Todd Hilsmeier replied that it depends on the fire PRA model and expert panel discretion. Biff Bradley indicated that the conservatism and modeling in fire PRA also impact the categorization of 50.69 for new reactors.	67-71
Don Dube discussed an example of the desk-top exercises for RG 1.174 application. He pointed out that even when the CDF changes are small, there are other principles (e.g., defense-in- depth) that would have to be considered before the changes can be made. Member Sieber and Don discussed how the operator actions and PRA updates are captured for new reactors.	72-87
Don Dube and members discussed the different definitions used for the description of LERF and LRF. Charlie Ader of NRO gave the history of the LERF and LRF development.	87-103
Don Dube described the three options transitioned from LRF to LERF. The staff recommends transition from LRF to LERF at or prior to initial fuel load and discontinued the use of LRF thereafter.	104-112
Don Dube stated that in RG 1.174 there is a need for licensee to discuss the impact of containment performance in addition to the LERF calculation.	113-124
Don Dube and members discussed the change process under Part 52 Appendix A, VIII.B.5 (c) for ex-vessel severe accident (EVSA).	124-139
Don Dube discussed how the containment bypass should be treated for new reactors since the containment bypass was not an ex-vessel accident event in the rule. Don discussed the results of the gap assessment and found that there are sufficient details in tier 1 document to preclude a significant design change without prior NRC approval.	140-151
Ron Frahm of NRR discussed the background information of ROP (Reactor Oversight Program). Ron discussed the results of tabletop exercises on SDP, the SDP analyses could be augmented with additional qualitative considerations, such as deterministic backstops, to ensure that the NRC appropriately addresses performance issues.	152-167
Chairman Stetkar asked what the percentage increase of CDF is for the HPCF pump out of service for an ABWR plant in the tabletop exercise. Don replied that the CDF increase is about 50 percent. For the US-APWR vessel head degradation case, the CDF increase is about 14 percent.	167-173
Rani Franovich of NRR discussed why the deterministic consideration might help the decision-making for the inspection program.	179-181
Chairman Stetkar stated that any risk model needs to go through peer reviews and be consistent with RG 1.200 requirements before the results can be reasonably applied to risk-informed applications. Members and staff discussed the validity of using risk numbers and how the inspection process should proceed using both risk insight	182-188

and defense-in-depth concept.	
Staff discussed the results from the MSPI tabletop exercises and members and staff felt that additional performance indicators may be needed.	189-204
Chairman Stetkar asked if the staff has considered the relative changes of risk metrics instead of absolute changes. Charlie Ader stated that the staff has considered the relative measures but the Commission reaffirmed the guidelines in RG 1.174 which is an absolute measurement of risk metrics. Members asked if it is feasible to propose another option (relative change of risk metrics) for the Commission to consider. Members and staff discussed the possibility of proposing this addition option.	205-231
Don summarized the key points in the Draft SECY paper that the staff plans to submit to the Commission.	232-251
Ron Frahm summarized the proposed options for ROP for new reactors.	251-256
Biff Bradley of NEI provided the industry's view on the draft SECY paper.	257-270
Members had no further comments on the presentation.	271-272
Members and staff discussed the proposed topics to be presented at the Full Committee meeting in April, 2012.	273-277

Table 2. Action Items

ACTION ITEMS	
Action Item	Reference Pages in Transcript
None	

<u>Documents provided to the Subcommittee</u>

1. Draft Commission Paper, "Risk-informed Regulatory Framework for New Reactors", February 3, 2012 (ML12011A191).
2. Summary of Public Meeting to Perform Tabletop Exercises Regarding Guidance on 50.69 and Draft NEI 96-07 Appendix C Related to Ex-Vessel Severe Accident (EVSA) Features for New Reactors, Summary Package on 08/09/2011, dated August 17, 2011 (ML112290891)
3. Summary of the Public Meeting Held on August 25, 2011 to Discuss Preparations for Upcoming Tabletop Exercises for New Reactor Risk Applications in the ROP, dated September 14, 2011 (ML112490043).

4. Summary of Conference Call to Further Discuss Preparations for Upcoming Tabletop Exercises for New Reactor Risk Applications in the ROP on September 7, 2011, dated September 16, 2011 (ML112590197).
5. Summary of Public Meeting to Perform Tabletop Exercises to Complete Licensing Issues and to Discuss the ROP for New Reactors on October 5, 2011, dated October 18, 2011 (ML11291A076).
6. Summary of the Public Meeting to Further Discuss Key Points From Tabletop Exercises for New Reactor Risk Application in ROP on October 26, 2011, dated November 4, 2011 (ML11308A542)
7. Summary of Public Meeting to Discuss Changes During Construction, Draft ISG-025 and NEI 96-07, Appendix C on October 15, 2011, dated November 28, 2011 (ML113320197).

Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: Advisory Committee on Reactor Safeguards Reliability and PRA Subcommittee

Docket Number: (n/a)

Location: Rockville, Maryland

Date: Wednesday, March 7, 2012

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

+ + + + +

RELIABILITY AND PRA SUBCOMMITTEE

+ + + + +

WEDNESDAY,

MARCH 7, 2012

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ROCKVILLE, MARYLAND

+ + + + +

The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 8:30 a.m., JOHN W. STETKAR, Chairman, presiding.

MEMBERS PRESENT:

JOHN W. STETKAR, Chairman

SAID ABDEL-KHALIK

J. SAM ARMIJO

DENNIS C. BLEY

MICHAEL CORRADINI

HAROLD B. RAY

JOY REMPE

MICHAEL T. RYAN

MEMBERS PRESENT (Continued):

STEPHEN P. SCHULTZ

WILLIAM J. SHACK

JOHN D. SIEBER

GORDON R. SKILLMAN

NRC STAFF PRESENT:

JOHN LAI, Designated Federal Official

CHARLES ADER

DONALD DUBE

TODD HILSMEIER

EDWIN FULLER

RONALD FRAHM

ALSO PRESENT:

BIFF BRADLEY

VESNA DIMITRIJEVIC

TABLE OF CONTENTS

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

(8:30 a.m.)

1. OPENING REMARKS

CHAIR STETKAR: The meeting will now come to order. This is a meeting of the Reliability and PRA Subcommittee. I am John Stetkar, Chairman of this Subcommittee meeting.

ACRS members in attendance are Jack Sieber, Steve Schultz, Dick Skillman, Dennis Bley, Harold Ray, Sam Armijo, Mike Ryan, Said Abdel-Khalik, Bill Shack, Joy Rempe, and Dr. Michael Corradini. John Lai of the ACRS staff is the designated federal official for this meeting.

The Subcommittee will hear the staff's discussion of the draft Commission paper and the results of the tabletop exercises in response to the Commission Staff Requirements Memorandum of March 2nd, 2011 on SECY-10-0121 regarding risk-informed regulatory guidance for new reactors. We will hear presentations from the NRC staff and the Nuclear Energy Institute.

There will be a phone bridge line. To preclude interruption of the meeting, the phone will be placed in a listen-in mode during the presentations and Committee discussions.

We have received no written comments or requests for time to make oral statements from members of the public regarding today's meeting. The entire meeting will be open to public attendance.

Secondly, we will gather information and analyze relevant issues and facts and formulate proposed positions and actions as appropriate for deliberation by the full Committee. The requests for participation in today's meeting had been announced as part of the notice of this meeting previously published in the Federal Register. A transcript of the meeting is being kept and will be made available, as stated in the Federal Register notice. Therefore, we request that the participants in this meeting use the microphones located throughout the meeting room when addressing the Subcommittee. The participants should first identify themselves and speak with sufficient clarity and volume so that they may be readily heard.

We will now proceed with the meeting. And, Charlie, do you want to say anything? I will turn it, ask Charles Ader if he will want to say something.

2. INTRODUCTION

MR. ADER: I actually just appreciate the

opportunity to present. Staff has done a lot of hard work here. And I think we're bringing this issue hopefully to a close. Other than that, I'll turn it back to Don.

CHAIR STETKAR: Don, it's all yours.

MR. DUBE: Thank you, John. I want to acknowledge my colleague from Nuclear Reactor Regulation Ron Frahm, who is going to cover the reactor oversight process and tabletop results. I think you'll find that really interesting and a different direction than the licensing.

I also want to acknowledge Eric Powell, who, unfortunately, took ill, literally ill. So I'm going to be covering his topic. I will do the best I can given a little bit of last moment. He was going to cover 50.69. Fortunately, I noticed in the audience the very active participation on 50.69 tabletop in August with the Electric Power Research Institute as well as Vesna Dimitrijevic from AREVA, who actually did some simulations for an active new reactor design. And so if we get into detail, I'll ask them to help answer some questions.

3. 50.69 AND RG 1.174 TABLETOP EXERCISES

MR. DUBE: So, with that, we'll move on. It's going to be, you know, a pretty complete day. I

don't think we'll go until 4:00 or 5:00, but it's going to be a very long day.

CHAIR STETKAR: Don't challenge us.

(Laughter.)

MR. DUBE: What we're going to plan to do is discuss the second series of tabletop exercise results and the staff's response to the SRM, basically going over the draft Commission paper. The Commissioner paper, the final, I'm sure we're going to get stakeholder response. and plan to change it in terms of content. Structurally it might change a little bit, but the options are the options and the recommendations are pretty much the recommendations.

So on the agenda today, we will discuss 50.69 really at a high level, what we concluded from the tabletop exercise. This is on classification of structures, systems, and components, and special treatment thereof. Reg guide 1.174, it's really the process, the thought process, that went through hypothesizing some license amendment requests, risk-informed, and what's the thought process by which the staff would review those and would it make any difference if we have a new reactor proposing a license amendment.

Large release frequency to large early

release frequency, this is a legacy going back 20 years or more. We felt it was part of the assignment, if you will, in the Commission SRM. And we want to bring some closure to this at some point.

We talk about the process for ex-vessel severe accident features, the change process. We'll talk about a gap that the staff identified in the current rule, if you will, and a recommendation to address that.

I think you will find reactor oversight process with very interesting results. We did dozens upon dozens of scenarios and cases and with some interesting results.

In the afternoon, we'll talk about conclusions, the options, recommendations in the draft paper and then the next step. So sit back and relax. It's going to be a full day.

Going back over a year now, a year and a half, in the original Commission paper, 0121, staff proposed to the Commission three options to address the risk-informed framework for new reactors. And this included status quo or treat new reactors the exact same way as the current operating reactors.

Option two was, well, let's go a little bit further, let's do some investigation, let's look

at the existing guidance, make sure there's no significant decrease in the enhanced level of safety of the new reactor designs as a result of implementing risk-informed guidance. This was the staff's recommendation.

Option three was even more radical, which was let's just use new numeric thresholds for new reactors, both in licensing and in reactor oversight.

MEMBER CORRADINI: And that was too radical for the staff?

MR. DUBE: For the Commission. So the Commission approved a hybrid of options one and two. It's probably more option two-ish, but they said continue the existing risk-informed framework pending a series of tabletop exercises that test this guidance. And so that's what the tabletop exercises were.

To test the guidance, we did not test every risk-informed application. I mean, it's just not feasible to do it in the time and resources. We did test the ones that were of greatest interest to the Commission that they spelled out explicitly.

But the Commission went further than to just say, "Go do these tabletops." They said that they reaffirmed the existing safety goals in terms of

core damage frequency and quantitative health objectives, safety performance expectation, subsidiary risk goals and associated risk guidance, the key principles in reg guide 1.174 -- and I'll refresh your memories on that -- and specifically the quantitative metrics.

So that set the boundary conditions for the tabletop. I mean, you know, to be true to the Commission SRM, we had to work within those boundaries, which means we're not going to change quantitative metrics, period. So the metrics that aren't used for current operating fleet we're going to use for the new reactors. But maybe there are some additional qualitative considerations that we could work into it. But that set the stage for the affirming.

CHAIR STETKAR: Don?

MR. DUBE: Yes?

CHAIR STETKAR: And we'll probably get into this as we go on. Your interpretation of quantitative metrics shall not be changed is that every single number that appears anywhere in any regulatory guide is cast in concrete or the concept of the quantitative metrics shouldn't be changed?

MR. DUBE: Some of the key measures, like

a change in core damage frequency, change in large early release frequency, I interpret that to mean those are firm.

CHAIR STETKAR: Okay.

MR. DUBE: The Commission expected advance technologies and the new reactors will result in enhanced margins of safeties. And, at a minimum, new reactors have the same degree of protection of the public and environment as current fleet.

And they went further, and they said, "New reactors with these enhanced margins of safety should have greater operational flexibilities than current reactors."

MEMBER CORRADINI: So that would mean that, even though you stick with the same numbers, you ought to see a larger margin when you look at new reactors?

MR. DUBE: Right.

MEMBER CORRADINI: And it's up to the staff to decide what large is? In other words, even if you fall within a CAP, the expectation of staff is that they're actually going to see when you do an exercise to see a larger difference?

MR. DUBE: Yes. One should not be pushing the goals as aggressively as currently, but it is

allowed. So, I mean, it's a very tough fine line to walk.

MEMBER CORRADINI: I understand. Understood. But, to put it differently, if you have some leading indicators of how things might change; in other words, you have a new plant, you want to change something and the change increases the risk, the allowable risk increase might be different for a new plant versus an old plant because the expectation is the margin should be larger?

MR. DUBE: Right.

MEMBER CORRADINI: Okay.

MR. DUBE: So how that reflects itself is the current operating fleet with nominal risk profile and core damage frequency and large early release frequency, a doubling or tripling of core damage frequency really would be an eye-opener.

A new reactor -- let's take ESBWR, which goes to internal events. It says they're 10-8 per year. We're not going to be so worried if there's a doubling of that because of some change because, quite frankly, there are probably other external events that are dominating anyway. And there is a large margin there. A doubling of that is not as much of an eye-opener as it might be for the current fleet.

MEMBER CORRADINI: But -- okay. That's enough for now. I'll just --

MEMBER SCHULTZ: Don, you had -- excuse me.

MR. DUBE: Go ahead.

MEMBER SCHULTZ: Don, you haven't added that last bullet. That comes from the --

MR. DUBE: Yes.

MEMBER SCHULTZ: -- Commission's expression.

MR. DUBE: This is the --

MEMBER SCHULTZ: So if I take what is there on the slide at face value, the new reactors would have additional margin of safety than existing reactors against the current guidelines and goals, but there is greater operational flexibility allowance.

MR. DUBE: Exactly.

MEMBER SCHULTZ: And so within that additional margin that is available, there would be the opportunity for the operator to use that operational flexibility.

MR. DUBE: Right. And I think that's consistent with what I was just saying.

MEMBER CORRADINI: And when we're done, we're going to hear how you're going to trip that

light fantastic between the hard --

MR. DUBE: Well, yes. I mean, I think we'll see that this recurring theme as we went through these tabletop exercises --

MEMBER CORRADINI: Okay. All right.

MR. DUBE: -- defense-in-depth margin of safety were really two key principles.

MEMBER ABDEL-KHALIK: You said that for the -- you gave the ESBWR as an example. And you said the doubling of the 10-8 would not raise any eyebrows. How about a hundred-fold increase?

MR. DUBE: Yes, that would.

MEMBER ABDEL-KHALIK: It would?

MR. DUBE: That would.

MEMBER ABDEL-KHALIK: But it would still be within the numerical guidelines.

MR. DUBE: Right, but in order to do that --

CHAIR STETKAR: A 10,000-fold increase would still be within.

MR. DUBE: Right. In order to do that --

MEMBER ABDEL-KHALIK: I forgot, just trying not to push that --

MR. DUBE: In order to do that, what probably occurred along the way is a significant

decrease in defense-in-depth and other principles. And, again, this will be a recurring theme. Having a small change in core damage frequency is for -- small risk increase is necessary but not sufficient. And if there's a significant decrease in defense-in-depth, staff would have consulted that.

MEMBER ABDEL-KHALIK: And you'll tell us?

MR. DUBE: Yes. I will try to.

MEMBER ABDEL-KHALIK: Okay.

MEMBER CORRADINI: Can I ask a question?

MR. DUBE: Go ahead.

MEMBER CORRADINI: What you are really saying is there is a larger uncertainty with the smaller the number. And, therefore, if you were ever able to compute the 95/95 using Professor Wallis' approach to life, it wouldn't be 10-8. It might be much higher.

MR. DUBE: Yes.

MEMBER CORRADINI: That's really what you're saying.

MR. DUBE: Yes, that and also a significant increase probably means that in order to get that, a large amount of very important equipment would be -- its performance would have changed or might be out of service or special treatment thereof.

And that's probably what it means. And so defense-in-depth may have been eroded along the way.

So, again, I mean, we'll talk about the key principles. But a small risk increase is just one of the key principles.

So what were the tabletop exercises? It was pretty aggressive, but it was fun. But it was a busy 12 months.

At the September 20th Subcommittee meeting of the ACRS, we talked about the first four or five tabletops, but back before the SRM, we had started looking at changed process for ex-vessel severe accident design features. And we identified a potential gap here. We'll talk about that. So I won't go any further because we did talk about these first few bullets.

At the last Subcommittee meeting, we talked about risk-informed inspection of piping on May 4th. We did risk-informed tech spec initiative 4b. This is on changing completion times where allowed outage times is what's recalled in the past. And we did that simultaneously with the maintenance rule (a)(4), which is managing risk joint online situations. They kind of go hand in hand. We gave examples and talked quite a bit of detailed

unbelievable participation by virtually all of the reactor vendors, had well over 100 scenarios and cases that we analyzed.

At the end of June, we did surveillance frequency control program. We'll highlight those results, but we talked about them on September 20th. We found that that was really more governed by performance monitoring than raw changes in risk, if you will. What we did talk about on September 20 was 50.69. And I will go into it, at least at a high level. We finished up on the change process for ex-vessel severe accident features.

Then October 5th was a very busy day. We did some scenarios on reg guide 1.174. Basically we said let's hypothesize eight kinds of changes, license amendments on a variety of plants. And we're not going to get hung up on what the bottom line change in core damage frequency or risk was. Let's go through the thought process and the staff's thought process on how we would review that and what would be some of the considerations besides risk. And we really drew upon some expertise in NRR with review of dozens of license amendment requests. And they gave us some of the thought process that we would carry over to the new fleet. So that was a very good opportunity.

We tackled this large release frequency as a risk metric to large early release frequency. We have come up with three major options and a recommendation to finally handle this question.

And then a large portion was on the reactor oversight process. There we did a significance determination process for inspection findings, reactive inspections under management directive 8.3. This is like sending out a special inspection or augmented inspection team because of a plant incident. And then the mitigating systems performance index and took actual examples from the current fleet and said, well, if we applied something like this to new reactor, what would the results kind of look like? And does the response look like what we would expect, you know, judgmentally? And I think you'll find some interesting results.

We had a follow-up discussion on the ROP on the 26th of October. So that was a very busy day.

So we'll talk about 50.69 at a high level. Again, I'm filling in a little bit. I mean, I am familiar with them obviously, but I'll be drawing upon my colleague Eric Powell's notes a little more than I might have otherwise. If there are real detailed questions on the process, you know, we have got some

representatives here I think.

So at a high level, 50,69 is a classification of structures, systems, and components, putting them into one of four bins and considering what special treatments one may be able to eliminate.

At the upper left are the traditional safety-related. And what the PRA might say as safety-significant have always been thresholds for using PRA metrics, like Fussell-Vesely or risk achievement works that would put them in this bucket. That would basically have the same treatment as the current fleet, the current requirements.

RISC-2 is the non-safety-related but, for whatever reason, safety-significant. It could be a standby feedwater pump that's not an emergency feedwater pump but because of the risk profile at a plant plays an important role in mitigating loss of feedwater events. And they may end up having risk metrics above a certain value.

In new reactor space, at least for the passive plants, there has been a great deal of overlap between RISC-2 and the regulatory treatment of non-safety systems. We have this category RTNSS. And they do have special treatment for the passive plants, like the AP1000 and ESBWR, which rely on passive

safety features but have non-safety systems that are not credited in the chapter 15 accident analysis but provide a great deal of defense-in-depth. And these might be a start-up feedwater pump, non-safety diesel generator, non-safety residual heat removal system, and the like. And those get special treatment. They have typically reliability targets, if you will, and unavailability targets. And so there is a good deal of overlap, at least to the passive plants.

MEMBER SHACK: There is DRAP and ORAP for the non-passive plants --

MR. DUBE: Yes, DRAP and ORAP.

MEMBER SHACK: -- which have a kind of a similar --

MR. DUBE: Exactly, similar kind of situation with targets like that. Right, exactly. Thanks.

The next easy one, RISC-4, are non-safety-related, low safety significance. So those, there is not too much controversy.

MEMBER CORRADINI: Are there any kinds of examples of those? It seems like depending on how I do it, everything would fit in the other three bins. What are some things that --

MR. DUBE: A lot of the things, most of

the things fit, many of the things fit, in that category.

MEMBER CORRADINI: This has the power generation unit, I assume.

MR. DUBE: Yes, power generation.

MEMBER CORRADINI: But it isn't --

MR. DUBE: Drains pump? I don't know. Something like that, some kind of --

MEMBER CORRADINI: Just so I've got it right, relative to two to four, the difference in two to four depends upon the parameter you use to decide. In other words, if I turn the screws, a lot of four becomes two.

MR. DUBE: Some fours become two.

MEMBER CORRADINI: Have I got it right?

MR. DUBE: Yes. I mean, there are risk metrics, Fussell-Vesely of greater than particularly 0.005, which is half a percent contribution to core damage frequency.

MEMBER CORRADINI: Especially ESBWR.

MR. DUBE: Right. Thank you.

CHAIR STETKAR: I knew that would come up.

(Laughter.)

CHAIR STETKAR: I mean, if I didn't say it, he would say it. So I waited for the slide.

MR. DUBE: Risk achievement worth greater than 2 or common cause failure risk achievement worth of greater than 20, but there are also some other factors that may come into play. For example, there could be --

MEMBER CORRADINI: That's fine.

MR. DUBE: Something might have a big roll. It's a reliability impact, reactor trip frequency. You know, this thing trips, kind of take down the feedwater system and result ultimately in a reactor trip. I mean, that could be enough to put it over. And in the integrated decision-making panel, they may have these qualitative and other considerations of why they want to move it up.

MEMBER CORRADINI: That's fine. That's fine.

MR. DUBE: RISC-3 is, you know, where all of the issues have been in the past going back, you know, almost a decade now, because these are safety-related and low safety-significant. And this is a category which is "Okay. How do we handle this?"

Fortunately, I mean, there is a regulation in 50.69 of what to do with RISC-3. So it's pretty firm. There's a fair degree of considerations that go into what gets into RISC-3 and how it's treated.

And just because it's in RISC-3, one cannot ignore it. One has to monitor its performance pretty aggressively if a RISC-3 has a component, has a trend, an adverse trend in terms of failure rate, corrective action has to be taken. It can't just be left to degrade, even though it's determined to be of low safety significance.

So I got permission from EPRI to use this slide. This is just a high level --

MEMBER CORRADINI: Written?

MR. DUBE: No. This is at a high level. And the guidance in NEI, 004 risk categorization of the process. So at the extreme left, one looks at the risk category characterization. That's a starting point, everything from internal events to external events and fire shutdown risk. And then depending, it could take several paths, could go directly to high safety significance, RISC-1 and RISC-2.

But, just because something is below the quantitative guidelines and the guidance and starts out as low safety significance, that's the garden path LSS. There are a number of considerations, everything from defense-in-depth; in other words, you know, maybe PRA says, you know, this particular equipment doesn't contribute that much but is it the only one of a kind

and if that equipment had some kind of degradation, it might impact because there's nothing else backing it up, maybe it's a high-pressure injection function, if you will, or something along those lines.

So there's defense-in-depth characterization. Furthermore, there are new risk sensitivity studies to see if the plant risk is sensitive to performance of that equipment. And then, really, that blue block is the integrated decision-making panel review.

This is actually required by regulation. And, in fact, it even spells out in the regulation the characteristics, the expertise that has to make up this IDP: operating experience, engineering review, licensing requirements, PRA. It's a regulation. It's a part of regulation.

And often they will take something for other consideration besides risk that says, well, as I mentioned earlier, maybe, you know, all PRA numbers aren't above the threshold, but, you know, it's really heavily relied upon for the design basis accident analysis, defense-in-depth, for just power generation, if you will, or other considerations. And they can add to that, and they have added to that. And that could bump it up into the high safety-significant

category.

So what comes out of that, then, is the RISC-1, the RISC-2, the RISC-3, and the RISC-4.

MEMBER CORRADINI: Don?

MR. DUBE: Yes?

MEMBER CORRADINI: Before you leave this -- and I have to apologize because I know you are just filling it. So you may actually need some help from the back. I've not read or studied anyway any I/O before. And, as I was going through the slides from the workshop on this process, I came across something that I found pretty interesting there, actually, in one of EPRI's slides.

In particular, it has to do with the risk sensitivity study results. They had kind of an interesting plot showing as you uniformly increase failure rates or something like that by factors of one and a half, two, three high, and so forth, there is kind of an anomaly or behavior. And that's sort of -- it's interesting for the IDP to look at that because it gives you a sense of rates of change and margins, if you will.

And I was curious. Is the guidance in NEI 0004 explicit about performing that type of sort of progressive sensitivity calculation or is this just

something that was done for this particular application?

MR. DUBE: No, no. I think --

MEMBER CORRADINI: Is everybody doing it?

MR. DUBE: They do some kind of sensitivity. The question is where do you stop? I mean, it all -- increase the failure rate by one and a half, there's not much controversy, then two, then three. I mean --

CHAIR STETKAR: Well, that's exactly my question, is, well, but I don't necessarily care about the endpoint. And one could say, well, how big could it be to trip over the acceptance criterion. I mean, you know, that's --

MR. DUBE: Yes.

CHAIR STETKAR: That's a possible endpoint. But what I was more concerned about, does the process actually generate this type of progressive information for input to the integrated decision-making panel or do people only do a single snapshot? Well, we'll give you the sensitivity study with a factor of two increase, period, and not show that progressive behavior.

MR. DUBE: I can't say that it's actual practice. I don't know if --

CHAIR STETKAR: That's what I say. I'm not --

MR. DUBE: -- representatives from EPRI, Patrick O'Regan or Dr. Dimitrijevic want to add anything. Dr. Dimitrijevic did -- well, okay. Biff Bradley from NEI.

MR. BRADLEY: I'm Biff Bradley, NEI. Vesna may be better speaking to the specific thing that John raised, but in 00-04, there was a tremendous amount of discussion that went into that since the factor and ultimately what we put in there was a range. I think it was three to five. There is no specific number. It was sort of left open for the pilot to deal with.

CHAIR STETKAR: Yes, Biff. I am less interested -- I don't care about the specific numbers, whether they're one and a half or 1,500.

MR. BRADLEY: Yes.

CHAIR STETKAR: I am more concerned, does the process create this type of plot that I pulled up here that was shown in the EPRI results to show essentially as you increase from one and a half, two, three, five, how linear or non-linear are the results to that theme because that is a piece of, the non-linearity of that is a piece of, information that

could be useful for the IBP to show, you know, how fast are you eroding margins as a function of -- granted, you know, equipment failure rates aren't going to change by uniformly a factor of ten overnight.

On the other hand, if suddenly the curve goes vertical between five and seven, that's sort of an interesting piece of information.

MR. BRADLEY: Right. Yes. I'm pretty sure the 00-04 doesn't speak explicitly to that non-linearity, but there is additional EPRI guidance out there.

Also, the issue that drove this was primarily cross-system common-cause failure. So we have a lot of guidance in there about grouping similar types of components across systems and tracking failure rates. That was the big controversy at the time of 00-04.

I don't know if Vesna wants to add something.

DR. DIMITRIJEVIC: This is not typically --

CHAIR STETKAR: Even though we all know you, you have to identify yourself because Charles doesn't know you.

DR. DIMITRIJEVIC: The NEI doesn't actually specify the requirement for this sensitivity. We did it in purpose of this tabletop. In that, you will have to do these. So you will have to make an assumption on some. And maybe in purpose of presenting at the risk of this application, you can choose to do the sensitivity to see what is --

CHAIR STETKAR: No. That helps. I was just curious because I found it kind of an interesting --

DR. DIMITRIJEVIC: I know.

CHAIR STETKAR: -- plot. You know, if I was sitting back in that blue box there saying, "Well, I meet all of these other criteria, but, see, I am uncertain about the failure rate of a particular piece of, oh, let's say, a very large squib valve."

And if there is some important non-linearity in terms of delta risk as a function of that failure rate, this gives me sitting in that box another sense of quantitative information that I can use to say, well, if I was close to the margin of including that in one or another category, it might affect my decision.

So from what I'm hearing, it's something you did for the tabletop. It isn't necessarily

something that --

DR. DIMITRIJEVIC: Yes.

MEMBER SHACK: It is in the guidance. I mean, a risk sensitivity study is performed to investigate the aggregate impact of potentially changing treatment of these low safety-significant components. That really sort of gets at what you're looking at. You're really not just looking at even one at a time, which is always a problem with the sensitivity studies.

MR. DUBE: Except for the fact you can interpret that by just assigning one number, you know, just --

MEMBER SHACK: Risk sensitivity study I suppose you could.

CHAIR STETKAR: I will look at the sensitivity to the aggregate change in the failure rate of everything by a factor of three, period. And this exercise did that but at several different values of that multiplier so that you could see you had a different sense of -- as I said, if this suddenly went vertical if you changed the failure rate from five to seven, that's -- there's nothing to believe that everything would change by that amount, but it does give you some sense of sensitivity.

MR. DUBE: I don't want to get too mathematical here, but mathematically it makes sense because if you start getting important cut sets, which is an initiating event, A train and B train fail in a cut set, it's a quadratic because if you increase the failure rate of A and B, at some point that quadratic starts to dominate.

CHAIR STETKAR: That's right. That's right.

MR. DUBE: So if you push this limit far enough, you really transition from linear to --

CHAIR STETKAR: Yes, yes.

MR. DUBE: But our challenge, back to 50.69, our challenge in here was, unlike risk-informed tech spec, where there's maybe a dozen, probably several dozen are implementing some form of risk-informed tech specs, risk-managed tech specs, and risk-informed and service inspection where I think more than half of the fleet are implementing it.

We actually don't have anyone who has actually implemented 50.69 per se. Now, South Texas 1 and 2 implemented something similar, graded QA. And we have that experience. And Vogtle 1 and 2 are volunteering, are going to be in a pilot I believe for 50.69. But we don't have a lot of experience to draw

upon. So it was a little bit of a challenge.

So some of the ground rules on applying it to new reactors, one does have to start out with an initial classification in the traditional safety-related, non-safety-related. What we were told is on the tabletop, it would be nice to apply 50.69 before the equipment was purchased. I mean, that's where the potential benefit is. But it's too late for the current wave of the new reactor designs because, you know, they have to send out the specs long ago, long ago before they were in the case of Vogtle 3 and 4 were issued the COL. So it's actually probably too late for that purpose. That was an interesting incident.

MEMBER SHACK: But the passive plants have cut back the scope of safety equipment rather drastically anyway.

MR. DUBE: Yes, yes. Right. That's true. And I'm not sure where they would --

MEMBER SHACK: EPR might benefit.

MR. DUBE: Perhaps, yes.

MEMBER SKILLMAN: For the new designs that have not yet achieved design certification, why wouldn't 50.69 be a requirement?

MR. DUBE: It's a requirement? It's

voluntary by regulation. It's a voluntary rule. They don't have to implement 50.69.

MEMBER SKILLMAN: Okay. Thank you.

CHAIR STETKAR: The only thing they do have to do is the new plants have to -- if they're a passive plant, they have to develop this --

MR. DUBE: The regulatory achievement and not say --

CHAIR STETKAR: If they're an active plant, they have to have a design reliability assurance program.

MR. DUBE: Yes and operational -- so they are required to have --

CHAIR STETKAR: They are required to have that --

MR. DUBE: Yes.

CHAIR STETKAR: -- by rule.

MR. DUBE: Exactly.

CHAIR STETKAR: But it's kind of like 50.69, yes, except it doesn't have the vertical component to take things out. It doesn't recategorize things from one to three or two to four in principle.

MEMBER RAY: Well, Don, yesterday in the context of ROP, I asked about how does it share with appendix B. Now, we're just now talking about the

benefits that arise from 560.69, the implication being the benefits would be for the safety-related components, I take it. I mean, that's the area of benefit that I can see.

Does that mean that the requirements for criteria of appendix B somehow would not apply to a pump valve component of some kind?

MR. DUBE: If it is in category RISC-3 and the applicant for this risk-informed initiative, 1 and 2 change the treatment of -- and they choose this -- of that component and say, you know, we will change the treatment under appendix B or environmental qualification or ASME IEEE codes and standards, a whole list, appendix J testing, it could.

MEMBER RAY: They could do that unilaterally or does that have to be --

MR. DUBE: The license amendment. Staff has to review it by the license.

MEMBER RAY: So this is simply an opportunity to seek a reduction in what otherwise remain the regulatory requirements of appendix B. And so that's where the --

MR. DUBE: The benefit in terms of procurement would be a) if it's of this large -- if there were a large set of equipment where they could

buy equipment commercially, as opposed to under an appendix B program with all the bells and whistles and documentation.

MEMBER RAY: Even though it's safety-related.

MR. DUBE: Yes.

MEMBER RAY: Okay. Well, that helps answer the question I was trying to get at yesterday, which is that this is a mechanism that has to be actually used and approved --

MR. DUBE: Yes.

MEMBER RAY: -- in order to reduce what are otherwise the requirements that continue to apply.

MR. DUBE: Exactly. And RISC-3 may remove requirements RISC-2 adds. That's the benefit of this.

MEMBER RAY: Yes, I know, but that is similar to what John is saying about RTNSS requirements already exist. I'm trying to probe the other side of it, which is the means by which this relief is obtained. And I understand it to be what the NRC has to approve.

MR. DUBE: Yes.

MEMBER RAY: Okay.

MR. DUBE: The staff will approve the process and then ask a large number of questions that

give examples of --

MEMBER RAY: Yes. I understand, but it seems as though it's quite ad hoc, I guess I would say, at this point. In other words, you've got to make -- a specific licensee makes a specific proposal to do something very precise and gets that approved, but that's just a one-off kind of --

MR. DUBE: I wouldn't call it ad hoc. I mean, there's a lot of guidance here. An actual regulation --

MEMBER SHACK: Plant-specific. It's plant-specific.

MEMBER RAY: All right. That's fine. I chose the wrong word, then.

MR. DUBE: Yes.

MEMBER RAY: But the point is it's plant-specific, then.

MR. DUBE: Yes.

MEMBER RAY: Okay. Under part 52, maybe it applies to a lot of people.

CHAIR STETKAR: What it says is component X at plant A may be in category 3; whereas, component X, if you went and looked at that component, what looks exactly like component X in plant A, in plant B, it might still be in RISC-1.

MEMBER RAY: Well, it really doesn't automatically do anything in my mind, John. That's why I'm asking the question the way I am. If it's safety-related, appendix B still applies until you get some relaxation.

CHAIR STETKAR: Well, but that 50.69 rule gives you that relaxation.

MEMBER SHACK: It isn't in RISC-3 until the NRC agrees it's in RISC-3.

MEMBER RAY: Well, I'm going to persist here because it is something I'm still not satisfied. But what RISC-3 means when it comes to a programmatic requirement, which is in the regulations and is very explicit and has forever been a programmatic issue as far as enforcement is concerned, what exactly the relief is needs to be identified. You don't just say it's in RISC-3. Right?

MR. DUBE: Mr. Bradley from NEI is very familiar with the guidance.

MR. BRADLEY: Yes. Biff Bradley, NEI.

Just for everyone's recollection, there are high-level treatment requirements in 50.69 that replace those of the regulations that you are exempted from. And it is reasonable confidence versus reasonable assurance. And it's test inspection and

corrective action. And the plant has to set up programs to address those for RISC-3.

Those are certainly auditable. And we're going through all of that right now at Vogtle 1/2, as someone mentioned. NRC has been heavily involved observing the categorization as well as the treatment aspects of it.

MEMBER RAY: Well, Biff, that does go to the question I'm asking. At what point are the exact programmatic requirements that are then subject to enforcement or inspection and enforcement as a program established at Vogtle?

MR. BRADLEY: It's a performance-based rule. There is no reg guide that describes programmatic expectation for RISC-3 treatment. It's performance-based. So it's the performance monitoring. And then for the accident functions, which you can't necessarily glean from normal testing, you have to make the case that you're providing reasonable confidence of that. So there is no reg guide that says, "Here's the RISC-3 program that we're going to go inspect you to."

MEMBER RAY: That answers my question better than anything else so far. So basically you are exempted from the end, the RISC-3, but you then

need to replace it with something. And the something that you replace it with is a performance-based thing of your own development, I guess I would say.

MR. BRADLEY: Yes.

MEMBER RAY: It doesn't come out of the agency here. But programmatically it's up to you. You no longer have to comply with appendix B if it's in RISC-3, but you need to have something.

MR. BRADLEY: Correct.

MEMBER RAY: Okay.

MR. BRADLEY: And you're still subject to if it fails whatever --

MEMBER RAY: Yes, yes, yes.

MR. BRADLEY: -- exposure you have there.

MEMBER RAY: Yes. Well, I understand, but I was just asking about the programmatic issue --

MR. BRADLEY: Right.

MEMBER RAY: -- by itself.

MR. BRADLEY: Okay.

MR. DUBE: Good. I am glad that helped.

Really moving along, so the process assumes a full-scope PRA or at least some consideration if, say, for example, we have seismic margins, instead of seismic PRA, that have to take that in consideration and the same thing with

shutdown. And then all functions are noted, including beyond design basis functions.

So what we did during the exercise, during the tabletop -- and we have, as I said, participants -- Electric Power Research Institute gave an overview of the process. And I'm not going to dwell on it.

Then a representative from AREVA talked about, well, what if one would apply this process to new active PWR? What would be the results in terms of the classification of SSCs? What would the distribution look like? Is it radically different from our one test case past experience, South Texas, or somewhat different? And General Electric-Hitachi also did it on an ESBWR.

So we compared the new build and an operating plant to try to --

MEMBER CORRADINI: Repeat that last part. You were talking about the active PWR, but then you parenthetically --

CHAIR STETKAR: That's on the next slide.

MEMBER CORRADINI: Oh. That is on the next slide. Never mind.

MR. DUBE: GE-Hitachi also looked at --

MEMBER CORRADINI: Okay. It's in the next slide. I missed that. I'm sorry.

MR. DUBE: Now, the interesting --

MEMBER CORRADINI: I just won't ask.

MR. DUBE: -- thing to really keep in mind, which brought home the point, is to think about this. How many components are there in a new nuclear power plant?

MEMBER CORRADINI: How do you define component?

PARTICIPANT: Really.

MEMBER CORRADINI: Oh, for Christ sake.

MR. DUBE: Millions, right? Yet, in a typical PRA, roughly 2,000 SSCs are modeled in the PRA. They might have super-components, like a pump, not a peace bar.

CHAIR STETKAR: Or a black box for the entire I&C system, for example.

MR. DUBE: But the point is only a very small fraction of the entire plant is modeled in the PRA, which means that most -- if you just start with the raw risk numbers, you're starting a very small population that might populate RISC categories 1 and 2, if you will.

And so, really, a lot of emphasis is on using the risk PRA results, what to do with those 2,000. Really, the bigger question is what do you do

with the remaining several million minus 2,000 that are --

CHAIR STETKAR: Well, but, Don, you know, you have to be a bit careful when you start throwing those numbers around because you have to presume the people who do risk assessment are fairly intelligent and they understand risk and they understand safety and they understand power plants. So that, for example, the motor operator on a particular door that goes into the turbine building isn't in the PRA. And, yet, it's probably one of your one million components in a power plant.

So that saying that 2,000 over one million is a small fraction of things is true in that sense, but it's not true. You know, the assertion that there are a lot of other things that you need to look at that might be potentially important to risk doesn't quite follow I don't think.

MR. DUBE: No. I'm sorry. I didn't mean to say it's not important to risk. What I meant to say is you'll have in the classification -- you don't have PRA numbers to assign. And so you have to rely more on this blue box --

CHAIR STETKAR: Yes, but the --

MR. DUBE: -- the defense-in-depth and the

integrated decision-making panel. That was my --

CHAIR STETKAR: True. And there certainly are at the margins things like radiation monitoring, for example, some of those functions that don't get --

MR. DUBE: Sorry if I came across --

CHAIR STETKAR: You know, your example of the heater drain pump or, you know, some waste-drain transfer pump or something like that. In principle, the blue box folks need to think about that but not too hard.

MR. DUBE: Right. Okay. Thanks, Biff, for helping.

And I'm going to give just a high level what were the results of looking at the active pressurized water reactor and comparing the categorization to the operating reactors, roughly the same categorization/distribution result. There were some differences but in terms of which equipment or SSCs felt in box 1, 2, 3 or 4 on a percentage basis, recognizing that in this case it was not the entire plant that was looked at.

And also the fact that in the categorization that was done for the new reactor PWR, they didn't go through the step for the decision-making panel, which would tend to add to

categories 1 and 2, but the categorization was roughly equivalent to what was seen in South Texas project 1 and 2.

CHAIR STETKAR: Don, could you leave that one up for a second? Because I wanted to ask a question that EPRI may want to address. Because you weren't directly involved, you probably can't. I was looking at the results from that exercise. And the slide that you had -- I guess go up to whatever slide you had that -- the summary slide, that one right there.

The second sub-bullet there says, "Maximize the percentage of SSCs in RISC-3." There might be some reduction in the future. And I read some of that stuff.

And on the RISC-1 and 3 side of the vertical line, the results sort of mirror what you were saying qualitatively, that for what's characterized as operating plants -- I'll call it South Texas, and I'll use round numbers kind of -- about 75 percent of the safety-related stuff was in RISC-3. And about 80, let's say 85 percent, was in RISC-3 for this particular new-build active plant. Those numbers are different, but they're not all that different, --

MR. DUBE: Yes.

CHAIR STETKAR: -- especially given the uncertainties. I was more interested on the right side of the vertical line, where for currently operating plants, about 99 percent of the non-safety equipment is in RISC-4; whereas, about 80 percent of the non-safety-related equipment for the new-build plant is in RISC-4. And that's a 20 percent in RISC-2 versus one percent in RISC-2. That's quite a large difference.

And I was curious why that large difference. If you are looking at the EPRI slides -- we don't have them in front of us here, unfortunately. So I don't want to bring them into this discussion if we don't need to do that.

MR. DUBE: Right.

CHAIR STETKAR: But I was somewhat curious about why that large difference. I mean, on the safety side of the line, the conclusion is, well, there isn't anything different, you know, substantively different from the exercise that South Texas went through to lead us to believe that the metrics wouldn't work as applied for differentiation between RISC-1 and RISC-3, at least in terms of populating bins of equipment. That doesn't seem to

necessarily be true on the other side, though. I was just curious --

MR. DUBE: For the specifics, Dr. Dimitrijevic?

DR. DIMITRIJEVIC: Yes.

CHAIR STETKAR: I was hoping you would come up.

DR. DIMITRIJEVIC: Yes. This is because it is very clear, actually, because there was some component modeled in PRA. The South Texas result includes everything.

CHAIR STETKAR: Okay.

DR. DIMITRIJEVIC: So, therefore, because we didn't have a disintegrated decision panel --

CHAIR STETKAR: Oh, okay. Thank you. That I understand. Thank you.

MEMBER SCHULTZ: Don, for clarification, can you go back the EPRI slide, 10? So is it assumed that the disintegrated decision panel is going to only consider the LSS arrow there that comes into the blue box, and, therefore, the percentage of high safety significance items can only increase --

MR. DUBE: Yes.

MEMBER SCHULTZ: -- as a result of the panel or are they also looking at the input in other

areas that are above that that come into them as marked by the other evaluations as high safety significance. Could they not put those in the low safety significance category?

MR. DUBE: I believe in most cases, it would bump up from LSS to HSS.

MEMBER BLEY: But they do look at the others.

MR. BRADLEY: They look at everything. Presented to the IDP as low, they can decide to make high. And there are a whole bunch of provisions in the guidance for reasons you might want to do that, even though it came out low. If it's presented as high, they cannot make it low.

MEMBER RAY: Right.

MR. BRADLEY: There are a handful of conditions where if it's high for some particular initiator but low overall, the IDP can consider that. But basically it's a one-way gate. If it's presented high, they can confirm that. If it's low, they can either confirm that or decide to make it high for any reason they choose.

DR. DIMITRIJEVIC: But also in this case, Steve, what is happening is that what is not in PRA would be assumed to be LSS, to go there to be decided

on. And this wasn't done in this example because we did not really look in anything which was not in the PRA. So that why, actually, the population moved up, instead of down --

MR. DUBE: Yes. I mean, that was my point about the --

DR. DIMITRIJEVIC: -- in the low, in the low.

MR. DUBE: -- the million components minus 2,000 is that they're on the low track there until decided to bump up. And so if it were decided to extend 50.69 to that whole population that's not in the PRA, that's a lot of work to do. And so, you know, it's a judgment call. I mean, it's a cost-benefit trade-off between putting in the effort to evaluate and changing the categorization of this equipment versus what does one gain from it.

MEMBER SCHULTZ: That is very helpful. Thank you.

MR. DUBE: So, moving along, I'm going to really go quickly. You know, the ESBWR, GE-Hitachi discussed the results. In some ways, the results in terms of the distribution were similar to the active pressurized water reactor, same magical 2,000 components, rather than the PRA. It's not a law of

nature, but it turns out to be not too far off.

They did the sensitivity study to look at the number of components that would be categorized as low safety significance in the safety-related group.

MEMBER SHACK: Did they change their criteria, the way they did for the DRAP program?

CHAIR STETKAR: They applied the same criteria as they did for the DRAP program. And that's what I want to ask the staff about.

MR. DUBE: Okay.

CHAIR STETKAR: Because when we get into the Commission paper, you -- well, we'll talk about the Commission paper this afternoon. So let's leave it for that.

It's my understanding that ESBWR, I mean, when you preface it for the record that currently operating plants and the active new-builds that I'm aware of have applied numerical criteria for the PRA of Fussell-Vesely important greater than .005, risk achievement worth of greater than 2, and common cause risk achievement worth of greater than 20 for their determination of risk, RISC-1 versus RISC-3 and RISC-2 versus RISC-4.

MR. DUBE: ESBWR did, yes.

CHAIR STETKAR: Safety significance.

ESBWR uses Fussell-Vesely importance of greater than .01, a factor of 2 higher; risk achievement worth greater than 5, a factor of 2 and a half higher; and common cause failure risk achievement worth of greater than 50, a factor of 2 and a half higher. Therefore, they are not using consistent measures of safety significance.

MEMBER CORRADINI: Different measures.

CHAIR STETKAR: They are not using consistent measures of safety significance. They are using much broader acceptance criteria. And, therefore, their populations of equipment in RISC-1 and RISC-2 are affected by that numerical value, are smaller than they would be if they applied the numbers that everybody else is using.

MR. DUBE: Right.

CHAIR STETKAR: Why is the staff comfortable with that? And why is the staff comfortable with that is question number one. If the staff is comfortable with that, why shouldn't the regulations -- regulatory guides, not regulations, be changed to embrace that notion so that they can be used uniformly by all new-build plants, not only the ESBWR in isolation because there is some sort of special case.

MR. DUBE: Right.

MEMBER CORRADINI: Good question. Who are you going to get to answer that one?

MR. DUBE: I have a man on deck there because this was a comment in response to the original Commission paper back in SECY-10-0121.

MEMBER CORRADINI: I'll bet you I know who it is.

MR. DUBE: I'm going to call upon -- you know, I have three help lines.

(Laughter.)

MR. DUBE: -- Todd Hilsmeier, who --

MEMBER CORRADINI: This is not surprising.

MR. DUBE: -- can directly answer this one.

MR. HILSMEIER: My name is Todd Hilsmeier from NRC Office of New Reactors.

And could you repeat your question again?

(Laughter.)

MR. HILSMEIER: I was in a happy space.

CHAIR STETKAR: It is kind of a two-point, is why kind of in an absolute sense does the staff accept the ESBWR's use of different numerical importance measures compared to what are currently being used in the current operating fleet and by other

new-build plants. So that is in an absolute sense.

MEMBER SHACK: Not the ABWR.

CHAIR STETKAR: Not the -- I said "current." I'm pretty careful sometimes.

MEMBER CORRADINI: But it's clear from the explanation standpoint the ABWR sits in the same category.

CHAIR STETKAR: ABWR, yes. But, in particular, EPR and US-APWR do not. And, actually, ABWR as it is being applied for the current in-progress COL does not because the COLA applicant, COL applicant for the ABWR is using what's called the commonly applied numerical screening values. So, even though in the design certification for ABWR they are anomalous, the applicant is using the lower value.

So why is the staff comfortable with that? That is one question. If the staff is comfortable with that -- and you must because the design is certified. And we have raised this question before.

MR. DUBE: Not quite yet certified.

CHAIR STETKAR: Yes. I'm sorry. I'm sorry. Thank you. It's nearly there. This is not the point of contention on that design certification.

If the staff is comfortable with that difference, that is a philosophical difference, if I

will, because now for this one particular new reactor, the staff has accepted essentially a sliding scale on those relative risk metrics --

MR. DUBE: Right.

CHAIR STETKAR: -- to determine risk significance. And if the staff has accepted that, you agree with that, why is that not part of your Commission paper in terms of options to the Commission? We will discuss that this afternoon, but I might as well get it out this morning.

It's not mentioned in the Commission paper. It only says, well, because the 50.69 process consistently uses relative risk, we don't need to say anything about that. The process as it's applied works okay. But it doesn't consistently apply relative risk. We have examples where it is inconsistently applying those measures.

Those are my questions basically. Number one, if you buy into the process that was used for ESBWR, if you do.

MR. HILSMEIER: First, for the ESBWR, they use reg guide 1.174 to justify their RAW threshold corduroy of five. This is in I believe it is appendix A of reg guide 1.174. It says that the RAW and Fussell-Vesely threshold criteria should be a function

of baseline CDF, instead of the same for every plant. And so since the ESBWR has a lower baseline CDF, they used a higher RAW and Fussell-Vesely criteria.

MEMBER CORRADINI: So they have bought into a sliding scale?

MR. HILSMEIER: Correct.

MEMBER CORRADINI: Okay.

MR. HILSMEIER: And for RAW 5, if a plant has a RAW of 5 and is failed, the CDF would increase from 1E-8 to I believe 5E-8, so about 1E-7 delta CDF while in reg guide 1.174 and for current operating plants, acceptable delta CDF if 1E-6. And so with a RAW of 5 for ESBWR, there is a smaller delta CDF, much less than the 1E-6 in the reg guide 1.174 --

MEMBER CORRADINI: For internal events at power.

MR. HILSMEIER: Excuse me?

MEMBER CORRADINI: For internal events at power?

MR. DUBE: Yes.

MR. HILSMEIER: Yes, correct.

MR. DUBE: But event the external events have --

MR. HILSMEIER: Yes. They looked at external events, too.

MEMBER CORRADINI: Maybe John is going to go there, but he is just clever. So what you are really saying is staff has already ad hoc developed a sliding scale philosophy for design certification. So why hasn't staff accepted that sliding scale philosophy for solving this problem? I mean, am I missing something?

MR. HILSMEIER: No. It's just -- as I said during the ACRS meeting on ESBWR, we could not say that their method was not appropriate. And I did agree with their approach. I mean, I thought it was acceptable.

MEMBER BLEY: So you are also saying any applicant could go to 1.174 and make the argument.

CHAIR STETKAR: Well, but you're saying, if that's true, why not build it in here? If that's true, why not build it in here and at least provide some coherence in terms of what that sliding scale is?

Because if I'm now applicant X with a completely different design -- let's not get into ESBWR or any of the current, I might say, "Well, okay. I'm going to use a Fussell-Vesely importance of 7 and a risk achievement worth of 600."

And will the staff then now need to evaluate on a per-case basis each of those numbers to

determine whether they're applicable or should there be some general guidance, regulatory guidance, saying, "Here is the sort of sliding scale that the staff deems appropriate"? What are those risk-significant values as a function of baseline CDF and LERF?

MR. DUBE: Good points. We have this competition here between this statement, which reaffirms the existing goals and metrics and the bottom bullet, which says, "Okay. Enhanced margins that ESBWR has should be available for operational flexibility, design flexibility." And so this would be where this competition is, if you will.

There is nothing in the regulation that says risk achievement this, Fussell-Vesely this. It's in the guidance. And guidance is guidance, which is we start from this framework and the expression is license applicant proposes and staff disposes.

So I'm not sure, number one, we have to tie the hands of the applicants. Number two, as Todd mentioned, there is this guidance in the reg guide 1.174 that allows some variation depending on the baseline core damage frequency.

I mean, I can say from our old charter, which is to look at the processes, the risk-informed processes, and say, because of this, is there

potential for significant decrease in the enhanced level of safety of any particular plant, ESBWR, because they use thresholds that are a factor of two different than some of the other plants, active plants, when their baseline core damage frequency is at least an order of magnitude different? I mean, the answer to that is staff feels comfortable that there is no decrease enhancing level of safety just because of different thresholds.

CHAIR STETKAR: Suppose the baseline core damage frequency is 10-6.

MR. DUBE: And?

CHAIR STETKAR: Suppose it was 10-6, still well below currently operating plants but a factor 100 higher, and they came in with their risk significance values. Is that okay?

MR. DUBE: We would take a look at it. And it might not be.

MEMBER CORRADINI: But your answer, then, implies what John was asking before, which is you are okay in this case, but until somebody else brings you a different RAW, you are not sure you will be okay in the next case. That's what I just heard.

MR. HILSMEIER: Yes, I agree with that.

MEMBER CORRADINI: So, I mean, maybe John

has brainwashed me here, but I guess that strikes me as develops a level of uncertainty for the incoming applicants that is necessary. Why is that necessary? Why is that? It seems to me in some sense you have ad hoc developed a logic which I personally like. Why not codify it and explain it to create certainty and some understanding in the industry?

CHAIR STETKAR: Such that if I come in with 3 times 10-6, I at least know that okay. Here is roughly the range of values that I should interpret as staff's knowledge of risk-significant.

MEMBER CORRADINI: Right. I know we brought John over on this one because I don't remember him being so happy during the ESBWR times. But I think your explanation to me makes sense. Then that leads to the fact that -- and you actually have developed a thinking philosophy that ought to be -- I'll use the word "codified" but at least explained so industry knows going in what to expect.

MR. HILSMEIER: Yes. In my personal opinion because under maintenance rule, there is a risk significance methodology, methodology for defining risk significance under 50.69 space. There is a slightly different methodology for defining risk significance.

CHAIR STETKAR: In our guidance, there are different numbers.

MEMBER CORRADINI: John has educated us on all of this.

MR. HILSMEIER: Right. In my personal opinion, since risk significance is used so much, it would be nice to have a general guidance document to define risk significance.

Another point I want to make is per the design centers with very low CDF and LRF, they use a RAW of two, Fussell-Vesely of .005. And their risk-significant list is very, very large. And it may mask the really important risk-significant SSCs.

MEMBER SHACK: So, on the other hand, it sort of puts an importance on the bottom line numbers that could be -- would Tippen Point, for example, with 10-7 CDF suddenly qualify for a 50.69 process equivalent to an AP1000?

You know, the industry is willing to live with these numbers for the maintenance rule in 50.69. I would be a little reluctant to let the staff deal with them on a case-by-case basis, but you could make --

CHAIR STETKAR: On the other hand, Bill, this is for a risk-informed application. You ought to

have a PRA that satisfies the quality requirements of reg guide 1.200. So if the PRA spoke level of detail, quality, data, yadda yadda yadda, satisfies those requirements, then, you know, if that's the number, that's the number.

I'm skeptical, I'm always skeptical, of small numbers. And if they don't quantify uncertainty, there is real reason for skepticism. But as long as they can show the PRA doesn't have any fundamental flaws in it, then you ought to use it as a quantification tool. And that takes the metrics out of it.

If you use a meter stick to try to measure fractions of a millimeter and the scale was not all that well-developed, you ought not to use that scale.

MR. DUBE: One final word on -- Biff Bradley was --

MR. BRADLEY: Biff Bradley.

Just a very minor point, a clarification. What we have been talking about is the RAP, not 50.69. ESBWR is not implementing 50.69. I understand the philosophy is the same. It's not directly -- there is no operating plant RAP. So this isn't a new plant versus old plant guidance issue.

This is a little different from 50.69. I

would expect if a new plant were to implement 50.69, they would get the same importance measures that we have in the guidance for the operating plant. It wouldn't be different, just to clarify.

CHAIR STETKAR: Well, Biff, be careful because I have raised this question with one of the COLA applicants that as they transition, for example, from the certified design into their maintenance rule program, you know, are they going to then apply the operating types of numerical values and suddenly expand the list of equipment that's under the maintenance rule compared to that set of equipment that was in the certified design using the different methods. And that can result in a discontinuity as you go from the design certification to the operating programs.

MR. BRADLEY: It may. I don't think there's anything in our maintenance rule guidance that lets you go out of the normal factors for those importance measures.

CHAIR STETKAR: There isn't at the moment.

MR. BRADLEY: Right. And certainly 50.69, the importance measures are just a minor part. I mean, there's like --

CHAIR STETKAR: Yes.

MR. BRADLEY: -- 100 pages of guidance. So they don't know how to categorize. But it's not my expectation that new plants would use different values than the ones that we have listed there. If they did, I think they would have to come in and make that case.

CHAIR STETKAR: But, I mean, you do get this kind of philosophical difference that those four categories, although we're particularly looking at 50.69 in the slides here, risk-significant things under 50.69, the argument has already been made by the staff that RTNSS or DRAP looks like RISC-2 under 50.69. So I don't understand numerical discontinuities simply if we're talking about 50.69 for a new plant or RTNSS or DRAP for --

MR. BRADLEY: There would be other discontinuities because RTNSS or DRAP are based solely on those importance measures. And 50.69 has got 100 other things we have got to consider. So you are going to have all of these qualitative considerations, EID and other things, that are different from the criteria for RAP.

CHAIR STETKAR: I thought at least DRAP goes through the --

MR. BRADLEY: It may have other criteria.

CHAIR STETKAR: -- qualitative -- I'm just

saying decision-making. I'm not sure about the --

MR. DUBE: It is not as formal as regulatory requirement.

CHAIR STETKAR: I know we have talked to applicants. And they claim that they're applying that type of qualitative panel. So, again, it's sort of philosophically the same, perhaps not as well-defined.

MEMBER BLEY: It hasn't come up really operational yet.

CHAIR STETKAR: No, it hasn't. Well, that's one of the reasons for raising questions now, to get a bit ahead of the curve.

MR. DUBE: So, in conclusion --

(Laughter.)

MR. DUBE: Remember, our charter was not to solve world hunger, but --

MEMBER CORRADINI: It wasn't? It wasn't?

MR. DUBE: -- a series of tabletops that test existing guidance specifically, though, to identify where there was the potential for significant decrease in the enhanced level of safety as a result of new reactor risk profiles.

And the staff reached a conclusion on 50.69 that based on the limited sizes and the limited comparison with the ones that result from South Texas

1 and 2, that there are no gaps, there are no considerations other than risk that factor into the categorization. We're showing a diagram where considerations impact on design basis accident analysis, defense-in-depth, margin of safety taken into consideration.

And specifically with the concern being on RISC-3, which are the safety-related but low safety significance, it's actually codified in regulation what has to be considered in terms of performance monitoring, addressing any failures in the Corrective Action Program, and taking appropriate action that, specifically for 50.69 stuff, is not performing any changes to the guidance.

MEMBER BLEY: Let me go back to world hunger just a second. The arguments we have been making around the table about we're talking the specificity on how this so-called sliding scale ought to work for component performance strikes me as one of those places where suddenly we get every different aspect of regulation gets its own set of guidance with its own sliding scales. Wouldn't the appropriate place if you want to see this more formalized be back at 1.174 --

CHAIR STETKAR: Sure.

MEMBER BLEY: -- so that we don't have it scattered all over the place?

CHAIR STETKAR: Sure.

MEMBER BLEY: That would just be worrisome.

CHAIR STETKAR: You know, I certainly think that's to do it as long as everything else points to 1.174.

MR. DUBE: And they all do pretty much, yes.

CHAIR STETKAR: They all do pretty much. So yes.

MR. DUBE: I mean, it's the --

CHAIR STETKAR: That would be the fundamental guidance doctrine, I suppose.

MEMBER BLEY: I just didn't want anyone leaving here thinking we want to see this guidance fragmented.

CHAIR STETKAR: No. I mean, I think that is why we are where we are is that we have different guidance for different focused purposes and numbers have evolved --

MEMBER BLEY: Right.

CHAIR STETKAR: -- for each of those sets of guidance. And in many cases, the numbers are

generally consistent. But in a few cases, there is room for different interpretations.

MR. DUBE: I understand your point. It is well-taken. From that viewpoint, maybe there is the possibility of making some enhancements to the guidance, but from the viewpoint of is a factor of two on the threshold for risk achievement work going when applied to the ESBWR going to result in a significant decrease in the enhanced level of safety of the design? I would say no.

CHAIR STETKAR: Yes. I understand. I don't think any of us are arguing -- you know, at that level of specificity, --

MR. DUBE: Yes.

CHAIR STETKAR: -- we are looking at sort of consistency in the whole process as it's applied, current operating reactors in the context of 50.69, in principle, to new reactors if they were going to adopt 50.69 or to new design certifications for reactors that we haven't even thought about yet coming forward that might have wildly different core damage frequencies, different risk contributors, things like that -- we sort of have a shot at working on this problem now.

MR. DUBE: That's a lot of what I had to

say on 50.69. That's as much as it --

CHAIR STETKAR: You didn't think you were going to get out of saying less, though, did you?

MR. DUBE: Any other comments from any of the members on 50.69? No? Okay.

MEMBER SKILLMAN: I do. On your Slide 9, please, where do the fire systems reside?

MR. DUBE: Fire protection system, you usually have the wrong quality assurance program because they're not credited in a chapter 15 accident analysis, for example. I think it's fair to say that they're in the right-most column. So it's either a two or a four, I would think.

MR. BRADLEY: That is correct. It's typically important to quality. So they can be RISC-1-ish or RISC-2-ish.

I think with the fire PRAs now that we are having to live with, you are going to see on plants like Vogtle some fire SSCs show up in RISC-2.

We're actually struggling a little bit with Vogtle right now because of what we believe are conservatisms in the fire PRA. And the impact on the denominator as well for the importance measures is causing us some concern right now. But fire systems typically right now are not safety-related. They're

important to safety.

CHAIR STETKAR: And I believe, Biff, for new plants, when you look at DRAP -- and I know you remember these things better than I do. Fire protection systems are explicitly called out as one of the sets of systems that you need to look at for DRAP, aren't they? I mean, there are those. Are they or are they not?

MR. BRADLEY: That I can't --

CHAIR STETKAR: Well, there's a set of -- station blackout, ATWS, and fire protection I think are called out explicitly.

MR. BRADLEY: That's extremely dangerous.

CHAIR STETKAR: I was hoping you would remember because it's been --

MR. DUBE: Todd Hilsmeier might be able to answer with a great degree of confidence.

CHAIR STETKAR: Todd, I know it's not 50.69, but, again, in the sense that this box sort of seems similar.

MR. HILSMEIER: Yes. My name is Todd Hilsmeier again.

Yes. The fire water pumps are generating DRAP. It depends on the design in DRAP for alternate injection into the reactor vessel. I'm thinking for

like STP --

MR. DUBE: Three and four.

CHAIR STETKAR: Right.

MR. HILSMEIER: Yes, three and four.

MR. DUBE: ABWR.

MR. HILSMEIER: Yes.

CHAIR STETKAR: But not necessarily for the fire protection functions? I can't honestly recall. This is not a leading question.

MR. HILSMEIER: Right.

CHAIR STETKAR: This is trying to see if we can get a little more information into Dick's question for at least new plants.

MR. HILSMEIER: It's a fire PRA model of credits. It would like --

CHAIR STETKAR: Yes. That certainly would be pretty clear.

MR. HILSMEIER: And also, I mean, it depends on the expert panel. The expert panel --

CHAIR STETKAR: Yes.

MR. HILSMEIER: -- may also add the fire protection system for fire suppression. I think, for one, the design centers that I'm reviewing, they included fire protection pumps for fire suppression.

CHAIR STETKAR: Okay. Thanks.

MEMBER SKILLMAN: Biff, if I could ask you, based on a comment that you made just a minute ago that there was some difficulty in determining the categorization of the fire equipment, would you expand on that a little bit? Is this an economic consideration or is this a philosophical consideration?

MR. BRADLEY: It is not really an economic consideration. We have briefed this Subcommittee previously over the last couple of years about some concerns we have about conservative bias in fire PRA. Some of this is a state of knowledge kind of challenge. Other is some regulatory elements that desire conservatism in these models for the 805 applications.

So we believe we developed fire PRAs that may have some bias with respect to the other model we're using; for instance, internal events at Vogtle for categorization. So we want to make sure we can properly account for that bias and that is not negatively impacting the categorization result.

And we didn't anticipate I think when we wrote 00-04 that -- this was one problem that we didn't explicitly address in the guidance. And now we're coming back to look at that.

But we understand for the 805 application the staff desires conservatism directly in the PRA for various reasons, but when you try to use that PRA for other purposes, especially when you're comparing it with your internal events risk and trying to make a decision based on that combined risk, then it gets a little more challenging. And that's what we're dealing with right now at Vogtle. That's what I was referring to.

MEMBER SKILLMAN: Thank you, Biff.

MR. BRADLEY: Sure.

MEMBER SKILLMAN: Thank you.

CHAIR STETKAR: Any other questions or comments among the members for 50.69?

(No response.)

CHAIR STETKAR: If not, we're going to take a break. So we will recess until 10:15.

(Whereupon, the foregoing matter went off the record at 9:59 a.m. and went back on the record at 10:15 a.m.)

CHAIR STETKAR: We are back in session. Let's hear about 1.174.

MR. DUBE: Okay. We had an interesting workshop, tabletop exercises on October 5th on 1.174. And I'll just at a high level discuss what we did and

what was the outcome.

Recollect there are five basic principles in reg guide 1.174. This is the risk-informed license amendment request. And specifically these five principles, as stated in reg guide 1.174, are generally applicable to all risk-informed initiatives. The proposed change meets regulations unless it's specifically asking for an exemption consistent with the defense-in-depth philosophy.

And, as an aside, reg guide 1.174 -- and you'll probably see it -- has been revised. It's going through the final concurrence process. But there was a task to provide more discussion in a hierarchical delineation of defense-in-depth and with examples. So that is being beefed up.

Maintains sufficient safety margins, results in an increase in CDF or risk that is small and consistent with the Commission's Safety goal policy statement, then monitor the performance going forward.

Again, these are the risk-informed acceptance guidelines; whereas, on the upper right is the core damage frequency. On the lower left is large early release frequency.

And the x-axes of both are the baseline

core damage frequency representing theoretically the total core damage frequency as much as it is quantified as possible. On the y-axes are as a result of the change, some measure of the change of core damage frequency that might result as a result of the proposed implementation of the license amendment.

Region I are regions where the no changes would be allowed. Region II says, "Well, you know, they're allowed, but there should be a good, compelling reason why." And region III and below is where most of the change historically has taken place, the license amendments have been proposed.

Now, again, early on in this project we looked at what options might there be should new reactors have different thresholds for regions I, II, and III. And, of course, the Commission SRM said no. So this plot is not going to change, certainly not as a result of our response to the SRM.

So what we did do in the tabletop is we hypothesized eight different cases, exercising pretty much the full range of the new reactor designs. And they were hypothetical but reasonable. They were hypothetical in the sense of what if the ESBWR proposed the change that would have this kind of an impact on core damage frequency or large release

frequency, large early release frequency.

And we weren't so much interested in the numerical value and where it was on this plot so much as what is the staff's thought process, what has it been for the current fleet, and then if we carry this over to the new fleet, what would be the staff's process and what would be some considerations that might come into play to be sure that the enhanced level of safety of the new design was not weakened.

And so I view these guidelines or filing in the region III or typically the low range of region II as necessary but not sufficient conditions. In other words, those principles are a series of principles. These are and between typically and not or. So the risk change can be small, but once there, they have to maintain defense-in-depth and have to maintain safety margins.

So we did a number of exercises. I'm not going to go through the eight cases that would have been in your background material that was handed out. I will take one example here. On the ABWR. Again, this is just a hypothetical case to exercise the thought process on how the staff and industry would respond to this kind of a change.

So here in the ABWR, we are proposing

hypothetically that a change was proposed to the containment over-pressurization system, which is the wetwell venting, to reduce the rupture disk setpoint from 104 psia to some lower value and change 2 isolation valves from normally open to normally closed with operator action required for venting.

So in the current design that has been certified, there are two isolation valves. But they are normally open. And they are too isolated in the event that decision was made, for whatever reason, that needs to stop making releases.

Basically it's a passive design with actually two ruptured disks: one ruptured disk at 104 psia. And the second is just there against protection against pigeons coming into the vent line that's going out to the atmosphere. And it's entirely passive in nature. And there's nothing right now as it's currently designed the operator needs to do.

We're just hypothesizing what if a change were made, for whatever reason, and it had this kind of order of magnitude.

CHAIR STETKAR: Don, do you happen to know off the top of your head, just out of curiosity, what percentage change that was in core damage frequency? Do you remember what the baseline --

MR. DUBE: Well, this is -- and this is roughly kind of a number that it might result from this of a change.

CHAIR STETKAR: I understand. Were you starting from 10-3 or --

MR. DUBE: Oh, the internal events core damage frequency of the ABWR is somewhere in the 3 to 4 times 10-7 range.

CHAIR STETKAR: So it's about 25 percent roughly. Okay. Thanks.

MR. DUBE: So it would lie right here at the bottom level of region III. In fact, you know, 10-7 here for a reason, but it would be -- and vertically it would be right at the bottom of that graph.

And I show two X's there. And that's because since we don't have a seismic PRA for ABWR, I hypothesized a range of what the seismic CDF might be. And I show a range of what is the baseline CDF might be, plus or minus about an order of magnitude.

So this was a good example where the risk is small. It's in the lower region, lower end of region III. So in terms of the principles, would one conclude that the risk increase is small? The answer is yes. But those are the factors, such as

defense-in-depth.

And one of the elements of defense-in-depth is does one possibly defeat or exchange a passive function with an active function? And, therefore, a situation where there was essentially very little operator error unless one assumes operator error of commission to isolater, but aside from that, to substitute a passive function with an active function is not in keeping with one of the elements of defense-in-depth. And this is a situation probably where unless there was an extremely strong compelling reason, staff probably for defense-in-depth reasons would not approve this license amendment.

MEMBER SKILLMAN: So this is just a hypothetical situation.

MR. DUBE: Yes, no one is claiming --

MEMBER SKILLMAN: All right. Okay. Thanks.

MR. DUBE: And all of these eight examples were that, and some people got nervous. We wouldn't do that, but it wasn't so much to, as I said, in all of these examples, to, say, "Is it realistic?" and "Would someone do that?" It is to exercise the thought process. So in all of these examples, we exercise a number of considerations.

Am I up to here already?

CHAIR STETKAR: Don't feel bad about getting ahead of the curve.

MR. DUBE: Okay. So in all the examples, we observed the change in core damage frequency risk to be low, but what we saw in all of these examples is that there often were these other principles that would come into play.

And, again, I will emphasize small risk increase is necessary but not sufficient. In a lot of these examples that we hypothesize, there would be everything from substituting active function for what was previously passive or degrading defense-in-depth and that possibly removing equipment from service that provides a defense-in-depth for high-pressure injection, low-pressure injection, or loss-of-feedwater events might come into play.

And in all of the examples that we went through, we saw the theme where defense-in-depth really had an active role in all of these considerations and under the staff's current reasoning would prevent a number of changes that based on just risk numbers are quite small but defense-in-depth played an important role.

MEMBER ARMIJO: Don, could you go back to

your example on the ABWR change? I would want to understand exactly the reasons why that particular change would not have been accepted. If you take two active valves which are normally and then you --

CHAIR STETKAR: Close them and make an operator --

MEMBER ARMIJO: You have to have an operator action. So you --

MR. DUBE: To open those two valves.

MEMBER SHACK: So he could fail to do it.

MEMBER ARMIJO: Okay. Okay. Okay. I read it the other way around. They were normally closed. Okay. So in this case, you would have to have an active --

MR. DUBE: Right. You are relying on -- you know, the design certification and reflecting the combined license application was reviewed by the staff. And the design was certified based on passive heat removal function as a last, one of the last, measures and substituting now requiring operator action for what was previously approved passive would probably get very close scrutiny by the staff.

MEMBER ARMIJO: But it wasn't related to the reduction in the setpoint on the disk rupture.

MR. DUBE: No.

MEMBER ARMIJO: It was related to the valve --

MR. DUBE: Yes.

MEMBER ARMIJO: -- being normally open and all this?

MR. DUBE: Right.

MEMBER ARMIJO: Okay. I understand it now. Thanks.

MR. DUBE: Thank you.

So we did these eight exercises and then looked at 1.174 in general. And I know, aside from the issue of reg guide 1.174 should have some guidance on risk importance measures, the staff concluded that there were no gaps in the reg guide 1.174 from the viewpoint of line tape in reg guide 1.174 in those five principles that have been used quite successfully for current operators for current reactors and apply them to new reactors. And the answer was yes, the staff does not feel that using the existing guidance and applying it, considering those five principles, would result in a decrease in enhanced level of safety of the new reactor designs.

CHAIR STETKAR: And, Don, was that conclusion basically uniform for your eight examples in terms of the staff's judgment regarding

defense-in-depth or some of those other qualitative measures being the -- I don't want to call it the backstop that would --

MR. DUBE: Yes. I mean, I think we felt comfortable in all of those examples.

MEMBER ARMIJO: If it wouldn't take too much time, do you have an example that you accepted as a backup slide, one of these changes, just to calibrate me on --

MR. DUBE: You know what?

MEMBER ARMIJO: Even if you don't have a slide, maybe you could just describe it.

MR. DUBE: Well --

CHAIR STETKAR: I don't immediately recall any.

MEMBER ARMIJO: Like I said, I didn't see any.

MR. DUBE: Either enclosure 3 or enclosure 4.

PARTICIPANT: It's not 4. Four was the ROP.

CHAIR STETKAR: Four is the ROP. It's not in those enclosures, Don.

MR. DUBE: You don't think so?

CHAIR STETKAR: Well -- oh, wait a minute.

MR. DUBE: I believe it was that --

CHAIR STETKAR: I think it was in the ROP because it was rolled up in that same meeting, wasn't it?

MR. DUBE: It was in the material that you had.

CHAIR STETKAR: Yes. It was in the material that we had, but I have renamed all of my files.

MR. DUBE: Oh, great. We're on the wrong --

CHAIR STETKAR: Well, you can rotate. Here it is.

MEMBER SIEBER: Either way down or --

MR. DUBE: Change this. Rotate clockwise.

CHAIR STETKAR: Just keep that.

MR. DUBE: Clockwise. So this is the handout. This was the activity.

CHAIR STETKAR: And we had this material.

MR. DUBE: Yes, yes.

CHAIR STETKAR: So you can read it.

MR. DUBE: This might be --

MEMBER ARMIJO: Okay. So here is an --

MR. DUBE: This might be an example that we -- this was actually pretty close to an actual

example because early in the design phase because of design basis accident considerations and concern of what happens if there were a steamline break inside containment and the impact on containment performance and the need to isolate feedwater. This was actually an early change that some of you probably are familiar with.

The PRA influenced the actual modification that was implemented. After they fine-tuned the change, the change resulted in a core damage frequency change of the order of 10-8.

I note here that the core damage frequency for all the quantified events and molds is around 10-7. This does not include seismic. I assumed this so we could get a baseline case.

We went through the thought process. And I think for this particular change because what they did was compensate for the need to isolate it by providing a means of bypassing the isolation of feedwater because the feedwater system provided some defense-in-depth for decay heat removal.

MEMBER ARMIJO: They didn't substitute an active system for --

MR. DUBE: Yes. It was in many ways, in an actual event, you've got to cut back on feedwater

or, else, you are grossly overflowing the reactor vessel. So it was more or less an actor for an actor. It was not an actual license amendment request, but we used it as an example of what if --

MEMBER ARMIJO: Because this happened during the certification.

MR. DUBE: Yes. This happened during the certification.

MEMBER ARMIJO: Okay.

MR. DUBE: So this would have been an example of where we felt comfortable with the change.

MEMBER ARMIJO: Okay. That kind of calibrates me.

MR. DUBE: Good.

MEMBER ARMIJO: Thank you.

MR. DUBE: Well, where am I?

MEMBER SIEBER: Back to slide 21. If you go back, let me ask you a question on the previous one. You talk about changing from passive to active, which means operator action. Do you take into consider over time how many new operator actions you add and what effect that has on the ability of the operator to perform them without making errors?

MR. DUBE: Well, I mean, that should be a consideration.

MEMBER SIEBER: It should be. The question is, is it explicitly a consideration? Because I can imagine as plant operation continues over the years to make changes to move passive systems to active systems because things don't work exactly the way you want them to. And to me, you reach a point where the operator just can't understand or handle a whole bunch of things --

MR. DUBE: I understand why.

MEMBER SIEBER: -- that require his manual actions.

MR. DUBE: It should be reflected in the PRA. And the PRA is required -- for new reactors, it is actually a requirement by regulation to be updated and maintained and meet the standards that have been endorsed by the staff. And the ASME/ANS standard has a series of high-level and implementing requirements on the human reliability analysis, including taking into account dependencies across operator action. The fact is that if you have several operator actions that are not independent of each other; that is, one operator action may be dependent on the other and how you quantify it.

So the long and short of your answer is the PRA that's maintained and maintained to the

standards has to take that into account. And so if a large number of dependencies of operator actions eventually manifested themselves in exchange, it should be captured by the PRA.

MEMBER SIEBER: Should be. And new regulations require PRA to be up-to-date --

MR. DUBE: Yes.

MEMBER SIEBER: -- periodically? If so, how often --

MR. DUBE: They have to maintain it consistent with the degree of the change to reflect current design and operational experience. No less frequent than every four years they have to do an --

MEMBER SIEBER: You can do a lot of changes in four years.

MR. DUBE: Yes, but they typically accumulate. I mean, the way it's done is you have a record and maintain a running list. And most licensees have a threshold. You do a back-of-the-envelope calculation. And there's a threshold that if a core damage frequency might change by some amount, I know at one particular licensee that that is ten percent CDF. Then an update is going to be required by procedure.

MEMBER SIEBER: Yes.

MR. DUBE: I mean, I used to do that. You maintain a running list of what changes need to be incorporated at the next update.

MEMBER SIEBER: I just want to make sure all licensees do it.

MR. DUBE: Well, for new reactors, it's --

MEMBER SIEBER: What you're describing is the responsible thing to do.

MR. DUBE: It's a requirement.

MEMBER SIEBER: Yes, for four years, long tim.

MR. DUBE: That's true.

MEMBER SIEBER: Okay. Thank you.

MR. DUBE: So any questions on reg guide 1.174? We made up some time.

(No response.)

4. LRF TO LERF TRANSITION

MR. DUBE: The next topic is transition from large release frequency to large early release frequency. This is not in keeping with the other exercises in the sense of, you know, this is not really a risk-informed activity, but it's a longstanding issue, which is -- you know, the new reactors have been licensed using large release frequency as a metric. And current operating fleet

use large early release frequency in all of their risk-informed applications and acceptance guidance and the reactor oversight process.

And we're long overdue on addressing -- since the Commission directed us to use the same risk metrics, the fact that new reactors have been using large release frequency, how does one transition to large early release frequency?

So we took it up as an exercise to look at various options that one might use. And I'll go through the options and staff's recommendations.

Basically a real quick history. The Commission goals for new reactors, as you might recall, are based on a conditional containment failure probability of less than 0.1, large release frequency of less than 10-6 and a 10-4 per year for core damage frequency. Whereas, operating reactors use core damage frequency and large early release frequency as the primary risk metrics, those are used in reactor oversight process in reg guide 1.174.

So some of the issues with continuing use of large release frequency is -- well, issue number one is these have not been defined by the staff. And each design center has chosen a different definition.

Now, for the purpose of the staff's review

of the design certification application, the COL application, the determination of whether they met the Commission's objectives, staff's been able to review it, notwithstanding not having a universal definition because all of the design certifications have chosen very conservative definitions of a large release.

For example, ESBWR says anything more than tech spec leakage through containment is a live release. And that's sort of by orders of magnitude.

MEMBER ARMIJO: Yes. You can't do better than that.

MR. DUBE: And, correspondingly, others have used any release resulting in more than 25 rem at site boundary as a --

MEMBER CORRADINI: Can you give me a few other, can you repeat that example with a few other, examples that show the other extreme? I mean, you once did this I can't remember how many presentations ago.

MR. DUBE: Right. Okay.

MEMBER CORRADINI: It was a good illustration, but the range of extremes are anywhere from anything above tech specs to --

MR. DUBE: Well, the worst, I mean, there's exposure-based guidance, like 25 rem at the

site boundary, like 0.5 or 0.6 miles, like a kilometer or something like that.

MEMBER CORRADINI: Okay. Fine. That's the other extreme.

MR. DUBE: And others have used a containment failure mode-based definition of large release being any kind of containment failure: Containment bypass, containment isolation failure, or any kind of large-scale containment failure other than like basemat leakage. Some have even included basemat leakage as part of a definition of large release. So they generally have been pretty conservative.

MEMBER CORRADINI: Don, the reason I ask that under your first bullet is -- oh, no. Let me just wait. I'm sorry. Excuse me.

MR. DUBE: Yes.

MEMBER CORRADINI: I'm sorry.

MR. DUBE: Those are some of the issues. So for the purposes of certifying a design and issuing a combined license and meeting the Commission's objectives, the fact that there is no universal definition has not been an impediment, if you will, to licensing requirements.

But, going forward, there are issues because the ASME/ANS level 1 PRA standard, which the

staff endorsed, uses large early release frequency. And it's used, large early release frequency is used, in risk-informed guidance, including reg guide 1.174, and all the subsidiary guidance and the reactor oversight process. So we add this discrepancy.

And then also no existing or proposed level 2 PRA standard -- and I have reviewed the level 2 PRA standard -- provides a universal definition of large release or large release frequency.

So we've got this dilemma of do we perpetuate large release frequency going forward or do we say at some point since the Commission directed us, you know, move from one metric to another.

MEMBER ARMIJO: Don, just to make sure I understand, for the ESBWR, does that mean LERF and LRF are the same since it's anything other than this leakage?

MR. DUBE: No. I wouldn't say they are the same, no.

MEMBER ARMIJO: Then somewhere along the line, you are going to have to explain what is going on.

MR. DUBE: There's guidance in reg guide 1.174 on what has traditionally been used as large early release frequency. There is also guidance in

the ASME/ANS standard.

CHAIR STETKAR: Well, but I think what Sam is asking is LERF in the particular application of what ESBWR has called LERF is a subset of that.

MEMBER ARMIJO: It should be a precursor to LRF.

CHAIR STETKAR: No. It's a subset of it, right?

MR. DUBE: You would think it's a subset, but about three years ago, Doug True from Erin Engineering gave a presentation and says --

PARTICIPANT: We had a long talk about it.

MR. DUBE: -- in fact, it ends up being, LRF ends up being, a subset of LERF.

PARTICIPANT: We will calculate that.

MEMBER CORRADINI: What?

MR. DUBE: Yes.

CHAIR STETKAR: Even if they have defined it as any leakage above tech specs as being LRF?

MR. DUBE: Well, I can't speak on a specific basis, but they use some calculations from NUREG-1150 and showed that it was contrary to thought process.

CHAIR STETKAR: That's true for the way NUREG-1150 defined things, but Sam was asking in the

1 particular context of the ESBWR, which, as you defined
2 it --
3 MR. DUBE: You should think LERF would be
4 a subset, yes.
5 CHAIR STETKAR: As any of the greater than
6 tech specs. It's different from I think the example
7 --
8 MR. DUBE: Right. You would take it --
9 MEMBER BLEY: It took us several hours to
10 try to figure out what was going on.
11 MR. DUBE: Yes.
12 CHAIR STETKAR: Yes, it is convoluted at
13 best, but I think we identified the fact that there
14 could be real subtle anomalies in some cases. But in
15 this particular example, I can't see how that would
16 apply.
17 MEMBER CORRADINI: Yes. It should. But
18 can I just --
19 CHAIR STETKAR: Other things, though,
20 where people have defined it perhaps at 25 rem at some
21 other context, what is large, then there might not
22 necessarily be a --
23 MEMBER CORRADINI: Correct. But it is a
24 function of what John is saying. The definition of it
25 if they're inconsistent, one could be encapsulated in

the other, but if you think of it from a logical standpoint and you want to define it such that LRF would encompass.

CHAIR STETKAR: In principle, that would make sense. That's right.

MEMBER SCHULTZ: So, Don, clearly these are distinctive and very important issues. These have not yet been defined by the staff.

MR. DUBE: Right.

MEMBER SCHULTZ: But of the issues: the large release frequency as well as the conditional containment failure probability.

MR. DUBE: Right.

MEMBER SCHULTZ: At least not defined consistently because the next bullet says everyone seems to have a different definition.

MR. DUBE: Everyone does.

MEMBER SCHULTZ: Thank you.

CHAIR STETKAR: That makes it interesting.

MEMBER CORRADINI: I am trying to understand, though, that if I were out there, let's just say for the moment that there is some logical consistency. Get across that one. I'm sure you guys will figure that out. To satisfy it because of the way it's defined, you could essentially improve upon

prevention and not have to prove that you have a containment system and associated gear that would essentially because -- I mean, one way to say that is that the -- I'm just looking at your first line -- is that I just reduced the core damage frequency enough that I still at least get a ten percent chance of, no more than ten percent chance of, failure of the containment system.

MEMBER SHACK: Conditional containment.

MEMBER CORRADINI: It's conditional, conditional. I understand.

MEMBER SHACK: Yes. So it doesn't matter how much you --

CHAIR STETKAR: It doesn't make any difference what the absolute core damage frequency is.

MR. DUBE: There's qualifiers on this, by the way, because the way he --

MEMBER CORRADINI: I am trying to understand all the games I can play with this.

MR. DUBE: Yes.

MEMBER CORRADINI: That's what I'm trying to get at.

MR. DUBE: Because the way many have defined conditional containment probability, they have taken large release frequency into the numerator and

divided it by core damage frequency in the denominator or is that -- but yes. Large release frequency in the numerator --

CHAIR STETKAR: Yes.

MR. DUBE: -- and core damage frequency in the denominator and showed that that is less than .1. You could be in a situation where you reduce core damage frequency, which is a good thing. You are right. You reduce the denominator. And CCFP goes up.

CHAIR STETKAR: Well, it depends on the sequences that are going into containment failure.

MR. DUBE: Yes, yes. And so if you have less than .1 and doing a good thing, which is reduce core damage frequency, make CCFP go to .1. Is that a reason not to approve a change? The answer is no.

CHAIR STETKAR: Don, for those, just out of curiosity -- and we've got to be a little bit careful about time.

MR. DUBE: Yes.

CHAIR STETKAR: The example you just mentioned was suppose I define large release frequency in whatever way I want to define it, some way of defining it. That is less than all of the sequences that result in any release. So, for example, I define large release frequency as some subset of sequences

that result in an off-site release because of some criteria I set on size or something. But there are sequences that have failure to isolate small lines that I have not included in what I have defined as large.

Is the notion of conditional containment failure probability only restricted to apply to something that's defined as large or is it this -- okay.

So your example, if I have cleverly defined what I call large, will underestimate conditional containment failure probability if that actually means anything that results in a release from the containment, small releases, whatever small is?

MR. DUBE: I can't think of any situation where CCFP is not conservative.

CHAIR STETKAR: Okay. But just in practice and what has been done so far?

MEMBER CORRADINI: By the time I -- it strikes me -- again, you guys will figure this out, but it strikes me that the staff ought to come up with a consistent definition

CHAIR STETKAR: I think if Don gets further, you'll see how they are --

MEMBER CORRADINI: And the definition of

LRF ought to be consistent with the definition of containment failure. You're not going to have these inconsistent if --

MR. ADER: Mike, if I can interject, it's probably time to go back. The history of LRF was LRF was really LERF. It was never called LERF. It came out of safety goal as a surrogate for the safety goal.

So all the effort we spent to try to define LRF, it was in terms of an early fatality. And at 10-6, it becomes a de facto new safety goal. So the recommendation was to terminate the effort to do that.

There were others. There was containment failure probability and 24-hour intact containments that were judged as reasonable containment performance goals.

So people get confused that LRF was intended to be a latent containment failure metric. It was always intended to be LERF.

MEMBER CORRADINI: That's fine, but the last time I paid attention to this and tried to understand it was in 1992. All right? And the Commission SRM or something -- I thought it was the Commission SRM that wrote this -- was that it was more the latter: ample time for some sort of external

action, such as evacuation, emergency preparedness, and on the order of a day or two before the containment starts releasing large amounts of radioactive material.

And that's how at least it was started. And then all of these definitions, at least in my memory, started being promulgated.

MR. ADER: The day or 2 was the 24-hour --

MEMBER CORRADINI: Correct.

MR. ADER: -- service level C.

MEMBER CORRADINI: Correct.

MR. ADER: The LERF was a surrogate for the early fatality metric in the safety goal. So if you go back and look at all of the papers that went back and forth, staff proposed it would be a release in curies that would result in an early fatality. It would be a containment failure that would result in early fatality. We tried to come up with a surrogate and equivalent curies of iodine. And where we ended up is at 10-6, it was a de facto new safety goal.

And the other constraint we had from the Commission is don't come up with a definition that became a de facto new safety rule.

So in the end, staff recommended terminating the effort because it didn't seem to be

needed at that time because we had the other containment performance metrics that achieved the purpose of having something for new reactors to review, acceptability of containment performance.

The unfortunate part is this kind of lingered out there. At 10-6 LRF, it's given staff a way to judge have the new designs provided enhanced safety, enhanced margins, enhanced severe accident protection so we can judge that. It's not been a hard criterion, but the issue now if we went back and said we should come up with a new definition, we're going back 20 --

MEMBER CORRADINI: And you're saying there's no reason to replow that ground?

MR. ADER: There are other activities I think going on in the discussions of some of the Fukushima recommendations that talk about is there a need for another metric that would deal with long-term. And those are on a different track.

MEMBER CORRADINI: Okay.

MR. ADER: So I would --

MEMBER CORRADINI: Okay. That's fine.

MR. ADER: I would clearly wait for those.

MEMBER CORRADINI: But can I just restate? Can I just restate kind of in the middle of what you

were talking about to make sure I've got it in my head?

At least in the spirit of it, staff was looking at essentially that with the new designs and with the containment systems within those designs, the chance of early fatality would be less than this metric, 10-6?

MR. ADER: We judge them are the -- you know, they have given us different definitions of LRF. And we judge that they all are below this metric.

MEMBER CORRADINI: Okay. Thank you.

MR. ADER: And on a later slide, you will see the proposal deals with the defense-in-depth issue on containment of how we are proposing to address that.

MEMBER CORRADINI: Okay.

MR. ADER: So I prefer not to go back 20 years and go through that whole effort that we concluded we couldn't come up with a definition before tied to the safety goal.

MR. FULLER: Yes. This is Ed Fuller from Severe Accident and TRA Branch in the Office of New Reactors.

I just wanted to point out a couple of practical aspects of this in terms of how we do our

new reactor reviews. First, turning to how large release frequencies are, in fact, evaluated in practice is that you have to come to grips with the fact that you have a limited number of release categories that the applicant has chosen to characterize the accident scenarios in. And when you look at those, those release categories, you invariably find that all but the one with tech spec leakage would lead to releases that one could probably pretty definitely say are large in the sense that they would be greater than 25 rem at the site boundary you have taken out. And that makes kind of a calculation.

So you pretty much direct it down to the conservative end of things when you look at what the condition of containment failure probability is. So it, in effect, is the total. The denominator is the total core damage frequency. The numerator is everything except tech spec leakage is what it comes down to.

And the other aspect, practical aspect, is that since we are looking at a variety of initiators, not just internal events of power but internal floods, internal fires, and in principle other external events, what you find is that you really can't -- and low-power shutdown events. What you find is you

really can't put low-power and shutdown events or external events into the equation because the external events are very, very site-specific. And, of course, the conditional containment failure probability for low-power and shutdown events is usually very high, if not close to unity. So you have to just throw those out.

So as a practical matter, you are left with a fairly easy way to evaluate both metrics. And, of course, LRF is not the same as large early release frequency because you're not concerned with when the release happens or if evacuation takes place or not.

All I'm trying to say is that in practice, it's not really an impediment.

CHAIR STETKAR: Okay. I don't want to hold up the discussion too much, but we do need to be cognizant of time. And I think we have to break at 12:00 because we have another --

MR. DUBE: Okay.

CHAIR STETKAR: -- meeting to attend to. And I'd like to see if we can get through the ex-vessel severe accident --

MR. DUBE: Okay. We will.

CHAIR STETKAR: -- stuff by 12:00. So if you can --

MR. DUBE: I'll speed it up.

CHAIR STETKAR: -- keep that in mind and that the members --

MR. DUBE: We came down to three --

CHAIR STETKAR: We're going to run, you know, the full afternoon anyway, but there's a lot to discuss on the ROP, --

MR. DUBE: Right.

CHAIR STETKAR: -- tabletop, and the Commission paper.

MR. DUBE: We came down to three possible options of -- I'll call them group of options. One is to continue using large release frequency and CCFP indefinitely -- well, that doesn't seem to solve it -- continue to use LRF and CCFP indefinitely but add in LERF at some point and propose initial fuel load for a number of reasons or transition from LRF to LERF at or prior to initial fuel load and then discontinue the use of LRF and CCFP thereafter.

So graphically this would be option 2A. So you have design certification COL application, a diamond at COL issuance. You have a nice diamond at Initial fuel load, which is a milestone that can use CDF and LRF and CCFP for the whole range.

In the interest of time, I won't go

through all the advantages and disadvantages, but there are a lot more disadvantages than advantages for option 2A. So we ruled that out.

Option 2B is a perturbation, which is continue to use CDF and LRF and CCFP indefinitely and add LERF at initial fuel load or perhaps a little bit prior to it depending on whether an applicant proposed, a license holder proposed risk-informed initiative.

That had advantages and disadvantages. Advantages are supports late calculation, continue to use large release frequency and CCFP, which were used in the original licensing using LERF as consistent with reg guide 1.174 in the ROP. The disadvantages are now we have two books, license holder has two books: a large release frequency book and an early release frequency book. It could be viewed as consistent with the SRM because now going forward, operating reactors only use LERF. New reactors are using both LRF and LERF. It seems to be consistent with the Commission direction. And it's an added burden on licensees.

MEMBER ARMIJO: When would you have to use LRF again at that point?

MR. DUBE: Never.

MEMBER ARMIJO: Never? Even in an amendment to a license --

MR. DUBE: No.

MEMBER ARMIJO: -- application, you would --

MR. DUBE: No. That is our option 3, our 2C.

MEMBER ARMIJO: So you would just finish with an LRF since that one is --

MR. DUBE: That's it.

MEMBER ARMIJO: They crossed the line. And we would never have to look at that parameter again.

MEMBER CORRADINI: That's 2C.

MR. DUBE: Right. That's option 2C, right here.

MEMBER CORRADINI: The graphics help.

MR. DUBE: CDF and LRF and CCFP have been used. They served their purpose, got the plant license. And now somewhere at or prior to initial fuel load, we have closed that chapter. And we say we are going to be consistent with operating reactors and just use CDF and LERF.

We choose initial fuel load for a number of reasons. First is a regulation saying that the

license holder has to have a level 1 and level 2 PRA at initial fuel load that meets NRC-endorsed consensus standards one year prior. So initial fuel loads are already on a very magic milestone.

I have a dashed line that says, well, one could transition anywhere before based on -- you know, if someone proposed some risk-informed initiative, like risk-informed tech specs, that uses LERF as a metric, one might want to have done this transition somewhat before, depending on when they submitted a license amendment request. So we're not going to say necessarily just only at initial fuel load. There can be some overlap, if you will. And that's option 2C.

The advantages, it's consistent with the SRM directions. Well, that's good. It harmonizes metrics for all operating reactors, current and new, going forward.

There are some disadvantages because the LRF and CCFP were part of the original design objective in the certification. They would no longer be tracked. You know, there is a -- you probably remember this. Six months ago there was a provision in reg guide 1.174 to look at the impact on late containment failure. And RF could be helpful in that regard, but, as Charlie Ader alluded to, you know,

staff is proposing that we add discussion along containment performance in section 2.2 of reg guide 1.174 by referring to the containment performance objectives that are in these two very important Commission papers and associated SRM, SECY-90-016 and 93-087. Those are the Commission papers that specify all of the containment performance objectives for new reactor design.

So we think we can address the issue of containment performance by making this one change to reg guide 1.174 that mentions this containment performance objective just for the new reactors that have been licensed.

And because this is existing Commission-approved policy for new reactors, we don't think we could get permission approval per the SRM to put this in reg guide 1.174. I mean, we'll mention it in the paper. And, of course, the Commission can always come back nd say, "Yes" or "No." But we think this is fair game to add to reg guide --

MEMBER CORRADINI: The fact that you explained this tells me this is the one you want.

MR. DUBE: Yes.

MEMBER CORRADINI: Okay.

MEMBER ARMIJO: Is there any way for if

you do this where something can happen or a series of things can change that would lead to undermining the licensing basis for the plant based on CDF, LRF, and CCFP, somebody come back and challenge you, say, "All the changes you have made since fuel load have undermined or violate licensing basis of this plant"? Is that a possibility?

MR. DUBE: I never say never, but, I mean --

MEMBER ARMIJO: Oh, no. I'm not asking you --

MR. DUBE: -- I think we felt comfortable again on the reg guide 1.174 exercise we talked about 20 minutes ago in all the activities that we did that there's repeatedly -- you know, RISC metrics is a necessary -- you know, load change in RISC is a necessary but not sufficient condition. And we have, you know, defense-in-depth requirements in here reg guide 1.174 long-term containment performance.

Specifically containment performance is a defense-in-depth measure that I think we feel comfortable that there is a reasonable assurance that enhanced level of safety wouldn't be degraded by this transition.

MEMBER ARMIJO: I like the idea of closing

the books and archiving it at that point as long as somebody can't come back and challenge a licensing basis.

MEMBER CORRADINI: Sam's point I think is fair. You're going to -- even though you think you -- excuse my English -- finessed it, you've only finessed it to the extent that you have defined it in a way that is consistent so that somebody can understand how you went from method of measurement 1 to method of measurement 2.

MEMBER ARMIJO: Right. And I think that's what I -- yes. That's exactly -- and the staff is comfortable that that is unlikely?

MR. DUBE: Right, and especially because of the provision, long-term containment performance, to continue to meet the containment --

MEMBER ARMIJO: By adding that at --

CHAIR STETKAR: Don, do those SECY papers have a numerical performance criterion like the ten percent CCFP?

MR. DUBE: Yes, that is where the ten percent comes from. That is where the ten percent comes from.

CHAIR STETKAR: Thank you. Thanks.

MR. DUBE: Well, but when we were looking

at containment performance objectives, we were looking more at this. We have a footnote in the Commission paper. I'll call it more the deterministic containment performance objective, which is the containment should maintain its role as a reliable, leak-tight barrier by ensuring that containment stresses do not exceed ASME level C for containment performance, so on, so forth, for concrete for approximately 24 hours following the onset of core damage under the more likely severe accident challenges. And following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products.

You might recall six or eight months ago staff created a new reg guide to address this new reactor designs. And remember we came up saying that one way of defining more likely severe accident challenges is roll-up the core damage sequences that amount to, for example, 90 percent of the sequences, which is consistent with the .1 conditional containment failure probability.

So we are not proposing that one necessarily explicitly going forward every time a change is made come back against that CCFP. We're not proposing that we continue with that. But the license

holder could continue to demonstrate that they meet this containment performance objective.

CHAIR STETKAR: I guess I am a bit confused because I read the discussion of this in the paper. And I guess I didn't understand what you just said, that although you are making reference to the SECY papers that do address some sort of quantitative conditional containment failure probability, whatever that means, you note that the level 2 PRA standard when it's released and however it is reviewed and endorsed by the staff will define releases according to release categories.

MR. DUBE: Right.

CHAIR STETKAR: And in principle, one could then provide guidance about the characteristics of the release categories that you roll up into large early release frequency or each containment failure of any form.

MR. DUBE: Each licensee will have to -- if they're going to use it, large early release frequency will have to roll up the level 2 PRA endpoints, which are release categories --

CHAIR STETKAR: Into something that they call large early release frequency.

MR. DUBE: Right.

CHAIR STETKAR: They could do a similar roll-up into something that is any containment failure for the containment, conditional containment, failure probability.

MR. DUBE: They could do that.

CHAIR STETKAR: But you're not proposing that they do that.

MR. DUBE: No.

CHAIR STETKAR: So, in principle, they could make changes that make the conditional containment failure probability for small releases 1.0. And they would never trip over a threshold.

MR. DUBE: I mean, I can't think of any realistic situation.

MEMBER SCHULTZ: But they would leave this -- this is a design criteria. And they would leave that behind upon operation.

MR. DUBE: Right. It was an original objective for licensing.

MEMBER SCHULTZ: Right.

MR. DUBE: But they still need to track large early release frequency.

MEMBER SCHULTZ: Correct.

MR. DUBE: And the goals for large early release frequency are an order of magnitude lower than

core damage frequency.

CHAIR STETKAR: But that will be -- the thing that is called large on a basis will be defined conditions. What I'm concerned about is this notion of what does conditional containment failure probability mean in the context of all of this?

MEMBER SCHULTZ: Design and operation?

CHAIR STETKAR: Design and operation. That's right. You know, can we get into a situation where I still meet the criteria for what I call large early release frequency --

MEMBER SCHULTZ: Right.

CHAIR STETKAR: -- which is a box that I have thrown things into --

MEMBER SCHULTZ: Right.

CHAIR STETKAR: -- and, yet, have a large fraction -- that's a pejorative term but a fairly highly measurable fraction of core damage frequency that doesn't give a large release, really, but has failure to isolate small containment penetrations; small late releases, if you will?

MR. DUBE: Yes. I mean, you know, the --

CHAIR STETKAR: From what I asked you earlier, you said, well, conditional containment failure probability isn't restricted to just large

early release or large release divided by total. It's any failure of the containment.

MR. DUBE: Right.

MEMBER CORRADINI: But the way this is defined is historically consistent and, similar to what Ed just said up here, which is they just take the radio of the LERF to the CCFP.

CHAIR STETKAR: That's correct.

MEMBER CORRADINI: But this wouldn't be maintained after that fact anyway. This is in a performance objective independent of all of these numbers.

CHAIR STETKAR: It depends on what you define as reliable leak-tight barrier. If reliable leak-tight barrier is only a barrier against large early releases, I understand that. If it's a barrier against any release --

MEMBER CORRADINI: That's not the case. That's not the case. I mean, the context, at least the context I -- I mean, Charlie has got the best history on this, but the context on an earlier thing was it's all relative to what would need to be done prior to a new set of plant designs and what were the expectations of -- what was the performance of the old plants coming out of the NUREG-1150? And it was

essentially that you had time to essentially effect some sort of action in those first couple of days. Do I have it approximately right?

MR. ADER: In terms of coming up with these objectives?

MEMBER CORRADINI: Yes.

MR. ADER: Yes. I mean, this was part of -- as I remember it, there were a number of lessons learned post-TMI, the severe accident policy statement, here are some challenges, you know, core spreading, containment performance long term. And these were put in. This one was put in to maintain that containment barrier for the 24 hours. This is a way to test the containment.

CHAIR STETKAR: I mean, I understand this. My only concern is, is there a gap in terms of intent of providing confidence in any releases from the containment as you transition from --

MR. DUBE: Right, but recall, too, in reg guide 1.174, they have to describe --

MR. ADER: It's whatever.

MR. DUBE: Yes. In reg guide 1.174, there's a section 2.2, "Need to discuss the impact of the change license amendment request, containment performance."

MEMBER CORRADINI: So that would give you the opportunity to --

MR. DUBE: That's an opportunity for the staff to issue a --

CHAIR STETKAR: Yes, yes.

MR. DUBE: -- request for additional information, saying, "Exactly what does that mean? What calculations, what results have you done to demonstrate that is the case? And did you impact this aspect of containment?"

MR. ADER: Don, you can correct me if I'm wrong, but what we're trying to do here, the Commission guidance was they reaffirmed the existing metrics, which would say, "Go away from 10-6 LRF."

1.174 has the part of it is you address an impact of the change on long-term containment performance. And from what I understand -- I haven't been involved in it as many of them -- that that is kind of ambiguous.

MR. DUBE: Right.

MR. ADER: What does that mean in practice? For new reactors that have done these calculations so they have a basis to refer back to so we're trying to, at least for the new designs, provide a little additional guidance of what they could use to

address that long-term containment performance because they can go back to what they have calculated when they came in regionally?

So our attempt here is to try to narrow this ambiguity in one part of the 1.174 by adding this or proposing to add this given constraints we have with 10-6. So it may not be perfect, but --

MEMBER CORRADINI: Yes. It's not perfect --

CHAIR STETKAR: And I guess I have to be careful in our time here, but is there a down side or did you think about LERF and CCFP? I've forgotten. That's not one of your mix of three.

MR. DUBE: No, that's not.

CHAIR STETKAR: That's a different --

MR. DUBE: Hybrid, yes.

CHAIR STETKAR: Did you think about that? Is there a down side to doing that?

MR. DUBE: Yes. I mean, the down side is --

CHAIR STETKAR: You have to define what CCFP means.

MR. DUBE: Yes, universal definition.

CHAIR STETKAR: Yes, but in principle, if you accept the notion that the level 2 PRA standard

will define end states and you can not kind of ambiguously put end states in a box, I can define to you what I feel is containment failure. You can look at that and say, "Well, why didn't you add this end state?" --

MR. DUBE: Right.

CHAIR STETKAR: -- or whatever.

MR. DUBE: All right. So CCFP is calculated, but it's not used in any risk-informed application, including reg guide 1.174 and the acceptance guidelines or reactor oversight process. So it's a truly academic exercise.

MR. ADER: Hey, Don --

CHAIR STETKAR: Well, but it's a quantitative measure of defense-in-depth.

MR. DUBE: Yes. And this is, too, I guess.

MR. ADER: Don?

CHAIR STETKAR: Charlie?

MR. ADER: We had had these discussions before. And I think Don had convinced me we're talking plants that have been built. Someone wants to come in and do a design change. So they have the good design change. Say they're meeting the .1 CCFP. They want to do a design change that lowers core damage

frequency that somehow they would not be able to meet the CCFP anymore.

We had some examples, which I can't think of at this time. Don may --

CHAIR STETKAR: Yes. In principle, you could think of ways that do that.

MR. ADER: I want to say I can lower CCDF, but I will trip this threshold. So I don't want to make that improvement.

CHAIR STETKAR: Yes.

MR. ADER: So at least he convinced me maybe that was not the best hard-line --

MR. DUBE: Okay. And the Commission policy statement said CCFP. I mean, that's an objective, but it should not be used to discourage accident prevention, you know. Accident mitigation should not take precedence over accident prevention. And if you're not careful, one could play that game. You're right at the border of .1.

We found that, you know, the fact that the reg guide says you have to look at the impact of containment performance and the fact that we are proposing to insert this kind of a containment performance objective, which is current policy, into the reg guide 1.174 we think meets the intent without

having all the burden of trying to track LRF and CCFP indefinitely into the future and have new reactors diverge from current reactors.

Yes. There's no perfect solution. That's why I first want to try to tackle this in 20 years. So, I mean, these are the advantages and disadvantages. I don't know what more to say. I think that's all I --

CHAIR STETKAR: Anybody else? Any members? Because we --

MEMBER BLEY: One little thing. If you wanted clarification on that, a several year ago discussion with Doug True can provide it in two sentences.

CHAIR STETKAR: That would be very useful, I suspect, for those of us who don't remember, which is me, and those of us who were here, which is a fraction of the people in the room.

MEMBER BLEY: Maybe it's four sentences. Doug organized his talk to prove his claim that everybody has got it wrong and that LRF is a subset of LERF. When we just talk conceptually, that's not right, but where it came from was some calculations Doug provided for us.

And, although these definitions weren't

quite around when -1150 was done, he went back to the -1150 results, found the scenarios that would be tagged as LERF now, calculated frequency, and then he provided a staff calculation that had calculated LRF from the results and showed that the frequency of LERF was much less than LRF.

The trick here, the problem, is that that calculation by staff of LRF from -1150 got its syllogism backwards. It calculated LRF as the frequency of one or more people dying. So it's an overestimate of LRF.

Not every LRF leads to somebody being killed, but you are right in saying if a person is killed, it is an LRF. So it is a reasonable calculation for what they were trying to show, but when you overlay those two, you get silly results. And that's because not having really killed somebody.

So it's a calculation that wasn't really the frequency of large release. It was the frequency of one or more being killed, just got language all screwed up.

MR. DUBE: Yes.

MEMBER BLEY: It took a long time to figure out what was going on that day.

MR. DUBE: Yes. The question is, do we

want to propagate this into the future?

MEMBER BLEY: Heavens, no.

(Laughter.)

CHAIR STETKAR: No, no.

MEMBER SCHULTZ: Don, for clarity of clarification --

MR. DUBE: Yes? Sure.

MEMBER SCHULTZ: -- I would recommend that on the slide that you have for option 2C, beyond the time when you leave behind the CCFP that you add to the going-forward requirements that you are going to augment. You are going to augment the discussion on long-term containment performance.

MR. DUBE: Good point, yes.

MEMBER SCHULTZ: Because it looks like you are leaving everything behind on containment, and you're not.

MEMBER CORRADINI: You are trying to say that the paragraph we just discussed really is the --

CHAIR STETKAR: And the paper builds that.

MR. DUBE: Yes. The paper does.

MEMBER SCHULTZ: The slide looks different. That is to make sure that it's shown that there is a requirement related to containment that's going to continue.

MR. DUBE: Okay. Good point. This slide may only be used one time. And that would be at the full ACRS.

MEMBER CORRADINI: We'll make sure that you remember that.

CHAIR STETKAR: We've got a good fraction of the Committee here. Some important people are not here.

MR. DUBE: Remind me because I have short-term memory loss.

CHAIR STETKAR: Some important people are not here.

MR. DUBE: Thank you. Okay.

So tier 2 changes. I am going to get us within the half-hour. Okay. This was a specific request or direction from the Commission to look at. We knew there was a working group at NRC called Change During Construction. And the changes for ex-vessel severe accident features is a subset of that larger change during construction. So we knew we were going to have to address this issue, even before the Commission SRM. So this is why I said we had this public workshop in December 2010, before the SRM, because we knew we were going to have to address it regardless.

6. NEI 96-07 APPENDIX C

MR. DUBE: NEI 96-07 is a nice one-stop shopping document for all of the change processes for the part 52 plants. It discusses tier 1 changes, tier 2, tier 2* -- I'm assuming I don't have to get into all of that -- impact of design basis accidents, even aircraft impact assessment, loss of large areas.

So what enters into this appendix C to NEI 96-07, which is the guidance 50.59 and then it may direct the licensee to different directions, but in different guidance reports, but it is a one-stop shopping. And there is guidance on the tier 2 changes to ex-vessel severe accidents. And these are codified in the rulemaking for each of the certified designs.

So we had a second workshop on August 9th. And we had a series of public meetings looking at appendix C in general. And there have been three since the past summer.

So, to refresh your memory, this is an advanced boiling water reactor from -- this is appendix A to part 52. So this is, in effect, a rule or regulation.

And VIII.B.5.c says, "A proposed departure from tier 2 affecting resolution of an ex-vessel severe accident design feature identified in the

plant-specific DCD requires a license amendment if" one of two conditions is met. "There is a substantial increase in the probability of an ex-vessel severe accident such that a particular ex-vessel accident previously reviewed" that was "not credible" becomes "credible or there is a substantial increase in the consequences."

Now, one of our assignments was to tackle what a substantial increase is because that went unanswered for 15 years. So I feel good about this.

I don't plan to talk about it because I think we talked about it a little bit before. But we believe what's in the guidance is good.

CHAIR STETKAR: Do you know what the word "credible" mean?

MEMBER CORRADINI: I understand number 2. I have read through number 1 a couple of times. What does that even mean?

MR. DUBE: Sometimes in a design control document, you may not find the word "credible." You may find words like "not physically possible," "has been ruled out," "not possible," "beyond belief" or some terms like this.

CHAIR STETKAR: "Not possible" and "impossible" are pretty well-defined words.

MR. DUBE: Yes. But basically "not credible" means -- and we'll talk about this. There are a number of severe accident challenges by regulation that the COL holder as design certification has to address. And they specifically have created a design feature to address those. And because of those design features, a severe accident challenge is not likely to occur. I mean, I don't know what more to say.

Sometimes you will see the design certification document actually an actual probability, like we believe that high-pressure melt ejection is 10-8 or less. And because of that, we are not going to further analyze these scenarios. Sometimes you'll see the word "credible."

I don't know if I answered your question, but a design feature has been implemented to address a severe accident challenge. And because of that, there is no further analysis done.

MEMBER CORRADINI: That's what number one is. And some parts will be taken that will then reverse that.

MR. DUBE: Exactly. So --

MEMBER BLEY: To make it more likely.

MR. DUBE: Right. Let me give you an

example.

MEMBER BLEY: Right.

MR. DUBE: I have a pressurized water reactor. I am worried about high-pressure melt ejection and direct containment heating. So I installed depressurization valves using squib valves so that in a severe accident, operator opens these depressurization valves that crashes the reactor coolant system pressure down to very low value. And so even if the molten debris melts to the bottom of the reactor vessel, we're not going to have the high-pressure melt election. And so that --

MEMBER CORRADINI: And somehow the valve design has changed, and that changes the probability of that event?

MR. DUBE: Exactly. Exactly. So in so many words, as a design feature that addresses high-pressure melt ejection and now I'm looking at a change which is I had this feature. I do something to that, turn something that has been not credible, high-pressure melt ejection, do something that makes this, for whatever reason, credible.

MEMBER CORRADINI: Thank you. Thank you. That helps.

MR. DUBE: And we struggled with this. We

didn't want to come up with a quantitative definition. Basically an example of number one would be a severe -- I have two parallel valves, let's say, paths, put as depressurization. And I do something to the design that severely degrades that capability.

MEMBER CORRADINI: Yes. Okay.

MR. DUBE: And it's like beautiful art. I know it when I see it, but, I mean, I'm not going to put quantitative numbers on it. And then two is substantial increase in the consequences. It's the same thing. No one has defined substantial increase, but we think we have zeroed in on a definition.

So that --

MEMBER BLEY: Can I back you up to the one you just had?

MR. DUBE: Okay.

MEMBER BLEY: Trouble. We all have trouble with this idea of credible, but it's something that you think is very, very unlikely or the designers thought. You could get quantitative if you talked about some kind of fractional degradation or something like that.

This still stays -- I mean, this has always been fuzzy, wherever we have had things like this. You know, Impossible is not bad. It's just

hard to prove and then anything that challenges that.

But it just seems to me that, at least on some relative basis, you could be quantitative and not leave quite as fuzzy a --

MR. DUBE: Good point. We struggled with this. So what's a substantial increase? Is it a twofold increase? Ten percent? One percent? Probably not. Two percent increase? Probably not. Ten percent? Probably not.

The other extreme, a tenfold or 100-fold increase in probability? Probably. Where do I draw that magic line?

MEMBER CORRADINI: You are saying substantial falls between two to ten. That's what you just told me.

CHAIR STETKAR: Two to 100. But, I mean, that is something that could be quantified as an input to the decision process. Even in your example there, the frequency of high-pressure melts is not zero. It's something that can --

MR. DUBE: Right.

CHAIR STETKAR: -- and is quantified in the PRA. There's some chance that the square valves won't open or --

MR. DUBE: Right. Exactly.

CHAIR STETKAR: -- you know, whatever. And any change is measurable as a delta.

MR. DUBE: Right.

CHAIR STETKAR: I mean, you know --

MR. DUBE: But there were --

CHAIR STETKAR: It is a credible event in the sense that it can be quantified.

MR. DUBE: There were a lot more disadvantages putting a quantitative number to this than --

MEMBER CORRADINI: Can I give you another example that troubles me more since you picked that one? So I changed the insulation on the vessel for AP1000.

MR. DUBE: Right.

MEMBER CORRADINI: And in-vessel retention is threatened.

MR. DUBE: That is a substantial increase.

MEMBER CORRADINI: But it is my impression -- I used this example specifically because it is my impression in-vessel retention is not part of the license. It was an additional feature that fell outside of it and was only in the PRA.

MR. DUBE: The defense-in-depth.

MEMBER CORRADINI: So that would not fall

into this category.

MR. DUBE: Well, we worked it. We had this example. And we said that would be some feature like that that in-vessel retention and external reactor vessel cooling is a defense-in-depth measure that if one were to make a change that one lost confidence that one would be able to do that, it would be a substantial increase in probability of --

MEMBER CORRADINI: So I'll give you another one.

MR. DUBE: We end up using this as an example in the guidance. I don't remember now. But we specifically talked about that example.

MEMBER CORRADINI: Because where I am going with this is if these measures were the uncertainty of the physics leads you to put it in as a defense-in-depth measure, but I wouldn't apply this rigor to it because the uncertainty of what is occurring is too mushy. Excuse my English. I think of this one. I think of the BiMAC. I can come up with a few of these.

But they look better. Do I want to apply this sort of rigor to it? That would get me nervous.

MR. DUBE: Well, that's why the writers of the rule specifically chose the word "substantial" for

that reason, because of --

MR. ADER: Yes. This was --

MR. DUBE: -- severe accident uncertainty.

MR. ADER: I mean, as I read the statement of considerations, this was originally going to be under the normal change process. And, for the reasons you said, severe accidents too uncertain that they put it in a change process that has more --

MEMBER CORRADINI: Okay.

MR. ADER: -- substantial increase, as opposed to significant. Don't ask me --

MEMBER CORRADINI: This is why you said 2 to 100. Let's just like leave it there.

MR. ADER: I think most of these features are in tier 2.

MR. DUBE: Some of them are in tier 1. So they would get --

MR. ADER: Yes, some are in tier 1.

MR. DUBE: It turns out as we did a review, a lot of these area actually mentioned in tier 1. So they need a license amendment. This is what can the COL holder do on their own without prior staff approval?

MR. ADER: So it is more than just being reflected in the PRA. It is reflected in the design

documents.

MR. DUBE: Yes.

MEMBER CORRADINI: Okay. Thank you. Thank you.

CHAIR STETKAR: Ed?

MR. FULLER: Yes. Ed Fuller again.

I would like to point out to the group and Mike, in particular, that those SECYs 90-016 and 93-087 are used to define what the severe accident challenges are and to define what, in turn, to direct people who design these reactors to include mitigation systems to address these challenges and that 10 CFR 52.47(a)(23), I guess, basically addresses very specifically the challenges.

One of these is the ability to cool debris. And if you'll look at the AP1000 with its insulation system, that is a system, a mitigation system, that is designed to make sure that you don't have core-debris-concrete interaction that leads to basemat melt-through.

So I would say in the context of what we are doing here that, indeed, this is an ex-vessel severe accident mitigation feature and that if it's taken out of its degree of effectiveness, that you could get significant increases in consequences of

such an accident.

MEMBER CORRADINI: But since you -- he'll stop me, but let's just push on this one. So when the staff was going through the AP1000, there were some contractors that were asked to look at the license analysis and found that the chance of this not working was more than ten percent. We'll say one out of ten just for the sake of argument. So the chance is essentially the in-vessel retention design in a tier 2 is one out of ten.

Now, they changed something. What I'm trying to get at is that the change takes me from one out of ten to two out of ten. Do I get nervous or I only get nervous when it becomes nine out of ten?

And the reason I am asking the question is the uncertainty in the one out of ten is large.

MR. FULLER: Right.

MEMBER CORRADINI: It's not one out of ten. It's not ten percent plus or minus one percent. It's more like ten percent plus or minus, plus a factor of four.

MR. FULLER: Right. Okay. In this particular case, there's a backup. That is, if the system doesn't work and the vessel fails, you still have a flooded cavity, which is large enough to meet,

for example, the utility requirements document definition of what might be coolable.

And since then a lot of experiments have been done. And the indications today are that you would be coolable. So it's more than just the insulation system. It's the backup, too.

MEMBER CORRADINI: I understand that, but I just want to press the point because I am just trying to understand the thinking process here because my other way of saying it is that I am saying it now. I will change to the ESBWR.

When we went through this discussion there, what was the final point that made some of us -- I don't know which of the some of us it was, but it wasn't me -- feeling good about it, even if the BiMAC didn't work, I still fell within the fact that within the first 72 hours, I wouldn't have a failure.

So I am going back to the previous history of how I would address this. That is, even if the added feature didn't function as expected, I still fell within essentially a bigger principle that is maintained.

MR. FULLER: Yes. In that case, again, you have enough floor area and height and provided you can keep water in there, you can cool the debris.

MEMBER CORRADINI: Okay. Fine. I understand. Thank you.

MR. DUBE: Well, I mean, without holding me to numbers, going from .1 to .2, the staff might not be as concerned, then. But certainly going from .1 to .9 or almost 1, that would certainly represent a substantial --

MEMBER CORRADINI: Fine. I'll stop. Sorry.

MR. DUBE: So I know I've got to move on.

MEMBER RAY: Did you ever figure out what credible meant?

MEMBER CORRADINI: We'll just leave that aside.

MR. DUBE: Yes. I mean, we have some guidance in there. And we said it's a feature that's been installed to address one of the containment challenges and has -- and use terms like "made it physically impossible," "extremely unlikely," "below some numerical threshold."

MEMBER RAY: Okay. Well, the reason I'm asking -- again, I don't want to screw us up here, Don -- is that the word "credible" appears in many places, as we know, not just in this application here. And I was trying to figure out if those definitions that you

were referring to just now were meant to be more universally applicable.

MR. DUBE: I've got to be careful because someone way beyond my pay grade has not been able to define credible for 50 years. So I'm not going to be able to do it in this project.

MEMBER RAY: Well, sure. I can show you licenses that say "maximum credible" and then if it's greater than that. So, I mean, obviously it's not impossible to be more severe than maximum credible.

MR. DUBE: But, I mean, we do state in the guidance that if a feature has been installed to address a severe accident challenge, there is reasonable expectation that it would work. That means that accident sequence resulting from that challenge is not credible.

We also made it clear that because it wasn't a universal definition of credible, the various design control documents have used other terms, everything from, like I said, unlikely to physically impossible and so forth.

CHAIR STETKAR: We haven't seen teeny-tiny or itsy-bitsy yet.

MEMBER RAY: We all acknowledge that the word "credible" is problematic. And I just wondered

if this were some magical answer to the question of what does it mean? I'm going to say no.

CHAIR STETKAR: No. I understand that at least the staff is not trying to tackle that issue. They're just propagating it.

MR. DUBE: Yes. Okay.

CHAIR STETKAR: I don't know if you can make it all the way through, Don, but see how far you can get. We do have to end at 12:00.

MR. DUBE: Okay.

CHAIR STETKAR: That is a magic time.

MR. DUBE: All right. So there is a definition of what is meant by "ex-vessel severe accident" in the statement of consideration and states explicitly the core has melted to the reactor vessel and containment is being challenged. And I'll just leave it at that.

These are the five specific containment challenges that are in 52.47(a)(23) and .79(a)(38). We reached a consensus during the workshops that containment bypass is not necessarily by the definition of statement of consideration an ex-vessel severe accident feature or features to address containment bypass because you can have a containment bypass, like an ISLOCA, where the molten debris is not

1 necessarily melted through the reactor vessel.

2 So the result of that is we found that

3 certain accident features do not address ex-vessel

4 conditions and the VIII.B.5.c don't apply. And I

5 mentioned ISLOCA/containment bypass. This is a gap in

6 the -- and, worse, it's a gap in the rule, but it is

7 certainly a gap in the guidance.

8 We reviewed NEI 96-07. We made comments,

9 and those have been incorporated. So we're happy with

10 the ex-vessel portion. But the real bottom line is

11 that last bullet: In a worst case, a significant tier

12 2 change to a non-ex-vessel severe accident feature,

13 up to and including permanent removal from service,

14 could be made without prior NRC approval. And this is

15 what staff is concerned.

16 Now, this is theoretical, but the gap --

17 and I'll show you a Venn diagram and a process

18 diagram. I mean, it is possible in the worst case to

19 visit a feature there that is put in to address severe

20 accidents. Because it's not addressing an ex-vessel

21 severe accident, as defined in the regulation here --

22 MEMBER BLEY: When you say, "could be

23 made," could be made without having to raise a flag

24 about it under risk-based?

25 MR. DUBE: Yes. It could be screened out

by the current process.

So here is a Venn diagram, going back to ninth grade mathematics. These are the five --

CHAIR STETKAR: Not my ninth grade.

(Laughter.)

MR. DUBE: Challenges. If you look at what ex-vessel severe accident is defined per the statement in its consideration, certainly core-concrete is within that definition. High-pressure melt ejection is within that definition. You could have a hydrogen explosion without having an ex-vessel severe accident. So the red line circle crosses through there.

It wasn't clear -- and I looked very closely whether one was referring to in-vessel steam explosion or ex-vessel steam explosion --

MEMBER CORRADINI: Your red is right. It is a good red line.

MR. DUBE: So I divided that. Now, I cannot for the life of me think of a design feature that a designer specifically put in just for the purpose of mitigating or preventing in-vessel steam explosion. I mean, I just can't think of it, and I can't think. But, nevertheless, the line does go through there.

And containment bypass, as I mentioned, is to induce thermal steam generator tube rupture or interfacing systems LOCA is not an ex-vessel severe accident necessarily by the definition. So there is a gap.

Okay. Another way to look at it is a process diagram. And I am going to point out where these two issues can occur. So let's start at the change. There is a change that a license holder wants to make.

First question, is it tier 1 or tier 2? Remember, tier 1 required prior NRC approval. And tier 2* is something in between tier 1 and tier 2, but it requires prior NRC approval. This is typically anything having to do with the fuel design and other things.

CHAIR STETKAR: They typically disappear after the first operating cycles.

MR. DUBE: Yes. If the answer is yes, a license amendment request is necessary. If the answer is no, then by definition, what's left is it's tier 2. Then the question is, is it an ex-vessel severe accident feature as defined in the statement of consideration because we have no other definition?

If the answer is yes, then you go to NEI

96-07, appendix C, 4.4.2.3. Then the question is, is it something that has been evaluated and deemed credible? If the answer is yes, VIII.B.5.c criteria are used. If the answer is no, you only evaluate the impact on the probability per A-B.5.c. So remember this right here.

You only look at the consciences. There is a substantial increase in the consequences to the product of a severe ex-vessel severe accident previously reviewed. So my thought process is really --

MEMBER CORRADINI: You are saying something was missed? Is that what you're saying?

MR. DUBE: Yes. I mean, it's possible that one would only look at the impact on probability and not the consequences. If, as the AP1000 states, that they have no credible ex-vessel severe accidents, you never ever look -- you only look at the impact on probability and not on consequences. So you only look at on the VIII.B.5.c -- or if the answer was no, you would only evaluate impact on probability for VIII.B.5.c. You never look at the impact on consequences.

But that's not the root of real concern. The root of real concern is if the ex-vessel severe

accident diamond, the answer was no, there's no impact or it's not addressing an ex-vessel severe accident feature, that means, then, the change might affect some other severe accident feature. If the answer was no, one would use VIII.B.5.b criteria. The problem with that is the VIII.B.5.b creature is for design basis accidents, not for severe accidents.

MEMBER CORRADINI: Okay. I'm really getting long-winded here, but you could have a change to a severe accident feature that's not ex-vessel severe accident. And you would be using design basis accident criteria, not the ex-vessel severe accident criteria. VIII.B.5.b is for design basis accidents. So one would not be asking these questions for a tier 2 change to a severe accident feature that's not ex-vessel.

The other concern is where I said that if one could make a change to a severe accident feature where one would then be led to use appendix C, 4.4.3.2. And that would allow screening out from further review because there's no impact on design basis accidents.

An example would be if you had a feature to prevent core damage and it was only in tier 2, there was not a discussion in tier 1. If you go

through the process since it's not there to mitigate an ex-vessel severe accident and it's not there for design basis accident purposes, the guidance would have allowed this, the change, to be screened out and not require any prior NRC approval.

Does that logic follow or should I go through it again?

MEMBER CORRADINI: Keep on going.

MR. DUBE: That's why I said --

CHAIR STETKAR: Silence is complete understanding.

MR. DUBE: That's why I said in the worst case --

CHAIR STETKAR: You could take that approach, too, but --

MEMBER CORRADINI: The Venn diagram helped me. After that --

CHAIR STETKAR: Yes.

MR. DUBE: It's hard to parse an example here.

MEMBER BLEY: It's hard to point you without a specific example.

MR. DUBE: I'll give an example.

CHAIR STETKAR: You've got an example?

MR. DUBE: Yes.

CHAIR STETKAR: Let's do that.

MR. DUBE: The point here is because of the definition of ex-vessel severe accident -- and those criteria only apply to ex-vessel severe accidents -- number one, severe accidents that are not ex-vessel in nature, such as containment bypass, do not have an appropriate set of review criteria commensurate with this for severe accidents.

CHAIR STETKAR: For example, if you made your low-pressure --

MR. DUBE: Right.

CHAIR STETKAR: -- isolation valves out of tissue paper, they probably would not --

MR. DUBE: And the only thing that is left since they are not using criteria for severe accidents is design basis accidents. Well, the guidance says, "I'm not using this for design basis accidents."

CHAIR STETKAR: So an RHR system is a non-safety-related system.

MR. DUBE: Screen out the change. Now, for a number of reasons, a number of these features are in tier 1. And that helps prevents a change that wouldn't get prior NRC approval.

CHAIR STETKAR: It's fortuitous.

MR. DUBE: Yes. It may be or may not be.

So we did a gap assessment. We reviewed severe accident features for three plants. We found no significant gaps of concern for a number of reasons. Either it's an ex-vessel severe accident feature and VIII.B.5.c criteria would be used for tier 2 changes or if it's not a non-ex-vessel severe accident feature, there's enough detail in tier 1 to preclude a significant design change without prior NRC approval. And we're verifying these conclusions. We'll eventually look at the other standard designs. So we have a recommendation 1 to the Commission to fill in this gap.

So here is an example why for the ABWR we don't think there is a gap. And here is a concrete example. The ac-independent water addition system provides the ability to flood the lower drywell via drywell sprays. It can also provide injection emergency makeup water to the reactor by cross-connection division C of RHR to the reactor building.

The first question is, is it an EVSA feature? Is it an ex-vessel severe accident feature? Well, some of it is, and some of it isn't. Those aspects for flooding the lower drywell via drywell spray meets the definition of ex-vessel severe

accident. And so any change you have to look at the impact on consequences. And in design control documents, there is discussion in tier 1 and tier 2 and so forth. So there's enough discussion in tier 1 that there is no gap in the change process. And the license holder could evaluate this under A, B.5.c for tier 2 changes. And any tier 1 changes would require NRC approval.

But those aspects for injecting emergency makeup water into the reactor is not an ex-vessel severe accident feature. All right? It's not addressing molten debris melting through the bottom of the vessel. Okay? So any change neither impacts probability nor consequence. This feature contributes to severe accident prevention but does not meet the definition of ex-vessel severe accident feature.

Fortunately, there is enough discussion in tier 1. And I'll show you an example. A license holder could make very much of a change without impacting tier 1. But if there was no tier 1 discussion, only tier 2 -- let's just say hypothetically there was no tier 1 discussion. They could in theory defeat this function and not require prior NRC approval. That's the gap.

Now, we went through all of the design

features and made sure that we had either a tier 1 discussion in sufficient detail in tier 1 or it was an ex-vessel severe accident feature, that it would get some kind of review.

CHAIR STETKAR: You say that's for the currently certified designs?

MR. DUBE: Those three designs that I mentioned. And this is why in tier 1 there is a lot of detail. So if they made any change to this paragraph here on this slide, they require prior NRC approval.

I won't go through all of it, but it discusses -- you know, it can be used for injecting emergency makeup water. It goes into the detail of having manually opening two in-series valves and a cross-connecting pipe. It's accomplished by local manual action in the valves. Fire protection water can be directed to either the reactor pressure vessel or drywell sprays and so on. So this is a lot of detail that a license holder doesn't have a lot of room to make changes that doesn't impact that.

So the long and short of it is yes, there is a gap to the process. It turns out for the three reactor designs that are certified in near certification, there is enough detail in tier 1. It

may be deliberate, but you can't rule out some other new reactor design coming in there and not having enough detail in tier 1.

So we'll discuss later this afternoon how we propose addressing this gap in the recommendation.

MEMBER REMPE: You said the staff reviewed it. You went back and got each of the design --

MR. DUBE: Yes.

MEMBER REMPE: -- certification teams involved in everything?

MR. DUBE: We got the design control documents and went through every severe accident feature.

MEMBER REMPE: And it was the PRA staff or --

MR. DUBE: Yes.

MEMBER REMPE: -- actually going back and getting the people that were involved in design certification from the NRC staff?

MR. DUBE: Just current NRC staff. But we're going to be getting an independent set of eyes to look at it.

MEMBER BLEY: Don, when you come back this afternoon, could you take just a few minutes and help clarify for me the definition of the ex-vessel severe

accident design feature? Is it where the accident occurs, which it must be. Just it would help clarify things a bit for me. I'm a little fuzzy on it.

MR. DUBE: Okay.

CHAIR STETKAR: Anything else?

(No response.)

CHAIR STETKAR: Don, you did a great job. We are recessed until 1:00 o'clock.

(Whereupon, a luncheon recess was taken at 12:02 p.m.)

A-F-T-E-R-N-O-O-N S-E-S-S-I-O-N

(1:01 p.m.)

CHAIR STETKAR: We are back in session.

MR. DUBE: Are there any lingering questions on ex-vessel severe accidents? Again --

CHAIR STETKAR: Dennis had asked for a brief synopsis of what those scenarios are.

MEMBER BLEY: How you decide some things are ex-vessel severe accident feature.

MR. DUBE: Okay. I'll do that.

MEMBER BLEY: Does it depend on where the accident begins? What's it about?

MR. DUBE: It's defined in the statement of considerations.

MEMBER BLEY: Yes.

MR. DUBE: A feature of where the intended function of the design feature is relied upon to resolve postulated accidents when the reactor core has melted and exited the reactor vessel and the containment is being challenged.

MEMBER BLEY: Okay.

MR. DUBE: So something like a BiMAC on ESBWR, which is a core-catcher in fancy terms, is an ex-vessel severe accident feature.

Some aspects of squib-activated

depressurization valves in the ESBWR, those aspects related to preventing high-pressure melt ejection to be an ex-vessel severe accident. Those aspects for design basis accident mitigation -- I mean, that's just part of the pressurization -- are not. Now, fortunately, I mean, it's hard to separate the two.

MEMBER BLEY: It strikes me that nine-tenths of your problem is due to definition. We didn't even need that.

MR. DUBE: And I mentioned the containment venting would be an ex-vessel sever feature. But the fire water addition system in ESBWR for flooding the lower dry well should the core become damaged and relocate into the containment --

CHAIR STETKAR: Is.

MR. DUBE: -- is.

CHAIR STETKAR: But not to prevent core damage.

MR. DUBE: Yes, in the ABWR not to prevent core damage. So you see the trap that we, plural we, worked ourselves into, those features specifically to address the accidents when the molten debris has exited the vessel are captured by this change process. Some of these are preventing core damage. And they're not safety-related. The added-on features like this

are not. So where do they lie? They don't have a happy home for a change process. And that's where the gap is.

Now, because a lot of these things are in tier one, you know, we don't believe --

MEMBER BLEY: And it doesn't matter.

MR. DUBE: Maybe it's a moot point, but, you know, who is to say somewhere down the line some new reactor comes in, the line comes in and --

MEMBER SHACK: I assume that you will check to make sure that --

MR. DUBE: Yes. We will. And we'll talk about the recommendations on that. I mean, I know we have a sort of recommendation. NEI has a little bit of a difference of opinion on that.

So I hope that helped. You know, I have other examples on other designs, but that's the definition, the working definition.

Do you want me to put it through for you, Ron?

MR. FRAHM: Yes, please.

MR. DUBE: So unless there are other questions, we will move on to ROP. We will touch upon some of these other topics again when I -- after ROP, I'm going to wrap it up with the basic conclusions and

the Commission paper, recommendations, and summary. But I think we are ready to move on to ROP if that is okay, Mr. Chairman.

CHAIR STETKAR: That is wonderful.

MR. FRAHM: I hope you still feel that way in a few minutes.

7. REACTOR OVERSIGHT PROCESS TABLETOP EXERCISES

MR. FRAHM: Good afternoon. I am Ron Frahm. I'm in the Performance Assessment Branch in NRR. And I have the lead for the ROP portion of the tabletop exercises.

And I did want to acknowledge that Rani Franovich, the Branch Chief for the Performance Assessment Branch, is with us this afternoon as well, as is Steve Vaughn, our lead for the significance determination process. So all hard questions will be forwarded over to that side of the room.

Well, we talked quite a bit about licensing tabletops. So we also ran several tabletops on the risk-informed aspects of the ROP. I did want to go into a little bit of ROP background and framework to kind of set some context and perspective on the ROP and how it all fits together. And then I plan to go through and discuss the existing risk-informed guidance for those risk-informed aspects

of the ROP, those being the significance determination process, mitigating systems performance index, and the management directive 8.3 for event response.

Then next I'll go over the approach that we use to perform these tabletop exercises, but the real focus of today's discussion will be on the results and the conclusions from the tabletops. You'll see there are several slides on that.

And then, lastly, we'll present the ROP options and recommendations, but I believe we will actually do that in a later session, where we talk about the Commission paper itself.

With that little bit of ROP 101, background on the ROP, it was first implemented in April of 2000. So we place the previous oversight processes, which we believed to be a little bit too subjective and unpredictable. And one of the primary objectives of the ROP was to improve the objectivity of oversight and minimize the subjectivity in our decision-making.

Another important objective was to improve the scrutability of our actions. And I'm not sure if we invented that word "scrutability" or not, but what it really means is to make our actions understandable and predictable. And our regulatory responses would

have a clear tie to licensee performance.

Another key objective and really the integral part of today's discussion is that the ROP be risk-informed and that the NRC and licensee focus on the issues with the greatest impact on safety.

This next slide provides the ROP framework. And I really wanted to put this up to demonstrate that there are seven cornerstones of safety that go through two-thirds down the page there. And these cornerstones support the strategic performance areas of reactor safety, radiation safety, and safeguards, which, of course, support the NRC's mission to protect public health and safety. The seven cornerstones are initiating events, mitigating systems, barrier integrity, emergency preparedness, public and occupational radiation safety, and security.

CHAIR STETKAR: Ron, be careful. The mike is real sensitive.

MR. FRAHM: Oh, okay.

CHAIR STETKAR: It kind of explodes --

MR. FRAHM: Thank you.

CHAIR STETKAR: -- in people's ears here.

MR. FRAHM: Thank you.

Okay. Moving on to this next slide, what

I wanted to show here is that within each of those seven cornerstones, there are objectives that need to be met to indicate that the licensees are safe within the boundaries of the cornerstone. And we do that through two different means that are equally weighted.

We have NRC inspections. We have performance indicators that are submitted by the licensees. This data is run through our significance determination process for inspection findings and then predetermined thresholds for the performance indicators.

And based on the level of significance, they get colors. Green is the best. Then progressively white, yellow, and red are more safety-significant. Then these inputs feed the ROP action matrix there in the middle in the pink.

And based on the column in that ROP action matrix, I hope everybody is familiar with what that looks like. I did not have a slide on that. But based on the column in that action matrix, that will determine our regulatory response. And one of the keys here is that that regulatory response is predictable. You can see exactly why we're doing what we're doing, why we're doing this.

And if we were to deviate from the column

in that action matrix, we would have to get approval from the EDO to deviate. And that's intended to be very rare. I think over the 12 years of the ROP, it's been done 18 to 20 times maybe. So we try to be very predicable and repeatable, et cetera, understandable.

MEMBER SIEBER: When the EDO approves an exception, is that to be more severe or less severe?

MR. FRAHM: Either way. We have gone both ways.

MEMBER SIEBER: Out of the 18 times it's been done, how many times did you --

MR. FRAHM: I don't --

MS. FRANOVICH: We can go back and check, but based on my recollection, I would say maybe 20 to 25 percent of the time it's to have a regulatory response that's less than what would be detailed by the action matrix.

MEMBER SIEBER: Okay.

MS. FRANOVICH: We can give you some specifics. We'll follow up with that.

MEMBER REMPE: Could you just give one example?

MR. FRAHM: I would say that is fair. Let's see. A good example of a deviation: groundwater contamination at Indian Point.

That didn't trip any thresholds, but we felt we needed to do a little bit more than what the action matrix would have told us to do. So region 1 requested a deviation. The EDO approved the deviation to go and do the additional inspection.

MEMBER SIEBER: So 25 percent of these decisions lead to lower and 75 percent lead to increases?

MR. FRAHM: That's pretty solid math.

(Laughter.)

MR. FRAHM: I would say yes. And we're guessing at those number. I would say somewhere between a quarter and a half. I'd say it's more likely to increase regulatory response than -- based on my recollection and experience.

MS. FRANOVICH: Or at least that's the historical precedent that we're aware of.

MR. FRAHM: Right.

MEMBER SIEBER: Thank you.

MR. FRAHM: Sure. Okay. Moving on, I want to provide a little bit of the guidance on each of the three main risk-informed aspects of the ROP. For the SDP, the process is described in IMC-0609. And that really focuses on the cornerstones of initiating events, mitigating systems, and barrier

integrity. I'm sorry. Appendix A of 0609 focuses on those cornerstones. And that's really the risk-informed aspect of the SDP. A lot of the other SDPs are a little more deterministic in EP and public radiation safety, et cetera.

And the risk thresholds are a function of changes in CDF and LERF against the plant's baseline risk. That's all I wanted to say there.

CHAIR STETKAR: He's coming across the table.

MR. FRAHM: Okay. For mitigating systems performance index, that guidance for all PIs is contained in IMC-0608, which, of course, includes MSPI. But the real details of the PI guidance are contained in NEI 99-02, which is controlled by obviously NEI and the industry. But we meet on a periodic basis and agree on the content of that document.

The MSPI does cover five systems that are important to safety. And, of course, they are all in a mitigating systems cornerstone. They use a calculation to track the availability of the monitored trains and the reliability of the monitored components within these safety systems. The MSPI reflects the deviation of a specific unit's performance from an

industry baseline and converts that into a simplified change in CDF.

In addition to the risk calculation, there is a performance limit or a deterministic backstop that's used for determining grade, performance. And exceeding this limit in any one of the MSPI indicators would result in a white indicator.

CHAIR STETKAR: Ron, I have to admit ignorance here. Are there five specific systems that apply to everyone because I think it must because --

MR. FRAHM: P's and G's are a little bit different.

CHAIR STETKAR: What are those systems? We have MSPI review but --

MR. DUBE: Emergency AC power, heat removal. So it's either for the PWRs it's emergency feedwater, aux feedwater, or RCIC for BWR, high pressure injection, residual heat removal, and then something called cooling water. So that may be actually a combination of service water, component cooling water. Some plants have raw water. So all the cooling waters are going to wind up in the --

CHAIR STETKAR: Thank you.

MR. FRAHM: Okay? Moving on to event response, that implementation guidance is in

management directive 8.3 with supplemental guidance in inspection manual chapter 03.09 for determining our response to events.

The risk-informed thresholds used for determining which reactive inspection to perform are a function of CCDP, conditional core damage probability, and conditional large early release probability.

And there is an overlap within these options based on the uncertainty and deterministic insights that provide a little bit of flexibility in determining an appropriate response. And there is also additional deterministic criteria that is reviewed and documented as the basis for our decision within that overlap region. And I'll attempt to demonstrate this on an upcoming slide. In fact, it is the next slide.

So on this slide, what I did, I actually put this together myself based on the three guidance documents to try to give a one size fits all demonstration of the risk aspects of the ROP. And you'll see the SDP in the thresholds on the left, MSPI in the middle. And the asterisk on the white there is based on the performance limit that I talked about. If that performance limit is reached, you are

automatically white for MSPI.

And then if you look at MD 8.3, you'll see there is a range for the IIT and the is and the AIT. So based on your risk number, if you fall within a range that crosses two of the different types of reactive inspections, you can select the inspection based on uncertainties and deterministic insights.

MEMBER SKILLMAN: Ron, may I ask you to go back one slide, please, to the second bullet, regarding reactive inspections? What assures that the NRC's calculation, condition of core damage probability and LERF and the licensee's calculation of those two parameters, are essentially the same number?

MEMBER SIEBER: They aren't.

MS. FRANOVICH: This is Rani Franovich from NRR staff.

I don't know that we seek input from licensees on reactive inspections. I think that that is purely an NRC-generated decision-making insight. When it comes to significance determination process, we do engage with licensees and have enforcement conferences, receive information on the docket that they like to provide to inform our risk-informed decision, but when it comes to reactive inspections, the same process does not apply.

MEMBER SKILLMAN: Thank you. Thanks.

MEMBER SIEBER: Your basic decisions are based on SPAR models, right?

MR. FRAHM: I believe. I believe so.

MR. DUBE: Fundamentally, yes.

MEMBER SIEBER: Yes. Right.

MS. FRANOVICH: Thanks, Don.

MR. FRAHM: Okay. That's really all I had on this slide. We may want to come back to it when we go over some of the examples.

Now to get into the actual tabletops themselves. They were conducted on October 5th and a follow-up meeting on October 26th, as Don previously noted. And, similar to the licensing tabletops, we tested various realistic scenarios. We wanted them to be realistic. In fact, the SRM required that they be realistic to confirm the adequacy of the ROP risk-informed processes or to identify areas for improvement and address any noted gaps.

We developed several examples across a broad cross-section of well-vetted cases, actual cases across several designs and vintages and developed from these actual examples, we ran through SDP, MSPI, and MD 8.3 case studies.

We applied similar situations to the new

reactor designs and then filled in some gaps and inconsistencies with the designs with some realistic hypotheticals and reasonable assumptions. And then we compared the risk values and resultant regulatory response to ensure that we're in the right place.

And a summary of these case studies was included as an enclosure to the October 26th meeting minutes. So it should be in your packages. And that was actually distributed ahead of time before the tabletops so that folks could digest it and come to the tabletops with some informed insights.

So the first set of tabletops that I would like to talk about today was the SDP tabletops. They did indicate that the existing risk thresholds for determining the significance are adequate. I think it's safe to say that some of us were surprised at the fact that we were able to get several greater than green findings, even given the more robust designs for the new reactors.

The tabletops demonstrated that these thresholds could be crossed and produce a regulatory response, but they were limited primarily to common cause failures across multiple safety systems or long exposures of risk-significant components.

We found that the existing process does

not always ensure an appropriate regulatory response for the degradation of passive components and barriers. And I'll discuss an example on the next slide.

We concluded based on these tabletops that the SDP analyses could be augmented with additional qualitative considerations, such as deterministic backstops, to ensure the NRC appropriately addresses performance issues.

And these preliminary results and conclusions were really kind of discussed and agreed to, for lack of a better term, at that October 26th --

CHAIR STETKAR: Ron, before you go to the vessel head one, which I think we want to understand, I went through the examples. And you mentioned that the conclusion was that if you applied enough common cause failures or applied a long enough exposure period, you transcended the greater than green thresholds.

I wanted to ask you a question about -- one of the examples was high-pressure core flow, ABWR. And you ran two exercises: one where you took out one high-pressure core flood pump. And this apparently occurred at the Perry plant and was a white finding, I think.

And for the ABWR, it didn't even close to exceeding green. If you took out both high-pressure core flood pumps, you just made it over and above the green.

Question I had -- and this will be pertinent later. You may not have it at your fingertips, but you might be able to bring to bear someone who could have it. What was the percentage change in the Perry core damage frequency for the actual event? And what was the percentage change in the ABWR core damage frequency for the same goal, high-pressure core flood pump?

MR. FRAHM: That I could not answer for you. I'm not sure if anybody is here that --

MS. FRANOVICH: Don, do you have that information in the tabletop exercise summary that you've got there or do you --

CHAIR STETKAR: The summaries do not. I could have calculated in my -- I know what the delta is in an absolute sense is, but I don't know what the denominator is.

MR. DUBE: I am going to take a good judgment call here. In the Perry case, the delta CDF was 5E-6, right?

CHAIR STETKAR: That is correct, yes.

MR. DUBE: Apart from the fact that it was for 23 days, you increase 5E-6 from some baseline. And then you integrate that over 23 days is how you get the change in core damage probability, convert from change in core damage frequency to a core damage probability by multiplying by the exposure time. But I am going to take a good engineering judgment guess that 5E-6 per year change in core damage frequency is 50 percent or even somehow of the order of 50 percent, 100 percent --

CHAIR STETKAR: I am just looking for, you know --

MR. DUBE: Yes.

CHAIR STETKAR: If you had it precise, that's great, but --

MR. DUBE: I mean, typical BWRs of this vintage have baseline internal CDFs in the -- the best are in the low to mid to the -6 and to the low to the -5. So, I mean, this is the kind of --

CHAIR STETKAR: Any sense for the ABWR example? Take the comparable single high-pressure core flood pump for 23 days. Don't extend it out to the year.

MR. DUBE: Well, you know, the baseline CDF for the ABWR internal events I am pretty familiar

because we ran those models. These are like something around three or four E-7 internal events. And we said that we're not delta over one year 2.2E-7. So it's --

CHAIR STETKAR: Yes. But, just for the record, I'm asking for the comparable 23-day, which is 1.4E-8.

MR. DUBE: Yes. 1.4E-8 is the change of core damage probability. So that's --

CHAIR STETKAR: Frequency.

MR. DUBE: Well, 23 days would be core damage probability.

CHAIR STETKAR: I think it was frequency in the tabletop, wasn't it?

MR. DUBE: You take the delta CDF times time. You end up with a core damage probability. I know it says delta CDF here, but that's really a --

CHAIR STETKAR: It's important to get units consistent.

MR. DUBE: Yes. Any time you multiply a --

CHAIR STETKAR: I don't care how you did the calculation. I'm asking for the ratio of whatever it was over 23 days to whatever the baseline was over 23 days if that's what you want to give me.

MR. DUBE: Yes.

CHAIR STETKAR: I can take it in frequency divided by frequency. I can take it in integrated probability divided by integrated probability.

MR. DUBE: During that time, the core damage frequency went up roughly 50 percent due to internal --

CHAIR STETKAR: About 50 percent? Thanks. Thank you. Now we can talk about that. I'm sorry.

MR. FRAHM: Thanks. Okay. The best example that supported our conclusion was the case that involved the vessel head degradation, which resulted in a marginally white finding for the AP1000 and a green finding for the ABWR based on the risk numbers.

And, in fact, I wanted to point out that the draft of the paper that was forwarded to the ACRS actually had this reversed. It was a mistake. We noted it after we forwarded the paper out. And it was actually the AP1000 that had the marginally white and not the APWR. So we'll make that change, obviously, before we send the paper to the Commission.

As we know in the ROP, the resultant regulatory response based on a white finding is to just go in the regulatory response column and perform a 95001 inspection, which is really a pretty focused

inspection of about 40 hours or so, I believe.

And then for the green finding at the APWR, we would do no additional inspection. So, as the last bullet states, we believe that a more robust and diagnostic supplemental inspection, such as a 95002 or a 95003 team inspection, which would correspond with columns 3 and 4 of the ROP action matrix, would give us a better idea that the root causes and degradation are adequately identified and corrected.

CHAIR STETKAR: Ron, I am going to ask you the same question about this. Do you have a sense of what percentage increase in core damage frequency for both US-APWR and AP1000 or US-APWR if that's the only green one because that's sort of the flag for concern?

MR. FRAHM: I do not, but hopefully my friend and colleague Mr. Dube does.

MR. DUBE: Okay. What now? The AP1000?

CHAIR STETKAR: Well, US-APWR on the slide, the corrected slide here, was the one that didn't transcend the green. So I guess I'm more interested in that one.

MR. FRAHM: It did not transcend to white?

CHAIR STETKAR: Did not transcend to white. While you're looking that up, Ron --

MR. DUBE: Their baseline internal event CDF is 10-6. So it's like a 14 percent increase.

CHAIR STETKAR: About a 14 percent.

MR. DUBE: For the APWR.

CHAIR STETKAR: About a 14 percent?

MR. DUBE: Yes.

CHAIR STETKAR: Okay.

MEMBER SHACK: What did those turn out to be for conventional PWRs? White?

MR. DUBE: Red.

MEMBER SHACK: Red?

MR. FRAHM: Red for the similar case at Davis-Besse a few years back.

CHAIR STETKAR: Why in this case, as I understand it -- I'm not familiar with all of the guidance. I'm not familiar with any of the guidance, to be more precise, for these calculations. When I was reading the summary, it said that you basically took this event and increased both the medium and large LOCA frequencies by two orders of magnitude. Is that right? Okay. And that's some --

MR. DUBE: We did that because that is what was done for Davis-Besse originally.

MR. FRAHM: Right.

MR. DUBE: So we were consistent with --

CHAIR STETKAR: Yes. No. I --

MR. DUBE: We can argue whether that is --

CHAIR STETKAR: What I was going to ask is --

MR. FRAHM: But it is within process.

MR. DUBE: Yes.

CHAIR STETKAR: What I was going to ask is some plants are more forgiving of large LOCAs than they are of small LOCAs. So does the process also address this type of event as a small LOCA vulnerability?

MR. DUBE: No. We just increased the medium and large LOCA initiating event frequencies.

CHAIR STETKAR: Okay. Okay. I mean, that was more of a process question. I had no idea how it would come out.

MR. DUBE: We wanted to do as much of an apples to apples comparison --

CHAIR STETKAR: No. I understand. If that's what you did for Davis-Besse, --

MR. DUBE: Yes.

CHAIR STETKAR: -- that's what you should do for this example. Just when I was looking at it, it struck me that you didn't look at the delta risk from small LOCAs, which is not at all clear.

MR. DUBE: Right.

CHAIR STETKAR: You know, I had no idea how that would fit into the --

MR. DUBE: Yes. They have a lower condition of core damage probability. So it might have changed the conclusion a little bit. It would have been more green than --

CHAIR STETKAR: Well, I'm sorry. Some advanced designs --

MR. DUBE: Yes.

CHAIR STETKAR: -- I think can handle large LOCAs and medium LOCAs better than they can handle small LOCAs. There are more things that have to happen to --

MR. DUBE: Yes. I don't --

CHAIR STETKAR: Okay. That was more of a curiosity.

MR. DUBE: Yes.

CHAIR STETKAR: Thank you.

MR. FRAHM: Okay? Moving on, we did identify some potential deterministic backstops as a result of these tabletops -- and we did mention these in the draft Commission paper -- to potentially explore further in the upcoming months or years to identify the shortfalls, assuming the Commission goes

with our recommendations. But these backstops will be consistent with the deterministic criteria in reg guide 1.174, defense-in-depth, maintain safety margins, et cetera.

And one example is that we could have a potential backstop that could be developed to emphasize the importance of maintaining barrier integrity for the fuel cladding, RCS pressure boundary, and containment.

Another possibility would be to potentially address the extensive equipment outage times resulting from degraded conditions, similar to the RITS 4b backstop completion time that was discussed I guess actually last time during that tabletop.

Another potential backstop could be to address repetitive equipment failures that could degrade the reliability or availability of important systems from performing their intended functions.

And, regardless, it is important to note that and really emphasize that these backstops would be designed to capture the infrequent, yet potentially significant performance issues that would not otherwise be captured by the risk calculations to ensure that we take the appropriate regulatory

response.

The goal here isn't to get more whites, reds, and yellows. The goal is to make sure that we don't miss anything and that we're looking at all available information to make an informed decision.

The example -- I'm sorry. Moving on to the MD 8.3 tabletops, these tabletops demonstrated that the existing risk thresholds for invoking reactive inspections would be adequate. And the deterministic criteria already does play an important role in the determination of event responses.

The tabletops did reveal, though, that deterministic criteria are used initially for event screening and then considered again within a range of response determined by risk values, as I pointed out earlier. So these risk values actually heavily influence whether or not a reactive inspection is warranted and, if so, at what level.

So, lastly, the tabletops revealed that variations in or minor revisions to the risk models used can potentially result in a different and potentially inadequate response.

MEMBER ABDEL-KHALIK: If we look at the first bullet, if everything sort of comes out green or if the majority of things turn out green, how would

that sea of green help you?

MR. FRAHM: Well, for one, they did not in this case. For the event responses, we actually were able to hit AITs and more intense follow-up inspections.

MEMBER ABDEL-KHALIK: Comparatively speaking, these are very severe scenarios.

MR. FRAHM: But keep in mind, too, these are events, not inspection findings. So they're not color-coded with green, white, yellow, red. So we would go to that chart that Don just pulled up.

Based on the CCDP, we would see where we fall in the chart and then pick from available options based on the range.

MS. FRANOVICH: This is Rani Franovich.

If I could just add that I think that that is the basis for the staff's recommendation that we not let the risk values overly influence our decision-making with respect to regulatory response? So you'll see in the staff's proposal that because we consider these to be heavily influenced by risk values, the deterministic considerations are going to help us achieve an appropriate regulatory response in a risk-informed framework if that helps answer your question.

MEMBER ABDEL-KHALIK: I fully understand. I am just sort of concerned that this would render this process irrelevant because you always fall back to the deterministic back side.

MS. FRANOVICH: Potentially. But the tabletops did not indicate that that would be the case. The tabletops indicated that risk thresholds would, similarly, be tripped using realistic scenarios from events we responded to with the current fleet and translating those to comparable scenarios for a new reactor design. white or green, depending on the reactor design

MEMBER ABDEL-KHALIK: If you look at the example that we just looked at with the vessel head degradation, the comparison was that it was red.

MS. FRANOVICH: Agreed. Agreed.

MEMBER ABDEL-KHALIK: It turned out to be either white or green depending on the reactor design. So you're not comparing apples and --

MS. FRANOVICH: So let me try to explain. It's a very good question. When it came to the management directive 8.3 reactive inspection decisions, all of the scenarios we looked at, as I recall, indicated that we would, similarly, have a reactive inspection for a new reactor.

The vessel head degradation scenario was one that was used for significance determination process, --

MEMBER ABDEL-KHALIK: Right.

MS. FRANOVICH: -- not management directive 8.3. But we recognized from that example the potential that the risk values would not lead us to the appropriate regulatory response, which is why we think those deterministic considerations will potentially be helpful to us but not always.

There will be times when the risk thresholds will also get us there based on the tabletops for 8.3. Based on the tabletops for SDP, we recognize that it may not always get us there.

MR. FRAHM: Much less frequently and less likely.

MR. DUBE: If I could generalize, it seems, the numerical risk thresholds seem, to work very well for mitigating system failures, let's just say, for SDP but not so much for barriers. And that's where often found the disconnect, for significant barrier degradation.

MEMBER BLEY: Barriers are your defense-in-depth.

MR. DUBE: Yes, right.

MEMBER BLEY: So that's not a surprise.

MR. FRAHM: And that's why we --

MEMBER BLEY: I guess I am a little uncomfortable with the --

CHAIR STETKAR: I would like them to get through that. I would like them to get through the MSPI because there are a couple of examples. And then I think we should come back and discuss some of this a little bit more, only because it's only a couple of slides and the MSPI is a little bit different than everything.

MR. FRAHM: Okay. It is actually the most straightforward.

CHAIR STETKAR: That's why I wanted to get through these two slides, so we can come back and have a --

MR. FRAHM: Just real quick, the conclusion on the MD 8.3 was that the contribution of the existing criteria could be modified, as we discussed, or new criteria could be developed similar to that that we presented on a previous slide for the SDP.

So, to go through the case study example, this really demonstrates the over-reliance on the risk numbers. And that over-reliance might put us in the

wrong place and not result in the adequate regulatory response that we would expect.

So in this example, for steam generator tube rupture in AP1000, we originally used a submitted Westinghouse PRA. And that indicated that only a special inspection could be performed in accordance with the guidance because it fell below the 1E-5, which did not take you up into considering an AIT.

And then for the ROP tabletops, though, we actually used the more conservative SPAR model. And that resulted in just crossing that threshold so that we would be able to consider an is or an AIT based on the uncertainties and the deterministic factors.

So this over-reliance on risk really could put us in the wrong place. And there's no real reason for it.

CHAIR STETKAR: I am going to challenge you really strongly on this one because I read those things and neither of the PRA models have been reviewed for technical adequacy for this application. You're comparing results from two preliminary unreviewed, incomplete PRA models and drawing some fundamental conclusion about the adequacy of risk indices based on that comparison. And that is not fair. It's not valid. It's not fair.

One must presume that any risk models -- and this comes back a bit to what Dick had asked -- any risk models used for input to regulatory decision processes ought to, ought to, satisfy some minimal quality requirements. They ought to be consistent with reg guide 1.200, as will every PRA that is produced prior to fuel load for the new reactors, as would I hope, although they are not, the SPAR models.

So comparing two numbers that come out of two equally bad PRAs and drawing a conclusion that you can't use risk numbers for anything is absurd. And I'll use that word. It's absurd. And I want some feedback on that.

MR. DUBE: I don't think anybody from Westinghouse would defend it, but, you know, I don't -- I mean, I --

CHAIR STETKAR: Their model, by the way, hasn't been through the peer review process. So I'm not --

MR. DUBE: I mean, it at least meets their quality assurance. It doesn't meet, you know, fully met, reg guide 1.200.

CHAIR STETKAR: That's what I was talking about.

MR. DUBE: Yes. For this particular

scenario, the conditional core damage probabilities were very, very close and within a factor of two. Given the uncertainties, I would say that is pretty darned good.

Now, it just turns out because of the nature of hard and fast thresholds, one happened to be below and one happened to be above.

CHAIR STETKAR: I am using this exercise to say that we really can't rely on risk numbers.

MR. DUBE: That may be strong words. That may be strong words.

MR. FRAHM: We don't want to base our decision solely on risk numbers.

CHAIR STETKAR: We don't want to base them solely on risk numbers, but we don't want to necessarily abandon risk as --

MR. FRAHM: We're not suggesting that, no.

CHAIR STETKAR: -- we go through the process.

MR. DUBE: See, I had done similar calculations some time ago. And what we found was that the conditional core damage probability for the new reactor designs for steam generator tube rupture are about an order of magnitude lower than for the current. Just given a steam gen. tube rupture was a

condition because that is more means of mitigation. So naturally --

CHAIR STETKAR: And they tend to have more loops.

MR. DUBE: Yes. So naturally you might be doing an augmented inspection for tube rupture at a current reactor. It might fall into the special inspection only for a new reactor. But the real question is, why would you -- given a number of concerns, the fact is you have reached the reactor coolant system barrier. And this is what I said earlier about barriers.

You know, this is probably a case where you probably would expect to treat a tube rupture at a new reactor design. Probably the responses should be the same as a tube rupture at a current plant. I mean, you've got an off-site dose release, you know, maybe not a significant exposure, but people get upset about a little tritium in the air. And you would expect the staff's response to be virtually identical and probably should override the calculated core damage probability.

CHAIR STETKAR: And that is fine. And I'm not arguing with that. All theatrics aside, that is consistent with the basic framework in reg guide

1.174, the multiple defense-in-depth barriers and so forth.

On the other hand, using examples like this to say that we don't have any confidence in risk numbers or that we ought not to rely on them is pretty condemning of the whole notion of quantitative risk information as not the basis, the basis, but input to that decision process.

MS. FRANOVICH: This is Rani Franovich.

Let me interject here and be very clear. I don't think that the staff has expressed a lack of confidence in the RITS numbers. They are what they are. What the staff has articulated is they may not get the regulator to the right level of response if that's the only consideration.

CHAIR STETKAR: Okay. I am sorry, Dennis. I cut you off, as usual.

MEMBER BLEY: Yes, you did. You know, I have read that as well. And hearing the discussion here, I don't care what you say. The way it's written, the way you talk, it sounds like this over-reliance is really saying "We don't trust this stuff much at all. Oh, and sometimes it works out that the RITS numbers are helpful."

Now, most of the time it does. I think it

would be very helpful. And I think it links to everything else that's out there to tie this to barriers. I think barriers are there for defense-in-depth.

CHAIR STETKAR: Right.

MEMBER BLEY: That's where they come from. And the reasons you want to elevate an example this is because you are breaching a barrier.

CHAIR STETKAR: Right.

MEMBER BLEY: That's perfectly reasonable, but that doesn't --

CHAIR STETKAR: That doesn't condemn the risk values.

MEMBER BLEY: Yes. And you say it's not condemning, but the tenor of this sounds like "Oh, we're just getting too wrapped up in risk." There are some places where, in fact, that's not -- the defense-in-depth is the key issue. And you can find those things.

And, rather than -- I also agree a bit with John that these aren't the risk assessments you have used for risk-informed applications. Yet, you are kind of right for the specific case you are looking at, it probably won't be much different.

But, again, the barriers it seems to me

can be a challenge to the defense-in-depth. And I don't know anywhere, at least in the last many years, where anything from the Commission or the staff has backed away from defense-in-depth. Well, I know of one.

CHAIR STETKAR: Except they said for that one --

MS. FRANOVICH: Let me respond again. I think what I'm hearing is violent agreement. That's what I'm hearing because I think that is exactly the way the staff sees it.

MEMBER BLEY: It isn't the way the staff has talked to us. And it isn't the way the letter reads.

MS. FRANOVICH: Okay. Well, maybe we need to go back and look at the way we have crafted the letter, but if we can get to the conclusions and the proposal, I think we can revisit whether or not we are that far off from each other.

MR. FRAHM: This discussion actually supports our conclusions, I believe.

MS. FRANOVICH: Yes.

CHAIR STETKAR: Let's see if we can get through the MSPI ones. And then I do want to come back to this notion because I want to ask you a

question that is sort of an overriding general question before we get into the details on the letter, you know, the Commission paper itself.

MR. FRAHM: And I can't find it in the paper itself, but I know it was in the minutes or I believe it was in the minutes from the meetings that we did point out that the PRAs had not been QV&Ved except --

CHAIR STETKAR: No. I believe it's in the paper itself.

MR. DUBE: In the paper itself?

CHAIR STETKAR: Yes, yes.

MR. DUBE: Okay.

CHAIR STETKAR: Yes. They just weren't called --

MR. DUBE: I'll pull that out.

CHAIR STETKAR: -- preliminary models or something like that.

MR. DUBE: Right, right. Okay. So, moving on to MSPI, these case studies probably, much unlike the others, actually showed that the MSPI is really not adequate. And it wouldn't be effective in determining an appropriate regulatory response for active new reactor designs and that it wouldn't be meaningful at all or possibly even possible for

passive systems and passive designs. We do have numerous case studies that demonstrate it as shortfalls.

CHAIR STETKAR: Is that mostly because of the higher degree of redundancy in the active system designs that don't exist? I mean, the current index was based on essentially operating reactors with two trains of equipment, right?

MR. DUBE: Right. My uneducated answer would be that sounds right to me, but I would --

CHAIR STETKAR: That was the sense.

MR. DUBE: The real reason for the passive is, I mean, when you have --

CHAIR STETKAR: No, no. The active stuff. I understand the passive.

MR. DUBE: Yes because you have so many more trains in the active systems.

CHAIR STETKAR: Yes. Okay. So it's just that the index didn't anticipate more than two --

MR. DUBE: Yes. I mean, I was one of the authors of the current approach in the MSPI, quite frankly. And we went to an unbelievable pilot project with 20 plants. And we did benchmarks. We did Monte Carlo simulations of projecting years into the future. And we instituted a backstop. And these things were

fine-tuned in some ways against the current risk -- risk profiles of the current fleet with core damage frequencies upper 10-6 to 10-4. And, you know, we didn't look at the 10-8 and 10-7 core damage frequencies.

And the number of failures to reach the threshold of white is inversely related, inversely proportional almost to the core damage frequencies. So in round numbers, if it's typical operating plant is three failures to white, that's a reasonable thing. A new reactor might be 30 to 300 failures to white. Well, that one before you get to 30 failures is a --

CHAIR STETKAR: I was just curious. My intuition was it had to do with a degree of redundancy.

MR. DUBE: Performance issue.

CHAIR STETKAR: Okay.

MR. FRAHM: Okay? So we did note that the existing performance limit or backstop could potentially be further leveraged for the active new designs, but we really don't think that is going to get us where we need to be either.

We did conclude that alternate PIs in the mitigating systems cornerstone could be developed. If you remember from the earlier chart, we have PIs and

inspection findings. So we could come up with alternate PIs and mitigating systems and additional inspection to compensate for the lack of insights that we have currently gained through the MSPI where we meet the goals of the mitigating system --

MEMBER SKILLMAN: Why wouldn't the treasure chest of information that would guide this come out of your maintenance rule components? I mean, where you're tracking your key components, you mentioned the five for your MSPI. Your maintenance rule ought to push out in the A1 your nonconformances. And those ought to come up just like a radar blip. And so your idea that it would take 3 to 30 or 300 in order to cross this threshold, it would seem that there's an abundance of information that comes out a whole lot sooner.

MR. DUBE: Yes, you are right. Long before you reach an MSPI threshold, you will reach some other action level in online risk monitoring or containment group, yes.

CHAIR STETKAR: But that is simply implemented under the plant-specific Corrective Action Program, right?

MR. DUBE: Yes.

CHAIR STETKAR: It doesn't necessarily --

MR. DUBE: Right.

CHAIR STETKAR: -- reach the staff attention unless you do an audit or something or you trip over the index.

MEMBER SKILLMAN: Maybe the staff ought to be looking at something more than just this.

(Laughter.)

MR. DUBE: Yes. That is on the list. That is exactly right. That is what you will see in the recommendations.

MEMBER SKILLMAN: Thank you.

MR. FRAHM: And isn't the safety system functional failure PI -- would that possibly catch --

CHAIR STETKAR: Well, that would again have to have multiple -- it's not going to catch your 25 failures of an individual diesel, right, because that doesn't take up a whole bunch --

MS. FRANOVICH: Yes. The SSSF PI really today only has a green to white threshold. They can have 100 failures of an SSFF and not trip a yellow threshold.

But, back to the maintenance rule, that would be an input through the inspection program. And it's been my experience that maintenance rule inspection findings are never more than green. And so

if you have a lot of unavailability or excessive failures of risk-significant components, that input to the action matrix will not generate a regulatory response given that significant lack of availability or reliability.

So we might be able to look at maintenance rule for insights for how we can better inform the action matrix with that information. But that will be something the staff will have to work with industry to explore as an alternate to MSPI for new reactors.

Does that make sense?

MEMBER SKILLMAN: Yes, it does. Thank you. I understand the words. What I'm really thinking about is in these new reactor designs, particularly where there is so much equipment. The CDF is so low that it can be the sense of where everything is broken. I'm not worried about it.

MS. FRANOVICH: Right, right.

MEMBER SKILLMAN: Whereas, if the bat came out at the second A1 component, an individual could say, "Hey, we had better pay attention."

Now, I think in the industry's defense, they are already doing that. The better-run plants, the better managements understand anything in A1 is an immediate attention item.

But if the real issue here is to raise the threshold or raise the bar in terms of safety, then a thicker magnifying glass and what is popping out of maintenance rule does it for the industry. And it does it for the NRC. At least that is this man's opinion.

MS. FRANOVICH: I understand.

MEMBER SKILLMAN: Okay. Thank you.

MS. FRANOVICH: Thank you.

MEMBER SKILLMAN: Thank you.

MR. FRAHM: Okay. As I said, there were several examples in MSPI that demonstrated the shortfall in providing any insights, really, into plant performance. The two examples that we put on this slide are that it would take greater than 25 emergency diesel generator start failures or greater than 25 run failures for the EPR to exceed the green-white threshold. And then it would take 12 failures to reach that performance limit or backstop that exists in MSPI.

And in the second example for an APWR, it would take greater than 14 turbine-driven emergency feed pump failures or greater than 25 motor-driven pump failures to exceed the green-white threshold. And, again, it would take at least six failures to

reach the performance limit using the existing MSPI backstop.

So, for a little perspective, the last bullet, just off the top of our heads, we would say for current operating reactors, about two to five of these failures would typically result in a white indicator and trigger a regulatory response for the existing fleet.

So that's --

MEMBER BLEY: Maybe I'd better wait until you get to your conclusion.

MR. DUBE: There is no conclusion here.

MEMBER RAY: Okay.

MR. FRAHM: We will talk about those later today when we talk about the paper.

MEMBER BLEY: The smaller number doesn't seem to get at the issue. You talk in some of your options of needing inspections for the passive functions. What it seems to me one needs to think hard about is what would that mean. Anything that would degrade any of the passive protection, which sometimes is a very delicate -- the dynamic balance sometimes that's something like maintaining the film on the outside of the containment.

All of these things that are small margins

to the way they're designed, looking for anything that could disrupt, though, it seems to me what we really ought to be thinking about ways to do that to look for those things.

I'm not sure saying, well, you need 25 failures. Let's look at two or three. I don't think that helps with the issue. I think the issue is more and you've got it there, but it's only in little brief phrases like inspections of the passive process, that sort of thing.

MR. FRAHM: Right.

MEMBER BLEY: But it's that stuff that could either break down redundancy or could defeat some of these passive features over time, but we need to think real hard about what do you need to look at to see if there is a challenge to those things. This one doesn't help me very much.

MEMBER SKILLMAN: I like the way into this because it's a thought that's been resident in my mind for a couple of years. With the part 50 licenses, we're focused on SSCs, structures, systems, components, things that do something for us; whereas, in the fuel plants, they focus on IROFS, items relied on for safety.

And it seems to me that with the passive

plants, we're almost in a situation where we need the SFCs plus perhaps a subset of IROFS. Maybe the film is an IROF. And maybe there's a hybrid going forward where we identify those safety features on which we depend, translate those to the components that have as another layer, perhaps as important as the structures, systems, and components, some set of IROFS that are measurable, that are actually defined in the license but somewhat subtly now because of the way the part 50 licenses are written or part 52 license or design certifications that will be provided. It seems like maybe as we move ahead, there is room in a new generation of reactors for accommodating IROFS that kind of cut between the analysis, the SFCs, and some thermal hydraulic feature or some physical feature that is just as important as the component that isn't the component, like the film.

MR. DUBE: Right, right. I mean, that is why. I mean, we have said that more or less in so many terms that for the passive plants, where you have a system, you have a tank and an explosive valve into the core.

The only thing you can monitor is the explosive valve. And you can't test that online. You can only test that sample on --

MEMBER SKILLMAN: Right. You can see if that were to --

MR. DUBE: So reliability is meaningless. And I'm not sure even availability because you don't track on -- you don't remove this from service online typically. And, as you pointed out, here it's a degradation of some of the physical parameters. you know, friction coefficient, check valve resistance, so on and so forth, that you are concerned because that is what give the good performance.

So the mitigating system performance index paradigm of looking at reliability of diesels and motor-operated valves, air-operated valves, pumps doesn't fit passive plants. And that's why we're saying that's exactly the point we're making, which is we need some other kind of indicator. For the MSPI as formulated, current active plants, it doesn't carry over to the passive plants.

MEMBER SKILLMAN: I respect what you say. And I understand it. What I'm suggesting is that maybe we ought to be working with some form of IROFS or translation of the IROFS over into this new environment that we were moving towards.

MR. DUBE: Yes. That's the next step. Well, the next step is to get the --

MEMBER SKILLMAN: Yes. Right.

MR. DUBE: -- Commission to agree with us. And then the follow-up step will be what do we consider to replace MSPI for new reactors.

MEMBER SKILLMAN: Thank you.

MEMBER ABDEL-KHALIK: But if you look at this from a very high level, we're doing all of this to protect the health and safety of the public. The new systems are so forgiving that people can screw up without it getting to really trigger anything that would raise your concern.

So perhaps what we ought to focus on would be cross-cutting issues, rather than individual triggers. Is there a way that you can do that without looking at these individual MSPI and all of that that would never be triggered?

MR. FRAHM: Well, that is already built into the current ROP that we look at cross-cutting issues across all the cornerstones --

MEMBER ABDEL-KHALIK: Right.

MR. FRAHM: -- and we do the same for the new reactors.

MEMBER ABDEL-KHALIK: But those come into play after you trigger the cornerstones.

MR. FRAHM: But even a green inspection

finding or anything would actually trigger assigning a cross-cutting aspect that could result in a cross-cutting issue. So you would just need the performance deficiency to get you there. So I think we're covered there. That should be consistent with the current reactors.

MEMBER SCHULTZ: Are we coming to a conclusion that matches something like what Dick has proposed or suggested or is it still an open item that's just an item to be considered by the Commission?

MR. DUBE: These are just -- you know, the step is Commission agree that we need to do something different in ROP for new reactors. And we are listing these kinds of examples, some potential examples.

MEMBER ABDEL-KHALIK: Backstops.

MR. DUBE: They tell us "Be specific" or they say, "Okay. Now go off and do it." So, I mean, it's --

MR. FRAHM: So that will be in the --

MR. DUBE: It's like a two-step process.

MR. FRAHM: The next several years of developing the process, et cetera, with interaction with industry and other stakeholders.

MEMBER ABDEL-KHALIK: But it would be

heavily dependent on these backstops.

MR. DUBE: It could be more dependent, yes, than the current fleet.

MR. FRAHM: Certainly more than the current fleet. In fact, we would -- well, I am kind of jumping ahead to our conclusions and recommendations, but we would envision that we would incorporate these deterministic backstops into our existing guidance and that existing guidance applies to the current fleet as well as the new reactors.

But we wouldn't expect the current fleet to reach the deterministic backstop. They'll get there by tripping risk thresholds before they ever get there. So we are just looking at a single process, which I think is a little bit different from where we were a year or so ago when we were considering what to do here.

That's all I really had for the discussion of the ROP case studies. As I said, we'll talk a little bit later about the options and recommendations in the paper, but you will see that they are pretty closely aligned with the conclusions we had so far here.

MEMBER SCHULTZ: I wanted to revisit Said's comment related to the cross-cutting issues

because perhaps there's a rationale, even though the current threshold for consideration is low for the current operating fleet. If you had a fleet of passive reactors, perhaps there's a rationale that says the thresholds should, in fact, be lower.

MR. FRAHM: I think I would put that on the things to consider list going forward, but I don't think we have -- you know, we are not there at making decisions at that level.

MEMBER SCHULTZ: You know, I understand, but it is just something to keep in mind. If you look at the difficult events --

MR. FRAHM: Right.

MEMBER SCHULTZ: -- that the industry has experienced, it has been in the cross-cutting issues, the complacency, and self-worth. And there is some tie-in to the passively safe reactor fleet and perhaps a lack of appropriate focus associated with cross-cutting issues.

MR. FRAHM: And they may become that much more important in the --

MEMBER SCHULTZ: You see them in the threshold as what it is because you are going to have things that happen. You do have things that happen in the fleet that yes, it can be a simple green

inspection finding, but more likely today it is going to be a wide area finding that is going to definitely put an operator in that position.

MR. FRAHM: Right.

MEMBER SCHULTZ: So it's something to consider.

MR. FRAHM: That's a good point. And I wrote it down on my things to consider going to forward. Thanks.

MEMBER SCHULTZ: Thank you.

CHAIR STETKAR: I would like to ask both of you since before we discuss the actual paper and the kind of details in it -- I wanted to save it until now because the ROP gets sort of the most attention in the paper if I can characterize it that way. And I want to understand the ground rules here or your interpretation of the ground rules from the SRM.

Did you look or think about things, rather than -- let me preface this. My impression of many of the sited perhaps deficiencies if you want to call them that in the numerical risk metrics, for example, SDP or other issues, may derive from the fact that the current guidance is based on absolute measures of delta core damage frequency and delta LERF.

It says that if I have a delta core damage

of 1.0 times 10-6, I transition from green to white. 1.0 times 10-4 gets me from yellow to red. Those are absolute measures. They're not relative measures. I've suddenly been asking you questions about what are percentages of changes.

And in some of our discussions, we ask you all, how bad would things need to get? And all of your answers have always been, well, if it increased by a factor of two or three or five, that might get our attention, ten percent increase. Those decisions and those discussions are not based on absolute values. You make decisions about importance based on relative measures.

The exercise is something you felt pretty comfortable with under 10 CFR 50.69 based on relative metrics. You found those to be reasonably good. We had this discussion about should there be a sliding scale.

MR. DUBE: Yes.

CHAIR STETKAR: But a relative measure of risk significance, you know, you felt pretty comfortable there. Did you look at under the ROP process, in particular, changing that -- and the extension would be to reg guide 1.174, which is kind of the fundamental basis for all of this -- changing

those absolute metrics on the ordinate to relative changes? And how would that affect some of your conclusions, some of your numerical conclusions, from your tabletop exercise? Instead of saying, "Well, you know, the delta CDF was 2.3 times 10-8," if that were a factor of 2 in the core damage frequency, you might have reached a different conclusion about how useful that numerical risk metric would have been, that relative change. Did you think at all about that? And if not, why not?

MEMBER ABDEL-KHALIK: Is that option at all open? The Commission said --

CHAIR STETKAR: It is an interpretation of what the SRM --

MR. DUBE: The answer is no, it is not.

MR. ADER: I am going to disagree. The answer was we thought about it a lot when we were here the first time on the first paper going up asking the Commission for guidance.

MR. DUBE: Yes, right.

MR. ADER: We talked about we could make it relative. We could make it absolute, some combination of did they want us to go off and develop that framework. They came back and said, "We're reaffirming the existing metrics. Now let's go

tabletop to see if you can definitely using realistic examples tell us that there will be a significant decrease in public health and safety."

So in the tabletops, we didn't, but if we had considered them --

MR. DUBE: Right.

MR. ADER: -- that they do that when we put the first paper up, then that would have been an option we might have pursued if we had gotten a different answer. So this one we felt more constrained with reaffirming 1.174 guidelines, tabletop -- we have gotten a lot of value out of the tabletops, too. So we did accomplish a lot of our objective.

CHAIR STETKAR: I guess I hear you, Charlie, but this morning when I was discussing GEH's use of different relative measures, you went to an appendix in 1.174 and said, "Well, we will accept the sliding scale on those measures because sort of the spirit of that appendix is that we could use different scales as a function of core damage frequencies."

MR. ADER: I almost interjected there because the paper if you remember the discussion we had is we had a 10-4. We had the 1.174 metrics for the existing processes. Everything tends to point

back to reg guide 1.174: 10-4 CDF, 10-5 LERF.

For new reactors, we had been evaluating things against a 10-4 CDF. And there was this 10-6 LRF undefined that we had been using to judge the enhanced margins to safety, enhanced severe accident performance.

So we had this quandary. And that's why we went to the Commission the first time to say, "We have some disconnect here." We did not have the 50.69 relative metrics, the Fussell-Vesely is part of that discussion --

CHAIR STETKAR: Right.

MR. ADER: -- because this was really what is the level of safety, not what is the relative importance of equipment at some lower level.

CHAIR STETKAR: Right.

MR. ADER: That was --

CHAIR STETKAR: And I understand that part of the discussion.

MR. ADER: -- the exercise.

CHAIR STETKAR: We're talking here at a different level in terms of a staff set of recommendations for Commission consideration that in many cases, at least as I read the paper -- and we'll talk about the paper -- says, "Well, numerical

quantitative risk information is of limited use for new reactors because if we strictly apply the numbers that are in reg guide 1.174 and in the guidance for the ROP, we don't have enough fidelity to make the types of decisions that we want to make that we feel are necessary."

And my question is, well, I ran out a bunch of examples. And transitioning from an absolute change in core damage frequency as a measure of fidelity to a relative measure of change as a measure of risk significance might very well get you through that nature of the metrics don't give you enough fidelity. The quantitative metrics don't.

Now, determining what those relative -- you know, do you get interest of a factor of red? When core damage frequency increases by a factor of 100 or 50 or 30, that's something that would need to be worked out. But it's sort of a different framework of thinking about what is risk significance. Is it an absolute change or is it a relative change from the standpoint? It could also preserve this notion of greater margin for reduced absolute levels of risk because you can have a sliding scale.

You know, someone whose baseline core damage frequency right now is 10-4. If you double it,

you're going to get into the red regime. Well, fine. You might decide that you could allow the baseline core damage frequency for a 10-8 plant to increase by a factor of 100. So that gives you greater flexibility. However, it would require kind of changing the framework of 1.174 to measure risk significance always on a relative basis to a baseline risk.

That isn't quite the same as 10-6 LERF or 10-4 as linked to the quantitative safety goals or necessarily even the basic framework of 1.174 because perturbing --

MR. ADER: From a licensing side -- and Don can disagree with me this time. I think from a licensing side in the tabletops, we felt there were enough controls in the process for the new plants, some of them -- tier 1 stuff is not going to be changed without tier 2*. Going through what's realistic, what was viewed as -- we could come up with scenarios that would be viewed as unrealistic.

CHAIR STETKAR: Sure.

MR. ADER: And the SRMs in the discussion --

CHAIR STETKAR: Yes.

MR. ADER: -- be realistic and be

definitive that you can show us that there will be a significant decrease in enhanced safety. We didn't feel we could pass that test. There are enough controls there we think that give us assurance that there won't be significant decreases.

CHAIR STETKAR: I think from a licensing perspective, I certainly understand that. However, in particular, for the ROP, now you're transitioning in terms of using these metrics to inform agency decisions regarding levels of inspection transitions to enhanced inspection. And to a lesser extent, you know, I will bring in populating things like the RTNSS list and the DRAP list, which aren't necessarily a strictly pass/fail safety, licensing issue, but they do have effects going forward in terms of how you treat equipment.

MR. ADER: Yes. We have identified some issues in implementation of some of these for new reactors and some things, like RTNSS, that may not be in the guidance for current initiatives that we need to figure out where that fits.

CHAIR STETKAR: Yes.

MR. ADER: Tabletops have informed us of some of the other issues we need to deal with going forward. At some level, you will have a PRA that is

updated.

CHAIR STETKAR: Yes.

MR. ADER: So if they make significant design changes over time, you would expect to see that PRA -- for new plants, its regulation, you would expect to see changes there. If there's a lot of unavailability, they would have to factor that. It's a lagging indicator.

CHAIR STETKAR: Sure.

MR. ADER: It does provide a little bit more of a longer-term backstop. There will be a reasonable way of measuring some of the impacts of some of this stuff, which gave me a little bit of a warmer feeling.

CHAIR STETKAR: It is. There are some of those, but it kept hitting me here, especially because I think some of the words in the paper -- that's why I kind of wanted to bring it up before we get into the paper discussion -- that the paper seems to say, "Well, we don't have confidence in quantitative risk measures, in particular." Again, the ROP kind of stands out in my view here -- I might be wrong -- as the issue of most contention.

As the quantitative risk measure is providing adequate -- and I'll call it fidelity, not

enough information for us to make adequate decisions. And, therefore, we need to fundamentally rethink certain elements of the reactor oversight process to insert more deterministic backstops, different -- I don't want to -- different indices, different ways of thinking about things to address these deficiencies, if you will, in the quantitative metrics.

I'm not arguing. And I'm not saying that those might not be necessary for things that Dennis has mentioned, you know, barrier to performance or some of the more really difficult, thorny issues of purely passive safety functions.

However, there may be ways to change the nature of the way those metrics are calculated and treated to improve the fidelity in the quantitative measures that might alleviate some of the concerns about, gee, it didn't address this. So we need to really think about this issue in the sense of what type of deterministic backstop or what other types of guidance do we need for reactor oversight or some other kind of decisions going forward.

MEMBER ABDEL-KHALIK: I guess they are taking this approach perhaps based on an interpretation of the wording.

CHAIR STETKAR: I think it is. I think if

you go back to your reg guide 1.174 slides, it sounds like a little interpretation that the staff said the Commission wanted to retained precisely those numbers.

MEMBER ABDEL-KHALIK: Right. I mean, if I were to just read this, "Continue existing risk-informed framework pending a series of tabletop exercises that test existing guidance," does that mean go test this, come back, we may change our mind and let you use a different framework or use a relative risk metric, rather than an absolute risk metric because you are interrupting this as, okay, we are going to stick with what we have and since it didn't work quite as well as we expected based on these tabletops, we are going to create these deterministic backstops. The Commission may be totally open to the idea of using a different approach.

CHAIR STETKAR: We can't speculate on that. It's not job to --

MR. DUBE: Charlie and I sat before the Commission on October 14, 2010. I remember very well. We didn't convince them of anything whatsoever, despite white papers and the Commission paper and so forth. And, you know, it came back pretty firm, you know, did not change the safety goals, safety performance expectations, goals, associated values,

key principles, and quantitative metrics. So, I mean, we were pretty faithful to that.

We did do one exercise --

MEMBER ABDEL-KHALIK: Because they didn't have the benefit of these reports.

MEMBER SHACK: I suppose that's what we're suggesting, that there's another option, one option to fix the --

CHAIR STETKAR: I mean, the question is, could there be another option in your mix that elaborates on the use -- you know, given the experience from the tabletops, I'm aware --

MR. ADER: We did -- Don had written a white paper which was sent up, which I think you had the benefit of one of the other ones, that went through similar types of examples and showed the types of changes that could be seen with the different risk --

MR. DUBE: Yes. We had an MDA .3 and an MSPI and a --

MR. ADER: So some of that information was there. Our interpretation of that was also -- my -- I'll speak for myself -- was flavored by reading the vote sheets, reading the various versions of the SRM that was being crafted. I'll say cobbled together

because I'm sure they don't fit crafted. The vote is just go option 1.

MR. DUBE: Yes.

MR. ADER: Some do option 2. So it was not option 3 to 2 someplace. You know, I'm saying the way we approach the tabletops and the interpretation from my point of view was all of this kind of taken together, what we saw, where they were going.

CHAIR STETKAR: Okay. I mean, none of us can -- you had the benefit of having much more discussion certainly than we have with the Commission and perhaps additional insights.

The discussion, many of the words that you can read. You know, I understand maintaining the safety goals. I understand maintaining -- if you use the words "risk metric" as core damage frequency and large early release frequency for the moment, changing the way you measure changes, significance, doesn't affect that.

I'm not saying "Change it to 'fatalities'" or anything like that. Maintaining the notion of current safety margins versus safety margins, you know, increased safety margins for new reactors but still allowing flexibility doesn't say how you do that.

MS. FRANOVICH: Yes. So if I could repeat back what I think I've heard for the last hour or so, it's that there may be a fourth option on the table. And it's not about changing the risk metrics that we're using. It's about using those risk metrics in different ways to look at relative decreases in safety as a function of performance.

CHAIR STETKAR: And I think you have characterized it very well. And I would call it a different way of quantitatively measuring risk significance, --

MS. FRANOVICH: Yes.

CHAIR STETKAR: -- not on an absolute value but a relative value.

MS. FRANOVICH: Well, we don't go all the way back to deterministic. We still use some qualitative methods that yield more understandable, predictable, transparent outcomes that have a nexus to risk.

CHAIR STETKAR: You may still need -- sure, this notion of preservation barriers and perhaps special considerations for truly passive functions may need some attention in reactor oversight process, for example. But there may be some usefulness. And I don't know in terms of the time you have available of

at least examining the results of some of the tabletops and seeing how your conclusions about the usefulness of the numerical metrics might change.

As I said, I ran a couple of examples. And I could get a lot more into the white and yellow areas, where you were saying they were concerned because they were uniformly green. Using kind of an arbitrary scale, I'm not going to propose a scale here. Obviously that is not what we do, a percentage change.

MS. FRANOVICH: Right, right. And so I think that is a concept that we really did not consider that I think could still fit within what the Commission established as its expectations. And we can go back and consider another option, the fourth option, if you will, that really kind of applies that concept. I think it's something we can definitely work with.

MEMBER RAY: What is the motive, John?

CHAIR STETKAR: Well, the motive in -- other than, you know, I am a champion of risk assessment, the motive is to reinforce. If you go back, some of the basic principles -- and I'll come back. I'll come back to ROP because I see this is the area.

The basic principles are maintaining a consistent use of -- you just passed it -- that third bullet there, "Risk inform a process and risk information in a quantitative sense," adds input to that process consistently, regardless of the plant designs. We use it very effectively today for the current operating fleet. We all understand how to do that. And, regardless of who divined why 10-4, 10-5, 10-6 ought to be those thresholds, it works okay for the current operating fleet.

The sense from the current paper is that, again, in the reactor oversight process that it doesn't work so well there. And so my motivation is to say, can we think about how those metrics are calculated to maintain sort of that same level of confidence in the quantitative risk input, recognizing that for some of the new reactor issues, you may still need --

MEMBER RAY: There is always a tension between why should I invest in one of these exotic plants -- and having gone through the AP1000, I'll call it exotic. I think I can do that. Why should I do that if my life isn't going to be made any easier as a result?

CHAIR STETKAR: Oh, I want your life to be

made easier.

MEMBER RAY: Well, I'm wondering. That's why I'm asking the question.

CHAIR STETKAR: I want your life to be made easier. And I want the reactor oversight process to maintain its objectivity and its quantitative nature. And I'm not saying that if you have a -- let's take a current operating plant with a 10-4 core damage frequency. If it has a change that increases core damage by 10-6, it transitions from green to white. That's a one percent change. I'm not saying that you transition from green to white with a one percent change for your reactor. You might not transition from green to white until you had many, many percent change.

MEMBER RAY: So you get the benefit. We've had this discussion before. I just wanted to hear you say that you acknowledge what you just now did, which is you had to take away --

CHAIR STETKAR: No, no, no.

MEMBER RAY: -- the advantages that would lead me to make this kind of a commitment in the first place.

MEMBER SHACK: But there is a tension between that and the --

MEMBER RAY: Of course.

MEMBER SHACK: -- unless you are going to maintain the enhanced safety. While they are going to maintain the enhanced safety by applying these sort of deterministic things -- and I guess we could argue we could do it much more directly by deciding just how much enhanced safety we wanted to maintain and building that into the process.

MEMBER BLEY: Let me try something that is bothering me a little bit. I mean, there are a lot of places in today's whole day's discussion where I'm pretty well aligned with what John said, but the thing that bothers me a little bit here is if we are thinking of this to address the issues where we have extensive redundancy -- and losing two or three things really doesn't change the calculation of overall risk much, but you could see it a little bit -- I think what we're doing in this case by suggesting the relative risk, I think, is actually trying to using that relative risk as a surrogate for something else. And the something else is -- maybe there's a better way to deal with this something else.

Now, if we have lots of redundancy, we really don't care if we lose a couple of things. It really doesn't change the overall very much. But if

those things that we have lost are symptoms of something that would create dependency across all of these, symptoms of a potential common cause failure, you really want to get them flagged.

Now, one way to flag them is to use a small change as a surrogate and begin, but the other way is to somehow -- we've already got a catalog of the things that mostly contribute to common cause of mechanical equipment anyway. For the kind of failures that crop up that would threaten that redundancy have a special evaluation of this could be quantitative, could be qualitative, but at least in this one area where we are looking at extensive redundancy, I think our idea of relative risk is really a stand-in for something that you might do better in another way because we're really worried about losing all of the redundancy or a lot of it.

CHAIR STETKAR: And that is essentially -- that is the MSPI, is a clear example of that. I think that --

MEMBER BLEY: The fact that we have changed, but the overall risk doesn't change much. Really, if that is all that's there, it isn't a big change. It's probably not worth worrying about because a symptom of something that could defeat the

redundancy even greater. You really want to be able to flag that.

CHAIR STETKAR: Yes.

MEMBER BLEY: I don't know if this is the best way to flag it or not. It's a way.

CHAIR STETKAR: I think it's a way that it doesn't completely resolve the problem. It is a way I think that resolves issues related to single-component importance and maintains kind of a consistent treatment across, you know, kind of a broad variety of plant designs or even plant-specific issues. It doesn't fully address the notion of the thing I'll call common cause failure. On the other hand --

MEMBER BLEY: My worry is that it wouldn't put the focus of the inspection on the right thing if it isn't couched in the right way.

CHAIR STETKAR: On the other hand -- well, the good news is you're going to have two things that happen when you start to think about common cause failures today.

MEMBER BLEY: Sure do. And if you only have two, then it's real easy to --

MEMBER ABDEL-KHALIK: Your argument about perhaps the lack of utility of the relative risk

measure implies that there is a cliff-edge somehow.

MEMBER BLEY: It implies there is an underlying cause that could -- I wouldn't call it a cliff edge.

MEMBER ABDEL-KHALIK: But it would imply that --

MEMBER BLEY: It is sort of like one.

MEMBER ABDEL-KHALIK: -- a gradual change in risk --

MEMBER BLEY: If they're not focused on what is going on underneath in a much shorter time than would expect, we might see --

MEMBER SCHULTZ: I don't think the relative approach meets the expectations that were set out in the SRM by the Commission. The SRM is saying what is existent today is okay and what advantages we have from the new plant design, we want to be sure we license a plant that's got better performance and better designs. It's going to have more margin than current safety limits.

We want to come up with a system that maintains that. But we will give the operator of the facility the allowance to have operational flexibility within that. That's what Harold was getting to.

And so we don't want to set up a system

that takes that away at the same time the plant was licensed to have additional safety margin. And so the system needs to assure that that safety margin is not eroded on a design basis.

MEMBER SIEBER: That's the key.

MEMBER SCHULTZ: So I don't know that the relative approach captures that exactly.

CHAIR STETKAR: Well, I mean, we can -- I tend to think it does.

MEMBER SCHULTZ: I could tell.

(Laughter.)

CHAIR STETKAR: And obviously the good thing about Subcommittee meetings is we are all individuals in subcommittee, but this is certainly not ACRS. And you have the flexibility to have these kinds of discussions in the subcommittee meetings.

I don't think that this sort of relative measure of risk importance, risk significance solves all the problems. I do think -- and I would need to be convinced otherwise -- that it does preserve a consistent way to factor in quantitative measures of changes in the risk into the regulatory process, regardless of the issues. Again I'll bring up all of B because it's the most evident. And both maintain the desired absolute improved level of safety while

providing increased flexibility for the owner-operator in the sense that they are not burdened with excessive frequency of entering white and yellow findings because this relative -- it's a scale.

You don't say one percent change is the same for everybody. One percent change might trigger a white finding for a current operating plant, but you might not trigger a white finding for a 10-8 person until you get a factor of 10 increase. And that will take an awful lot. And, in fact, it probably would take some common cause failures to get me that big, which is a little bit of why it might be a partial surrogate for the concerns about common cause failures. If you got two out of the four of your things fail, you might see a factor of ten increase. And you might then look at those two to see, ah, is there some sort of common cause issue going on here that could have affected all four?

MEMBER SCHULTZ: But you have to design that very carefully.

CHAIR STETKAR: It has to be designed. I didn't say it was easy.

MEMBER SCHULTZ: It has to take into account, as we have discussed today, the quality of the risk assessment methodology and all of the pieces

that fit into that.

MEMBER SHACK: That's been today --

CHAIR STETKAR: That's right. You know, the agency is making regulatory decisions based on the quality of in some cases only the SPAR models. And I think it's more often a comparison between a SPAR model and an individual licensee's calculations. So that's not an issue that is unique to new reactors.

MEMBER SCHULTZ: That is correct.

MR. ADER: For what it is worth, I had the original paper with me. And it had Don's white paper attached to it, which had some options which were independent of the Commission paper. One of the options -- and this was in licensing context -- was convert to relative risk changes in 1.174. So the first go-around consisted of that, and it was raised. I mean, it was part of the package that went to the Commission.

CHAIR STETKAR: Yes. I think that --

MR. ADER: But my earlier answer, I was focusing more on the licensing side of 1.174 change.

CHAIR STETKAR: Yes. I think at this stage, all I am proposing is, does this concept merit enough attention to be added to the paper as an option for consideration by the Commission with some

supporting information?

You know, the Commission may very well come back and say, "Hey, we told you that we didn't want to consider this. And, yes, pressed us. We understand absolutely what this means." Fine.

MS. FRANOVICH: I think the biggest challenge for the staff at this juncture would be putting enough meat on the bones for the Commission to really have a good understanding of what exactly we would be proposing. We'll have to go back and see if we have enough information from the tabletops to put together a reasoned methodology that would get us to a different decision standard using risk insights, if you will.

CHAIR STETKAR: Yes. I think, again, if this is an option in a Commission paper, you certainly can't solve the notion of exactly what are those trigger points.

MS. FRANOVICH: Right.

CHAIR STETKAR: And you ought not to do that.

MS. FRANOVICH: Right.

CHAIR STETKAR: It's just a question of would a different way of measuring that risk significance --

MS. FRANOVICH: Right.

CHAIR STETKAR: -- on a relative basis provide you a vehicle that might give you more confidence in those numerical measures down where you need increased fidelity and then leave it. You know, obviously it's at the same point right now.

If the Commission said, "Well, yea, verily, go ahead and come up with new deterministic backstops and things like that," you are not proposing what those backstops are. You just signal forth you need to think pretty hard about what they might be.

MS. FRANOVICH: True. We've been able to say, you know, we're thinking they may be around barriers or common cause failure, that kind of thing. But you may be right. Just conceptually here's another option that the staff has formulated with input from the ACRS. And, you know, we would still be able to achieve the ROP fundamental tenets of being risk-informed, predictable, understandable, and objective. So if we put those parameters around it, it would give I think the Commission a better understanding of what at least we would be trying to achieve.

Biff?

CHAIR STETKAR: Biff, I was ignoring you,

but I just thought you had something to say.

MR. BRADLEY: I would like to raise an administrative point. If the staff intends to add that option to the paper, I would like to request that we go back through the public interactions because the interactions we have had so far have not included that. We would certainly have comments on that proposed approach.

MS. FRANOVICH: Another potential impediment with the time that we have. So we will have to see what we can achieve.

MR. BRADLEY: All right.

CHAIR STETKAR: And I appreciate that process, by the way. It's important.

MR. FRAHM: And I would anticipate that there would be others, too, that say that would be outside of what the SRM directed the staff to do --

CHAIR STETKAR: Fine.

MR. FRAHM: -- just a consideration, including some commissioners, you know, in reading the vote sheets and whatnot. I mean, they were very clear in many ways that the risk is what it is. And I don't think this is part of the --

CHAIR STETKAR: Is there a full Committee meeting scheduled? There is a full Committee meeting

scheduled on this in --

MR. FRAHM: April.

CHAIR STETKAR: -- April. So it's soon. We needed to do that because they certainly want a letter. And that gives us enough time to get a letter in a timely fashion for them.

I just wanted to float that. We do need to go obviously through the paper. Is there any more discussion on my ranting?

(No response.)

CHAIR STETKAR: Thank you. Let's take a break until 3:00 o'clock. We're recessed.

(Whereupon, the foregoing matter went off the record at 2:44 p.m. and went back on the record at 3:03 p.m.)

CHAIR STETKAR: We're back in session. Let's hear about the paper unless there's anything more to discuss about the -- by the way, just for the record, Bill reminded me that for clarification, when I talk about relative measures, there's -- what I am staring at here and what is inappropriate to kind of discuss in detail, the relative measure's on a log-log scale.

So they provide a -- it's not a uniform one percent change or 20 percent change or 50 percent

change across the board. They're consistent percentage changes on a log-log scale as a function of CDF or LERF or whatever measure. So if there's any concern about the motion of keeping the same one percent change, regardless of what your core damage frequency is, that certainly doesn't satisfy the conditions. And that's not what I am proposing at all, not at all.

With that, let's hear about the paper.

9. CONCLUSIONS, OPTIONS AND RECOMMENDATIONS TO COMMISSION IN DRAFT PAPER

MR. DUBE: Okay. Good. So the major conclusions were from the tabletop exercises for licensing applications, staff did not identify any potentially significant decreases and enhanced safety margins for new reactors, didn't identify a gap in the process, the guiding severe accident features that are not related to ex-vessel severe accident prevention and mitigation.

And we've got a recommendation along those lines. And the current risk thresholds are appropriate for ROP. However, the changes to the ROP may be warranted consistent with the integrated risk-informed principles of 1.174.

MEMBER ABDEL-KHALIK: That's a mild way of

putting it.

MR. FRAHM: It was high-level --

MR. DUBE: So on risk-informed in-service inspection, we talked about this, the original tabletop back in September. We found that it's a risk-neutral effect for new active plants and new passive plan, even when we did -- we did do sensitivity studies here where we use more restrictive acceptance criteria didn't change our overall conclusions.

There are numerous regulatory and programmatic controls; for example, inspection of a minimum set of weld locations, regardless of risk level. It's a dynamic process ten-year ISI program, incorporates lessons learned and updates to risk ranking. It is also very consistent with part 52 requirements for new reactors to upgrade and maintain the PRA.

On the risk-informed completion times, time and time again as we did -- well, between us and the industry participants, well over 100 case studies, the risk-informed completion time is limited to a deterministic maximum of 30 days, often called the backstop completion time, from the time the tech spec action was presented.

And voluntary use of the risk-managed tech spec for a configuration which represents a loss of total tech spec safety function or inoperability of all required trains is not permitted. And we saw these again in all of our kinds of exercises that we did.

We do mention and we mentioned in the paper we did identify some configurations of equipment outages that would represent, could represent ten years worth of core damage probability.

Repeated entry over time could increase core damage frequency by an order of magnitude or more, which could approach the baseline CDF of currently operating plants. But these are extreme stretch cases of configurations that are unlikely or unrealistic, found the additional regulatory and programmatic controls that would limit the aggregated risk increase; for example, under part 52, as we mentioned, under 50.71(h), part 52 plants that are licensed. You know, performance monitoring is required. Periodic PRA maintenance and upgrade are required.

I don't have a fourth bullet, but a point that often gets brought up is, even under the current standard tech specs, it can be in these

configurations. And what you find under risk-informed tech specs is a process and programmatic controls in place to limit the risk increase; whereas, in current tech specs, you could have these hypothetical configurations and with lesser degrees of control. So I think the staff feels very comfortable with risk-informed tech spec 4b.

So we conclude there are no substantive changes to the methodology. And we get some implementation issues but, as I said, no fundamental change to the methodology.

MEMBER SCHULTZ: Don, on that slide, ten years worth of core damage probability. So that's going back to a relative measure there. Are you not?

MR. DUBE: The point was, you know, in a hypothetical situation where a lot of equipment out of service -- and these were like in three-train plant difference, two trains out of service. And they find themselves in an unplanned outage of a third train. I mean, those circumstances, even if you're in there for a short period of time, the increase in core damage frequency and the time duration, the integral could give ten years worth of core damage probability.

And the point I made was it's extremely unrealistic. And this situation can occur in a

current plant under current tech specs. But that's what that meant. We weren't talking relative here necessarily.

CHAIR STETKAR: Don, one brief editorial comment. You may just want to make a note of this. It could be caught in the final version. But in appendix A, where you discuss the issue of 4b, it says, "MHI Application for RITS 4b and the US-APWR Standard Design and Luminant COL Application for Comanche Peak Units 3 and 4."

In particular, when we assess, in essence, pilotless efforts with new designs, it's my understanding that the US-APWR-certified design will not have risk-informed tech specs. They are formulating their tech specs with an option for risk information, but the standard design won't have risk-informed tech specs. But they have a framework that allows it. But for the certified design, at least current understanding is they're only going to use the standard tech specs completion time.

So it is true. Comanche Peak, at least today, still is saying they are going to risk-inform their tech specs in the CLL. Just, you know, that is a bookkeeping issue.

MR. DUBE: Yes.

CHAIR STETKAR: It doesn't change at all the technical essence of the discussion or anything like that.

MR. ADER: We'll check. The last I was thinking is -- and we try to still have them in.

CHAIR STETKAR: The version I have seen says use the standard tech specs or a risk-informed value derived from this other document. They're not going to develop that other document. They're simply providing the shell of the tech specs to allow someone -- it's kind of like the setpoint-change methodology. Use this setpoint or a setpoint derived from a to-be-inserted-later setpoint development document.

So I don't believe that MHI is going to be presenting any actual AOTs that are different --

MR. ADER: No. They wouldn't.

CHAIR STETKAR: -- from the standard design.

MR. ADER: And that's where the implementation issues that we have been dealing with, and even coming out of the tabletops, is for a plant that is not yet built and operating without operating experience, you can't get the final quantification. But they want that option. So you have to put it in there that refers to a defined process. So they kind

of do, kind of don't. But they haven't gone all the way.

MR. DUBE: Okay. I made a note. On the maintenance rule, 50.65(a)(4), staff found no gaps in the assessment and management of risk. And we'll talk.

During the exercise in the tabletop, we actually exercised an online risk management tool, for example, or contracted it. When the PRA approach is combined with other inputs such as the degree of defense-in-depth and plant transient assessment, factors other than PRA were often found more limiting in terms of the risk management action level. So they look at not just the PRA risk values but also what's the impact on defense-in-depth? Is defense-in-depth still maintained? And what about the impact on plant transient situations? If you are working on main steam isolation valves, obviously there is an impact on reactor consistency and what have you. And often those were more limiting than the change in core damage frequency, if you will, based on the existing guidance.

And the other thing is -- and the Chairman here will be very happy with this paragraph -- NUMARC 93-01, section 11 explicitly acknowledges there is

acknowledged variability in baseline core damage frequency and large early release frequency, blah blah blah, determination of the appropriate quantitative risk management action thresholds are plant-unique activities. This means, for example, that there isn't a one size fits all level for action that -- I hate the word to use "relative" because I can see from the audience -- is that --

CHAIR STETKAR: That's okay. He gets a chance to come up after you guys are through.

MEMBER SHACK: But there's not a whole lot of guidance to tell you how to pick those.

MR. DUBE: Right, correct.

MEMBER SHACK: And the suggestions are obviously geared towards current.

MR. DUBE: Yes. We do know that some changes may be necessary to the guidance because you get some new SSC, structures, systems, and components, in the new reactor designs. And the staff right now is reviewing the guidance documents to see if some changes might be necessary because of scope issues, but I will just note that these aren't related to what we have been talking about in terms of risk metrics or thresholds or increased enhanced level of safety. They're kind of more administrative in nature, I would

say.

On surveillance frequency control programs, staff found no gaps in here. And we have listed some of the summary documents, seven, eight reasons. They are getting core here, some of the basis.

Surveillance frequencies that are controlled by other programs are excluded. And so what is covered, equipment covered by in-service testing, for example, major pumps and valves, tend to have some of the highest risk importances, but these are excluded from this particular initiative. In fact, what is left usually are things like, you know, control wide movement or MSPI testing or some radiation detection equipment or something along those lines.

So what remains to be implemented under RITS 5b generally are lower risk-important components. So what happens to it usually is not going to impact the baseline risk very much.

But more importantly is that fourth bullet. And that is unlike tech spec initiative 4b, 5b is much more deterministically oriented, with risk impact only a secondary condition in the criteria for changing surveillance test interval. And I think we

talked about this during the September subcommittee meeting, but basically it's highly dependent on performance monitoring, that any particular change in surveillance frequency usually goes in a phased approach if it's a monthly testing. They may go to maybe two months or three months. And it's a gradual change. At every step they hold, observe, that they identify changes in performance. The IDP, integrated decision-making panel, may decide, you know, let's look at the trends, look at the costs, and may decide to actually go back if an adverse trend is resulting.

Some examples, some equipment you want to exercise off frequently because things stick. Seats stick in valves. That's solenoid-operated valves and what have you. And the testing almost has a preventative maintenance function as well as a testing function.

So staff found no gaps if applied to new reactors.

We talked about 50.69. I won't dwell on that. Staff believes there are no gaps.

On 174, we've talked about this all through this morning. So I won't emphasize this. Again the staff found no gaps.

We have talked this morning about the

change process. It's called 50.59-like. We found a gap in that. You know, this definition of ex-vessel severe accident and the applicability of the tier 2 change process to just ex-vessel severe accidents are not other features that may be there to prevent or mitigate accidents, severe accidents, that are still retained within the reactor vessel seem to be excluded from the process. And the criteria that might be used are more applicable to design basis accident analysis evaluation assessment, not severe accidents.

So we do have a recommendation to address this potential gap by ensuring that there are sufficient details on all key severe accident features in tier 1 and including a change process in future design certification rulemaking in section VIII for non-ex-vessel severe accident features similar to section VIII.B.5.c for ex-vessel severe accident features.

You know, because we did review those designs that have been certified, we're not proposing any kind of backfit because, actually, something like this is actually a rule. I mean, it's right in the Code of Federal Regulations, part 52. I mean, it actually would require a backfit analysis. We're not proposing to go to that extent because we feel

confident for the current reactors that have been certified that we don't have this gap.

CHAIR STETKAR: Don, the way this one is presented, here it says A and B.

MR. DUBE: Yes.

CHAIR STETKAR: Is A sufficient? I mean, as I understand it, you made your determination --

MR. DUBE: Very good. That's a very good point because we had an "or" here before.

CHAIR STETKAR: Ah. Okay.

MR. DUBE: And the staff discussed it. And management change discussed it. And I know industry will recommend an "or," at least an emphasis on A.

The thing is if you speak to those who have been involved with this 52 process since day one, I mean, they will admit that the way they define ex-vessel severe accident in the statement of considerations, they should have used different language. That was all-encompassing. That should have encompassed non-ex-vessel, should have encompassed those five -- remember, those five containment challenges --

CHAIR STETKAR: Yes.

MR. DUBE: -- should have referred back to

whatever that -- those regulations that describe those five. They should have actually had that.

So, in answer to your question, showing that it is in tier 1 is a good stopgap measure, it will address staff's concern, but it is a work-around. And, really, the thing is in our opinion, you know, the rules should be changed. Going forward, the rule should be made right.

CHAIR STETKAR: Don?

MR. DUBE: There was a lot of -- I mean, excuse me -- a lot of discussion with OGC General Counsel on this particular -- in fact, I mean, 90 percent of our comments on the draft Commission paper has been on this very issue. So it's generating a lot of interest.

MEMBER SKILLMAN: From your slide 41, I thought that the real thrust of this was, if you will, the sneak path of the work-around on tier 2 for the ex-vessel.

MR. DUBE: Right. That's why by ensuring it's in tier 1, tier 1 requires staff to require approval. It makes sure that significance --

MEMBER SKILLMAN: What you are doing is lifting them out of tier 2 into tier 1.

MR. DUBE: Yes, yes. Any significant

change is going to change tier 1. Right now you will find in tier 1 statements like -- I showed you that example on the reactor vessel injection using firewater for APWR. That's pretty explicit, I mean.

MEMBER SKILLMAN: Yes.

MR. DUBE: It says there are valves there. There are two valves, manual operation, so on and so forth. You are not allowed the flexibility to change that. But there are some tier 1 statements, like "This feature is used for this," you know, literally ten words or less. And all the rest of the details are in tier 2. And so change tier 2 under a licensee control process. There's a lot of room for a lot of license, if you will --

MEMBER SKILLMAN: It seems to me the words are still vague. It would seem to me that --

MR. DUBE: What words are vague?

MEMBER SKILLMAN: Under A, under alpha, I would think you would want to say ensuring that all details regarding key severe accident features are in tier 1.

MR. DUBE: Just that all key severe accident features are in tier 1, including details.

MEMBER SKILLMAN: Yes. But what is absent in A at the current time, if you will,

comprehensiveness so that details cannot be varied in tier 2.

MEMBER SCHULTZ: That's not how I read it, but if that's what you intended, that's not the words.

MR. FRAHM: "Ensuring that all key accident features are included in tier 1" or something like that and take out "the sufficient details."

MEMBER SKILLMAN: That would certainly do it for me. It seems that --

MR. FRAHM: Do you see the difference?

MEMBER SKILLMAN: The details are obscure.

MR. DUBE: No, but I only got a 500 on SAT Verbal. So I --

(Laughter.)

MEMBER SKILLMAN: That's pretty good. That's pretty good.

MR. ADER: One of the concerns when you're going back and forth with OGC was are we being inconsistent in the level of detail, what we have done on the ones we have certified?

MEMBER SKILLMAN: Right.

MR. ADER: We were looking at putting it in tier 1. The subset in tier 1 has been through the three we have certified. So we're still saying keep it in tier 1 with the high-level important stuff

because that would require a real change to change the certification but not that it had a balance. So we don't have to put excruciating details into tier 1 and give it the tier 2 flexibility to change if there was a gap, there was no change process to evaluate it under. So that's why we --

CHAIR STETKAR: And this keeps you covered in -- you can still have a fairly high-level description in tier 1 consistent with a lot of the other material in tier 1.

MEMBER SKILLMAN: But, still, the real change from the present wording that you have presented in A needs to be the comprehensiveness of A. A has got to cover them all.

CHAIR STETKAR: A does say, though, "all key severe accident features." It doesn't say, "all severe ex-vessel severe accident features." So the "all key severe accident features," my interpretation is that comprehensiveness.

The level of detail that's in tier 1 for these particular things ought not to be any more detailed than is for any other safety system or function.

MEMBER SKILLMAN: Well, let me push a little bit further. If A is written, what is in tier

2?

CHAIR STETKAR: The details.

MEMBER ABDEL-KHALIK: The rest of the details.

MEMBER SKILLMAN: I understand that because I did this for the US-APWR. Where I'm going is on your page 41, you showed a reasonable work-around. And I think what you are trying to do with this particular set of statements is to preclude this from occurring.

MEMBER SHACK: No. He is happy that it's there.

MEMBER SKILLMAN: Pardon?

MEMBER SHACK: He is happy that it's there. The statement that's in tier 1 now is what saves him in the ABWR. And you wouldn't want to --

MEMBER SKILLMAN: Right.

CHAIR STETKAR: Putting it in tier 1 hopefully kicks you out of the first diamond to the right on page 41. Even if it doesn't, still if you make a change to some subtle design detail that you didn't quite capture in tier 1, the tier 2 requirement would kick you out to the right.

MR. ADER: I can't think of anything -- tier 2 pretends to provide more detail on the feature

of what's in tier 1.

MEMBER SKILLMAN: I understand.

CHAIR STETKAR: I'm sorry I asked, but I do understand now sort of the basis for "and."

MR. DUBE: Okay. You still don't like the words?

CHAIR STETKAR: No, no, no, no. I understand the -- I only asked whether or not they were essentially redundant when simply the first would be necessary and sufficient. But I can see given the possibility of fairly limited design information in tier 1, it may not be necessary and sufficient.

MR. ADER: What you may have in tier 1 is provides sufficient flow to provide cooling. In tier 2, it may say, "Our pump is 450 gpm." You make a design change that takes it to 400. I don't want to have to do a rule change if 400 is also sufficient.

CHAIR STETKAR: But you do want to have them do the evaluation.

MR. ADER: I do want to have them do the evaluation if that makes sense.

MEMBER SKILLMAN: It does. Thank you.

MR. DUBE: Good. And on the large release frequency, the LRF operations, have talked about this earlier. So I won't dwell on it, the options.

I acknowledge Steve Schultz's comment on this particular thing, but we are doing something additional. I may not ever use this slide again, but --

(Laughter.)

MR. DUBE: -- if I do, I made a note.

CHAIR STETKAR: You will be before the full Committee.

MR. DUBE: Yes. I don't know how much --

CHAIR STETKAR: Steve made notes.

MR. DUBE: And the recommendation is option 2C, to harmonize risk-informed applications for the new reactors consistent with the risk metrics used by the current operating fleet. So 2C is this with the provision for leaving performance objectives.

We did not do our risk-informed initiative, had neither time nor resources. We did focus on every single process that was identified in the Commission SRM. So that we did do. And then we did beyond that.

We didn't do in-service testing of pumps and valves. We did not do integrated leak rate testing interval, didn't do whatever 50.46a may have evolved. We didn't do equivalent of NFPA 805, which is risk-informed fire protection, did not do 806.

There was enough to do.

Some of the reason was there appeared to be little short-term interest by COL applicants. When I say "short-term," I mean something that we have to worry about here in the next couple of years.

Some applicants were actually COL holder now. So maybe down the road they want to go all risk-informed on everything, but we'll leave that to the next generation of reactor analysts, the risk analysts that do these tabletops. We didn't do it.

I do note that alternative source term, reg guide 1.183, is implemented at all the new designs with COLAs except ABWR. So that activity has moved forward.

And then I'll turn it over now to Ron Frahm.

MR. FRAHM: Okay. As we discussed just earlier today, based on the ROP tabletops, we did come up with three options for the Commission to consider. Under all of these options, we tried to meet these three objectives and then some that are pointed out in the paper.

And we believe that these are consistent with the SRM direction, and that is to maintain the current risk thresholds for the new reactor designs to

be consistent with reg guide 1.174 and our integrated risk-informed decision-making and to not infringe upon the greater operational flexibility that's afforded the new reactors based on their more robust designs.

The first option is basically use as is, don't make any changes under this option. An obvious advantage is that there are no additional resources needed to implement this option, but at the same time, it falls a little short. And we might not always provide for an appropriate regulatory response.

MEMBER ABDEL-KHALIK: I would have used the words "may never," instead of "may not always," in reference to that.

MR. FRAHM: Well, not always. Well, it's pretty close to it. But we don't want to downplay the feasibility of all the options. We had a flavor. We didn't want to leave --

CHAIR STETKAR: They found some things where the existing framework indeed triggered things that they felt were appropriate.

MR. FRAHM: We want to provide actual options that are reasonably feasible. Some might be better than others.

MEMBER ABDEL-KHALIK: Be careful what you ask for.

MR. FRAHM: Okay. Moving on to option B or second option is to augment the existing processes. You'll notice that --

CHAIR STETKAR: Don, hit the button.

MR. FRAHM: You'll notice that the implementation here is very similar to the conclusions we have from the tabletops that were presented earlier.

Under this option, we would use the existing risk-informed SDP but augment it with deterministic backstops, as we have talked about. Under MD 8.3, we would modify the existing contribution of deterministic criteria or potentially develop new backstops consistent with the SDP. And under MSPI, we would develop an alternative indicator or additional inspection in the mitigating systems cornerstone to compensate for not getting a whole lot of insight out of MSPI. And they we would also have to increase the inspection of passive mitigating systems for the passive designs.

An obvious advantage to this option is that it aligns very closely to the conclusions from the tabletops. Another advantage is that these proposed enhancements could be developed using existing resources that are already budgeted and could

take place over the next several months or even years prior to new reactors going into operation and work with external stakeholders and internal stakeholders, for that matter, to develop the specifics of the deterministic backstops.

The third option, option C, is to develop deterministic tools and essentially not use the existing risk-informed SDP tools but, instead, develop new tools that are specifically designed for new reactors. They would capture risk insights but using deterministic guidance consistent with the licensing basis and design certification.

A disadvantage is that these tools would be less risk-informed than for the current fleet, obviously. And then another disadvantage is that we probably would require additional resources above and beyond what is currently budgeted to research and develop the new guidance documents.

CHAIR STETKAR: And, as usual, it seems to fly in the face of Commission policy on risk-informing the regulatory process, but there's that.

MR. FRAHM: Right. But it does meet the SRM of maintaining the current risk thresholds because they are still there. Pretty hard to come up with three options.

CHAIR STETKAR: Oh, okay.

MR. FRAHM: In case you hadn't guessed, we're going to recommend option B, to augment the existing processes. And prior to doing so, obviously we would obtain Commission approval. And, in fact, at least one year prior to implementation, assuming the Commission directs us to go this way, or one year prior to implementing it, we would go back to the Commission and say, "Here is what we plan to do" and then get their blessing going forward.

And, as part of the ROP, we have built in a continuous improvement process. So these enhancements could be refined over the years as we learn lessons and further tweak to get where we need to be.

CHAIR STETKAR: I was going to ask you, Ron, rather than running some formal pilot process over a period of time, you would rely on that normal improvement process to sort of work out the details of the rough edges.

MR. FRAHM: That was our thinking. The pilot idea I wouldn't discard.

CHAIR STETKAR: Okay.

MR. FRAHM: Yes.

MS. FRANOVICH: That could be a management

decision that --

CHAIR STETKAR: Yes. No. I was just curious in terms because I've got the words about --

MS. FRANOVICH: We could do it that way.

MR. FRAHM: We often run pilots for implementing new things. So that would probably make quite a bit of sense, and I could see that. That's a good point.

And, with that, that's the end of the ROP recommendations and options.

MR. DUBE: Yes. Well, we ain't got a lot of next steps left. A year ago this was a dozen, but this is the full ACRS 12, finalize the Commission paper. And the paper is due to the Commission early June but actually due to the EDO, the Executive Director for Operations, I believe it is, May 24th. So it's tight. It's very tight.

That's it.

CHAIR STETKAR: Are there -- any members have any comments or questions for the staff?

(No response.)

CHAIR STETKAR: If not, thank you. Before we close, I want to make sure NEI has their time --

MR. DUBE: Right.

CHAIR STETKAR: -- and we get input from

the public if there is any. There's a lot of material to cover. And we may need to have a little bit of close-out discussion about what is appropriate to bring under the full Committee meeting --

MR. DUBE: Okay.

CHAIR STETKAR: -- because we have got a very limited time slot there.

MR. DUBE: Right, right.

CHAIR STETKAR: But I want to make sure that we have an opportunity to hear from NEI and anyone else who might want to provide anything. So if we can do that, Biff, do you want to come up?

MR. BRADLEY: I saw somebody do some kind of ALT-CTRL-something.

CHAIR STETKAR: Ron is adept at this.

MR. BRADLEY: Are you ready to go?

CHAIR STETKAR: We're ready.

MR. BRADLEY: Okay.

10. STAKEHOLDER PERSPECTIVES ON RISK-INFORMED REGULATORY FRAMEWORK FOR NEW REACTORS

MR. BRADLEY: Thanks for the opportunity to present to the Subcommittee today. I just have a few slides, really. Hopefully it will be far briefer than the 81 slides we just went through.

Just sort of my apolitical announcement

here is, you know, we really believe that plants should actually be encouraged to pursue these applications. They have PRA requirements under 50.71 to meet endorsed standards and to periodic updating. It's all built directly into the regulations for these plants.

So it would be a shame for them not to benefit from those models and use them for these applications. It makes the models better, and it makes the whole objective of having these models much more attractive to the plants.

So it probably goes without saying, but sometimes I get the impression from talking to the COL community that it is sort of like a heavy lift to try to implement these. And I really don't think it should be.

As I mentioned, there are a lot of things available, both voluntary. One point I want to make is that there are certain mandatory uses of risks for all plants. And those include the maintenance rule, the reactor oversight process, I guess the DRAP and other things for the new plants.

And I know the staff has suggested that there needs to be some period of where we accumulate data before we can use applications. I would just

note that it seems like that would apply to mandatory as well as voluntary applications. And somehow we have got to come to grips with that. You know, I would like to think we don't have to operate these plants for years and years before we can take advantage of some of these things.

MR. ADER: It's a good point, Biff. Is NEI's or the industry's position that you ought to be able to use the voluntary initiatives essentially from day one, fuel load?

MR. BRADLEY: That's a good question. I don't think we have established a formal position on that and really gone back, but I think our thinking would be that that would be desirable. But we would have to see if we could justify that. But I really would like to think that we could -- certainly if there are certain things you are doing from day one with the ROP and the maintenance rule, it seems like whatever rationale you have for that ought to apply to the other applications as well.

MR. ADER: Yes. And I personally saw some of those concerns. And the issue is that, by and large, I don't want to say every SSC, every component in a new reactor design is very similar to an existing reactor. You can think really big squib valves and

things like that. But the normal complement of pumps and pipes and valves and diesel generators and circuit breakers and wires and all that sort of stuff --

MR. BRADLEY: Materials.

MR. ADER: -- materials pretty much look the same. And it's not clear to me how operating a fixed number of new plants for a few years will give you very much useful operating experience unless, for some reason, the equipment is really, really bad. If it's really, really good, as is most equipment, accumulating a few more years of no failures doesn't do anything for you. If it's really, really bad, you are going to see it.

MR. BRADLEY: Right.

MR. ADER: So it's not clear to me what that operating -- with the exception of some perhaps very different pieces of equipment, for which there isn't a comparable --

MR. BRADLEY: I understand. I know there has been a lot of discussion on the squib valves. I guess we need to take some initiative, both we and the staff, to come to grips with this as sort of an aside after a period of what we're talking about.

MR. ADER: It's interesting that you brought it up because it was one thing that I kind of

stumbled over a bit also.

PARTICIPANT: That's one of the issues that we're trying to work through on the tech specs is how they account for lack of their own operating experience. The staff work is working that issue. How it comes out is --

CHAIR STETKAR: It's an important issue because, you know, the question is how long do you have to operate until enough is enough? If it's 35 years, it's not very much of an incentive. If enough is enough is three years, it probably doesn't make any difference compared to zero.

MR. BRADLEY: And I would note some of the initiatives; for instance, 5.b, are dependent on accumulating some operating experience. Don talked about how that was primarily a deterministic initiative, and he's correct. Some of these are somewhat application-specific, but I think in general we should be able to use these relatively early.

Getting back to the tabletops, I just wanted to note that I thought this was a very effective and well-conducted exercise. I thought both industry and NRC staff had good participation and that we really gave these a hard scrub.

We stayed consistent with the Commission

directive. We had times where there was pressure to go back and rethink the Commission decision. And we all did a good job of keeping it constrained to the direction the Commission gave us. We saw a little of that even here today I think.

MEMBER SHACK: We didn't get the message.

(Laughter.)

MR. BRADLEY: Not that I would have expected otherwise from this group.

And, as Don spoke, you know, I thought the examples, while they were relatively stretched cases, we also realized they had to be somewhat realistically constrained. As, as he mentioned, a lot of these same things you could set up under the current tech specs or the current regulations. And you could create similar extreme kinds of situations, where you just know they don't happen. So I thought we did a good job of that.

And my take-away here is we are in general agreement with the conclusions that were presented by the staff this morning. We really don't have any major, substantive disagreements with what's in the draft SECY, and I would note, you know, in the licensing applications, there were very minor gaps or essentially no real gaps identified. I think there

were some minor things, maybe the maintenance rule. We're obviously satisfied with that conclusion and in agreement with it.

Having written a lot of these guidance documents over the years, we went to great efforts to build in backstops and deterministic considerations. They weren't risk-based. And I guess I'm not surprised with this result. That just confirms that when we wrote these, we had the right considerations in place.

On the ROP, again, I think we're in essential agreement with the conclusions that were presented that the certain elements of MSPI are not going to be applicable to the new designs. And there will need to be some thought given to some replacement for that.

And, again, the idea that you need some quantitative considerations to account for things such as diversity, redundancy, what have you, that the fundamental safety philosophy of the plant, we're supportive of that. I think there have been some qualitative considerations that are in the current ROP. And as long as there are good, objective, clear, qualitative, there's nothing wrong with something being qualitative versus quantitative. We don't want

subjective, fuzzy stuff in there, but good qualitative considerations can sometimes be very effective. And I think this may be a case where qualitative considerations relative to those design philosophies or features are a good way to revise the ROP for these plants.

Again, this is a devil in the details kind of question. And, as the staff alluded, there will be a significant amount of interaction to come to grips with exactly what those would be.

MEMBER RAY: Well, Biff, let me try and get out the details here and just say it's asserted -- I've had this assertion made to me in the course of our conducting reviews here that "Oh, that is investment protection" -- you've heard that term used, I'm sure -- "investment protection. And, therefore, it's our business what we do with its operability and procurement and design" and so on, which is fair enough, I think.

I have gotten into some controversies, though, when credit is taken for investment protection when it comes to calculating risk. And I think that is where this dilemma arises.

If one did not take credit for investment protection and took the position that "It's my secret

what emergency diesel generators I provide. It's none of your business" and I don't take any credit for it, if one were to do that, would that change what you just said about these features and how we all should learn to get along, I guess, together and taking appropriate attention of them and so on?

MR. BRADLEY: That is a question I guess that sounds rather profound. I don't know if I want to answer it quickly.

MEMBER RAY: It's very real. In this very room here, we've had --

MR. BRADLEY: I think it would. I mean, I guess my sense is that, well, one, we don't do that. I mean, these things are treated. Even though they are just for investment protection, there is regulatory treatment of these items.

MEMBER RAY: Well, that is what we are talking about, an aspect of the regulatory treatment, aren't we?

MR. BRADLEY: Whether we do -- well, your question, I'll have to give it some more thought.

MEMBER RAY: Yes. I just want you to know that there are times when your associates would like us to view them as merely investment protection.

MR. BRADLEY: I guess that's not my

personal view, --

MEMBER RAY: Okay.

MR. BRADLEY: -- you know, that all of this is investment protection, but I will take that back and see if we can get a good answer.

MEMBER RAY: Well, this is not easy. Okay? In other words, I don't mean to say we shouldn't recognize the risk-benefits of these things. But, you know, there come times when you seek to impose what seem to be rational requirements on things and they're off the table, prohibited, "Don't talk about that because it's in this category of investment protection."

So, anyway, I'll leave it there. You don't need to answer it. I just think it's a part of the dilemma that we struggle with here.

MR. BRADLEY: Right. I mean, it even I think comes up a little bit back to this relative risk discussion we have been having. And if you were not going to credit those additional equipments for whatever risk reduction that they provide --

MEMBER RAY: Say I don't.

MR. BRADLEY: -- then maybe you should be under -- you know, maybe your ROP should take that into consideration. If you desire to do that, that's

sort of making a case in a way for the relative -- generally, as you might have figured out, I'm not a big fan of that.

And we went back. We did spend quite a bit of thumb back a year or two ago making the case on why we thought that wasn't a good way to go. And we had a paper that was I think discussed with this group as well as provided with the Commission that laid all of that out.

Primarily it just comes down to, especially in the ROP, I think the concern with punishing the good deed of making the plant safer. And so that --

MEMBER RAY: Believe me, if I had an AP1000, I definitely would not want to have some of the safety-related systems activated. I would much rather use my investment protection equipment. And so I have a motivation to put it there other than safety. It's to make it so that I have some chance of operating the plant in the future. So it's not just altruistically installed.

And I'm sympathetic to the idea that, wait a minute, I put this in here because I wanted to not activate those damn --

MR. BRADLEY: Squib valves?

MEMBER RAY: -- squib valves. And so now why are you punishing me because I haven't? So okay. Never mind. We've talked about it enough.

MR. BRADLEY: I think I understand where you are coming from. And maybe that's an angle we can think about.

With regard to the ex-vessel severe accident change process, again, I don't have a lot to add, as Don indicated, I guess I haven't had the benefit of lengthy discussions with OGC or anything, but it's still not totally clear to me why you need both the tier 1 and the tier 2 controls that are being proposed. I guess we believe if you put an adequate description in tier 1, the problem is solved.

It appears to us that, in reality, this has not been a gap. Thais is a gap on paper. And the designs that have been looked at, those features that were important, whether it's ex-vessel or in-vessel, found their way into tier 1. So it seems to me that just continuing that or making sure that that continues solves the problem.

Obviously, you know, the change process is getting pretty cumbersome for these plants. We more than double the volume of change control guidance going from operating plants to new plants already. So

we have a whole other set of stuff in tier 2, I guess. You know, in my mind, if we could avoid that, that would be beneficial. And probably we would prefer to make sure the tier 1 description is going to do the job.

So, all of that being said, you know, again, a great effort, good interaction. And we really don't have substantive policy disagreements or substantial disagreements with where this has come out. I will say I think if we are going to rethink this now and possibly go back to putting relative risk in as another option I would certainly not make that same statement. I would have to go back and look at what is getting added and what other options are being put in. And we would have to rethink these conclusions that you see here.

But, as the paper is currently written, you know, we're substantively okay with it.

CHAIR STETKAR: Any members have any questions for NEI? If not --

MEMBER SCHULTZ: I have one question. Biff, in the last bullet, what type of interaction? You had this opportunity for the industry-NRC workshops that occurred in the last year and a half. So in terms of the upcoming dialogue and

participation, what would the industry be looking for in terms of interaction with the staff?

MR. BRADLEY: This is in ROP space you are referring to?

MEMBER SCHULTZ: Yes.

MR. BRADLEY: I would assume it would be through public meetings and through the normal processes we have set up right now. I think we have monthly ROP interactions on operating plants. But just through the normal public process to build the guidance, just that's what I had in mind, I mean, nothing extraordinary.

MEMBER SCHULTZ: Thanks.

CHAIR STETKAR: Anything else?

(No response.)

CHAIR STETKAR: Biff, thank you very much. I appreciate it. And I think we understand each other.

Right now is there any member of the public here who would like to make any comments or ask any questions?

We had a bridge line, phone bridge line, open, trying to get that opened up for two-way communications. If there indeed is someone out there listening, could you do us a favor and just make some

sign so that we know that the bridge line is open? We have no way of knowing that it's actually open in this direction. If somebody is out there, just anybody say something, just anything, so we know it's open.

(No response.)

CHAIR STETKAR: Hearing silence, I'm going to interpret that as there's no one there. So, with that, we've satisfied that.

Now, a couple of points of business. Let's go around the table as well and see if any of the members have any final comments or questions on anything. And I'll start with Jack.

11. DISCUSSION AMONG MEMBERS

MEMBER SIEBER: Okay. I generally agree with what the staff has done. And I think we are ready to proceed ahead with the overall plan.

CHAIR STETKAR: Steve?

MEMBER SCHULTZ: I concur. I think that I really appreciate the workshop approach that has been used here. I think that the staff has gained great value through that process and has very informative information to bring forward to the Commission as a result.

CHAIR STETKAR: Dick?

MEMBER SKILLMAN: I commend the staff for

having completed a significant amount of important work. Thank you.

CHAIR STETKAR: Dennis?

MEMBER BLEY: A very good open interaction today. I have nothing to add.

CHAIR STETKAR: Harold?

MEMBER RAY: I think I have expressed my -- I am more trying to learn what is a changing environment and what the implications of it are. I mentioned this.

CHAIR STETKAR: Said?

MEMBER ABDEL-KHALIK: I have no additional comments.

CHAIR STETKAR: Bill?

MEMBER SHACK: No.

CHAIR STETKAR: Joy?

MEMBER REMPE: No comments.

CHAIR STETKAR: And I have nothing to add. I've said all I need to.

So, staff, do you have a decent idea? I don't know how long we have allocated, probably an hour and a half, I'm guessing, at the full Committee meeting.

MR. FRAHM: That sounds right.

CHAIR STETKAR: You know, obviously, the

full Committee will be interested in the paper itself, the conclusions and the recommendations. And I would say certainly the tabletop exercise experience that drives some of the major conclusions, the ROP information, and the ex-vessel severe accident, I think you certainly want to highlight those.

MR. FRAHM: Okay.

CHAIR STETKAR: Some of the other staff as far as details of what was done in the tabletops, it's really interesting information, but I don't want to get hung up on that as a prelude to making sure that the full Committee understands the basic thrust of the Commission paper and the different options and the final conclusions and the basis for how those were derived.

MR. DUBE: Okay. Yes. I think if you look at this afternoon's session where I summarize the results and the recommendations, that's like 45 minutes or 50 minutes. And then I can augment that with general tabletop experience, a little more detail on ROP and ex-vessel. That will take us into an hour and a half.

CHAIR STETKAR: Well, is NEI going to have a -- have you talked to John at all? Do you want to say anything?

MR. BRADLEY: My sense is we will be at the meeting, but I don't think we need to present. But we will be here, and we will stand up if we need to.

CHAIR STETKAR: Okay. Thank you.

MEMBER SHACK: What do we hear from the peanut gallery? We're kind of arguing that maybe the ex-vessel -- you could spend your time better on the desktop exercises.

CHAIR STETKAR: That may be very well true. The ex-vessel in my mind is not directly relevant to the fundamental issue of risk metrics --

MEMBER SHACK: Risk metrics.

CHAIR STETKAR: -- and how they are used.

MEMBER SHACK: Cleaning up.

CHAIR STETKAR: Do the other Subcommittee members agree to that?

(Whereupon, there was a chorus of "Yes.")

MEMBER BLEY: I think focusing on the tabletops and where that led --

CHAIR STETKAR: Okay. In that case, since the full Committee hasn't heard anything about the first set of tabletops -- remember, you came last whenever.

MR. DUBE: Correct. September, yes.

CHAIR STETKAR: September. Let's decide to do that. Kind of downplay the ex-vessel. Even though it is kind of highlighted --

MR. BRADLEY: Yes.

CHAIR STETKAR: -- in the paper itself, I think you can work through that. I think that the message from the RITS 4b and the tabletops for that is important, only because of the extensive work that you did do.

MR. BRADLEY: Right.

CHAIR STETKAR: And the basis for the conclusions that the 30-day backstop and other considerations provide the limitations on just infinite allowed outage times for some -- so I think that would be useful in terms of that whole first set of things.

MR. BRADLEY: How about the LRF to LERF transition? Was that high on the list or low on the list?

CHAIR STETKAR: I personally think that's low on the list only because I can see the full Committee getting into an excruciating discussion about the philosophy. And that's not directly germane to kind of this issue of risk metrics.

MEMBER SIEBER: I agree with you.

CHAIR STETKAR: What do the other members think about that?

MEMBER BLEY: Yes. I agree. I don't see that as --

MEMBER SIEBER: What's hard to explain is the switchover.

MEMBER BLEY: And it could be a great time sink.

CHAIR STETKAR: I think it could be a great time sink. I agree.

MR. DUBE: Okay. So we'll emphasize the key tabletop exercises, certainly from the first set, especially RITS 4b, and some of the interesting ROP results.

MEMBER BLEY: And how they track the recommendations.

MEMBER SKILLMAN: I think we've got to mention LERF, but I think it can be offered in passing, as opposed to strong margin on it.

CHAIR STETKAR: They have to mention LERF, and they have to mention the --

MR. BRADLEY: CCDF.

MR. FRAHM: Ex-vessel.

CHAIR STETKAR: -- EVSA, only because they are elements of the paper going up. So there has to

be, you know, some de minimis discussion of that topic in the context but not to the level of detail --

MR. BRADLEY: Sure.

CHAIR STETKAR: -- that we heard today.

MEMBER SCHULTZ: Or the conclusions and --

CHAIR STETKAR: Or the conclusions and kind of -- yes.

MR. BRADLEY: Got it.

CHAIR STETKAR: And, you know, as you presented the LRF to LERF, you know, just noting the kind of consistency across the fleets.

Anything else? Any other member comments?

(No response.)

CHAIR STETKAR: If not, I would like to thank the staff.

MEMBER SIEBER: Thank you.

CHAIR STETKAR: You covered an awful lot of material today. I think we have --

MR. DUBE: That was a lot of material.

CHAIR STETKAR: I think we learned a lot. I think we really appreciate, you know, all you have been through through the last year with all of these tabletops and the industry, too. I think that this is an example of really good cooperation to address, you know, an important topic that does affect not just

licensing basis. But, more importantly, I think it affects how people proceed after the new plants are operating.

And I would really like to congratulate everybody on a job well done. And I think you did a very good job summarizing things today.

And, with that, we're adjourned.

(Whereupon, the foregoing matter was concluded at 4:07 p.m.)

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UNITED STATES NUCLEAR REGULATORY COMMISSION

Protecting People and the Environment

RISK-INFORMED REGULATORY FRAMEWORK FOR NEW REACTORS

Advisory Committee on Reactor Safeguards Subcommittee on Reliability and PRA

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March 7, 2012

Meeting Purpose

Discuss second series of tabletop exercise results and staff's response to the SRM on SECY-10-0121

Agenda

- Tabletop exercise results
  - 50.69 SSC classification
  - RG 1.174
  - LRF to LERF transition
  - Change process for ex-vessel severe accident features
  - Reactor oversight process
- Conclusions, options and recommendations in draft paper
- Next steps

Options Provided in SECY-10-0121

1) No changes to existing risk-informed guidance (status quo)
2) Implement enhancements to existing guidance to prevent significant decrease in enhanced safety (NRC staff recommendation)
3) Develop lower numeric thresholds for new reactors

Commission SRM Dated March 2, 2011

- **Commission approved a hybrid of Options 1 and 2**
  - Continue existing risk-informed framework pending a series of tabletop exercises that test existing guidance

- **Commission "reaffirms" existing**
  - safety goals
  - safety performance expectations
  - subsidiary risk goals and associated risk guidance
  - key principles (e.g., RG 1.174)
  - quantitative metrics

SRM (cont.)

- **Commission expects:**
  - Advanced technologies in new reactors will result in enhanced margins of safety
  - As a minimum, new reactors have the same degree of protection of the public and environment as current generation LWRs
- **New reactors with these enhanced margins and safety features should have greater operational flexibility than current reactors**

Tabletop Exercises

- December 2, 2010: 50.59-like change process for ex-vessel severe accident (EVSA) design features under Section VIII.B.5.c of each design certification rule
- May 4, 2011: Risk-informed inservice inspection of piping
- May 26, 2011 and June 1, 2011: Risk-Informed Technical Specifications (RITS) Initiative 4b on completion times and the Maintenance Rule (a)(4)
- June 29, 2011: RITS Initiative 5b (surveillance frequency control program)
- August 9, 2011: 50.69 and guidance in NEI 96-07 Appendix C on the change processes for Part 52 specific to EVSA design features
- October 5, 2011: RG 1.174; transition options from large release frequency (LRF) as a risk metric to large early release frequency (LERF); and ROP risk-informed case studies including SDP, reactive inspections under Management Directive 8.3, and MSPI
- October 26, 2011: Follow-up discussions with stakeholders on the ROP

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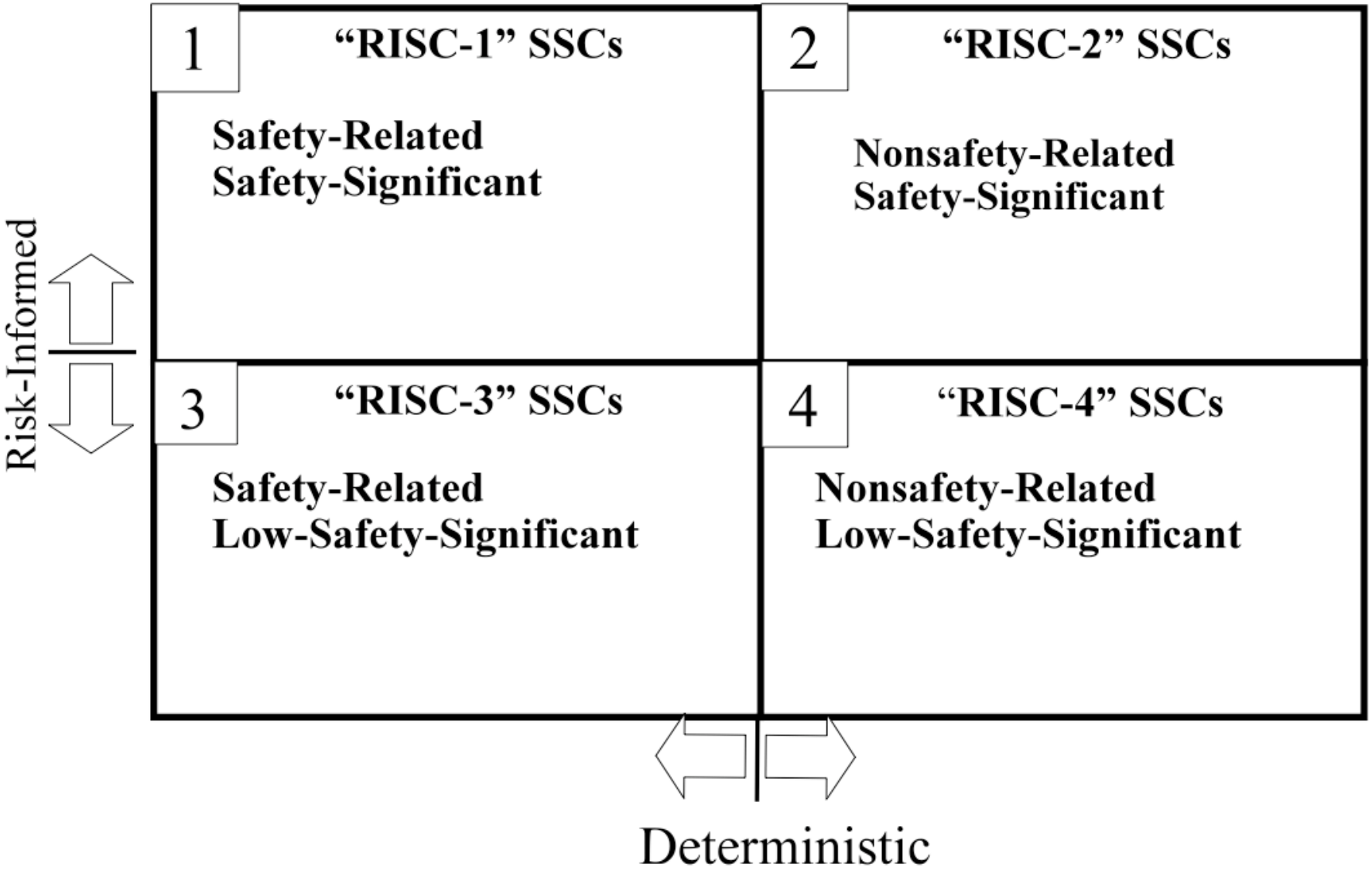
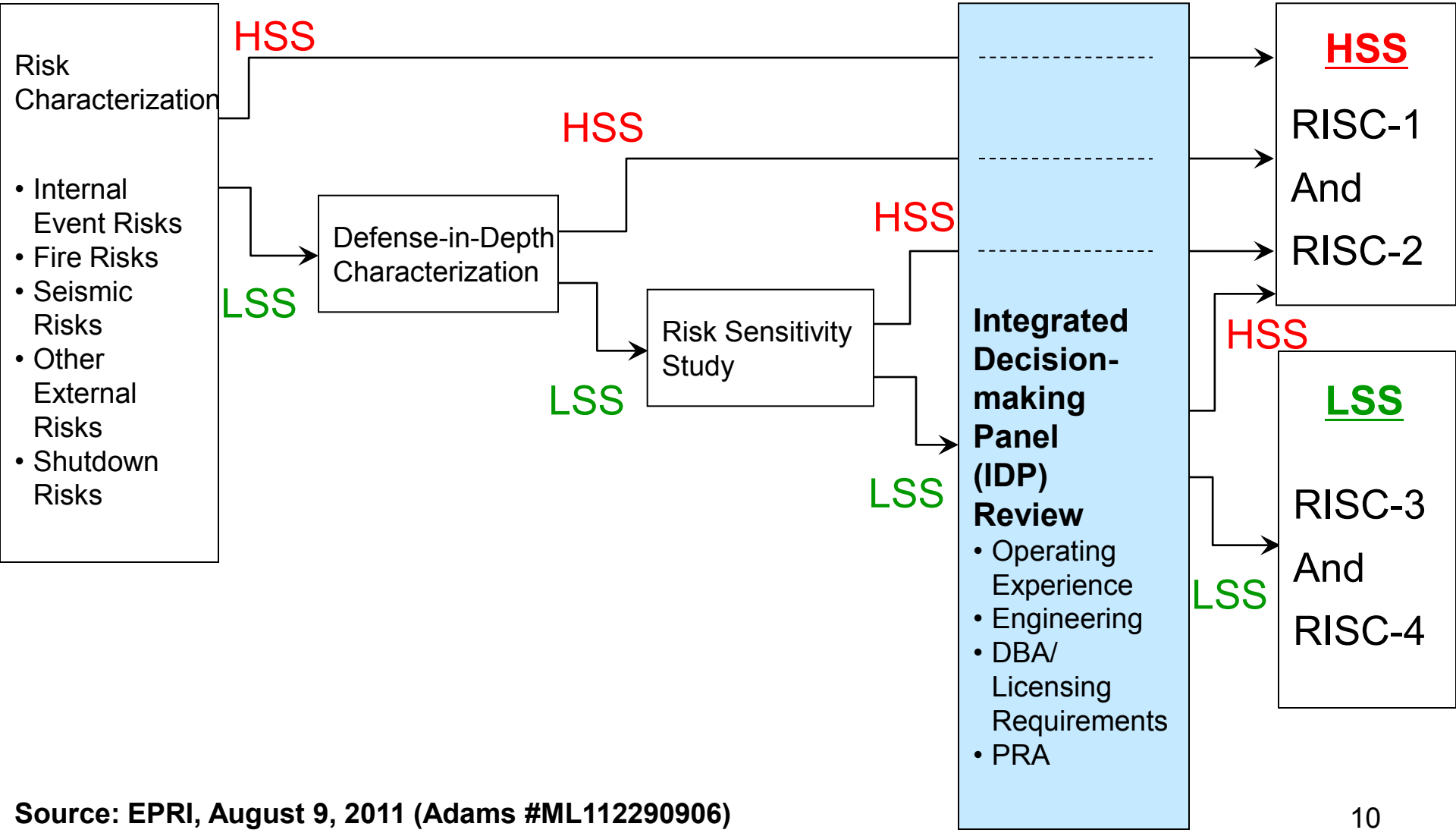
10CFR50.69 - Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors

10CFR50.69 – SSC Classification



NEI 00-04 Risk-Informed Categorization



Source: EPRI, August 9, 2011 (Adams #ML112290906)

Application of 10CFR50.69 to New Reactors

Process assumes:

- Initial safety related/non-safety related classification of all equipment
  - Beneficial during procurement if SR equipment goes through 50.69 process prior to purchase
- Full scope PRA
  - PRA standard leans toward use of plant specific data for failure rate
- All functions are known
  - Beyond design basis functions

New Reactor Results

- Active PWR
  - New build vs. operating plant
  - Maximized percentage of SSCs in RISC-3
    - Likely reduced in the future

New Reactor Results (cont.)

- ESBWR
  - Focus on ESBWR
  - Components moved from RISC-1 to RISC-3

10CFR 50.69 Tabletop Results

- **50.69: No gaps**
  - New reactor design (PWR with active safety systems) compared to South Texas 1 & 2 pilot
  - Built-in measures to monitor RISC-3 components

Regulatory Guide 1.174

RG 1.174 – Five Principles

- **Five principles for making risk-informed decisions**
  - **The proposed change:**
    - **Meets current regulations (unless exemption request)**
    - **Is consistent with the defense-in-depth philosophy**
    - **Maintains sufficient safety margins**
    - **Results in an increase in CDF or risk that is small and consistent with the intent of the Commission's Safety Goal Policy Statement**
    - **Will be monitored using performance measurement strategies.**

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From RG 1.174

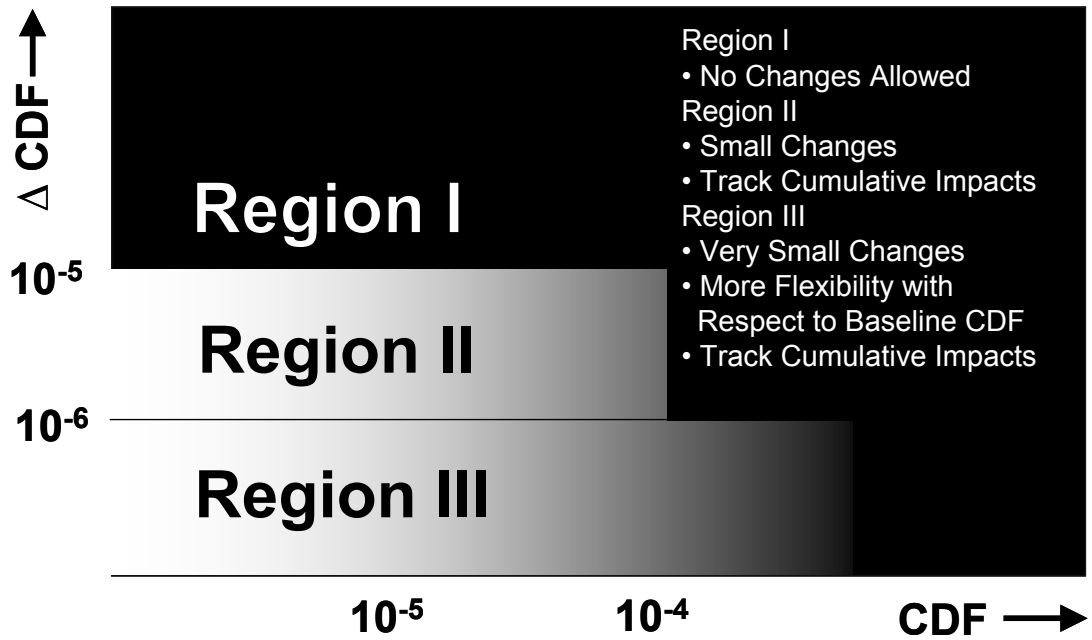


Figure 3. Acceptance Guidelines for Core Damage Frequency (CDF)

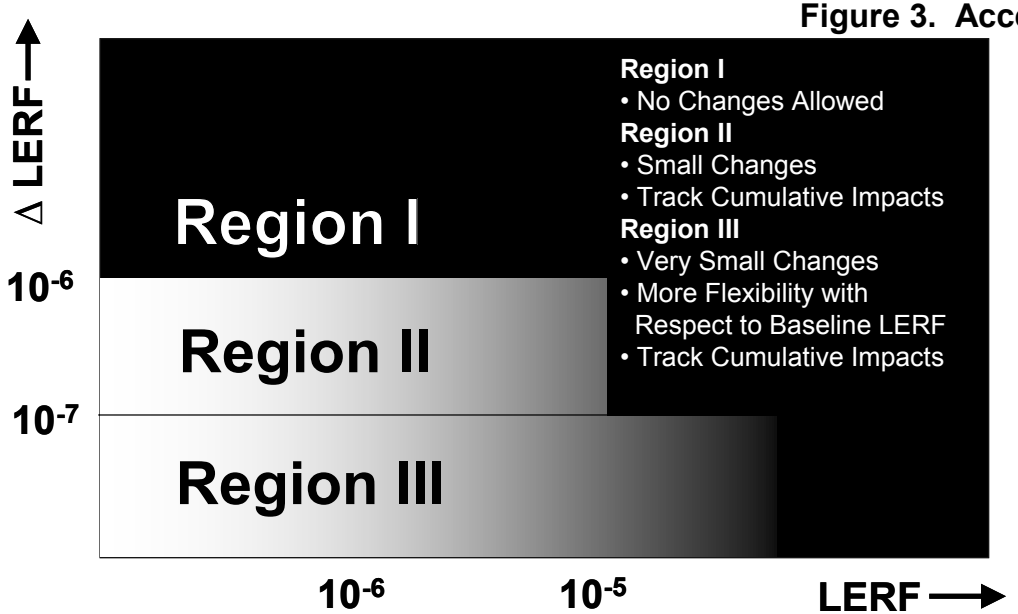


Figure 4. Acceptance Guidelines for Large Early Release Frequency (LERF)

RG 1.174 Cases

- ESBWR
- ABWR
- U.S. EPR
- US-APWR
- AP1000

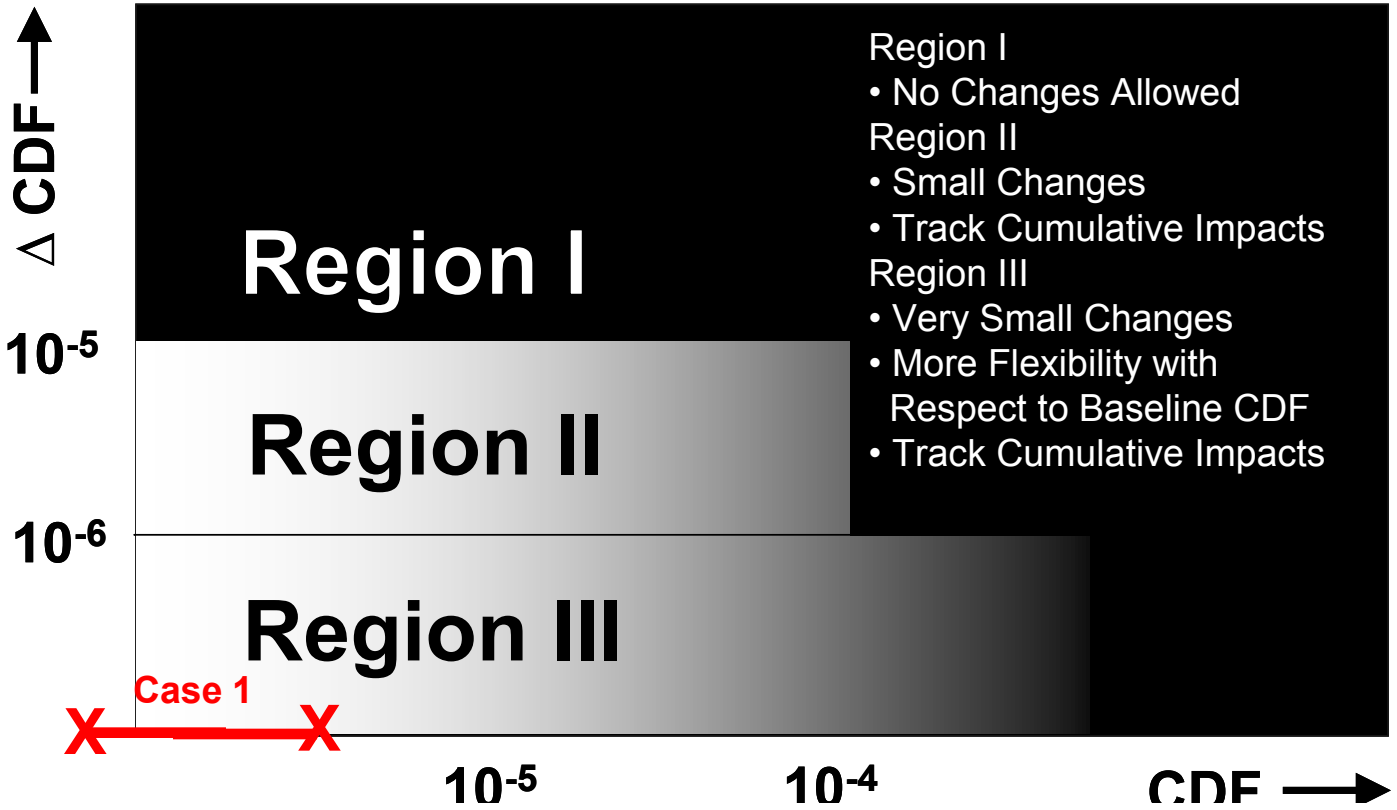
Hypothetical Case Study: ABWR

Case 1: Tier 1 change to COPS (wetwell venting) to reduce rupture disk setpoint from 104 psia to lower value, and change two isolation valves from normally open to normally closed with operator action required for venting

$$\Delta CDF \sim 10^{-7} /yr$$

Sources: ABWR SPAR Model, and DCD rev. 4 Section 2.14.6

ABWR Hypothetical Case



RG 1.174 Tabletop Results

- RG 1.174: No Gaps
  - ΔCDF observed to be very low
  - Close review by the staff:
    - Degradation of defense in depth
    - Changing from passive to active
    - Changes near boundary of region II

Transition from LRF to LERF

LRF-to-LERF Transition

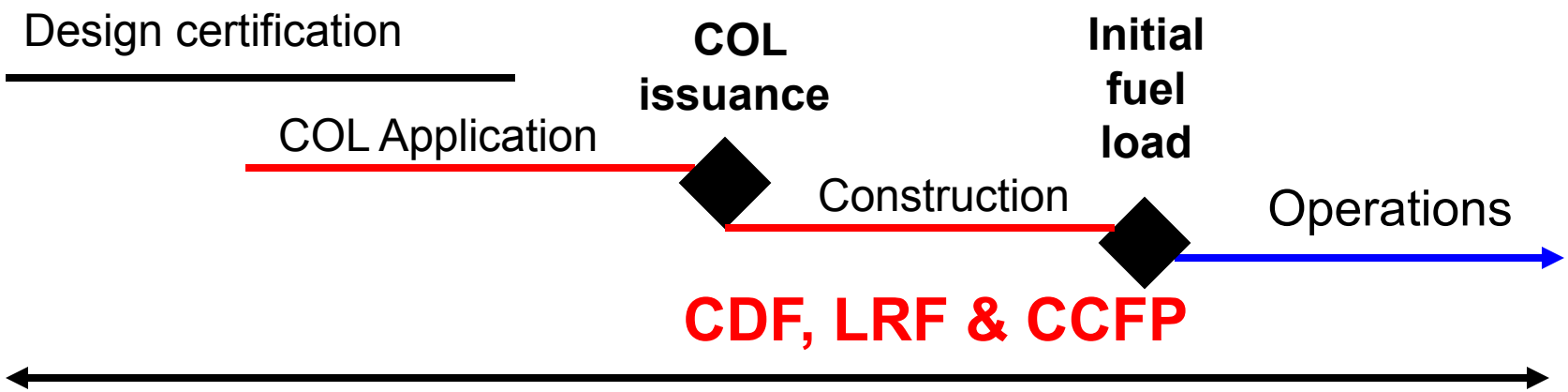
- **LRF vs. LERF**
  - Commission goals for new reactors are based on a conditional containment failure probability (CCFP) of less than 0.1, and a LRF of less than 10^{-6}/yr, as well as 10^{-4}/yr for core damage frequency (CDF)
  - Operating reactors use CDF and LERF as risk metrics

- **LRF issues**
  - LRF (and CCFP) have not been defined by the staff
  - Each design center has chosen different definitions
  - LERF is used in the ASME/ANS level 1 PRA standard, in risk-informed staff guidance (e.g., RG 1.174), and ROP
  - No existing or proposed level 2 PRA standard provides a universal definition of LRF

LRF-to-LERF Options

- Option 2A: continue use of LRF (& CCFP) indefinitely
- Option 2B: continue use of LRF (& CCFP) indefinitely and add LERF at initial fuel load
- Option 2C: transition from LRF to LERF at or prior to initial fuel load; discontinue regulatory use of LRF (& CCFP) thereafter

Option 2A

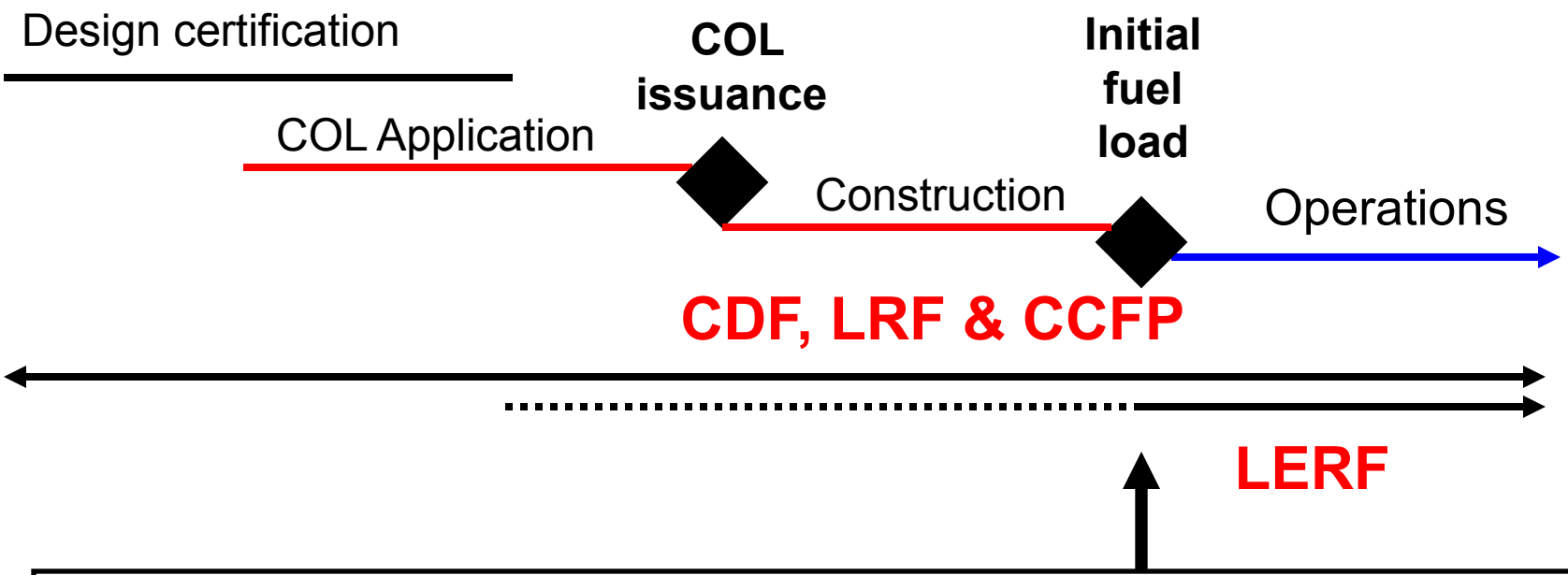

Design certification
COL Application
COL issuance
Construction
Initial fuel load
Operations
CDF, LRF & CCFP

Option 2A

- **Advantage**
  - **Maintains definition and use of risk metrics consistent with original license application and staff review per FSER**

- **Disadvantages**
  - **No existing definition & guidance on use of LRF**
  - **May be inconsistent with SRM direction in which Commission "reaffirms that the existing … quantitative metrics for implementing risk-informed decision making, are sufficient for new plants"**
  - **Issues with RG 1.174 and ROP which use LERF**

Option 2B



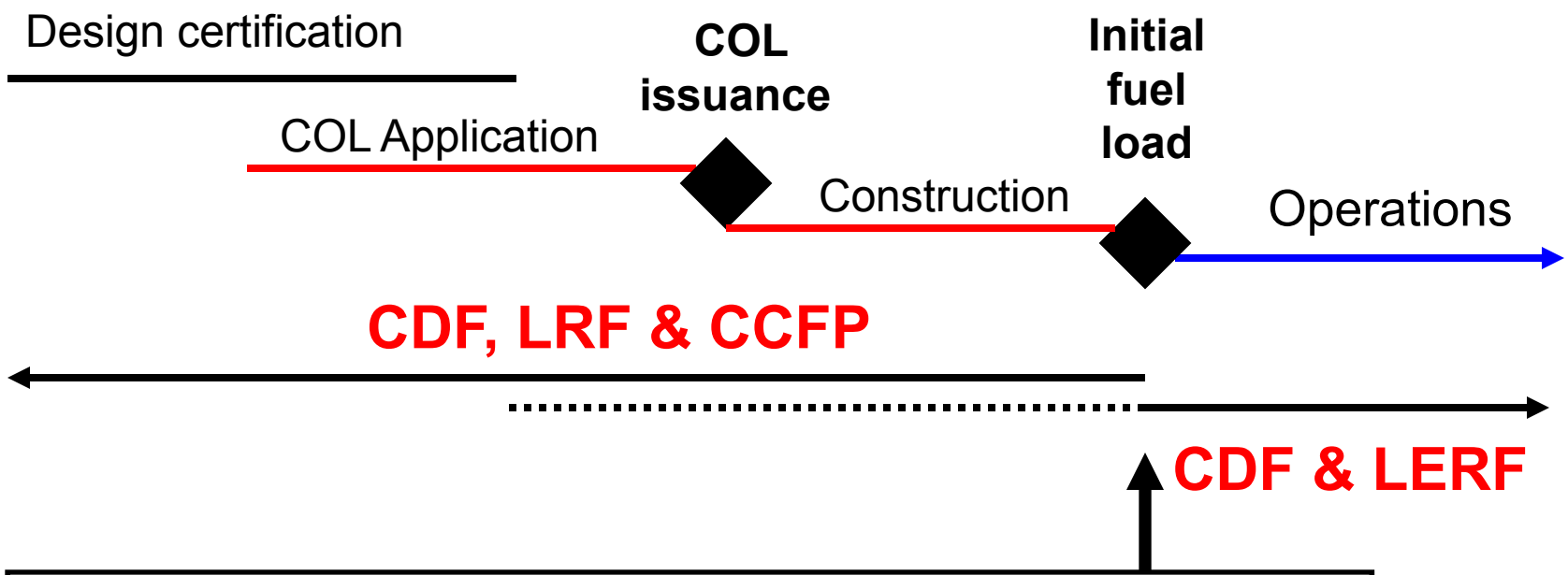
- **LERF calculated at or prior to initial fuel load.**
- **LERF and CDF used for RG 1.174 acceptance guidelines going forward.**

Option 2B

- **Advantages**
  - **Continued use of LRF & CCFP maintains definition and use of risk metrics consistent with original license application**
  - **Supports calculation of late containment failure impact per RG 1.174 rev. 2**
  - **Use of CDF & LERF for risk-informed changes to licensing basis consistent with RG 1.174 acceptance guidelines for currently operating reactors, as well as ROP**

- **Disadvantages**
  - **Added confusion by tracking both LRF & LERF**
  - **May be viewed as inconsistent with SRM direction**
  - **Added burden on licensees**

Option 2C



- **LERF calculated at or prior to initial fuel load. CDF & LERF used for RG 1.174 acceptance guidelines going forward.**
- **Last regulatory use of LRF & CCFP**

Option 2C

- **Advantages**
  - **Consistent with SRM direction**
  - **Harmonizes metrics for all operating reactors, both current and new, going forward**
- **Disadvantages**
  - **LRF & CCFP, part of original design objective in design certification, no longer tracked**
  - **LRF not available to assist in determining impact on late containment failure in RG 1.174**
    - **Augment discussion on long-term containment performance in Section 2.2 of RG 1.174 by referring to the containment performance objectives in SECY-90-016 and SECY-93-087**

Option 2C (cont.)

- **Containment performance objectives per SECY-90-016 and SECY-93-087:**

  The containment should maintain its role as a reliable, leak-tight barrier (for example, by ensuring that containment stresses do not exceed ASME Service Level C limits for metal containments, or Factored Load Category for concrete containments) for approximately 24 hours following the onset of core damage under the more likely severe accident challenges and, following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products.

Tier 2 changes to ex-vessel severe accident (EVSA) design features

Background

- NEI 96-07, Guidance on 50.59, new Appendix C regarding Part 52 change process, for example:
  - Departures from Tier 1, Tier 2, and Tier 2*
  - Effect on design basis accidents
  - Aircraft impact assessment and loss of large areas
  - Tier 2 changes to ex-vessel severe accident (EVSA) design features
- Public workshop on EVSA December 2, 2010
- Public meeting on EVSA draft guidance Aug 9, 2011
- Public meetings on NEI 96-07 Appendix C, in general, Nov 15, 2011, Jan 17, 2012, and Feb 14, 2012

From Part 52 App. A (ABWR)

- **VIII.B.5.c. A proposed departure from Tier 2 affecting resolution of an ex-vessel severe accident design feature identified in the plant-specific DCD, requires a license amendment if:**

  (1) There is a substantial increase in the probability of an ex-vessel severe accident such that a particular ex-vessel severe accident previously reviewed and determined to be not credible could become credible; or

  (2) There is a substantial increase in the consequences to the public of a particular ex-vessel severe accident previously reviewed.

What is an EVSA Design Feature?

Per the Statement of Considerations for the ABWR Final Rule, the change process for EVSA applies only to "severe accident design features, where the intended function of the design feature is relied upon to resolve postulated accidents when the reactor core has melted and exited the reactor vessel and the containment is being challenged"

EVSA Design Feature (cont.)

"In addition, the Commission is cognizant of certain design features that have intended functions to meet 'design basis' requirements and to resolve 'severe accidents.' These design features will be reviewed under either VIII.B.5.b or VIII.B.5.c depending upon the design function being changed."

10 CFR 52.47(a)(23) and 10 CFR 52.79(a)(38)

Design certification document to address challenges to containment integrity caused by:

- **core-concrete interaction**
- **steam explosion**
- **high pressure core melt ejection**
- **hydrogen combustion, and**
- **containment bypass***

*** Consensus of December 2010 workshop participants that design features that prevent or mitigate containment bypass events are <u>not</u> in and of themselves EVSA features, and as such Section VIII.B.5.c criteria do not apply**

Tabletop Results on EVSA Design Feature Change Process

- Certain severe accident features do not address "ex-vessel" conditions and VIII.B.5.c criteria do not apply (e.g., features to prevent ISLOCA / containment bypass)
  - Generally, enough details in Tier 1 that such features can not be removed and significant design changes are precluded
- Staff satisfied with "ex-vessel" portion of NEI 96-07 Appendix C Section 4.4.2.3; staff comments incorporated
- In a worst case, significant Tier 2 changes to non-ex-vessel severe accident features, up to and including permanent removal from service, could be made without prior NRC approval

Containment Challenges per § 52.47(a)(23) & § 52.79(a)(38)

Core-concrete interaction

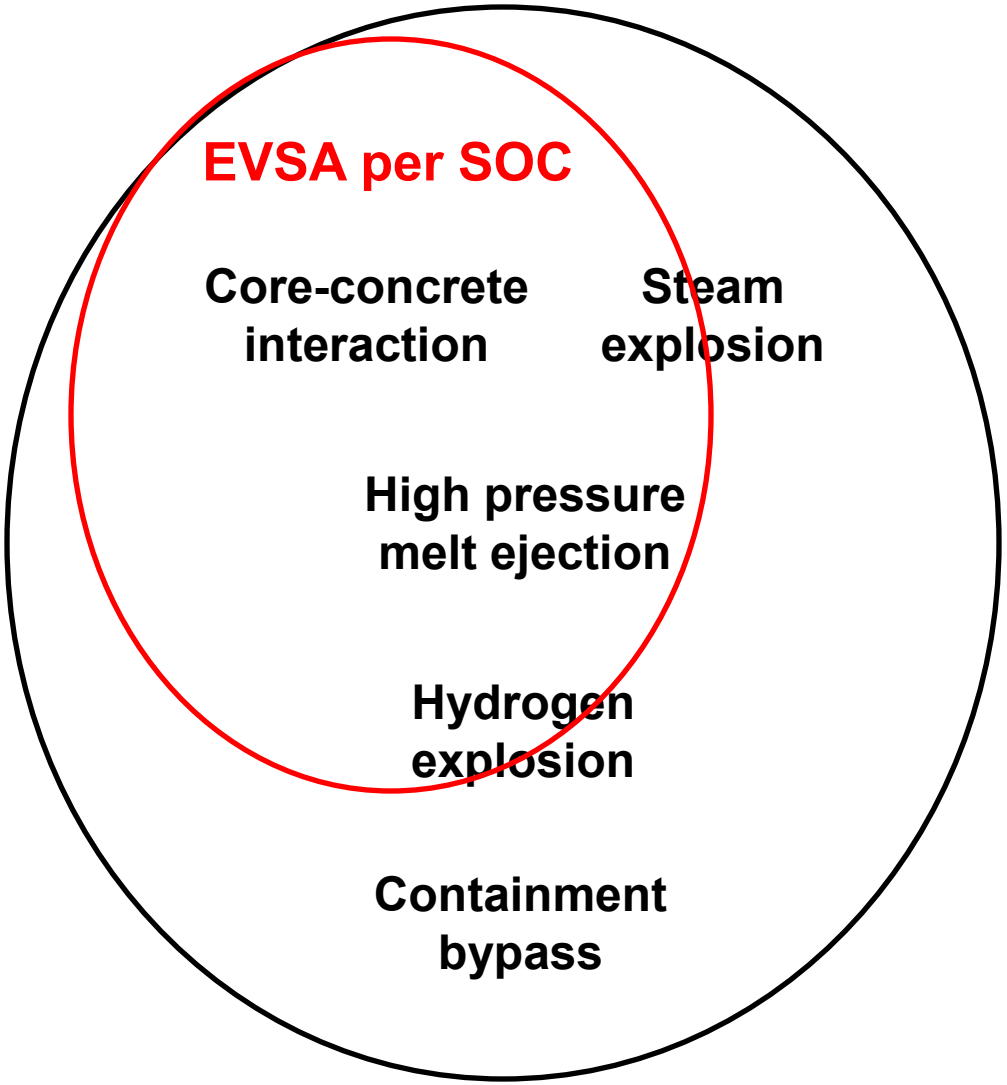
Steam explosion

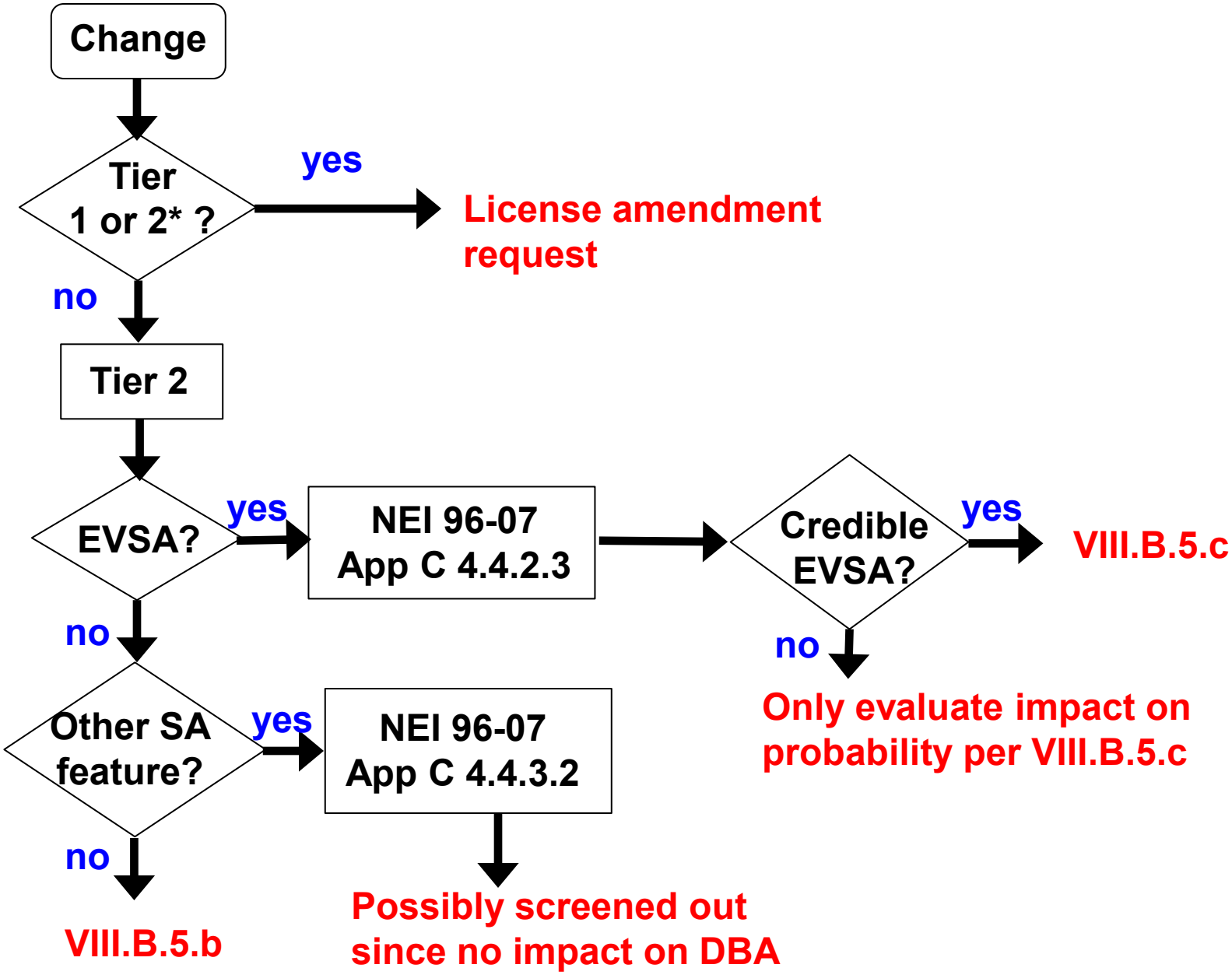
High pressure melt ejection

Hydrogen explosion

Containment bypass

Gap Identified




Change
Tier 1 or 2* ?
yes
License amendment request
no
Tier 2
EVSA?
yes
NEI 96-07 App C 4.4.2.3
Credible EVSA?
yes
VIII.B.5.c
no
Only evaluate impact on probability per VIII.B.5.c
no
Other SA feature?
yes
NEI 96-07 App C 4.4.3.2
Possibly screened out since no impact on DBA
no
VIII.B.5.b

Staff's Preliminary Gap Assessment

- Staff reviewed severe accident features for ABWR, AP1000, and ESBWR
- No significant gaps of concern
  - Either it is an EVSA feature and VIII.B.5.c criteria will be used for Tier 2 changes, or
  - If it is a non-ex-vessel severe accident feature, there is generally sufficient detail in Tier 1 as to preclude a significant design change without prior NRC approval
- Staff is verifying preliminary conclusions
- Other standard designs eventually to be addressed
- Await Commission direction per Recommendation 1

Example for ABWR

DESIGN	DESIGN FUNCTION & SSCs	EVSA * FEATURE?	AFFECTS PROBABILITY OR CONSEQUENCE OF EVSA?	DCD SECTION	GAP IN CHANGE PROCESS?
ABWR	AC-independent water addition system (ACIWA) provides the ability to flood the lower drywell via drywell sprays. It can also provide a means of injecting emergency makeup water to the reactor by cross connecting Division C of the RHR System to the Reactor Building Fire Protection System header, or alternately utilizing additional sources of water from an external connection just outside the Reactor Building.	YES, those aspects for flooding the lower drywell via drywell sprays	Consequences	Tier 1, Section 2.15.6 (very limited details) Tier 2, Section 19.8.7 Tier 2, Table 19.8-7	NO Evaluate under VIII.B.5.c Tier 1 changes require NRC approval
		NO, those aspects for injecting emergency makeup water into the reactor	Neither. This feature contributes to severe accident prevention but does not meet the definition of an EVSA feature.	Tier 1, Sections 2.4.1 and 2.15.6 Tier 2, Section 19.8.7 Tier 2, Table 19.8-7	NO Evaluate under VIII.B.5.b as appropriate only with regard to design basis accidents Tier 1 changes require NRC approval

ABWR Tier 1 Section 2.4.1

AC Independent Water Addition Mode

Division C of the RHR System also functions in an AC independent water addition mode. This mode provides a means of injecting emergency makeup water to the reactor by cross connecting the Reactor Building Fire Protection (FP) System header, or alternately utilizing additional sources of water from an external connection just outside the Reactor Building. This makes it independent of the normal safety-related AC power distribution network. This mode is accomplished by manually opening two in-series valves on the cross-connection piping just upstream of the tie-in to the normal RHR piping. This is accomplished by local manual action at the valves. Fire Protection System water can be directed to either the RPV or the drywell spray sparger by local manual opening of the Division C RHR injection valve or the two Division C drywell spray valves. “Local manual” as used in this paragraph means manually operating the valves at the valves.

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ROP Tabletop Exercises

ROP Tabletop Exercises

- ROP Background and Framework
- Existing Risk-informed Guidance and Thresholds
- ROP Tabletop Approach
- Tabletop Results and Conclusions
- ROP Options and Recommendation

Background: ROP Objectives

- Improve the Objectivity of the Oversight Processes - Subjective Decision-making is Minimized
- Improve the Scrutability of NRC Actions – Regulatory Response and NRC Actions Have a Clear Tie to Licensee Performance
- Risk-inform the Processes - NRC and Licensee Resources are Focused on Performance Deficiencies With the Greatest Impact on Safe Plant Operation

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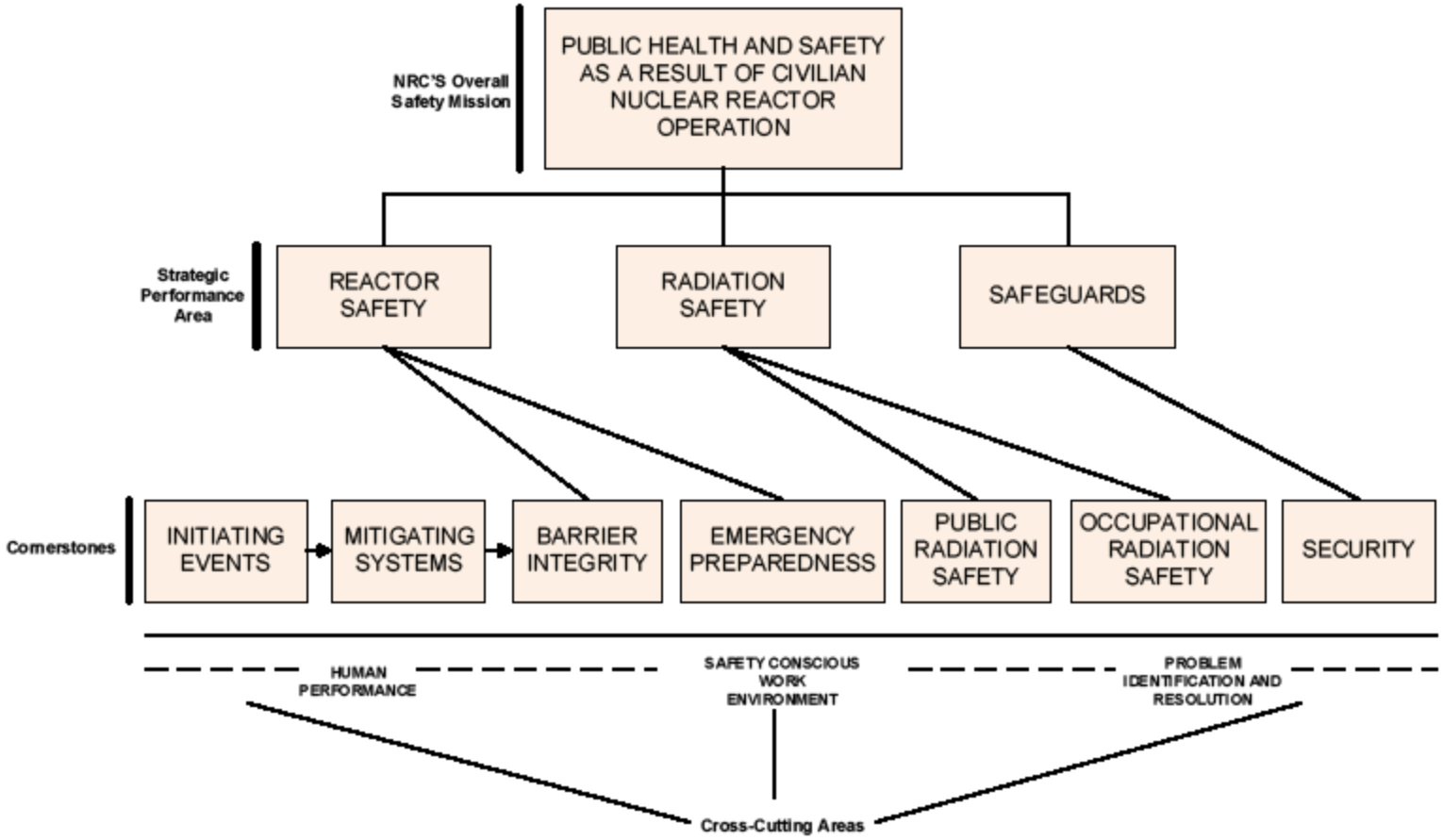
REGULATORY FRAMEWORK



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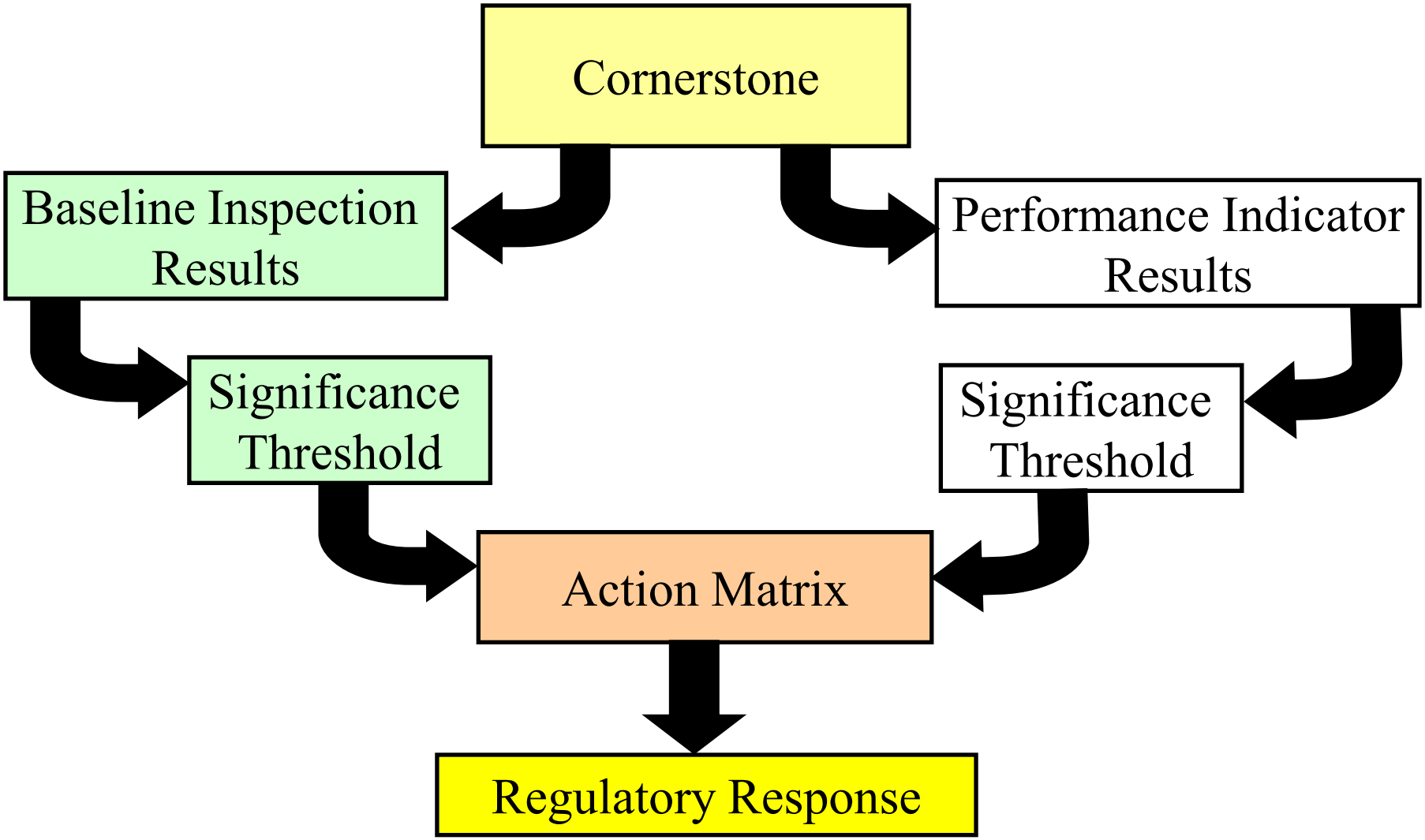


SDP Guidance

- Implementation Guidance in IMC 0609, “Significance Determination Process,” and IMC 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations”
- Appendix A and a few others use risk insights to inform regulatory response. Several other SDPs are more deterministic
- Risk thresholds are a function of changes in core damage frequency (CDF) and large early release frequency (LERF) against a plant’s baseline risk

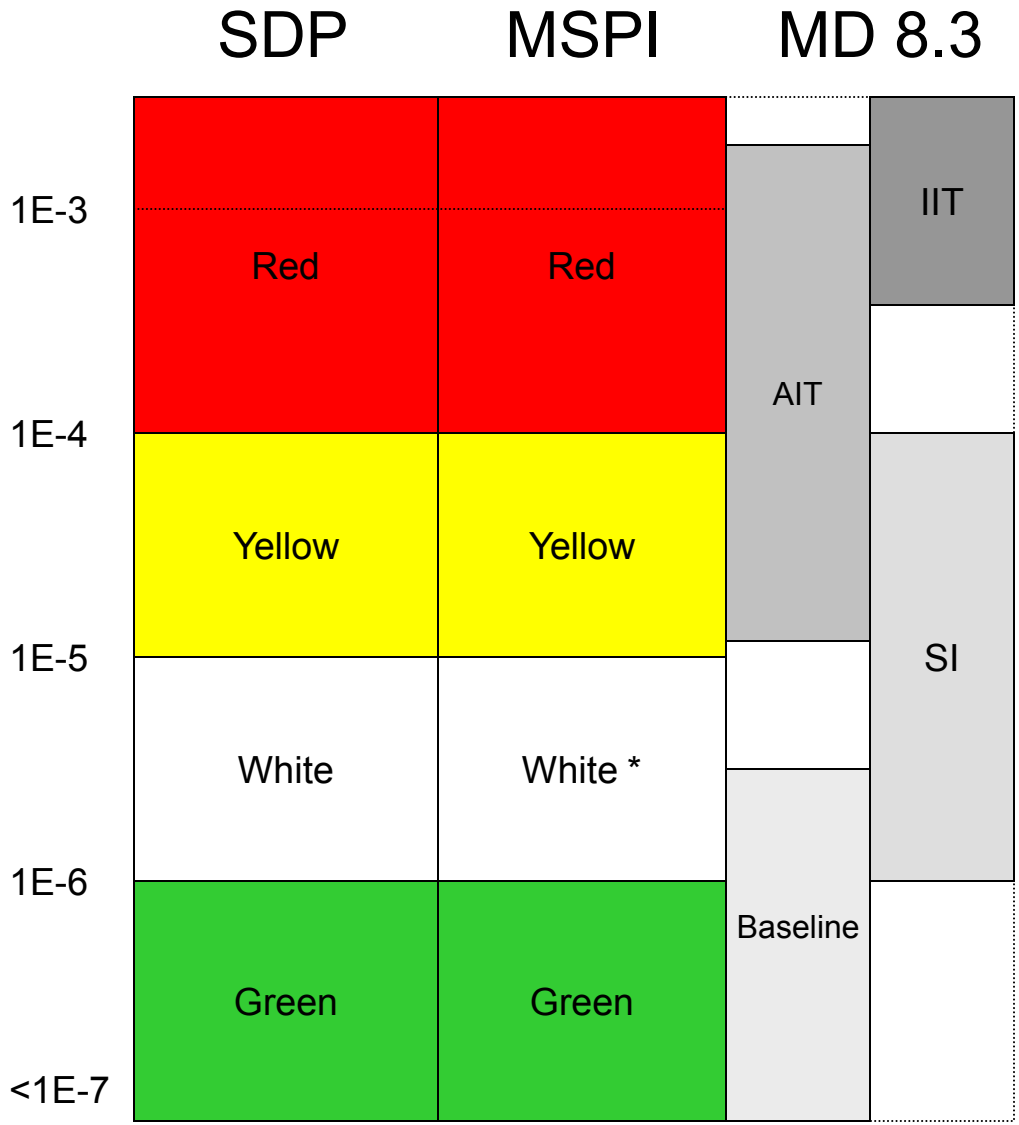
MSPI Guidance

- Implementation guidance in IMC 0608, “Performance Indicator Program,” and NEI 99-02, “Regulatory Assessment Performance Indicator Guideline”
- Covers five systems important to safety
- Tracks the availability of monitored trains and the reliability of monitored components
- Reflects the deviation of a specific unit’s performance from an industry baseline, converted to a simplified change in CDF
- A performance limit is also used for determining degraded performance

Event Response Guidance

- Implementation guidance in MD 8.3, “NRC Incident Investigation Program,” and IMC 0309, “Reactive Inspection Decision Basis for Reactors”
- Reactive inspection thresholds are a function of conditional core damage probability and conditional large early release probability
- Overlap of options based on uncertainty and deterministic insights provides flexibility
- Additional deterministic criteria reviewed and documented as basis for staff decision in overlap region

ROP Risk-Informed Thresholds



ROP Tabletop Approach

- Tested various realistic scenarios to confirm the adequacy of the current ROP risk-informed processes for regulatory decision-making or identify areas for improvement
- Used a broad cross-section of well-vetted cases, developed from actual greater-than-green examples from the current fleet of reactors:
  - Significance Determination Process (SDP) findings
  - Mitigating Systems Performance Index (MSPI) data
  - Management Directive (MD) 8.3 event response
- Applied similar situations to the new reactor designs, filling in gaps with realistic hypothetical situations and reasonable assumptions, and then compared the risk values and resultant regulatory response

SDP Tabletops

RESULTS

- Existing risk thresholds for determining significance of inspection findings are adequate
- Greater-than-green inspection findings would likely involve common cause failures and/or long exposures of risk-significant components
- Existing process does not always ensure an appropriate regulatory response for degradation of passive components and barriers

CONCLUSION

- SDP analyses could be augmented with additional qualitative considerations (deterministic backstops) to appropriately address performance issues

SDP Case Study Example

- Vessel head degradation resulted in a marginally white finding for the AP1000 and a green finding for US-APWR for LOCAs based on risk numbers
- Resultant regulatory response would be to move the AP1000 facility to Regulatory Response Column (column 2) of ROP Action Matrix and perform a 95001 supplemental inspection
- A more robust and diagnostic supplemental inspection (such as a 95002 or 95003) would better ensure root causes that led to the degradation are identified and corrected

Potential Deterministic Backstops

- Maintaining barrier integrity for fuel cladding, reactor coolant system pressure boundary, and containment
- Extensive equipment outage times resulting from degraded conditions (similar to the RITS 4b backstop completion time)
- Repetitive equipment failures that could degrade the reliability or availability of SSCs from performing their intended safety functions
- Designed to capture the infrequent yet potentially significant performance issues that would not otherwise be captured by the risk calculations to ensure an appropriate regulatory response

MD 8.3 Tabletops

RESULTS

- Existing risk thresholds for invoking reactive inspections are adequate for new reactors
- Deterministic criteria used initially for event screening and then within a range of response determined by risk values
- Risk values heavily influence whether or not a reactive inspection is warranted and, if so, at what level
- Variations in or minor revisions to risk models used can potentially result in an inadequate response

CONCLUSION

- Contribution of existing deterministic criteria could be modified or new deterministic criteria developed for initiating reactive inspections

MD 8.3 Case Study Example

- For steam generator tube rupture at an AP1000 facility, the submitted Westinghouse PRA indicated only an SIT could be performed in accordance with the existing guidance
- The slightly more conservative SPAR model resulted in just crossing the threshold to consider an AIT
- An AIT would better ensure a broader and more comprehensive understanding and analysis of causes, conditions, and circumstances of event

MSPI Tabletops

RESULTS

- Existing MSPI is not adequate and would be largely ineffective in determining an appropriate regulatory response for active new reactor designs
- Meaningful MSPI may not even be possible for passive systems using the current formulation of the indicator
- Existing performance limit (backstop) could be further leveraged for active new reactor designs

CONCLUSION

- Alternate PIs in the mitigating systems cornerstone could be developed and/or additional inspection could be used to supplement insights currently gained through MSPI

MSPI Case Study Examples

- EPR emergency diesel generator (EDG)
  - > 25 EDG start failures or > 25 EDG run failures for the EPR to exceed the green-white threshold, - or –
  - 12 failures to reach performance limit
- US-APWR emergency feedwater pump (EFP)
  - > 14 turbine-driven EFP failures or > 25 motor-driven EFP failures for the US-APWR to exceed the green-white threshold, - or –
  - 6 failures to reach performance limit
- Current operating reactors: 2 to 5 EDG or EFP failures would typically result in white indicator

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Draft Commission paper:

Conclusions, options and recommendations to the Commission

Major Conclusions

- During the tabletop exercises for licensing applications, the staff did not identify any potentially significant decreases in the enhanced safety margins for new reactors
- Identified potential gap in the Tier 2 change process regarding severe accident features that are not related to ex-vessel severe accident prevention and mitigation
- Current risk thresholds are appropriate for ROP; however, a few changes to the ROP may be warranted consistent with the integrated risk-informed principles in RG 1.174

Specific Results

- **Risk-informed ISI: No gaps**
  - Risk-neutral effect for a new active plant and a new passive plant, even when sensitivity studies used more restrictive acceptance criteria
  - Numerous regulatory and programmatic controls (e.g., inspection of a minimum set of weld locations is required regardless of risk levels)
  - The 10 year ISI program is dynamic and allows for incorporation of lessons learned and update to risk ranking consistent with Part 52 requirements for PRA maintenance/upgrades

Results (cont.)

- **RITS 4b (completion times): Two key programmatic controls**
  - The risk-informed completion time is limited to a deterministic maximum of 30 days (referred to as the backstop completion time) from the time the TS action was first entered
  - Voluntary use of the risk-managed TS for a configuration which represents a loss of TS specified safety function, or inoperability of all required safety trains, is not permitted

Results (cont.)

- **RITS 4b staff exercises**
  - Staff identified some configurations of equipment outages that would represent 10 years' worth of core damage probability
  - Repeated entry into such condition over time could increase CDF by one or more orders of magnitude, which could approach the baseline CDF of currently operating plants
  - Staff believes these configurations are unlikely or unrealistic, and that there were additional regulatory and programmatic controls that would limit the aggregated risk increase (e.g., performance monitoring, periodic PRA maintenance and upgrade under 50.71(h))

- **Staff concludes no substantive changes to methodology is necessary**

Results (cont.)

- **Maintenance Rule 50.65 (a)(4): No gaps in assessment and management of risk**
  - When PRA approach is combined with other inputs such as the degree of defense in depth and plant transient assessment, factors other than PRA are often more limiting in terms of the risk management action level
  - NUMARC 93-01, Section 11 explicitly acknowledges “there is acknowledged variability in baseline core damage frequency and large early release frequency… determination of the appropriate quantitative risk management action thresholds are plant-unique activities”
  - Some changes to NUMARC 93-01 may be necessary to address changes of scope because of new and different SSCs in the new reactor designs

Results (cont.)

- **RITS 5b (surveillance frequency control program): No gaps**
  - Surveillance frequencies that are controlled by other programs are excluded from the SFCP
  - Equipment covered by inservice testing, for example major pumps and valves, tend to have some of the highest risk importances but are excluded
  - What remains to be implemented under RITS 5b generally are lower risk importance components
  - Unlike RITS 4b, RITS 5b is much more deterministically oriented, with risk impact only a secondary consideration in the criteria for changing surveillance test interval

Results (cont.)

- **50.69: No gaps**
  - Sample application to new reactor design (PWR with active safety systems) shows approximately same categorization distribution (RISC-1,2,3 & 4) as South Texas 1 & 2 pilot based on importance measures
  - Rule has built-in measures to monitor RISC-3 components and take corrective actions (e.g., periodic program review every 2 refuel cycles)

Results (cont.)

- **RG 1.174: No gaps**
  - In many of the examples during the exercise, the estimated change in core damage frequency (ΔCDF) was observed to be very low and well below Region II of the acceptance guideline per Figure 3 of RG 1.174
  - Degradation of defense in depth would be an area of close review by the staff
  - Changing a plant feature from highly passive to active thus placing greater reliance on key operator actions would be an area for close review by the staff
  - Proposed changes in or near the boundary of Region II would undergo close scrutiny by the staff, and there would need to be a compelling reason on the part of the license holder for the proposed change. Serious consideration of alternatives with lower risk impact would need to be assessed by the licensee

Results (cont.)

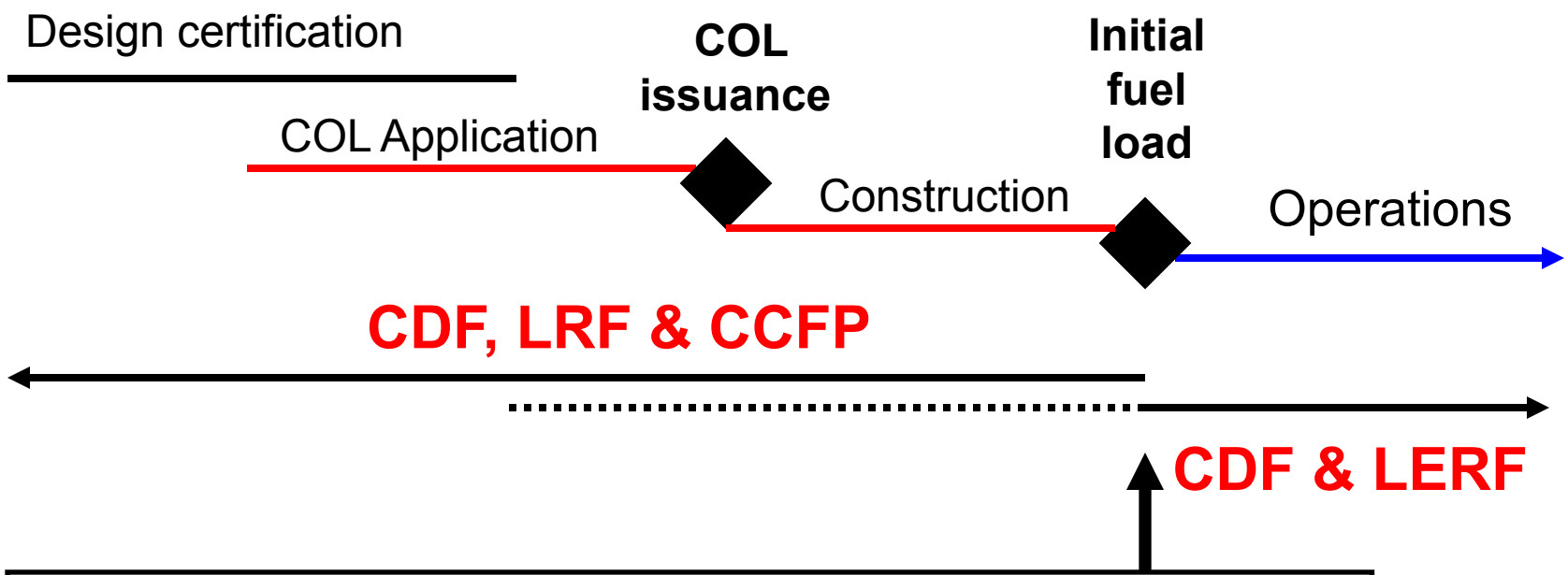
- **Part 52 50.59-like change process: Gap identified**
  - Staff generally satisfied with “ex-vessel” portion of NEI 96-07 Appendix C; staff comments incorporated
  - However, changes to severe accident design features that are not specifically intended to address EVSAs (e.g., containment bypass) are not addressed using severe accident criteria as in Section VIII.B.5.c.
- **Recommendation 1**

  Address the potential gap, by a) ensuring that there are sufficient details on all key severe accident features in Tier 1, and b) including a change process in future design certification rulemakings in Section VIII for *non-ex-vessel severe accident features* similar to Section VIII.B.5.c for *ex-vessel severe accident features*

LRF-to-LERF Options

- Option 2A: continue use of LRF (& CCFP) indefinitely
- Option 2B: continue use of LRF (& CCFP) indefinitely and add LERF at initial fuel load
- Option 2C: transition from LRF to LERF at or prior to initial fuel load; discontinue regulatory use of LRF (& CCFP) thereafter

Option 2C



- **LERF calculated at or prior to initial fuel load. CDF & LERF used for RG 1.174 acceptance guidelines going forward.**
- **Last regulatory use of LRF & CCFP**

Results (cont.)

- **Recommendation 2**

  Staff recommends Option 2C to harmonize risk-informed applications for the new reactors consistent with the risk metrics used by the currently operating fleet

Results (cont.)

- **Other programs not assessed in tabletops**
  - Risk-informed inservice testing of pumps and valves (RG 1.175)
  - Integrated leak rate testing interval extension (NEI 94-01)
  - 50.46a
  - NFPA 806
- **Little short-term interest by COL applicants**
- **Alternative source term (RG 1.183) implemented at all new designs with COLAs except ABWR**

Objectives for ROP Options

- Maintain current risk thresholds for new reactor designs
- Consistent with integrated risk-informed decision-making concepts in RG 1.174
- Afford greater operational flexibility based on enhanced safety margins

ROP Options

OPTION A: USE AS IS

- Use the existing risk-informed ROP tools for new reactor applications without making any changes
- No additional action or resources needed, but existing tools may not always provide for an appropriate regulatory response

ROP Options (cont.)

OPTION B: AUGMENT EXISTING PROCESSES

- SDP: Use existing risk-informed SDP, but augment with deterministic backstops to ensure an appropriate regulatory response to address performance issues
- MD 8.3: Modify the contribution of existing deterministic criteria or develop new criteria for determining the appropriate regulatory response to plant events
- MSPI: Develop alternative to MSPI or augment existing guidance to emphasize performance limit for active new reactor designs, and increase inspection of passive mitigating systems for passive new reactor designs
- Proposed enhancements could be developed using existing resources and working with stakeholders

ROP Options (cont.)

OPTION C: DEVELOP DETERMINISTIC TOOLS

- Do not use the existing risk-informed ROP tools
- Capture risk insights to a lesser extent than the current fleet using deterministic guidance consistent with new reactor design certification and licensing basis
- Additional resources may be necessary to research and develop the new guidance documents

ROP Recommendation

Staff Recommendation: Option B

Augment Existing Processes

- Staff would obtain Commission approval for proposed changes to ROP at least one year prior to implementation
- Process enhancements could be further refined based on experience and lessons learned

Next steps

- Full ACRS April 12
- Finalize Commission paper based on ACRS and stakeholder feedback
- SECY due to be issued early June, 2012

New Plant Risk Framework

ACRS PRASC
March 7, 2012

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Risk-Informed Applications for New Plants

- New plants have PRA requirements under 10 CFR 50.71
- New plants have safer designs benefiting from PRA insights
- New plants should be encouraged to use available risk-informed applications

NEI

Risk-Informed Applications for New Plants

- Significant voluntary risk-informed regulatory applications are available for operating plants
  - Technical Specifications, Inservice Inspection, Risk-Informed scope for special treatment regulations
- Mandatory risk uses include maintenance rule, reactor oversight process

NEI

NRC Tabletop Exercises

- Consistent with Commission directive
- Well organized
- Good participation from NRC staff and industry
- Examples were realistically constrained while still testing guidance
- Industry in general agreement with NRC findings in draft SECY

NEI

Industry Observations

- Agree with NRC conclusions relative to licensing applications
- Agree that elements of existing reactor oversight process will need to be replaced or rethought for new plants
  - Details of additional qualitative considerations are important

NEI

Industry Observations

- Agree that there is a potential gap in change process considerations for new plants with respect to “other than ex vessel” severe accident mitigation features
- This gap has been addressed in practice thus far, and can be dealt with through Tier 1 descriptions in future DCRs

Conclusions

- Thorough effort to address gaps in guidance, and responsive to Commission direction
- Industry supportive of conclusions
- We look forward to implementing risk-informed applications and supporting dialogue on revised ROP considerations and any necessary enhancements to change control guidance

NEI