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

December 21, 2017

MEMORANDUM TO: ACRS Members

FROM: Quynh Nguyen, Senior Staff Engineer ***/RA/***
Technical Support Branch, ACRS

SUBJECT: CERTIFIED MINUTES OF THE ACRS PLANT OPERATIONS AND FIRE PROTECTION SUBCOMMITTEE MEETING ON SEPTEMBER 20, 2017

The minutes of the subject meeting were certified on December 15 2017, as the official record of the proceedings of that meeting. Copies of the certification letter and minutes are attached.

Attachments: As stated

cc w/ att. A. Veil
M. Banks

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

MEMORANDUM TO: Quynh Nguyen, Senior Staff Engineer
Technical Support Branch, ACRS

FROM: Gordon R. Skillman, Chairman
Plant Operations & Fire Protection Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE PLANT OPERATIONS AND FIRE PROTECTION SUBCOMMITTEE MEETING ON SEPTEMBER 20, 2017

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

/RA/ **December 15, 2017**

Gordon Skillman, Chairman Dated
Plant Operations and
Fire Protection Subcommittee

Certified on: December 15, 2017
Certified by: Gordon Skillman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF THE ACRS PLANT OPERATIONS AND FIRE PROTECTION SUBCOMMITTEE MEETING
SEPTEMBER 20, 2017

The ACRS Plant Operations and Fire Protection Subcommittee held a meeting on September 20, 2017 in T2-B1, 11545 Rockville Pike, Rockville, Maryland. The meeting convened at 12:59 PM and adjourned at 5:37 PM.

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/Consultants/Staff
GORDON R. SKILLMAN, Chairman
RONALD G. BALLINGER, Member
DENNIS C. BLEY, Member
WALTER L. KIRCHNER, Member
HAROLD B. RAY, Member*
JOY REMPE, Member
JOHN W. STETKAR, Chairman
MATTHEW W. SUNSERI, Member

Designated Federal Official:
QUYNH NGUYEN

NRC Staff:
ODENAYO AYEGBUSI, NRO
STEVE CAMPBELL, NRR
CHRIS CAUFFMAN, NRR
RUSSELL GIBBS, NRR
MICHELLE HAYES, NRR
MATT LEECH, NRR
DANIEL MERZKE, NRR
CHRISTOPHER MILLER, NRR
JEFFREY MITMAN, NRR
ANDREA D. VEIL, Executive Director, ACRS
SEE-MENG WONG, NRR

*Present via telephone

SUMMARY

The purpose of the meeting is an information briefing regarding the staff's proposed updates to the reactor oversight program (ROP) for new reactors. 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.

SIGNIFICANT ISSUES	
Issue	Reference Pages in Transcript
The staff deliberately focused on AP1000 because it is the one currently under construction.	11
Member Stetkar questioned the staff's statement that the baseline risk for new reactors was lower than the current operating fleet. The staff responded that all modes of operation were accounted for and that the current ROP is robust factoring internal events and external hazards.	15-17
In the current ROP, the focus is on SSCs (structures, systems and components). For new reactors with passive systems, Chairman Skillman asked how are subtle things, such as degradation of these passive systems, detected before a bigger problem occurs. The staff responded that the key is the inspection program. Chairman Skillman brings up the passive systems discussion several times.	18-21, 46-51, 172, 196
Member Bley stated that SPAR models need to be completed for new reactors.	22, 44
Mr. Merzke gave an overview and background of the Commission direction regarding the ROP. Member Rempe began a discussion about the lack of qualitative factors incorporated in the program. The Commission disapproved of using qualitative factors to supplement quantitative risk evaluations.	23-29
Mr. Merzke provided an overview of the ROP, including the Action Matrix.	35-44
Mr. Ayegbusi discussed performance indicators. Since the ROP is technology neutral, Member Kirchner asked what happens for different plant designs which may remove traditional barriers. The staff answered that when a new design is considered, the ROP is re-evaluated; performance indicators may be replaced by new or increased inspections.	57-59, 204
Member Stetkar began a discussion regarding the perception of arbitrarily determining the color of findings.	60-67
Member Stetkar started a discussion regarding squib valves, a component that is not tested often but could be an important contribution	68-81, 189

to overall reactor risk. He believes that squib valves in a new reactor with low core damage frequency is not an appropriate argument for infrequent or non-existent testing.	
Discussion regarding the maintenance rule.	82-90
Member Sunseri stated that the ROP is effective due to vast operating experience and wondered how the ROP for new reactors will be implemented since no operating data exists currently.	90-95
Member Ray wanted to remind the Committee to not mix design certification and reactor oversight concerns.	107
Mr. Campbell discussed additional inspections to cover for performance indicator shortfalls. Member Stetkar asked how the staff knows when they have enough operational experience. He also asked how the staff can anticipate preventive instead of reactive thinking. The discussion continued regarding mitigating safety performance indicators (MSPI). Mr. Mitman stated that there are no PIs for shutdown (119). Chairman Skillman stated the staff determined that additional inspections are not needed to compensate for not using MSPI indicators; why are MSPI needed (123).	108-124
Member Stetkar asked for clarification of inspection resources. Mr. Miller offered an explanation that most of the systems are within containment so inspections can only occur during shutdown. Chairman Skillman believes that there is an aversion to at-power containment entries.	125-141
Mr. Gibbs, the staff lead in the cornerstones, and Mr. Wong in risk assessment begin a discussion regarding the significance determination process (SDP). Member Powers discusses the process in which the SDP is improved and Member Rempe discusses SPAR models.	142-158
Chairman Skillman wants the staff to consider passive physical functions as physical conditions.	162
Chairman Skillman asked about the status of Appendix M. Member Rempe wanted to clarify that staff would engage with external stakeholders.	180-184
Mr. Leech begins a discussion on Appendices G and H.	186-194
Member Ray clarified that there are other systems and components in containment.	191
Mr. Merkze makes concluding remarks.	194

Documents provided to the Subcommittee

1. White Paper with Staff Recommendations for the Reactor Oversight Process for New Reactors
2. SECY-10-0121, “Modifying the Risk-Informed Regulatory Guidance for New Reactors,” dated September 14, 2010.
3. SECY-13-0137, “Recommendations for Risk-Informing the Reactor Oversight Process for New Reactors,” dated December 17, 2013.

Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: Advisory Committee on Reactor Safeguards Plant Operations and Fire Protection Subcommittee

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

PLANT OPERATIONS AND FIRE PROTECTION SUBCOMMITTEE

+ + + + +

WEDNESDAY

SEPTEMBER 20, 2017

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ROCKVILLE, MARYLAND

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The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 12:59 p.m., Gordon R. Skillman, Chairman, presiding.

COMMITTEE MEMBERS:

GORDON R. SKILLMAN, Chairman

RONALD G. BALLINGER, Member

DENNIS C. BLEY, Member

WALTER L. KIRCHNER, Member

DANA A. POWERS, Member

HAROLD B. RAY, Member*

JOY REMPE, Member

JOHN W. STETKAR, Chairman

MATTHEW W. SUNSERI, Member

DESIGNATED FEDERAL OFFICIAL:

QUYNH NGUYEN

ALSO PRESENT:

ODENAYO AYEGBUSI, NRO

STEVE CAMPBELL, NRR

CHRIS CAUFFMAN, NRR

RUSSELL GIBBS, NRR

MICHELLE HAYES, NRR

MATT LEECH, NRR

DANIEL MERZKE, NRR

CHRISTOPHER MILLER, NRR

JEFFREY MITMAN, NRR

ANDREA D. VEIL, Executive Director, ACRS

SEE-MENG WONG, NRR

*Present via telephone

T-A-B-L-E O-F C-O-N-T-E-N-T-S

PAGE

P R O C E E D I N G S

12:59 p.m.

CHAIRMAN SKILLMAN: This meeting will come to order. This is a meeting of the Plant Operations and Fire Protection Subcommittee of the Advisory Committee on Reactor Safeguards.

I'm Gordon Skillman, Chairman of the Subcommittee. ACRS Members in the room are Ronald Ballinger, Matt Sunseri, Dana Powers, Joy Rempe, Dennis Bley, Walter Kirchner, and we anticipate we will have John Stetkar.

We also have Harold Ray participating by the bridge line. Quynh Nguyen of the ACRS Staff is the designated federal official for this meeting.

The Subcommittee will hear Staff presentations on the proposed update of the Reactor Oversight program in regards to new reactors.

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

The ACRS was established by a statute and is governed by the Federal Advisory Committee Act. This means that the Committee can only speak through its published letter reports.

We hold meetings to gather information to support our deliberations. Interested parties who wish to provide comments can contact our Offices requesting time after the meeting announcement is published in the Federal Register.

That said, we also set aside time for spur-of-the-moment comments from members of the public attending or listening in. Written comments are also welcome.

The ACRS Section of the U.S. NRC public website provides our charter, bylaws, letter reports, and full transcripts, of all full and Subcommittee meetings, including slides presented at the meeting.

The rules for participation in today's meeting were previously announced in the Federal Register. We have received no written comments or requests for time to make oral statements from members of the public regarding today's meeting.

We have a bridge line established for interested members of the public to listen in. To preclude interruption of the meeting, the phone bridge line will be placed in a listen-in mode during the presentations and Committee discussions.

We will unmute that bridge line at a designated time to afford the public an opportunity to

make a statement or provide comments.

At this time, I request the meeting attendees and participants please silence your cell phones and any other electronic devices that are audible.

A transcript of the meeting is being kept and will be made available as stated in the Federal Register Notice.

Therefore, we request that participants in this meeting please 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.

Make sure that the green light of the microphone is on before speaking, and please turn that green light off when not in use.

We will now proceed with the meeting and I call on Chris Miller, Director of Inspection and original support of the NRC Staff, to begin. Chris?

MR. MILLER: Thank you, Chairman Skillman, and Members of the ACRS.

First of all, we just appreciate your time and the look that you're going to be giving to

this program as we're modifying the Reactor Oversight Process to accommodate new reactors.

The Staff values your insights in the discussion that we have about our progress so far. Of course, this has been a multi-year effort involving stakeholders, internal and external, and a number of public meetings.

And to be frank, we have a few more years before we'll be putting this version of the ROP into practice. So, we have some time to make some tweaks. So, this is a good time to be discussing this process.

While the paper and the recommendations the Staff developed to modify the ROP focused mainly on AP1000, we do think that we have a model that can be applied to any of the Advanced Reactor Series. So, we could use this as we go forward for other designs.

The Staff has concluded that only modest changes to the ROP should be necessary to ensure adequate oversight of new reactor designs which have a much lower baseline core damage frequency.

In developing the recommended changes, Staff ensured that we followed Commission direction in maintaining the existing ROP framework and existing risk goals, and the guidance associated.

The Commission affirmed in SRM for SECY-

10-0121 that at a minimum, new-generation reactors afford at least the same degree of protection of the public, and the environment that is required for current-generation light water reactors.

But since these reactors have enhanced margins and safety features, we think the program could perhaps allow for more flexibilities. And we'll be discussing some of those.

And finally, we note that the ROP is a living process. We continuously assess the effectiveness of the ROP and the changes to the ROP, and modify when necessary.

We fully expect when the new reactor designs come online, the Staff will evaluate the operating experience and recommend adjustments as part of the annual ROP self-assessment process.

So, again, we look forward to your insights and a healthy discussion. I'd like to turn the meeting over to Dan Merzke.

MR. MERZKE: Thank you, Chris. Good afternoon, my name's Dan Merzke. I'm from the Office of Nuclear Reactor Regulation, NRR.

I'm the current project lead for developing recommendations to modify the Reactor Oversight Process, or henceforth the ROP, to ensure

appropriate oversight of new reactor designs, as the current project lead because I took over the project from Ron Frahm, who some of you may have worked with in the past.

And he took early retirement and bagged me with it. I'd like to thank the Committee for meeting with the Staff today to hear --

MEMBER POWERS: Is early retirement an occupational hazard here?

MR. MERZKE: He left with a smile on his face.

Again, I want to thank the Committee for meeting with Staff to hear your recommendations -- to hear our recommendations and share any insights you have that might contribute to informer file recommendations that we'll be sending to the Commission by the end of the year.

With me today are the leads for the individual ROP program areas, who will be also presenting tonight, or today.

Immediately to my right is Ayo Ayegbusi, from the Office of New Reactors, not nuclear reactor regulation. But he'll be discussing performance indicators.

I would like to note that Ayo is not the

PI Program Lead for NRR. The gentleman who did most of the legwork on this effort was recently transferred to Three Mile Island as a Senior Resident Inspector or so.

He said he's too busy into an outage this week to participate, so Ayo graciously stepped up the plate and helped us out here. So, I thank him.

MR. MILLER: And he also went with a smile on his face.

MR. MERZKE: Yes, he did.

CHAIRMAN SKILLMAN: I will note in advancing that the TMI shutdown went with some fanfare; what's being advertised is this might be the last of the last. That's correct?

And it came down over the weekend. I think it came down either Friday night or Saturday night.

But the Harrisburg community is watching what is happened at Three Mile Island very closely because of employment and also industry interest.

A lot of people after the accident became quite interested in what was happening in Middletown.

So, what's going to happen in this month and then in the two years that follow may have some national significance in terms of preserving one nuke,

a merchant nuke.

MR. MERZKE: And he may be back with us in a couple years after they shut down.

(Laughter.)

Also, to my right is Steve Campbell from NRR, who will be discussing the Baseline Inspection Program changes.

And Russell Gibbs, See-Meng Wong, and Matt Leech, all from NRR, will be discussing the plan changes for the Significance Determination Process.

Last month, the Staff delivered a white paper to the ACRS with the Staff's conclusions and recommendations, modifying the ROP for new reactors.

Members may have noticed that the white paper focuses mainly on the Westinghouse AP1000 reactor design, and does not acknowledge other new reactor designs.

The Staff review deliberately focused on AP1000 because those are the only units that are currently under construction.

And after recent developments, we thought maybe this was just going to be an academic exercise after all. But Vogtle continues, so we'll see.

But the review process that we put in place for this effort, all those processes, the GAP

analyses we did to determine what was missing and what needed to be addressed for the ROP to support these new reactor designs.

That process will be easily transferrable to any of the new reactor designs.

MEMBER POWERS: What kind of lead time before operation do you think you would need?

I mean, you clearly don't know, but do you think you would need to set up an ROP for something like a new scale design?

MR. MERZKE: Well, I don't think the Gap analyses that was conducted for the various programs, the programs under PIs, based on inspection and STP, I don't think would take much more of a year of effort to go through that Gap analysis again, based on --

MEMBER POWERS: But through the Significance Determination Process, you'd need a PRA, and I mean it clearly takes some time.

MR. MERZKE: Yes, we're not done with the Significance Determination Process either. We know what we need to do, what changes need to be made.

We actually have put pen to paper over the next year and actually identified the specifics as to ensuring we've got the right words down.

So, there's still work to be done in those

areas. But at this point, we just know what we need to do with each of the STPs.

MEMBER POWERS: I mean just as a true way would be two years?

MR. MERZKE: That would be a good guess. Thus, we concluded the ROP is robust and flexible enough to provide adequate oversight of the AP1000 with just relatively modest provisions.

Moving on, the purpose of today's meeting is to discuss the Staff's evaluations, conclusions, and recommendations, as noted in the white paper, to modify the ROP for new reactors.

And this is in response to the Staff Requirements Memorandum on SECY-13-0137, which we'll talk about in the background.

For an agenda, I will be discussing the background and overview of the ROP, and I will give a brief tutorial of the ROP for those folks who might be attending that are not very familiar with the ROPP.

I'll turn it over to Ayo, who will discuss the performance indicators and thresholds. And then Steve Campbell will discuss the baseline inspection program changes.

And Russell, See-Meng, and Matt, will discuss the Significance Determination Process issues

that need to be addressed.

And I'll wrap up with conclusions and recommendations that we've developed in the white paper and the steps we're going to be taking following.

How did we get here? The baseline risk estimates for most new reactor designs are significantly lower than those for current-design power pressurized water reactors and boiling water reactors.

Back when the Staff recognized the new designs were going to have significantly lower baseline pro-damage frequencies, there were some questions raised as to whether the correct guidance, risk-informed guidance, for licensing changes in reactor oversight would be sufficient to address these new reactor designs.

So, in the past several years, Staff has corresponded with the Commission and the ACRS to address Staff's recommendations for developing or modifying risk-informed guidance and the oversight process for new light water reactor applications.

MEMBER STETKAR: Dan?

MR. MERZKE: Yes?

MEMBER STETKAR: I don't know when it's

appropriate to ask this, but tell me if you're going to address it later.

But you said that the baseline risk estimates for new reactors are, you used the term, much lower, you just used the term lower here, than currently-operating reactors.

And you used that argument in several of the papers that I've read about concerns about specific metrics and things like that.

The current ROP measures things against the core damage frequency from internal events during power operations. Is that correct?

MR. MERZKE: Yes, sir, in general, yes.

MEMBER STETKAR: Okay, well, one might find that for new reactors, it's more important to consider all hazards and all modes of operation.

Because for example, systems like service water systems and complement cooling water systems, that you note are not important at all for the AP1000 in the way they measure importance during shutdown modes, are really, really important during shutdown modes.

And one might find that the risk during shutdown is comparable to or greater than the risk during power operation, if one actually had a shutdown

risk assessment, which they don't.

And one might find that the important of some systems and components to large release frequency or large early release frequency is much more important than it is to core damage frequency.

So, how does the Reactor Oversight Process capture those elements of whether you want to call it plant safety or plant risk or what attributes you want to give to those things?

Simply because you're looking at one small chunk of risk and saying that somebody's calculation of that small chunk is smaller than somebody else's calculation of the small chunk for a currently-operating reactor.

MR. GIBBS: So, Dan, may I?

MR. MERZKE: Yes, go ahead.

MR. GIBBS: So, the first thing we want to point out is the current STP and the ROP looks at all modes of operations, not just power operation. That's the first point.

And we have our Staff from our Division of Risk Assessment here, who is actually our shutdown specialist in covering shutdown and, actually, low power. So, that's important to acknowledge that.

So, then the second point is that we have

many, many tools to address all the aspects of the current operating fleet, and we're going to be looking at the tools for new reactors.

Now, we believe the current ROP, STP, is robust, as Dan indicated earlier. That would address operational aspects of new reactors as well.

And we're going to talk through that sum today about what we believe needs to change, and what we believe does not need to be changed.

And so, also, I wanted to share with you, when we do an analysis, particularly those that are more we'll call them risk-informed or more based on probabilistic risk assessment, we're looking at more than the internal events' analysis.

We're also looking at external hazards. We're looking at large early release frequency.

So, each of the inspection findings that we examined for significance that are not screen to green, or very low safety significance, we are looking at all those other aspects of risk, not just the internal events' analysis.

I just wanted to make sure we were very clear about that.

MEMBER STETKAR: Okay, well I'll wait until later in the discussion because I'm more

concerned about --

CHAIRMAN SKILLMAN: I'd like to build on John's question if I could, please? So much of what we've experienced for the past couple decades on ROP has been focused on SSCs, structures, systems, and components.

We know if an RHR pump is operable or not operable, operable but degraded. We know how to make operability calls. We know how to exigent-request those types of things.

But let me propose a situation to you, and ask you how the ROP is going to review it? We've got an up-and-running AP1000. It's maybe on a 24-month field cycle.

It's out 21 months, 100 percent power, not a blip. And it gets a retroactive bump lubricating oil fire, and so they file their text specs, they go into either reduced power shutdown.

But that fire has coated the inside of that containment with something.

It's going to be debris, it's going to be organic debris, and I would offer, having been part of a couple fires, I now have a degraded thermal condition to the wall of that containment.

But it depends on that thermal

conductivity for my emergency core cooling system.

How do you consider obscure, passive, physical characteristics as part of your consideration for inspection and for significance determination, when you look at the review of an event like that, and particularly, the question of continuing to operate?

MR. GIBBS: All right, so if I may, why don't we address the inspection piece of that first, because that's, if you will, the flow of our program? We inspect and then if there's a problem, we determine significance.

So, Steve, did you want to comment about what we're doing in this inspection side with respect to specifically passive components?

Because I will tell you, from a risk-assessment, and this is a very general statement, you know, we have said these new reactors are safer. Well, one of the main reasons they're safer is because they have passive components.

And so from an STP perspective, what I sort of anticipate, if you will, is we need to ensure these passive components are performing their function.

And if they are not, then those types of inspection findings could, indeed, be very risk-

significant. So, I'm agreeing with you in that regard.

CHAIRMAN SKILLMAN: Building on John's comment --

MR. GIBBS: And it's a matter of how we go about assessing significance.

CHAIRMAN SKILLMAN: What I'm suggesting is that there can be some extremely subtle conditions that present themselves to the plant, that really present a risk or a hazard that we've never had to deal with before because we've always depended on big, heavy, well-designed components, pumps, valves, heat-exchangers, fans, blowers.

And now we're depending upon a film coefficient or a column height for natural circulation to ensure that we are going to get the decay heat removal, or whatever else it is, upon which we are depending.

How are those physical phenomena addressed?

MR. MERZKE: I'll just go ahead and pick it up from there, Russell. This is what we would term an event at a plant.

Okay, and after an event occurs, the regional management will be directed to 8.3 and

they'll go through the requirements of evacuating to determine if a reactive inspection is warranted.

In a case like what you're talking about with the reactor coolant pump fire, I would suggest that the core damage frequency, or conditional core damage probability I think is what they used for Management Directive 8.3, would be sufficient to at least launch a special inspection, at the very least.

So, we'll be sending a team of inspectors there to evaluate the events and the licensee's root cause and corrective actions.

In part, those corrective actions would be to expect them to do an analysis or something to fix the containment.

Would somebody recognize the fact that there's a film buildup on the inside of containment? That's a good question. How good are our inspectors? I have to rely on their knowledge and skills.

Most of our inspectors probably are going to be coming from other reactor plants. We probably won't be getting any new inspectors sending out to the AP1000. They'll have experience in other PWR applications, I expect.

But a lot of that, those insights, will have to be -- hopefully, the licensee will catch, if

not, the inspectors will catch.

But those are two levels of eyes who are hopefully going to catch an issue like that where we don't miss it, and expect corrective actions to be taken by the licensee before we allow them to restart.

MEMBER BLEY: Let me get a comment on the writing that I'd like to place there. Number one, as of right now, we don't have any, the Staff doesn't have any, SPAR models for these new plants.

And the PRAs for them by the vendors aren't the complete PRAs that need to be there before fuel load. For every, I think every design cert, at least in the last ten years, that's come up.

We've put a comment in the record that the PRAs, as the State re-saw them as they went through the design cert, had not really looked at these passive features for things that could degrade them.

So, if we have the PRAs that exist now and you had SPAR models as they would exist now if they were based on those, and if you had a finding that we had a fire that surfaced and they went to the PRA, they'd say, well, there's nowhere to put this in so there's problem.

So, between now and whenever this goes into place, those risk assessments need to be upgraded

to the point that you could address these issues. The issues aren't simple or subtle but they're knowable, and they really need to get factored in.

So, that's not something that you guys are doing right now but I think everybody needs to stay aware of that.

MR. MERZKE: Yes, sir. Thank you, it's noted.

CHAIRMAN SKILLMAN: Please proceed.

MR. MERZKE: Thank you. And I guess we'll go onto the next slide.

Quickly going over the background of how we got here, the first paper that the Staff submitted was back in 2010, the SECY-10-021, modifying the risk-informed regulatory guidance for new reactors.

The Staff provided the Commission options to modify the risk-informed regulatory guidance for new reactors.

And that included options for modifying the Reactor Oversight Process to adjust for the lower baseline core damage frequency of these new designs.

The Staff recommended in that paper working with stakeholders to identify and implement appropriate changes to the existing risk-informed guidance, because of those lower baseline risk

frequencies.

In the SRM, the Commission disapproved the Staff's recommendations and reaffirmed that the existing safety goals, safety performance expectations, subsidiary risk goals, and associated risk guidance, key principles of quantitative metrics for implementing risk-informed decision-making, are sufficient for these new plants.

I quote that because this Staff hangs its hat on those words and maintaining the risk thresholds that the current ROP possesses.

In a subsequent paper to the Commission, SECY-12-0081, Risk-Informed Regulatory Framework for New Reactors, the Staff provided additional recommendations on both licensing and oversight processes.

Staff recommended the development of quantitative risk insights, originally termed deterministic backstops, to supplement the probabilistic risk assessment information to determine significance of inspection findings.

In that SRM, the Commission also disapproved the Staff's recommendation and directed the Staff to consider using relevant note risk metrics, in order to provide a technical basis for why

this option was not viable.

I believe the relative risk metric was an idea provided in our letter from the ACRS at the time. And the Staff subsequently did an evaluation of those risk metrics in the following SECY paper, Relative Risk, SECY-13-0137, Recommendations for Risk-Informing the Reactor Oversight Process for New Reactors.

In that paper, the Staff developed a technical basis for the proposal to use qualitative considerations for characterizing the significance of inspection findings.

The Staff also performed a technical evaluation of the use of relative risk measures for characterizing the significance of inspection findings. In that paper, the Staff recommended against using relative risk metrics.

And I would also note that the ACRS disagreed with the Staff's evaluation at that time, and the Commission did not press the Staff in its SRM to continue efforts in that area.

Also in that paper, the Staff evaluated the appropriateness of the existing performance indicators and the related thresholds for new reactors.

MEMBER REMPE: So, remind me, because I'm

older and forgetting things, but I do recall another example where the Staff and industry worked together with qualitative considerations or factors, and they set a path to use them.

And at the end, industry said, well, we don't want to do something because of a qualitative consideration and things just kind of didn't work out at that point.

And what is the industry interaction with using something that's qualitative?

And what assurances do you have that if you determine something was significant, based on a qualitative consideration, that industry might not do the same thing?

MR. MERZKE: Industry has been opposed to the additional use of qualitative factors in determining significance of anything.

MEMBER REMPE: Right.

MR. MERZKE: So, the Commission has also weighed in on this area and seems opposed to the Staff taking further measures and using qualitative insights in using risk-informing our guidance.

So, we're taking a cue from them and we're not pushing that for the recommended changed to the ROP. However, there's still the Appendix M to IMCO-

609, STP 609, which is the STP which uses qualitative measures, which we'll be discussing later on in this briefing.

That is being reviewed right now for potential revision, but if that revision goes forward, it will be going to the Commission for approval.

So, the Commission will weigh in at that time as to whether or not the Staff is off base with the use of qualitative measures in determining significance.

MEMBER REMPE: Okay, but again, you know industry doesn't like using qualitative considerations but you're still, as a Staff Member, pursuing that, is kind of where I was at.

I was reading this material to prepare for this meeting and --

MR. MERZKE: Well, 609 Appendix M is kind of like a last resort. We use it because there are such large uncertainties in the assumptions made in developing a significance determination that it could go anywhere from green to red.

So, we need to start looking at other factors to zero in on what really is the significance of that issue.

MEMBER REMPE: But when push comes to

shove, industry will probably come back and say, no, it's qualitative. Right?

MR. MERZKE: Not really. We've used Appendix M in several cases.

In most cases, for flooding analyses where several white findings were assigned using Appendix M.

MEMBER REMPE: And they didn't protest it?

MR. MERZKE: And they did not protest those, no.

MEMBER REMPE: Okie-doke, thank you.

MR. MERZKE: Sure.

MR. GIBBS: Dan, may I? Just one comment just for clarity, and thank you for that question. We're going to speak to that more as we get into the Significance Determination Process.

But the Commission, based on their Staff Requirements Memorandum, clearly indicated that they want us, the Staff, to continue using quantitative approaches, to emphasize quantitative approaches, and acknowledge that. And we will do that when it's appropriate.

But the Commission also in the same SRM said to develop a qualitative framework for certain aspects of new reactors.

And they also said to improve in the

clarity on the use of qualitative factors for the current operating fleet.

So, I just want to be clear that the Commission does acknowledge that qualitative framework is needed on occasion, when the quantitative approaches are not, if you will, productive or fruitful in the Significance Determination Process. We'll speak more to that in just a moment.

MEMBER KIRCHNER: Mr. Chairman, can I risk a digression here? Could you elaborate on your second bullet here? Not being part of the ACRS until recently, what are you really saying here about recommending against using relative risk measures?

MR. MERZKE: Currently the Significance Determination Process uses absolute values for the change in core damage frequency to trigger thresholds for significance.

MEMBER KIRCHNER: I'm presuming you didn't qualify what new reactors are, so I'm assuming you're trying to come up with a technology-neutral revision to the ROP?

MR. MERZKE: One of the considerations was if we did go to relative risk metrics, that we would have to apply them to the entire operating fleet, not

just new reactors.

It was going to be a challenge.

MEMBER KIRCHNER: But for new reactors that are by and large -- as it was pointed out earlier, until they're much further along in the design certification stage or almost ready to load fuel, your PRAs are based on paper reactor designs for non-LWR technology.

So, wouldn't you want to look at relative risk metrics there, where you might have absolute values for core disruption frequency or large release that are 10 to the -7 or less.

MR. MERZKE: That's a good question.

MEMBER KIRCHNER: The certainty band on the 10 to the -7 might be several orders of magnitude until the design is mature.

MR. MERZKE: I will say that the ROP provides opportunities for the Staff to exercise additional regulatory action when we think it's necessary in the instance of an ROP deviation.

So, if were to look at an inspection finding that in today's ROP would be classified as green or one of these new reactor designs because, let's just throw out a number there, the delta CDF was calculated at 2E to the -7th, which in today's

environment would be green.

But it might be double the magnitude of what the baseline core damage frequency is for the AP1000. In this case, I think that's pretty close.

Is that significant? Yes, some folks would say that might be significant.

Do we have an option of doing anything in addition to just assigning a green finding, and telling them that they've got to put in their corrective action program, and we're done with it? Yes, we have other options.

Again, like I said, we can use the ROP deviation process to exercise that and say, listen, we think this is really significant, we're going to -- even though the action matrix, and we'll get to the action matrix shortly, defines baseline inspection, because they're in Column 1, we could actually do a 95001 if we thought it was necessary.

Because that's what the deviation process allows us to do. But a lot of that will be a judgment at that point.

CHAIRMAN SKILLMAN: Thank you.

MR. MERZKE: Sure.

CHAIRMAN SKILLMAN: In those cases where you exercised your Appendix M and you were in

qualitative space, did you fire off a 95001 for those white findings for flooding?

MR. MERZKE: Yes, sir, absolutely.

CHAIRMAN SKILLMAN: You did?

MR. MERZKE: Yes.

CHAIRMAN SKILLMAN: And what was the licensees response to that?

MR. MERZKE: They prepared for the United 500 inspections and followed through. As for their root cause evaluation, they knew they had issues, they knew they were wrong.

So, we called them on it and they accepted the consequences.

CHAIRMAN SKILLMAN: Okay, thank you.

MR. MERZKE: Yes, sir?

CHAIRMAN SKILLMAN: Were there any yellow findings out of that? I think there might have been a yellow findings at Monticello?

MR. GIBBS: I believe we did have one yellow finding. Jeff Mitman, are you here? I think we may have but I don't --

(Simultaneous speaking.)

Jeff Mitman, with our Division of Risk Assessment, may have more information.

MR. MITMAN: Yes, this is Jeff Mitman, I'm

a Senior Risk Analyst with the NRR Division of Risk Assessment. There were multiple yellow findings along the way.

Most of them were qualitative, but we always were able to at least get some bounding information about quantification.

The problem tends to be on whether we can come up with a good hazard curve or not. And when you're extending hazard curves out to low frequencies, the uncertainty is very high, there's not a lot of data, there's a lot of skepticism on it.

And that's often why we would go into Appendix M, because of the uncertainty on the hazard curve.

We pretty much understand how the plant responds to a flood, what equipment is lost at what elevations. But the hazard curve becomes problematic, and understanding where the hazard curve is.

And that can drive the risk results directly. So, that's we were over in Appendix M space and there were multiple yellow findings along the way.

CHAIRMAN SKILLMAN: Thank you.

MR. MERZKE: All right, moving on, I'll discuss the SRM-SECY-137.

In that SRM, the Commission disapproved

the Staff's recommendation again to use qualitative measures to supplement quantitative risk evaluations, and directed the Staff to enhance the existing STP for conditions that are not currently modeled in the PRA.

The Commission also noted the overall structure of the existing ROP should be preserved.

And the Commission also directed the Staff to develop appropriate performance indicators and thresholds for new reactors, specifically in Initiating Events and Mitigating System Cornerstones, or to develop additional inspection guidance to address any shortfalls to ensure that the cornerstone objections are adequately met.

And finally, the Commission directed the Staff to explore how the current safety system functional failure of PI would be applied to the passive safety-related components in Generation 3 Plus reactors.

And those are the AP1000 ESPWR-type reactors. So, that was our last direction from the Commission.

In response to that SRM, the Staff is delivering a notation vote paper to SECY for signature at the end of this year, December, 2017, to respond to the Commission direction.

How we got here is we involved internal and external stakeholders, including Nuclear Reactor Regulation, the Office of New Reactors, the Regions, industry, ACRS here, and the public.

We had several public meetings. The Staff maintains existing risk thresholds, consistent with Commission guidance and the SRM-SECY-10121.

We're preserving the existing ROP structure and we'll be providing the Commission an integrated description of the ROP for new reactors, which basically is the white paper which we provided to you all for review.

Moving on, I'd like to give a quick tutorial of the ROP, again, for those folks who are not totally familiar.

The ROP is the NRC's program to inspect, measure, and assess the safety and security performance of commercial nuclear power plants and respond to events in any decline in performance.

It has several objectives. It needs to be risk-informed, objective, predictable, understandable, open and transparent.

And I think we heard mention of the previous oversight program prior to the ROP being in existence, and it probably didn't meet any of those

objectives.

A quick tutorial, the ROP framework is in front of you. It starts with the NRC's safety mission of protecting the public health and safety.

How do we do that? We need to ensure the safety of the public in three specific areas -- reactor safety, strategic performance area, radiation safety and safeguards.

Under those strategic performance areas, we have seven cornerstones of safety we called them. Each cornerstone is there.

The performance of the licensees measured in each of these cornerstones of safety, and they consist of the initiating events.

And the initiating events or the objective of the cornerstones is to limit the frequency of those events that upset plant stability and challenge critical safety functions.

In the event that we have an initiating event, the next cornerstone moves to mitigating systems.

And mitigating systems, the objective is to monitor the availability, reliability, and capability, of systems that mitigate the effects of initiating events to prevent core damage.

In the event of the mitigating systems fail, we move to the Barrier Integrity Cornerstone.

The objective of this cornerstone is to provide reasonable assurance that the physical design barriers protect the public from radionuclide releases caused by accidents.

When all else fails, the final safety net is the emergency preparedness cornerstone, and that, the objective is to ensure licensees are capable of implementing adequate measures to protect health and safety during a radiological emergency. Those cover reactor safety strategic performance area.

The cornerstones for radiation safety involve public radiation safety and occupational radiation safety.

Both of those have the same objective, similar objectives, to protect the public from exposure to radiation. And the occupational radiation safety is to protect the workers.

And under safeguards, we have one cornerstone, security.

And security, the objective is to provide assurance the licensee's security system, material control and accounting programs use a defensive, in-depth approach, and can protect against the design-

basis threat from radiological sabotage from external threats and the loss of radiological materials.

Underpinning all those cornerstones are what we call the cross-cutting areas. There are three -- the human performance, safety-conscious work environment, and problem identification resolution.

Where did those come from? When the ROP was developed, it was kind of determined that plants that had significant performance problems in the past generally exhibited significant performance problems in one of these three areas, or more than one of these three areas.

So, those three cross-cutting areas actually underpin all the cornerstones and can affect performance in each of those cornerstones.

What do the cornerstones mean? Satisfactory licensee performance in each of the cornerstones provides reasonable assurance to the Staff that the facilities are being operated safely and that the NRC's safety mission is being accomplished.

So, how do we measure performance in those cornerstones? We measure them in by two inputs; each cornerstone has baseline inspection results and performance indicators.

Performance indicators, briefly, are voluntary data reported to the Commission on a quarterly basis on various plant parameters, which gives us some reasonable assurance of performance in those areas.

The baseline inspection in those areas complement the performance indicators, and inspections conducted in areas that are measured by those performance indicators.

In those two inputs, inspection and performance indicators, there are significant thresholds associated with those.

Baseline inspections result in findings, findings we need to determine the significance of, and that's what the Significance Determination Process is.

And once we cross the threshold, that feeds our action matrix, and the action matrix determines our regulatory response, which provides the predictable objective and open objectives of the ROP.

Similar to the baseline inspection results, performance indicators do the same thing.

They are preset thresholds for our significance, and if a licensee reports data that is exceeding the threshold, that data will report it as either -- I cover that as green, white, yellow, or

red.

And again, once those significance thresholds are tripped, that input goes into the action matrix, again, to use for our regulatory response.

So, that's how we measure performance in each of those cornerstones.

And critically, the action matrix concept, I spared the Committee Members the actual action matrix because it's very busy and hard to read on a slide. So, I'll throw the concept on here.

MEMBER POWERS: But it is so nifty. The beauty of the action matrix is it's so clear --

MR. MERZKE: It's very prescriptive, it's predictable, and objective.

MEMBER POWERS: Predictable and reliable.

MR. MERZKE: Yes. There are five columns in the action matrix.

Columns 1 through 5, starting on the left, licensee's response, that's when the licensee data inspection findings performance indicators report is green.

That licensee will get baseline inspection program and nothing more, as far as the regulatory response.

When a licensee starts getting findings or performance indicators across thresholds, they'll start moving right in the action matrix.

Column 2, the regulatory response column, might be one or two white findings, or a white input from the PI and a white finding.

Moving to Column 3, the greater performance column, again the significance increases here.

There might be a yellow finding or a yellow performance indicator, or multiple white findings.

As we move further right, Column 4, multiple-repetitive-degraded cornerstone, licensee gets a red finding or PI would fall into this category.

Licensees that move to Column 5 will be determined to be an unacceptable performance. Licensees are not allowed to operate in that column, so they would be ordered to shut down and we'd move them to the IMC 053 process.

As you'll note, as we move to the right in the action matrix columns, it's increasing safety significance, where we're increasing NRC inspection efforts.

As the licensee moves to each of those columns to the right, there's additional supplemental inspections, 95001, which is maybe another 40 hours of additional inspection.

A degraded performance licensee would have a 95002 inspection, up to 200 hours of additional inspection.

And the licensee in Column 4 might see up to 3000 hours of additional inspection, which is a significant team inspection.

So, there's significant inspection effort increase as the licensee moves right in the action matrix.

Also, there's an increasing NRC licensee management involvement in increasing regulatory actions associated with the licensees with degrading performance.

The ROP has four specific program areas. They consist of the assessment program, the performance indicators, the baseline inspection, and the Significance Determination Process.

I'm the Assessment Program Lead so I'll cover assessment, and at this time, the Staff is recommending no changes to the Assessment Program or the action matrix for assessing licensee performance

for new reactor designs.

There's no technical basis to make any changes to the action matrix based on what we know. So, that concludes my portion of the background and ROP tutorial.

Does anybody have any questions before we move on to performance indicators?

MEMBER POWERS: Well, I would just say that Members are not familiar with the action matrix to look at it.

MR. MERZKE: The action matrix can be found in Inspection Manual, Chapter 0305. I don't know if it's on the public website other than being in that Inspection Manual chapter.

But it's very prescriptive on communications, management oversight, and regulatory actions associated with that.

MEMBER POWERS: There's a great deal in it.

MR. MERZKE: There's a lot of information in it. It's very busy.

MEMBER POWERS: It's precisely says how the agency's resources are marshaled when there's degrading performance.

MR. MERZKE: Thank you, sir.

MEMBER BLEY: I have one more at a high level. I'd like to ask it now rather than in the Significance Determination Process talk.

We were talking to some other groups of Staff recently. This isn't really your bailiwick I guess in developing this, but where in the coordinated effort trying to get ready for the first new plant to begin operation does the development of the Staff's PRA models fit?

It seems to me, to be able to carry out what we'll see in the Significance Determination Process, Staff needs a PRA. It's going to be a fair amount of work to develop a new PRA for one of the new plants.

And I don't think anybody's working on that now.

At least that's the impression I got, and maybe we won't get one actually operating but we better have one quite a time before it begins operating for you to be able to carry out your plan.

CHAIRMAN SKILLMAN: I'd like to make a high-level comment too, back on Slide 15, please, Dan.

MEMBER BLEY: Can he answer mine?

CHAIRMAN SKILLMAN: Please, I was reading the body language up here and I thought that had been

absorbed and stopped.

MEMBER BLEY: That's all right.

CHAIRMAN SKILLMAN: I'm sorry.

MR. MERZKE: I guess to answer that, I don't know what work is being done to develop the PRA in the SPAR models.

MEMBER BLEY: But where does this come together? That's what I'm asking.

MR. AYEGBUSI: So, I'm not sure that I have an actual --

MEMBER BLEY: That's all right.

MR. AYEGBUSI: -- I have the direct answer to your question. I mean, what I do know is we do have SPAR models for the design and like you said, we do need SPAR models for the plants, specific to the plants.

And I know Andrew Johnson is here from DCEP and there's an implementation plan for the transition to operations, and what I'm not sure of is if that includes developing the SPAR models for individual sites.

MEMBER BLEY: I'm not sure either because we haven't gotten a hint.

MS. HAYES: I'm Michelle Hayes, I'm an ex-Branch Chief for NROSPRA, and research has a contact

to develop SPAR models for the individual sites.

MEMBER BLEY: They do? Okay. That's good. Some of them didn't know that, though, the last time we talked with them. I won't say who, but I'm glad it's there. I'm sorry, Dick.

CHAIRMAN SKILLMAN: I'm responding to your comment, Dan, that no change to the assessment program. Let me bring your attention back to Slide 11 just for a second.

I remember when this concept became firm guidance from the Regions, and we all learned about the seven cornerstones. And it's been some number of years.

What's changing in my mind is mitigating systems. There are mitigating systems, but there are also mitigating physical phenomenon to which requirements are attached.

So, to your comment, Dan, probably no changes to the assessment program, I'm wondering if m mitigating systems now has two pieces?

One piece is that population of SSCs upon which we depend for protection of the barriers, the fuel boundary, the reactor coolant system pressure boundary, and the containment boundary.

But if now there isn't a new subgroup of

physical phenomenon that are just as important, heat transfer across the containment for an AP1000, passive flow rate for plants where there will be no pumping function for decay heat removal.

In the first case, it's probably BTEs per hour per square foot, or kilowatts per square meter per second, or whatever the metric might be, that needs to be confirmed from time to time to determine or to prove that the design basis physical phenomenon remains capable of providing the function for which it is intended.

Hence, I'm thinking that when we say there's no change to the assessment program, here's my real concern. I want to talk about getting caught up in the ether.

So, when you go to buy a new truck, colors are just right, wheels are just right, engine's just right, all the trim is just right, and it's about $10,000 more than you really want to spend.

And boy, that salesperson is really, really good and you end up with keys for a brand-new truck and you say I don't know how I got drawn into that, I never should have done that.

I feel as though we can get drawn into a fog here with this low CDF and low initial LERF, and

not step back and say wait a minute, even though those numbers are really mighty low, there are other features for which we must be very cautious.

And we must make sure that those features are able to deliver what it is we are intending them to deliver in safety space.

And I think that's a mind change, and that was suggested to me that there might be a change to the assessment program called Passive Features Recognition or Passive Features Accreditation.

Something that kind of says to everybody this is not the same game.

For these new reactors, there's another element of thinking that we've got to be sensitive to.

MR. MERZKE: I recognize exactly what you're saying. I'm not sure what you're implying belongs in assessment space as opposed to inspection space.

CHAIRMAN SKILLMAN: Well, it's a program change from the perspective of we've become so familiar with SSCs, we depend on them in every plant in the country today.

And we're moving in the future to a design where the SSC is no longer that feature that is most important to us.

In fact, it's passive heat transfer, and oh, by the way, a valve on a tank that's sitting up on the top of the containment.

And unless we're thinking about those passive features as a physical phenomenon, we're probably back in the good old structures, systems, and component, space.

MR. MERZKE: What I was going to suggest is that under inspection space, the design basis assurance inspection, which is formerly the component design basis inspection, would probably be the most appropriate mechanism.

The team that goes on site would be looking at exactly what you're talking about.

CHAIRMAN SKILLMAN: But that's fine from your perspective; how about the poor guys, men and women, in the control room where they have that fire?

And someone says, you know, I'm not so sure we ought to keep operating. The fire's out, all of our systems are operating normally. There isn't any real casualty other than we're minus one pump and our tech specs allow us to run with three pumps.

Someone finally says do you think we degraded the inside of containment where we're no longer operable relative to heat removal for ECCS?

I'm suggesting that is a mental change that ought to be somehow embedded in the assessment program. Leadership's understanding of this subtlety of that type of phenomenon, how important it is.

Because that's what's really different.

MEMBER RAY: Can I make a comment?

CHAIRMAN SKILLMAN: Go ahead, Harold.

MEMBER RAY: It seems to me like the pressure regaining function of the containment is a passive function, very similar to what is now going to be relied on in terms of heat transfer.

And that's validated periodically by a pressure test, as we all know.

I think what you're talking about, and I believe I'm agreeing with the Staff's comment, is more are we testing this capability often enough, not are we assessing it in this program that we're talking about today. That's my two cents' worth.

CHAIRMAN SKILLMAN: Thank you, Harold. The difference between what Harold is communicating and what I'm communicating, I understand containment leak rate tests and proof of the pressure function.

I'm suggesting that there can be a subtle change to the physical phenomenon upon which the new design is depending. And its degradation might not be

intuitively obvious.

You've got to be thinking about it and saying, you know what, we might have harmed that and we wouldn't know because we're sitting in this air-conditioned control room.

That's the point I'm trying to make.

MR. MERZKE: And it's a good point and I agree. And again, we're going to have present inspectors on site.

In the event that something like that does happen, well, the licensee's going to have to declare at least an usual event.

And likely, there'll be a reactor shutdown, and most likely, there'll be a special inspection team launched from the Regions, would be a guess just based on your scenario.

CHAIRMAN SKILLMAN: And my response would be if that's codified, then I agree with you.

MR. MERZKE: Okay.

CHAIRMAN SKILLMAN: I agree with you.

MR. MERZKE: That's our SOP at this time.

CHAIRMAN SKILLMAN: And see, what that might suggest is that where before, one might have taken a fire as an unusual event and declared operability and kept on operating, one might now say

fire, unusual event inside containment for an AP1000, that's probably an inspectable event.

And we might have to bring in an inspection team for that.

MR. MERZKE: Correct, I would agree with that.

CHAIRMAN SKILLMAN: And if that's codified, then I'm with you.

MR. MERZKE: Okay, sir.

CHAIRMAN SKILLMAN: Okay? Thank you. Thanks, Harold, thank you.

MR. MERZKE: Okay, I guess I'll turn it over to Ayo now to discuss the PI program. Ayo?

MR. AYEGBUSI: Excuse me. Good afternoon, my name is Ayo Ayegbusi. I am a Risk and Reliability Analyst in the Office of New Reactors, NRO, not NRR.

And so I'm going to be going over the appropriateness of existing PIs and thresholds for new reactors.

Dan covered SECY-1281 earlier. I just want to point out here that SECY did evaluate, again, the systems performance indicator index.

And part of that evaluation was the Staff pointed out that the indicator is risk-informed, meaning some quantitative risk assessment is used as

input into that indicator and used to set the thresholds for that indicator.

That evaluation determined that the MSPIs would be an effective, and determined an appropriate, regulatory response for new reactor designs.

And the reason for that was because it would be very unlikely to cross the green-light threshold, and it would require significant change to the NEI-9902 guidance as it is written.

All right, so for the next bullet, this SECY-1281 did not evaluate all the other PIs.

I should say here that the MSPIs, there are five of them, but in total, we have 17 PIs so the 12 other PIs were not evaluated in this SECY.

The Commission came back and directed the Staff to provide a discussion of the appropriateness of existing performance indicators, in this case, the other 12 PIs and the related thresholds for new reactors.

I should say I'm going to go into a lot of background before we get into what we did for this current paper.

So, going on into the background, in response to the Commission's directive from SECY-1281, the Staff developed SECY-13137.

And in this SECY, we reviewed all the existing PIs and the related thresholds to determine their appropriateness for new reactor designs.

The conclusion was that the PIs would remain applicable to new reactors, with minor adjustments, particularly to the NEI guidance as I mentioned earlier.

We also would like to point out that the plant scrams complication indicator would need to be supplemented with additional guidance, particularly for new reactor designs, to account for passive systems and the reliance on emergency DC power instead of AC power for current plants.

In response to that SECY, the Commission came back and directed the Staff to develop appropriate performance indicators and thresholds for new reactors, or develop additional inspection guidance to address any identified shortfalls.

Continuing on the background, I just wanted to discuss a little bit about the PI program, just to give you more details about how the program works.

So, the PI program provides a sample of objective data to assess reactor facility performance in each cornerstone there that Dan went over.

Along with inspection findings, the PI data serves as inputs to the ROP assessment process, and could potentially lead to additional inspection efforts, known as supplemental inspections.

The performance indicator data is voluntarily collected by the reactor facility and reports it to the NRC on a quarterly basis.

For each indicator, there are objective thresholds that establish a level of regulatory engagements for reactor facility performance in each cornerstone area.

The data that is reported to the NRC is verified by resident inspectors on a frequent basis.

The PI performance bands, as you see here, are similar to those used for inspection findings. And how it typically works is the plant stays in the green band, and when the threshold is crossed, the plant is moved to either the white or the yellow.

I should say that most of the PIs do have thresholds going from green to white, yellow, and red. There are a few that are just green to white. Sorry.

One correction on this slide, it's supposed to say risk-informed PIs and deterministic PIs, not versus deterministic PIs. The reason for that is obviously the whole process is risk-informed.

What we wanted to emphasize here was a situation or cases where quantitative risk assessments is factored into the development of the PI threshold, or is used as an input to determine if the plant performance is across the threshold.

So, many of the PIs are not directly risk-informed in the sense I just described, but they're based on regulations and standards that would apply to new reactor design.

For example, emergency response, the requirements were a number of drills that would apply. And then as Dan mentioned earlier, occupational and public radiation safety would also apply in this case.

The PIs or the performance indicators that are directly related to risk or are risk-informed are the MSPIs and plant scrams. As I mentioned, there are many PIs and their thresholds are more deterministic.

They're based on quantitative risk assessment, assessment of risk, to determine those thresholds and agree to apply them between NRC and industry.

CHAIRMAN SKILLMAN: Would you please explain -- what you said is you introduced the slide, you said there is an error. And I'm not quick enough to understand what it is that you were pointing to,

and I don't want to miss it.

MR. AYEGBUSI: That's fine. So, the title of this slide, which says Risk-Informed PIs Versus Deterministic PIs. We wanted it to say Risk-Informed PIs and Deterministic PIs.

CHAIRMAN SKILLMAN: Thank you very much.

MR. AYEGBUSI: So, the point of this slide is to show you all the performance indicators that we have for each cornerstone.

And the performance indicators that are not marked by an X are the indicators that have, as I described earlier, some risk-informed, quantitative risk-informed, application to it, whether it's in the threshold or the input into the indicator.

MEMBER KIRCHNER: Can you show that slide without the Xs? Thank you. So, maybe this is a good point to ask a question.

Since you're developing a technology-neutral approach, some of the new designs would remove one of the traditional barriers.

Instead of having the three that we conventionally think of, fuel pressure boundary for the coolant and containment, some of the new designs would not have a containment or a confinement in its place.

Some of the really new designs wouldn't have fuel as we know it, it might be in a liquid form.

So, my question is how, if you're doing this in a risk-informed manner, do you change the ROP to adjust to the fact that some of the new designs will be even more dependent on the remaining two barriers than the traditional three that we've seen in the LWR fleet?

Do you change the inspection frequency?

What changes in the application of this when these often passive components now are more important in the defense of the public safety, et cetera, et cetera, radiation safety, and safeguards?

How does this adapt, how would the ROP adapt, to the fact that these new designs may not have the traditional defense in-depth?

MR. MERZKE: Again, we focused our review specifically on the AP1000, but recognizing there are other technologies that are out on the table right now and even on the drawing board stage right now.

We would do the same thing as far as making recommendations to change the ROP for every one of these new technologies. We look at each of these PIs; would they be appropriate PIs for that new technology or not?

In some cases, we determined that, like for the AP1000, the MSPI indicators were not appropriate. So, we'll eliminate them. What do we do when we eliminate a PI?

We replace it with inspection or ensure that the current inspection program, in this case, ensures the objectives of the mitigating systems cornerstone are met either through inspection or through those performance indicators.

So, we will either develop new performance indicators based on what we know about that technology if we can, or we will supplement that to oversight with inspection.

MEMBER KIRCHNER: So, the inspection frequency could be higher, for example --

MR. MERZKE: Frequency and scope.

MEMBER KIRCHNER: -- and scope for a particular system component. Okay, thank you.

MR. AYEGBUSI: Okay, so I'm going to focus on the risk-informed PIs.

The Mitigating System Performance Index measures readiness of systems to perform their safety function by focusing on the unavailability and unreliability of systems and components that meet one of the five systems listed here.

The Commission's safety goals are used to establish the PI threshold for the Mitigating System Performance Index.

For the plant scrams, they measure the rate of scrams per year and provide an indication of core damage frequency.

The data is normalized to 7000 critical hours based on expected capacity fact of plant operation of 80 percent. The impact of plant trips on industry-wide plant risk was used to inform the PI threshold.

So, this is one of the cases where a simplified risk assessment model was run to determine how many plant scrams, how many scrams will cause the plant to -- the change in core damage frequency to trip the threshold.

And that number was used to inform the thresholds for this performance indicator.

MEMBER STETKAR: Before you leave this, I need some help. you characterize the current, in this PI, index as a risk-informed index. And certainly, I'm not an expert on this, I don't do this kind of stuff.

But reading the white papers that you're going to talk about in a second here, there's

discussions of things like front-stops and backstops. And if it doesn't do this then you put this number in, and if it doesn't do that, you put this other number in, and you make what seems to be patchwork decisions, even in the current reactors. That's what I want to get to.

For the current operating fleet, you've got some sort of numerical criteria like, well, we won't assess it as more than 5 times 10 to the -7 because we won't.

So, could you explain just how those are applied in practice? What are those front-stops and backstops, and that jargon that tends to be used?

MR. AYEGBUSI: Okay, so --

MEMBER STETKAR: So, you basically can get a green to white transition when you want it, even though the numbers don't show it?

Or you can prevent the green to white transition, even though the numbers on a current snapshot from the three-year running total might indicate that you might get one?

So, how are you artificially kind of kicking stuff up and artificially suppressing stuff with those other things that you apply?

MR. AYEGBUSI: Are you referring to --

MEMBER STETKAR: I don't know what I'm referring to because I don't actually apply this.

I've never heard about it until I started to read the white papers, and said, well, yes, we get into this situation and we apply this backstop, or we have this front-stop, or we apply this 5 times 10 to the -7.

It's got a name that I can't find in my notes here, risk limit, or something like that.

Which, to me, sounds like a quite arbitrary process but I want to understand it because you're arguing that the whole MSPI framework cannot apply for new reactors.

And I'm not sure how it's applied for current reactors.

MR. MERZKE: I'm not a PI expert.

MEMBER STETKAR: No, I'm not either, and I naively thought that these were objective quantitative measures that were updated periodically.

And I'm finding that there's an element of truth to that, but there's apparently some other pieces that get applied.

MR. MERZKE: These are values that are calculated and reported by licensees --

MEMBER STETKAR: I understand that, and I

also am beginning to understand that there's some sort of really complex algebra --

MR. MERZKE: I think you're right.

MEMBER STETKAR: -- that's more complex than what I naively thought was simply looking at the performance of residual heat removal or something like that on a three-year running average.

And saying, okay, I had one failure in the last year, and therefore, my estimated failure rate on my throughput running is X. And I'm still under the industry average, and therefore, I'm still green.

I had one failure and I'm above X but, gee, nah, I'm going to apply some sort of suppression effect.

MR. GIBBS: Okay, so, if I may, I recall, and I think we owe you something on this --

MEMBER STETKAR: I have no idea, because honestly, I'm being a bit provocative here, but I only in the last week tried to read through this stuff.

And we have so much stuff to read that I can't read everything, that I stumble across these other -- in some places, there's things that are characterized as a front-stop, there are other things that are characterized as a backstop.

And I can't find the term because I'm

searching through my notes here, but there was something like a 5 times 10 to the -7 risk limit or risk something or other, that sounded like it was plugged in, in some cases, so that you couldn't get any more detriment than that amount, if you will, that numerical amount.

MR. GIBBS: And I will admit, my vague recollection, back many years ago when the performance indicators were established, we, indeed, and someone who works in our division of risk assessment may recall, or not, that we did types of sensitivity studies with the information to help us risk-inform these performance indicators.

So, you would take the information from a performance indicator, and do that sensitivity study using a probabilistic risk assessment to check where possibly maybe a backstop might exists. Honestly, that seems like a reasonable approach that I think we may have done.

But I think, Dan, we owe maybe what was done to risk-inform these numbers, and where, what, was the basis of these backstops? I just don't know.

MEMBER STETKAR: The reason I ask is people are using this process today, and however the algebra evolved over time, it's an evolved algebra

that people are using.

And there's some level of comfort of with it, both from the industry and the inspectors, the Staff.

The white papers, both the industry's and the Staff's white paper, regarding MSPI have basically said, well, that concept is not going to work for new reactors for a variety of reasons that I could take issue with individually.

But it sounds to me, and I might be wrong, that there were equal problems with just a naive application of a risk-informed index based on some sort of rolling measure of equipment or system performance as the current process evolved.

And there was some sort of negotiated set of values or backstops or something that got plugged into that algebra.

And if I were more familiar with that and could think of that in the context of some of the arguments that were made why the whole concept of an MSPI can't be applied to new reactors, it might help.

And as I said, I literally was not aware of these other numbers, whatever they are.

MR. AYEGBUSI: Let me try. Like Dan said, this is definitely not my area of focus, but let me

try to give a perspective.

So, looking at the white paper, right, not the MSPI program, looking at the white paper for this activity, there are two aspects to the NEI guidance that we have to consider.

MEMBER STETKAR: Okay, for a second, from my perspective, I don't care what the NEI guidance says. I'll set my mental model with that.

The NEI guidance was developed for current operating reactors, and it kind of works for that.

MR. AYEGBUSI: Correct.

MEMBER STETKAR: And so as far as I can see, my opinion is people can rewrite a report, and train new people on new reactors to think a different way.

So, I don't care what the old NEI guidance says when I'm thinking about new reactors.

MR. AYEGBUSI: I understand that.

MEMBER STETKAR: I understand your perspective but I'm just giving you my perspective on it.

MR. MERZKE: This is my understanding of the MSPI. These are active safety systems, and again, I stress safety systems, high-pressure injection, heat removal, residual heat removal, emergency AC power,

those are the ADGs.

These are all component systems that are testing, regular maintenance. They assume run failures, that's adds to the reliability index.

Unavailability because of maintenance, charging pumps out of service for six weeks because there's being maintenance being done on it, whatever.

All this adds to what we call unavailability and that time is punched into a computer somewhere and it comes out with this number that says, okay, this is what your MSPI is because of a number of failures or whatever.

MEMBER STETKAR: There's some algebra someplace.

MR. MERZKE: It's a black box to me, it's a little computer. None of these are what we consider safety systems anymore.

What these are now is what we call RTNSS, Regulatory Treatment of Non-Safety Systems. These are still important systems but they're not considered safety-related systems.

Now, the safety-related systems are the passive safety systems. Passive safety systems I guess are really what, and I hate to use the word inherently reliable because how do you test those?

MEMBER STETKAR: This is going to devolve into something that I was going to try to stay away from, but I can't.

So, how does the reactor oversight process for, and I'll limit it to the AP1000 rather than trying to get it to any new reactors, assess the grading performance of squib valves? How do you assess that? How are you going to assess that?

MR. MERZKE: I have the very same question.

MEMBER STETKAR: Tell me exactly how you're going to do that. I'm not going to get into safety systems or any of the other philosophical stuff about importance to shutdown risk.

I'm not talking about RTNSS. Squib valves --

MR. MERZKE: And neither are we.

MEMBER STETKAR: -- I'll call them the whole set of I think 12 explosive valves in that plant. How are you going to do that?

MR. MERZKE: You can't test the squib valves because they're one and done.

MEMBER STETKAR: Well, I'm sorry.

MR. MERZKE: If you (Simultaneous

Speaking) them off, you have to replace them.

MEMBER STETKAR: All right, but you must monitor squib valves for standby liquid control systems and boiling water reactors today, and the last time I checked, the last time you could check those, you had to blow them up and they weren't there either.

So, the whole concept of not being able to test a squib valve like a motor-operated valve applies to currently operating reactors.

Everybody's comfortable with standby liquid control systems for boiling water reactors with squib valves?

MR. MERZKE: My understanding with the squib valves is that the detonators that'll be used in those valves are basically produced in lots and the vendor inspectors are pulling samples from those lots to test the explosives.

MEMBER STETKAR: And when you say those valves, you mean the standby liquid control for boiling water reactors?

MR. MERZKE: I'm referring to the squib valves for the AP1000. This is my understanding.

MEMBER STETKAR: I'm trying to challenge you as to conceptually why is a big guy squib valve for something called an AP1000 conceptually different

from a little guy squib valve for a boiling water reactor, in terms of the ability of the licensee and the NRC Staff to monitor its performance?

And have confidence in the fact that they're not degrading, or that a specific plant's treatment of those valves, whether it's testing or maintenance or inspection or whatever, is not degrading?

MR. AYEGBUSI: so, if I can answer that --

MEMBER STETKAR: That doesn't have anything to do with lots of explosive charges or anything like that.

MR. AYEGBUSI: So, if I could answer that question. I've worked with boilers before and I've looked a little bit at the AP1000's requirements and tech specs, right?

And from what I see, I don't see any difference or any significant difference in the requirements for testing, such as checking electrical continuity and things of that nature, online.

And requirements for doing testing, I forget the frequency and tech specs. And then you follow the standard for if you do have a failure to increase a population.

So, I don't see that that process is going

to be any different from what's being done for the screw valves in boiling water reactors.

MEMBER STETKAR: Okay, so if I hear you say that, then why in your white paper do you say, well, there's no way we can develop and index for what I'll call the first bullet there, a high-pressure injection, which you've used the squib valves as a shortcut for, for new reactors because they don't test the valves and the valves are going to be very, very, very reliable?

MR. AYEGBUSI: Well, so the first thing there is we actually developed an index. What we said was that that index would not be a good enough index to determine appropriate plant performance, something of that nature.

So, we did develop the index, but when you look at trying to exercise the index, and I forget which one of the -- there were four, one for black.

And the lowest threshold you would have to cross from green to white would be six failings of a valve, which, that was high, one.

Two, at that point we thought there would be other inspections, other activities going on, right, that we felt that index would not be a good indicator of plant performance.

MEMBER STETKAR: Okay, and that brings me back to the negotiated algebra for the current MSPIs, that if an algebra was negotiated to put in front-stops and backstops, or whatever they're called, for current MSPIs, why couldn't a similar algebra, conceptual algebra, be developed in the new plants?

Such that if you feel uncomfortable that you need six or more failure before you can transition to a green to a white, you develop a different algebra, so that Staff has comfort that we're tracking degrading performance, that we're measuring degrading performance?

MR. AYEGBUSI: So, a couple of things to that.

MEMBER STETKAR: And then I'll come back, if you could give me assurance, the answer to my first question is how will the Reactor Oversight Process for new reactors provide assurance that we do not have a plant-specific degradation of the squib valves?

What elements of the Reactor Oversight Process, given the fact that we're not going to have some sort of numerical index for those valves?

How will the Reactor Oversight Process raise the flag if I'm starting to see increasing numbers of failures of those valves, or increasing

unavailability?

Because I can take a squib valve out of service, I can take firing circuits out of service. It's a very normal operation.

I'm allowed to do that by my tech specs, but there's an unavailability and unreliability conflict.

MR. GIBBS: Right, let me try this.

First of all, the existing Reactor Oversight Process, we intentionally provide overlap between the performance indicators and the inspection program.

That's intentional for what we don't know really, and so that's very important, the first point.

MEMBER STETKAR: Please, if you have a --

MR. GIBBS: No, no, keep --

MEMBER STETKAR: Well, I was going to say, so what I'm hoping I'm going to hear you say is that you're going to explain to me how the inspection program now will provide that surrogate confidence?

Because I don't have an indicator right now.

MR. GIBBS: No, I don't know, Steve may know, but I do not know what we plan to do with the squib valves specifically with respect to inspection.

I don't know if we're making adjustments to our program in that regard. Steve, could you comment on that?

MR. CAMPBELL: We have not looked at squib valves specifically for the inspection program.

The way I understand it's done today is they take a sample from a lot, for the BWR and the standby liquid control system, and do a sample of that to see if they explode.

I know that there's been an information notice back in the '80s on squib valve failures. I haven't looked at that in too much detail but we have not looked at that from an AP1000 perspective.

Now, so if there's some degradation of a squib valve, yet, in the current program, this is the way we would do the squib valves for standby liquid control, if that degradation does not result in that valve being non-functional or inoperable, that issue would screen to green of very low safety significance.

Now, that does not mean that we're done. That means that the licensee is required to correct that problem.

So, right now, in the Manual, Chapter 609, Appendix A, there are a number of screening questions, and they're very powerful screening questions in that

regard. And so there's the specific question about degradation, yet, operable.

If that is the case, it would screen to green in the Significance Determination Process. Of course, we as an agency are growing, a learning agency.

If we begin to see problems, we're going to take the necessary actions, meaning more squib valves are failing than we expected for example.

And so I just wanted to clarify the existing process, make sure you understood how that works and how it is intended to work for new reactors for the light valves.

Jeff, do you have anything?

MR. MITMAN: Yes, I would add that if the problem with the valves rises to the point of a failure, we can identify performance deficiency, and then we can take the performance deficiency, put it in a PRA model, and calculate the impact on the risk.

And from there, we can get a potentially greater-than-green run. I haven't looked at the AP1000 design but I understand that the squib valves are quite important, and so that's why there's a lot of attention on them.

So, we start to see a lot of failures, the

failure rates are going to be higher, and we can modify the -- do an on-the-fly Bayesian update of the failure probabilities, come up with new failure rates and see what the risk significance is.

And if it gets really high risk significance, that's going to drive a plant over in the action matrix, which will draw more attention, more resources, and more oversight on the process.

So, hopefully, the mixing of the inspection process and the oversight portion of it, the STP portion of it, will ensure that if there is a higher failure rate than what we were anticipating, it would get the regulatory scrutiny and oversight.

MR. MILLER: Without trying to add too much fuel to the discussion here, but trying to point something out, as far as what I understand, in the current MSPIs.

As opposed to what we're looking for for new reactors, current MSPIs are things, high-pressure injection, the heat removal, RHR, emergency AC power.

We have tests that we can do with the plant operating, that give us good indication of the performance of those systems.

We also do inspections of those but the PIs are measuring things that are testable.

The squib valves, as you already aptly pointed out and Ayo backed up, slick system and BWR, you don't test those, you don't have a good index for them, but you, during the inspection and the outage, would blow a sample similar to what you do with SRVs.

You don't light off an SRV and see if it relieves at the right set point. You take them off during an outage and you go set them to a bank to test them.

These systems here, in most of the safety systems, not the RTNSS system, but the safety systems, are inside containment, and you don't have access to them.

You don't have inspection access and you don't have tests that you can test, of course, a lot of the safety, or the passive systems.

So, to develop something, a PI that can help you trend before you can get in there at each outage, you really don't have anything that can give you additional information that you can't get by inspecting during the outage inspection.

That's the difficulty of coming up with a risk-informed MSPI for these, is that really, when you're going to get your information is during the outage, and you've got eyes on during an inspection

during the outage to see that.

MEMBER STETKAR: The problem that I have is I hear all of these arguments as excuses for not wanting to do something.

And I use squib valves only because they're kind of a poster child, or they're not tested very often, and yet they can be important contributors to overall risk.

And in fact, somehow the industry and the Staff are dealing with an analog of them in BWRs today and have been dealing with that analog for 30 years or more. And yet, you're saying, well, we can't deal with them for the new plants.

I don't get that. I don't get why you can't. The other systems that are listed here, I read in the white paper arguments it says, well, residual heat removal.

Well, yes, you know, the passive residual heat removal system just has a couple of valves that have to open and we don't test those valves very often.

And the active, the normal residual heat removal, system is only operated during shutdown, so we don't get many operating hours on that. So, how can we develop an index in case one of those pumps

fail?

Well, of course you can. Normal residual heat removal systems in operating plants only operate during shutdown. That's when you gather all of the operating hours and most of the starts and the stops on the bump.

So, there's no difference in the normal residual heat removal system on the AP1000 versus a normal heat residual system. Compound cooling water system and service water system on the AP1000 are going to be operating all the time.

They're not called safety systems but they're going to be operating all the time when that plant is running.

So, you're going to have a whole bunch of operating hours and starts and stops on pumps and opens and closes on valves and those systems. So, they're not like squib valves.

So, saying that, well, the only time they might be important is shutdown, and by the way, we don't have a shutdown risk assessment so we don't know how important it is for overall risk.

I don't get the whole rationale for saying we can't monitor these systems because they're not safety-related and it's a new reactor.

MR. MILLER: Could I ask you a question on that?

MEMBER STETKAR: Sure.

MR. MILLER: This is Chris Miller -- are you saying that RTNSS -- really, is your point more that RTNSS systems could be and should be monitored? Is that what you're saying?

MEMBER STETKAR: My point is that arguments saying that we can't do it simply because it's a new reactor and the core damage frequency is small is not an appropriate argument to me.

And, yes, we can develop meaningful measures of availability and reliability of RTNSS systems because we have data for that.

Now, the unavailability might be higher than currently operating plants because they're not safety-related and tech specs allow you to have it -- there are no tech specs on it basically.

So, the unavailability of those numbers might be different. But just because the numbers are different doesn't mean that we can't monitor that as a measure of performance of the plant.

Are we letting that stuff degrade?

And without the numerical index, I'm looking for what part of the Reactor Oversight Process

is providing us comfort that people aren't allowing the service water and complement cooling water and normal residual heat removal systems to just sit there and degrade, despite the fact that somebody hasn't named it safety-related?

MR. MERZKE: I didn't do the leg work on the white paper but I'm going to throw out a guess here in that they didn't develop a performance indicator for the RTNSS systems because they're not safety-related. However, we will adjust, and it's partly based on the inspection presentation coming up, we will be inspecting RTNSS systems at a minimum, as part of the minimum scope of the inspections.

MEMBER STETKAR: And I'll be quiet now because I've made my point. I'm really interested when you get to the baseline inspection about how you're going to address this.

And I haven't heard anything yet that gives me comfort about what you're going to do about the squib valves.

Because they're not RTNSS, they are safety-related pieces of equipment, so you can't dismiss them as non-safety-related.

MR. GIBBS: So, let's reflect on the

maintenance rule, which the maintenance rule, of course, will apply to new reactors and the maintenance rule specifically talks about non-safety-related systems that are important to safety.

And licensees are required to maintain these systems effectively.

So, that's not the whole answer, I think part of its inspection.

But I think from a regulatory perspective and what we have within the ROP, the maintenance rule is a very important rule that we affirmed some years ago in that regard, I believe in '94 or so.

MR. CAMPBELL: Technical specifications requires them to test their safety systems, or I don't know what they're going to lead in for AP1000, but they also have the onus to put degrading issues into the corrective action program, which is also available for inspectors to review and determine whether they've been addressed.

CHAIRMAN SKILLMAN: I want to just comment that I'm in agreement with John, and this goes back to my comment about the ether getting so comfortable with this notion that we have such low initial CDF and LERF that these, if you will, systems for which MSPI has been, if you will, abandoned, probably shouldn't have

been.

We do have data on these systems that are RTNSS, that are important to safety. They play a minor role, if you will, in terms of how the license is written, but they can play a major role in terms of damage reduction or damage prevention.

So, it seems to me it would not be difficult simply to say even though these systems are not required in terms of the license for the, if you will, safe shutdown of the plant or the prevention of core damage, nevertheless, they present the defense in depth, that is part of almost the culture of this business, and we're going to keep track of this stuff.

And it's easy to do that. I mean, it really is easy. I spent years at the plant; there are individuals who can take that reliability and availability data and pull that number together in a matter of 30 minutes for a quarterly report.

MR. AYEGBUSI: So, going back to your point, the big thing we have to remember about RTNSS is it's the regulatory treatment of non-safety systems. Regulatory treatment is critical to the whole thing. The first part is those, the Commission has required us to have those because of the uncertainty of the passive systems.

For example, you might have insulation on the passive pipe that might degrade and might not give you that thermal degrading that you need. So, these systems were identified.

There's a criteria of, a set of, five items that they have to meet if they're needed to meet the safety goals, respond to I think it's a station blackout, post 72-hour actions. So, RTNSS components were identified using a certain criteria because of that concern.

The other aspect to RTNSS components or SSCs is that the three -- part of the regulatory treatment, three items are required that the licensee has to take, one of which is they have to monitor their availability and reliability of those SSCs, right?

And one of the things we did was require that for passive plant designs, you have to identify RTNSS systems using the process, the set of five items, you have to identify them, and then you have to include them as SSCs that have to be monitored in making sure.

And so the availability and reliability of those SSCs will be monitored, right? There are two other requirements as far as quality assurance. Well,

as far as regulatory treatment. So, one is availability and reliability controls.

The other is quality assurance. For quality assurance, all RTNSS SSCs identified have an augmented quality assigned to them. And there's one other item which I'm trying to remember.

I don't know remember that off the top of my head, but there's one other regulatory treatment requirement that the plants have to meet. So, they've just been thrown to the wayside.

Now to John's comment, the paper. I worked with the folks that developed the paper and our conclusion was not that we can't monitor.

Whether it's RTNSS SSCs, or we can't come up with an index, our conclusion is yes, we can. We can come up with an index for the safety valves, which we did, we evaluated it. We can come up with some kind of MSPI equivalent for RTNSS SSCs.

However, our conclusion by going through that process was that if we did an MSPI-like program for RTNSS SSCs, the more likely it is to never cross the threshold. Right?

Based on the fact that there's a heavy reliance on the passive system, which is driving the lower risk profile of the plant. That's one.

So, that takes care of MSPI. For the valve index, it wasn't that -- the conclusion wasn't based on not having enough testing or operation of those valves.

The conclusion was based on if we wanted to follow the MSPI process, that was one aspect.

If we were to follow the MSPI process, we wouldn't be able to implement the program as is because it looks at a three-year period and you look at what testing and operation of those components have been done.

But if you look at just the valve index alone, all it was was to look at how many failures did you have and how many times did you -- how many demands did you have on those components, right?

And when we looked at that, one thing we had to remember is we've said testing is only going to occur during an outage when you take the valve out, right?

So, if you have a set of 12 valves and you test 20 percent, say 3 valves. If one fails, you have to expand the population, right?

So, any of the testing, if you end up having four or five failures, for example, before you hit a satisfactory number, all that information on

data would be acquired during an outage.

And so what we found by coming up with the index is that to cross from green to white, the least number you would need to cross from green to white for any one of those indexes would be six failures -- well, seven failures of a valve, right?

And our conclusion was that at that stage, there is going to be a significant inspection effort going on that just having that information during an outage window, as opposed to being able to track -- you're not going to have the opportunity to track failures over time.

Because the failures that would have occurred during an outage window would not be sufficient to support tracking plant performance, from a PI perspective. Because it's supposed to track performance over time.

So, that was our conclusion. Now, I think I would agree with the comment from Mr. Stetkar that there is potential to have unavailability, right?

I don't remember what the tech spec says for squib valves, for example, but there is that potential to de-energize a squib valve. I don't know what the tech spec says on that, and that may be something to look at.

However, that is covered in tech specs, for example, but that may be something to look at. But that was where we came out with RTNSS and the white paper.

MEMBER STETKAR: Let me just finish a thought here. There are a lot of concepts in those white papers. There's the concept of you don't want an index that is too numerically volatile that suddenly tosses you way up or throws you way down, just because you start counting one or two more failures.

And I get that, but there are ways to dampen that behavior, numerical ways to dampen that behavior.

The issue that I'm trying to grapple with is not the issue of, well, you have to get to six failures in a given outage before you trip a green to white transition.

Because I'll grant you that if you had six failures in an outage, it's going to get somebody's attention somewhere.

I at my plant in this outage have one failure and I do some more testing and I didn't have any more. The next outage, I have another failure. I've got to do some more testing and I don't have

anymore.

The next outage I got two failures, I've got to do some more testing. That's not what I would expect from squib valve.

That's not consistent with the very low failure rates that have been used for those valves. And how does that type of behavior trip some attention that I might have degraded performance?

And if it's through the maintenance rule, I'm happy to hear that. If it's not through the maintenance rule, then where is that raising inspectors' attention?

MR. AYEGBUSI: So, if you look at the inspection process, I doubled this multiple times as an inspector. The way you would have safety valves, every outage, there's always one or two that fails.

And so the way process works is these are safety-related SSCs and if you look at criterion 16 of Part 50 of Appendix B, you can't have a repeat failure over 60 related -- I forget what we call it now. A SCAQ, a significant condition of adverse quality.

So, what we looked at there was if during -- so, for your example, you test the squib valve, you have a failure of a valve. That failure is going to get inspected.

Why? Because it's a safety-related system, it's an outage. That's the only time you're going to get access to inspect the valve. What are the causes of the failure of that valve that are going to get inspected?

Three years later, there's another test, if that's what the tech spec requires. You have two more failures. The first failure that would have required a root cause evaluation.

The next failures that you have requires another set of root cause evaluations. What I looked at was have you identified a different root cause for this failure?

If you have, then it's just another condition-adverse quality. If you have not, and it's similar to the previous failure, the way the process works is now it becomes an SCAQ.

And that's a whole other process because that would be a violation of that particular requirement.

So, that's a whole other process and that's something that we'll definitely be accounting for in the inspection space.

MEMBER STETKAR: Thanks.

MEMBER SUNSERI: So, I'm going to ask a

question here that there's probably not a good answer for, so I'm maybe not really looking for an answer.

I'm going to preface my remarks with I'm in favor of the Reactor Oversight Program and the benefits it's made to safety of the operating fleets.

So, with that said, I think part of the success of the ROP, as it currently exists, is because we had a good foundation of operational history with the existing fleet.

We understood the risk, we understood how the system failures affected the risk, we knew what kind of inspection results we were getting.

We know how it all tied in and we created a nice, action matrix that allowed the regulator to become more intrusive in situations where more intrusiveness was necessary in order to protect people and the environment when a reactor wasn't working right.

So, what it seems to me is we don't understand the new plants well enough to apply the existing Reactor Oversight Program and the Gap analysis you've done.

And I know that sounds harsh and critical but from listening to the discussion here that we've had, it almost sounds like we're in a situation of

bending to fit and paint the match so we can link up the action matrix with these new plants and make our nice model work so that we can become an intrusive regulator.

And I'm not sure -- I'm not against being an intrusive regulator when it's necessary, but I think in some of these cases, it seems like we're being intrusive for the sake of being intrusive and actually not benefitting, protecting people and the environment on the associated risk.

So, I don't know if that means we need to make more investment in smaller models for specific plants or whatever model it needs to be to help us understand the new plants better, and how these individual failures like John's bringing up is probably a good example.

If we run that through this whole process, does it work? Is it inspectable? Does it give you the right risk numbers? Do we understand what that risk magnitude is?

And does that warrant intrusiveness on the part of the regulator and everybody involved in this process for the protection of the people and the environment, which is our ultimate mission?

In my superficial example here, let's say

that I'm a safety regulator and I get a papercut. Well, a papercut might be 10 to the -7th issue, right? Well, if I get two papercuts and it's twice that, do I require more intrusiveness on the regulation? I say no.

But yet, I heard an example here where if something's 1 times 10 to the 7th, and we double it and 2 times to the -7, then maybe we ought to be more intrusive in that because we double the risk factor, which I'm not sure that warrants the benefit that we're getting.

But anyway, so I've made my point. So, the question is are we putting the energy into the right space?

Should we be learning more about how they're anticipating to operate, and what kind of failure modes and risk levels are there associated with the new design?

So that we can actually see how this is going to work, similar to what we did with the existing fleet.

MR. MERZKE: Just to address your -- you referenced the I guess example I threw out there earlier on when we were discussing relative risk, and how maybe doubling the risk might lead us to maybe

consider additional regulatory action.

I was just throwing numbers out of the air.

MEMBER SUNSERI: And I wasn't picking on you, that was just a good way of trying to illustrate my point.

MR. MERZKE: An example of relative risk.

MEMBER SUNSERI: Right.

MR. MERZKE: And really, that's one of the reasons why this Staff kind of rejected it, is it's very difficult to set, establish a technical basis of what percent is significant.

MEMBER SUNSERI: And don't get me wrong here, I know we need to do something, we need to have something.

We need to have a strong and robust regulatory oversight process for the new reactors.

I just don't know if we're heading in the right direction with this one, because we don't know enough about the output.

MEMBER STETKAR: I'll just throw out further that the folks who put together the construct in Reg Guide 1.174, which is now accepted by everyone as fundamental guidance for the agency, had to struggle with why did they put the grey transitions

where they put the grey transitions?

That wasn't an easy decision, but somebody had to make it. And after a lot of discussion in the agency and discussion with stakeholders, those decisions were made and they were determined to be reasonable and pragmatic.

So, just because something is difficult to do doesn't mean you ought not to try to do it. Because we have experience of people making those decisions in the past.

It took time to do it, though. It's not something that you do in six months to get ready for a plant starting up two years from now.

You need to start thinking about it early on and you need to start thinking about it in the context of not just an AP1000 squib valves.

But if this is going to apply to a broad spectrum of new reactor designs, many of which have passive systems, some of which don't --

MR. GIBBS: So, Dan, I want to address your thoughts there on --

CHAIRMAN SKILLMAN: Please let me interrupt for a second. I would like to ask Odunayo if you could do your next four slides quickly so that we can take a 15-minute break, and then get back to this?

May I ask?

MR. GIBBS: May I just quickly move forward?

CHAIRMAN SKILLMAN: If you can keep that to a minimum.

MR. GIBBS: Right. The first thing I'd like to say is that I believe the NRC is a very flexible regulator. And flexible, what I mean by that, I mean there are things that we don't know, but when we find out information, we do respond.

I'll give you an example of the process that is called Tipper Instruction.

When we uncover information, we formulate that in the way of a new inspection, and then we go out and we get more information to help us better inform what our inspection program may need to be changed to adapt to that new information.

That's the first point. Because I don't know that we know everything and we have to maintain that flexibility.

The second point is I think the Commission acknowledged this whole notion of lack of operational experience.

In fact, the Commission directed the Staff to produce a qualitative framework for situations for

new reactors that have limited operational experience.

And indeed, we intend to do that. I just want to make those two points to address your specific concern about the direction we're heading with new reactors.

Thank you.

CHAIRMAN SKILLMAN: Okay, thanks. Ayo, go ahead, please?

MR. AYEGBUSI: Okay. All right, so what covered previously to this slide was all background information of what got us to the recommendations we were making in the next SECY paper.

And the first thing I want to address is there were two industry white papers. They evaluated all performance indicators for applicability to AP1000.

The one white paper covered all PIs for the MSPIs. And then the other white paper covered the MSPIs. And the conclusion of both white papers was that all the PIs except MSPI are applicable with limited changes to the PI program.

And it was determined that the MSPI would be an adequate tool to measure performance for the AP1000 design. And again, that was due to a lack of data and limited testing and cycling of SSCs.

In addition to that, the NRC did come up with a white paper that arrived at a similar conclusion to the industry white papers, but went a step further to evaluate if there was the potential of -- a potential indicator or index that could give us insight on plant performance over time.

And one of the things we looked at was safety valve and reliability index.

And what that really was was identifying all the safety valves and figuring out which valves had some active component to them, and then binning them as far as whether it's an air-operated, motor-operated screw-valve, or solenoid-operated.

And then evaluating if each index would be of any value to marginal performance again over plant operation.

MEMBER STETKAR: Ayo, just for the purposes of the record and clarity, when you say safety valve on reliability index, you're not talking about a spring-loaded safety valve?

You're talking about unreliability index for safety-related valves, because you said motor-operated valves, solenoid-operated valves, air-operated valves, squib valves?

Is that correct?

MR. AYEGBUSI: That is correct.

MEMBER STETKAR: Okay, thank you.

MR. AYEGBUSI: That's correct. All right, so this index, after looking at it, it was proven to be impractical due to limited testing of the valves during operation.

Again, we currently don't have any industry data and we would need -- if we wanted to follow the MSPI program, but in general -- we would need three years to collect data, which is one of the guidance in the NEI document.

And then, it has little sensitivity to failures, as we talked about earlier.

You would need, at the very minimum, six failures to cross the green-to-white threshold, and I believe for one of the indexes, you could have an unlimited number of failures without crossing the green-to-white threshold.

The other aspect is we identified 45 valves, 32 of which we identified had an active component. And compared to an MSPI index or any type of index, that's a limited set of components to monitor.

And we felt that you were looking at very few risk-significant components and the index around

that would also not provide the type of performance feedback that we're looking for.

This slide is pretty much what I've talked about before. I just wanted to point out on plant scrams that the thresholds there were conservatively set for the existing fleet.

And so after we looked at it, we think that the existing thresholds of performance bound the lower risk of new reactors.

In conclusion, the SECY paper that will be going up to the Commission, overall, the Staff is recommending a relatively modest adjustment to the program areas of performance indicators.

In the case of plant scrams, it will be applied as is with the same thresholds -- we recommended it's applied as is with the same thresholds.

In the case of plant scrams with complications, we also recommend to apply as is but as we discussed earlier, tweaking the NEI guidance or adjusting NEI guidance to account for passive systems.

And the difference is what a complication would look like for a new reactor versus a current fleet.

And then we're recommending not applying

the MSPI index to the AP1000 design, and along with that, we did not identify any new PIs that we felt would fit under the mitigating system cornerstone, although, we determined it to be needed.

But going back to some of the conversation we've had, we're definitely going to be looking for opportunities once we have some operating experience.

And on the current plants, the new reactors are required to have a PRA, which, part of that would be collecting data on all SSCs' failure data, data on availability, data on reliability.

We can then use that to inform the process over time. And that's all I have.

CHAIRMAN SKILLMAN: Okay, questions? Any questions before we take a 15-minute recess, please?

MEMBER KIRCHNER: I had one. It would seem to me that with regard to the MSPIs, the five systems that you identified for the existing fleet are, in one fashion or other, active systems.

But if you step back from how active they are to function, it would seem to me that even the new reactors in one form or other will require these functions. And that would provide a basis for defining performance indicators.

You are going to have to have -- as John

gave an example, even the passive plants when shut down will have some residual heat removal trying to function while they're doing refueling, et cetera.

So, there's a function that is almost traceable back to the GDCs for each of those five systems that will still, in one form or other, have to be addressed in these new and advanced plants. So, that becomes an open-ended question.

Well, that would be my approach to the MSPIs for new plants to look at it from a function standpoint.

And because you're at a disadvantage also; a lot of these are paper designs so there isn't any detail as to many of the supporting systems and plant layouts that will come only much further on, and with the PRA, that will indicate which, or if, any of these systems are important to safety.

MR. AYEGBUSI: If I may respond to that? We did evaluate exactly what you just mentioned, which is looking at the function, right?

So, in the white paper, we looked at the current function, we looked at what those functions are, or would be for the AP1000 design.

And some of them would have safety systems, and some of them would have non-safety

systems.

But like we've discussed, when you look at the non-safety systems specifically, there's really no feasibly situation where you would actually cross from green to white.

So, then that index wouldn't be as meaningful as what we have now. So, we did evaluate that.

We evaluated a couple of instances or options given the fact that the design is different, even though the functions are the same.

MEMBER KIRCHNER: -- appropriate to pick on one design, but just to make it tactile, with AP1000, you'll still require DC power, right?

MR. AYEGBUSI: That is correct. That is the only safety-related power.

MEMBER KIRCHNER: So, therefore, I mean, that's the surrogate for emergency power, and I would hope that would be on your inspection program and your monitoring program, right?

MR. AYEGBUSI: Yes, so the inspection program, the inspection program takes a list of all the risk-significant SSCs. And so all those things always -- inspection items are picked from that list.

So, the DC power, the safety-related DC

power, will be inspected. There's no two ways about it. Excuse me.

But then even the monitoring of the DC power, specifically for AP1000, is actually risk-significant.

And so it will be inspected because it's part of that list, and that's the list from which to do inspections.

MR. MERZKE: Most of that list really comprises the RTNSS systems. They will be inspected.

We're adding those to the scope of the inspection procedures as minimum sample sizes for our baseline inspections.

MR. CAUFFMAN: Hi, my name's Chris Cauffman, and I just wanted to add just one thing here so that we don't lose sight of it.

When the ROP was formed, they came up with PIs, and the idea is the PIs could be used to assess these areas under the cornerstones of safety, which was communicated by Dan earlier here.

And then what couldn't be assessed by PIs was to be examined by inspection, okay?

So, in this case, there may be a PI, an area of PIs, where we're not maybe doing inspections here, and people seem to be very concerned that we may

be losing something in this area.

But inspection can be used to supplement areas where we're not doing PIs.

Typically, when we have a PI, it's not valid when we start operating earlier. We do additional inspection to cover the absence of the PI to try to get assurance in those areas where we don't have the PI.

With the AP1000, we think that it will probably less sample opportunities. There's more passive system reliance and things like that we can look at.

There won't be as many, for example, surveillance tests and things like that being done continuously, which inspectors were looking at with these more active-type components.

So, we sort of think we might have even some oversampling for the opportunities or have a robust sampling, just using the existing baseline program that we have right now, until we can learn more.

So, personally, I wasn't too concerned with the lack of the PI, you know, the PI not being there, because I think that we can cover that inspection space.

That's all.

CHAIRMAN SKILLMAN: Thank you. With that, I would like to take a 15-minute recess. Let's reconvene at 25 minutes after the hour on that clock.

(Whereupon, the above-entitled matter went off the record at 3:10 p.m. and resumed at 3:25 p.m.)

CHAIRMAN SKILLMAN: We're back in session. Harold, go ahead.

MEMBER RAY: Thank you. This is Harold Ray. Several times along the way here, I thought, as one who went through the design certification process for AP1000, that we were revisiting things that were discussed then and seemed to have been resolved.

Now, that's not entirely the case. I understand the difference between reactor oversight and design certification.

But I just wanted to introduce a question, that when we talk about things, as we have, and I think they're important things to talk about, that we are mindful of what was discussed during the certification process about stuff like squib valve testing and so on.

And I know John's talking about how these things are modeled and represented in the risk analysis, which is not the same thing as I'm talking

about, but nevertheless, just always wanting to remember that we're talking about a certified design where some of the issues we're talking about and oversight, may have been discussed ad nauseum during the certification.

And you want to look back and keep that in mind so that we're not doing something that was discussed previously. That's all I wanted to say at this point.

CHAIRMAN SKILLMAN: Harold, thank you. Thank you for that. Steve, you're up.

MR. CAMPBELL: My name's Steve Campbell. I'm in the Reactor Inspection branch under Division of Inspection and Regional Support NNR. I was responsible for reviewing baseline inspection procedures and how they apply to AP1000.

So, a little bit of background, something we've talked about quite a bit is SECY-0137, and specifically, in my area, it has to do with developing additional inspection guidance to address identified PI shortfalls to ensure that all cornerstone objectives are adequately met.

We have done that, and as I go through the, go through the presentation, you'll see where that guidance has been implemented as through samples

and also through resource allocations.

MEMBER BLEY: Bas that been worked into the inspection manual already, or is that --

MR. CAMPBELL: No.

MEMBER BLEY: -- you have drafts that'll go in?

MR. CAMPBELL: We have drafts, yes. What we foresee is taking the existing inspection procedures and making changes to those to address certain areas to add samples or add --

MEMBER BLEY: Okay.

MR. CAMPBELL: -- more resources to it. Background, also, we've also talked about MSPI quite a bit.

And just a reminder, it's to monitor the availability, reliability of safety systems necessary to mitigate accidents.

We also touched a little bit about RTNSS, and they are non-safety-related structure systems and components that perform risk-significant functions, and therefore, candidates for regulatory oversight.

And we also talked about the last bullet, which is the industry white paper that provided aspects of the ROP for the AP1000 design.

And I seem to remember reading that, and

that said, basically, that the use of MSPI right now, without having operational experience, isn't a good indicator until we get that operational experience.

MEMBER STETKAR: The, I'll just ask the rhetorical question is, how do you know when you have enough operational experience that you turn it on?

The Germans, for example, claim that they never have enough data. They didn't develop risk assessments for years and years and years because they claimed they never had enough data.

MR. CAMPBELL: We have a --

MEMBER STETKAR: So, how do you know you have enough operating experience and you can --

MR. CAMPBELL: Because there was also --

MEMBER STETKAR: -- turn something on?

MR. CAMPBELL: -- Davis-Besse lessons learned program. We have an operational experience branch in DERs.

They look at events throughout the, internationally and also nationally. And anything that's, in my opinion, more than twice, is a trend.

And then there would be some discussion and some decision on how we would handle that potential trend that's starting to occur. Either issue some type of generic communication, like

information notice or policy.

MEMBER STETKAR: That's more a reactive, once something goes bad, I react to it, perspective. Having four or five years of experience with no failures is also operating experience.

MR. CAMPBELL: Correct.

MEMBER STETKAR: It gives you confidence that things aren't getting worse. So, just, it was a rhetorical question because you often, in the white paper uses this notion of, well, once we get enough operating experience, we'll go revisit this.

MR. CAMPBELL: Right. I was, I duly note what you're saying and this is something that --

MEMBER STETKAR: Yes, don't wait for the bad day before you said, well, you know, two years ago, we should've started looking at this thing.

MR. CAMPBELL: I was paraphrasing what NEI, the, NEI had come up with, in industry, in terms of how we handle MSPI.

MR. MILLER: I will just point out that, you know, it's not the first time we'll bump into that.

You know, when you come up out of a, you know, a new startup like, you know, what's part two, or when you, when you have a plant that's been shut

down in the 0350 process for a long time, you, we've had those same discussions.

And you know, there's some that would say that trend is a year. Some would say three. I don't think anybody advocates for, you know, your German model of it.

(Laughter)

MR. MILLER: But typically, we're looking at, you know, three years to get enough data to say it's significant. But there's probably, there's probably work and discussions that could happen on that.

MR. CAMPBELL: Okay, next slide. Part of evaluating for AP1000 design only, we looked at, it was me and another person for New Reactor, Office of New Reactors and I think somebody from Region 2 construction branch.

We reviewed 20 baseline inspection procedures for changes to accommodate the AP1000 design.

Results of that review, we found that review, we found that few changes are required. We probably, we will probably make adjustments to sample sizes and resource estimates.

And we would also include an inspection of

RTNSS in the sample. And the sample ranges are based on, that there are fewer components and there's a lower baseline risk for the AP1000.

And also, the range is going to include, we're going to propose risk-important high and intermediate systems to inspect as well.

I'm not saying that's a definite, but that's something we're proposing from the program, from the program branch.

MEMBER STETKAR: I'll also come back to my former statement where you convinced me, or tried to convince me, that you don't only look at delta CDF. Everything I read focused on CDF, and as does this slide.

MR. CAMPBELL: Okay.

MEMBER STETKAR: Our experience, from even operating plants, is some stuff is not very important to core damage, but can be really important to public health risk.

And that might be even more emphasized in some of the new reactor designs. So, when you start thinking --

MR. CAMPBELL: You don't agree with the delta CDF in terms of --

MEMBER STETKAR: Delta CDF, there is not a

linear relationship between core damage frequency and large early release frequency, and certainly not when you look at importance of systems and equipment, that there isn't, some stuff is very important to large early release frequency.

Just think of steam generator 2 rupture. It's really, really important to offsite releases. It's not very important to core damage frequency at all.

And so, if you only focus your attention on core damage, you might be missing stuff. And in the stuff that you miss, on a relative, I hate to use that term. Bad use of terminology.

On a, it might be more important on the new plant designs than it is on current, on the current operating fleet because of the integrated notions of maintaining passive core heat removal and containment heat removal using the same stuff. So, just, I'll just throw that out, is --

MR. CAMPBELL: Right.

MEMBER STETKAR: -- think about --

MR. CAMPBELL: And we had to have a starting point, and it would be based on the safety verification matrix that Office of New Reactors had come up with.

And I think it is also in the inspection manual, Chapter 2519. That list of systems and what their importance are. And we use that as what we need to do for samples. Also, the sample range --

MR. MERZKE: Can I, can I just toss out a, I just want to throw out a quick comment here. You know, when, our inspection procedures are, offer a lot of flexibility to the inspectors out there.

The resident inspectors, they get to choose the samples that they select. And a lot of it's risk informed, will go out there and, okay, what's the most risk-significant system that I haven't inspected in a long time?

Or what does the, what's the licensee planning on doing here in the near future that could get everybody into trouble? Maybe I should take a look at that before it happens.

With, when we're referencing here, the risk, the high and intermediate risk-important systems, those things just can be referenced in the samples, recommended samples as, okay, you should at least be taking a hard look at these and the RTNSS systems because they're very important.

But that, it's not going to inhibit an inspector from going out there and choosing samples

that are more appropriate.

MEMBER STETKAR: It isn't, and I'm not implying that it does. What I am trying to challenge is a mindset that says, core damage, core damage, core damage.

MR. MERZKE: I understand that. I just --

MEMBER STETKAR: So, that, when I'm, when I as an inspector am trying to make that decision of where am I going to focus my resources, I might want to think more broadly.

MR. MERZKE: Yes, I understand that. And I think our resident inspectors do that.

MR. CAMPBELL: That's really interesting, because my division director and I went out to Vogtle and interviewed the resident there who was at the Chinese reactor, and we also learned that if they don't close a certain door, that it would flood down a sub-compartment B, and within, you know, a cooling for the core. So, this door has to be closed.

You know, so those are the things we're learning, and we're going through the training courses to pick up what we need to do to incorporate these procedures. And as the Chinese reactor is getting more operating experience, we'll learn from them.

(Laughter)

MR. CAMPBELL: Do you like that? Okay. So, also, our sample range is based on RTNSS, which is what I'd mentioned before. And also, the accessibility of components, whether they're inside and outside containment.

In the AP1000, the passive significant components are inside containments, so we'd need to have probably a little bit more inspection focus when that door gets opened up and we have opportunities to look at those components.

We looked at the comment on the MSPI, and we looked at the breadth of the baseline inspections and assessed the availability, reliability, and capability of mitigating systems that meets the MSPI purpose.

I did a quick review of all of our procedures. It's about 60, 70 percent of these procedures cover that cornerstone.

About six of them do not, and those are like fire protection procedures, and service inspection activities, and operator re-qualification.

So, the passive systems are expected to be reliable. Compared with the AP1000 baseline, lower baseline risk, we feel that the inspections we have right now are more than adequate to compensate for

MSPI omissions.

And we determined that additional inspections are not needed to compensate for not using MSPI indicators for the AP1000 design.

I talked about outages. We assessed the staffing during outages. Again, safety-related systems, high and intermediate risk-important systems will be identified, as well as RTNSS, that are in containment.

This will require an additional staffing resources be applied there. We would foresee a team being augmented with an additional inspector to assist with conducting equipment walk-downs, surveillance testing, post-maintenance testing, and containment closeout.

The other team members would be focusing on outage-related engineering programs. So, in conclusion, we conducted the gap analysis.

We found that there were fewer changes to the inspection requirements, and added guidance as necessary.

We anticipate adjustments to sample sizes, risk importance, and RTNSS, to be specific. We determined that MSPI not applicable to AP1000 design.

AP1000 design has a lower baseline risk,

and IPs, as written, are sufficient to assess licensee performance in mitigating system cornerstone without MSPI.

Inspect for resources, we are proposing that an outage inspection team be dispatched from the region to cover outage-related operational and program inspections. And that's it for the inspection part.

CHAIRMAN SKILLMAN: Steve, I've got to ask this. You make the, make the argument that the baseline risk is so low and the RTNSS systems will be inspected.

But by and large, because the risk is so low, you really don't need the MSPI. What happens if you have your RHR system fail repeatedly during an outage on a fully constructed AP1000 that's run a full fuel cycle, so it's got a full burden of decay heat?

MR. CAMPBELL: We'd handle that, sorry, we'd handle that like we do a conventional reactor. We would either listen to it and find out about it through interviews or attending plan of the day meetings, or outage meetings, review condition reports, follow up on how the licensee corrected that condition per their corrective action program.

There is a regulatory requirement and criteria in 16 to do that. They would assess that for

performance deficiency, whether that performance deficiency was more than minor.

And if it is more than minor, it would be considered a finding, which then would have to go through the SDP. That's how we would handle it, just like we handle conventional reactors.

CHAIRMAN SKILLMAN: If that plant were a conventional plant, the consequence of a repeated failure of a safety system can be a yellow finding, or in some cases, a red finding.

And so, in the shutdown condition, we can have the identical condition, or the, almost the identical circumstances in terms of risk that we would have, we could have in an, in an AP1000 that we would have in a, in a large four loop Westinghouse somewhere else.

So, I'm struggling with this notion, therefore, that MSPI really doesn't have to apply. It seems like it should.

MR. MITMAN: If I could add to this, Jeff Mitman again, with the existing fleet, there are no PIs for shutdown.

We, when we set up the ROP, we looked at creating PIs for shutdown, and after a long discussion, they pondered it, and they didn't create

PIs.

And they relied upon the inspection program and the SDP process to monitor risk and plant performance during shutdown.

So, in that aspect, there's no difference between new reactors or the AP1000 and the existing fleet.

But you're absolutely right. If there's, if there's repeated failures of a, of a monitoring RHR system during shutdown with the new reactors, there's going to, there is an SDP with the existing fleet.

We will modify that and there will be discussion of that later this afternoon about what we're going to do that.

And we'll monitor any proposed deficiencies, and you can them graded a green finding on the existing fleet for, or on the, on the AP1000 for problems of an RHR system.

So, for shutdown, I think we've got it covered. There aren't any differences conceptually in the process for the, for the AP1000 versus the existing fleet.

MR. AYEGBUSI: Can I, if I, if I may, can address --

CHAIRMAN SKILLMAN: Please, yes.

MR. AYEGBUSI: -- your comment? So, I think we need to really be clear about MSPI and how inspection factors in.

The aspect of, the way the ROP is set up is, if you don't have PIs, right, then you should develop inspection procedures, inspections to cover, to cover the areas in each cornerstone. Right?

So, what Steve is saying is we're looking at the inspection process or the inspection program and saying, now that we've determined that MSPI would not be of any, would not be of value for monitoring plant performance, are we adequately covering the mitigating system cornerstone in the inspection program?

So, it's not to say that the MSPI would not be valuable. It's saying, we recognize that it wouldn't give us value in a performance indicative space. I hope that clarifies things.

CHAIRMAN SKILLMAN: I understand the words, but I almost feel as though this is a walnut and pea game and we're just moving the, we're just moving the pea around independent of where the walnuts are.

When you, it's almost a word game as opposed to a focus on the function that needs to be

performed, and monitoring that function to ensure that it is provided when it's supposed to be provided.

MR. CAMPBELL: There's really nothing different we can do. If you look, take the squib valve for example.

What can an inspection do other than follow up on a failure that occurred on the squib valve because they can't light those off? They can't do that.

So, if they find in a lot that one didn't work and they have repetitive failures, the inspectors would go in and review what the licensee did and whether their corrective actions were adequate.

If they weren't, then there would potentially be a performance deficiency and possibly a finding, and then that would be evaluated for a color significance. There is nothing else we can do with the squib valve. I mean, MSPI --

CHAIRMAN SKILLMAN: I would just respond, Steve, that that situation, for a squib valve that remains idle for long time periods and is called upon infrequently, and either succeeds or doesn't succeed when it's challenged, is a very different situation than having an RHR system that is known to be required when the reactor is shut down.

MR. AYEGBUSI: So, for the AP1000, right, the way it's going to be operated is the non-safety systems, which you do have a non-safety RHR system, right, will be used for operation. Right? So, that's one aspect.

There is also a passive RHR system, right, which it seems they're related. Right? So, they're two, they're two different systems, but the approach is to use the non-safety systems when at all possible, without challenging the safety system unless required.

So, I don't know if you, so that, but the passive RHR system is not an active system. It's just a bunch of squib valves and MOVs that have to be opened to get flow into the core and back into --

(Simultaneous speaking)

CHAIRMAN SKILLMAN: I'm thinking of shutdown cooling is what I'm, what I'm thinking about.

MR. AYEGBUSI: Yes. So, in that case, it would be the non-safety RHR systems.

MEMBER STETKAR: Okay. Let me, let me try something here. Two slides ago, Slide 31, says that staff determined that additional inspections are not needed to compensate for not using MSPI indicators.

And as I said earlier, I can, you know, I get that if I can gain some confidence from you that,

indeed, I understand how the inspection program for the AP1000 will more appropriately compensate for that.

Let me flip it around. Why the heck do we have the MSPI for the current operating reactors if the scope of the current inspection program is adequate to catch everything that I need to catch?

In other words, what benefit am I getting today from the MSPI for currently operating reactors? I must be saving inspection resources, or I must be getting some early warning, or I must, there must be some benefit from it because it's there.

You're, as I understand this, you're saying, I don't need the MSPI for whatever reason. I have all kinds of rationale.

But I can't do it, I don't want to do it, I, you know, can't develop the numerical metrics. Fine. I can live with that.

Now, you're saying that you don't need to enhance the inspection program to compensate for that because the current inspection program is adequate.

So, well, if that's the case, why the heck do I need the MSPI for current operating reactors, because the current inspection program must be adequate for that without the MSPI for current

operating reactors. So, what do I buy?

MR. MERZKE: Let me take a stab at this one. When we say no additional inspection, that means not creating any new additional inspection procedures to monitor mitigating systems.

What we, the inspection procedure we have, currently, you know, addressed the availability, reliability of those safety systems.

What we need to do and will do is adjust the guidance in those inspection procedures to ensure that we follow up and monitor those systems that are important, like the residual heat removal systems, and the diesel generators, and things that we're not going to, they're not considered safety systems anymore.

But we're going to adjust the guidance to make sure that the mitigating systems cornerstone objectives continue to be met with that additional guidance and --

MEMBER STETKAR: Let me try this, because we might be talking about semantics here. When you say that no additional inspections are needed, do you mean no more inspector person hours or do you mean no more writing an additional chapter of the inspection manual?

MR. CAMPBELL: I'd say yes to, I'd say yes

to both. No more additional, well, additional inspection resources, what I had talked about during the outage.

If these plants are a lot, we look at them as being much more safer than the conventional reactors.

MEMBER STETKAR: I look at them as being absolutely as safer as the conventional reactors, and I have no idea whether they're safer or less safe during shutdown because, for example, I haven't seen a shutdown risk assessment on them.

As best as I can tell, they're no more or less safe during shutdown than a conventional plant. And it's not clear to me whether their risk is substantially lower during power operation, because I haven't seen anybody do an external hazards analysis of them. So, much safer is perhaps your opinion. It's not mine. And certainly not during shutdown.

MR. MILLER: So, if I could help a little bit with a couple of the points. One is, to get back to your question --

MEMBER STETKAR: But that, I'm not quibbling over shutdown versus power operation. I'm quibbling over the process.

There's something here and you just said,

no more inspection resources measured by inspection person hours will be needed to compensate for the fact that I don't have these indices because you've already said, well, if I've got an index, I can use that as a surrogate for inspections.

I thought that that's what I understood. And I don't have that index, so I'm going to rely on the inspection program to, the inspection process as a surrogate, for the lack of that index.

And as I said, I can, I can live with that. I can get it. I don't know how you're going to do it.

MEMBER BLEY: Chris has been --

MEMBER STETKAR: Yes.

MEMBER BLEY: -- add to this for a while.

MR. MILLER: So, the, yes, I think I, perhaps the way I look at how it should be characterized, we took a good look at how you should inspect these plants differently.

The first aspect is, you know, they're built differently, and the most important systems are inside containment.

And so, how the heck do I get, and the containment is tight and hard to get in, you know, it's going to be hard to get around and whatever. So,

you have, so, you have some challenges there.

So, but we do, we do have our, and we may not have a new procedure number, necessarily, but we have changed, you know, the samples and the types of things that need to get inspected at different times.

We realize that, really, I mean, you know, the traditional walk-downs that I, as an inspector, both in ROP and pre-ROP, you know, would do. You don't have those opportunities now.

We don't think that the PIs are going to give us the same valuable information in the ROP. So, we have to focus our inspections more on the times when we do have that availability. That's during shutdown.

And so, we, those procedures are being adjusted. Those samples are being adjusted. I don't think we've come to a, you know, a final point on the number of hours.

Steve was giving you a speculation of where we might end up because there's, what, 40 percent less active components that you'd, that you'd look at and walk down.

But the passive systems, we know that those are very important, and we know that that inspection during shutdown, the last person out of

that containment is going to check some very specific things.

Steve mentioned one of them that he and I saw when we were down there. It was like, if you don't get that one right, then, you know, you might as well give up on some of your core cooling.

So, there is very specific changes. I don't want you to get, walk away with the idea that we're not making any changes to the inspection procedures. They are.

Now, as to value of the, you know, of the PI, why did we, why did we put the PIs in in the first place? And I did inspections before we had PIs in pre-ROP. Right?

You know, we found that, you know, if we could give, get somebody to point us to the right systems to put more emphasis on, and that's what the PIs really do.

They give more emphasis on the really high risk-significant systems, and they give you more emphasis such that, if you trip, then you, then you do more inspection, 40 hours more, you know, or whatever, 120 hours more, depending on if it's a 95001, 95002, or 95003.

So, it points you to it, but if you don't

have an indicator that trips, it'll never point you to it.

Or if you don't have an indicator that trips until, you know, it takes six times or eight times, and you're not going to see that kind of thing.

It doesn't give you a valid method to point you to another inspection. And if you can't do the inspection anyway because, I mean, if you're only going to be doing the inspection during an outage, then it, the PI really doesn't help you.

If that's, you know, that's when you're going to get the PI information, and that's when you're going to be getting the inspection, it really doesn't assist you that much if you're going to be looking at that thing, those kinds of things during an outage.

So, those are the things you have to weigh when you're balancing PIs to inspection. And I think we've, you know, even pre-ROP, we did a really good job of inspecting and finding areas that, you know, that we felt were risk-significant, and had ways of, and now we have better ways of assessing the risk.

So, we'll still have those capabilities with inspection. It's just that we won't have some of the PIs that have been handy in the past, because they

don't have as much meaning as they had before.

So, that would be my take on, we have, we have changed, we will change the inspections. We just may not change the inspection numbers.

We'll change the sampling, we'll change the focus, we'll change the outage focus clearly on that type of inspection.

MR. MERZKE: Maybe I can offer up an example. Currently, inspection procedure 71-111.04, equipment alignment. Okay.

The purpose of the MSPI is to ensure, well, availability and reliability of those safety systems. Now, how do we do the same thing for passive safety systems?

Well, if those systems aren't aligned when you close out containment, well, they're not going to be available for an entire cycle, and that's a bad thing.

So, we can use that same inspection procedure that's already written, which requires two full walk-downs of, system walk-downs during a, during a year, and usually during that containment walk-down, and it doesn't require any extra inspection resources to do that.

It's just, that's just going to be our

sample for that particular effort. So, that's just one example of how we'll use the current inspection procedures to, you know, for that particular thing. And I'll measure availability of passive safety systems.

MR. CAMPBELL: And also, at power of staffing, with the current fleet, the components are outside containment, so they're accessible to inspectors.

So, you don't really need to have the amount of inspection, like resident inspectors onsite, because those, that equipment isn't accessible on an AP1000.

Whereas, on a conventional reactor, they are accessible and you need to have more inspectors look at them. So, there's a difference between what we have.

We have the kind of leverage, what we have now for inspection procedures, and whether those are adequate now for the current operating fleet.

And then, you know, try and use that in considering where the equipment's located, outside containment or inside containment, and make your decisions based on that.

So, the resources, we don't see as, like,

the same number of residents being stationed at an AP1000.

It's only when an outage occurs and they can get into those areas where there's a higher significant of passive components that we need to go look at.

MR. MILLER: Yes. Let me, let me just give a comment on that. And that, so, we're still, we're still working that piece too as far as staffing, and we're working Region 2, and we're working with NRO, and we're working among ourselves for what the staffing needs are.

I think we left it in the paper in the version that you have, but put some, we put some words in there saying that, you know, this is a, this is, this is new technology, and you're rightfully flagging, you know, some areas where we don't have all the knowledge that we should.

We don't think we're going to, you know, go through the first cycle with the same number of inspectors that we might go through, you know, five cycles down the road. So, we do intend to have some additional staff looking at, certainly in a startup --

MR. CAMPBELL: Initially.

MR. MILLER: -- but also as we're walking

through in the first, in the first cycle or two to see, well, what is the right number?

How do we balance that, you know, outage inspection versus how many samples we need to do during a, during pre-outage, during the regular operation? So, I'd say that's still an ongoing thing that we're looking at.

CHAIRMAN SKILLMAN: I'd like to ask a question here because I, I'm sensing a theme that I had not sensed earlier of, in the comments that have been made, it's clear in my mind that there is either an aversion or a prohibition for at-power containment entries.

And I worked at a plant where we made at-power containment entries. We did not go into the primary shield, but we certainly went into the operating flat, and we had access to the reactor building coolers, and to other equipment. And we made routine visits.

So, what I'm hearing here, almost, is this idea that, once you start up, you do not enter that containment.

Is that an overwriting guideline for how you're approaching inspections? And I would be quick to point out, if it is, then I can recall a very

powerful inspector from Region, from Region 1, whose counsel was, shame on you. You would, you would not go in a containment for a year? Shame on you.

You ought to go and touch the tags. You don't have to stay long, and you don't have to get exposed, but you ought to go take a look.

I thought that was wise counsel from one of your best. It was an NRC inspector. And we did do that at TMI1 routinely.

And so, I'm wondering if, for a few more pieces of shielding, for choice of several routes, there could be some actual inspections that ensure that the safety systems we're talking about, RTNSS and otherwise, are fit for duty.

MEMBER STETKAR: But, I'll just make, I hear a lot about, well, we can't get inside the containment and everything's inside the containment.

If I restrict my focus to safety-related stuff, I'll call it that, then a lot of the safety-related stuff on the AP1000 is inside the containment.

And on different new plant designs, all of it is inside the containment. Safety-related stuff might be squib valves, or it might be valves to actuate the passive RHR heat exchangers.

Best that I can tell, almost all of the

RTNSS stuff is outside the containment. I can go walk and touch that stuff pretty much 365 days out of the year.

So, this notion of, how am I balancing inspections of RTNSS and safety-related, and what's the scope of my inspections, and how much resources, from what I read in the white paper, you do have a target list of RTNSS stuff that will be included, at least under the initial inspections, if not the ongoing ROP. And all of that stuff is outside the containment.

So, this notion about the fact that I can't get to it and the inspections that I'm going to be doing during an outage when I have that, you know, hoard of characters inside the containment, we'll get it done, doesn't --

MR. CAMPBELL: All samples --

MEMBER STETKAR: -- doesn't apply for the RTNSS stuff, which I can take samples and go put my hands on it and --

MR. CAMPBELL: All samples are, even outside containment, we recognize RTNSS is outside containment, and that would be included in the sample population for inspection --

MEMBER STETKAR: And things like power

supplies for squib valves and that sort of stuff is all outside containment.

MR. CAMPBELL: Yes.

MEMBER STETKAR: Best as I can, you know, there are wires that go through the containment. Can't make penetrations, but that's all outside.

MR. CAMPBELL: We recognize there's still going to be inspections outside containment. But we know that the important passive ones are inside containment.

CHAIRMAN SKILLMAN: So, are you at least mentally prohibiting inspections of those?

MR. CAMPBELL: As an inspector, I would take advantage whenever the licensee opened up containment to enter.

I would take those opportunities during a forced outage or if they needed, I actually went in and, at ANO, and looked at a leaking valve with, at a power entry, and I was gassed up in my lungs for about a week.

You know, so, going in and, you have to consider the health of the, and safety of the inspectors and --

CHAIRMAN SKILLMAN: Well, I inspected --

MR. CAMPBELL: -- you know, going in

there.

CHAIRMAN SKILLMAN: I understand that. I mean, I've done those entries. I understand.

MR. CAMPBELL: But we have not considered whether, would we, I'm kind of hearing, would we ask the licensee for an AP1000 reactor to, permission to go in at-power? Is that, is that what you're asking? Is that what you're inquiring? That's --

CHAIRMAN SKILLMAN: I'm curious whether or not you would expect them to go in and look at their own equipment.

MR. GIBBS: May I? I'm a former operator and I've made multiple at-power entries. That's, that was part of the job.

As an inspector, I certainly wouldn't want to do that. I didn't want to do it as a, as an operator. So, of course, a licensee has the primary responsibility for safety.

MR. CAMPBELL: Right.

MR. GIBBS: No question about that. I think it would certainly be a, very much a change for us to, I would characterize that as having an NRC inspector go in containment for a routine inspection.

I don't believe that we're going to be doing that. And I haven't, I haven't spoken with

Steve about this particular issue.

However, having said that, if something happens in, that we know about, in containment, then we would, we would have to look at that on a case by case basis.

But I suspect, because of ALARA, that we would be biased towards not going in containment at power, and that we would rely on the licensee to provide us the information that we would need, such that we could do our job. I don't know if that addresses your thoughts about these inspections.

CHAIRMAN SKILLMAN: It does. And I'm certainly not promoting at-power entries. But what I would communicate is these new containments are well built and they're well shielded, and there are areas inside the containment that are, at background, as we are right here right now.

And so, I think, now, one should not view entering containment as a near death experience. One can go in containment safely, dressed out safely, and do so, ALARA, and accomplish a mission as long as it's planned properly.

And so, if that's what it might take to look at some of this equipment to ensure its operability, I would say, okay. Maybe that's part of

how the new plants are operated.

MR. MERZKE: I think in most cases, even with the operating plants, if there's a containment entry being done by the licensee for some reason, generally, a resident inspector's going to accompany him, or that person, as they, as they make that containment entry.

A lot of times just to see, you know, what's going on in there, if there are any leaks, and to do that containment closeout and make sure it's done properly.

CHAIRMAN SKILLMAN: But containment closeout is a little different. I'm talking about an at-power entry to go and inspect something.

MR. MERZKE: Well, I know. I'm just saying, there, licensees do that on occasion, and I'm, I mean, of the, some resident inspections actually accompanied those licensees when they make those containment entries.

CHAIRMAN SKILLMAN: Okay, let's move on. I was just struck by what I thought was a theme here. What I hear you say is, yes, sometimes it is appropriate to make an at-power containment entry. It's not normal and it's not necessarily championed, but it's possible.

MEMBER RAY: This is Harold. I want to just point out that, in containment at-power is different in AP1000 than the other plants you're talking about because of the residual heat removal function of the blow down to containment.

It doesn't mean you can't do it or wouldn't do it. I don't know what the status of that is, but it's different anyway.

CHAIRMAN SKILLMAN: Okay. Thank you, Harold. Let's proceed. Go ahead.

MR. CAMPBELL: I'm done, Daniel.

MR. MERZKE: I guess, sorry, I guess that wraps up the inspection procedure portion, and we'll be moving on to the significance determination process.

MR. GIBBS: Good afternoon, everyone. Thank you for listening to us, and we appreciate what you have to say.

And I think I heard something in, a moment ago said, at the risk of being provocative, well, I like being provocative.

I think it helps us, helps us understand what we're doing and actually has a, you know, have, maybe ask ourselves questions that we haven't thought about.

So, I think we, I can speak for everyone to say that we appreciate what you, what you have to say here today. I'm Russell Gibbs. I'm a senior reactor operations engineer in the Division of Inspection and Regional Support.

Dan and I work in the same division. I am the program lead for the Agency on the significance determination process.

Not just for those that are more, you know, associated with PRA or probabilistic risk assessment, but also those that are more deterministic in nature.

So, I'm the Agency's lead for all seven cornerstones, and we have some representatives here today in some of these various cornerstones. And I'm joined by See-Meng Wong, who is in our Division of Risk Assessment.

See-Meng and I work very closely together in terms of our program and its successful implementation. And Matt Leech is also here today to talk about one of our procedures.

I wanted to reflect on a couple things about the SDP in general. And just to, just to make sure we all are on the same page about what the SDP's trying to do.

And of course, reflecting on our principles of good regulation, the SDP needs to be reliable.

It needs to be as good as it can be, but at the same time, it needs to be efficient. And so, why, you know, this, so, this balance between reliability, us getting it right, and us doing it in an efficient way is something that we have to balance in our program.

And it's very important that we get that sweet spot right, because of course, our decisions have, can have significant implications for licensees. But you know, we also need to make these decisions in a timely manner.

Now, it's important that we do it in a timely manner for, one reason is that we need to make sure we know what we need to do to follow up.

And we need to, for example, if a, if an inspection finding is greater than green, we need to conduct that follow up inspection in a timely manner to make sure the licensee's taking the right action.

The second thing, and some people oftentimes forget this part of the SDP, and that is, we need to be timely such that we can be in step with the assessment process.

The assessment process is done quarterly, semi-annually, and on an annual basis. So, if our decisions are not in step with the assessment process, then we have to wonder, are we being relevant with respect to our decision making in the overall course of understanding overall licensee performance.

And then thirdly, we have an obligation to inform the public about licensee performance. So, I say that because it's, again, it's striking of this balance between reliability and efficiency.

And so, we try to do that when we develop our program, and we have to be mindful of that. Okay. With that, slide, next slide, please.

MEMBER POWERS: You've made, of course, a very important point here of the balance --

MR. GIBBS: Yes.

MEMBER POWERS: -- that we have to get there. How are you doing?

MEMBER POWERS: I, you know, that's, this is something we're looking at currently. We have, I believe, and many of the folks involved in the significance determination process here in the room, it's a good, a very good program, but I will tell you, we think we could improve.

MEMBER POWERS: Yes. There's certainly

horror stories about SDPs taking a long time.

MR. GIBBS: Right.

MEMBER POWERS: And you know, that, when the ROP was first set up, we, everybody said, what else do you, tell me something else that's new because it was new. But I wondered how it was doing now that we've had some substantial experience with it.

Now, of course that's complicated because the risk assessment tools that you have keep getting newer and better and --

MR. GIBBS: Right.

MEMBER POWERS: -- more complex.

MR. GIBBS: Exactly.

MEMBER POWERS: And your expertise, which, by the way, in my experience, the guys that's doing it are really outstanding.

And, but they want to do more sensitivity, how close, or were we near cliffs? Is there something here more?

And they're always looking to do more and more and more with it. I wondered if you tracked and just how well are you doing on making this balance?

MR. GIBBS: Right. Let me, let me offer you a bit of data to help you, to help you appreciate how we've been, how we've done.

And I would agree with you, our tools, over the years, have got, have improved. Our staff has become actually quite sophisticated --

MEMBER POWERS: They are, they absolutely are.

MR. GIBBS: -- with respect to how we go about performing these risk calculations for the initiating events, mitigating systems, and barrier integrity cornerstones.

So, there's an initiative underway that was actually started as part of Commission direction to streamline the significance determination process.

That was actually started as a result of a specific SDP that took us over a year. In fact, it took us about a year and a half to make a decision. This is the Arkansas Nuclear One.

And so, since that time, what we've been doing is we've been looking at our program to identify ways that we can become more efficient, keeping the reliability. We want to get it right. But how can we do this work in less time?

And so, the streamlining initiative is now part of what we're calling inspection finding resolution management.

The data that I would offer to you is, as

we looked in the past, about 30 percent of the time we reached, that we made a decision for a greater than green inspection finding, it took us over a year.

About 40 percent of the time, it took us over eight and a half months. And so, we stepped back from that, and we said, we can do better.

And so, what we're doing now, we're in a trial period to identify ways that we can actually improve our performance.

And you know, I'm happy to tell you that from the, from the, now, we're in about the eighth month of this trial period, our performance is improving.

And the one reason is for improving, mainly, I suspect, is one, we have better, we're seeing better management oversight of the work that the staff is doing.

But two, really important, is that the front end of the whole inspection and SDP, we have established a trial period, a metric, of 120 days.

And so, that metric is, once we become, once we become aware of an issue of concern, we're looking at about 120 days to make some kind of decision and inform the licensee with a formal exit meeting. In the past, we had no metric for that.

And so, that one metric is really helping us in combination with improved management oversight of these greater than green issues. I hope that, I hope that explains --

MEMBER POWERS: Well, I mean, that's of course the step to take. There's another step is, are you asking, especially your analysts themselves, are the tools adequate for the chore, the task you have, or are there things that would make it possible to take that 120 days and turn it into 60 days if the tools --

MR. GIBBS: Right.

MEMBER POWERS: -- were improved?

MR. GIBBS: Okay. So, if the 120 days is about inspection.

MEMBER POWERS: Yes.

MR. GIBBS: That's an inspection focus. With respect to the significance determination process and time, we believe the current SPAR models are very good in helping us assess the delta core damage frequency, or the change in risk from the baseline risk.

We believe those tools are really good. Sometimes though, where we get into some inefficiency is when we try to use those tools in situations that

they're not really meant to be used for.

I think we talked about early, about external flooding. We tried our very best in the past to, if you will, quantify external hazards from a, from a significance standpoint.

MEMBER POWERS: I see what you're saying.

MR. GIBBS: And then, we would go back and realize that, you know, this is not really helping us. So, we would use this Appendix M approach, which is a more qualitative approach to assessing the significance.

MEMBER POWERS: You know, you're talking to the people developing the codes and saying, look, this is a problem I had. I was using your tool in the wrong place and I had to resort to something else. It would really be nice if I could just use the tool.

MR. GIBBS: Correct. Exactly. And that's exactly what we're doing right now to help us improve our performance.

And we're very, I think we're very hopeful about where this is going to take us over the next months.

MEMBER REMPE: So, have your discussions led to a user need or how far are those discussions going? Because, I mean, not much gets done unless you

have a user need, right?

MR. GIBBS: No. Not, no. That's, we haven't a user need with respect to reaching out to --

MEMBER REMPE: The tools are inadequate and something else needs to be done?

MR. GIBBS: Okay. So, the tools themselves is an ongoing initiative with the SPAR models through Idaho National Lab, they are the ones that develop these tools, the SPAR models.

MEMBER REMPE: Right, I know.

MR. GIBBS: And then, we have, for example, See-Meng Wong here is the lead staff member in the Division of Risk Assessment to help with that, working with the Office of Research. So, that is an ongoing project every year.

MEMBER REMPE: So, it's a, I don't know. It's a base interaction, so you don't have to have a user need --

MR. GIBBS: Correct.

MEMBER REMPE: -- is what you're telling me. But if you're trying to go beyond the maintenance of the codes and you want a new model, that's still covered by it too?

MR. GIBBS: For example, if we wanted shutdown risk models that we currently don't have --

MEMBER REMPE: Right.

MR. GIBBS: -- we would have to have a very concerted effort to develop those models, recognizing that very few licensees have shutdown models. But yes, we would have --

MEMBER REMPE: Then you would have to have a user need.

MR. GIBBS: Absolutely.

MEMBER REMPE: Or if you wanted, what about if you want to try and expand them for evaluating them, FLEX strategies? Is that covered by your base --

MR. GIBBS: I believe --

MEMBER REMPE: -- allocation?

MR. GIBBS: -- that would actually be covered by the base, but See-Meng Wong, you might want to --

MR. WONG: Okay.

MR. GIBBS: -- comment.

MR. WONG: If I may, in response to your question, ma'am, we talked about this question. We do have a continuing user need to our colleagues in the Office of Nuclear Regulatory Research.

It's a multi-year program where we indicate to our RVS colleagues to provide support to,

on an ongoing basis.

In fact, a 24/7 kind of a support. Not only to maintain and enhance the SPAR models to assure the fidelity of the SPAR models when we know that there are modifications or changes and all that is going on.

The ongoing user need has also very specific tasks that we ask our folks to improve the, especially in the methods in the PRE technology and so on.

So, we do have in place this mechanism and this vehicle to ensure that our tools that we use to support the implementation of the ROP and SDP process is continuously improved.

Another thing I want to add is that, to make our risk analysts and our senior reactor analysts, which is really the core of individuals in the Agency who process the issues using the significance determination process tools, we have in place, and I believe there was a presentation on a, what we call a risk assessment standardization project handbook, which is a codification of best practices of our experience in SDP, MDA.3, and the accident sequence precursor program.

So, and these handbooks, there are four

volumes, Volume 1, 2, 3, and 4, that address internal event risk assessments. Volume 2, external event risk assessment methods and guidance.

Volume 3 is a, sort of a very detailed SPAR model, kind of an execution manipulation that gives guidance to the analysts. And then, Volume 4 is, address for shutdown risk assessment.

So, these, in relation to the tools that we have, we have a document in place, and also our continuous effort, as Russell's mentioned earlier, the ROP, SDP is a living program.

We continuously upgrade and improve our MIC guidance documents in addition to your last handbook where we do engage the external stakeholders to seek their views as we try to keep improving the way on how we process our inspection findings more efficiently, and so that, just kind of a knowledge management tool in place. That's, does that answer your question?

MEMBER POWERS: Well, it's excellent. I mean, I'm glad everything's on, but I think Russell's hit upon a key that we need to make sure these people providing you the tools understand the metrics you're imposing on your own activities so that they understand what you're trying to achieve.

Otherwise, they end up playing in a sandbox a lot because they need to set their own metrics and performance goals and things like that.

So, yes, I think you've hit upon a strategy, and I'll be delighted to hear how this comes out as you go through your trial period and as you extend these metrics and what you'd like, what's your goals, your stretch goals and things like that, and communicate it to people providing your tools.

Because I think there's opportunity, they have to proceed kind of blind because they don't know what you're trying to achieve.

And that communication, which has already started, as Mr. Wong pointed out, that you're, you've got ongoing dialogue with them.

MR. GIBBS: We absolutely do.

MEMBER POWERS: And --

MR. GIBBS: And in fact, the risk, the risk analyst community is no fault.

MEMBER POWERS: But I know this for a fact --

MR. GIBBS: Right.

MEMBER POWERS: -- I mean, myself, but I think you've hit upon a tactic that actually helps them when they understand what you're trying to

achieve.

MR. GIBBS: Yes. Bingo.

MEMBER POWERS: Yes. Yes.

MR. GIBBS: You got it right, as far as I'm concerned. Okay, yes, indeed, we have been engaged with the risk community.

They know exactly what we're trying to do and why we're trying to do it. And so, this is over a two year process, and we're, again, we're hopeful for the outcomes.

MEMBER POWERS: Well, I hope that you will take the initiative to ask to come to us again and talk to, once you've had a chance to digest what your two year process has led to.

MR. GIBBS: I would be happy to do that.

MEMBER POWERS: And how you, because I don't think it ends. I mean, in two years, I think that's where you're getting your feet wet --

MR. GIBBS: Yes.

MEMBER POWERS: -- and what you can do to go in for a full swim here.

MR. GIBBS: Indeed, it's a living program.

MEMBER POWERS: And because this is really the key to success on a lot of this ROP, is understanding well, where the resources should be

focused and not trying to inspect every damn thing in the world.

MR. GIBBS: And knowing when to spend how much time on assessing a performance deficiency and its significance, how much time is appropriate?

MEMBER POWERS: That's right.

MR. GIBBS: That's what we're trying to do.

MEMBER POWERS: Because the Agency does not have infinite resources --

MR. GIBBS: Correct.

MEMBER POWERS: -- nor does the licensee.

MR. GIBBS: Correct. Okay. If, any other thoughts about that, we'll continue.

MEMBER REMPE: Oh, okay. So, a few weeks ago, we were at a meeting and I remember one of my esteemed colleagues making a comment about a lot of resources were going to get a fifth significant figure out of a SPAR model.

And again, I think what Dr. Powers is saying could be really honed in if there's some significant, I mean, maybe that user need ought to be really focused on trying to make sure that you're getting the bang from your bucks to have some process improvements and efficiencies.

And so, that would be something that would be good to see coming out of some of these discussions.

MR. GIBBS: I appreciate that. In fact, you know, early on in the SDP, I remember a letter that we were sending to the licensee about the significance of an inspection finding, and I believe it had four to five significant digits in the, in the risk estimate.

And all of you, of course, know that that is not helpful when describing the significance of a performance deficiency.

And so, that was many, many years ago, and can assure you that now we have, we have, we are improving, have improved greatly with respect to understanding really that point estimate or that change in risk from the baseline risk. Really, the, you know, the perspective of what that number really means.

In fact, we're taking that further in our decision making process by doing a number of things, such as conducting training for our decision makers about what is a PRA and how to go about making an effective and efficient decision in the SDP process.

MEMBER POWERS: Boy, is that music to the

former Commissioner Apostolakis' ears --

(Laughter)

MEMBER POWERS: -- because, I mean, he was really concerned about decision making --

MR. GIBBS: Right.

MEMBER POWERS: -- in a risk-informed context. And again, that would be, that's a whole new subject for this crew to come talk to us about is decision making in a risk-informed context.

MR. GIBBS: So, the findings --

MEMBER POWERS: I don't know how to do it.

(Laughter)

MR. GIBBS: Well, right. Thank you for your, for your comment. I appreciate that.

MEMBER REMPE: He started it.

MR. GIBBS: And all, both of you. So, the SDP for new reactors, I'll just, you know, the bottom line here is, you know, we don't see much change needed, and those are the procedures in the second bullet there that we believe need to be changed.

So, we'll, let's get into some more of that detail. Let's go to the next slide. Of course, the background on this, as has been mentioned a number of times today, we received, I believe, some very good direction from the Commission, and that is addressing

these circumstances, and this has come up today, that are unique to new reactors.

Uncertainty of, or uncertainty of reliability of passive systems, structure systems and components, or those SSCs with limited operational experience.

This is specific direction from the Commission, and we've actually talked about that in our discussion today. So, that, I thought that was a very good dialogue to sort of tee up this discussion.

And then, as I pointed out earlier, the Commission, I believe, was smart to tell us that we needed a structured qualitative framework as well for certain events of conditions that are not evaluated in supporting risk models.

I think what that's saying is, if there are situations that there are no risk models available.

And let me, let me make a comment back again to the SDP. The program itself, it really encourages the staff to use the best available information in the process.

Using best available information help, enables us to make our decisions in a more timely way. Again, balancing reliability and efficiency in that

regard.

And so, I'll talk a little bit about what we're doing to address these specific, these specific direction. But importantly, is that we continue to place emphasis on the quantitative approaches.

So, if there are risk models that we can use, we should use them, and we should try to, we should try to produce a quantitative result.

And why is that so important? Because quantitative results, in their proper perspective, are objective, and they are repeatable.

And that's very important with respect to the reactor oversight process as we interact with licensees. Such that, we, the dialogue that we have licensees on performance deficiencies, that we're both on the same page, if you will, about understanding the significance of any given performance deficiency.

So, this is, this is the specific direction provided by the Commission, and we certainly believe it's appropriate. The next slide, please.

So, you know, I was reflecting on our discussions about the same safety expectations, and that whole, that whole discussion that we were having about the SRM SECY 10-0121, and reflecting on the current SDP.

And would the SDP change dramatically if, indeed, the Commission adopted, you know, a lower threshold, if you will, for new reactors? And honestly, I don't think it would.

I think the program itself is very robust indeed. And it, but of course, we would have to make some changes with respect to some calculations and outcomes of those calculations.

But, so, as I'm looking at the program, and that, you know, and that third item there, the existing program is very robust.

In fact, I counted up, I counted all the SDP documents, and not to say that numbers are, should impress you, but we have 34 SDP-related manual chapters.

Now, that includes very specific guidance for the various cornerstones. But also includes our program guidance as well.

And so, that's a, that's a, that's, that's a fairly large program. And some of our SDPs are very specific.

For example, there's an SDP for steam generator 2 rupture. There's an SDP for containment. There's an SDP, as Jeff Mitman pointed out earlier, for shutdown. There's an SDP for at-power, et cetera.

And there are SDPs for emergency preparedness, security, radiation protection. So, it's a very robust program, and so, we're continuing to evaluate how we might, how we might change that program. And that's exactly what we did.

CHAIRMAN SKILLMAN: Russ, let me jump in here --

MR. GIBBS: Sure.

CHAIRMAN SKILLMAN: -- because the comment that I want to make is precisely in this area of specificity of the SDPs. From your white paper, just let me read a sentence, and then I'll make my comment.

In your white paper, you write, and this is on page, it's under significance determination process. It's the end of the second paragraph.

The necessary modifications include new screening questions for the safety cornerstones of initiating events, mitigating systems of varied integrity, as well as addressing findings associated with the reliability of passive SSCs, digital instrumentation and control, and human performance issues uniquely associated with operational practices in gen 3 reactor designs.

So, my rhetorical question is, what about passive physical functions credited in the license?

For example, now, you just mentioned 34 SDPs.

My thought is, there needs to be a significance determination process that is designed around assessing the significance of the failure of a physical condition.

For instance, the containment surface, heat transfer, cleanliness. For instance, the ability of the water tank valve to be, for certain, open when commanded.

It's not too different than flight time testing for a control rod. You're not really looking at the rod, you're looking at the tech spec requirement for 90, 95 percent inserted in 2.2 seconds. You're looking at the function, not the device.

So, what I'm, what I'm suggesting and challenging you with is the notion that because the passive designs are crediting passive features, passive functions, thermal hydraulic features, other such features that are really not devices.

They're actually physical conditions. That your SDP should have the capability and perhaps the guidance to assess the significance when that, when that physical phenomenon doesn't do what it's supposed to do, much like when the flight time test

comes out four seconds to 100 percent insert when it should've been 2.2. Right?

At that point, you're saying, you know, we know the rod's stuck or there's something going on, but what we're monitoring is the time, not the device.

I would suggest that you are actually going to need to monitor a physical phenomenon, and it is as important in the passive design as SSCs are in the active design.

MR. GIBBS: Okay. So, and thank you for that. I want to make sure I get that. And I see See-Meng Wong is writing very quickly over here.

(Laughter)

CHAIRMAN SKILLMAN: I hope I'm not being confusing.

MR. GIBBS: No, no, no. No, you're not. And so, what we're trying to do in using the current framework, what we'd like to do as best as we can, and as efficiently and reliability as we can, as we look at a degraded position caused by a licensee performance deficiency, and we try to make a determination if it is a very low safety significance.

In other words, does it screen to green? That's the first step after we have what we call an inspection finding.

And so, See-Meng is going to talk about manual Chapter 6 and 9, Appendix A, which is the at-power SDP.

We will also talk about Appendix G, which is the low power and shutdown. Matt will speak to that.

And so, what our challenge is, is to, is to design a screening question that would actually take into account what you just said.

But keep in mind that the program, if the function itself is not lost, and consistent with other aspects of our program, that would screen to green. That's the way our current program is now.

However, if the function is lost, and by the way, as I mentioned earlier, if we lose function of these passive systems, this could be a very significant inspection finding.

CHAIRMAN SKILLMAN: Well, see, it's not just loss of the passive system, it can be compromise of the physical feature on which --

MR. GIBBS: Sorry.

CHAIRMAN SKILLMAN: -- the phenomenon depends.

MR. GIBBS: Degraded, if you will?

CHAIRMAN SKILLMAN: Yes, it could be.

MR. GIBBS: Right.

CHAIRMAN SKILLMAN: A three percent degradation in heat transfer could result in failure in your ECCS performance.

MR. GIBBS: Right. Right.

CHAIRMAN SKILLMAN: It could be just as subtle as subtle can be, and unless there's a change in the SDP process to have a magnifying glass that's thick enough to recognize that, it'll get lost.

MR. GIBBS: So, when we get to Appendix A, maybe you can address that further. But we understand. I believe that we understand what you're talking about.

And it's a matter of the level of detail that you would need in your tool to understand the change in risk, not just from it being lost, but from it being degraded.

CHAIRMAN SKILLMAN: Compromised.

MR. GIBBS: Compromised in some way.

CHAIRMAN SKILLMAN: Yes.

MR. GIBBS: Correct.

CHAIRMAN SKILLMAN: That's the point I'm trying to make.

MR. GIBBS: Yes, yes, yes.

CHAIRMAN SKILLMAN: And I'm trying to say,

it can be very subtle.

MR. GIBBS: Yes.

CHAIRMAN SKILLMAN: For instance, in a passive design where you're depending on the siphoning factor, thermal density, it could, it could require that there needs to be a manometer.

It will tell you, you still have an inch and a half head driving end. But that's not a whole lot of driving head, but it may be enough to prevent from having a LOCA, or prevent from having --

MR. GIBBS: Right.

CHAIRMAN SKILLMAN: -- fuel damage. What I'm trying to say, as we move into this realm of passive features, the sensitivity of those, the failure of success, becomes very important. And unless the SDP is kind of looking at it that way, it might miss it.

MR. GIBBS: Well, you know, the SDP does rely, in many ways, on the licensee's rigor, if you will, in determining operability of a, of a structured system or a component, most of which we would be saying here are structures, I suspect.

CHAIRMAN SKILLMAN: Well, you're saying structured system component, and I'm resisting that, because I think it's structure system component and

passive phenomenon, about passive phenomenon.

MR. GIBBS: Right, right, right. And does that passive phenomenon actually cause, have an effect on the structure system or component?

CHAIRMAN SKILLMAN: Well, it certainly carries the day for ECCS in the AP1000.

MR. GIBBS: Right, right, right.

CHAIRMAN SKILLMAN: It's the heat transfer into the building.

MR. GIBBS: Right. Okay.

CHAIRMAN SKILLMAN: In the surface of the building.

MR. GIBBS: We noted that.

CHAIRMAN SKILLMAN: Thank you.

MR. GIBBS: We appreciate that.

CHAIRMAN SKILLMAN: Okay, okay.

MR. GIBBS: Okay? Okay. So, you know, again, we looked at every SDP. We have inspect, we have leads for every SDP that we have, and a gap analysis was done.

And as it turns out, because the SDP is essentially design neutral, design neutral, we, we're not, we don't believe we need to make many changes.

For example, the deterministic SDPs, the SDP for emergency preparedness, Appendix Bravo, the

SDPs for radiation protection, both worker and public.

Appendices Charlie and Delta, the Appendix for security, Appendix Echo. We looked at those and we don't believe any changes are needed as the basis we're providing in the document we provided to you. But in general, it's a design neutral consideration.

Now, we also looked at SDPs for fire, which is manual Chapter 6 and 9, Appendix F. Operator re-qualification, which is Appendix I or Indigo, steam generator 2 rupture, maintenance rule, B5(b) and mitigating strategies.

We believe that none of those SDPs are going to require revision for new reactors. However, in looking and doing this gap analysis, we do believe the at-power, as you point out, possibly for passive structures, the SDP for at-power, Appendix A, the SDP for shutdown, Appendix G, and that for containment.

And then finally, and a very important SDP we use for qualitative considers, Appendix M, we believe all four of those documents will require revision for new reactors.

And we're going to give you some more detail in that regard in just a moment. Next slide, please.

So, I'm going to turn this over to See-

Meng. He's going to talk to you a little bit about Appendix A and Appendix M, and then, and then Matt will follow with Appendices G and H. See-Meng?

MR. WONG: Okay. Thank you, Russ. Good afternoon. My name is See-Meng Wong and I work in the NRR Division of Risk Assessment.

I'm a senior reactor analyst, and I've been in this business I think since the implementation of the ROP, providing support in the development of some of the SDP tools, as we have it today for the current operating fleet, as well as some of the specific SDPs like Appendix F, Appendix M.

So, that's the experience that Russ has, all his division has assigned me to work on to develop or improve the SDP tools that we have to customize it to accommodate the AP1000 design, or the new reactor designs.

So, as Mr. Skillman, that you've commented, just very, very clearly, very specific features in the AP1000 design and the new reactor designs that we have to have a greater understanding to make sure that findings or issues, when the plant is in operation, that we have a good microscope.

So, this, as a step back, the IMC 0609, Appendix A, is really our framework and guidance to

screen what we call at-power inspection findings. And then, directing the user to other applicable SDP appendixes, in case for specific areas.

For example, if it's a fire issue, fire brigade issue, we have got specific direction to say, please proceed to go to that specific appendix.

And then, and if it filters through what we call the screening process, which we consider it to be the central foundation for the Appendix A to perform a detailed risk evaluation if the issue is screened greater than green.

So, what we have today in the Appendix A framework, so to speak, is that we have a Phase 1 screening process.

And the Phase 1 screening process will look at the various cornerstones, initiating event, mitigating systems, barrier integrity, and external events.

And so, your finding is identified, and so, the inspectors, and more often than not, the inspectors, once a performance deficiency has been identified, will work with the senior reactor analysts at our regional offices to process it to determine whether that issue is screened to a low risk significance issue.

So, that's a logic that we apply, kind of a systematic logic. So, if it is looking at a degraded condition, there are filters. Okay?

There are what we call simple logical or illogical and questions or conditions, so, like Mr. Skillman said, if there is an issue, a particular issue that you are talking about where, I don't remember, but one example I can think of, for example, containment coating.

CHAIRMAN SKILLMAN: It's a physical phenomenon.

MR. WONG: Physical phenomenon.

CHAIRMAN SKILLMAN: I'm trying to, I'm trying to reinforce this notion. We talk about structure systems and components.

MR. WONG: Right.

CHAIRMAN SKILLMAN: That's dandy for active plants. The passive plants are depending upon physical phenomenon, not necessarily the components.

MR. WONG: Correct.

CHAIRMAN SKILLMAN: And so, it seems to me that that is a, that is a flag that needs to be recognized in the ROP.

MR. WONG: Yes. So --

MEMBER RAY: This is Harold. Do you think

it needs to be recognized in the ROP primarily, or in the required surveillance testing, which is what I keep thinking you're talking about?

CHAIRMAN SKILLMAN: I think it's both. It needs to be recognized in the surveillance testing, but the --

MEMBER RAY: Well, if it, if it's in the surveillance testing, then the ROP is, for sure, going to pick it up.

As a long-term licensee, and I know you were too, I just don't like to think about requirements being imposed as the oversight process.

CHAIRMAN SKILLMAN: Fair enough. I understand your comment.

MR. WONG: Yes. Thank you very much. So, in looking at what we have for the operating fleets SDP for Appendix A at-power, so, the one big change that we are going to be going forward would be a new set of screening questions to help us to screen the issues, whether it is going to be green or greater than green, or to other specific SDPs.

So, the areas that we're looking for is, again, as mentioned, the passive SSCs, and also the physical phenomena issues that is being discovered.

In addition, we will design screening

questions to screen out issues related to, for example, systems that have got limited operating experience.

One thing that comes to mind is the digital I&C systems that is going to be as part of the AP1000 design or the new reactor design.

The other area that we think that we need to put our eyes on is human performance issues. Potential loss of command and control issues may come up. We may not know, but is this something that we need to look at?

And finally, there are indices that got, and he may have addressed your question, that has got plant-wide implications or effects.

For example, cabling issues, okay? We have an example, for example, in Oconee, we have got cable issues where, you know, that were installed, and we, it is this type of issues that we need to find a way to design the system that we can screen it out, you know, that if we don't have a risk model or we don't have a supporting risk model as the SRM has indicated, then we would specifically direct the analysts and, or the management panel reviewing it to go to Appendix M. Okay?

MR. GIBBS: Well, to say that differently,

to go to a qualitative structure.

MR. WONG: Qualitative structure.

MR. GIBBS: It just so happens that Appendix M is the current framework for that structure, just to be clear.

MR. WONG: So, I will pause here for any questions before I go to the more exciting SDP.

CHAIRMAN SKILLMAN: Please continue --

MR. WONG: Oh, okay.

CHAIRMAN SKILLMAN: -- See-Meng. Please.

MR. WONG: All right. Fifth, the next SDP to, that we currently have in the books is what we call an IMC 0609 Appendix M. It's decision making using qualitative criteria.

Now, as I've looked back at the SRM to, the Commission SRM to SECY 13-0137, and just to make sure that we understand what the Commission direction, I also went back and looked at the voting records that the Commission has provided, as to their recommendation.

And so, I need to add that Commissioner Apostolakis was the lead commenter in circle, providing the direction.

So, and that direction is that, and in my slide is, I will quote directly, and it is quoted

throughout the voting record is that, the Appendix M should be a structured qualitative assessment for, they say events, but more so for degraded conditions, that are not evaluated in supporting risk models.

So, my understanding, and interpretation, is that we should look for a way to use qualitative factors to assess issues that is beyond the capability of, say, a PRA model where most of our SDP tools today are very, what I call, PRA-centric.

That means it's got a PRA framework that leads ones through to make, you know, a quantitative outcome. So, that's how I read into it.

So, following through to that direction, this is what we are proceeding in our activity to try to develop this Appendix M, and provide the Commission as one of the options to answer the question of having a tool that can support assessing findings for the new reactors. Okay?

So, if you were to read in the SRM, there is also a kind of an additional direction. I call it a separate initiative to provide clarity on the use of the qualitative factors for upgrading reactors, because under the current, for the current operating fleet, our Appendix M is a very short, is only a four-page guidance that is, that has been established to

allow the analyst or the senior analyst, when they use it, only in very, very specific entry conditions for situations where there's no PRA model.

For example, issues that's related to reactor management, you know, spent fuel pool reactivity issues. Okay?

Or there was one entry condition where the PRA process, the assumptions that go into the quantitative SDP tools has a lot of uncertainty that gives the, you know, the calculation is going to take a long, long time. Okay?

So, you've heard previously, Russ mentioned, with regard to our current use of the Appendix M.

So, over the past, since 2006, with the current Appendix M, there's been 20 issues that has been processed with Appendix M.

And in that database, there, a few examples were related to flooding issues, some pre-Fukushima, some post-Fukushima.

But in those issues, a lot of effort was initially undertaken to try to find out, you know, what is the best available data to use to define or to establish the flood frequency, you know, 20 year versus 1,000 year rainfall data, which one is better.

But that kind of analysis is what is causing some of the comments that we're having of concerns that it's taking a lot, long, long time.

So, where all this is going to is that Russ and I, we're working to try to come up with better ways, and if Appendix M is one potential option that we think the Commission should review, not only to support, to meet the SRM for the new reactors, but also the benefit of trying to improve the guidance for our decision makers in reviewing issues that are very difficult and complex for the current operating reactors.

So, my next bullet is essentially, we have a general structure, and one of the things, the challenges that we have is to really try to define clear entry conditions, when we can use it, when we should not be using it if, for situations where we know it's going to, you know, it's a better way than spending a lot of time, spending, you know, nine months to a year, you know, on a project to come up with a better or more definitive number to reduce the uncertainty and confidence.

So, where we are going is that, in addition to the defining clear entry conditions, the structure that we are, we are thinking of, the quality

of structure is to follow the Reg Guide 1.174 structure.

That has been reviewed, vetted by the ACRS, and the revision activities has, we have engaged with the industry to make sure that, you know, there's a good understanding on the major decision attributes that needs to go into this qualitative structure.

And so, we are not stepping out on developing something new where it's going to go through a lot of challenging comments of how valid that's going to be. So, we are taking existing processes that we have in place.

Now, having said that, there's definitely, in adopting that approach, that methodology, we need to kind of help refine and make it clear, you know, the questions that is in the decision attributes when we ask the decision makers to kind of put it in a context of a little bit more common sense. So, this is something that we are working on.

So, in addition to the entry conditions, we need to define, provide guidance to define and assess the decision attributes, which, in the current version, there is no detailed guidance.

So, how decision makers have lamented the fact that sometimes we are not very sure, but we still

out go out with the, with the best information that we have with the decision that we make. So --

CHAIRMAN SKILLMAN: What is the status of Appendix M right now?

MR. GIBBS: Do you want me to --

MR. WONG: Yes.

MR. GIBBS: -- speak to that?

MR. WONG: Go.

MR. GIBBS: Okay. So, we have been working on this for a number of months, and we are trying to reach an internal alignment about how to carry forward this direction by the Commission to produce this qualitative framework for both new reactors and to provide clarity for the qualitative framework for the existing fleet.

We believe Appendix M is a very viable option, but we have to get that approved by the Commission. This is a significant change to the significance determination process.

And so, we don't want to, if you will, pretend that the Appendix M and what we, what we plan to do with this is the, is the, is the, is the right option, because we need to propose to the Commission, others as well.

And so, we've been working with our

internal stakeholders. In fact, we just a meeting last Friday with all of our division directors in the regions, and we solicited to them to provide some comments or ideas about other ways that we might be able to do this.

The approach that See-Meng is, has shown you is one of the options that we intend to propose to the Commission.

We have, right now, a due date of June of next year to produce a Commission paper with those options.

We have a lot of work to do to develop each of those options and provide, if you will, the pros and cons, the merits of each. So, June of next year is the Commission paper due date.

CHAIRMAN SKILLMAN: So, your Appendix M is really your response to the Commission's comments and the SECY 13-0137 regarding this development of a structured assessment tool for qualitative assessment?

MR. GIBBS: It is. At this point, it's the best we have, but we're reaching out to our stakeholders to see if there might be other ways that we might be able to accomplish the same thing.

CHAIRMAN SKILLMAN: But this is really the, at this point in time, your Appendix, your

emerging Appendix M is your response to the NRC's guidance regarding SECY?

MR. GIBBS: Yes. It is.

CHAIRMAN SKILLMAN: Okay. Now, I understand the level of detail and the guidance that See-Meng is --

MR. GIBBS: Right.

CHAIRMAN SKILLMAN: -- communicating here.

MR. GIBBS: Okay. So --

MEMBER REMPE: So, out of curiosity, what are the other stakeholders, internal stakeholders proposing?

You said that there are other options. Can you, I know it's preliminary, but give us an idea of what --

MR. GIBBS: I'll give you an idea. What if we wanted to have an integrated approach to decision making for all inspection findings using qualitative information? Not just relying on a point estimate from a PRA.

That's just another idea. Another option is, for these situations that are not necessarily amenable to a probabilistic risk assessment.

Maybe we should define those as greater than green and just go do an inspection to see what

additional information we might need before we go forward with a final decision.

I mean, we're really, right now, in the creative mode of trying to understand what some of those options might be.

MEMBER REMPE: And have, is your process going to, after you finish looking at the internal stakeholders, are you going to start socializing this with the external stakeholders --

MR. GIBBS: Absolutely.

MEMBER REMPE: -- before you go to the --

MR. GIBBS: Because we want their ideas as well.

MEMBER REMPE: Yes. Okay.

MR. GIBBS: Yes. That's a must.

MEMBER REMPE: Yes, I would think so. Okay.

MR. GIBBS: Particularly for this, for this exercise.

MEMBER BLEY: You generated a few questions from me here. There is the existing Appendix M, which you referred to here. Your last comment stirred me. There are more than point estimates from PRAs. If it's a decent PRA, you get more than a point estimate. And thinking about

the uncertainty you got there is useful.

This is a more general question. The paper you're sending up in December clearly won't have this new appendix to a chapter in the inspection manual, but you'll hint at what's going to be in it, I take it.

The other draft inspection manual changes, are they going to be attached to this paper, or is that just going to refer, is the paper essentially going to look like the white paper we had for today?

MR. MERZKE: It's a plan. It is going to be --

MEMBER BLEY: It's a plan, yes.

MR. MERZKE: -- a draft paper. But the Commission didn't ask the staff to provide a paper on the proposed changes to the baseline inspection procedure.

We just added those to this particular paper to give them an integrated look at, okay, we're not, you know, proposing to do MSPI for AP1000.

This is what we plan to do in inspection space, and then in SDP space. We just wanted to give them an integrated look at the whole ROP for the new reactor.

MEMBER BLEY: Okay. I don't want to

divert us here, but somewhere at the end today, I want to come back to MSPI and PIs and ask a question because I've confused myself and I'm going to need a little help.

(Laughter)

MR. MERZKE: Join the club.

CHAIRMAN SKILLMAN: Let's continue, please.

MR. WONG: Okay. So, that's all I have for these two appendixes, so --

CHAIRMAN SKILLMAN: Okay.

MR. WONG: -- I will turn it over to Matt Leech who's a risk analyst in our Division of Risk Assessment.

He's assigned the responsibility to work on any revisions to Appendix G, which is the shutdown SDP, and Appendix H, which is the containment integrity SDP. These two SDPs has been sitting on our books since, for a long time.

And so, as part of our continuing improvement process, these two SDPs were identified as areas for improvement.

So, and if I might mention, we do have a, kind of a feedback process on our use that we rely on to make improvements to SDP tools.

We have an ROP feedback form that the users, the risk analysts, the inspectors, when they use it and they come across situations where they need it, you know, additional guidance or clarifications.

And so, they send it to us and we use that as a, kind of a foundation of how way we can improve the tools that we have in place. So, Matt.

MR. LEECH: All right. So, as See-Meng mentioned, my name's Matt Leech. The first procedure I want to talk about for accommodating new reactors, and specifically AP1000, is our Appendix G, which is our shutdown procedure.

The general approach of the existing Appendix G is going to still work for AP1000s, but the way our current Appendix G works, it was designed years ago before we had any SPAR models for shutdown. And we still only have a few.

So, it's got an intermediate Phase 2 process, which is pre-solved of entries and tables that we use if an operating reactor had a shutdown event.

And we will not need to use that Phase 2 process because there actually is an AP1000 SPAR model. I wanted to clarify that.

Idaho National Labs has developed an

AP1000 SPAR model that's available for our use, and it does have a shutdown event also that can be assessed.

Now, how good it is currently, I don't, or how vetted the SPAR model is at this point, I'm not sure. I know the Office of Research is going to develop a more specific one with INL. This one is just generic.

But the idea will be for Appendix G, that for AP1000, if an event occurs, we can use the SPAR model to assess the event. And we're going to do some additional things as well.

Appendix G has some screening questions in it that will be looked at and revised to see if anything needs to change for the AP1000.

And we're going to, we're going to highlight, in the basis technical document, IMC308, Attachment 3, we're going to highlight some of the differences and improvements that have been made for shutdown in AP1000s. And that's the plan for Appendix G.

CHAIRMAN SKILLMAN: Thank you.

MR. LEECH: All right.

CHAIRMAN SKILLMAN: Let's continue.

MR. LEECH: Okay. The next on is Appendix H, which is containment-related issues. And once

again, the general approach that we use for our current Appendix H, we believe will work for AP1000.

We will have to modify and make some changes here and there. Basically, there's a process for screening events, and for containment in Appendix H that will need to be revised to fold in the new AP1000 for a Phase 1 and a Phase 2, which we actually assess kind of an initial significance of the event and highlight the differences that need to be spoken about in an Appendix H basis document. But our basic approach, with some modification and, will work.

CHAIRMAN SKILLMAN: You know, that kind of sounds like a guy that opens a toolbox on a Saturday afternoon and says, you know, I got this wrench. It's not exactly the right one, but I can make it work.

(Laughter)

CHAIRMAN SKILLMAN: Well, and then half an hour later, he's got a broken wheel stud and he says, that was the wrong tool.

MR. LEECH: I, let me clarify. The reason why it's, will be relatively easy to fold into Appendix H is because even for containment-related events, you start out with a SPAR model and you assess, there's two types of findings.

There's a finding that affects core damage

frequency, and there's a, there's a, there's events that don't affect core damage frequency, but could contribute to LERF, like a stuck open containment isolation valve or something like that.

So, most of the events for Appendix H start out where you need to assess what the core damage frequency is, and that's where having a SPAR model to work with already will make folding it into Appendix H easy.

CHAIRMAN SKILLMAN: But isn't this what John communicated earlier? You can have events where CDF really isn't the driver. LERF is the true driver, and that might involve RTNSS equipment.

MR. LEECH: Yes. Jeff's nodding.

MR. GIBBS: It's not, we don't see these very often, but it is certainly possible.

MR. MITMAN: Yes. We, in fact, we had one within the last year where, during shutdown, the licensee couldn't close the equipment hatch, so, during certain periods of the outage.

So, it had zero impact on core damage. All right? But it has an impact on this with large early release frequency.

So, Appendix H is currently set up to divide the issue into two spaces. One where you have

an impacted in, one where you have a performance deficiency that impacts both the core damage frequency and the LERF capability. And so, we assess that one way. Okay?

And it also has a section that says, for those issues that do not impact the core damage frequency but do impact large early release frequency, we go to this other approach.

So, we've bifurcated the methodology to address the two approaches. Now, the thing that's fundamentally, I shouldn't say fundamentally, the thing that is potentially different with the AP1000, with its lower core damage frequency, is even though you have a finding that does not affect core damage, the LERF, the LERF criteria is in order of magnitude smaller, all right, so the threshold between green and white for a LERF issue is 1 EMI 7.

So, you base that on a core damage frequency, and then you look at how, what the impact on the containment was. All right?

So, in the case we had earlier this year, if the containment hatch can't be closed, then your containment has failed.

It has failure probability of 1, and in that case, the core damage frequency, or the LERF

frequency is equal to the core damage frequency.

So, we have to, we have to consider the case that the core damage frequency for the AP1000 is different than the core damage frequency, say, for a Westinghouse large dry containment.

And then, we have to look at those issues on containment and how that factors in. Now, the containment response is shutdown.

In an AP1000, I expect to be different than the containment response for large, or for containment issues at-power. You know, you can be relying upon different systems to do that.

So, we've got to take a look at that and think about how those things are impacted. But the general approach of dividing performance deficiencies into two categories, one, that affect both core damage and LERF, will be assessed, and then another portion of the procedure that addresses those issues that'll get containment.

MEMBER RAY: This is Harold Ray. Let me just say, this containment isn't just a containment. It's also the ultimate heat sync heat exchanger, and I think that repeats comments that have been made earlier in this meeting.

So, when you say that containment,

existing procedures and protocol for containments are applicable here, just keep in mind that this isn't just a containment. It's also a heat exchanger.

MR. LEECH: Yes.

CHAIRMAN SKILLMAN: Thank you, Harold

MR. LEECH: I understand that.

CHAIRMAN SKILLMAN: Thank you.

MEMBER RAY: It's what Dick was saying earlier, basically.

CHAIRMAN SKILLMAN: Yes. Thank you. Let's proceed, please.

MR. GIBBS: So, just to, just to be clear, we have work to do. When I, when we say the procedures are going to be modified, we're not exactly sure how they're going to be modified.

But what we're saying is that we have a framework to work within, as Jeff just pointed out, acknowledging that, indeed, we're going to have to make some changes.

We're not exactly sure what those changes are. But we believe the framework itself is solid for us to continue within that framework. Meaning, no new SDPs are going to be required, we don't believe. Okay.

So, the path forward, final slide for the

SDP portion is, of course, we're going to have to reach some internal alignment on these proposed changes. And I presume you're aware of the SRM/COMSECY-16-0022.

There's direction from the Commission where changes, for example, to the significance determination process, are going to require Commission approval.

We believe, most assuredly, that what we do to address the qualitative structure for new reactors, and what we do to provide clarity for the current operating fleet will require Commission approval.

I don't know about the others, but I suspect that the other changes that we're proposing may not require approval, but certainly would require some type of Commission notification, and that would be consistent with that particular SRM-16-0022.

And the, particularly, you know, the value in communicating with, working with our external stakeholders and conducting these public meetings and tabletop exercises, by the way, that's required by our, by our program, remainder of Chapter 0609 is very clear about that, particularly when you're making substantive changes that you do some benchmarking,

that you do case studies, and of course you do some type of tabletop exercise to see if what you want to do seems to, seems to work and make sense.

And then, right now, the schedule is that this work will be done by December of next year. We have a lot of work to do.

We'll see, we'll see where that goes with respect to the schedule, you know, based on what's happening with the construction of the, of the new plants.

And I'll just leave it at that and turn it over to Dan, in case you have any final questions about the SDP program itself.

MR. MERZKE: I see none.

CHAIRMAN SKILLMAN: Please proceed.

MR. MERZKE: Okay.

CHAIRMAN SKILLMAN: Thank you.

MR. MERZKE: I'm just going to roll up, I'm sorry. I'm just going to roll up the summary, basically, of what you've already heard here.

So, I'll kind of just go through these slides relatively quickly. Staff inclusions, we're going to maintain.

We've just seen risk thresholds consistent with Commission guidance preserving the existing ROP

structure as is, and we are proposing no changes to the assessment program, as currently specified in 9005.

Under the performance indicators, many of the PIs are based on regulations and standards that also apply to the new reactor designs.

Twelve of the 17 current PIs remain valid for the AP1000. The five PIs that comprise the MSPI indicators would not be valid.

The staff has identified no new PIs that would be effective in monitoring performance to the AP1000 at this time.

And maintaining existing PI thresholds pending staff evaluation after some operating experience is gained.

So, we'll, I mean, it's part of our regular ROP self-assessment process. We'll always come back on an annual basis and review the performance indicator program.

Are the PIs appropriate? Are they giving us, are they measuring what we want as a measure? Is there, is there another new PI that might be possible? I mean, these are all questions we ask annually, on an annual basis. So --

CHAIRMAN SKILLMAN: And let me ask you

this. Have you heard anything today that would cause you to rethink your second bullet there regarding new PIs or regarding those MSPI indicators?

MR. MERZKE: I heard you say that there's no way why we couldn't make valid MSPI indicators out of the RTNSS systems.

We'll go back and take a look, I know the white paper specified that failure of the RTNSS systems would never cross a threshold.

I guess we could take another look at the thresholds. I need to look at that evaluation. I was not around, or I wasn't part of the development of that paper.

But it's certainly worth taking another look at to make sure that, you know, we have our, we have a good technical basis for deciding not to have an MSPI.

CHAIRMAN SKILLMAN: Well, that was one comment. It had to do with the RTNSS system. The other had to do with considering whether or not there needed to be recognition that the physical phenomenon for the passive plants is just as important as the SSCs are for the active plants.

MR. MERZKE: I did hear that, sir. And I'm, we'll have to take another look at that and

evaluate it under a PI program to see if there's something that we could use to develop that might measure that performance.

CHAIRMAN SKILLMAN: I mean, like Harold just said, for the AP1000, the containment is the heat exchanger.

MR. MERZKE: The heat exchanger. Right.

CHAIRMAN SKILLMAN: And so, the fowling on the heat exchanger surface is important.

MR. MERZKE: That is, I agree.

CHAIRMAN SKILLMAN: And so, when you say, it really screens out, I've got to think of my life back in Region 1.

Not many things screened out. We were in trouble all the time. So, I'm thinking it's the same kind of a deal. And so --

MR. MERZKE: My initial --

CHAIRMAN SKILLMAN: -- I'm sorry, but --

MR. MERZKE: My initial impression is that would be captured more under inspection space than PI space.

CHAIRMAN SKILLMAN: And concur with you, and Harold's right. That's in surveillance. But somehow, this program is intended to flag a departure from fully satisfactory to threshold deficient.

And if I have a fire in containment and I fowled my containment surface, I believe I've got a threshold deficiency.

And I believe that requires some attention. Maybe even scrubbing down the inside of the building. And --

MR. MERZKE: I think you make a valid argument. I don't disagree.

CHAIRMAN SKILLMAN: Thank you. Let's proceed. Thanks.

MR. MERZKE: All right, sir. Thank you. For the inspection program, the staff concluded that there are very few changes needed to the inspection requirements.

There will be additional guidance added, as necessary. We'll be adjusting sample sizes to specify, including, at a minimum, risk important and RTNSS systems.

That doesn't mean that there's, we do say that there's no new inspection required, but that doesn't mean that we're not going to update the guidance to ensure that we focus our appropriate attention on those systems that are important to safety, even if they are not safety-related.

Under MSPI, the IPs, as written, are

sufficient to assess licensee performance and mitigating cornerstone without MSPI.

Again, the guidance will specify the appropriate inspection samples to cover those passive safety systems, where applicable.

And you just heard from the significance determination process, the existing SDP program is robust and inclusive, and no new SDPs will be developed.

And we will be making modifications to the SDPs for at-power shutdown, containment, and use of qualitative criteria.

And those will all be, hopefully, worked off by the end of next year, but we also recognize that we've got some time because the schedule for completion for the Vogtle plants are sometime out yet.

So, we're going keep a push on the schedule, and it's better to get done sooner rather than later.

So, finally, our staff recommendation is for the Commission to approve the staff's plans to modify the ROP for new reactors, as described in that white paper in front of you.

We will be seeking additional Commission approval for specifics, to specific changes to the

SDPs, specifically for the Appendix M.

And if any of the other ROP changes that we're, we identify as part of this process, meet the criteria of significant changes as specified in the SRM, the COMSECY-16-0022, then we will seek Commission approval for those issues too.

Our next steps are to finalize the Commission paper based on your comments and stakeholder feedback, and the paper's due to the Commission by the end of this year. And that concludes our formal presentation.

MEMBER STETKAR: I know Dennis had a question --

MR. MERZKE: Right.

MEMBER STETKAR: -- on indicators.

MEMBER BLEY: Actually Daniel kind of covered mine, so I don't need to ask.

MEMBER STETKAR: You're not confused anymore?

MEMBER BLEY: I think --

(Simultaneous speaking)

MEMBER STETKAR: I'll talk to you after the meeting.

(Laughter)

MEMBER STETKAR: I'll have pencil and

paper ready. What I wanted to ask is, and I wanted to wait until the end to not confuse things.

The whole discussion that we've had today and the discussions in this white paper and the proceeding, the NEI white papers, and the staff's white paper on MSPI, have all focused on that bugaboo of AP1000 because, well, it's the only thing that we have to hang our hat on and, by God, you know, we're going to build four or maybe two or maybe one of them, or maybe none of them.

NuScale seems to be raising its ugly head as something that might actually be built somewhere, sometime, by somebody.

Have you thought at all about that design? Because we actually know something about that design, at least technical information about it.

MEMBER POWERS: About this week's version.

MEMBER STETKAR: That's true. But I mean, you know, we know it doesn't look --

MEMBER POWERS: Well --

MEMBER STETKAR: -- like a cardboard box. That's what I'm trying to get at. We kind of, sort of, they're going to change fundamental elements of the way that machine is supposed to work, at least in terms of safety functions and heat removal functions

and things like that.

Have you thought at all about that in the context of this, to see if any additional, or tweaks are needed?

MALE: I'll be honest --

MEMBER STETKAR: And I, and I, and I'm not advocating thinking about the other 47 or whatever, you know, bizarre designs that have been formulated out there, or necessarily gas reactors.

MR. AYEGBUSI: So, I've been part of this working group for a couple of years now on ROP for new reactors. And so, specifically, the Commission has us working on ROP for new reactors, right?

MEMBER STETKAR: Right.

MR. AYEGBUSI: SMRs don't fall into that. They fall under advanced reactors, right? So, all the work we've done with this working group has been, has been focused on things like ABWR, ESBWR, AP1000-type plants.

MR. MERZKE: So, the short answer is no, we've not really assessed SMRs under --

MR. AYEGBUSI: Because --

MR. MERZKE: -- advised in the ROP.

MEMBER STETKAR: Thanks. You've helped me because I've never drawn the distinction between a new

reactor and a not new reactor because it's an advanced reactor.

MR. MERZKE: I think we kind of touched on this earlier though that, given the processes that we use to make these recommendations for the AP1000, they can easily translate to any of the other reactor designs. And I would say, I don't see why we couldn't use the same procedures for the SMRs. And --

MEMBER STETKAR: That's what I was hoping you would say, but --

MR. MERZKE: And it would be, maybe, a two to three year overall process to lock down a good ROP process for the SMRs. But again, we'd probably end up going to the Commission with those recommendations.

MEMBER STETKAR: All right. Thanks.

CHAIRMAN SKILLMAN: Okay. Chris and Dan and the entire staff, thank you very much for a very informative, very fruitful afternoon. This has been an excellent set of presentations, and I want to thank you very much.

MR. MERZKE: Thank you, appreciate it. And we do appreciate the questions and the scenarios you threw out.

Now, I'll be honest, some of those things you threw out, I wasn't thinking about, but you know,

maybe there are people smarter than me because I just got thrown into this new reactor thing a few months, so I'm still learning. But --

CHAIRMAN SKILLMAN: Thank you.

MR. MERZKE: -- good insights, and I appreciate it. Thank you.

CHAIRMAN SKILLMAN: Let me ask my colleagues, if any has a comment, Dr. Rempe, would you please lead?

MEMBER REMPE: Oh, sure. I also appreciated the presentation, and I found it very informative, and I don't have any additional comments.

MEMBER KIRCHNER: Thank you for the presentation. Just the, one thing I was thinking, looking ahead to new reactors, you know, you had a chart early in your presentation about the cornerstones, and obviously one of them is barriers, although we spent more time on mitigating systems.

Without getting into any one design, characteristic of most of the proposed designs for new reactors or advanced reactors, something would eliminate some of the traditional defense and death barriers that we have with the existing fleet.

Most would take advantage of either inherent feedback affects or other passive design

considerations for ultimate heat rejection, et cetera, reactivity control, and so on. These are, by and large, not, well, passive. So, stationary systems.

And it just seems to me that the, although reportedly, they have a lower risk that remains to be determined, but it would seem to me, just in general, that the inspection, especially inspection, maybe it's, it begins with Appendix B in the actual design and construction, but it continues into operational space, becomes more critical, and yet, perhaps more difficult.

And some of the systems will have what I would describe as, if they fail, they're not going to fail gracefully. That they will have cliff phenomena. And that becomes a little more difficult in inspection space.

You know, you, and I hesitate, and I will not use an actual example, but everything will go along okay, but then all of a sudden, you might see something like increased coolant activity or something.

And then you're quickly beyond, you know, where you wanted to be, so to speak. So, I see that as a challenge to you in trying to come up with a generic approach, looking down the road to new

reactors that are not OWR derivative. And thank you.

CHAIRMAN SKILLMAN: Thank you, Walt. John?

MEMBER STETKAR: Yes, thanks. Again, I really appreciate the discussion. I, and despite my comments on indices and whether they ought to have indices.

I don't advocate calculating seven significant figure numbers according to some algebra, just for the sake of doing that.

I can see where reason changes to the inspection program can accomplish the same goals, and maybe even more efficiently.

And with that, I'm really looking forward to seeing the changes that you're going to be making to the inspection manual so that we have some confidence that, if there is value added today by the MSPI, that we can retain that value through the enhanced, whatever you want to call it, enhanced inspection program. So, thanks a lot.

CHAIRMAN SKILLMAN: Thank you, John. Dr. Bley.

MEMBER BLEY: It's been an interesting day. Although I said I'm not still confused, I've --

(Laughter)

MEMBER BLEY: -- going to try to put all this together and try to get, is a bit difficult. I kind of get why you can't get an index over different kinds of things that go together for which you have some qualitative, some quantitative.

Other, I need to study this stuff a bit more. I'm thinking about, we're writing a letter next month, right?

CHAIRMAN SKILLMAN: No.

MEMBER BLEY: No? Oh, I thought you had a letter coming up on this.

CHAIRMAN SKILLMAN: No, this is a plain information briefing.

MEMBER BLEY: Just information briefing.

CHAIRMAN SKILLMAN: Yes, sir. Yes, sir.

MEMBER BLEY: So, we'll, will we have another meeting on your submittal to the Commission, either before or after you send it out? No?

MR. MERZKE: This is, this paper's going up in December, so it was decided there was no reason to have a formal meeting.

MEMBER STETKAR: The white, the white paper will have a SECY --

MR. MERZKE: That is correct.

MEMBER STETKAR: -- number attached to,

right?

MR. MERZKE: That is correct.

(Laughter)

MR. MERZKE: That is correct.

MEMBER BLEY: Well, we'll surely want to see that white paper when it goes up, and I'm sorry.

MEMBER STETKAR: The white paper that we saw for this subcommittee meeting --

MR. MERZKE: That is the draft SECY paper that's going to the Commission.

MEMBER BLEY: I understand that.

MR. MERZKE: Okay. All right.

MEMBER BLEY: It's not a SECY paper today. It will be a SECY paper when it goes up in December. Oh, okay.

MR. MILLER: There will be some more work on it, but it has the essential elements --

MEMBER BLEY: Yes. I understand that.

MR. MILLER: -- that you see in the white paper. Yes.

MEMBER BLEY: Mr. Stetkar's frowning at me and raising his eyebrows, and I'm not sure if he wants to talk about that. I look forward to seeing its final form and digging back through some of the basis.

You know, John asked a lot of questions

and I started looking around, trying to track down where the thresholds come from, and I guess, going through the, some of the inspection procedures, and you track it back, it seems to go back to SECY-99007. And --

MR. MERZKE: 99007 was the original basis document for the --

(Simultaneous speaking)

MEMBER BLEY: Yes. And some of the language has changed, but the basic ideas --

MR. MERZKE: That is correct.

MEMBER BLEY: -- are still all there. But when you look in the inspection procedures and see, here's a threshold, it says, go back and see that SECY to figure out how we came up with this, and I don't quite have all of that, so I want to go look at that.

Eventually, this will come back to us in some form or another at some point in time, and I hope to be better informed when you come back.

MR. MERZKE: That SECY's about 400 pages long.

MEMBER BLEY: Yes, I got it.

(Laughter)

MEMBER BLEY: It's got a lot of attachments.

MEMBER STETKAR: There's a couple of big new regs that get referred to that --

CHAIRMAN SKILLMAN: Thank you, Dennis. Dr. Powers.

MEMBER POWERS: Well, I'm certainly glad we had this information briefing. I found it very useful.

I am especially excited about what's going on, and with Mr. Gibbs and Mr. Wong in their shop. And my personal belief that we have not optimized our SPAR modeling, PRA modeling to suit the needs of the inspection and enforcement parts of the NRC. And I think we can. And I think they've hit upon a mechanism to drive that if they'll take advantage of it.

So, I'm very hopeful we can have another information briefing once you've gone through your trials and experiments, because I think that's a mechanism that will allow us to start pressing the model developers to get one, get models that are less focused on CDF, and more focused on what it is that the inspection people need to focus their activities in the most productive area.

CHAIRMAN SKILLMAN: Thank you. Dr. Ballinger.

MEMBER BALLINGER: Well, I found this extremely informative. Of the Members sitting around the table today, I'm the least, I have the least knowledge. I'm hopeful that by the time I exit the carbon cycle, I'll know enough to be able to make --

(Laughter)

MEMBER BALLINGER: -- a SECY-an comment on this. But I found it really very interesting. Thank you very much.

CHAIRMAN SKILLMAN: Okay, thank you. Harold, are you still with us?

MEMBER RAY: I am, and I guess I won't add anything that I've said already other than to say I think we should all think about, how do we engage with the, given the comments that have been made, the determination of what surveillance programs are required for licensees of new plants.

I don't think we get into that in the certification process or any other time that I'm familiar with.

So, even though I don't think it's, belongs in the oversight, I think it is an important matter to consider at some point. And with that, I'll end.

CHAIRMAN SKILLMAN: Harold, thank you for

sticking with us. Thank you very much. With that, I would like to invite any member of the public in the audience to come to the microphone and make any comment, please.

Seeing none, we're going to make sure the microphone is opened. On the, on the call-in line, is there anybody there? If so, would you just simply say hello?

Hearing none, we can close the line. With that, everybody, thank you for an excellent meeting, and we are adjourned.

(Whereupon, the above-entitled matter went off the record at 5:37 p.m.)

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REACTOR OVERSIGHT PROCESS FOR NEW REACTORS

Advisory Committee on Reactor Safeguards
Subcommittee on Plant Operations

Contact: Dan Merzke, NRR/DIRS

September 20, 2017

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Meeting Purpose

Discuss staff's evaluations, conclusions and recommendations as noted in white paper to modify the Reactor Oversight Process (ROP) for new reactors in response to the SRM on SECY-13-0137

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Agenda

- Background and overview of the ROP for new reactors
- Discussion of appropriateness of existing Performance Indicators and thresholds
- Baseline Inspection program changes
- Significance Determination Process (SDP)
- Conclusions and recommendations in white paper
- Next steps

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Background

- Baseline risk estimates for most new reactor designs are lower than those for a design similar to that of the current fleet
- Lower risk values raised questions about how to apply acceptance guidelines for changes to licensing basis and regulatory response in ROP
- Over past several years, staff has corresponded with Commission and ACRS to address staff's recommendations related to risk-informed guidance for new light-water reactor applications

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Background (cont.)

- SECY-10-0121, "Modifying the Risk-Informed Regulatory Guidance for New Reactors," issued September 2010 to provide options to modify risk-informed regulatory guidance for new reactors.
- Staff recommended working with stakeholders to identify and implement appropriate changes to the existing risk-informed guidance based on lower baseline risk of new reactors.
- In its SRM, the Commission disapproved the staff's recommendation, and reaffirmed that the existing safety goals, safety performance expectations, subsidiary risk goals and associated risk guidance, key principles and quantitative metrics for implementing risk-informed decision making, are sufficient for new plants.

Background (cont.)

- SECY-12-0081, “Risk-Informed Regulatory Framework for New Reactors,” issued June 2012 to provide staff recommendations on both licensing and oversight processes
- Staff recommended development of qualitative risk insights (deterministic backstops) to supplement probabilistic risk assessment (PRA) information to determine significance of inspection findings.
- In its SRM, the Commission disapproved the staff’s recommendation, and directed the staff to consider using relative risk metrics, or to provide a technical basis for why this option was not viable.

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Background (cont.)

- In SECY-13-0137, "Recommendations for Risk-informing the Reactor Oversight Process for New Reactors," issued December 2013, the staff:
  - Developed a technical basis for its proposal to use qualitative considerations for characterizing the significance of inspection findings.
  - Performed a technical evaluation of the use of relative risk measures for characterizing the significance of inspection findings. Recommended against using relative risk metrics.
  - Evaluated the appropriateness of the existing performance indicators (PIs) and the related thresholds for new reactors.

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Background (cont.)

- In its SRM to SECY-13-0137, the Commission:
  - Disapproved the staff's recommendation to use qualitative measures to supplement quantitative risk evaluations, and directed the staff to enhance the existing SDP for conditions not currently modeled.
  - Noted that the overall structure of the existing ROP should be preserved.
  - Directed the staff to develop appropriate Performance Indicators (PIs) and thresholds for new reactors, specifically in Initiating Events and Mitigating Systems cornerstones, or develop additional inspection guidance to address shortfalls to ensure all cornerstone objectives are adequately met.
  - Directed the staff to explore how the current Safety System Functional Failure PI would be applied to the passive safety-related components in Generation III+ reactors.

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Staff Approach

- Deliverable is a Notation Vote SECY for EDO signature in December 2017.
- Involve internal and external stakeholders, including NRR, NRO, Regions, Industry, ACRS, and public
- Maintain existing risk thresholds consistent with Commission guidance in SRM-SECY-10-0121.
- Preserve existing ROP structure.
- Provide to the Commission an integrated description of the ROP for new reactors (i.e., AP1000)

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Reactor Oversight Process

- Reactor Oversight Process (ROP)
  The NRC's program to inspect, measure, and assess the safety and security performance of commercial nuclear power plants and to respond to events and any decline in performance

- ROP Objectives
  Risk-informed
  Objective
  Predictable
  Understandable
  Open & transparent

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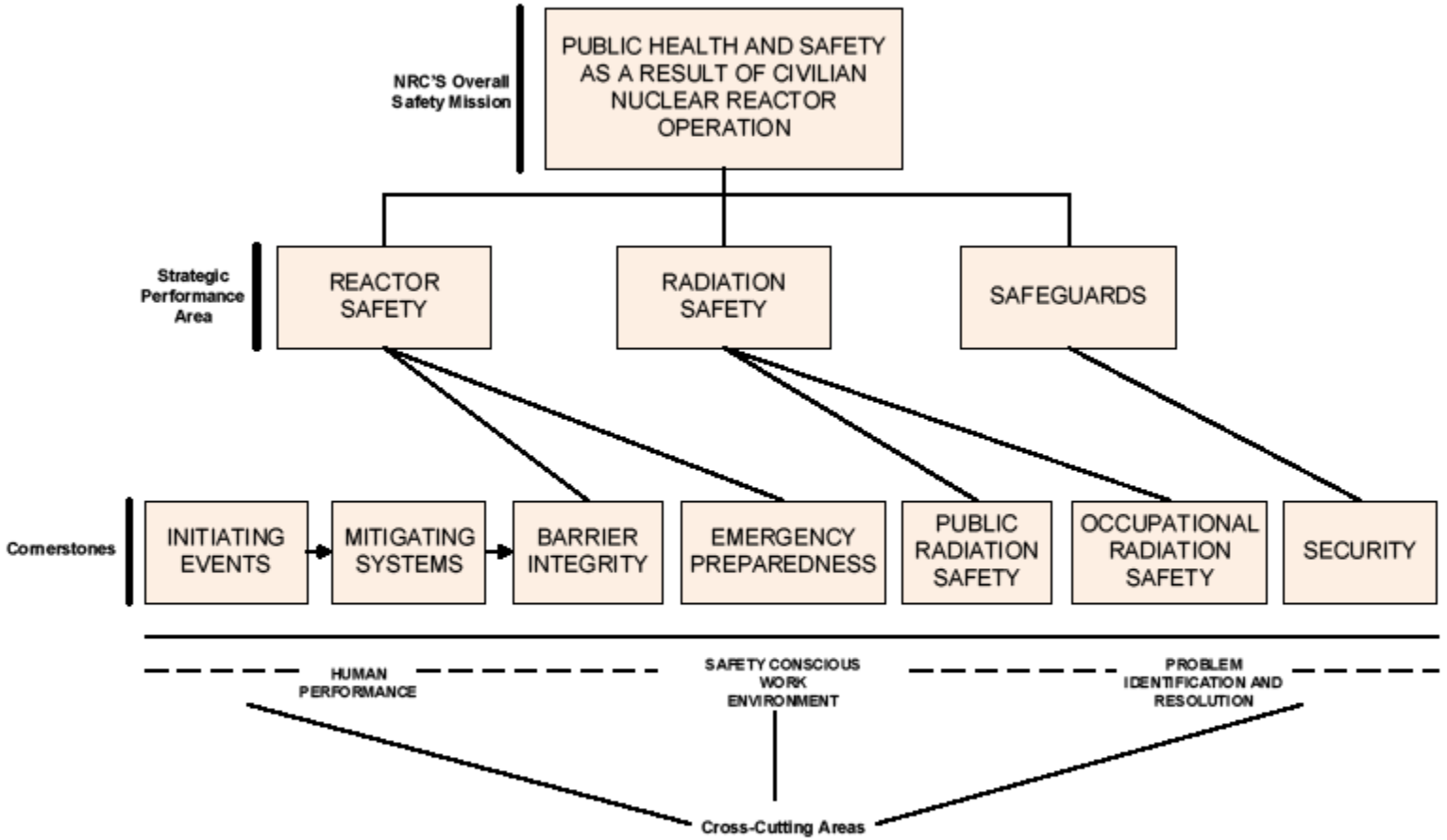
REGULATORY FRAMEWORK



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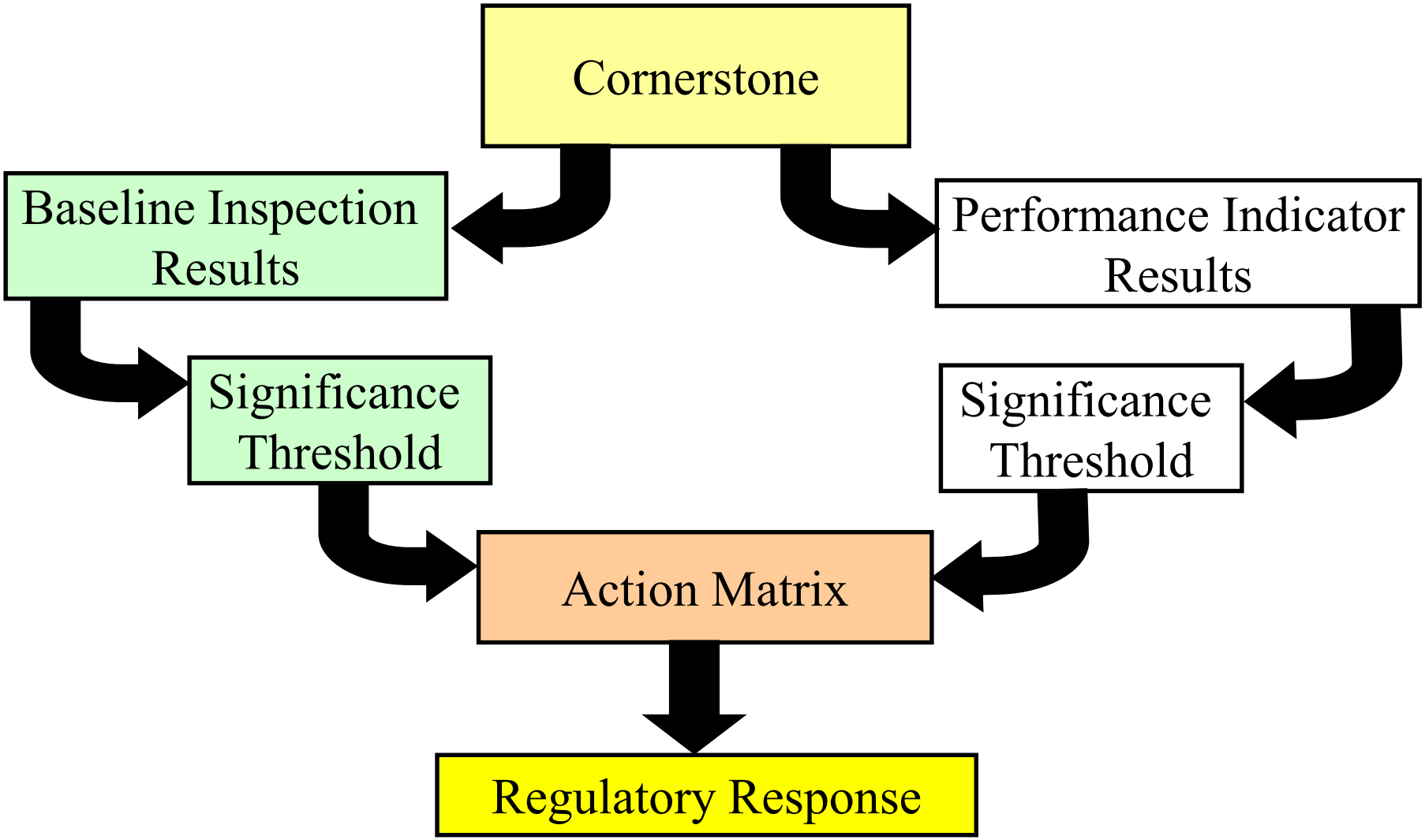

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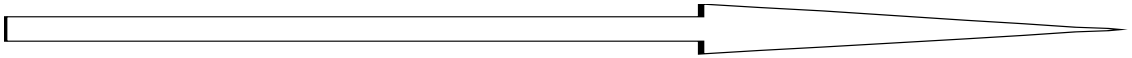
Regulatory Response

Action Matrix Concept

Licensee Response	Regulatory Response	Degraded Performance	Multiple/Repetitive Degraded Cornerstone	Unacceptable Performance

Column 1 Column 2 Column 3 Column 4 Column 5



- Increasing safety significance
- Increasing NRC inspection efforts
- Increasing NRC/licensee management involvement
- Increasing regulatory actions

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ROP Program Areas

- Assessment
- Performance Indicators
- Baseline Inspection
- Significance Determination Process

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Assessment

- The staff is recommending no changes to the assessment program for assessing licensee performance for new reactor designs.

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Appropriateness of Existing Performance Indicators and Thresholds

Ayo Ayegbusi, NRO/DSRA

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Background

- Mitigating Systems Performance Index (MSPI) evaluated in SECY-12-0081, “Risk-Informed Regulatory Framework for New Reactors”
  - MSPI indicators are risk-informed
  - Determined to be ineffective in determining an appropriate regulatory response for active new reactor designs
- Remaining PIs not evaluated in SECY-12-0081
- SRM-SECY-12-0081 directed the staff to provide discussion of the appropriateness of existing performance indicators (PIs) and related thresholds for new reactors

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Background

- Appropriateness of Existing PIs and thresholds evaluated in SECY-13-0137, “Recommendations for Risk-Informing the Reactor Oversight Process for New Reactors”
  - The staff reviewed the basis and related thresholds for the remaining PIs to determine appropriateness for new reactor designs
  - Staff concluded that the PIs remain applicable to new reactors with minor guidance adjustments
  - Unplanned Scrams with Complications indicator would need to be supplemented with additional guidance for new reactor designs to account for passive systems
- SRM-SECY-13-0137 directed the staff to develop appropriate Performance Indicators (PIs) and thresholds for new reactors or develop additional inspection guidance to address identified shortfalls

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Performance Indicator Program

- Provides a broad sample of objective data to assess reactor facilities performance in each cornerstone area
- Along with inspection findings, serve as inputs to ROP assessment process and additional inspection efforts
- Performance indicator data voluntarily collected by reactor facility, reported to NRC on a quarterly basis
- Objective thresholds establish the level of regulatory engagement appropriate to reactor facility performance in each cornerstone area
- Inspection to verify performance indicator data

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PI Performance Bands

Green: performance within an expected performance level where the associated cornerstone objectives are met
White: performance outside an expected range of nominal utility performance but related cornerstone objectives are still being met
Yellow: related cornerstone objectives are being met, but with a minimal reduction in the safety margin
Red: significant reduction in safety margin in the area measured by the performance indicator

Risk-Informed PIs vs. Deterministic PIs

- Many of PIs are not directly risk-informed, but based on regulations and standards that would also apply to new reactor designs
- PIs directly related to risk or risk-informed
  - Mitigating Systems Performance Index (5)
  - Unplanned Scrams per 7,000 Critical Hours
- Remaining PIs and thresholds are more deterministic
  - Thresholds based on industry performance and agreed upon by experts (industry and NRC)

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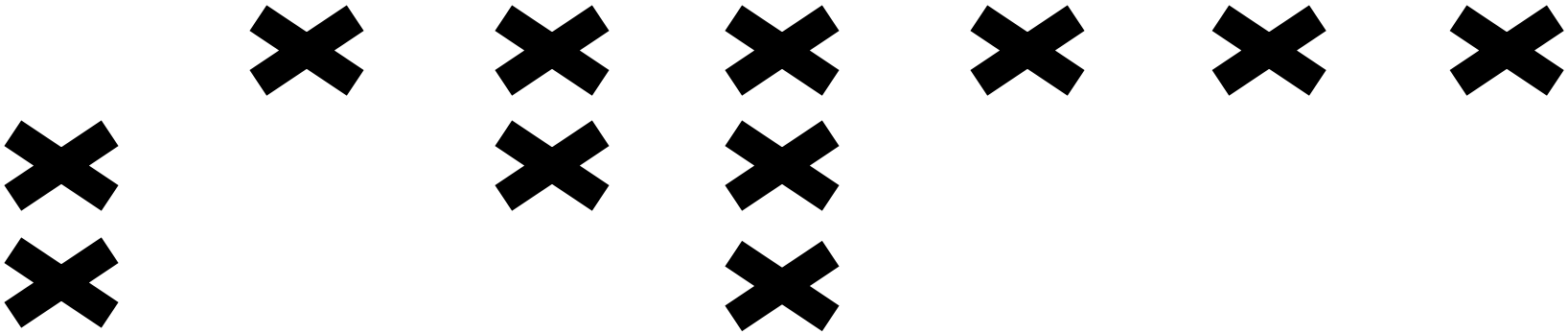
Performance Indicators



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Risk-Informed PIs

- Mitigating Systems Performance Index
  - Measures readiness of systems to perform their safety function (availability and reliability)
    - High Pressure Injection
    - Heat Removal
    - Residual Heat Removal
    - Emergency AC Power
    - Support Cooling Water
- Unplanned Scrams per 7,000 Critical Hours
  - Measures the rate of scrams/year and provides an indication of initiating event frequency
  - Normalized to 7,000 critical hours (80% capacity factor)
  - Impact of plant trips on industry wide plant risk was used to inform the PI thresholds

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White Papers

- Industry White Papers
  - Evaluated all Performance Indicators for applicability to AP1000
  - Determined all PIs except MSPI are applicable with limited changes to the PI program
  - Determined MSPI was inadequate
- NRC White Paper
  - Arrived at similar conclusion with Industry white paper
  - Further evaluated potential to replace MSPI with Safety Valve Unreliability Index
    - This proved to be impracticable due to limited testing and operation of the valves.

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Evaluation of PIs

- Mitigating Systems Performance Index
  - Application evaluated in SECY-12-0081, “Risk-Informed Regulatory Framework for New Reactors”
  - 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
- Unplanned Scrams per 7,000 Critical Hours
  - CDF sensitivity studies conducted to inform initial threshold setting
  - Conservative thresholds set for existing fleet
  - Existing thresholds of performance bound lower risk of new reactors
- Unplanned Scrams with Complications
  - Potential adjustments to the guidance (NEI 99-002) for new reactors

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Conclusions

- Overall, relatively modest adjustments to the program areas of performance indicators (PIs)
- Unplanned Scrams per 7,000 Critical Hours
  - Can be applied to new reactor designs
  - Threshold values are set conservatively and will account for lower risk of new reactors
- Unplanned Scrams with Complications
  - Propose developing new reactor specific guidance in NEI 99-002 for PI reporting guidance
- Mitigating Systems Performance Index
  - Not applicable for AP1000 design
- Remaining PIs can be applied to new reactor designs to determine an appropriate regulatory response

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Baseline Inspection Program Changes

Steve Campbell, NRR/DIRS

Background

- SECY-13-0137, “Recommendations for Risk-Informing the Reactor Oversight Process For New Reactors”, staff recommended developing additional inspection guidance to address identified PI shortfalls to ensure that all cornerstone objectives are adequately met.
  - Commission approved this recommendation (Recommendation 2).

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Background

- Mitigating Systems Performance Indicator (MSPI) monitors availability and reliability of safety systems necessary to mitigate accidents.

- Regulatory Treatment of non-Safety System (RTNSS) are non-safety-related Structures, Systems, and Components that perform risk-significant functions and, therefore, are candidates for regulatory oversight.

- Industry White Paper provided aspects of the ROP for the AP1000 Reactor Design.

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Evaluation

- Gap analysis conducted on 20 Baseline IPs for changes to accommodate inspecting AP1000 design:
  - Few changes are required.
  - Possible adjustments to sample sizes and resource estimates.
  - Possible reference to inspecting RTNSS.
  - Sample range:
    - Adjust sample ranges based on fewer components and lower baseline risk.
    - Based on the risk importance (high (ΔCDF greater than 1E-4) and intermediate (ΔCDF less than 1E-4 but greater than 1E-5)).
    - Based on system classified as RTNSS.
    - Accessibility of component (inside or outside containment).

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Evaluation

- MSPI:
  - Breadth of baseline inspections assess the availability, reliability, and capability of mitigating systems and meets MSPI purpose.
  - Passive systems are expected to be reliable and the existing baseline minimums paired with the lower AP1000 baseline risk is more than adequate to compensate for the MSPI's omission.
  - The staff has determined that additional inspections are not needed to compensate for not using the MSPI indicators for the AP1000 design.

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Evaluation

- Inspection staffing during outages:
  - Safety-related systems, high and intermediate risk important systems, and systems identified as RTNSS become accessible for inspection.
  - An outage inspection team will be formed to support the inspections of those systems.
  - Team augmented with an additional inspector to assist with conducting as-left equipment walkdowns, surveillance testing, post-maintenance testing, and containment closeout.
  - Other team members would conduct outage-related engineering and radiation protection inspections.

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Conclusions

- Gap Analysis completed
  - Few changes to inspection requirements, add guidance as necessary.
  - Anticipate adjustments to sample sizes (risk importance and RTNSS).
- MSPI
  - Not applicable to AP1000 design
  - AP1000 has lower baseline risk.
  - IPs as written are sufficient to assess licensee performance in MS cornerstone without MSPI.
- Inspection Resources
  - Outage team dispatched from Region to cover outage-related operational and program inspections.

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Significance Determination Process (SDP) Changes

Russell Gibbs, NRR/DIRS
See-Meng Wong, NRR/DRA
Matt Leech, NRR/DRA

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Purpose

- Discuss the plans and basis for the development of SDP tools for new reactors (i.e., AP 1000 plants)

- Provide an overview of revisions to IMC 0609 Appendices A, G, H, and M used for operating reactors that can be adapted to reflect the unique design and operational practices of advanced reactor plants

Background

- Commission Staff Requirements Memorandum (SRM) on SECY 13-0137, dated June 30, 2014

  - Develop guidance to address circumstances that are unique to new reactors, e.g., uncertainty of reliability of passive SSCs or other SSCs with limited operational experience.
  - Enhance the SDP by developing a structured qualitative assessment for events or conditions that are not evaluated in the supporting plant risk models, e.g., performance deficiencies associated with passive safety systems, digital I & C, and human performance issues.
  - Continue emphasis on the use of quantitative measures for both operating and new reactors.

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Considerations

- Commission direction to use same safety expectations as those for operating fleet (SRM/SECY-10-0121)
- Gap analysis performed identifying SDPs that need revision to accommodate new reactors
- Existing SDP program robust and inclusive – no new SDPs will be developed
- No deterministic SDPs require revision - design neutral
- SDP's for fire, operator requalification, steam generator tube rupture, Maintenance Rule, B.5.b and mitigating strategies are not affected
- SDPs for at-power, shutdown, containment and use of qualitative criteria need revision

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IMC 0609 Appendix A

At-Power

- Only change to Appendix A will be new screening questions
  - Initiating Events, Mitigating Systems, Barrier Integrity
  - External Events
- Guidance for detailed risk evaluation, or go to IMC 0609 Appendix M

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IMC 0609 Appendix M
Qualitative Measures

- Appendix M should be "a structured qualitative assessment for events or conditions that are not evaluated in the supporting risk models."
  - Separate initiative to provide clarity on use of qualitative factors for operating reactors
- General structure similar to IMC 0609 Appendix M for operating reactors, but enhanced to:
  - Define clear entry conditions
  - Define and assess decision attributes
  - Integrate decision attributes

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IMC 0609 Appendix G
Shutdown

- General approach of existing Appendix G will work for AP1000
- No need for a Phase 2 procedure for AP1000's because generic shutdown SPAR model already exists for AP1000 that the SDP can use
- Procedures including IMC 0308 Attachment 3, Appendix G Basis Document, will be modified as needed to document differences, and how to address AP1000 for shutdown

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IMC 0609 Appendix H
Containment

- General approach of existing Appendix H will work for AP1000 with some modification
- A new plant type for AP1000 in Phase 1 screening and Phase 2 assessment will be added
- IMC 0308 Attachment 3, Appendix H Basis Document, will be modified as needed to document differences, and how to address AP1000 containment issues

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PATH FORWARD

- Reach internal alignment on proposed changes
- Engage Commission on proposed changes (SRM/COMSECY-16-0022)
- Plans for public meetings and “tabletop exercise” workshops
- Planned completion date for SDP tools for new reactors – December 2018

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Conclusions and Recommendations to the Commission

Dan Merzke

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Staff Conclusions

- Maintain existing risk thresholds consistent with Commission guidance in SRM-SECY-10-0121.
- Preserve existing ROP structure.
- No changes to the assessment program are needed.

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Staff Conclusions (cont.)

Performance Indicators

- Many of the PIs are based on regulations and standards that also apply to new reactor designs
- 12 of the 17 current PIs remain valid for the AP1000. The five PIs comprising the MSPI indicators would not be valid.
- The staff has identified no new PIs that would be effective in monitoring performance of the AP1000.
- Maintain existing PI thresholds pending staff evaluation after some operating experience is gained.

Staff Conclusions (cont.)

Inspection Program

- Few changes to inspection requirements, add guidance as necessary.
- Anticipate adjustments to sample sizes (risk importance and RTNSS).
- MSPI
  - IPs as written are sufficient to assess licensee performance in MS cornerstone without MSPI.

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Staff Conclusions (cont.)

Significance Determination Process

- Existing SDP program robust and inclusive – no new SDPs will be developed.
- SDPs for at-power, shutdown, containment, and use of qualitative criteria need some revision.

Staff Recommendations

Recommendation: Commission approves the staff's plans to modify the ROP for new reactors as described.

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Next steps

- Finalize Commission paper based on ACRS and stakeholder feedback
- SECY due to be issued in December 2017