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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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608TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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WEDNESDAY

OCTOBER 2, 2013

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ROCKVILLE, MARYLAND

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The Advisory Committee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 1:30 p.m., J. Sam Armijo, Chairman, presiding.

COMMITTEE MEMBERS:

J. SAM ARMIJO, Chairman

JOHN W. STETKAR, Vice Chairman

HAROLD B. RAY, Member-at-Large

RONALD G. BALLINGER, Member

SANJOY BANERJEE, Member

DENNIS C. BLEY, Member

CHARLES H. BROWN, JR., Member

MICHAEL L. CORRADINI, Member

DANA A. POWERS, Member
JOY REMPE, Member
PETER C. RICCARDELLA, Member
MICHAEL T. RYAN, Member
STEPHEN P. SCHULTZ, Member
GORDON R. SKILLMAN, Member
WILLIAM J. SHACK, Consultant

NRC STAFF PRESENT:
EDWIN M. HACKETT, Executive Director, ACRS
CHRISTOPHER L. BROWN, Designated Federal Official
WEIDONG WANG, Designated Federal Official
RAJ AULUCK, NRR
SUDHAMAY BASU, RES
JEROME BETTLE, NRR
BOB DENNING, NRR
DONALD HELTON, RES
STEVE JONES, NRR
NAGESWARA KARIPINENI, NRR
TIM McGINTY, NRR
DAVID PELTON, NRR
BILL RECKLEY, NRR
FRED SCHOFER, NRR
KEVIN WITT, NRR

ALSO PRESENT:

ROBERT ALVAREZ, Institute for Policy Studies

PHIL AMWAY, Constellation Energy

RANDY BUNT, Southern Nuclear

STEVEN KRAFT, NEI

DIANE CURRAN, HCS&E

JOHN KESSLER, EPRI (via telephone)

GREG KRUEGER, BWROG

EDWIN LYMAN, Union of Concerned Scientists

TOM PARKER, BWROG

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

(1:28 p.m.)

CHAIRMAN ARMIJO: Good afternoon. The Committee will now come to order.

This is the first day of the 608th meeting of the Advisory Committee on Reactor Safeguards. During today's meeting the Committee will consider the following: first, spent fuel study and expedited transfer of spent fuel to dry cask storage; second, development of guidance in support of Order EA-13-109 on reliable hardened containment vents; and, third, preparation of ACRS reports.

The meeting is being conducted in accordance with the provisions of the Federal Advisory Committee Act. Mr. Christopher Brown is the designated federal official for the initial portion of the meeting. He is out right now. Oh, no, he did show up. Thank you, Chris.

Ms. Diane Curran has submitted written comments and requests an opportunity to make an oral statement during this session on the spent fuel and expedited transfer topic. In addition, Mr. John Kessler from Electric Power Research Institute has submitted material for members, information, an EPRI report on impacts associated with transfer of spent nuclear fuel storage pools to dry cask storage after five years of cooling.

There will be a phone bridge line, and so to preclude interruption of the meeting the phone will be placed in the listening mode during the presentations and Committee discussion.

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

At this point, we will move into the Tier 3 analysis of expedited transfer of spent fuel. I am Chairman of that Subcommittee, so I guess I'll turn it over to myself. And we just informed attendees at our Materials, Metallurgy, and Reactor Fuels Subcommittee -- reviewed the regulatory analysis on July 9th and again on September 19th. These were preliminary versions of the analysis.

These two meetings were closed to the public, since at that time the staff was presenting predecisional and official use only information. The regulatory analysis assesses whether any significant safety benefits or detriments would occur from expedited transfer of spent fuel to dry cast storage for the reference plant as modeled, and the potential costs associated with expedited transfer.

Now, the analysis we will be talking about is a generic analysis applicable to I believe most of the plants. There will be additional remarks by Don Helton of the staff, as well as -- in addition to the comments by Ms. Curran and Mr. Kessler, who is on the bridge line.

At this point, I'd like to turn the meeting over to Tim McGinty of the staff for opening remarks and to introduce the speakers. Tim?

MR. McGINTY: Thank you. Good afternoon. My name is Tim McGinty. I am the Director of the Division of Safety Systems in the Office of Nuclear Reactor Regulation.

I would like to thank the Chairman and the members of the ACRS for the opportunity to present the staff's evaluation of the near-term task force two-three action to recommend whether regulatory action will be warranted for spent fuel at power reactors to be transferred from wet to dry storage on an expedited schedule.

To determine whether regulatory action might be warranted, we followed our regulatory decisionmaking procedures to determine whether there is a substantial safety enhancement as well as a cost-benefit analysis to determine whether the benefits of the proposed regulatory action outweigh the costs.

Today Kevin Witt will be covering the Tier 3 plan and objectives. Steve Jones will be covering the Tier 3 analysis. And Fred Schofer will be supporting the discussions on the regulatory analysis.

Our evaluation confirms that both spent fuel pools and dry casks provide adequate protection of public health and safety and the environment. And the likelihood of an accident involving a significant radiological release from spent fuel pools remains extremely small.

After these presentations, Don Helton from the staff will also be presenting perspectives that are the outcomes of the healthy dialogue that the staff often has on important issues. He intends to provide additional emphasis on particular aspects of the regulatory analysis for which he and some other NRC staff feel that the Committee should be made aware of.

With that, I'd like to either turn it over to Kevin to kick off or back to you, Sam. Your choice.

CHAIRMAN ARMIJO: No, go ahead. Kevin, you've got it.

MR. WITT: Thank you. My name is Kevin Witt. I'm a project manager in the Japan Lessons Learned Project Directorate. I'm responsible for coordinating the staff activities on this Tier 3 issue. I'll be going over the background and objectives, and then I'll turn it over to Steve Jones to cover the analysis.

During our briefing for you this afternoon, we are hoping to cover a number of things with you in terms of what we did on this Tier 3 issue. First, we will go over the objective and background for this Tier 3 issue. Then, we will talk about the analysis process that we followed to determine whether regulatory action might be warranted to require the expedited transfer of spent fuel to dry cask storage.

To talk some more about that analysis, we are going to go over the key inputs that we used in that analysis, as well as the assumptions that we made in that analysis. Then, we will talk about the results and other insights that we evaluated during this analysis. Finally, we will talk about the stakeholder feedback and how we address that, and also the next steps.

The objective of our presentation today is to go over this Tier 3 activity on expedited spent fuel transfer and talk about what all we did with this issue and how we came up to a conclusion on it. We are going to talk about how the spent fuel pool study, which the ACRS previously reviewed during their July 9th meeting, talk about how that was used in the analysis as well as a broad history of other spent fuel pool studies that we have done over the years to inform the analysis that we conducted on this Tier 3 issue.

And by the end of the presentation, we hope to gain ACRS endorsement of the analysis that we conducted here. So feel free to ask any questions that you may have during this presentation.

A little bit of background in terms of how we followed -- how we did this analysis. We had to really determine what the proper process was followed to -- or what process we would follow to determine if regulatory action would be warranted. All along during this process we pretty much concluded that spent fuel pool safety is -- adequately protects public health and safety. And all of the research that we've done continues to confirm that conclusion.

So the next step that we went to was to

determine if there is any substantial safety enhancements by requiring the expedited transfer of spent fuel from the pools to the casks. And to determine whether this type of action would be warranted, we followed our established processes for making regulatory decisions as outlined in the regulatory analysis guidelines stated here as NUREG/Brochure-0058.

This is the process that we follow for determining whether regulatory action is warranted on a number of safety issues. And in that guideline we follow the -- we utilize the Commission's safety goal policy statement. And it's really a quantitative mechanism to determine whether a safety improvement would surpass what the Commission expects the staff to follow in terms of adequate protection of public health and safety and improving public health and safety in the operation of nuclear reactors.

And so the first part of that safety goal policy statement is to not have a significant increase in risk to public health and safety. And this can be measured by the likelihood of early fatalities from a potential accident as well as latent cancer fatalities, lifetime chances of getting cancer from the operation of nuclear reactors.

In addition to that, there is a broader guideline that we use in terms of societal risks for the use of nuclear power. And that is reasonable assurance that a large radiological release or a core damage event would not occur in the United States.

And to really determine whether this type of regulatory action would meet those criteria, we utilized information from the history of spent fuel pool studies that we have done over the years, in addition to the recently completed spent fuel pool study that was done by the Office of Research.

Now, when we came up with the process for doing this, we really had a broad perspective in terms of how we would figure out if regulatory action would be warranted. And it was really taking a lot of -- we thought that it would take a really long time to go through all of that evaluation. Our initial project plan had a five-year timeframe on that.

Now, a number of things happened after that plan was sent to the Commission in July of 2012. The first was that the spent fuel pool study was completed or the draft was completed back in July of this year. And so we wanted to make sure that the spent fuel pool study could be utilized, and the information could be analyzed in our regulatory framework to determine if this type of action could meet the criteria, as well as the waste confidence generic environmental impact statement.

We figured that the information that we were doing here, the public could really benefit from understanding the processes that we used in this analysis to inform -- to participate in the waste confidence rulemaking proceedings.

To help out with all of this, we've tried to make all of the information that we have had publicly available as quickly as possible. And we did release a draft analysis to the public, and we also provided that to you before this meeting, in terms of what our analysis found on this issue.

Now, when I mentioned the waste confidence, the staff has concluded that there is no impact from what we did here on the conclusions made in the waste confidence decision. And what we are really trying to do is making sure that the public is aware of all of these issues, so that they can participate in both this and the waste confidence proceedings.

MEMBER CORRADINI: If I might just ask, since I -- that last part I don't completely understand. Going into this, there was a chance that it would affect the waste confidence? I don't see the connection.

MR. WITT: Well, the waste confidence is really relying on a broad history of spent fuel pool studies, kind of similar to what we have utilized for the Tier 3 analysis.

MEMBER CORRADINI: Okay.

MR. WITT: And so they have gone ahead and moved forward with what they were doing. Now, subsequent to the work that they were doing, we initiated this work. And so there was a question about whether the work that we would do would impact the waste confidence.

Now, we have been in communication -- we have worked very closely with the waste confidence staff in terms of what they are doing, and the conclusion has been made that there is no impact from what we are doing and what waste confidence is doing.

MEMBER CORRADINI: Okay. Okay. I guess I'm -- okay. I guess that makes sense to me. But if I could just say it differently, whether the spent fuel sits in a dry storage or wet storage, going into it I'm not sure how that would affect the waste confidence. Am I missing something?

MR. WITT: Well, I think --

MR. JONES: This is Steve Jones in NRR. I'd just mention that I guess the spent fuel -- the generic environmental impact statement that goes with waste confidence did consider the older spent fuel pool studies and looking at those types of events in pool storage. And just to make sure that the spent fuel pool study and other analysis we perform as part of Tier 3 did not change any of the assumptions drawn from those older studies, was important to support the waste confidence going forward.

MEMBER CORRADINI: Okay. Fine. Thank you.

MR. WITT: So to give a little better sense of how all of these issues have an effect on each other, this chart shows the different levels that we went through in doing this analysis. The first level that you see there is the spent fuel pool study, which has been released in draft form. And that was a study done on a specific plant for a specific scenario. There was a seismic -- beyond the design basis seismic event for a BWR Mark I type reactor.

So following that analysis that we did, or the study that we did on the spent fuel pool, we added in a regulatory analysis to that study. That's Appendix D of the spent fuel pool study. So we utilized that information from the study, and put it into our regulatory framework to give an idea of where this information would come out in terms of meeting the criteria for regulatory actions.

And we have also included an expanded set of scenarios in that analysis that was part of the spent fuel pool study. Steve will talk about that a little bit more in terms of the expanded set. And then, finally, what we did on this Tier 3 is we took that information from the spent fuel pool study, primarily Appendix D, which was done on that specific plant, and we expanded that out to all of the plants.

And so that's why we're calling this a generic analysis is because it's really the analysis for all plants, whereas the spent fuel pool study was done for one plant.

CHAIRMAN ARMIJO: Okay. Steve, I just -- you've got to answer a question that I was confused in reading the document. And the question is, are the plants, west coast plants that are not central and eastern United States, are they or are they not covered by this generic analysis?

MR. WITT: Well, in our draft paper, we do have a commitment in there to go back and look at the western plants. They are not specifically considered in the analysis right now, but we still have confidence that they are safe. But we do commit to getting back to that, primarily because the seismic information is not --

CHAIRMAN ARMIJO: So it's being updated.

MR. WITT: Right.

CHAIRMAN ARMIJO: Okay.

MR. WITT: It's not of the same level that --

CHAIRMAN ARMIJO: Okay. So at this point, there is a certain number of plants, specific plants, that are not yet covered by this analysis. But you will analyze them in a similar way or some other way when you get the updated seismic data.

MR. WITT: Correct.

CHAIRMAN ARMIJO: Okay.

MR. WITT: Yeah. There's three plants, I believe. Three or four.

CHAIRMAN ARMIJO: Is Washington, the California plants --

MR. JONES: Yeah. There's Columbia in Washington, and then the Diablo Canyon in California, and Palo Verde in Arizona, are really the affected plants that are still operating.

CHAIRMAN ARMIJO: Okay. Thank you.

MR. WITT: Okay. And now I'll turn it over to Steve.

MR. JONES: Good afternoon. I'm --

MEMBER POWERS: It's kind of interesting that the problem is the seismic -- the western United States isn't up to snuff.

MR. JONES: That's not correct. It's up to the same --

MEMBER RAY: It's not covered by central and western. It's its own unique analysis. There are those who would argue that it's more current and more precise --

MEMBER POWERS: Far more current.

MEMBER RAY: -- than central and eastern. But central and eastern is covered by the generic information that is readily available. That's really what's true. It's not that it's out of date in the west. That's another debate we can have.

MR. JONES: Okay. Good afternoon. My name is Steve Jones. I'm responsible for spent fuel storage and handling in the Office of Nuclear Reactor Regulation. I'd just first like to revisit the spent fuel pool study results.

That study updated public consequence estimates for specifically beyond design basis seismic event affecting a spent fuel pool. And I looked at both high and low density conditions in the existing racks. The low density was defined as removal of all fuel with more than five years' decay. And then also looked at conditions both with and without deployment of the existing mitigation equipment that is required under post-9/11 actions that the agency implemented.

That study determined that the frequency of the release is really independent of the density of the fuel storage for those two conditions. It is driven by the hot fuel assembly -- you know, the decay heat of the most recently discharged fuel, and that is roughly equivalent since that is the heat that drives some portion of the fuel to reach the ignition temperature required for the Zircaloy oxidation.

And then it was -- once the oxidation occurs, though, the heat from the oxidation really carries the reaction forward. And so that drives the magnitude of the release, which is the last bullet there. Many scenarios result in no release, but in some cases if oxidation begins in the most recently discharged fuel the heat can propagate and affect other assemblies.

The study, together with previous research, confirmed that spent fuel pools adequately protect public health and safety. And the regulatory analysis that was provided in Appendix D of the study showed that faster spent fuel pool transfer does not substantially enhance safety.

Next slide, please.

Okay. To expand that to a generic level, the staff built on the regulatory analysis in Appendix D of the spent fuel pool study. That study -- the studies of Appendix D analysis already included cask drop and loss of power and loss of coolant inventory, in addition to a larger seismic event as initiating events.

And then, the Tier 3 analysis, more generic analysis, expanded that to all spent fuel pools, including pressurized water reactors and boiling water reactors with Mark III containments, as well as considering the plants with the combined operating licenses right now, which are both -- are all AP1000 units.

I want to note early on that the assessment of security events has been handled separately, and that resulted in numerous regulatory changes with respect to Part 50 licenses. In particular, 10 CFR 50.54(hh) imposed the license condition for mitigating strategies that helped deal with damage to the spent fuel pool.

Also, the effect of these security changes was modeled to some extent in the regulatory baseline and considered in the alternative.

Next, please.

MEMBER BANERJEE: Excuse me. You said you took into account the spent fuel pool with and without mitigatory measures. Did you do with and without mitigatory measures for the cask, any accidents that might --

MR. JONES: No. The scope of this study was focused on the pools themselves, because right now we are I guess neglecting whatever additional risks may be associated with the actual transfer of the fuel, the dry storage, and the placement of the fuel in dry storage on the pad outside.

MEMBER CORRADINI: And to follow up Sanjoy, I guess he was thinking what I was thinking, so I would think there would be compensating risks. There was not even an estimate of what those compensating risks are?

MR. WITT: That is correct.

MEMBER CORRADINI: By definition or just by judgment. Were you instructed not to worry about it, or did you decide it's just outside the scope?

MR. SCHOFER: This is Fred Schofer. It was a simplifying assumption. We were focusing primarily on the expedited transfer from high density to low density. And I fundamentally assumed that the risk for moving the cask to the S phase where the casks are stored, as well as any risk associated with the placement and storage of those dry casks above ground, the risk was zero.

MEMBER CORRADINI: Okay.

MR. SCHOFER: And, therefore, I was maximizing the delta between the two alternatives.

MEMBER BANERJEE: But you also took into account any risks associated with the transfer, right?

MR. SCHOFER: I only considered cask drop initiating events. I didn't include any incremental risk associated with moving any additional casks to the S phase.

MR. WITT: But the cask drop was based on a previous -- that probability was based on a previous PRA done on spent fuel pools, that we understood the probability of a cask drop event to be a certain amount just based on a normal operational basis.

So we didn't look at what the increase in that cask drop would be if we did expedited transfer.

MR. SCHOFER: Correct.

MR. WITT: That would just cut into the benefits that you would get out of the expedited transfer, since there would be additional risk.

MEMBER BANERJEE: So you looked at -- if I understand it, of course, if you leave everything where it is, you've got a pretty good idea of what can happen. If you move the stuff into the casks, you looked at what could happen related to the pool in terms of reducing consequences and risks, and so on, but anything which is associated with the cask you've taken to be zero.

MR. SCHOFER: I assumed was zero, just for this simplifying assumption for this --

MEMBER BANERJEE: To maximize the benefits.

MR. SCHOFER: To maximize the benefits.

MEMBER BANERJEE: But, in fact, there will be risks associated with --

MR. SCHOFER: There would be some --

MEMBER BANERJEE: -- so the benefits --

MR. SCHOFER: -- health risk.

MEMBER BANERJEE: -- you overestimated the benefits.

MR. SCHOFER: Yes.

MEMBER BANERJEE: I think we should be

clear on that.

MR. WITT: Yeah. The analysis that we did, we tried to put in as many words as possible to say we tried to do a conservative analysis here. And in some cases we did a bounding analysis as well. So we are really trying to maximize the --

MEMBER BANERJEE: So if I understand your point of view, which I don't know if it's correct or not -- correct me -- you are trying to make the strongest possible case for the transfer that you can. Is that correct?

MR. JONES: Yes. And that shows up I guess with the last bullet there, how we modeled the effect of the security changes, for example, in the regulatory baseline or the high density storage case, which we'll get to. We assumed no -- gave no credit for any mitigative actions that the plant staff may carry out.

In the alternative, the low density storage case, we did credit with a high level of effectiveness, the implementation and mitigation, so --

MEMBER BANERJEE: I don't understand the logic. I mean, you are trying to make sort of a judgment, which is important I think, but why would you give one potential way of doing things all these advantages? How can we get a clear picture from that?

MR. WITT: Maybe I can answer. Part of the original plan, when I said it was a five-year plan, that was originally intended to include all of these considerations. We were going to look at what the additional risk would be. We had -- you know, there's going to be changes in the regulatory framework eventually down the road. So we were going to try to include all of that.

But right now what we were doing is trying to get this information out to the public as quickly as possible just to do as best as we could and analyze in a conservative manner whether this would pass the criteria for regulatory action.

CHAIRMAN ARMIJO: Well, I can see some logic to your decision in that this fuel would eventually have to be taken out of the pools and stored. So there is -- the casks would have to be lifted out. So over time you'd have the same number of casks moved and everything else.

So there is -- the incremental risk is the fact that it's being done on an expedited basis, which -- as opposed to a -- but there is -- so, you know, overall same number of fuel rods have to be put in casks and put out on the pad, whether it's done over a five-year period or whether it's done over the life of the plant.

So you could argue that, yeah, it's about the same. But if you're trying to do it in a rush to meet an arbitrary schedule, then you are going to add risk. It's hard to measure. I don't know how you do that, but it's real.

MR. RECKLEY: This is Bill Reckley. Just to elaborate -- NRR. Just to elaborate a little bit on what Kevin was saying, in our phased approach here, the second phase, if we had determined that there was a potential that there was a significant safety benefit for moving the fuel, or if it was indeterminate, the next phase is additional study.

This was not intended to be a definitive analysis in this phase to say expedite or not to expedite fuel. This was the preliminary phase to see, should we go to the second phase, which is additional study? And so in that regard that's why we were very conservative, or we tried to be, to say we will -- where there's a doubt we'll maximize the benefit of expediting the transfer.

And then if it turned out, again, to be indeterminant, then we would address that in the second phase. And the second phase would have considered the risks of the transfers itself, the risks of dry cask storage, reevaluation of some of the assumptions that Steve is going to lay out that we used in this preliminary assessment, which, again, was just to determine whether we should do additional studies.

MEMBER BANERJEE: I think that answers my question. That's a good answer, actually. Clear. Okay.

CHAIRMAN ARMIJO: Go ahead, Steve.

MR. JONES: Okay. I'll go back to the groupings. The regulatory analysis was divided into several groups in order -- based on the characteristics of the individual units associated with it and their spent fuel pools.

The first group is BWRs with Mark Is and IIs. They all have elevated spent fuel pools, and that constituted that group. The second group is PWRs and BWRs with Mark III containments, with the spent fuel pools at or near grade, with at least one exposed side, and, therefore, that pool could leak at a relatively high rate.

And in both cases we are excluding the western reactors, as we discussed before, for the reason that we don't have seismic hazard data at the same level that we have for the other units.

And then the third group is the combined operating license holders, as I mentioned, the V.C. Summer Units 2 and 3, and Vogtle Units 3 and 4. The fourth group is PWRs with shared spent fuel pools. There are several of those units or several of those sites in the United States.

CHAIRMAN ARMIJO: How many plants do you have in each group? For example, the BWRs.

MR. JONES: I'll have to turn to Fred on some of these. BWRs with Mark Is and IIs, I believe it's 31 units.

CHAIRMAN ARMIJO: Okay.

MR. JONES: PWRs and BWRs with Mark III containments, I don't know, I think we're in the forties there somewhere.

CHAIRMAN ARMIJO: Okay. And that's shared --

MR. JONES: Okay. Forty-eight units in Group 2, the four AP1000 units I just mentioned in Group 3, and 21 reactors sharing 11 spent fuel pools for Group 4. And, you know, there are some plants that have somewhat separated pools. It's kind of hard to make a distinction at what threshold we say the pools are shared, but wherever the license capacity was based on a dual unit, basically that's where we considered that plant has a shared pool.

CHAIRMAN ARMIJO: So there must be one set of plants, three plants sharing one pool. Is there such a thing? Because 11 times two is 22.

MR. JONES: That's what I'm trying to say is there are -- there is an extra pool thrown in there. I'm not sure which unit that --

CHAIRMAN ARMIJO: Is that the Morris?

(Simultaneous speaking.)

There is a mystery pool there someplace.

MR. JONES: I mean, these are the ones that are in each group. That's --

CHAIRMAN ARMIJO: Okay.

MR. JONES: Is that something that maybe we could get back to you on?

CHAIRMAN ARMIJO: Yes. You can get back to us. Just a little bit of a puzzle for me that -- just keep going.

MEMBER CORRADINI: What page of the COMSECY were you pointing to, so we can look up?

MR. SCHOFER: It was Table 1 of the enclosure.

MEMBER CORRADINI: Thank you.

MR. JONES: In addition, there are a select number of sites that have the spent fuel pools located below grade with backfill on all four sides. These plants we don't consider credible for any significant leakage, and we excluded them from the regulatory analysis.

Also, we have several spent fuel pools at decommissioned plants where we have excluded them based on the low decay heat and the -- you know, the information from the spent fuel pool study indicating that after a certain point it is unlikely to reach conditions that would support oxidation.

CHAIRMAN ARMIJO: So they'd be air-coolable or something -- some argument like that or --

MR. JONES: Well, they'd have a much higher likelihood of being air-coolable. I guess there is no absolute assurance depending on the end configuration. But at a minimum, there would be a very long time to respond and apply mitigative strategies.

CHAIRMAN ARMIJO: Okay.

MR. JONES: Next slide, please. Okay. Got it. These are the two alternatives we considered. The regulatory baseline -- I have discussed some of this before, but it involves implementation of the strategies required by the license condition under 10 CFR 50.54(hh).

Part of that involves deployment of the fuel in a distributed pattern with hotter fuel surrounded by at least four colder assemblies.

MEMBER POWERS: And licensees do that?

MR. JONES: Yes. And there's some information from the spent fuel pool study about that particular unit. They actually go beyond one by four. It's one by eight in a repeating pattern.

MEMBER BANERJEE: So if you have this pattern, there has to be a continuum of sort of decay heat levels. So do they arrange it so that the hot bundles are surrounded by the next hotter, and then the next -- how do they do this? Is there a pattern?

MR. JONES: I don't think -- there is really no requirement for a continuum as far as that goes. There is some limit on the storage capacity in some plants, where exactly fuel can be stored based on its reactivity level. But the hottest assemblies, and the most recently discharged assemblies, have dramatically higher heat loads. And for each one of those it is basically a chess move, the knight move, between assemblies if you have a repeating one by four pattern.

MEMBER BANERJEE: Okay. So it's -- there is some -- because I guess it's like a two-year delay or something or a year and a half or something.

MR. JONES: Right. At least a year and a half, in many cases two years, between refueling. So there would be substantial decay for that, even for the next hottest assembly.

MEMBER BANERJEE: And when you said that you took into account the possible propagation from, say, what is happening in the hot assembly in cases like cooling or, you know, commencement of oxidation to the surrounding assemblies, was this a sort of radiation heat transfer calculation that was done, or how was this done?

MR. JONES: Well, really, for our purposes we used largely bounding assessments as far as looking at 100 percent non-coolable configurations, and then assuming a release fraction for all of the fuel in the pool. So it's very much bounding.

MEMBER BANERJEE: But that could be extremely conservative.

MR. JONES: Right. It is.

MEMBER BANERJEE: My concern --

MR. JONES: Compared to the spent fuel pool.

MEMBER BANERJEE: Mike is nodding his head there, because I don't necessarily see that happening.

MR. JONES: That's correct. The analysis in the spent fuel pool study, again, is the only really detailed analysis using state-of-the-art codes that looks at the propagation. And in that case, only under certain very rare conditions do you get that type of propagation between assemblies.

MEMBER BANERJEE: So let me understand what you did here. So suppose you get uncovery and you get partial uncovery, so that you actually get into sort of a rapid oxidation state. If it's -- the hot assembly is surrounded by the four cooler assemblies, do you assume that those four cooler assemblies will also go into a rapid oxidation state? Do you just make a probability of one that this will happen?

MR. JONES: For the high estimate and also -- we will get into the exact numbers, but, yes, for both -- really, the high and for every time that we're assuming that the pool is actually damaged for even the base case, except for one -- with one exception the actual case that was examined in the spent fuel pool study, where we have more detailed information.

MEMBER BANERJEE: And what did that more

detailed information indicate about the propagation? I mean, you did this in more detail with appropriate radiation codes, and so on. What happens there?

MR. JONES: Well, there is a difference I guess in the end state. We're not -- for that spent fuel pool study, the structural examination of the pool determined that it would only fail at the bottom corners.

Therefore, most of the events involved at the end of the analysis period a complete draindown. Some of them transitioned through, over a long period of time, a partial drain condition where the fuel would heat up, begin a steam oxidation, and then, once the bottom plate cleared, then oxygen could be admitted, and that would cause an air oxidation state to develop. And those cases did propagate to some of the surrounding assemblies because of the --

MEMBER BANERJEE: But due to the air oxidation.

MR. JONES: Right. The very -- the amount of heat released by the air oxidation would radiate --

MEMBER BANERJEE: It would be just a partial draindown. For example, if it wasn't right at the bottom, it's very unlikely that anything would happen to the --

MR. JONES: Right. It's a very complicated scenario because you do need --

MEMBER BANERJEE: You need a very slow draindown. Otherwise --

MR. JONES: Right.

MEMBER BANERJEE: -- the air cooling would simply not -- I mean, I'm just trying to understand how many impossibilities or improbabilities are --

MR. JONES: There are quite a few.

MR. WITT: If I could just add something. Also, I think this is one of the areas where we've utilized our history of doing studies on spent fuel pools. There has been a number of studies done. NUREG-1353 was a generic issue on spent fuel pools. NUREG-1738 was on a decommissioning reactor spent fuel pool. And I think what those studies show -- kind of inform what we have in the tables here in terms of the release fractions. And I think that's what you're trying to get to is how much of that fuel would --

MR. JONES: I guess what we did is we followed this -- a lot of the same conservatisms that were used in those earlier studies is what we were seeing. I did want to mention that, you know, it's a complex phenomenon as far as the oxidation developing, because -- and it's like any fire. You need fuel, you need a source of heat, you need to reach a certain activation temperature, and then you need oxygen to carry the fire forward.

MEMBER BANERJEE: Or steam.

MR. JONES: Well, yeah, steam works for a while, but it's a much slower -- it doesn't release as much heat --

MEMBER BANERJEE: Sure.

MR. JONES: -- per reaction. You are releasing hydrogen, though, which is a problem. But it's really when the oxygen can get into the fuel that it's -- I guess the oxidation rate obviously would go way up at that point.

That way, when we're looking at partial drain, we are being very conservative because the steam would tend to keep the air out for a long period of time, or maybe indefinitely. If it wasn't producing as much steam, then it would be a very slowly evolving event because you have low decay heat and you're not -- that's why you're not generating the steam. So there is a lot of conservatism there in terms of --

CHAIRMAN ARMIJO: Steve, you mentioned -- you know, we've put in orders, two orders, EA-12-051 and 12-049, Fukushima action items for improved spent fuel pool cooling, power supplies, monitoring of pool levels, and things like that. And you don't credit that at all in this for the regulatory baseline case, yet you appear to credit it for the expedited transfer case. Is that -- am I reading that right?

MR. JONES: Right. I guess what we're looking at predominantly is the spray capability, and that dates back to the 9/11 guidelines that were issued, post-9/11 guidelines. So, but that's correct. We only applied the -- again, to maximize the benefit, we only applied the mitigation to the low density case where we already have a lot lower likelihood of --

CHAIRMAN ARMIJO: Yeah.

MR. JONES: -- an event leading to oxidation.

MR. JONES: One important point on that is that the licensees haven't implemented those orders yet. They are in the process of doing that. And so it would be inappropriate for us to give credit for something that hasn't been implemented yet.

CHAIRMAN ARMIJO: Yeah. I understand that. But we're talking about a long-term issue here.

MR. JONES: Right.

CHAIRMAN ARMIJO: And I'm sure we'll --

MR. JONES: Well, I think the way we have treated it, it wouldn't matter so much --

CHAIRMAN ARMIJO: My issue is, why apply it to either one. If you're not going to apply it -- if you're going to apply it to the five-year expedited case, you should apply it to the regulatory baseline case or to neither. And at least it's kind of even-handed. You know, if they're not installed -- anyway, go ahead. I'd put that on my conservative side list of --

MR. JONES: Yes, it is. It is another conservative --

CHAIRMAN ARMIJO: Okay. Well --

MR. SCHOFER: This is Fred Schofer. I just want to clarify something. When I do reg analysis, I do it based upon all orders, all regulations that are in place currently, if it's implemented or not.

So with regard to FLEX equipment, what you're talking about, because they didn't have details with regard to how it would be implemented, as well as human performance analyses to determine the likelihood of it being effective, I addressed it qualitatively as other consideration, did not quantify it, but it is in the analysis.

Primarily, the difference between the

Alternative 1, where I took no credit for mitigation, and Alternative 2, which is the low density case, I took credit for the B5B equipment, which was the equipment that Steve was just talking about that was put in place following 9/11. That was primarily the makeup capability to the pool, as well as the spray nozzles to spray the fuel.

CHAIRMAN ARMIJO: So in neither case were the orders, the equipment and strategies that come with the others, credited to either case.

MR. SCHOFER: It was credited qualitatively, not in the numbers.

MR. WITT: Well, I think one other consideration there is that I think we are of the understanding that the orders are not going to increase the capacities that the spray or fill will have on the spent fuel pools. Is that correct, sir?

MR. SCHOFER: But you'll have more equipment --

MR. WITT: It's still under review, so --

MR. SCHOFER: You have more equipment, you have more power availability, whether it be, you know, gas, diesel, whatever, to drive motors or drive pumps. So the likelihood of having a greater ability to mitigate --

CHAIRMAN ARMIJO: It's going to be there, or why would you order them to do something you want done?

MEMBER SCHULTZ: It's also appropriate, if you -- we just referred back to studies that had been done in the past as some justification for the conservative assumptions we're using now. I think we need to put those in their place in terms of, what was the purpose of those evaluations as compared to something like the spent fuel pool study, which was a more thorough analysis, and its way of demonstrating the expected -- expected results of spent fuel pool accident.

That study was informed by these past studies. Those past studies were not informed by the spent fuel pool study. They couldn't have been, obviously. And so I think we ought to be focusing in terms of developing and recognizing the conservatisms that are in this analysis. We ought to be comparing it to the spent fuel pool study, and we see dramatic differences between these assumptions and the spent fuel pool study.

And that just demonstrates, as we said earlier, that the results are very conservative. And if they do demonstrate that the fuel should not be moved, then it's a clear demonstration that further study is not warranted.

CHAIRMAN ARMIJO: Go ahead, Dana.

MEMBER POWERS: Let me ask a couple of questions about a couple of comments you made in passing. You said that if there was steam it would generate hydrogen, and then kind of shrugged your shoulders. Are the containment volumes for the pools sufficiently leaky that you don't have to worry about the hydrogen?

MR. JONES: I guess the spent fuel pool study did address that for a BWR. It concluded that the leakage of the boiling water, secondary containment, is relatively leak-tight, and, therefore, it would not go away. For pressurized water reactors, that again may be -- it may be somewhat conservative to assume that, because there are -- they tend to be located very close to large truck bays or other large openings, although they are -- they were previously required to have controlled ventilation systems that could draw a vacuum to control any releases from the area. So, again, they are somewhat tight.

MEMBER POWERS: Steve, the other comment you said was one that I found a little bit remarkable. You said in the steam production, that would keep the air out. How do you know that, that it would seem to be a function of how rapidly steam was generated and what the geometry exactly was? Did you look at that in close detail?

MR. JONES: No. I haven't done any detailed analysis. We have a little bit of information from Research, just some preliminary analyses looking at partial drain conditions.

MEMBER POWERS: It seems to me --

MR. JONES: I'm talking I guess about, you know, when you're at the higher heat rates, the rate of steam production would be fast enough that it would make it difficult for air to penetrate.

MEMBER POWERS: It's a very difficult problem, because it depends on entrainment. It depends on configuration fairly dramatic or -- and ultimately you are in an unstable environment, because the steam is inherently lighter than the gas that is flowing into.

So you get into a Raley-Taylor instability problem, which -- and that instability is going to try to drive the oxygen down. And so, I mean, it's not transparently obvious to me that steam would exclude the gas. I have looked at it somewhat carefully for fuel in a reactor vessel with the head off where you have enough decay heat that you do create enough steam which probably could keep the air out. But it's not obvious to me that's the case for a spent fuel pool.

MR. JONES: Well, I guess what we are considering specifically is a high density storage rack that has closed cell walls with a single fuel assembly in that. And if you're generating -- if that assembly is hot enough that it's boiling the water in a partial draining condition near the bottom of the pool, and that steam is exiting that cell, that's really what we're looking at.

CHAIRMAN ARMIJO: Particularly if it's channeled.

MEMBER POWERS: Well, but that's -- that seems to me to be designed to entrain air.

CHAIRMAN ARMIJO: Yes.

MEMBER POWERS: I would think that that particular configuration would be unavoidable, that you would entrain air every time in that one. The one where you get a partial draindown, that one is not so obvious to me, but a low draindown I think you're doomed. That's going to get the air in through the --

MR. JONES: Okay. I think we're getting into really one of the reasons why we made a conservative assumption to look at just the fuel is damaged in these types of cases, because it would be a very complex scenario to evaluate requiring computational fluid dynamics to look at the flow paths for air and steam in that environment. And I don't even know if it's --

MEMBER POWERS: Well, it has an impact also on what you assume the radionuclide release is to that -- those releases, in a steam environment, are different than what you would expect there.

MR. JONES: Okay.

MEMBER POWERS: One with a higher oxygen -- higher oxygen.

MR. JONES: Okay. Move on to the next slide.

These are just the detailed inputs that were used in the regulatory analysis. I'll just highlight some factors on this slide. I don't want to go through every element. What you see on the top line are the seismic hazards.

For all of the plants, we used Peach Bottom as a roughly greater than median hazard for the low and base cases. For the high estimate, we used the highest seismic hazard for the plants within the group. In the case of Group 1, that was the Limerick site. And we'll go over Groups 2 through 4 a little bit on the next slide.

Down a little bit as far as the liner fragility, the top line is using data from the spent fuel pool study, since it's applicable to boiling water reactors. It's the best data we have for that.

For the best case and the high estimate, we are looking at bounding conditions on fragility. Basically, any of these initiating events that occur, the liner leaks at a rate beyond the makeup capability for the plant and it eventually drains.

And then for the -- insufficient natural circulation is addressing that ability to air cool the fuel under that condition, whether it's a partial drain or a complete drain.

With respect to the Bin 4 earthquakes and catastrophes that haven't been evaluated by the spent fuel pool study, we are looking at 100 percent probability for insufficient natural circulation, i.e. the fuel is hot enough to begin Zircaloy oxidation. And that really bounds all of the concerns you might have with different configurations in the plant, whether it's the fuel hasn't been dispersed into the required pattern or the pool only partially drains.

And it also addresses the -- whatever benefit might evolve from an open frame rack design that would allow horizontal airflow within the pool.

CHAIRMAN ARMIJO: Steve, now the 100 percent bounding case on fragility, is that the case that leads to what the staff calls a moderate leak in the spent fuel study? Or is it the smallest --

MR. JONES: It's just there is a leak somewhere. It is really kind of undefined because the way we are treating it is if --

CHAIRMAN ARMIJO: Yeah. You're saying --

MR. JONES: -- so it --

CHAIRMAN ARMIJO: -- it's not -- there is no issue about the rate of draining or anything else. It just --

MR. JONES: Right. For the Bin 4 seismic events, I guess we're considering that leak could be anywhere in the pool. It could be a partial drain condition, for example.

CHAIRMAN ARMIJO: Because you are going to --

MR. SCHOFER: Fred Schofer. If you take the two together, liner fragility and insufficient natural circulation, because it's 100 percent it means you are losing inventory and it may only partially drain, because you don't have coolability.

MR. JONES: I think what you're seeing

here, really, is the base case is just relooking at the spent fuel pool study results, because it's the BWRs. Every other case for base and high estimate, we really have a bounding progression to fuel damage and oxidation. And then it's the release fractions where you see a difference between the base and high estimates there.

What we're talking about in release fraction is just how energetically the fuel is oxidizing, how much it affects adjacent fuel assemblies, and then also the integrity of the building, if there is any hold-up, to what extent there is hold-up of aerosols or any release from the pool.

For the base case, you see we are using 40 percent for the high density case. That is representative of one of the worst results in the spent fuel pool study. Ninety percent we really consider to be basically bounding to get that kind of release from the pool and the high estimate.

And Alternative 2 is really looking at -- is looking at the low density case, and, again, draws heavily on the spent fuel pool study.

CHAIRMAN ARMIJO: Steve, just to refresh my memory, for the 40 percent base case for

Alternative 1, was that the situation in this early -- in this OCP phase where you had pretty much the peak decay heat?

MR. JONES: No. The 40 percent was --

CHAIRMAN ARMIJO: What case was that?

MR. JONES: It was the small leak case for OCP 3 where you go transition through a steam oxidation phase. The spent fuel pool study modeled the hydrogen exposure, and then an air oxidation phase.

CHAIRMAN ARMIJO: Okay.

MR. JONES: And that was really the worst analysis, and that was very unique circumstances I think to develop that scenario. You know, precisely a slow enough leak that it allows steam oxidation to develop.

CHAIRMAN ARMIJO: Well, that's important, though. Could you just give me the page in your report where that is discussed in more detail? Later. You don't have to do it right now, but I just --

MEMBER BALLINGER: I want to go back to this fragility thing. It is true that we -- that the earthquake at Fukushima had much larger ground motions. There is no indication that the liner failed. In fact, we know it didn't fail.

MR. JONES: Right.

MEMBER BALLINGER: But in this case, you assume that it did fail. For the spent fuel study, you had to -- you assumed that you had -- that's the only way you get a leak is for the liner to fail.

MR. JONES: That's correct.

MEMBER BALLINGER: So you had to assume that the liner failed for any of these events to even occur.

MR. JONES: Right. But there is --

MEMBER BALLINGER: What happens if the liner doesn't fail?

MR. JONES: You end up in a boildown scenario. All of this -- all of those considerations for the spent fuel pool study of the pool -- the fuel was not exposed enough to --

MEMBER BALLINGER: Okay. So end of story.

MR. JONES: -- fail in the 72-hour window.

MEMBER BALLINGER: So the whole -- these releases all are predicated on the fact that we had to assume a liner failure when we know that we didn't get liner failures under conditions much worse than this, and we also know that the stainless steel that's used for the liner is really tough.

MEMBER POWERS: I mean, why do you think

they are much worse than this? The .7 and 1.2 peak ground acceleration?

MR. JONES: There is --

MEMBER POWERS: Those are substantial earthquakes there.

MR. JONES: There is an explicit comparison of the .7 g peak ground acceleration. That is at 100 hertz. And the profile -- you know, the seismic spectrum that is associated with that, and the spectrums associated with the two major earthquakes in Japan that have occurred over the last 10 years. And that is in Chapter -- I believe Chapter 3 of the spent fuel pool study.

CHAIRMAN ARMIJO: But the Kashiwazaki earthquakes were very severe, more severe, and in a certain range of the spectrum it exceeded the .7. I don't know what -- how it compared to the 1.2 g in this study, but --

MR. JONES: Right.

CHAIRMAN ARMIJO: -- you know, those were not trivial earthquakes.

MR. JONES: No, they were not.

CHAIRMAN ARMIJO: And the performance of the liners was exceptionally good.

MEMBER BALLINGER: Well, this is a very

severe assumption.

MR. JONES: The Bin 3 100 percent on the high is definitely a very severe assumption. Bin 4 is a little bit less certain because we haven't seen an earthquake that severe near enough to a plant to cause that type of --

CHAIRMAN ARMIJO: But your argument is you haven't gone into the engineering details of, is this liner a little bit thicker? Is it connected the same way? So you are covering that --

MR. JONES: Variability.

CHAIRMAN ARMIJO: -- variability. Couldn't do it this way, you'd have to do a detailed engineering analysis like the spent fuel pool study.

MR. JONES: Right. And that's --

CHAIRMAN ARMIJO: Got it.

MR. JONES: -- a lot of resources.

MR. WITT: And that's also outlined in our SECY paper that we are going to have sent up to the Commission, that we did a review of operating experience history including the Japan reactors and basically confirms the conclusion that spent fuel pools aren't safe.

CHAIRMAN ARMIJO: Okay. Let's move forward.

MR. JONES: Okay. It is in the release -- just one last point on this slide. It is in the release fraction where you see that note about successful mitigation applied to the alternative to the low density case, which further reduces the likelihood of having a release in those cases as modeled in the reg analysis.

CHAIRMAN ARMIJO: Let me -- so that five percent would be reduced by a factor of 19, or has it already been reduced?

MR. JONES: No. It would be reduced further by --

CHAIRMAN ARMIJO: So if you did successful mitigation --

MR. JONES: I'm sorry. It's the frequency, not the magnitude, that was released.

CHAIRMAN ARMIJO: Oh, okay.

MR. JONES: The frequency. So, and maybe I put that in a bad --

CHAIRMAN ARMIJO: Yeah, okay.

MR. JONES: -- that note in a bad place, but --

CHAIRMAN ARMIJO: Got it.

MR. SCHOFER: In addition, responding to your question, page 45 of the enclosure talks about the release fractions.

CHAIRMAN ARMIJO: Okay. Page 45.

MR. JONES: Okay. And then going on to Groups 2 through 4, it's largely the same information as far as the insufficient natural circulation, except we go to -- even for the Bin 3 earthquake, we're assuming a 100 percent chance that there might be a partial drain case or some other configuration that would prevent adequate air cooling of the fuel.

The liner fragilities are a little bit different. They are generally lower because there is not the same level of amplification from the seismic event when the pool is near grade. The 25 percent in the highest is when it is actually drawn from a much earlier study, NUREG-1353, for the Generic Issue 82 beyond design basis accidents in the spent fuel pool.

And on the release fraction, you see higher numbers for the base case and highest base case, just representing the lack of detailed modeling for PWR fuel assemblies that we have available in the open literature.

CHAIRMAN ARMIJO: I thought you had a NUREG that had studied the PWR accident.

MR. JONES: There is some information that is done for security studies, yes.

CHAIRMAN ARMIJO: That you can't reference here?

MR. JONES: Correct.

MR. SCHOFER: Correct. We relied upon the values that were in prior studies, 1738, that had those values, the 90 percent.

CHAIRMAN ARMIJO: Okay.

MR. JONES: Okay. Next slide.

This table just indicates the relative amounts of cesium in the different cases considered, and they represent actual pools. They are derived from -- the values are derived based on the licensed thermal power of the reactors, the license capacity of the pools, assuming one core reserve capacity for an offload, and also considering different burnup cases for the high estimate. Of course, it's high burnup fuel throughout the spent fuel pool.

And you see really the highest case is Alternative 1 for Group 4. We have a shared pool. We have two units discharging to a single pool, and that obviously would produce higher results.

CHAIRMAN ARMIJO: Okay.

MR. JONES: Group 3 has generally the lowest results, because they are new reactors and it takes a while to generate a population of fuel in the pool.

VICE CHAIRMAN STETKAR: Steve, in this case, though, your low estimate really isn't the lowest in the group, and your high estimate really isn't the highest in the group. So for some reason you did these lows and highs differently than you've done other things. No, they're not, just -- I looked at the data. They're not. They're sort of middling low and middling high if you will.

MR. JONES: They were selected among the population of plants.

VICE CHAIRMAN STETKAR: The highest for Group 4, for example, is 175.4. The lowest for Group 4 is 42.7. So these lows and highs are somehow selected differently using a different thought process than the other lows and highs that you've given us.

CHAIRMAN ARMIJO: Is that only for Group 4?

VICE CHAIRMAN STETKAR: No, no, it's all four of the groups. Well, Group 3 doesn't have any data, so we don't know --

CHAIRMAN ARMIJO: Yeah. Right.

VICE CHAIRMAN STETKAR: -- how they select the data. But 1 and 2 and 4 -- for example, if I look at the data, the lowest for Group 1 is 24; the highest is 74.2. The lowest for Group 2 is 20.4; the highest for Group 2 is 115.1.

I know how you calculated the best because it's a linear average for the -- I can reproduce these numbers. I can find the numbers that you picked. But they're neither the lowest of the low nor the highest of the high. So for some reason -- my only point is that the way you're characterizing low and best and high, they seem to be different depending on which parameter you have decided to focus on.

And that doesn't really come out in your report, because you tend to characterize things using words like "bounding." Well, the highest is not the bounding, nor is the lowest, the bounding lowest, given the evidence.

CHAIRMAN ARMIJO: Was that done intentionally?

VICE CHAIRMAN STETKAR: It must have been done intentionally, or there's a lot of -- I could copy a table and rank order things and --

CHAIRMAN ARMIJO: Yeah, yeah. So, you know, Steve, do you have an explanation for that -- for those differences between what is really low and what's really high versus what's on these -- on this chart?

MR. JONES: I think what I used was the 90 percent value for the high --

CHAIRMAN ARMIJO: I'm sorry. I didn't hear you.

MR. JONES: -- a 90 percent value for the high and a 50 percent for the low. I'm just looking for it now.

CHAIRMAN ARMIJO: Why? You know, these are really small differences considering all of the other bounding stuff. What is wrong with just the lowest of the low and the highest of the high?

MR. WITT: Well, if I could just --

CHAIRMAN ARMIJO: As kind of an approach.

MR. WITT: -- if I could just offer, we are getting close to the time for concluding the presentation. Is this something that we could get back to you on?

CHAIRMAN ARMIJO: Sure. We'd like to get --

VICE CHAIRMAN STETKAR: Well, one of the things I want to pursue a bit -- and, unfortunately, I didn't have the opportunity to come to the Subcommittee meeting. I have to apologize for that. I tried to understand how you established these ranges. I understand it for the seismic stuff. I don't it for this. I mean, I understand where these numbers -- I can point to where these numbers came from. I don't know why you didn't do it the same way. I didn't have a chance to go through all of your other estimates of low, best, and high.

CHAIRMAN ARMIJO: Let's get their answer.

VICE CHAIRMAN STETKAR: Yeah. Well, no, wait a minute. Let me finish because --

CHAIRMAN ARMIJO: Well, make your point.

VICE CHAIRMAN STETKAR: -- it was important for me to try to understand how they then related to your sensitivity studies, which I had trouble following through, because I didn't understand how even your best estimate cases scaled in your various sensitivity studies.

MEMBER SCHULTZ: One point, Steve, real short. Here on this slide, suddenly we are talking about the middle case being best estimate versus base case. And in terms of our --

MR. JONES: My fault.

MEMBER SCHULTZ: -- communication to the public where the base case is very conservative, we don't want to miscommunicate that we are talking anything about best estimate?

VICE CHAIRMAN STETKAR: This is, though, the linear average cesium content for that group. So in a sense it is the best estimate.

MEMBER SCHULTZ: I understand. But the next slide also says best estimate where it -- we cannot get into that miscommunication.

CHAIRMAN ARMIJO: Yeah. Well, look, you'll get us a response to --

MR. WITT: Yes. We'll get back to you on that.

CHAIRMAN ARMIJO: Yeah. Okay. Let's keep going.

MEMBER POWERS: In the analysis of consequences where you had potential of air interacting with fuel, is cesium the appropriate surrogate radionuclide to look at?

MEMBER CORRADINI: Can you say that louder? I didn't hear you.

MEMBER POWERS: When we have the potential of air interacting with fuel, is cesium the radionuclide to pay attention to? We can concede that iodine is inappropriate, because there probably isn't very much unless the fuel has just been offloaded, but it's not transparently obvious that if there is the potential of air to interact with fuel that one should not look at the platinoids or molybdenum.

MEMBER CORRADINI: I thought you were going to say ruthenium.

MEMBER POWERS: Ruthenium is a platinoid.

MEMBER CORRADINI: Oh, sorry. It is.

MR. SCHOFER: Well, what was done, as discussed on page 28 of the enclosure, is we used cesium as the means to scale the inventories from between different types of assemblies, BWR to PWR, as well as the number of assemblies in the pool.

And we used that, we verified the accuracy of that type of scaling device based upon the origin code. And then the isotopes that actually came out of the MELCOR analysis were actually used in the MACCS runs to come up with the radiological consequences.

CHAIRMAN ARMIJO: And were they primarily cesium, or were they --

MR. SCHOFER: No, it was the whole spectrum of what --

CHAIRMAN ARMIJO: -- the whole spectrum of --

MEMBER POWERS: Well, the question is, I could get MELCOR code to -- on a good day, I can get the MELCOR code -- on the best day I have ever had, I can get the MELCOR code to release whatever I want. So it depends a little bit on where its release modeling is. And if you broke default MELCOR, then you're getting a steam release. You're not accounting for any air in it.

If I look at other sources of information, I can find that air does have an impact on the radionuclide splits that get released.

MR. JONES: Can we get back to you? We need to consult with the research folks that did the actual MELCOR analysis inputs to the MACCS2 runs.

Okay. Just moving on, these are the regulatory analysis inputs. I guess the main thing I'd just like to point your attention to is the habitability limits used in the health effects analysis. The analysis does assume relocation of people to limit the actual health effects, and that's really all I'd like -- other than that, there are the low, best, and high, or base case, and high estimates do consider different economic and demographic information.

VICE CHAIRMAN STETKAR: I'm sorry. This is different than what's in the report. In the report, the low estimate is 93, and the high estimate is 688, if I look at Table 31 in the report.

CHAIRMAN ARMIJO: On what, the population?

VICE CHAIRMAN STETKAR: Population.

MR. SCHOFER: I reviewed a number of different cases, and the earlier section where I was doing sensitivities for one at a time, just varying population density, I actually had four cases. I had Peach Bottom, Surry, Palisades, and Point Beach.

And then, toward the end where I was grouping to have a low estimate, base case, and high estimate, I used the top three which was Peach Bottom, Surry, and Palisades.

With regard to the numbers, if there is a difference it would be the difference from the last census versus the increase in the population since that last census. So I might have an inconsistency, but the results are the same.

VICE CHAIRMAN STETKAR: Fred, I'm sorry, just -- I'm looking at Table 31 that shows a distribution of the population density for 50 miles. And that table says, "I'm not going to use words like low. I'm going to use words like 20th percentile is 93." The median is 169. I see 169 here.

MR. SCHOFER: Right.

VICE CHAIRMAN STETKAR: The mean is 317. I see 317 here. And the 90th percentile is 688. I see 722 here. So I'm not quite sure how these numbers that we are seeing today on this slide jive with this table. And from the qualifications that you said orally, I have no idea, then, what was done in the population sensitivities because you're saying, "Well, I did this grouping and I did that grouping."

Because that's part of what I was trying to do in your sensitivity studies is to scale from this type of information, fixing the population, varying seismic frequency, and I couldn't scale those at all. So there must be something done intermediate that isn't explained.

MR. SCHOFER: The high estimate is Peach Bottom, the mean is Surry, the median is Palisades, and the low is Point Beach.

MR. WITT: Dr. Armijo, in the interest of time, I mean, we're running --

CHAIRMAN ARMIJO: Yeah. Get back to us on that, because, you know, the lack of -- we don't understand why the numbers don't all fit as means or medians or something. They bounce around a lot, and they're staying at least in the same order of magnitude. But I think we want to do a little better than that.

MR. WITT: Yeah. We do commit to getting back to you on these questions.

CHAIRMAN ARMIJO: Okay.

MR. WITT: The issue is whether we can do it right now or --

CHAIRMAN ARMIJO: No, no. No, no, no. We won't do it -- no need to do it right now. Let's get through the presentation. We're running a little bit behind schedule, so we're going to have to move along.

MR. JONES: Backfit analysis resulted in the conclusion there is no substantial increase in overall public health and safety. That's looking predominantly at a comparison to the safety goal of quantitative health objectives. Regarding the nature of the release, there is really not a significant immediate health effect to anyone based largely on cesium being the dominant nuclide of interest.

The individual risk for latent cancers is much lower than your objective, on the order of two or three percent of the qualitative health objective. And that is due in part to all cases being subject to the relocation of populations and that limiting the doses that people are exposed to.

And the individual risk is dominated by long-term dose in habitable areas. In other words, people that don't move are exposed to a low level for a long period of time. The model does use a linear no-threshold dose response model which maximizes the health effects.

CHAIRMAN ARMIJO: You didn't consider what -- some sort of a threshold model like -- similar to what was done in SOARCA? Particularly for habitability, long-term habitability, it really -- I just -- you just drive the numbers way up with LNT for that kind of thing.

MEMBER POWERS: Consistent with the best current understanding of health effects.

CHAIRMAN ARMIJO: Well, I disagree with that. I think it --

MEMBER POWERS: If you want to disagree with that, we will have a real long discussion. But, you know, I think the SOARCA thing did put in a --

CHAIRMAN ARMIJO: Capricious and arbitrarily selected.

MEMBER POWERS: I think it was more than that, but we'll --

CHAIRMAN ARMIJO: It was completely capricious and arbitrary.

MEMBER POWERS: Okay. We will agree to disagree on that one. Okay. So but I think your -- now you made me forget what I was going to say, so I'll keep -- I'll come back to it later. It isn't a debate on LNT, but I think it's -- oh, gosh darn it, just keep going and I'll remember it.

MR. SCHOFER: Well, quickly, just to address that point, there was a sensitivity done in the spent fuel pool study, Table 34, that looked at LNT versus looking at dose truncation comparison. And it looked at linear no-threshold, a 620 millirem year truncation, and a five-year or 10 rem lifetime truncation.

So you can look at that and you can see that, you know, there's several orders of magnitude difference between those phases.

MEMBER CORRADINI: The three, again, were the last -- I didn't hear. I'm sorry. You said one was of course LNT, one was --

MR. SCHOFER: 620 millirem per year.

MEMBER CORRADINI: Which is the addition of background and medical. And then what was the third? I'm sorry.

MR. SCHOFER: Five rem per year or 10 rem lifetime truncation.

MEMBER CORRADINI: Okay. Thank you.

MR. JONES: And the results of the analysis were that the costs outweigh the expected public health benefits, and we do consider the majority of the units evaluated to be bounded by the base case analysis due to the number of conservatisms, particularly in the accident progression portion.

The highest --

MEMBER CORRADINI: If I might just ask, because maybe I missed it, but instead of having -- you used LNT, but you had a distance truncation.

MR. SCHOFER: I had a 50 mile, but then I also have sensitivity that went out as far as possible.

MEMBER CORRADINI: Oh, you did. Okay. I missed that. Excuse me. Okay.

MR. JONES: These results are based on -- the last bullet does address that consideration beyond 50 miles for the low and base cases for all -- the costs outweigh the public health -- expected public health benefits. For the high estimate, we think that's way bounding and -- but as a result, the health benefits did outweigh the costs when consequences beyond 50 miles were considered for that case.

CHAIRMAN ARMIJO: Steve, just to make sure I -- clear this up for me. It's my understanding if you meet the health objectives with margin, on both fatalities -- early fatalities and latent cancer fatalities, if you meet those with a sufficient margin, that's as far as you have to go in this analysis.

MR. JONES: Right.

CHAIRMAN ARMIJO: By regulation. Now, so the cost-benefit work, that's purely discretionary. It's not the basis for your decision. Is that correct or not?

MR. JONES: Well, we're --

CHAIRMAN ARMIJO: It may be your practice, and you may do it all the time, but --

MR. JONES: -- the information possible to the decisionmakers I guess, and in terms of whether or not to pursue additional research and refine the numbers. But you're right, the guidelines for regulatory analysis do stop at a safety goal screen for reactor events.

CHAIRMAN ARMIJO: Okay. So based on meeting the safety goals, and, you know, the margin for -- in this analysis is smaller than it was for Peach Bottom. You know, Peach Bottom had a huge amount of margin. And all of your conservatisms have eaten up a lot of that margin, but you still have plenty.

So I -- so at this point, I'll ask again, why did you do the cost-benefit analysis? And the answer I got is, "So the Commission can take a look at it and see if they think you want to -- they want you to do more stuff." But it wasn't the basis for your decision or conclusion.

MR. WITT: I would say that it helped inform our conclusion. But when we follow the process, our regulatory analysis guidelines are very clear. You do the screening first and you --

CHAIRMAN ARMIJO: Yeah. Make a decision based on health and safety, not economic consequences.

MR. WITT: Right. But we continued on just to provide that information.

CHAIRMAN ARMIJO: Okay. I understand that. Thank you.

MR. JONES: Next slide?

Okay. The regulatory analysis adds in those economic factors, such as property damage and relocation costs and things like that.

CHAIRMAN ARMIJO: Sure.

MR. JONES: The base case and low estimate costs outweigh the benefits based on a $2,000 per person rem within 50 miles, and, really, in many of the other cases we'll get to in the sensitivity study.

The high estimate benefits appear to outweigh the costs, and we believe that is largely due to the conservatisms in the analysis.

Okay. The sensitivity analysis did look at a higher dollar per person rem, but it's not yet approved for use in consequences beyond 50 miles. In that case, the low estimate costs outweigh the benefits.

For the base case, the costs outweigh the benefits for Groups 1 and 2, which are the boiling water and pressurized water reactors that discharge to a dedicated spent fuel pool. The benefits marginally outweigh the costs for Groups 3 and 4, Group 3 largely because the fuel movement is further out in time, and, therefore, it requires less upfront investment to support, and Group 4 because of the larger inventory of fuel there.

The high estimate cases, again, when you go beyond 50 miles, appear to outweigh the costs because of the conservatisms and the -- oh, safety perspectives.

This next slide is --

CHAIRMAN ARMIJO: Before you leave that, in these various estimates, one thing that it seemed like it -- was invariant and that was the cost. It just seemed like it didn't move at all. And if there is a high, low, high -- I'm sorry, low, base case, and a high case, I can tell you from experience that cost estimates, particularly on nuclear work, moving things and all of this sort of stuff, are -- all of these cost estimates are really -- really should be adjusted to put more realism -- at least attempt to put more realism.

And the costs go up -- should go up for the high case as well as for -- they shouldn't stay the same. That kind of distorts the picture. It may not make much difference. You know, it may be that the benefits -- will still be higher, but the costs should be -- they can't be all the same. I guess that's --

MR. SCHOFER: I'll simplify an assumption. In some cases, you know, you could increase that by a factor of -- you know, the highest for the high would be factor two or more higher, for the low it might be, you know, a third or 50 percent lower. But because I got the -- a lot of the cost information from, you know, EPRI documents I kept that constant across all the cases.

CHAIRMAN ARMIJO: Well, is that consistent with this -- there's a number floating around -- I think it's in an EPRI report -- of $3.6 billion to do something like this, implement expedited transfer. I think Mr. Kessler may be on the phone. We might ask him, too. But are those the costs you used? That's all.

MR. SCHOFER: There are a number of EPRI documents that look at five-year, you know, movement of fuel, and it's from those reports that I got the cost information.

CHAIRMAN ARMIJO: Okay. Well --

MEMBER BANERJEE: Could we just look at the previous slide, please, for a moment for the beyond 50 miles? So when you look at these estimates, low, base, high, are there any sort of probabilities associated with these? Can you give us like what the chances of a low estimate might be compared to base estimate or high estimate? Because obviously you have made a bunch of assumptions, right, around the base estimate. So you have a base estimate which is already a very low probability event, because it's beyond design basis.

On top of that you made a bunch of assumptions, like things will fail and so on. So maybe the probability of this happening is 10 to the minus six or seven, or I don't know what that number is, but then what is the low estimate probability, and what is the high estimate?

Because now with the high estimate you

made a further set of assumptions that the liner will fail with a probability of one or whatever. I don't know what you've done, but without that, you know, these estimates just hang in the air. We don't even have a feel for is the high estimate 1,000 times less likely than the base estimates? Or is it 100 times less likely? And the best --

MR. JONES: I guess the release --

MEMBER BANERJEE: Is it 10 to the minus 10 instead of 10 to the minus seven?

MR. JONES: Yeah. We have some frequency information on this slide, and I guess I can talk to the different --

MEMBER BANERJEE: Okay.

MR. JONES: -- pieces a little bit.

MEMBER BANERJEE: That would be useful, yeah.

MR. JONES: Okay.

MEMBER BANERJEE: Thank you. I just want to fix what this high estimate is.

MR. JONES: Right. The pools provide -- we feel that the pools provide adequate protection and defense-in-depth. The overall estimated frequency of damage to the stored fuel was very low, and the base case we -- we have frequencies of release that's a few times in a million years. And then --

MEMBER BANERJEE: Ten to the minus six.

MR. JONES: -- the 10 to the minus six value and there is really a range based on, really, the seismic hazard curve inputs for the different groups.

MEMBER BANERJEE: It's 10 to the minus five, then, seismic hazard or what?

MR. JONES: Right. The seismic hazards are generally in the 10 to the minus five range, or 10 to the minus six range even with some of these.

MEMBER BANERJEE: And then you assume some probability that the liner will crack or whatever.

MR. JONES: Right.

MEMBER BANERJEE: Which is, what, about 10 percent of the time?

CHAIRMAN ARMIJO: In the base case, sometimes it's one.

MEMBER BANERJEE: No. I'm saying with the base case.

MR. JONES: For the base case, it's -- yeah, it varies, but it's --

MR. SCHOFER: Between eight percent and one, depending upon the case.

MEMBER SCHULTZ: What it's meant to do

here is to demonstrate that for the base case we are presuming a very, very conservative case. The base case is a very, very conservative evaluation tool.

MEMBER BANERJEE: Then we should call it a very, very conservative case.

MEMBER SCHULTZ: So there is not a practical way to establish a likelihood unless one goes back to the spent fuel pool study and makes a lot of different comparisons about a number of conservatisms that have been included in the low estimate, in the base case, and even more in the high estimate. And we are talking many, many orders of magnitude for the high estimate, several orders of magnitude for the base case, and maybe an order of magnitude or so for the low estimate.

MEMBER BANERJEE: So just to fix it in my mind, let's say for whatever reason that earthquake is 10 to the minus five to 10 to the minus six. Okay. So the low estimate now becomes what?

MEMBER SCHULTZ: It should be the same.

MEMBER BANERJEE: It should be one order of magnitude even less, is that the way I'm reading it?

VICE CHAIRMAN STETKAR: No. I don't think that's what Steve said, because it's -- you have to be careful because they never had a low estimate of the seismic -- it's -- you can't -- no matter what we do -- we could discuss this for days in this forum, and you'd never understand the information that you are trying to get, because, for example, their base case and low estimate use the same seismic hazard. So the low estimate seismic hazard is the same as the base case seismic hazard.

MEMBER BANERJEE: So what's different, then?

VICE CHAIRMAN STETKAR: It's the other numbers that they played with.

MEMBER BANERJEE: Okay. So just the difference in the other numbers.

MR. SCHOFER: Table 84 provides you --

MEMBER BANERJEE: It does?

MR. SCHOFER: -- a summary.

MEMBER BANERJEE: It would be -- you know, to me, I just don't understand what this means in terms of frequencies.

VICE CHAIRMAN STETKAR: So in other cases they -- you know, in some cases it's a mixture of -- and I'll call it "a number" times a lower number gives you one result, and a number times a different number gives you a different result. But the first single input number is the same for those cases.

So although you get a range of two different values, it isn't the low and the low. It isn't the low and the high. You know, it isn't --

MEMBER BANERJEE: So put it --

VICE CHAIRMAN STETKAR: -- something that you can get a handle on.

MEMBER BANERJEE: So trying to put it in my world, which I understand, which is of course thermal hydraulics, and this is related to thermal hydraulics in many ways. You have a distribution of some sort of various parameters which you then sample using non-parametric means or whatever.

So is that sort of what you're doing? Or why not?

MR. JONES: You mean like a Monte Carlo type --

MEMBER BANERJEE: But then, with a Monte Carlo, you need to do a huge number of --

MR. JONES: These are just some assumptions meant to really get a -- maximize the difference and look at the overall benefit from a slow leak transfer of fuel.

CHAIRMAN ARMIJO: But when you maximize the benefits of Alternative 2, you also -- you create what seems to be like an argument for doing Alternative 2. If people look at your high estimate and you look at the sensitivity studies, people that don't know what you've done would immediately say, "My God, what are they waiting for? We should do this right away. It's millions of dollars of benefit."

And yet I don't see that. It just -- so without having the frequency information that kind of puts it in perspective, you're really -- they're just numbers. We could -- certainly would like to get something like --

MEMBER BANERJEE: No. I think frequency would be --

MEMBER SCHULTZ: Well, the purpose of the study -- it's very clear -- it's biased in favor, well in favor, of removing the fuel from the pool. It is biased in the very, very conservative direction in terms of potential impact if you leave the fuel in the pool. And this is only a study -- on a study to determine if further study is warranted.

MEMBER CORRADINI: Yeah. I think it --

MEMBER SCHULTZ: It clearly determines that it is not.

MEMBER CORRADINI: I think that's the key to me is -- which is what Bill said at the very beginning. This was, if I understood staff's plan, is they were going forward with, pardon the expression, multi-year attack. This is Phase 1 of a multi-year attack at this, and it's a screening approach.

And if the screening approach shows that there are some outliers, then you've got to go back and look much more carefully, much more, as Sanjoy is asking all of these fun questions, as to whether you've got to look at it in more detail. And so --

CHAIRMAN ARMIJO: So is the answer that if the base case is solid, no outliers, even though it's conservative and shows no benefit, the other things are just interesting studies but of no significance or what?

MEMBER CORRADINI: I just don't know what to do with those, the high phase, and I certainly don't know what to do with the sensitivity.

MEMBER BANERJEE: We need to look at the outliers, because if the outliers are so outlying that they -- if we can say the outliers are 10 to the minus 10, I mean, who cares?

MEMBER CORRADINI: Well, I mean, you're asking an opinion. I don't know if this is the right --

MEMBER BANERJEE: No, it's not an opinion.

I'm saying --

MEMBER CORRADINI: But I think the staff is -- at least the explanation coming into this, the staff looked upon this as a first cut at the problem. And they are coming through this saying that there are some that come close to the boundary. And if they look beyond the 50 miles, it goes beyond the boundary in terms of the cost-benefit. But they don't see it enough to warrant any further action.

And then, it's a matter of --

MEMBER BANERJEE: But that's a matter of opinion.

MEMBER CORRADINI: Correct.

MEMBER BANERJEE: Now, if you file conservatism on conservatism, we can always arrive at scenarios which will give us $100 billion, you know, whatever. I mean, the east coast gets radiation or whatever. You know, so it's just a question of how much. Where is the question of degree here? That's my point.

MEMBER CORRADINI: I think maybe Steve was trying to take a whack at that when he was giving you the orders of magnitude.

MEMBER SCHULTZ: But it's only -- again, the perspective we must maintain is that this is an analysis that only -- only is for the purposes of determining whether further study is warranted. Where it's really dangerous is that these results could be interpreted as being meaningful with regard to potential consequences.

And when we have a forum in which both professionals as well as lay people see these results and believe that it represents something like the results of a spent fuel pool accident, it is -- that's very dangerous to put out there.

And so the purpose of this study and the conservatisms that have been applied really need to be explained very carefully in the front of the document and throughout the document, because, again, if you go through it and you compare the assumptions here with what has been derived in the spent fuel pool study, it is orders of magnitude in several different places, which have been placed in favor of Alternative 2, just to see if it possibly has any reason for -- it has demonstrated, based on what we're hearing today, that it doesn't pass, even with these orders of magnitude, thousands and millions of --

CHAIRMAN ARMIJO: Thank you. I think we'd better keep moving. We're just -- we need -- I want to try and stay on schedule, because we have some --

we need some time for further -- other people to talk. So I'll --

MR. JONES: We'll try to run through the last few slides. I guess the main point here is we believe the base case is representative of a reasonably conservative case of the set of spent fuel pools we're looking at. I mean, the entire inventory of spent fuel pools.

And there is defense-in-depth. You have a very low frequency of any real challenge to the spent fuel pool because it's designed that way. And on top of that, we have fuel dispersal in the "makeup and spray capabilities" to help address any event that may be well beyond the design basis and challenge those pools.

Okay. Next slide.

Therefore, our recommendations were that expedited transfer of spent fuel to dry cask storage did not provide -- would not provide a substantial increase in overall protection of public health and safety, and the safety benefit for the best base case does not outweigh the associated costs.

And, therefore, we would not recommend pursuing additional study to look at expedited transfer of spent fuel to dry storage, and, therefore, with this activity it would be closed.

Okay. Next slide.

Other alternatives were brought up I guess in the spent fuel pool study. And, again, these would involve lower costs but additional cost to industry, including alternative loading patterns, direct offloading into the distributed patterns, and enhancement of the mitigation strategies to be more reliable than what has been established through the orders.

But, again, the staff considers these changes are not warranted based on the overall results.

MEMBER BANERJEE: What does "enhancement of the mitigation strategies" mean? Just backup pumps for your spray or what?

MR. JONES: I guess efforts to make the reliability of their deployment more -- or higher reliability in their deployment. And I don't know --

MEMBER BANERJEE: So you didn't look at this in detail. You're just saying make the mitigation systems more reliable. However you choose --

MR. JONES: That could be pre-deployment or fixing certain elements of the equipment in place rather than having to bring them out of storage and position them.

MEMBER BANERJEE: So to be concrete on this, what is the likelihood of failure you assume for the mitigation strategies for the -- let's say your base case?

MR. JONES: Well, again, we did not really consider this except with respect to the two alternatives, and that's -- for the current regulatory regime, we assumed the mitigation equivalent would be ineffective, maximizing the benefit. And for the alternative where you have a low density case, we assumed that the mitigation equivalent would be effective in reducing the frequency of fuel damage by a factor of 19. So -- or 95 percent.

MEMBER BANERJEE: But, so again, you --

CHAIRMAN ARMIJO: None of these things would even come close to the costs of expedited transfer. So these have got to be very low cost. And if there was -- and so the decision can't be based on the -- the decision has to go back to your Slide 15.

If there is no -- if you meet the quality -- quantitative health objectives with margin, you shouldn't have to do anything. Even though it would be nice to do, and somebody may choose to use a one by eight loading pattern, because it does have real benefits.

MEMBER CORRADINI: Not only choose to, they are using.

CHAIRMAN ARMIJO: One utility gets that. So, you know, it seems to me like that's a good practice, but it's certainly not justified.

MEMBER BANERJEE: So going back to this, the high case, you also assume that mitigation would not work beyond -- for the greater than 50 miles case, or the case which is as is, and the case that you removed, you assumed the mitigation would work. And that's an enormous delta then.

MR. SCHOFER: Yes, it is.

MEMBER BANERJEE: All right.

CHAIRMAN ARMIJO: All right.

MEMBER BANERJEE: I think we really need to clear this because --

CHAIRMAN ARMIJO: Yes. I think we are coming together on that.

MEMBER BANERJEE: Getting there, yeah.

CHAIRMAN ARMIJO: Steve, we've got to -- if you can just get through your next two, we'll try not to --

MR. JONES: Sure. I'll just go over the

last two slides. We did have several public meetings as well as meetings with you all on this issue. And we have taken that information back, the comments and suggestions, and we have tried to include additional discussion in the COMSECY.

In addition, we have also received letters from stakeholders on this issue. We are responding to those letters, and they are being considered in our development of this COMSECY.

The spent fuel pool study that Research is working on, it's being finalized as we speak. They received a number of comments during a public comment period, and those have been addressed in the spent fuel pool study. In addition, we're aware of what those comments were, so we have considered that in our analysis.

CHAIRMAN ARMIJO: Okay. Kevin, we also received a submittal from a Dr. Gordon Thompson. Are you -- did you receive that as well, and are you preparing a response to his arguments? Or is that --

MR. WITT: I'll have to look back through my records. I don't recall --

CHAIRMAN ARMIJO: It relates to partial draindown of the pools. And I don't know if it's a spent fuel pool study issue or whether you are going to -- how the staff is -- whether the staff is going to address it and how.

MR. WITT: Well, some of the letters that we've gotten in were more directed at the spent fuel pool study. And so our Office of Research is considering that input and responding to that. So I'm not too sure if the Gordon Thompson was directed towards this Tier 3 analysis or the research study.

CHAIRMAN ARMIJO: No. I think it's more toward the research. Yeah, okay.

MR. WITT: But we are aware of all of the things that stakeholders have brought up, so we are --

CHAIRMAN ARMIJO: Okay.

MR. WITT: -- including that in our analysis here.

MR. SCHOFER: Although our coolability value of 100 percent not coolable addresses that partial cooldown.

MR. WITT: So real quickly, next steps. This paper will be finalized by October 11th. That's next week, considering we are still operating. And then we'll -- the Commission is planning to have a meeting on this issue by the end of 2013, which we will participate in.

CHAIRMAN ARMIJO: Very good. Well, what

I'd like to do now is we have -- we had a request for additional staff discussion by Don Helton of Research. And if Don is available, he has some remarks that he'd like to share with the Committee.

MR. HELTON: Just to clarify a point that came up a moment ago -- sorry, Don Helton, Office of Research. Clarify a point that came up a moment ago. The RES staff received a 40-page -- roughly a 40-page letter from Dr. Thompson as part of the public comment period on the spent fuel pool study.

CHAIRMAN ARMIJO: Yeah. That's what I was referring to.

MR. HELTON: Right. And so that's being responded to as part of the finalization of the spent fuel pool study itself. So the final document will have an Appendix E that responds to all comments received during the public comment period.

CHAIRMAN ARMIJO: And make sure to get that to us, Don.

MR. HELTON: Okay.

CHAIRMAN ARMIJO: Make sure we get a copy of that response.

MR. WITT: Yeah. I believe the spent fuel pool study was being sent to the ACRS.

CHAIRMAN ARMIJO: Okay.

MR. HELTON: Okay. So, again, my name is Donald Helton. I'm a staff member in the Office of Nuclear Regulatory Research. A couple of sort of preamble comments if you will before I get into some of the other remarks that I wanted to make.

First of all, the regulatory analysis represents a significant amount of work accomplished in a relatively short timeframe. And the NRR staff, in my view, should be commended for both the breadth and the complexity of what they have accomplished.

The remarks that I'm about to make are intended to provide additional emphasis on particular aspects of the regulatory analysis. They may not resonate with the Committee or the Commission as they are currently characterized in the draft Commission paper.

Mr. McGinty mentioned earlier, because of the expedited schedule here, we're using this forum as part of continuing the healthy dialogue that's going on between the NRC staff.

Finally, these represent my views. The are not the views of the Office of Nuclear Regulatory Research. Also, I need to make one point of clarification so that some of my later comments make sense. There were some statements in the earlier discussion about the substantial safety enhancement as part of the regulation and the QHOs, as part of that.

It is my understanding -- and certainly Fred Schofer can intervene if he thinks that I mischaracterize this -- the regulation requires a determination of substantial safety enhancement. The regulatory guidance recommends the use of the safety goals, the QHOs, as the means of making that determination. So I'm just drawing a distinction there between the regulation and the regulatory guidance.

With that, the regulatory analysis shows that the expedited movement of fuel older than five years from spent fuel pools to dry cask storage does not provide a substantial safety enhancement. It is important, in my view, for the reader to understand that the significance of the safety enhancement has been judged based solely on the risk to individuals living in close proximity to a nuclear powerplant.

This means that risk to the individual is assumed to be a reasonable surrogate for cumulative human health risk, even though these events are known to be low likelihood/high consequence events, high consequence in the unlikely event that they occur.

Point two, the regulatory analysis shows that the studied action is not cost beneficial when radiological release frequency estimates are biased in favor of a cost beneficial finding, while the total offsite impacts -- human health and otherwise -- are not comprehensively considered.

Specifically, a dated dose conversion factor and a 50-mile distance truncation are employed. The Commission paper acknowledges this and emphasizes the importance of the sensitivity analyses without informing the reader that, a) in many instances this is the difference between a cost beneficial and a non-cost beneficial finding; and b) it makes an order of magnitude difference in some results.

Point three, the staff's work to date does not provide a clear perspective on the cost beneficial results when both the conservatisms and the non-conservatisms are removed. Based on my own investigation, which involved constructing a cumulative distribution function from the low, base and high cases, and using the beyond 50 miles and $4,000 per person room sensitivities, I expect that the action would not be cost beneficial for a majority of the fleet, but that it could be cost beneficial for many plants.

Additional work to refine specific

simplifying assumptions in the regulatory analysis, such as the effect of mitigation in reducing the release frequency, or to perform a simplified plant-by-plant screening based on available information might alter this conclusion in a more non-cost beneficial direction.

Point four, the regulatory analysis does not consider related alternatives such as expedited movement of fuel older than 10 years or refinement of spent fuel pool heat management strategies. These might be more cost beneficial.

Point five, since on a whole there is no compelling evidence upon which to take generic regulatory action, I personally agree with the Commission paper's recommendation to close the Japan Lessons Learned Tier 3 item.

However, in light of the points raised above, I believe that the staff should advocate for continued staff activity under another appropriate regulatory program to assess whether actions would be cost beneficial for specific plants, when simplifying assumptions are refined, or when other contributing factors such as inadvertent criticality are considered.

This would be in addition to resolving the issue for western plants, as the Commission paper already envisions. This information would then be provided to the Commission.

Point six, I believe that the staff should also seek Commission direction on the use of quantitative health objectives for an individual as a suitable measure of substantial safety enhancement for classes of accidents known to be low likelihood/high consequence events, particularly when this determination causes the staff to dismiss cost beneficial or potentially cost beneficial actions.

Point seven, since future work is not expected to change the NRC's understanding of the fundamental processes affecting potential environmental consequences of spent fuel pool zirconium fires beyond the significant state of knowledge that already exists via this regulatory analysis, the supporting spent fuel pool study, and the numerous past investigations on this issue, I believe that activities related to the development of the environmental impact statement and proposed rule for waste confidence should proceed unencumbered by the follow-on activities recommended earlier in these remarks.

Finally, point eight, I believe the

characterization -- I believe that the characterization of the regulatory analysis in the Commission paper needs to be strengthened to capture the importance of these items, such that the Commission paper provides the Commission with a balanced perspective upon which to provide direction.

Thank you for your time.

CHAIRMAN ARMIJO: Okay. Thank you, Don. Interesting comments.

Okay. We're running a little late. Unless there's burning questions, I'd like to move to the next presenter or speaker, and that's Ms. Curran. Is she here? Ms. Curran, kindly speak -- well, go ahead. Sit down there. You're fine.

MS. CURRAN: Thank you. Although I appreciate this opportunity to talk to you all today, I have asked Robert Alvarez to come up with me because I'm going to share my time with him. And I would also like to share my time with Dr. Edwin Lyman of Union of Concerned Scientists.

CHAIRMAN ARMIJO: Okay. But, you know, just keep it to 10 minutes. Okay?

MS. CURRAN: We will do our best. I am Diane Curran. I am here representing 26 environmental groups across the United States. We consider the issue of spent fuel storage risk to be one of the most critical safety issues facing the NRC today, and we look to the ACRS for your independent, thorough assessment of the risks posed by this fuel storage method.

I have been representing environmental groups in spent fuel storage cases since the mid-1980s. And I can tell you for two decades, the '80s and '90s, the NRC systematically denied that spent fuel -- high density pool storage of spent fuel posed any risk to the public at all, wasn't an issue.

Dr. Gordon Thompson, our expert, who has done detailed comments on the draft consequence study, was my expert witness in a case in North Carolina involving a proposed spent fuel expansion, high density pool storage. He said, "High density pool storage poses a significant risk of a severe accident." He was told -- I was told -- I had hired somebody who was crazy basically, who didn't know what he was talking about.

The ACRS played a crucial role in changing that mindset, and, frankly, I don't think that any of us would be sitting here today talking about the draft consequence study if it had not been for the ACRS.

In the year 2000, the ACRS, which was then chaired by Dr. Powers, held a meeting at which Dr. Thompson was invited to give a presentation and presented his view that high density storage does pose a risk of zirconium fire and that the studies on which the NRC had relied for decades to say it wasn't a problem were based on studies of low density storage that were purported to address the risk of high density pool storage.

It was in the year 2000 that the NRC staff members came to that meeting and admitted that actually they couldn't rule out the potential for a catastrophic fire in a high density storage pool. Shortly thereafter, the September 11th attacks occurred. And the NRC basically admitted that there was a serious problem here and then decided to deal with the whole matter in secret.

So there has been a couple of turning points for the ACRS and the NRC. There was the 2000 ACRS meeting, the 2001 attacks, and then now we have the Fukushima accident, which has raised the profile of spent fuel pool accidents again. And, unfortunately, the process for looking at this issue has not been as open and thorough as it should be, given the importance of the issue.

You have had a meeting of your

Subcommittee that looked at the draft consequence study without the benefit of any outside participation. Dr. Thompson, Union of Concerned Scientists, other groups, had only received the draft consequence study when you had your meeting.

When you issued your report approving that study, they were still in the middle of reading it -- July 18th. With this meeting today, you know, it was my understanding that Dr. Thompson would have 50 minutes to present his views to you.

He spent a lot of time -- you know, we commissioned him to do a thorough analysis of this study. He found it had serious, serious shortcomings, and we asked for an opportunity for him to present his views to you today because he was not able to address the ACRS in the July meeting.

And he was later told, "Well, we'll see if we have time for you in the public comment period." That is not the kind of discussion and debate and exchange of ideas and information that is required for such an important issue.

So I have written a letter to you all -- I hope you have a copy of the letter -- asking you to please reopen this issue, have another look, take -- have a discussion with Dr. Thompson, have a discussion with Dr. Lyman and David Lochbaum about their concerns about this study.

For instance, how this study could use the term "low density storage" to describe high density racks with less fuel in them. It has a tremendous impact on the outcome of the study, but it is not talked about. It is an assumption that's buried in that draft consequence study.

So I don't want to take any more time, because I do want to share it with Mr. Alvarez and Dr. Lyman. But I just cannot urge you enough to reopen this study because we are counting on you.

Thank you.

CHAIRMAN ARMIJO: Okay.

MR. ALVAREZ: Very briefly, my name is Robert Alvarez. I'm a senior scholar at the Institute for Policy Studies. And just to pick up where Diane left off, my colleagues and I formed a working group in the summer of 2002 and issued a report, a peer review journal, looking at what the U.S. consequences would be to -- regarding an act of malice or acts of malice for spent fuel pools, particularly high density pools.

We looked -- we made a recommendation that these pools -- that the United States develop a more sensible and safe storage policy and return the pools to their original purpose, which was open rack, short-term cooling for one core. We also suggested that this could be done within a timeframe of 10 years, and could be done for somewhere in the range of $3.6- to $7 billion.

Subsequently, the Electric Power Research Institute came out with a report in August of last year indicating that it would be about $3.6 billion.

In looking at the draft study that has been presented to you by the NRC staff, or just to follow up on that, our report I guess, in lieu of better words, got us stricken from a lot of Christmas card lists. And it was not well received by the Nuclear Regulatory Commission or the industry, but it did cause a sufficient level of controversy where the National Academy of Sciences was called in to more or less referee this dispute.

And as some of you may know, the National Academy did release a report about a year later pointing out that, yes, indeed that our concerns are warranted, and that dealing with acts of malice against spent fuel pools would be very -- should not dismissed out of hand as the Commission has done.

Moreover, the NRC pointed to something

which this study has failed to address which is the risk of partial drainage. It has been assumed that there will not be oxidation sufficient to ignite the spent fuel during a partial drainage event.

We are not certain where that ignition temperature might be, depending on the level of drainage. Some say it's between 20 percent and 80 percent, somewhere in that range.

This study presumes that the pool itself will -- as it drains will remain in a confined environment, and, therefore, things like roof failures or entry of oxygen from the outside is not necessarily going to occur, which might enhance an ignition event.

The study also does not address what we had originally recommended, which was a comparison of the -- or at least it did not look at a comparison of open frame storage versus high density. It just simply looked at thinning out the existing high density storage racks.

The Academy and we pointed out that these racks interfere with convection, and can enhance the heat buildup in the spent fuel pools.

Finally, in terms of the regulatory analysis, this analysis looks at a timeframe of transfer of five years. We thought that -- in looking at that, when we did that study -- that that was a very unrealistic assumption, that this should be done over a period of 10 years, and that -- because of the availability of casks, the logistics of doing something like this.

And assuming -- I mean, not to be too polite, or impolite, I kind of looked at that as sort of like moving the goal line to the 50-yard line in this study. And I think that it really alters the cost-benefit analysis and the backfit analysis if you look at the recommendation we made with respect to a 10-year timeframe versus a five-year timeframe.

That's all I have to say. Thank you very much.

CHAIRMAN ARMIJO: Thank you, Mr. Alvarez.

MS. CURRAN: You have a little time, Dr. Lyman, for some comments.

DR. LYMAN: Thank you. I know there is not much time, so I'd just like to make two remarks. One, the issue of the SOARCA study came up, and I'm pretty puzzled by the different response that this Committee seems to have had to the current spent fuel scoping study as opposed to SOARCA.

When SOARCA was presented, this Committee was extremely critical of its methodology. It raised a number of important points that the study ultimately had to address, including incorporating an uncertainty analysis.

I don't see any of that skepticism being brought to bear in the current study, even though one might argue that it is even more inadequate, less complete, than SOARCA was in trying to actually do something resembling a complete set of initiating events and consequences with adequate uncertainty treatment.

So I don't know why the Committee now seems so willing to accept the outcome of this spent fuel study without at least applying the same standards that it did to the SOARCA study.

The second point I'd like to make is if there was any issue that seems to be appropriate for evaluation in a revised regulatory framework, that would be the issue of expedited spent fuel transfer. As you know, the near-term task force recommended a revision to the regulatory analysis guidelines that would incorporate greater emphasis on defense-in-depth, for example.

Now, at the same time, the Commission is evaluating different endpoints, including land contamination, the economic consequences, and to an extent that is not currently considered in the regulatory analysis.

It seems that if this issue of expedited spent fuel transfer were evaluated in the context of the revised regulatory framework the near-term task force had contemplated you might have other considerations that might lead you to a different conclusion. In particular, the defense-in-depth, the staff said that spent fuel storage currently has a lot of defense-in-depth.

I would say that the benefit of expedited spent fuel transfer to take -- to compensate for the uncertainties that are not being taken into account in the spent fuel study are valuable. And so that the defense-in-depth of thinning out the pools, reducing the source term, if there is a zirconium fire, reducing the possibility of a hydrogen explosion if there is a spent fuel fire.

And this is barely mentioned in the regulatory analysis, but the spent fuel study points out clearly that there is only enough hydrogen to cause an explosion in the high density fire scenarios. None of the low density scenarios generated sufficient hydrogen for explosion. It seems that that in itself is a qualitative aspect that you could consider.

So one last point on whether there are consequences that are not being adequately taken into account in the current regulatory analysis framework, I just beg you to look at what happened on the ground at Fukushima. You can read The New York Times today and read about the real impact of the release of probably 10- to 20,000 curies of cesium on the ground there.

Most of the cesium blew out to the ocean. Compare that to the enormous amounts of cesium that are being predicted to be released in the high density scenarios, and I would say that the tunnel vision of simply looking at the numbers that are being presented in this cost-benefit analysis do not give you the whole picture.

And I'll stop there. Thank you.

CHAIRMAN ARMIJO: Okay. Thank you very much.

All right. With that, we are behind schedule. But that was our fault. So let's take 15 minutes and be back --

PARTICIPANT: Sam, on the line --

CHAIRMAN ARMIJO: Oh, I'm sorry. Yeah. I think Mr. Kessler was on the line, but I don't know if he wants to make a comment. Is the bridge line open? Let's make sure. Thanks for reminding me.

(Pause.)

While we're waiting, if there's anyone else here in the room that would like to make some comments, please step up and identify yourself. If not, let's -- Mr. Kessler?

MR. KESSLER: I'm here. Can you hear me?

CHAIRMAN ARMIJO: Yes. Loud and clear.

MR. KESSLER: Okay. I just -- I want to just keep it real brief since you're running behind schedule. Yeah. We did do a study that I believe that you now have. Our study was limited to, you know, what are the costs and benefits of moving fuel five years old or older out of the pool.

We looked at dollar costs, we looked at increased cost to workers. And, in terms of benefit, we looked at the amount of spent fuel that would be removed from pools, the amount of decay heat that would be removed from the pool, the reduction in cesium source if we took five-year old or older fuel out.

And it's based on the assumption that it would take 10 years or 15 years, due to operational limitations, to get fuel that old or older out of the pool.

And just for the sake of getting you back on schedule, I'll stop there.

CHAIRMAN ARMIJO: Okay. Thank you very much. I believe we have -- no comments form the room. We've gotten the bridge line --

MR. KRAFT: Mr. Chairman?

CHAIRMAN ARMIJO: Yes.

MR. KRAFT: Thank you. Steven Kraft, Nuclear Energy Institute. At the risk of overstaying our welcome -- different topic.

The horrible events of Fukushima Daiichi, particularly Unit 4, are unfortunately the best laboratory we have for looking at this. And I guess maybe it's a matter of judgment/opinion, but it seems to me that that plant got hit with the fourth largest earthquake in recorded history.

You then racked it with a massive hydrogen explosion and collapsed one of the concrete walls that protect the pool liner. Is that a great day? Of course not. But it does demonstrate, we think, the robustness of these structures.

Since that time, and while the NRC studies as described only took into account what we refer to as the B5B capability post-terrorist attack, ability to put water in a pool, we have significantly enhanced that with our FLEX capability, we have added temperature requirements, there is a level requirement that NRC has added.

So I think at the end of the day our view is that you've got a robust structure protecting the fuel already, and we have the ability to deal with an event that we don't know what that event turns into, so we deal with this.

I just thought I wanted to put that out there just to sort of wrap up what our view was. And that's all contained in our letter to NRC on the topic, and I thank you for --

CHAIRMAN ARMIJO: Okay. Thank you very much.

Okay. Let's take a 15-minute break. Let's reconvene at 4:00.

(Whereupon, the proceedings in the foregoing matter went off the record at 3:42 p.m. and went back on the record at 4:00 p.m.)

CHAIRMAN ARMIJO: We're going to reconvene, and Dr. Steve Schultz will lead us through the next presentation.

Steve?

MEMBER SCHULTZ: All right. Chairman

Armijo, the Fukushima Subcommittee met on September 18th on this matter, and I want to just go through a couple high-level points before we turn to the presentations.

The staff has worked with industry and members of the public in several public meetings conducted before and after the order was published to develop the industry guidance that we're going to be discussing today.

The Interim Staff Guidance endorses, with certain conditions and exceptions, the NEI document 13-02 which provides the detailed guidance approach to implement the order. This Interim Staff Guidance addresses the Phase 1 Program within the order to address wet well venting enhancements. Approaches to address dry well venting guidance requirements are being addressed in Phase 2, and that's a continuing effort. So this is not a stopping point, but a point of delivery for the wet well venting guidance so that the phased scheduled milestones will be met. The approach the staff and industry has developed to examine these issues and document the resultant guidance has been very effective.

As I mentioned, several public meetings have been held, about a dozen since before and after the order was issued. When completed, the NEI document 13-02 is designed to include the guidance for both Phases 1 and 2, and this has helped to identify the interplay between the importance of features to both wet well and dry well venting.

At the Subcommittee meeting the industry provided and the staff concurred with the key issues short list that is active for Phase 2 resolution, and we're going to hear about that more today. As a demonstration of the progress that is moving forward, we're going to hear in the discussion today some updates of resolution moving forward on that short list that have occurred in the last two weeks.

And I do want to mention also that at the Subcommittee meeting and as delivered and presented by both the NRC and the industry this effort has been represented within the industry panel here and within the audience a demonstration that the work is supported by all of the effective CNOs, the BWR Owners Group, operations and engineering support staff, which have contributed to this effort.

So with that, I'll turn it back to you for the discussion today.

CHAIRMAN ARMIJO: Okay. I think, Steve, will you take the lead?

MEMBER SCHULTZ: Do we have staff remarks, Dave Pelton?

MR. PELTON: Not at this time.

MEMBER SCHULTZ: Okay.

MR. PELTON: We'll address the Committee.

MEMBER SCHULTZ: Okay. Yes. Then the first presentation is by the industry, Sam, and so I'd turn it over to Steve Kraft.

MR. KRAFT: Well, thank you, Chairman Armijo and Members of the Committee. As has been said, I am Steve Kraft. I'm a senior technical advisor at the Nuclear Energy Institute. It is a great pleasure to be here. I'm joined at the table by several colleagues who I will have introduce themselves, but they are in the Leadership Group of the BWR Owners Group playing a key role in preparing the guidance. Why don't we start with Tom?

MR. PARKER: I'm Tom Parker. I work at the Monticello plant for -- it's an Xcel Energy plant and I'm the chairman of the BWR Owner's Group Fukushima Response Committee.

MR. KRUEGER: Good afternoon. My name's Greg Krueger. I'm director of risk management at Exelon. I also chair the Containment Strategy Subcommittee working for Tom.

MR. KRAFT: Thank you. We also have several experts sitting in the audience. Phil Amway from Nine Mile Point, Constellation Energy Nuclear Group, and Randy Bunt, Southern Company, Fukushima lead. They will be available for answering questions. Randy in particular has the latest draft of the guidance in his hands and as we go through this, if you have specific questions about language, we're going to look to Randy to help us out with that.

Just echo something that Dr. Schultz said, this is a stopping point, but it is by no means an ending point. The documentation and the discussions with the staff have advanced since the meeting of the Subcommittee and we will discuss some of that as well. But for the benefit of members of the Full Committee who were not at the Subcommittee meeting, we will be repeated certain matters.

Generally speaking, this has been one of the most cooperative efforts between the staff and the industry that I've personally been involved in. Numerous public meetings, lots of dialogue. If we were to write a description of all the discussions back and forth and all the changes we made to NEI 13-02 since the beginning of this effort, we would write a document twice the size of NEI 13-02. That's how extensive it has been. And I think we have achieved good alignment with the industry view and the NRC. And there are two topics that are still currently under discussion that we will go over with you. And the staff was kind enough to share their slides with us in advance. We know they'll be talking about them as well.

As I was explaining to Tom prior to the meeting. The difference between "agree" and "align" is that align is you have a path forward. And I think that's where we are on these two issues, so I'm pleased to report that.

Reacting to a question that we were asked at the Subcommittee, the industry is working towards a common understanding of the elements that the vent system design should contain. We have announced our workshop November 12, 13, 14 in Baltimore. The 13th and the 14th will be devoted to a specific engineering and design set of discussions with the folks from the industry who actually have to do that work, vendors and what have you.

If I can have that next slide, please? At the time of the Subcommittee meeting and the issuance of the ISG, there was a number of issues, pretty much six in number, that we identify here as having been under discussion at the time. And so just to give you a status report, you can see that we've resolved one of them. What we mean by "addressed" is that NRC has indicated how they are handling the issue and we are satisfied that that is the correct way to do it. For example, the EPG/SAG, which are the guidance from the Owner's Group to the industry on writing procedures, emergency procedures and severe accident procedures, NRC staff has said, well, in the context of NEI 12-02 they're not going to endorse that, which is fine by us. So that's the point, is that they said they would do something different than we had originally thought. We think that's just fine. That's what "addressed" means.

Generic Letter 89-16, again we think it ought it to be rescinded. NRC staff said it's fine, but our document says you don't have to pay attention to it anymore. So it's the same sort of idea.

But the issues that are before us, and I don't want to mention this in a way that sounds like we are at vast differences here, it is just discussions that have not quite come to conclusion yet, and that is the dry well temperature design value. And we will talk about that at some detail here. How we're handling anticipatory venting as part of the FLEX resolution.

And this last issue which was discussed at the very end of the Subcommittee meeting, there is a statement in the ISG that takes the statement in the order, expands it a little bit about the use of a dry well vent with an engineered filter as a way around many of the requirements in the order. Our view is this is completely misplaced. We said so at the time. We were very open about that. We believe this is something that belongs in the rulemaking that we are not here to discuss today. Our comments on this will be in our former comment on the ISG.

Please do not take that as a point of serious contention. It is just we think it doesn't belong there. NRC says, well, it's there. It doesn't make a difference. So we'll just sort of work it out. But I just didn't want to leave you thinking we had come to some kind to resolution.

At the Subcommittee meeting there were a number of issues that were identified in discussion, and we wanted to report back how we dealt with those going forward and seek input. Also, if there's something that we missed, please let us know. We'll be happy to address them in the questions.

There was a lot of discussion about how

we've engaged the industry in developing the guidance. And as Dr. Schultz reported, we do have alignment -- I should say agreement in the industry. It is important to know that the BWR Owner's Group, which represents all the BWRs in this country and many abroad, had extensive meetings in July with the details on the table, with the people who need to deal with the details. And there was, you know, a lot of review input and changes as a result of that meeting.

We've had a lot of interactions among the different committees in the owner's group. We have a working group at NEI. It's more of an umbrella activity. And we've worked a lot with the people. Greg chairs a subcommittee, Tom chairs a committee, Randy's involved. There are other committees. I don't mean to bore you with the structure, but we did bring in a lot of people to bring views in on this very complicated subject.

Tom will talk later about anticipatory venting, but just to say we are using the generic issue resolution process that the Japan Lessons Learned Directorate has developed for this. And we will then also discuss a little further containment accident pressure. And just to point out here that in NEI 13-02 the way it stands now, there is protection for inadvertent actuation, and that of course is inadvertent actuation of the vents always being in place, but we included here -- and that of course protects the CAP capability when the vent is not needed.

Of course the flip side is that if you're venting you don't have power to run those pumps anyway, so it's not like it becomes an issue. So that's something that we would -- and the procedures require we re-closing the vent if you regain those capabilities.

All right. At this point let me turn it over to Greg.

MR. KRUEGER: Okay. Thanks. What I'm going to go through is some of the changes that we've agreed upon with the staff in the past seven or eight working days since the Subcommittee meeting and then talk a little bit about the dry well temperature capability of the hard vent.

The vent itself, this is a very unique engineered feature. It is something that's used for different modes of operation, if you will, saturated conditions all the way up through severe accident conditions where there might be more than steam, but steam and hydrogen and radionuclides, as well as the higher temperatures. So we've worked to make sure that the criteria in the document are such that we can assure that the vent is capable of operation under all of those conditions.

As noted here, we improved the severe accident definition. Up in the frontispiece of the document in Section 2 there's a discussion about severe accidents, those that generate core damage and subsequent hydrogen and radionuclides. With the staff we did make this definition a little more accurate with regard to the spectrum of severe accidents that could occur such that the vent would need to handle all of those.

Prior to this refinement basically it was looking at core damage retention in-vessel and the core damage with ex-vessel poured material on the floor. And in fact there could be a spectrum between those two that could exist that could create conditions that we need to deal with. So just the refinement.

We did clarify -- we did have a statement with regard to components. In a number of places in the document, maybe 15 or 20 sections, we had just the generic word components. With the interaction with the staff tried to make it much more clear in that components and instrumentation need to maintain their function for operation for the vents. So not just components as a generalization, but make sure that the instrumentation that's used to tell the operator when to vent, as well as any controls need to follow the same rules/design criteria as that of the pipe and valves and any other equipment.

We did correct -- in the overview section of 1.4 there was an introduction section that just mapped incorrectly, and that's pretty much an administrative issue. And the discussion on the dry well vent design and its operation and capability.

After the discussion we had a couple weeks ago, we did use a diagram which is two slides hence that we thought was useful in communicating the concepts of how we design the vent and how we then tried to assess its capability, which is beyond that design, and thought the document itself just in text didn't do it. After that discussion I had spurred us to put that diagram or a facsimile of that diagram, adjusted a little bit from a few weeks ago, into the document itself, which I hope is a useful addition to the designers.

MR. KRAFT: I would like to say at this point that for those of you who were not at the

Subcommittee meeting, it's sort of like you came in at the fifth reel of the movie here, the background on what's in the guidance and what the philosophy for that, those slides are included as background here. We thought it would be more important to get to the hot issues that we all talked about.

MR. KRUEGER: So with regard to why are we talking about dry well vent design temperature, as Dr. Schultz had mentioned, that's really a Phase 2 issue. But we do realize from a design perspective there's an interface or a likely interface between the wet well hardened vent and the dry well hardened vent, and that interface and the valves and the equipment that need to exist on the wet well hardened vent under Phase 1 do need to be designed as if the conditions in the dry well for severe accidents existed. So we have to make sure that the vent valve that separates the dry well and the wet well -- it may see temperatures, high radiation profiles on the back side of that valve that in fact we need to know from a design perspective, or at least feed that information in from a design perspective to assure that it would operate.

Again, this document will be revised when we do get to Phase 2 to be more encompassing of the dry well vent design.

But four concepts here is that we are picking a design value. We understand that yield or failure is beyond design. Normal engineering processes are such that we do not design to a failure point. We design to a point and then typically failure or yield is higher than that. And we're trying to assess or show that there's a capability range beyond the design that this hardened vent should be able to handle.

Since we are talking --

MEMBER CORRADINI: Greg, can I interrupt you?

MR. KRUEGER: Go ahead.

MEMBER CORRADINI: I want to make sure I understood what you just said. So you're eventually going to get to a number, but I want to understand. So you're trying to identify a number that's beyond the design value but does not necessarily guarantee failure. Is that I just heard you say?

MR. KRUEGER: It's beyond the design envelope of containment.

MEMBER CORRADINI: Right.

MR. KRUEGER: And it will have additional capability beyond that design point.

MEMBER CORRADINI: So for want of a better word it's not the design value, it's not the failure value, it's somewhere in between?

MR. KRAFT: Well, the number will be the design value. Designed to a number. And that design value, you'll see on the next diagram, is higher than the containment design value.

MEMBER CORRADINI: Understood.

MR. KRAFT: And there's a capability that goes beyond that, and that's demonstrated in the next slide.

MEMBER CORRADINI: Okay. All right. I'll just stop. I get it now. Thank you.

MR. KRUEGER: And as mentioned, the third bullet, the temperature that's being selected is significantly higher than the design value of containment. Mark I containments are typically designed to 281 or 340. We're talking about 545 degrees, as well as containment pressures above the containment design pressures that typically exist for Mark I containments.

Of importance and to note at the bottom here is that we do not expect any testing of the containment or vent components that will have to show this ultimate capability. In other words, we do believe that by picking a high enough design value that in and of itself it will have the capability to go beyond that of the containment. It will at least meet or exceed the components of the containment as they now exist.

MEMBER SKILLMAN: How will you know from one fuel cycle to the next that this equipment will operate?

MR. KRUEGER: There is a section in the ISG or in the guidance document that talks about all the testing requirements and the checks that we need to do when we shut down.

MEMBER SKILLMAN: I'm just reminded of the plant operators that said we really don't have to test that equipment. And so there it sat for 16 years resting and seizing. And when it was finally called on to operate, it was either filled with clams or mussels or rust or something. And so the component they were depending upon failed. There needs to be some exercise at some frequency that gives the operator bold confidence that the device is going to function the way they intended it to function.

MR. KRAFT: All of the Tier 1 Fukushima orders have in them a requirement for that kind of periodic testing. Spent fuel instrumentation. It's every cycle. FLEX, you have to drill. There's been a lot of discussion about, for example, the role of the isolation condenser at Fukushima Unit 1. In this country my understanding is is that operators have to understand the isolation -- it's never called into account really of the accident. But they know what it is. They know what it looks like. They know what it sounds like. They know what they have to do to keep it full. So I'm just saying that this is a common thing now in the industry to do exactly what you're requesting, and it's built into our presentation.

CHAIRMAN ARMIJO: I think we've got an operator comment.

MR. AMWAY: Good afternoon, my name is Phil Amway, Constellation Energy, and as far as testing we do have built into the Guidance 13-02 testing requirements for the HCVS System. One of those tests is to make sure that we cycle the HCVS valves, inter-spacing system valves used to maintain the containment integrity during operations with a frequency of once per operating cycle. And there's a matrix in there that defines that testing to make sure that the system has functional capability when it's called upon to be used.

MEMBER SKILLMAN: Thank you.

MR. BUNT: This is Randy Bunt. I think

what may have been misunderstood on that previous slide is we were talking about there's a requirement to test containment valves, the things we're putting in here, to a point of failure. Right, that's last bullet is to say we don't believe this should be a testing program, that we don't want to create a new testing program to show these components have a failure at this point. If they're designed at 545, they'll fail at 950 degrees. That's what -- that last bullet is not talking about the operational test of a vent system throughout this life of that vent system.

MEMBER SKILLMAN: Thank you. Got it.

MR. KRUEGER: Okay. This diagram is a stylized diagram and it is a composite from a number of sources of information with regard to containment, containment failures and the capability of containment. A lot of this information came from the 1980s, 1990s. There were some Sandia tests on containment. There's tests on containment penetrations and elastomers and what they can hold in terms of pressure retention, as well as temperature and degradation as a result of temperature.

What we show here on the bottom or the left corner is the design envelope of the containment itself. Most Mark Is are 56 to 62 psi containments that go to 280, 281 to 340 degrees. That's the typical design envelope for a Mark I containment. However, when we take a look at all of the tests and information with regard to what a severe accident could impose upon the containment, we find that the failure or the capability of containment is actually much greater than that design envelope. It really is way up toward this red line that goes across the top and angles down to the right.

And in fact the capability, what we're pointing out with these numbers here is that if we picked a high-design point for both temperature and pressure for the containment itself as point No. 1, we can compare that to point No. 3, which is the pressure and temperature capability of containment, and look at that range and understand that that range is well beyond what the design parameters were originally for the containment.

MEMBER POWERS: When you said point No. 3, did you consider your elastomeric seals to have both a dose and a temperature on them?

MR. KRUEGER: In the studies they were mostly temperature and pressure.

MEMBER POWERS: In fact the point No. 3 at 700 degrees Fahrenheit was strictly temperature. That was the INL test.

MR. KRUEGER: That's right. That's right.

MEMBER POWERS: What happens when you take into account dose?

MR. KRUEGER: I did ask GEH last week with regard to that, and in fact what I'll call the primary drivers for failure were temperature rather than dose. In other words --

MEMBER POWERS: Do you have data to back that up, because I think I have Japanese data that suggests that dose is very important. And we have access to that because they published it in the open literature.

MR. KRUEGER: We don't have access to the Japanese data, no.

MEMBER POWERS: They published it in the open literature and it would suggest that that point three is strongly dependent on the dose.

MEMBER BALLINGER: Does it depend on the material?

MEMBER POWERS: Of course it does. The material is constant here.

MEMBER BALLINGER: Well, I haven't seen to many elastomers that will run at 700 C.

MEMBER POWERS: No. Yes, they ran -- yes, in the course of doing the NUREG-1150 Study, venting in the head seal was identified as a potentiality. And what they found is that between Browns Ferry and -- I want to say Peach Bottom, but I'm not sure about that, that the head bolting was different. And so they ran some experiments. They were Sandia experiments that were run at INL. I can't imagine those two organizations interfacing at all for doing a test, but that's how it was done. And, I mean he's seen that.

MR. KRUEGER: Yes, right.

MEMBER POWERS: They're decent enough tests.

MR. KRUEGER: I will point out that this is a range. I mean this is stylized --

MEMBER POWERS: Yes, I mean it's a stylized drawing, but --

MR. KRUEGER: -- but there's a range of failure that goes backward in temperature as well.

MEMBER POWERS: But the Japanese did some experiments that are really quite interesting on the head seal elastomers in which they looked at temperature and nitrogen, temperature and steam, temperature, steam and dose. And that combination of steam, dose and temperature I think is fairly devastating for these elastomers.

Now of course there's another dimension here that we're making a projection because there's time involved here, too. And I don't know where you are on your time schedule, but I mean you put a lot of information, and I appreciate that.

MR. KRUEGER: Right.

MEMBER POWERS: But I wouldn't gain a lot of confidence about point 3 as some sort of fixed margin that you would have there because of the dose effect on --

MR. KRUEGER: Yes, definitely it's not fixed. I mean there is probably a range there.

MEMBER POWERS: The problem is that if you get dose-resistant elastomers, they don't work very well for sealing purposes and vice versa.

MR. KRAFT: So what that suggests, Dr. Powers, is that you have to set the operational point low enough so you never get into that test, or the test range, right?

MEMBER POWERS: And it's a fact that we know we do that --

MR. KRAFT: Right.

MEMBER POWERS: -- because we test the damn things every time we turn the plant on.

MR. KRAFT: And you'd see -- I think Randy is about to talk about the way the language of the -- of -- based upon this information, a combination of design and operation will keep us down to the left of all these numbers so we know we're in sort of a -- call it a safe range, if you want to.

Randy?

MR. BUNT: All right. This is Randy Bunt from Southern Nuclear. One point that I did want to make is that we are talking about the full blue range here as where the head gasket area would be -- start seeing leakage, and realize that that is a metal-to-metal-type surface with bolt tightness and stretch on it. And you've got to wait until that stretch gets elongated before the elastomer actually is going to be exposed to the temperatures in other regions. So that plays another factor into why this is a higher value before you see that versus a strict O ring failure point.

MEMBER POWERS: That depends on whether you're dry well cooler is operating or not.

MR. BUNT: From a --

MEMBER POWERS: It gets toasty up there if the dry well cooler is not on.

MR. BUNT: Yes, I don't disagree. You're

saying that this --

MEMBER POWERS: Then you get a thermal load on that elastomer. It starts squeezing.

MR. BUNT: Correct, inside the groove. This is a double-groove O ring that's metal-to-metal contact on either side that's pulled down. So, yes, the O ring will provide some benefit, but the main benefit for the sealing of that head is the metal-to-metal end of the bulwark that is going to get there. And this is a graduated leakage probability. As that temperature approaches to zone 3 and where we're planning to give direction for operation is to stay below those ranges so you don't start getting compromise. Again this diagram is primarily to show the deviation between a design input, a capability value and then use this to some degree to indicate where the operation, proper operation range would be, which is really still down in the design envelope window.

So I think that was the one key thing that we learned from our 23rd meeting with the staff is that we're talking three relatively new parameters that were intertwined with each other. One is where do you design it so a procurement engineer or a procurement person can buy something from a vendor and design it? Where do you want to say it would fail so that you make sure the vent's not the last component standing when everything else around it has already self-vented? And then where do you operate the vents so that you stay away from any of these colored regions that start giving you compromise of components?

MR. KRAFT: Very good. And looking at NEI 13-02, NEI ISG, the primary metric for protecting containment is that dome seal. So everyone is focused exactly on that problem.

MEMBER POWERS: Yes, what I don't understand is why they don't focus on that. I mean what gets you into trouble is failure of the dry well cooler up there.

MR. KRAFT: Looking at dry well cooling is contained in the rulemaking. That's one of the things we are considering. So that's not an issue we're ignoring, but just not part of this.

MR. KRUEGER: Right, we're trying to get initial guidance. And certainly Phase 2 will get into the stratification, right?

MEMBER POWERS: Well, I think I made my point that --

MR. KRUEGER: No, you did. I think we

agree, too, yes.

So again, going back to trying to design to something that designers and manufacturers can reasonably manufacturer, we're picking design points of PCPL, which is the primary containment pressure limit, which is a parameter that's calculated but very close to of the design pressure of containment and 545 degrees, which is also a calculated range in the EPGs. The reason we selected these is that the BWRs already, through the use of the EPGs, understand this envelope and use this envelope for what I'll call containment venting and containment venting strategies to make sure we don't challenge this point, point No. 2.

What we're trying to do with point No. 4 is show that there is some point again beyond its normal design that the dry well vent could be compromised based on high pressures and high temperatures, but in fact it is not the design point. And there's some range of capability there as well.

Again, the operational piece, which isn't shown on here, as Steve mentioned, really drives venting, the anticipatory venting we're going to hear about and any venting during severe accidents well into the -- for the design envelope side of this so that the operational piece along with the design aspects will keep us in an area in which we can assure that we can use the equipment as needed to mitigate the event.

So at this point let me turn it over to Tom Parker.

MR. PARKER: Anticipatory venting. Before the events in Japan I guess my personal thought on the containment vent was it was a great tool to protect containment from over-pressure. Subsequent to that I've gained a great appreciation for its ability to remove heat from the containment also.

MEMBER POWERS: Everybody goes through that lesson, don't they?

(Laughter.)

MR. PARKER: There are also many other features that we found that it helps us with, too. We have another group in the BWR Owner's Group that gives us guidance on the emergency procedure operation, and they've proposed a revision to that to suggest that we should lower the pressure than when we would be able to open the vent, provided the conditions merit that. And that's what we're referring to as anticipatory BWR venting.

The procedure changed that has been recommended by that committee to us to all the BWRs.

It is to change the setpoint or the -- the technical term is an "override" in the procedure that allows the operator, if certain conditions exist, to open the vent. And those conditions are if the containment pressure is greater than the SCRAM setpoint, which is roughly around two psig, and also that we don't have any heat removal terms or heat removal capability. Of course normally the tools that we would use to remove heat from the containment would be RHR pumps. And during this event of course we don't have RHR pumps because they're driven by electric motors.

So if you meet those conditions where you don't have those pumps available and you're above the two-pound setpoint, then it is up to the operator to decide when they're going to open the vent. But they're permitted by the procedures to open the vent there.

MEMBER CORRADINI: Can I ask a question about that, because I seem to remember when we were visiting Peach Bottom this was discussed, at least kind of in passing.

So that's the allowable start point. Is there a must-open point?

MR. PARKER: The must-open point would be as you're approaching PSP, the pressure suppression pressure, which is somewhere on the order of 25 to 30 pounds, psig.

MEMBER CORRADINI: Okay. And I guess I'm not enough of a BWR person to understand why that --

(Simultaneous speaking.)

(Laughter.)

MEMBER CORRADINI: No, but at this point though the assumption is saturated at those conditions. Is that the assumption?

MEMBER POWERS: No. No, deliberately not saturated.

MEMBER CORRADINI: Okay.

MR. KRAFT: If we could have Phil Amway talk to --

MR. AMWAY: Yes, this is Phil Amway, and my background was also -- has been in operations. And the reason why you want to vent to maintain below the pressure suppression pressure is for a variety of reasons. But if your plant conditions drive you to perform an emergency depressurization of the reactor vessel, then having the pressure in containment low enough; i.e., below the PSP, would make sure that the containment can receive the blowdown, the high energy from the reactor pressure vessel to prevent exceeding the design pressure of the containment.

MEMBER CORRADINI: So this is just the delta-H, the pre-calculation of what the delta-H is? Okay. Fine.

MR. AMWAY: That's correct.

MEMBER CORRADINI: All right. Thank you.

MR. PARKER: So one of the other aspects that the anticipatory venting will do is it will prolong the operation of the RCIC operations, since RCIC cooling is done by torus water and keeping and maintaining the torus water cooler provides cooler water to RCIC, which extends its operation.

We're in the process of working out a white paper with the staff to talk about anticipatory venting, some of the advantages and how our procedures will make sure that it's properly used. In fact, we were just about an hour ago or so meeting with the staff to get some additional comments on that white paper, which we'd be glad to make available. I think the Subcommittee mentioned you had some interest in that, so we'll certainly make that available to you when we get those comments resolved and issue a Revision 1 to that white paper.

Again, talking about some of the advantages of it. We get some core cooling out of this, because the problem here in this event is to get the decay heat out of the containment. And so the containment vent does a very good job of doing that. It provides also a pressure margin by reducing the containment pressure, staying away from the limits there, providing us margin, and the operator more time. We're using installed equipment as opposed to portable equipment at this point in the event, and that's the advantage of prolonging RCIC operation. We can certainly use portable equipment if RCIC should not be available, but our preference is to have the operator use the equipment that they're trained with. It's installed in the plant, and using the vent supports that. Also of course we're taking advantage of the late heat evaporization, which is a very great heat removal term for us.

Again, talking about referencing the Subcommittee discussion, we had some questions on containment accident pressure. And the override that I mentioned earlier specifically addressees that by saying you can only use that override if you don't have the normal core cooling functions available.

I guess one other thing to talk about is that generally when we open the vent, depending upon the size of the vent -- but in most all cases the pressure does not immediately drop to zero. It's basically when we open the vent we're going to leave the vent open for awhile so it's not going to -- because just to relieve the flow of steam out of there. There's a lot of energy to get out. So the pressure does not drop back down to zero right away. So there will be pressure available in the containment following the vent opening for some time. And furthermore, the venting capability of course is going to be enhanced by the order from the ISG that Greg was addressing.

So the bottom line is we feel that venting the containment is very advantageous for maintaining margin, giving the operator time to address the situation by extending RCIC operation, and removes the decay heat from the containment.

MEMBER POWERS: Even by assumption there's no dry well spray at all like this?

MR. PARKER: Well, the dry well sprays normally are supplied by pumps that are not available then. There is some talk of possibly using the portable pumps to supply flow through the dry well spray header.

MR. KRAFT: One thing we're looking at in the rulemaking is -- I hate to use the term FLEX-plus, but a FLEX-like capability to inject water through the spray headers in an ELAP circumstance if you then have a severe accident.

MEMBER POWERS: Yes, I mean the attraction I suppose of course you get a substantial amount of decontamination --

MR. KRAFT: Oh, absolutely.

MEMBER POWERS: -- associated with them. And they're very good. And the headers are very capable in the Mark Is. I really don't have --

MR. KRAFT: Unfortunately, they're not as high up in the barrel as you might like.

MEMBER POWERS: Well, yes. Well, the upper one is.

MR. KRAFT: Yes, a few models.

MEMBER POWERS: But I mean the problem is that it doesn't have very much spray, lateral through flow and --

MR. KRAFT: These are the questions we're looking at in the rulemaking. The photographs that Greg's provided us show how crowded it is up there.

MEMBER POWERS: Yes, but that interface of the spray and the emergency operations in the dry well vent, you know, I mean that needs to be worked out, and it needs to be worked out in the guidance, not in the rulemaking.

MR. KRAFT: Well, there will be guidance with the rulemaking, too, that, you know, we anticipate -- we talked about this with the Subcommittee, that water will get you so far. Then the rulemaking will complete and there will be some processes, analyses that will lead individual utilities to make determinations as to how they would manage a severe accident. Whether it would include a filter or not becomes their choice. This question of how you control an event with the water injection, how much water you inject. One of the great learnings from Fukushima is water control will become very important, probably something we've never really looked at. You know, Three Mile Island, as we know, 350,000 gallons of water. We felt we were drowning. Look what's going on over there in Japan now.

MEMBER POWERS: They're drowning.

(Laughter.)

MR. KRAFT: Water management becomes a much more important factor that we recognize we want to deal with in the context of the rulemaking.

MR. KRUEGER: I will say that the current EPGs do have spray limit curves, and they don't allow the operator to spray down to zero pounds either. I mean there's certainly a range that you want to stay above zero.

MR. KRAFT: Mr. Chairman, that completes our prepared remarks. As I said, you have background slides for more of the general information about the guidance. We're more than happy to answer some more questions or to yield the table to the staff.

CHAIRMAN ARMIJO: Questions?

PARTICIPANT: Maybe we can wait until the staff's presentation.

CHAIRMAN ARMIJO: Probably a good idea.

MR. KRAFT: We'll be here so we can answer questions.

CHAIRMAN ARMIJO: Okay. Thanks very much.

MR. PELTON: Yes, Mr. Chairman, this is Dave Pelton, the acting deputy director of NRR's Division of Safety Systems. And I just want to take a minute to say thanks to you and the rest of the Committee for taking the time to be us today so that you could hear about the staff's good work in developing an Interim Staff Guideline that will provide a means of assuring consistency with the order and will endorse the industry guidelines that you just heard discussed.

Consistent with the message we heard from the industry, we also appreciate the open collaborative manner with which staff and industry have worked together to develop these guidelines. It's really important. A lot of challenging technical issues. And the dialogue has been really very good, and we really appreciate it.

I won't go into a lot of detail; you heard from Dr. Schultz and from the industry already, but I will say that you're aware that during the previous Subcommittee meeting there were a number of issues raised. Staff's prepared to talk about those today, you know, including the dry well temperature design value issues. Again, we're looking forward to discussion and entertaining any questions, comments or concerns you might have. I'll turn to over to Bob Dennig; he's the chief of our Containment Ventilation Branch, to see if he has any opening remarks.

MR. DENNIG: No, Dave, thank you very much. I'll turn it over to Rao Karipineni and Jerry Bettle to take you through the technical presentation.

MR. AULUCK: Okay. Yes, good afternoon. My name is Raj Auluck. I'm an NRC project manager in the Japan Lessons Learned Project Directorate within the Office of Nuclear Reactor Regulation. With me today are lead technical staff members Mr. Nageswara Karipineni and Jerome Bettle who will be presenting the bulk of the staff's presentation. Other members who participated in completing and preparing this Draft Interim Staff Guidance are present in the audience and are prepared to answer any of your questions.

I'll briefly go over the agenda and provide a brief overview for the benefit of the Full Committee Members.

As you may recall, Commission paper SECY-12-0157 was issued in November 2012. It incorporated comments from stakeholder and the ACRS. The SECY paper provided options to address questions about maintaining containment integrity and limiting the release of radioactive materials if venting systems were used during severe accident conditions.

The Staff Requirements Memorandum on this SECY was issued on March 19, 2013. In it the Commission directed the staff to take certain actions, and these are noted on this slide. It required licensees to upgrade or replace the reliable hardened vents required by Order 12-050 with a containment venting system designed and installed to remain functional during severe accident conditions.

Second, it developed a critical basis for filtering strategies, the dry well filtration and severe accident management of containments and to provide developed proposed and final rules and separately seek Commission guidance on user qualitative factors in regulatory decisions.

As declared in the SRM the staff engaged external stakeholder throughout the development process. There were five public meetings held between the issuance of the SRM in March and mid-May when the draft was completed. The Revised Order EA-13-109 was issued on June 6th, 2013. It included a two-phase approach to ensure implementation of adequate protection provisions and cost-justified enhancement with minimal delays. This order superseded Order EA-12-050. Since the requirements in EA-12-050 were also reflected in the revised order, licensees were no longer expected to comply with the requirements of EA-12-050.

And under Phase 1, which involves upgrading venting capabilities from the containment wet well to provide reliable severe accident capable hardened vents to assist in preventing core damage and if necessary to provide venting capability during severe accident conditions. As noted on the slide, the revised order added severe accident capability.

And this slide provides a timeline of

implementation in Phase 1. The integrated plans are to be submitted for staff review by June 30, 2014.

Phase 2 involves providing additional protections for severe accident conditions through installation of a reliable severe accident capable dry well vent system or development of a reliable containment venting strategy that makes it unlikely that the licensee would need to vent from the containment dry well during severe accident conditions.

In a parallel activity staff is continuing discussions with the stakeholders on filtering strategies and severe accident management which would assist in the development of a technical analysis in support of a proposed rule. The rulemaking technical analysis is to be provided to the Commission in December 2014 and the proposed rule in December 2015.

The next slide provides the timeline for the implementation of Phase 2. As noted on the slide, the integrated plans are to be submitted to the NRC staff for review by December 2015. Focus of today's meeting is on the Phase 1 of the order only.

This slide shows the schedule of the ISG development. The staff briefed the ACRS Subcommittee on September 18, 2013. The draft ISG was published on

September 18 in the *Federal Register* for public comment. The *Federal Register* number is 78FR57418.

Again, I would like to highlight the fact that we had substantial interactions with the stakeholders in developing the order and the Draft Interim Staff Guidance. As noted and Dr. Schultz mentioned, we had many public meetings. There at least seven public meetings between the issuance of the order and finalizing the ISG. In addition this topic was also discussed at Senior Managers' Joint Steering Committee meetings between the NRC and NEI. In all, since January 2013 we have had 14 public meetings and 4 Joint Steering Committee meetings. There was also one Commission meeting in January focused on this issue.

The public comment period ends on October 18, 2013. Our next step is to wait for the public comments and make changes based on the public comments, as well as comments received today from the Full Committee. We will schedule a public meeting if needed later this month to finalize the ISG.

As stated earlier, and at our Subcommittee meeting, too, there were a couple of issues which required further discussions. And it has been already mentioned that these related to temperature in the dry well and the level of details needed for the instrumentations. We have made progress on these issues and we'll highlight these issues in our presentation.

Also as mentioned earlier, the Industry Working Group under the NEI initiative volunteered to develop a guidance document for the NRC staff review and endorsement. The scope of the guidance document NEI-13-02 is broader than the scope of Phase 1 of the order. The Draft ISG is endorsing this guidance document with clarifications and exceptions.

With this, I will introduce Rao Karipineni, who's a senior reactor systems engineer in the Containment Ventilation Branch who will lead the staff's presentation.

MR. KARIPINENI: And Jerome Bettle will also assist me as I go along.

Next slide, please? The primary objective of the vent, from the very beginning it has been preventing containment failure from both over-pressure and over-temperature conditions. The initial 050 was only for before core damage. The order was revised to go into severe accidents, and we clearly stated that the severe accidents include a breach of the vessel by the molten core debris.

The severe accident conditions relate to the dominant accident sequences and we believe the most dominant failure of this failure mode of the containment from over-pressure and over-temperature is the failure of the dry well head flange seal. This has been predicted before by calculations, as well as what has occurred at Fukushima. So we believe that the severe accident capable vent also should be designed to, among the other things, prevent the failure of the dry well head seal. And it has another function, which is basically the assisting in the removal of the decay heat. The requirement we put in in the documents is one percent decay heat at the PCPL, but removing decay heat also involves allowing the operation of pumps to operate to inject into the vessel, etcetera.

Next slide, please? The Phase 1 we are working on is the wet well vent. Phase 2, the dry well vent or reliable venting strategies that makes it unlikely that dry well venting is needed. So the NEI guidance document has places to be filled later to include the guidance for a dry well vent.

MEMBER BLEY: Can you tell us anything about the kinds of scenarios where we might need a dry well vent and the wet well vent be effective for us?

MR. KARIPINENI: Well, I'll just go -- there are a couple of scenarios here. One of them is dry well vent during a flood up of the containment. You will need that because you will be over-pressurizing as it fills in. Secondly, a dry well vent has the benefit of reducing temperature or assisting in the heat removal directly from the dry well when the core becomes ex-vessel, which is then more likely to get heated very quickly. Before the core becomes ex-vessel you could remove a lot of heat through the wet well vent, but after that, you know, it has a limited --

MEMBER CORRADINI: So just to get to Dennis' question, there would have to be a failure of the wet well vent or you just simply fill the inventory that this is the only pathway out? I'm trying to figure out where I would give up on the wet well vent and move to that. And so those are the only two circumstances that come to mind.

MR. KARIPINENI: Fill-up with the containment and which means that a wet well vent is not available.

MEMBER BLEY: Fill up with water?

MR. KARIPINENI: And also, you know, when you have a need really to remove a lot of heat from dry well.

MEMBER CORRADINI: Again; maybe I've missed a path, the second would be some failure of the wet well early on to remove your decay heat through the filtering of the pool.

MR. KARIPINENI: Well, we are talking now about post-core melt, because you're not taking all the heat and putting through a safety relief valve into the wet well, which allows you to heat up the wet well and get the pressure out. The core isn't outside of the vessel and there's a lot of sensible heat coming into the dry well that's heating up the dry well now.

MEMBER CORRADINI: So a core melt for which the wet well wouldn't have been successful in preventing the core melt, I guess.

MEMBER BLEY: So that --

MEMBER CORRADINI: So still -- I mean just let me press the point, then I'll stop. But if I didn't have it flooded up with water, I still would rather have it all flow through the wet well. So either it failed or I flooded it up. I wouldn't want to preferentially take it through the dry well. That would be my last resort.

MR. BETTLE: This is Jerry Bettle. Until

you --

MEMBER CORRADINI: I want just to make sure I'm not missing something.

MR. BETTLE: Until you get the vessel breach, if it's coming through the SRVs, it's going to come through the pool anyway and then come back through the wet well or dry well vacuum breakers and then back up a vent from the dry well. So you're not going to lose that pool scrub until you get the vessel breach.

MEMBER CORRADINI: Right. So I'll let you guys go on. I don't mean to hold you up at this point.

MEMBER BLEY: I don't think this is spelled out in anything we've read.

MEMBER CORRADINI: Yes. I think the point that Dennis was asking, I was kind of thinking the same way, which is where must I retreat to this phase in the process?

MR. KARIPINENI: Those kind of details were the exact things that were supposed to be worked on in Phase 2 --

MEMBER CORRADINI: Okay.

MR. KARIPINENI: -- on the rulemaking relief. So there's all kinds of thoughts about it floating around, but there's no answer to that yet.

MEMBER CORRADINI: That's fine. Okay. Thank you.

MR. KARIPINENI: Next slide, please? The different timelines between Phase 1 and Phase 2. We are trying to coordinate the Phase 2 in the rulemaking process so it all can result into a cohesive set of requirements at a later stage.

The order has divided the requirements for the vent into three major categories: Performance objectives, quality requirements, and programmatic requirements. And I mean most of you are aware of what some of these requirements are. I just repeated some here.

Under performance we need to minimize the reliance on operator actions, minimize plant operators exposure, account for radiological conditions that would impede any personal response, and also controls and indications shall be accessible and functional under a range of plant conditions.

And then under the functional requirements there is a subcategory called design features. And this is where most of the technical stuff goes in. The vent capacity, the one percent capacity I just alluded to before, the effluent discharge monitoring, minimizing the unintended cross flow between units, etcetera, into the plant, capability to operate from the control room at a remote location with the minimum capability to operate for at least 24 hours by means of permanently installed equipment. Also we address the flammability of gases inside the vent. That needs to be taken care of. And it has the operation, testing and inspection and maintenance requirements also.

The quality requirements basically are divided into two parts: The containment isolation barrier itself and anything beyond the containment isolation barrier.

Programmatic requirements to develop, implement and maintain procedures, training, etcetera.

We believe all these requirements have been worked out basically to the satisfaction of the staff and the industry. So the remaining items that are left, or the small items that were left that the industry has talked about on the next few slides.

Next slide? You have already heard what the industry has said about the EOPs/SAMGs. And we have not really reviewed them. We don't have them really. Our intention is not to review them at this point unless something else develops later. Our most important thing here is the vent is designed to the requirements of the order. And we were really concerned about the amount for discussion that was involved there about EPGs and Revision 3s, etcetera that we haven't reviewed. We don't have them even with us.

So we have asked for a statement that the requirements of the order takes precedence over any other features like these that they may want to have. And they have agreed to that and that statement has been included into the guidance document.

MEMBER SCHULTZ: Now, what is the schedule for addressing that?

MR. KARIPINENI: Well --

MEMBER SCHULTZ: It's got to before the end of Phase 2.

MR. KARIPINENI: I don't have an exact answer because particularly are they going to give us those things, number one? Number two, this is somewhat not along the precedence that has happened before. The NRC hasn't reviewed SAMGs, etcetera, and has written a safety evaluation or agreed to anything. This was an industry document basically from a long time. So I'm not in a position to really answer that question myself.

MEMBER SKILLMAN: I'd like to ask a question about your slides 14, 15 and 16. These words can give someone false comfort. What radiological burden in terms of curies are you considering when you write down these words? You spoke about wet well venting with part of the core through the reactor vessel.

MR. KARIPINENI: Right.

MEMBER SKILLMAN: And so I know a little bit about that. What kind of curie burden are you considering when you talk about minimizing plant operators exposure, accounting for radiological conditions and those types of things?

MR. KARIPINENI: I don't exactly remember, but I believe it's the ERO guidance that the industry wanted to rely on, and the staff has agreed to that.

MEMBER SKILLMAN: Does the staff appreciate what the numbers are?

MR. KARIPINENI: Exact numbers I don't recall.

MEMBER SKILLMAN: Well, let me help you: You're going to have between 15 and 18 billion curies in that core if you've got 100 days run time on the core, if it's a typical 3,000-megawatt-thermal core. And the bulk of that will be cesium, if it's 0.667

MeV-gamma. And you can get near that.

MEMBER POWERS: Very little of it would be cesium. Most of it would be xenon.

MEMBER SKILLMAN: You're going to have radiation levels that are just stunning, and you'll have enough radio-gas that makes the venting question very complicated.

MR. KARIPINENI: The location of the full panels, etcetera, including the shielding in the places where the operator has to go to to operate the system, were addressed in the guidance document by the industry. But absolute numbers, I don't recall having any numbers there. Only a guidance like emergency -- what is that called, Jerome?

MR. BETTLE: An emergency response organization.

MR. KARIPINENI: Emergency response organization values that they will accept under conditions of that nature.

MEMBER SKILLMAN: Thank you.

MEMBER POWERS: Then that is an extraordinarily important point to understand that when you blow this thing down and vent it, there's a formidable number of curies coming off that.

MEMBER SKILLMAN: That's what I'm trying

to communicate.

MEMBER POWERS: And just the shine alone --

MEMBER SKILLMAN: Off the top of the TMI2 building, four feet of concrete, it was 4,000 R per hour. We could not get near that building. And the water was lethal at a meter. I spent a whole life doing that. The numbers are staggering. And we are even seeing it now two-and-a-half years later from the Fukushima event.

So when we talk about enabling operators to get close, some of these human factors to enable operation of the vents, I agree with the need to do that, but I'm curious about the practical understanding of what those words entail. I was involved in building shields three and four feet thick and as big as the wall here so someone could get close to a valve. The practical implications of this are stunning.

MR. BUNT: If I could? This is Randy Bunt again. If you look at our 13-02 document under "Programmatic Controls," we do go back and reference the TID 14844 for calculations of distance factors and power, and also to accident source terms that are out there associated with existing lessons learned and accident scenarios, because those already have a core melt feature in them. So we're utilizing those as the bounding conditions. And those are referenced here as a starting point from there with other factors applied to it. We are looking at addressing that, I mean, and making sure we take into account all the lessons learned. And also as we learn lessons from Fukushima and also what we've applied to TMI or in these type documents, we do need to continue to go forward with that. But it is addressed in here, and that's what we're using as our bounding items for that value for those source terms.

MEMBER SKILLMAN: Thank you.

MR. KARIPINENI: The anticipatory venting. Again industry talked about it. And right now there is a white paper that is submitted and the Mitigating Strategies Directorate is reviewing that. And therefore we are not specifically reviewing that in our group at this point. And at some point there will be some result out of this review. And we'll take that into consideration when we write our ISG if there is a need to revise this or remove it, or whatever that is. And we expect that that would happen before we actually finally issue the ISG.

Interfacing requirements with GL 89-16.

There was an Appendix E that was included. We felt like there's really nothing there that we would have to get involved in about the design and implementation of the HCVS itself. And therefore, we did not review that. And we've some history and things like that on how they keep their housekeeping.

MEMBER SCHULTZ: With respect to anticipatory venting, we raised the issue of the containment accident pressure assumptions related to accident response, equipment response during an accident. We raised that in the Subcommittee meeting. And we didn't have a response either from industry or the staff at that point. Now industry has provided a response. Is that a response that the staff has reviewed and do you concur with the assumptions that industry has used with regard to this accident sequencing and that in fact is not a problem?

MR. KARIPINENI: I would expect some involvement from us eventually when we review this white paper when MSD comes back to us. And we'll have a position taken at that time. But I heard what they told us in the meeting and I am generally in line with that. I'm okay with what said there that you're not operating any of these ECCS pumps. The issue is if somebody makes a mistake and then he vents it. And that's why all these provisions were put in the order and the guidance that you have, you know, stick controls and then you open this vent, at least not an automatic opening.

MEMBER SCHULTZ: I think part of our concern is that it wasn't that we were in a situation where we wouldn't need that equipment or the equipment wouldn't operate anyway because it didn't have electric power, but in the event it was functioning/ operational. But for some reason there was a decision to do venting in any case that the CAP credit could be --

MR. DENNIG: And this is Bob Dennig, NRR. The CAP issue is of the bore core damage issue. And say that get's partitioned into the mitigating strategies bowl. So they are looking at the concern about making an error. And for those plants that need containment accident pressure, somehow the feeding that when it's needed for the normal ECCS pumps. So again when we say that it's being looked at over there, that's the rationale. We're assuming that there will be some procedural approach that will implement preservation of CAP under the appropriate circumstances.

MEMBER SCHULTZ: Okay. Thank you, Bob.

MR. KARIPINENI: There were some interlocks and some other switches. Sometimes you may have to operate them with two switches. There were some provisions that were included that give you some assurance that it won't be a very simple one quick flip a switch. Somebody tells you open the switch, and you open the switch. It may have to require a couple of minutes, a half a minute of thinking of something what I'm doing here by having this couple of actions that would be required, is the impression I got from the industry. And that was included in the guidance.

The next slide, please? The dry well temperature issue. The reason we are even looking at it now is because the fact that industry came forward and told us that there is a common portion of the pipe that they would have to design for the dry well conditions. Even though if you don't have the dry well vent, it's the only wet well vent. That wet well venting is about all that I could see. Should we have a dry well vent later, then it would see different conditions then and what are those conditions that they would like to design that common pipe for. And that's how it all started. Originally when we wrote the document at Diablo or the -- it never was in the thinking until they brought it up in the meetings we had in the beginning. So we all started facing a dry well question that was supposed to be part of the Phase 2 actually and the rulemaking process.

So given that, you know, they had to proceed because to make the wet well vent, to complete the wet well vent that little portion of the pipe, wherever they have some equipment and etcetera, they need to put those numbers into the designing process now. And therefore, the numbers of PCPL and 545 degrees were proposed. And we went through quite a bit of long discussions in several meetings with the industry.

And where we stand on that is for the part -- for the Phase 1 decision to whether we can accept that temperature, we believe that it is a reasonable proposition. For one thing, you know, you're only operating the wet well vent right now. You know, you don't have the dry well vent yet. That will come later. And we'll do all these evaluations in the future. Also, the numbers that were proposed are not design values from the design basis accident. They were already higher than those numbers. So in that sense there was a bit of margin there over the regular design basis accidents.

And also the discussion came up about the possibility that they can operate even higher than that. If there were treated for those numbers, they can actually operate for even higher numbers than that. Ultimate numbers, let's call it. Plasticity, elasticity, etcetera, they said. And all these things would allow us to include that number for just Phase 1 portion of the design.

However, we were really concerned that to design for these numbers there may be some acceptance on our part that licensees may not fully realize that we have other issues to look at associated with this in Phase 2 and rulemaking. And one of the issues is -- the biggest of them is the head seal, because it's the most dominant failure mode of the containment from over-pressure and over-temperature. As I said, this has been already proved by the severe accident calculations done by various organizations and also confirmed by the Fukushima event.

Therefore, what we felt is that this dry well vent, should there be one, or alternate filtration strategies that the industry is proposing -- they all have another function within them, which is they should operate and develop the strategies that are designed and operate the dry well vent in a manner that it protects the dry well head seal from gross leakage. That's where we are coming from.

The severe accident conditions, as they have shown in the graph that you looked at in the figure 2-1, can actually exceed these design conditions we are talking about. But we also believe that the dry well vent can be operated, and probably at slightly higher numbers than that, but it has another function in Phase 2, which is somehow you have to operate and develop your strategies in a manner that the dry well head seal doesn't get into a situation of gross failure.

What we are trying to say is basically that designing is one thing. What you are going to operate and what you're going to try to protect with that thing is something slightly different than a design number. Obviously we can't design it for 900 degrees or 1,000 degrees. There are very, very unlikely sequences. But, you know, for majority of sequences higher probability of success is assured.

When you look at the ultimate values of the design as well as the ultimate value failures of the seal and see how we are doing in this question, and that's going to be done in Phase 2. And we want to recognize that fact. And that's why we wanted to put some language into the ISG, into the guidance document. And the industry has put -- for most part include the discussion we had, but shied away from the final statement that somehow it has to be designed and operated in a manner that it protects the dry well head from gross leakage.

MR. KRAFT: Pardon me. Steve Kraft here. I'm sorry. That's just not right. I didn't want to let that go. The combination of design capability operations -- what we're talking about -- we proposed language. The issue for us; and again we could end up in a different spot at the end, is we believe by the information that we've presented based upon Peach Bottom work is that if you pick that design point, 545 at PCPL, you provide a capability such that you will be way -- in that block diagram way outside such that you will always be able to -- that vent will survive, will operate under conditions and operational.

The issue between us is not whether that's true or not. The issue is what sort of proof do I have to provide as a licensee when I submit my overall integrated plan? Our view is based upon what we know now. You pick the design point and you're done. Just go do the work. NRC's language; and we talked about it this morning, is no, no, no, you have to show more that you're actually doing it. And that's just going to lead to further complications and a never-ending discussion.

So that's kind of what the issue is. And I don't want it led to believe that we're somehow not agreeing with what Rao said. It's just how we get to the end here.

MR. DENNIG: This is Bob Dennig. Just to reframe it from another perspective, we are supporting the idea that there's a more extensive capability from the design point to some unknown point that we need to protect at temperatures we haven't figured out yet. But we don't have any. That figure doesn't demonstrate that capability. The statement that that figure demonstrates the capability is just not the case. And we were at one point promised additional information to support that thesis, but we haven't seen anything.

And lastly, we're getting way out in front of ourselves with trying to lock in something in this Phase 1 that was deliberately put off into Phase 2 because of difficulties in specifying these temperatures. So we don't know what the temperature of the environment is. We don't know what the capability of the seal is, but we're going to lock in the temperature. And we don't have anything really concrete to support the range of this capability. So for that reason, we need a straightforward statement that the system will be capable of protecting the seal and without specifying any details of that. And that's pretty much where we've been.

MEMBER SCHULTZ: Bob, what activities are ongoing to make the determination of what the failure temperature will be and what then is required for the vent temperature? Is that something that the staff has --

MR. DENNIG: Yes, Sud is here with us. He's doing the MELCOR analysis.

MR. BASU: I think Bob is putting me on the spot.

(Laughter.)

MR. BASU: This is Sud Basu from the Office of Research. We are doing some MELCOR calculations as part of the rulemaking technical support activities and we have done MELCOR calculations in previous phase that led to the making of SECY-12-0157. You have seen some of those calculations. Some calculations we have seen dry well temperature far in excess of 545F. Now, that doesn't mean that something that is designed for 545F will not survive. It only means that you are going to basically you erode the safety margin by that much more, depending on what temperature you're going to see.

So I think what Bob is trying to say is that there's some work in progress. What we can provide to the Committee, to the stakeholders is that for some accident sequences what kind of temperature do we see in the dry well atmosphere. And then the designer are to take that information and see how best to design the components, whether to design the components at 545F or some other number.

I don't know if that answers your --

MEMBER SCHULTZ: Well, my concern was that I thought, based on both the discussion of the Subcommittee and subsequent discussions, that the closure was closer here in terms of determining what this temperature would be. And I'm concerned that, you know, there is not a lot of time, 12 months, in the full period of Phase 2, and the decision time frame associated with this related to the Phase 2 is obviously much shorter than that. So I'm getting very concerned that we're still talking about safety margins that we haven't yet at least even estimated, let alone quantified.

MR. BASU: And you're making an excellent --

MEMBER SCHULTZ: And we're going to run into a crunch associated with the movement of this beyond the plan which licensees need to submit fairly shortly.

MR. BASU; And you're making an excellent point. And I think that probably -- and I can argue that will support actually deferring this discussion until the Phase 2 of this, which is looking into the dry well, looking into the rulemaking, filtration rulemaking. Because that's when all of these things are going to show up more tangibly.

Here we're talking about designing a portion of the vent system, component of the vent system that is common to both dry well venting and wet well venting.

Now, if I want to make the argument that the dry well venting is not needed, then in a way that goes away, because you will be only dealing with the wet well venting.

If on the other hand dry well venting is needed, then I need to know what sort of conditions that this dry well venting will be operating. So it becomes important that we come up with a number that we can stand on support, you know, from our perspective. As well, I'm sure industry will appreciate that, too, that if the component has to work -- operate in an environment that is temperature-wise more extreme than this component can withstand, I think they ought to know that.

So one way that I can think of is if you design something at 545F and you can tell me that, yes, it will withstand 900 degree F in terms of failure, and then we go back and see whether -- for all credible accidents scenarios whether we do get to a situation where the dry well temperature exceeds or even, you know, sort of closes in on 900F, then some dialogue at that point will be more meaningful. Right now by having a 545 degree F design temperature, we don't know what the ultimate failure temperature is. And we also don't know for all credible accident sequences what the dry well temperatures would be.

Am I making any sense?

MEMBER SCHULTZ: I appreciate the information, but, yes, we still have a program here and a schedule that is being endorsed. That is to say, the staff is saying we're done with Phase 1; we're ready to move to Phase 2. And the Phase 2 schedule is only a year in total before licensees need to make commitments, and I'm not convinced that we stated a schedule or an approach that's going to meet the schedule for completion of Phase 2. And incorporating discussion about the rulemaking side of this is that that schedule is much longer than the schedule for the completion associated with the venting installation.

MR. DENNIG: This Bob Dennig again. The technical basis is due to the Commission in a paper December 14th. So I mean that's the technical information that will inform either before that's done or while it's being done. That's the process that we're looking to inform the process.

And the integrated plans for Phase 2 according to my calendar are due the fourth quarter of 2015. So sometime before that licensees would have to have a pretty good idea of --

MEMBER SCHULTZ: For Phase 2, but for Phase 1 it's a year previous.

MR. DENNIG: Phase 1 integrated plans are due the second quarter of 2014. And that's the schedule --

MEMBER SCHULTZ: So, but what I heard from the industry was that it would certainly be very helpful if this issue was resolved by that time, not a year from now.

MR. DENNIG: I don't disagree with that. I think as far as we were willing to go with that is as much as you needed to know to complete your wet well with the option of having it tied to the dry well, the 545 was okay with us. But the rest of it we're reserving judgment until we know more.

MR. KARIPINENI: This is Rao. The other issue is also that whether you sell at 500, 550 or 600, that's the basic issue we need to look at eventually in Phase 2 and rulemaking. And I'm not saying just by decreasing a few degrees or increasing that by a few decreases is going to satisfy us. That issue is something that has to be looked at in the future. So for a wet well vent right now, that's okay. And the risk is there, that the unlikely circumstance that that doesn't work that portion of the pipe may have to be revised. I can't put it any other way other than that.

But to take that number and say we design it here and it has some more margin, that's enough that we don't have to do anything anymore, is not acceptable to staff. We're saying it has to be looked at and to eventually assure ourselves that, you know, for most of the sequences that the seal is not compromised. Whether you sell at 550, 500, 650, 600, it doesn't matter. That needs to be looked at.

MR. BUNT: This is Randy Bunt again, and I think we're talking possibly still around ourselves. This is part of why we believed coming out of the 23rd meeting was that this is three unique topics. One topic is truly the operation of the dry well vent and wet well vent and what do you need to do to ensure that your temperature stays within the capability so you don't get damage by controlling pressure? We totally agree on that. We believe that's a Phase 2 topic. That's something to discuss later on, and we'll cover that. We need to make sure that we state something of that order.

The second issue is that the order itself states that the vents, wet well and dry well, do not need to exceed the capability of containment. Okay. We then say based on that, if the containment was designed with basically the same type of construction components that the vent will be designed to, and it was designed at a design value less than we're proposing for the dry well design value, then there is inherent margin in the new vent from containment so that it would slightly exceed, even though it's not required to exceed. So that's the design point of it.

And then to the capability point of it, we're saying there is assurance because we're in severe accident land and beyond design basis land. We don't want to say any of the design basis terminology. But there is some assurance that because it's the same type components, because it's a higher select design value, that the capability then can be inferred will be higher than containment. Therefore, it complies with the order language to say that the design of the system does not need to exceed the limiting component inside containment.

So that's why we believe we're getting closer to the staff by breaking this into three distinct topics where one is how do we design it to go ahead and move forward so that when we design it and we've implemented our plant sites, we don't have rework, we don't have to go pull something back out, that we put something that's capable of performing and exceeding the capability of the limiting component of design because it has a design value higher than the design of containment components?

We realize that full implementation of the dry well vent is a Phase 2 and it has a longer duration. The ISG for Phase 2 is due by March of 2015, so the integrated plan can be issued in December of '15. In the same period there's about nine months to develop the details that go into the integrated plan, to develop the design, but there are many utilities that are wanting to design this one time, design the total interaction of the system. And it's a little disheartening from the utility standpoint to understand that when we put this value in to send the guidance document, that that value can change again. Because that's one of the points that put it here is that the common system portion of this will be locked in and would not be changed again, and the interfacing components would not be changed again.

So that's why we believe it's three unique topics that we keep trying to merge back and forth and they really need to be addressed three different ways. One is the design value. How do you go out and buy stuff? The other one is the capability value to get some assurance that we're higher there. And the other one is operating value so we make sure we operate it properly so we don't challenge those components. Thank you.

MR. DENNIG: And again this is Bob Dennig. I'm not sure why we're trying to lock in a number in Phase 1. Well, Bill, I said that we were willing to go along with the idea that for the common components that was acceptable. Right? That's where we are.

MR. BUNT: And that would be fine if --

MR. DENNIG: But the notion of something that would somehow tie our hands from doing something that we learn later we need to, the staff is just not comfortable with that and not comfortable with just the assurance that there will be this capability without any particular basis for it.

MR. BUNT: We understand that point, Bob. We also provided in the presentation here and also in the latest version of the document the list of references that make up that chart to provide some assurance in there for that documentation. And really if we're saying that we are assured and that because it's the common portion that we wouldn't be looking to redesign it, that's an assurance that we are looking for here so that we don't have to do that redesign for both sides going forward.

MR. RECKLEY: Bill Reckley from NRR, Japan Lessons Learned. As Raj mentioned, we have time for one additional meeting, and we'll use that to address this issue, but the one thing everybody needs to keep in mind is that Phase 1 included those actions that the Commission deemed necessary for adequate protection, which is the decay heat removal and other primary functions. This severe accident portion was an add-on that was cost-beneficial to the degree we could do it and was a safety enhancement. What we said in the order, what we'll continue to say here is we cannot allow the severe accident portions to delay the implementation of the adequate protection parts.

And so what I'll commit to you here is we're going to meet and we're going to get worked out to make sure that this issue does not in any way end up delaying the most important functions of the venting system. So again, Raj says we have one more time for a meeting. We'll meet. I tend to agree with probably everybody.

(Laughter.)

MR. RECKLEY: We're not as far apart here as it tends to sound as we're laying out. And so we will work to make sure we narrow those things. But certainly the NRC has no desire to have you design something and then have to replace it later on. We want to minimize any potential of that. But the primary thing is to not let this issue ultimately delay the implementation of the wet well vent because it has more important functions than even the severe accident discussion that we have going on here.

MR. BUNT: Thank you, Bill.

MR. AMWAY: This is Phil Amway, Constellation Energy. I just wanted to make also clear that a number of us plan to do not only the design work one time for both the dry well and the wet well, but also realistically the implementation to go in and modify the system at one time. So, you know, the need to resolve this goes beyond just the common piping. It's also the dry well vent for those of us that are planning to implement that not in two separate phases, two separate outages, but ultimately to do the design work and the installation as a single activity. And, you know, a lot of that has to deal with the implementation schedules of when our outages fall and to try to maximize the safety vent event of the hardened vent and to be able to do the wet well and the dry well vent together. Thank you

MR. DENNIG: This is Bob Dennig. And that's originally how the order was packaged was to do them both at the same time.

MR. KARIPINENI: The last items is the instrumentation reliability and operating environment. This was discussed in a little bit more detail with the industry in the meeting after we had with the Subcommittee, and most of these things were resolved to the best we can say. But the INC engineers felt that there may be some information that they would want to see at the time the submittals come in, because this is a higher-level approached that was given in the document. And I believe the industry is aware of that. And so therefore, we are considering either revising the section we have in the ISG on this portion or maybe potentially even deleting it. We haven't made that decision yet.

This is other observations. This is something industry has talked about. We believe that the statement has been there from the very beginning in the order, not in the order portion itself, but is in the preamble. And therefore, we have included this statement in the ISG also that if there are any takers to go ahead with the dry well vent and an engineered filter, that is something we would look at immediately. That option is there for them. It doesn't mean that we're asking them to do it or requiring them to do it.

MEMBER SKILLMAN: If they were to do that, do you have the tools to evaluate an installed engineered vent?

MR. KARIPINENI: There is substantial information that we have gotten from the sources, as well as filter vendors. Therefore, we believe we can accelerate that review and giving the guidance around those requirements.

MEMBER SKILLMAN: Thank you.

MR. KARIPINENI: That's the completion of my presentation. Thanks.

MEMBER SCHULTZ: Any other questions from the Committee?

(No audible response.)

MEMBER SCHULTZ: Sam, I'll turn it back over to you in case there's public comment.

CHAIRMAN ARMIJO: Okay. Is there anyone on the bridge line?

(No audible response.)

CHAIRMAN ARMIJO: No one on there? Okay. I think, listeners, any added questions, additional questions for the Members?

(No audible response.)

CHAIRMAN ARMIJO: Steve, I think we're ready to go.

Okay. Let's take a break for 15 minutes and then we reconvene and start talking --

MEMBER BLEY: Do we have another --

CHAIRMAN ARMIJO: I don't think so.

MEMBER BLEY: Do you want to close the meeting now? I mean --

CHAIRMAN ARMIJO: Oh, that's right. We're closed. You're right. Close the meeting. We're coming back and we have to do a lot of work. So 15 minutes, that's 6:00.

(Whereupon, the meeting in the above-entitled matter was adjourned at 5:43 p.m.)

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IMPACTS ASSOCIATED WITH EARLY TRANSFER OF SNF FROM POOL STORAGE TO DRY STORAGE

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International High Level Radioactive Waste Management Conference
Albuquerque, New Mexico
April 28 – May 2, 2013

Overview of Presentation

- Objectives of EPRI Study, *Impacts Associated with Transfer of Spent Nuclear Fuel from Spent Fuel Storage Pools to Dry Storage After Five Years of Cooling*, Revision 1, EPRI, #1025206, August 2012.
- Overview of Assumptions
- Results of Study
  - Impact on dry storage requirements
  - Impact on dry storage costs
  - Estimated radiological impacts
  - Spent Fuel Pool (SFP) decay heat and cesium inventory
  - Additional impacts
- Conclusion

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Objectives of EPRI 1025206

- Update of 2010 assessment of impact of a policy decision to transfer spent nuclear fuel (SNF) from SFPs to dry storage after 5 years of cooling. Assessment includes more realistic assumptions regarding impacts associated with worker dose, dry storage costs, cask manufacturing, and SFP and cask loading equipment availability.
- Impacts include:
  - Near term increase in dry storage systems loading requirements including impacts on cask manufacturing and DSC costs.
  - Decrease in DSC capacity needed to store the higher heat load 5-year cooled SNF and subsequent increase in the number of dry storage systems loaded.
  - Increase in worker dose associated with loading 5-year cooled, high-burnup SNF.
  - Decrease in SFP decay heat and cesium inventory.
  - Other costs and impacts (such as an increase in ISFSI decommissioning costs)

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EPRI Assumptions

- **Industry Base Case:**
  - SNF loaded into dry storage systems as needed in order to maintain Full Core Reserve (FCR) capacity in SNF storage pool.
- **Case 2: 10-Year Transfer of SNF Inventory to Dry Storage**
  - 2011 to 2014: SNF is transferred to dry storage as required to maintain FCR.
  - 2015 to 2024: SFP inventories, cooled at least 5 years by 2010, are transferred to dry storage over this 10-year period.
  - 2025 to end of study: 5-year cooled SNF transferred to dry storage
- **Case 3: 15-Year Transfer of SNF Inventory to Dry Storage**
  - 2011 to 2014: SNF is transferred to dry storage as required to maintain FCR.
  - 2015 to 2029: SFP inventories, cooled at least 5 years by 2010, are transferred to dry storage over this 15-year period.
  - 2030 to end of study: 5-year cooled SNF transferred to dry storage

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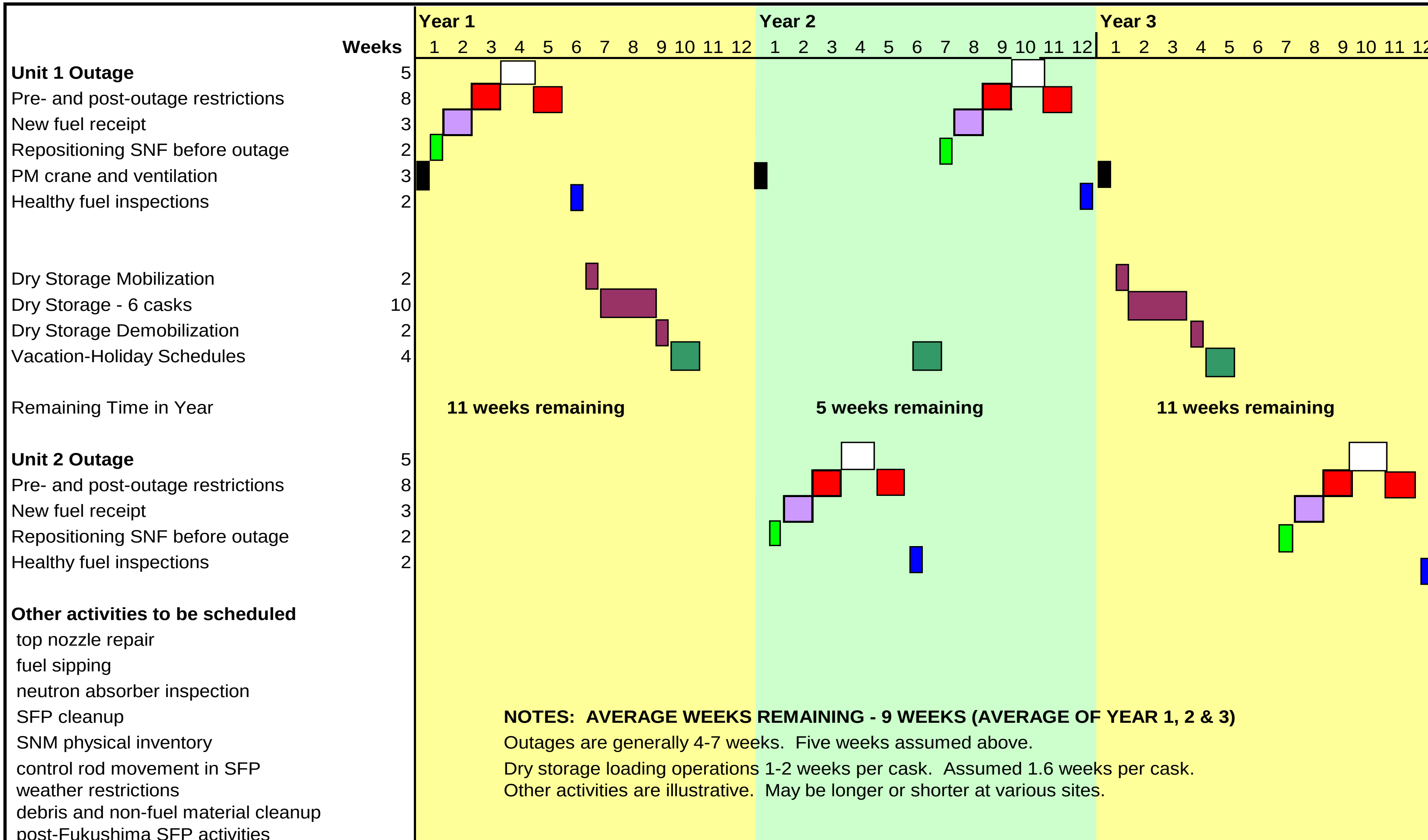
Industry-Wide Impacts – Assumptions Regarding Dry Storage System Capacity

Description	Dry Storage System Capacity (assemblies/system)		
		Current Capacity	Reduced Capacity
Industry Base Case	PWR	24 32 37	24 32 37
	BWR	61 68	61 68
Case 2: 10-Year Transfer Case	PWR	24 32 37	**24** **30** **30**
	BWR	61 68	61 68
Case 3: 15-Year Transfer Case	PWR	24 32 37	**24** **30** **30**
	BWR	61 68	61 68

Industry-Wide Impacts – Power Plant Activities that Limit Availability of SFP and Cask Handling Equipment

- SFP cleanup activities post outage
- Restrictions on pre-outage loading
- Repositioning of SNF in SFP in advance of refueling outage
- Refueling outage
- Restrictions on movement of heavy loads after an outage
- Healthy fuel inspections, special nuclear material (SNM) physical inventory
- Fuel sipping campaigns (periodic)
- Top nozzle repairs (PWR, may be done once or in stages)
- SFP neutron absorber inspections (SFP rack dependent)
- Maintenance, surveillance, and inspection of cask handling crane, ventilation systems, and other equipment
- Weather or seasonal restrictions (may prohibit dry storage loading in some locations)
- Debris and non-fuel related material cleanup and removal
- Control rod movement in SFP
- New fuel receipt and positioning of new fuel in pool
- Scheduled training, vacations and holidays

Industry-Wide Impacts –
SFP Activities and Scheduling for an Illustrative 2-Unit Site With Shared Cask Handling Crane, 18-Month Refueling Cycle



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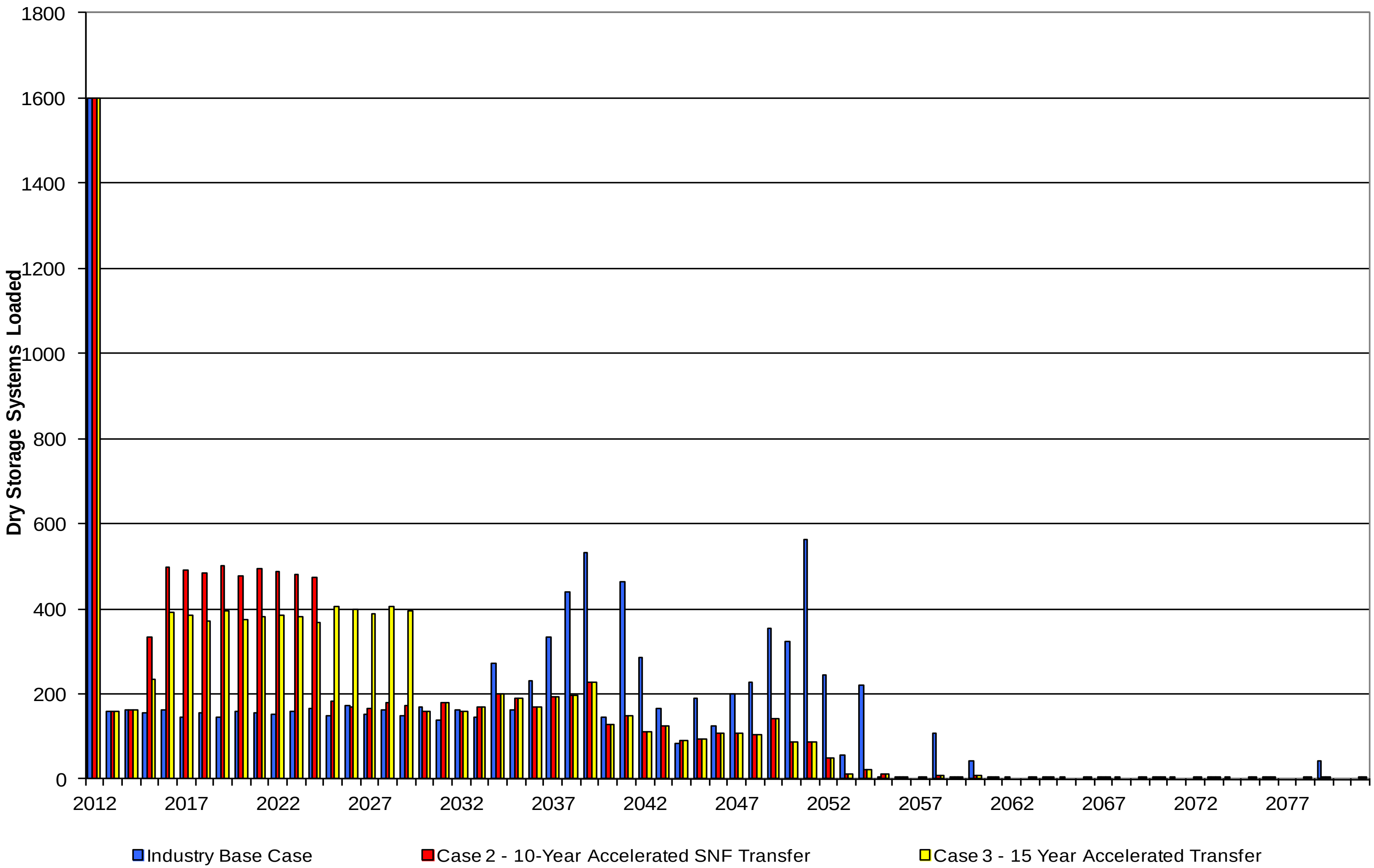
Industry Wide Impacts – Timing of SNF Transfer to Dry Storage

Description	Assemblies Discharged	# DSCs Loaded			
		Year-End 2012	During Operation	Post Shutdown	Total
Case 1: Industry Base Case	**475,600**	1,700	4,636	4,491	10,827
Case 2: 10-Year Transfer		1,700	7,934	1,321	10,955
Case 3: 15-Year Transfer		1,700	7,983	1,337	11,020

- Impacts:
  - Significant increase (>60%) in DSCs loaded in Case 2 and Case 3 during operation due to early transfer
  - More DSCs loaded in Case 2 and 3 due to reduced PWR capacity associated with storing higher heat load SNF.

Industry Wide Impacts – Timing of SNF Transfer to Dry Storage



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Industry Wide Impacts: Comparison of Dry Storage Costs

Description	Dry Storage Costs (Billions $)	
	Dry Storage Costs (Constant $2012)	NPV1 Scenario (Real Discount Rate: 5.8%)
Industry Base Case		
Upfront and Incremental Costs	18.0	5.8
Operating Costs	31.5	3.5
Total Costs	**$49.5**	**$9.3**
Case 2: 10-Year Transfer Case		
Upfront and Incremental Costs	19.4	9.6
Operating Costs	31.5	3.6
Total Costs	**$50.9**	**$13.2**
Case 3: 15-Year Transfer Case		
Upfront and Incremental Costs	19.6	9.2
Operating Costs	31.5	3.6
Total Costs	**$51.1**	**$12.8**
Increased Costs Associated with Case 2: 10 Year Transfer	**$1.4**	**$3.9**
Increased Costs Associated with Case 3: 15 Year Transfer	**$1.6**	**$3.5**

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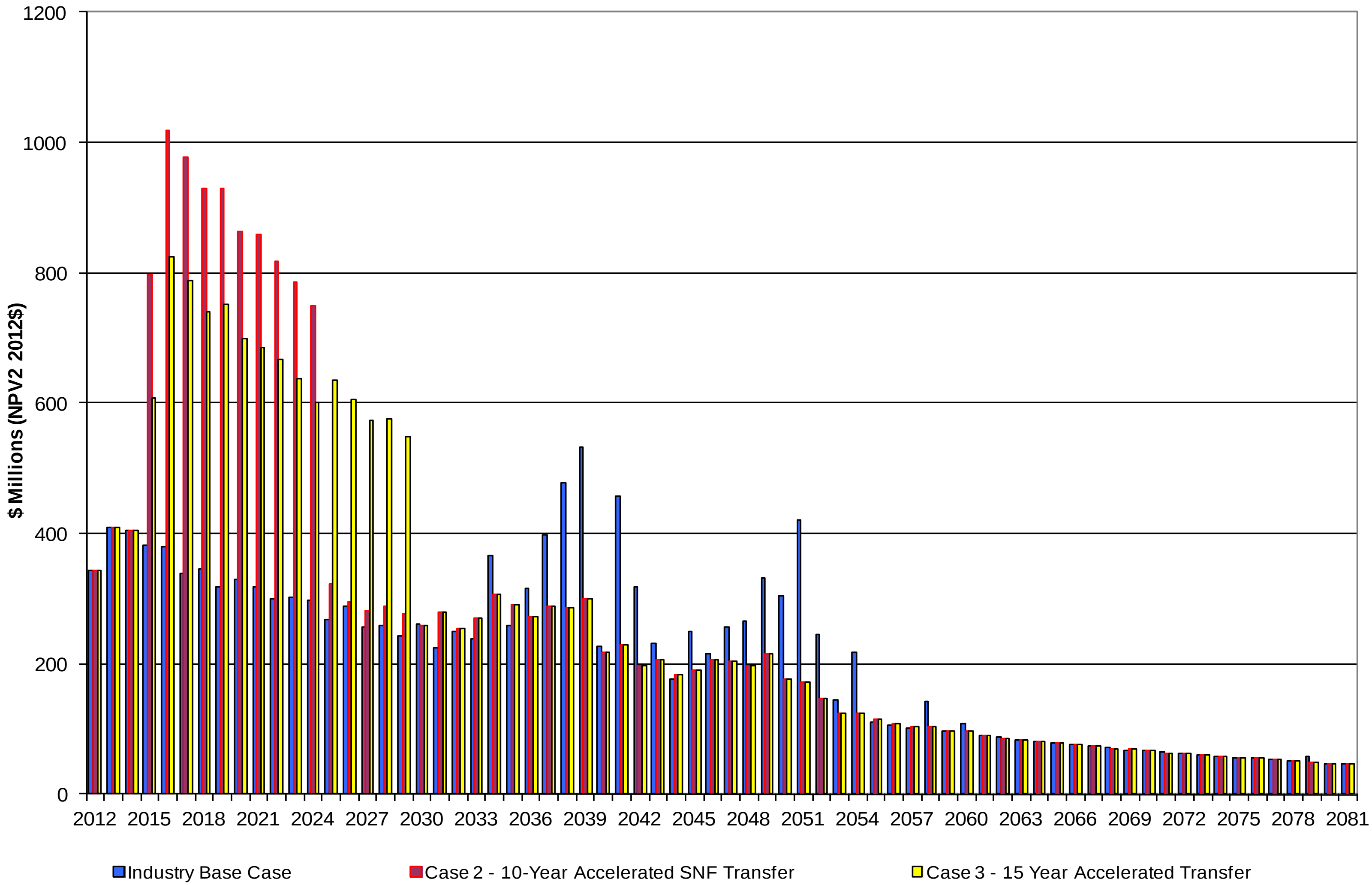
Industry-Wide Impacts: Comparison of Annual Cash Flow (Net Present Value 2012$)



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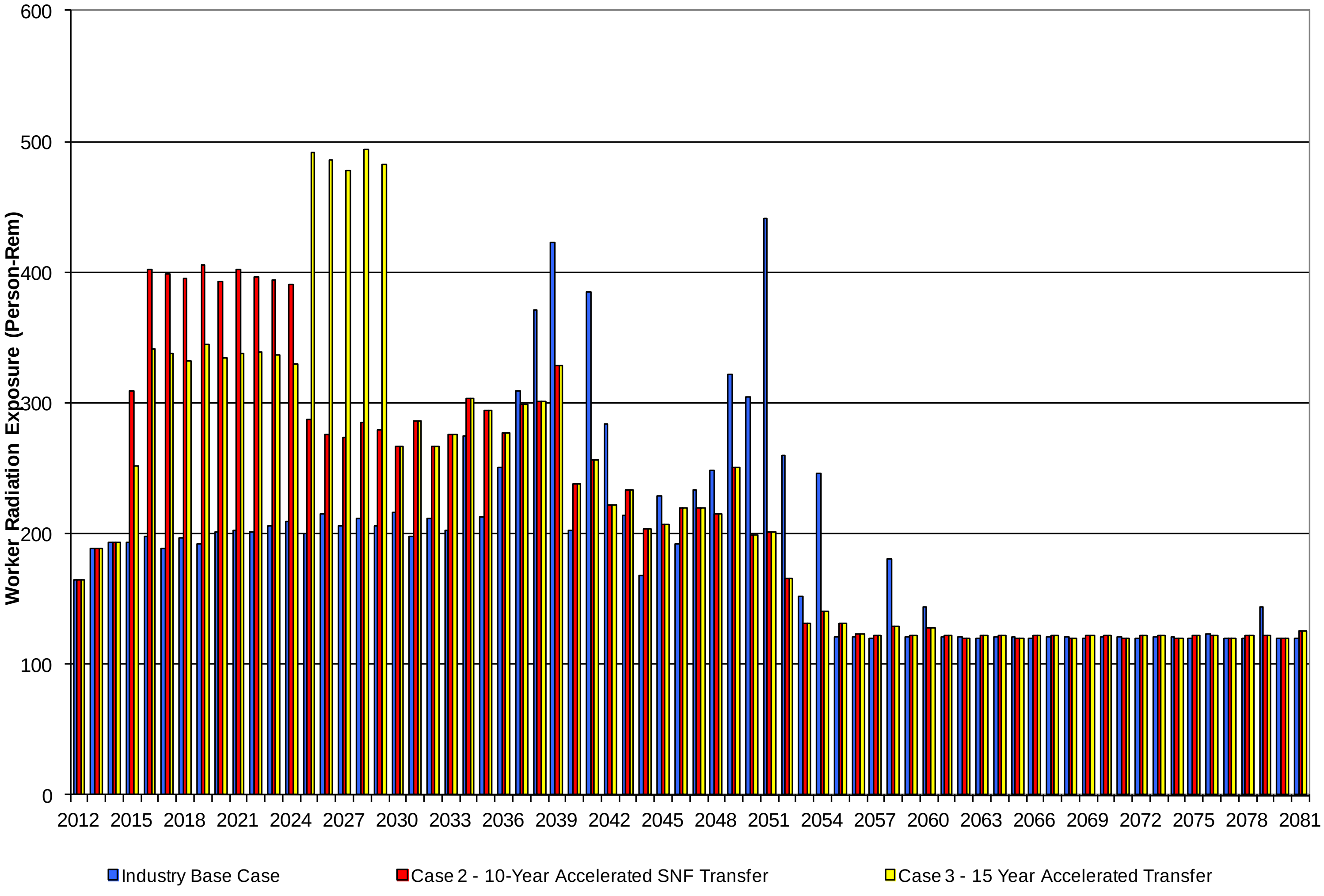
Industry-Wide Impacts: Comparison of Estimated Radiological Impacts to Workers (Person-Rem)

Description	Base Case	10-Year Transfer	15-Year Transfer
Dry Storage System Loading	3,750	5,270	5,690
Annual Maintenance and Inspection	10,460	10,570	10,570
ISFSI Expansion	1,590	1,610	1,630
TOTAL	15,800	17,450	17,890

- Base Case assumes 0.4 person-rem per DSC loaded
- 10-Year and 15-Year transfer cases assume 0.75 person-rem per DCS loaded during period when only 5-year cooled, high-burnup SNF is loaded (+2025)
- Doses for annual maintenance & inspection, and ISFSI expansion are the same in all cases.

Industry-Wide Impacts: Comparison of Estimated Radiological Impacts to Workers (Person-Rem)



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Representative PWR and BWR: SFP Decay Heat and Cesium Inventory

Parameters	**Representative PWR SFP and Inventory**	**Representative BWR SPF and Inventory**
Initial Plant Operation	Mid-1970s	Mid-1970s
Dry Storage Operation	Mid-1990s	Late 1990s
Dry Storage Inventory 2012 (assemblies)	860	1840
SFP Inventory 2012 (assemblies)	835	3200
SFP Inventory 2012 Decay Heat (kilowatts)	2,010	1,900
SFP Inventory 2012 Cs-137 Source Term (Millions TBq)	2.2	3.5
SFP Inventory After 5-Year Cooled SNF Transferred to Dry Storage (assemblies)	279	876
SFP Inventory Decay Heat After 5-Year Cooled SNF Transferred to Dry Storage (kilowatts)	1,540	1,300
SFP Inventory 2012 Cs-137 Source Term (Millions TBq)	1.26	1.85

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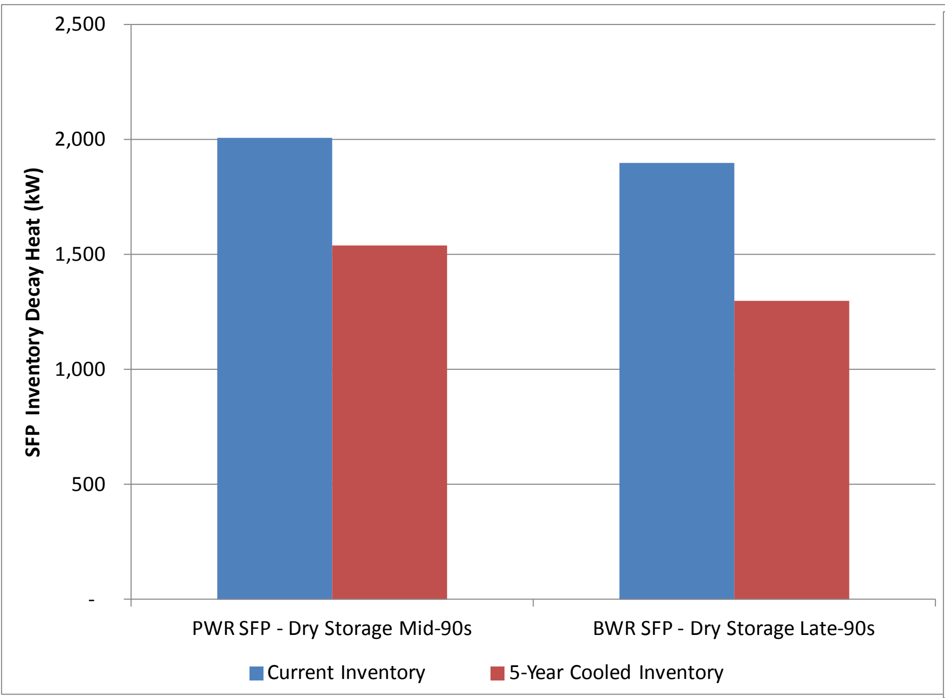
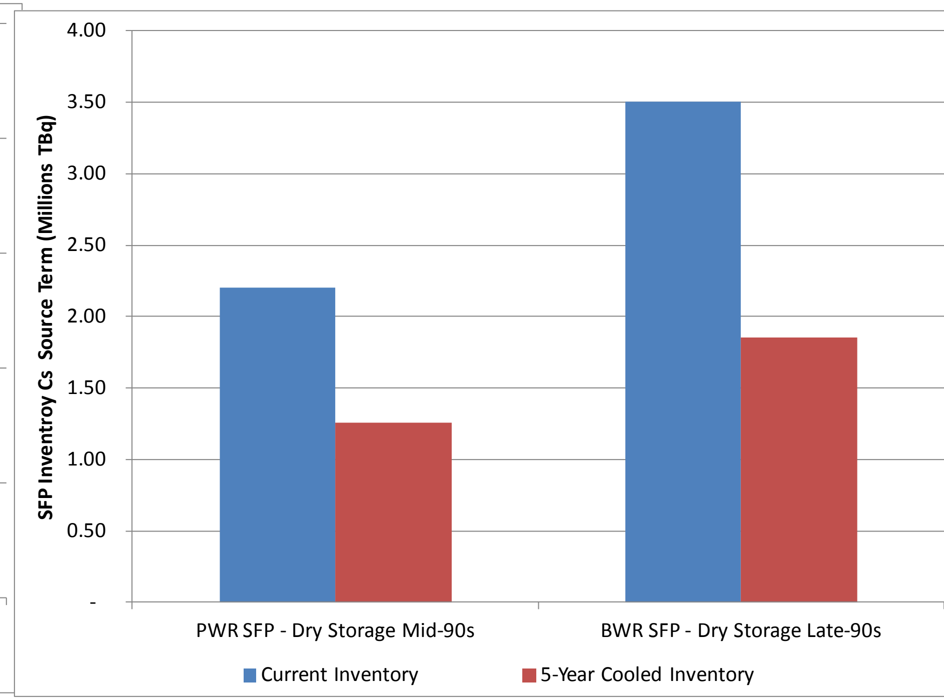
Representative PWR and BWR: SFP Decay Heat and Cesium Inventory



Comparison of Decay Heat
Before and After Transfer
of 5-Year Cooled SNF



Comparison of Cesium Inventory
Before and After Transfer
of 5-Year Cooled SNF

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Other Industry-Wide Impacts:

- Accelerated transfer results in 300 to 500 DSCs loaded per year
  - 2.5- to 3-fold increase in DCSs fabricated, and loaded
  - Increased NRC inspection and oversight activities for cask designers, fabricators, and during loading operations.
- Potential impacts associated with very large loading campaigns include:
  - Need for more management attention;
  - Impacts on plant outage schedules or maintenance schedules due to the increased need for staff to support dry storage operations;
  - Increased non-radiological risks associated with fuel handling and cask handling operations.
- Impacts associated with higher thermal loads for DSCs:
  - Increase in possible hydrogen generation during loading operations
  - Potential for water thermal expansion
  - Higher package and canister lid temperatures, resulting in occupational safety issues and impacts on cask loading operations

Study Conclusions

- Approximate 60% increase in number of DSCs loading during period of reactor operation; although small increase (2%) overall.
- Increase of 1,650 to 2,090 person-rem in worker radiation exposure resulting from loading more DSCs and handling SNF with higher heat load and radiation dose.
- The cost of early transfer of SNF to dry storage is estimated to be $3.5 to $3.9 billion (NPV $2012).
  - Higher costs for DSC fabrication, shielding and for loading short-cooled, high-burnup SNF.
  - Significant cost impact associated with time value of money for early transfer of SNF to dry storage.
- Transfer of 5-year cooled SNF reduces SNF inventory, SFP decay heat and Cs inventory:
  - 27-33% of initial SFP inventory remains
  - 68-77% of initial SNF decay heat remains
  - 53-57% of initial Cs source term remains

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Study Conclusions

- DSC designs may need to be amended, or new designs may need to be certified
  - This may require advances in the heat transfer capabilities of dry storage systems either through improved materials or improved methodology; lower cask capacities, etc.
- It is not clear whether the potential risk reduction due to lower SFP decay heat and Cs source term associated with accelerated transfer of SNF to dry storage offset the real increased risks associated with occupational safety hazards, increased operational impacts and increased cost.

References:

- ***Impacts Associated with Transfer of Spent Nuclear Fuel from Spent Fuel Storage Pools to Dry Storage After Five Years of Cooling*, Revision 1,** EPRI, 105206**,** Final Report, August 2012

http://www.epri.com/abstracts/Pages/ProductAbstract.aspx?ProductId=000000000001025206

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Japan Lessons Learned Tier 3 Issue: Expedited Transfer of Spent Fuel to Dry Cask Storage

Kevin Witt, NRR/JLD/PSB

Steven Jones, NRR/DSS/SBPB

Fred Schofer, NRR/DPR/PRMB

ACRS Meeting

October 2, 2013

Agenda

- Objective & Background
- Expedited Fuel Transfer Analysis Process
- Key Inputs and Assumptions
- Results and Insights
- Summary of Stakeholder Feedback
- Next Steps

Objectives

- Outline staff activities on the Japan lessons learned Tier 3 activity on expedited spent fuel transfer
- Discuss how the Spent Fuel Pool Study and past studies were used in the regulatory analysis for all spent fuel pools
- Gain ACRS endorsement of the Regulatory Analysis for the upcoming Commission paper on this issue

Background

- Tier 3 Project Plan:
  - Determine whether the NRC should consider expedited transfer of spent fuel to dry casks
    - » Regulatory Analysis Guidelines (NUREG/BR-0058)
    - » Commission Safety Goal Policy Statement
      - » No significant increase in risk to life and health
        - » < 0.1% increase in chance of early fatality
        - » < 0.1% increase in chance of lifetime fatal cancer
      - » Societal risks comparable to other sources of power
        - » Reasonable assurance a severe core damage event [large radiological release] will not occur in the U.S.
  - Utilizes information from past SFP evaluations and SFPS

Background

- Schedules have been aligned to facilitate the public's involvement in the Tier 3 issue, the SFPS, and ongoing Waste Confidence activities
  - Spent Fuel Pool Study (final with public comments considered) planned for public release in mid-October
  - Waste Confidence Generic Environmental Impact Statement and draft rule open for public comment (September 13 – November 27)
  - Draft Commission Paper and Regulatory Analysis on expedited transfer of spent fuel publicly released on September 26, 2013
- Waste Confidence documents developed independent of SFPS and Tier 3 activities
  - No explicit inter-relationship / draft rule relied on previous studies
  - SFPS and Tier 3 results align with previous studies

Overview

Generic Regulatory Analysis

- Regulatory Assessment
- Expanded Plants (Generic by Groups)
- Expanded Scenarios

Regulatory Analysis for Reference Plant (Appendix D)

- Regulatory Assessment
- Specific Plant
- Expanded Scenarios

Spent Fuel Pool Study

- Consequence Study
- Specific Plant
- Specific Scenario

Spent Fuel Pool Study Results

- Updates public consequence estimates of a beyond-design-basis earthquake affecting a spent fuel pool
  - high- and low-density loading conditions considered, both with and without deployment of existing mitigation equipment
  - frequency of release driven by presence of recently discharged fuel, not dependent on loading condition
  - many scenarios result in no release, but magnitude of release significantly affected by loading condition for several very unlikely scenarios
- The Study, together with previous research, confirms spent fuel pools adequately protect public health and safety
- The regulatory analysis for the reference plant indicates that faster spent fuel transfer does not substantially enhance safety

Generic Regulatory Analysis

- **The Study's (Appendix D) and Generic Regulatory Analysis consider other initiating events such as:**
  – cask drop
  – loss of power
  – loss of coolant inventory
- **Tier 3 Expand Evaluation to all Spent Fuel Pools**
  – PWRs and BWRs with Mark III containments (spent fuel stored in building separate from reactor building)
  – new reactors (AP-1000)
- **Assessment of security events handled separately**
  – regulatory changes implemented (e.g., 10 CFR 50.54(hh))
  – effect of security changes reflected in regulatory baseline

Groupings

1. BWR Mark I / II with non-shared spent fuel pool (SFP) located well above grade (Excluding Western U.S. Reactor - Columbia)
2. PWR & BWR Mark III with non-shared SFP located at grade with at least one exposed side (Excluding Western U.S. Reactors – Diablo Canyon and Palo Verde)
3. Combined Operating License Holder SFPs (AP-1000)
4. PWRs with Shared SFPs
5. SFPs located below grade with backfill on all sides (not evaluated based on low probability of inventory loss)
6. SFPs at decommissioned plants (fuel in pool) (not evaluated based on low decay heat rate)

Alternatives Considered

- Regulatory Baseline (No Action)
  - Implementation of fuel strategies required by 10 CFR 50.54(hh)
    - high-density storage with 1 hot assembly surrounded by 4 colder assemblies
    - mitigation capability assumed to be ineffective
  - Fuel transfer at rate to just maintain full core discharge capability
- Expedited Transfer of Fuel with > 5 Years Decay
  - Fuel stored in low-density configuration in existing racks with 1 hot assembly surrounded by 4 empty locations
    - Expedited fuel transfer completed over 5 year period for existing SFPs
    - Expedited transfer of all fuel as soon as it has decayed for 5 years for Combined Operating License holders
  - Mitigation assumed to be 95% effective
- Difference approximates maximum potential benefit

Accident Progression – Group 1

Parameter	Low Est./Base Case	High Est.	Notes
Site seismic hazard • Bin 3 (0.7g PGA) • Bin 4 (1.2g PGA)	Peach Bottom 1.65×10^{-5} 4.90×10^{-6}	Limerick 2.24×10^{-5} 7.09×10^{-6}	Limerick is Group 1 site with highest seismic hazard
Liner fragility • Bin 3 (SFPS) • Bin 4 • Cask Drop	 10% / same 50% / 100% (bounding) 100% / same	 100% (bounding) 100% (bounding) 100%	For high estimate, specified initiators always result in coolant inventory leak
Insufficient nat. circ • Bin 3 • Bin 4 • Cask Drop • Other Initiators	 8% / same 30% / 100% (bounding) 8% / 100% (bounding) 100% (bounding) / same	 100% (bounding) 100% (bounding) 100% (bounding) 100% (bounding)	High est. never air coolable – bounds: • uniform dist. • partial drain • closed cell racks
Release Fraction • Alternative 1 • Alternative 2	 3% / 40% 0.5% / 3%	 90% 5%	Alternative 2 models successful mitigation - additional factor of 19 reduction

Accident Progression – Groups2- 4

Parameter	Low Est./Base Case	High Est.	Notes
Site seismic hazard • Bin 3 (0.7g PGA) • Bin 4 (1.2g PGA)	Peach Bottom 1.65×10^{-5} 4.90×10^{-6}	[Highest in Group] 2.9×10^{-5} to 5.6×10^{-5} 9.1×10^{-6} to 2.0×10^{-5}	Highest Hazard Sites: Gr. 2: Watts Bar Gr. 3: Summer Gr. 4: Sequoyah
Liner fragility • Bin 3 • Bin 4 • Cask Drop	 2% / 5% 16% / 50% 100% / same	 25% 100% (bounding) 100%	Bin 4 Earthquake and cask drop always result in loss of coolant inventory
Insufficient nat. circ • Bin 3 • Bin 4 • Cask Drop • Other Initiators	 8% / 100% (bounding) 30% / 100% (bounding) 8% / 100% (bounding) 100% (bounding)	 100% (bounding) 100% (bounding) 100% (bounding) 100% (bounding)	Base & High case not air coolable – bounds: • uniform dist. • partial drain • closed cell racks
Release Fraction • Alternative 1 • Alternative 2	 10% / 75% 0.5% / 3%	 90% 5%	Alternative 2 models successful mitigation - additional factor of 19 reduction

Source Term (MCi Cesium)

Group	Low Est.	Best Est.	High Est.
Source term	**Alt 1/Alt 2**	**Alt 1/Alt 2**	**Alt 1/ Alt 2**
Group 1 (BWR)	40.6 / 19.8	52.7 / 22.0	63.3 / 26.4
Group 2 (PWR)	57.4 / 15.7	67.9 / 17.4	78.2 / 20.9
Group 3 (New)	33.7 / 15.7	44.4 / 17.4	54.2 / 20.9
Group 4 (Shared)	63.6 / 31.4	101.1 / 34.8	142.2 / 41.8

Regulatory Analysis Inputs

Parameter	Low Est.	Best Est.	High Est.
Dose Consequence Analysis			
Population density & demographics	169 people/sq.mi. (Palisades)	317 people/sq.mi. (Surry)	722 people/sq.mi. (Peach Bottom)
Weather conditions & modeling	Same as SFPS (Peach Bottom)	Same as SFPS (Peach Bottom)	Same as SFPS (Peach Bottom)
Habitability Limit & health effects	500 mrem annual - LNT	2 rem first year, 500 mrem thereafter - LNT	2 rem annual - LNT
Evacuation assumptions & modeling	Same as SFPS (Peach Bottom)	Same as SFPS (Peach Bottom)	Same as SFPS (Peach Bottom)
Offsite Property Analysis			
Economic data	Site specific using SECPOP2000) (Palisades)	Site specific using SECPOP2000) (Surry)	Site specific using SECPOP2000) (Peach Bottom)

Backfit Analysis Results

- No Substantial Increase in Public Health and Safety
- Comparison to Safety Goal Quantitative Health Objectives
  - No early fatalities predicted based on nature of release
  - Individual latent cancer risk is much lower than objective
    - Individual risk decreases with distance from facility
    - All cases are similar due to offsite protective actions
    - Individual risk dominated by long-term dose in habitable areas
      - Linear – No Threshold dose-response model used
      - Dose threshold would significantly reduce calculated health effects
- Costs outweigh expected public health benefits
  - Many facilities considered bounded by base case analysis
  - High estimate bounding and not representative of any site
  - For high estimate, health benefits outweigh costs when consequences beyond 50 miles are considered

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Regulatory Analysis Results

- Regulatory Analysis includes economic consequences not considered in Backfit Analysis
- Base case and low estimate costs outweigh benefits
  - Benefits based on $2000/person-rem within 50 miles
  - High estimate benefits appear to outweigh costs because of conservatisms
- Sensitivity Analyses ($4000/person-rem and consequences beyond 50 miles)
  - Low estimate costs outweigh benefits
  - Base case costs outweigh benefits for Groups 1 & 2, benefits marginally outweigh costs for Groups 3 & 4
  - High estimate benefits appear to outweigh costs because of conservatisms

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Safety Perspectives

- Pools provide adequate protection and defense-in-depth
- Overall estimated frequency of damage to stored fuel is low
  - Base case release frequencies are on the order of a few times in a million years
  - These frequencies exclude effective deployment of mitigation capability and generally exclude consideration of air cooling (SFPS)
- Spent Fuel Pool Maintains Defense-in-Depth
  - Defense-in-depth consists of layers of protection with reliability of each layer commensurate with the frequency of challenges
  - SFP designed to prevent coolant inventory loss under accident conditions, which results in a low frequency of coolant inventory loss
  - Fuel dispersal, coolant makeup, and spray capability have reliability commensurate with the low frequency of coolant inventory loss

Recommendation

- Expedited transfer of spent fuel to dry cask storage does not appear to provide either a substantial increase in the overall protection of public health and safety or a safety benefit that outweighs the associated costs

- The staff's recommendation is to not pursue expedited transfer of spent fuel to dry cask storage and close this Tier 3 Japan lessons learned activity

Other Alternatives

- Examples include:
  - Alternative loading patterns
  - Direct offload of fuel into more coolable patterns
  - Enhancement of mitigation strategies
- Staff has considered these possible changes but determined that they do not provide a substantial safety enhancement such that generic regulatory action would be warranted

Stakeholder Feedback

- Two public meetings held (August 22 and September 18)
- Letters received from stakeholders
  - Staff drafting responses
- Written comments received on Spent Fuel Pool Study
  - To be addressed in final study
- In response to stakeholder feedback, staff has provided additional clarification on specific issues in Tier 3 paper

Next Steps

- Issue Final Commission Paper
  - October 11, 2013
- Conduct Commission Meeting on Spent Fuel Safety
  - By end of 2013

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Washington DC 20036-4523
202.328.3500 | office
202.328.6918 | fax
HarmonCurran.com

September 17, 2013

Edwin Hackett, Executive Director
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
By e-mail to: Edwin.Hackett@nrc.gov

SUBJECT: *Request to participate in October 2 ACRS Meeting*

Dear Mr. Hackett:

On behalf of 26 environmental organizations across the United States[1], I am writing to request that you provide Dr. Gordon Thompson, President of the Institute for Resource and Security Studies, with an opportunity to address the Advisory Committee on Reactor Safeguards at its upcoming October 2 meeting regarding Japan Lessons Learned Tier 3 Issue: Transfer of Spent Fuel to Dry Cask Storage. In particular, we seek an opportunity for Dr. Thompson to present his criticisms of the NRC Staff's Draft Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a US Mark I Boiling Water Reactor (June 2013), on which the Staff proposes to rely in large part for its recommendation that you endorse the safety of continued high-density pool storage of spent fuel. A copy of Dr. Thompson's comments is attached.

The ACRS held a meeting on the Draft Consequence Study on July 9, at the beginning of the 30-day public comment period that was provided by the NRC Staff. Therefore it was not possible for us to participate in that meeting in any meaningful way.

My clients consider the issue of spent fuel storage risks to be one of the most important unaddressed safety and environmental issues facing the NRC today. Therefore, they are extremely concerned that, as discussed in Dr. Thompson's comments, the Draft Consequence Study is biased and incomplete, and therefore should not be relied upon for any regulatory decisions regarding management of spent fuel. We are also very concerned that in a July 18,

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[1] With Mindy Goldstein of the Turner Environmental Law Clinic, I am counsel in this matter to: Beyond Nuclear, Blue Ridge Environmental Defense League, Center for a Sustainable Coast, Citizens Allied for Safe Energy, Don't Waste Michigan, Ecology Party of Florida, Friends of the Coast, Friends of the Earth, Georgia Women's Action for New Directions, Green States Solutions, Hudson River Sloop Clearwater, Missouri Coalition for the Environment, NC WARN, Nevada Nuclear Waste Task Force, New England Coalition, No Nukes Pennsylvania, Nuclear Energy Information Service, Nuclear Information and Resource Service, Nuclear Watch South, Physicians for Social Responsibility, Public Citizen, Riverkeeper, SEED Coalition, San Luis Obispo Mothers for Peace, Sierra Club Nuclear Free Campaign, and Southern Alliance for Clean Energy.

2013 letter to NRC Chairman Macfarlane, the ACRS appears to have approved the Draft Consequence Study, without addressing any of the study's serious deficiencies, and without explaining how the study resolves the concerns about deficiencies in the NRC's analysis of the risks posed by high-density pool storage of spent fuel. *See* letter from Dana Powers, ACRS, to Richard A. Meserve, re: Draft Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (April 13, 2010) (copy attached). We request an opportunity for Dr. Thompson to address the respects in which the ACRS's concerns remain unresolved.

In making this request, I would like to clarify that we seek a good ten-to-fifteen minute period for Dr. Thompson to present his views and entertain questions from the ACRS. The brief opportunity for public comment that the ACRS typically sets aside at the end of its meetings would not be a sufficient period of time for Dr. Thompson to explain his opinion and engage with the NRC Staff and members of the ACRS.

I have discussed this request with Christopher Brown of the ACRS staff and Kevin Witt of the NRC Staff. Mr. Witt, who is responsible for proposing a meeting agenda to the ACRS, does not object to our request as long as the Staff has sufficient time to make its own presentation. In this regard, I have been informed by Christopher Brown of the ACRS staff that the meeting can be extended past 3:00 p.m. if necessary.

We would appreciate it if you would share this letter and the attachments with the members of the ACRS.

Thank you for your consideration.

Sincerely,

/s/
Diane Curran

Cc: Christopher Brown
Kevin Witt

INSTITUTE FOR RESOURCE AND SECURITY STUDIES
27 Ellsworth Avenue, Cambridge, Massachusetts 02139, USA

Declaration of 1 August 2013
by Gordon R. Thompson:

Comments on the US Nuclear Regulatory Commission's
Draft Consequence Study of a Beyond-Design-Basis
Earthquake Affecting the Spent Fuel Pool
for a US Mark I Boiling Water Reactor

I, Gordon R. Thompson, declare as follows:

I. Introduction

(I-1) I am the executive director of the Institute for Resource and Security Studies (IRSS), a nonprofit, tax-exempt corporation based in Massachusetts. Our office is located at 27 Ellsworth Avenue, Cambridge, MA 02139. IRSS was founded in 1984 to conduct technical and policy analysis and public education, with the objective of promoting peace and international security, efficient use of natural resources, and protection of the environment. My professional qualifications are discussed in Section II, below.

(I-2) I have been retained by a group of environmental organizations to assist in the preparation of comments invited by the US Nuclear Regulatory Commission (NRC).[1] Specifically, NRC has invited comments on a draft technical study, dated June 2013, that NRC staff has prepared.[2] The draft study is titled "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a US Mark I Boiling Water Reactor".[3] Hereafter, in this declaration, I refer to that study as "NRC's Draft Consequence Study" or "the Study".

(I-3) On 2 January 2013, I completed a declaration that set forth recommendations for NRC's consideration of environmental impacts of long-term, temporary storage

[1] These organizations include: Beyond Nuclear, Blue Ridge Environmental Defense League, Center for a Sustainable Coast, Citizens Allied for Safe Energy, Don't Waste Michigan, Ecology Party of Florida, Friends of the Coast, Friends of the Earth, Georgia Women's Action for New Directions, Green States Solutions, Hudson River Sloop Clearwater, Missouri Coalition for the Environment, NC WARN, Nevada Nuclear Waste Task Force, New England Coalition, No Nukes Pennsylvania, Nuclear Energy Information Service, Nuclear Information and Resource Service, Nuclear Watch South, Physicians for Social Responsibility, Public Citizen, Riverkeeper, SEED Coalition, San Luis Obispo Mothers for Peace, Sierra Club Nuclear Free Campaign, and Southern Alliance for Clean Energy.

[2] *Federal Register*, Volume 78, Number 127, Tuesday 2 July 2013, pp 39781-39782.

[3] Barto et al, 2013.

of spent nuclear fuel (SNF) or related high-level waste (HLW).[4] Those recommendations would apply to NRC's Waste Confidence Generic Environmental Impact Statement (GEIS), which has been issued as a preliminary draft report for comment dated August 2013.[5] Some issues addressed in my 2 January 2013 declaration are relevant to NRC's Draft Consequence Study. Accordingly, I incorporate here by reference the findings and recommendations in my 2 January 2013 declaration.

(I-4) Here, I comment on selected aspects of NRC's Draft Consequence Study. The scope of my comments is constrained by time and budget limitations. Absence of discussion of an issue in this declaration does not imply that I view the issue as insignificant, or that I have no professional opinion on the manner in which the issue has been addressed in NRC's Draft Consequence Study. Although I comment only on selected aspects of the Study, these aspects have comparatively high significance for public health and safety. Moreover, my review of the Study is sufficient to support the findings presented here.

(I-5) NRC's Draft Consequence Study examines, among other matters, the potential for self-sustaining, exothermic oxidation reaction of fuel cladding in a spent-fuel pool if water is lost from the pool. For simplicity, that event can be referred to as a "pool fire".

(I-6) A pool fire is a potential event at every nuclear power plant in the USA. That is so because the spent-fuel pools at all plants are equipped with high-density, closed-frame racks. The nuclear industry began installing these racks in the 1970s, to replace the low-density, open-frame racks previously used. The high-density racks offered a comparatively cheap option for storing a growing inventory of spent fuel.

(I-7) This declaration has the following narrative sections:

I. Introduction
II. My Professional Qualifications
III. A Brief History of Pool-Fire Analysis
IV. What Pool-Fire Analysis Should NRC Have Published Now?
V. NRC's Draft Consequence Study: Structure, Apparent Scope, and Messages
VI. NRC's Draft Consequence Study: Actual Scope, and Credibility
VII. NRC's Use of the MELCOR Code
VIII. Conclusions and Recommendations

(I-8) In addition to the above-named narrative sections, this declaration has two appendices that are an integral part of the declaration. Appendix A contains tables and figures that support the narrative. Appendix B is a bibliography. Documents cited in the narrative or in Appendix A are listed in that bibliography unless otherwise identified.

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[4] Thompson, 2013.
[5] NRC, 2013.

II. My Professional Qualifications

(II-1) As stated in paragraph I-1, above, I am the executive director of the Institute for Resource and Security Studies. In addition, I am a senior research scientist at the George Perkins Marsh Institute, Clark University.

(II-2) I received an undergraduate education in science and mechanical engineering at the University of New South Wales, in Australia, and practiced engineering in Australia in the electricity sector. Subsequently, I pursued graduate studies at Oxford University and received from that institution a Doctorate of Philosophy in mathematics in 1973, for analyses of plasma undergoing thermonuclear fusion. During my graduate studies I was associated with the fusion research program of the UK Atomic Energy Authority. My undergraduate and graduate work provided me with a rigorous education in the methodologies and disciplines of science, mathematics, and engineering.

(II-3) My professional work involves technical and policy analysis in the fields of energy, environment, sustainable development, human security, and international security. Since 1977, a significant part of my work has consisted of analyses of the radiological risk posed by commercial and military nuclear facilities. These analyses have been sponsored by a variety of non-governmental organizations and local, state and national governments, predominantly in North America and Western Europe. Drawing upon these analyses, I have provided expert testimony in legal and regulatory proceedings, and have served on committees advising US government agencies.

(II-4) To a significant degree, my work has been accepted or adopted by relevant governmental agencies. During the period 1978-1979, for example, I served on an international review group commissioned by the government of Lower Saxony (a state in Germany) to evaluate a proposal for a nuclear fuel cycle center at Gorleben. I led the subgroup that examined radiological risk and identified alternative options with lower risk.[6] One of the risk issues that I personally identified and analyzed was the potential for self-sustaining, exothermic oxidation reaction of fuel cladding in a high-density SNF pool if water is lost from the pool. That event is referred to here as a pool fire. In examining the potential for a pool fire, I identified partial loss of water as a more severe condition than total loss of water. I identified a variety of events that could cause loss of water from a pool, including aircraft crash, sabotage, neglect, and acts of war. Also, I identified and described alternative SNF storage options with lower risk; these lower-risk options included design features such as spatial separation, natural cooling, and underground vaults. The Lower Saxony government accepted my findings about the risk of a pool fire, and ruled in May 1979 that high-density pool storage of SNF was not an acceptable option at Gorleben.[7] As a direct result, policy throughout Germany has been to use dry storage in casks, rather than high-density pool storage, for away-from-reactor storage of SNF.

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[6] Beyea et al, 1979.
[7] Albrecht, 1979.

(II-5) Since 1979, I have been based in the USA. During the subsequent years, I have been involved in a number of NRC regulatory proceedings related to the radiological risk posed by storage of SNF. In that context I have prepared a number of declarations and expert reports.[8] Also, I co-authored a journal article, on SNF radiological risk, that received considerable attention from relevant stakeholders.[9] The findings in that article were generally confirmed by a subsequent report by the National Research Council.[10] As a result of my cumulative experience, I am generally familiar with: (i) US practices for managing SNF; (ii) the radiological risk posed by those practices; (iii) NRC regulation of that risk; and (iv) alternative options for reducing that risk. Also, I am familiar with the US effort since the 1950s to implement final disposal of SNF and HLW, and have written a review article on that subject.[11]

(II-6) I have performed a number of studies on the potential for commercial or military nuclear facilities to be attacked directly or to experience indirect effects of violent conflict. A substantial part of that work relates to the radiological risk posed by storage of SNF or HLW. For example, in 2005 I was commissioned by the UK government's Committee on Radioactive Waste Management (CORWM) to prepare a report on reasonably foreseeable security threats to options for long-term management of UK radioactive waste.[12]

III. A Brief History of Pool-Fire Analysis

(III-1) Any review of the merit of NRC's Draft Consequence Study should be informed by the history of analysis regarding the potential for a pool fire. Here, I provide a brief history from March 1979 through May 2013 (i.e., just prior to publication of NRC's Draft Consequence Study in June 2013). This history does not purport to be exhaustive. Instead, it addresses some important highlights.

(III-2) Two studies completed in March 1979 independently identified the potential for a pool fire. One study was by members of an international review group commissioned by the government of Lower Saxony, as discussed in paragraph II-4, above. That study was done under time and budget constraints, so it used simple, scoping analysis to address pool-fire phenomena. The second study was done by Sandia Laboratories for NRC.[13] In light of knowledge that has accumulated since 1979, the Sandia report generally stands up well, provided that one reads the report in its entirety. However, the report's introduction contains an erroneous statement that complete drainage of the pool would be the most severe mode of water loss.[14] The body of the report clearly shows that partial loss of water could be a more severe case, as was recognized in the Lower Saxony study.

[8] See, for example: Thompson, 2009.
[9] Alvarez et al, 2003.
[10] National Research Council, 2006.
[11] Thompson, 2008.
[12] Thompson, 2005.
[13] Benjamin et al, 1979.
[14] Benjamin et al, 1979, page 11.

(III-3) The 1979 Sandia report explicitly recognized a point that was obvious then and has remained so. The point is that the pool-fire issue became salient when the nuclear industry abandoned the use of low-density, open-frame storage racks and switched to high-density, closed-frame racks. The nuclear industry made this switch, beginning in the 1970s, because high-density racks offered a comparatively cheap option for storing a growing inventory of spent fuel. Figure III-1 shows a low-density, open-frame rack for pressurized-water-reactor (PWR) fuel. If water were lost from a pool equipped with such racks, fuel would be readily cooled by three-dimensional, natural convective circulation of air and steam. Human intervention would not be required. Contemporaneous racks used for boiling-water-reactor (BWR) fuel were not as fully open to three-dimensional convective circulation of air and steam, in the event of water loss, as would be the rack shown in Figure III-1. However, a BWR rack could be constructed with a configuration similar to that in Figure III-1. If necessary, channel boxes could be removed from BWR fuel assemblies before their placement in that rack, as discussed in the following paragraph.

(III-4) If low-density, open-frame racks were used, water loss from a pool would lead to fuel ignition only in very rare circumstances. These circumstances might include deformation and coverage of racks by a falling object, and/or the presence in the pool of fuel assemblies from a reactor shut down a short time previously. A thorough investigation of pool-fire risk would identify and characterize such circumstances. Also, such an investigation would determine the potential for ignition and fire propagation for cases in which channel boxes were, or were not, removed from BWR fuel. Convective circulation of air and steam, in the event of water loss, would be enhanced if the channel boxes had been removed. Overall, it is clear that re-equipping the present high-density pools with low-density, open-frame racks would dramatically reduce the risk of a pool fire. In the case of BWR fuel, removal of channel boxes might be an appropriate adjunct step.

(III-5) By the latter part of 1979, at least six points about potential pool fires were clear to any technically-competent person who was paying attention to this issue. First, loss of water from a pool with high-density racks could lead to exothermic air-zircaloy or steam-zircaloy reactions under some conditions. Second, the intensity of exothermic reactions could lead to propagation of ignition to some fuel assemblies that had not initially ignited. Third, a water-loss case involving the presence of residual water would be a more severe case than one involving total drainage, other factors being equal, because the residual water would inhibit convective heat transfer. Fourth, a pool-fire scenario would develop more slowly than a reactor core melt, because the output of decay heat would be smaller in the pool situation. Fifth, the fire threat could be dramatically reduced by reverting to low-density, open-frame racks. Sixth, the fire threat can be roughly characterized using simple, scoping analysis, but developing a thorough understanding would require sophisticated modeling backed up by experiment.

(III-6) Given these six points, one can easily identify a water-loss scenario that represents a test of the credibility of an analysis of pool-fire risk. Any such analysis fails if it does not characterize this scenario. This scenario is not necessarily the "worst" case of water loss from a pool. It does, however, capture the role of residual water in the pool. I refer to it here as the "Severe Reference" scenario of water loss. In the basic version of this scenario, water level would fall rapidly (i.e., within a few minutes) to about mid-height of the fuel. Variants of the scenario would explore the implications of different timing and magnitude for the initial fall of water level, and different outputs of decay heat.[15] After the initial fall of water level, water loss would be evaporative, driven by decay heat. There would be no water makeup. The exposed portion of the fuel would gradually increase in temperature. Eventually, a zircaloy-steam reaction could begin in this portion, commencing first in fuel assemblies with the highest decay heat. The availability of steam would initially limit the rate of this reaction. The fire could propagate across the pool. Over time, fuel and rack degradation, and evaporation of residual water, would alter the fire characteristics. Outcomes could include the initiation of a zircaloy-air reaction.

(III-7) A thorough and comprehensive investigation of pool-fire risk would begin by characterizing the Severe Reference scenario, its variants, and a range of other water-loss scenarios, in terms of phenomena related to zircaloy ignition, fire dynamics, and radioactive release. Then, and only then, would the investigators be ready to move to the next analytic step. That step would be to identify and characterize a full range of event sequences that involve water loss and could lead to a pool fire. The need to work in this manner – completing phenomenological analysis before proceeding to event analysis – has been clear to any technically-competent pool-fire analyst since 1979. I address this matter further in Section IV, below.

(III-8) A credible analysis of event sequences would certainly consider earthquake as a potential initiating event. However, other pool-fire initiating events, including accidents and attacks, would receive at least equal attention. Notably, a credible analysis would thoroughly examine potential situations in which a reactor adjacent to a spent-fuel pool experiences core melt and a substantial release of radioactive material. The onsite impacts of that release and associated phenomena (e.g., hydrogen explosion) could preclude actions, such as water makeup, that could prevent a pool fire.

(III-9) The physical proximity of spent-fuel pools to operating reactors, and their sharing of safety systems, means that the use of high-density racks creates strong linkages between reactor risk and pool risk. A reactor core melt – a comparatively fast-developing event – could enable a pool fire – a slower-developing event. This coupling could be manifested through an accident or an attack. The potential for pool-reactor linkages has, since 1979, been clear to any technically-competent person who was paying attention to the pool-fire issue. The Severe Reference scenario for water loss, as articulated in paragraph III-6, above, is particularly pertinent to these linkages.

(III-10) NRC has publicly postulated an attack on a spent-fuel pool, in its August 1979 GEIS on Handling and Storage of Spent LWR Fuel.[16] Table III-1 summarizes the nature

[15] Some variants would include a zero magnitude for the initial fall of water level (i.e., water would be lost only by evaporation).
[16] NRC, 1979.

of the postulated attack. NRC did not examine the potential for this attack to cause a pool fire. However, the adversary capabilities and other assumptions reflected in Table III-1 would be consistent with an attack that causes a linked core melt and pool fire as outlined in paragraph III-9, above. NRC is currently reluctant to discuss the threat of attack on a pool and/or reactor, but has not repudiated its discussion of attack in the August 1979 GEIS.

(III-11) After receiving the 1979 Sandia report described in paragraph III-3, NRC conducted and sponsored a number of studies related to pool-fire risk, which were published over a period of two decades. Unfortunately, those studies employed the erroneous assumption that complete drainage is the most severe case of water loss, until NRC indirectly corrected this error in October 2000. Thus, for two decades NRC personnel failed to acknowledge the effect of residual water on heat transfer, which is the third of six points I articulate in paragraph III-5, above. The studies also had other deficiencies. I provided a critical review of the various NRC studies in a February 2009 report.[17] In short, those studies did not provide a credible technical basis for assessing the risk of a pool fire.

(III-12) NRC's belated acknowledgment of the effect of residual water on heat transfer came indirectly. It came in the context of determining the maximum age of spent fuel at which the fuel could ignite if water were lost from a pool equipped with high-density racks.[18] If residual water were present, heat transfer from the exposed portion of the fuel would be comparatively feeble.[19] Thus, in the absence of sophisticated modeling of heat transfer, a prudent analyst would assume that the exposed portion of the fuel would be in an approximately adiabatic situation. It follows that comparatively old fuel – perhaps as old as 10 years – could ignite. This issue arose during a license-amendment proceeding in regard to the expansion of spent-fuel-pool capacity at the Harris nuclear power plant. I served as a technical adviser for Orange County, North Carolina, the intervenor in that proceeding. In filings during March and April 2000, the NRC staff repeatedly disparaged my statements that comparatively old fuel could ignite. A few months later, however, the staff adopted my position. NRC staff members stated that loss of water from pools containing fuel aged less than 5 years "would almost certainly result in an exothermic reaction", and also stated: "Precisely how old the fuel has to be to prevent a fire is still not resolved."[20] Moreover, the staff assumed that a fire would be inevitable if the water level fell to the top of the racks.

(III-13) In October 2000, NRC released a study, which was formally published in February 2001, that addressed the potential for a pool fire at a nuclear power plant undergoing decommissioning.[21] The study – NUREG-1738 – was in some respects an improvement on previous NRC studies that addressed pool fires. It reversed NRC's

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[17] Thompson, 2009.
[18] Here, "age" refers to time since the fuel experienced fission.
[19] Colleagues and I have addressed this heat-transfer situation in various documents. See, for example: Alvarez et al, 2003.
[20] Parry et al, 2000, paragraph 29.
[21] Collins and Hubbard, 2001.

longstanding, erroneous position that total drainage of a pool is the most severe case of water loss. However, it did not consider attack. Nor did it add significantly to the weak base of technical knowledge regarding the propagation of a fire from one fuel assembly to another. Its focus was on a plant undergoing decommissioning. Therefore, it did not address potential risk linkages between pools and operating reactors, as mentioned in paragraphs III-8 and III-9, above.

(III-14) The preceding two paragraphs show that, in October 2000, NRC suddenly reversed an erroneous technical position it had held for two decades. The context in which this reversal occurred is significant today. I return to this matter in paragraphs III-23 and III-24, below.

(III-15) After publishing NUREG-1738, NRC ceased publishing analysis on pool-fire risk, but claims to have done some secret studies. The US Government Accountability Office (GAO) confirms that NRC has, indeed, done some secret studies on pool fires. However, according to GAO, the NRC has lost track of those studies. An August 2012 GAO report stated:[22]

> "Because a decision on a permanent means of disposing of spent fuel may not be made for years, NRC officials and others may need to make interim decisions, which could be informed by past studies on stored spent fuel. In response to GAO requests, however, NRC could not easily identify, locate, or access studies it had conducted or commissioned because it does not have an agencywide mechanism to ensure that it can identify and locate such classified studies."

(III-16) I identified a similar problem in a February 2009 report that I mention in paragraph III-11, above. In that report, I examined statements, in two official NRC documents published in 2008, regarding secret studies allegedly conducted or sponsored by NRC in order to improve technical understanding of pool fires. I concluded:[23]

> "To summarize, the Draft Update, issued in October 2008, mentions one set of secret studies, while the rulemaking petition decision, issued in August 2008, mentions a different set of secret studies. This inconsistency represents, at a minimum, carelessness and a lack of respect for the public."

(III-17) Since 1979, NRC has consistently and unequivocally argued, in many contexts and with somewhat varying language, that high-density storage of spent fuel in pools protects public health and safety.[24] Yet, after the attacks of 11 September 2001 on New York and Washington, NRC placed its work on pool-fire risk behind a veil of secrecy. The lengths to which NRC would go to preserve this secrecy were evident from its confrontation with the National Academy of Sciences (NAS).

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[22] GAO, 2012, Highlights.
[23] Thompson, 2009, Section 5.2, pp 24-25.
[24] For example, NRC's Draft Consequence Study says (Barto et al, 2013, page iv): "The NRC continues to believe, based on this study and previous studies that spent fuel pools protect public health and safety."

(III-18) In 2003, eight authors, of which I was one, published a paper on the radiological risk of pool fires and the options for reducing this risk.[25] That paper aroused vigorous comment, and its findings were disputed by NRC officials and others. Critical comment was also directed to a related report I had prepared.[26] In an effort to resolve this controversy, the US Congress requested NAS to conduct a study on the safety and security of spent-fuel storage. NAS submitted a classified report to Congress in July 2004, and released an unclassified version in April 2005.[27] Press reports described considerable tension between NAS and NRC regarding the inclusion of material in the unclassified NAS report.[28] NRC was the party demanding greater secrecy.

(III-19) NRC has never explained how its ongoing statement that high-density pools protect public health and safety could be reconciled with its vigorous efforts to hide pool-fire risk behind a veil of secrecy. An adequate explanation is hard to imagine. If the pools truly posed an insignificant risk, then spent fuel in the pools would not ignite in the event of water loss, regardless of how that water loss proceeded or what was its cause. In that case, there would be no need for secrecy.

(III-20) Assessing the radiological risk posed by a reactor or spent-fuel pool involves science that was at the cutting edge a comparatively long time ago – mostly in the first half of the 20th century or earlier. Nevertheless, a risk assessment must conform to scientific principles if it is to be credible. Those principles include transparency, accountability, openness, support for independent teams of investigators who can critique each other's work, peer review, and opportunities for open dialogue among investigators.

(III-21) In theory, NRC has processes available to it that would allow some of the principles of scientific discourse to be applied to radiological risk assessment. One such process is an evidentiary hearing. Although that process is more legalistic than a scientist would prefer, it does allow for the public cross-examination of expert witnesses under oath. That cross-examination can help to elucidate the scientific reality underlying a contentious issue.

(III-22) Since the 1980s, I have been a technical adviser to various entities – state and local governments, and citizen groups – that have sought to intervene before NRC regarding pool-fire risk. These entities have repeatedly requested the holding of an evidentiary hearing, in the full knowledge that their own expert witnesses would be subjected to rigorous, public cross-examination. NRC has consistently denied these requests, on legalistic grounds.

(III-23) Over this period of three decades, I have had one opportunity to present my findings on pool-fire risk at an NRC-sponsored event that approximated the characteristics of a scientific dialogue. That opportunity came when I asked NRC's Advisory Committee on Reactor Safeguards (ACRS) if I could present my findings to

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[25] Alvarez et al, 2003.
[26] Thompson, 2003.
[27] The unclassified version was ultimately published as: National Research Council, 2006.
[28] Wald, 2005.

them. ACRS agreed, and I presented my findings at two public meetings of ACRS in the latter part of 2000. A remarkable feature of the first meeting was that NRC staff members who made presentations at the meeting suddenly reversed NRC's longstanding, erroneous position that total loss of water from a pool would be the most serious case of water loss. That reversal then made its way into the NRC staff position in the Harris license proceeding, and into NRC's report NUREG-1738, as discussed in paragraphs III-12 and III-13, above.

(III-24) This interaction before ACRS, unique in my experience with NRC, clearly demonstrated the efficacy of scientific discourse. NRC staff members, required for the first time in decades to justify their technical position in a public setting where they could be challenged, suddenly changed that position. Regrettably, however, NRC never repudiated the bad analysis it had done over the preceding two decades, based on its misunderstanding of the 1979 Sandia report. Also, from my observation, NRC has subsequently been careful to avoid placing itself in a similar public setting in which it could be challenged.

(III-25) As stated in paragraph III-5, above, it was clear in 1979 that the threat of a pool fire can be roughly characterized using simple, scoping analysis, but developing a thorough understanding would require sophisticated modeling backed up by experiment. When did NRC acquire the capability to perform such modeling and experiment? A reasonable case can be made that NRC had acquired an appropriate capability by the time of its work on reactor risk that led to publication of the NUREG-1150 study in 1990.[29] Regrettably, however, the NUREG-1150 work did not address pool fires.

(III-26) The history described in paragraphs III-1 through III-25 began in March 1979 and ended just prior to publication of NRC's Draft Consequence Study in June 2013. To summarize, at the end of that period NRC's technical credibility on the pool-fire issue was low. NRC had done demonstrably bad analysis that it never repudiated. NRC had claimed that high-density pool storage protects public health and safety while simultaneously demonstrating the falsity of that claim by hiding pool-fire risk behind a veil of secrecy since 2001. NRC had avoided scientific settings in which its technical position could be publicly challenged. When obliged by ACRS to appear in such a setting in 2000, NRC suddenly changed its position. NRC failed to conduct sophisticated modeling and supporting experiments that could have resolved technical issues central to pool-fire risk, despite having an appropriate capability prior to 1990.

[29] NRC, 1990.

IV. What Pool-Fire Analysis Should NRC Have Published Now?

(IV-1) As summarized in paragraph III-26, above, in May 2013 NRC's technical credibility on the pool-fire issue was low. If NRC had made a serious commitment to begin restoring its credibility, and to provide the public with useful information about pool-fire risk, what technical analysis would NRC have published in June 2013? This question assumes, of course, that NRC would have made its commitment well in advance of June 2013 and would have done the appropriate work before that date.

(IV-2) The answer to the question in paragraph IV-1 is that NRC should have focused its initial attention exclusively on establishing a solid technical understanding of phenomena directly related to a potential pool fire. To do this, NRC would have started with a clean slate and used the best available modeling capability backed up by experiment. This modeling and experimental work would have been done according to scientific principles that I discuss further in paragraph IV-3, below. Tasks in the investigation would have included:

1. <u>Identify a range of rack and pool configurations</u>: The key point here would be to compare a pool with high-density racks to a pool with open-frame, low-density racks. (See paragraph III-3, above.)
2. <u>Identify a range of rack loadings</u>: In the high-density cases, the range of rack loadings would include different phases of the reactor operating cycle, and different distributions of younger and older spent fuel across the pool. In the low-density, open-frame cases, the range of rack loadings would include removal of fuel from the pool if above a certain age, such as five years.
3. <u>Identify a range of water-loss scenarios</u>: Mechanisms for water loss could include various combinations of: leakage; evaporation; sloshing; displacement; siphoning; pumping; and tipping of the pool. To reflect the various combinations and their timeframes, the investigation would identify a range of water-loss scenarios. These scenarios would include, but would not be limited to, situations in which leakage occurred through a hole at the level of the pool floor. The scenarios would include the Severe Reference scenario, and its variants, as discussed in paragraph III-6, above.
4. <u>Identify collateral conditions that could affect fuel ignition or fire dynamics</u>: The potential for fuel ignition, in the event of water loss, could be affected by collateral conditions. Those conditions could also affect the development and propagation of a fire. Relevant conditions could include: the presence of extraneous objects in the pool (e.g., transfer cask, fuel-handling machinery, overhead crane, debris from the upper portion of the pool building); the ventilation status of the pool building; and deformation of racks.
5. <u>Determine combinations of conditions that would lead to fuel ignition</u>: Tasks 1 through 4, above, would identify ranges of rack/pool configurations, rack loadings, water-loss scenarios, and collateral conditions. The various combinations of conditions could be grouped where appropriate. Then, each combination would be examined to determine if, and with what timing, it would lead to fuel ignition.

6. <u>Predict fire behavior</u>: For each instance where Task 5 determined that ignition would occur, the development and propagation of the resulting fire would be predicted. Relevant fire characteristics would include the production of hydrogen and its behavior in the pool building.
7. <u>Estimate the atmospheric release</u>: For each fire sequence examined in Task 6, the resulting release of radioactive material to the external atmosphere would be estimated in terms of isotopic magnitudes, timing, and other relevant characteristics.

(IV-3) If NRC were truly committed to restoring its credibility and providing useful information, it would have performed Tasks 1 through 6 according to generally accepted scientific principles. As discussed in paragraph III-20, above, those principles include transparency, accountability, openness, support for independent teams of investigators who can critique each other's work, peer review, and opportunities for open dialogue among investigators. To satisfy those principles, NRC would have funded independent investigators and made its models available to them for their own use. NRC would have financed independently-run workshops where NRC investigators and independent investigators could engage in open, scientific discourse. NRC would have provided full documentation of all supporting experiments.

(IV-4) Further to paragraph IV-3, NRC would have performed Tasks 1 through 6 with explicit treatment of uncertainties. Also, NRC would have done sensitivity analyses to test the implications of changing modeling assumptions or input conditions. At this stage of risk assessment, however, modeling of mitigating actions would have been premature.

(IV-5) Completing Tasks 1 through 6, consistent with paragraphs IV-3 and IV-4, would have involved the publication of a number of documents, including NRC analyses, independent analyses, peer reviews, and responses to those reviews. The issues addressed would be purely technical, pertaining to Tasks 1 through 6 as described above. When all issues had been resolved to a reasonable scientific standard, a summary document would be published. Then, and only then, would NRC have been ready to move to the next analytic step.

(IV-6) The next analytic step would have been to identify and characterize a full range of event sequences that could lead to the combinations of conditions that would, according to the analysis done in Tasks 1 through 6, be associated with a significant radioactive release. Hereafter, for simplicity, I refer to this step as "event analysis". If assessment of pool-fire risk is to be done properly, it is essential that event analysis be preceded by acquisition of a thorough understanding of pool-fire phenomena. Otherwise, analysts would lack essential knowledge about how particular combinations of conditions could affect fuel ignition and fire dynamics. In the absence of such knowledge, it is likely that analysts would ignore or misunderstand some event sequences that are significant to pool-fire risk.

(IV-7) The event sequences addressed in a properly-executed event analysis would include a range of potential accidents and attacks. Earthquake would certainly be considered as a possible initiating event, but other types of credible initiating event would receive at least equal attention. Careful attention would be given to potential risk linkages between reactors and pools, as discussed in paragraphs III-8 and III-9, above. In this context, the 2011 Fukushima accident was a wake-up call. Figure IV-1 illustrates two aspects of such linkages. First, the Unit 4 building at Fukushima was badly damaged by explosion of hydrogen that has been attributed to core damage in Unit 3. Second, a concrete-pumping truck was, at the time of this photograph, providing makeup water to the Unit 4 pool, reminding us of several days of futile attempts, earlier in the accident, to provide makeup water to Units 1 through 4 by other means.

(IV-8) Fortunately, the Fukushima accident did not proceed to a pool fire. However, any competent analyst who thinks about the Fukushima accident could readily identify a range of event sequences in which a core melt would be linked to a pool fire. Such an event sequence need not involve an earthquake or tsunami. The key point is that the event sequence would involve a timeframe such that a portion of the fuel in the pool would be above water, in a situation involving limited heat transfer, for a period long enough that the youngest fuel would heat up to its ignition temperature. The Severe Reference scenario for water loss, as articulated in paragraph III-6, above, addresses this point.

(IV-9) This declaration is intended for general distribution. Accordingly, it does not contain any information that would assist persons who could plausibly attack a US nuclear power plant. A large body of information of this type is already in the public domain. Moreover, many persons in the USA and worldwide have already acquired, through military experience or otherwise, the knowledge and practical skills that would be needed to mount a plausible attack. At any given time, some persons in that group may have motivation and resources sufficient to mount an attack with a substantial conditional probability of causing a reactor core melt and/or pool fire. The feasibility of such an attack is illustrated by the publicly-available information presented in Tables IV-1 through IV-3 and Figures IV-2 through IV-5. The probability of such an attack is cumulative across the population of nuclear power plants and the years of their operation.

V. NRC's Draft Consequence Study: Structure, Apparent Scope, and Messages

(V-1) Section IV, above, explains why any NRC study on pool-fire risk that is published now (i.e., mid-2013) should have focused exclusively on establishing a solid technical understanding of phenomena directly related to a potential pool fire. Such a study, done appropriately, could potentially have established NRC as a credible source of information about pool-fire risk. NRC did not follow that path. Indeed, NRC took a radically different approach. It published a study that is misleading, incomplete in its examination of risk, and designed to support pre-determined conclusions.

(V-2) NRC's Draft Consequence Study is structured as though it were a comprehensive assessment of the risk of a pool fire. It begins by identifying a single threat – an earthquake – and proceeds through a series of steps that end with a "regulatory analysis" (Appendix D) to determine if the threat justifies expedited transfer of spent fuel to dry storage. The scope of the Study is actually much narrower than would be the case in a comprehensive assessment, as discussed in Section VI, below. The Study itself acknowledges this fact in its interior sections. However, the Study's initial sections – Foreword, Abstract, and Executive Summary – propagate a different story. As NRC personnel undoubtedly know, many readers of the Study will never penetrate beyond these initial sections. Such readers will receive strong messages that the risk of a pool fire is very low, that expedited transfer of spent fuel to dry storage is not necessary, and that further analysis would not alter these findings.

(V-3) One of the messages in the Study's initial sections is that, by considering a particular earthquake threat, the Study has addressed the major source of risk of a pool fire. In this context, the Study says:[30]

> "Previous studies have shown that earthquakes present the dominant risk for spent fuel pools, so this analysis considered a severe earthquake with ground motion stronger than the maximum earthquake reasonably expected to occur for the reference plant."

(V-4) To complement that message, the Study provides strong messages that the risk of a pool fire is very low, and expedited transfer of spent fuel is not necessary. In those contexts, the Study says:[31]

> "This study's results are consistent with earlier research studies' conclusions that spent fuel pools are robust structures that are likely to withstand severe earthquakes without leaking cooling water and potentially uncovering the spent fuel. The study shows the likelihood of a radiological release from the spent fuel after the analyzed severe earthquake at the reference plant to be about one time in 10 million years or lower. In addition, the regulatory analysis included with this study does not support accelerated spent fuel transfer to casks for the reference plant."

(V-5) Expedited transfer of spent fuel to dry storage would allow a pool to be re-equipped with low-density, open-frame racks. As discussed in paragraphs I-6 and III-3, above, the pool-fire issue became salient in the 1970s when the nuclear industry abandoned the use of low-density, open-frame racks and switched to high-density, closed-frame racks. Thus, if a concerned citizen learns that NRC is now studying the merit of a switch to low-density pool storage, that citizen could reasonably assume that NRC is considering the use of low-density, open-frame racks. Such a citizen, reading only the initial sections of NRCs Draft Consequence Study, would not encounter any information to contradict that assumption.[32] Moreover, the citizen would be told that a

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[30] Barto et al, 2013, Executive Summary, page vi.
[31] Barto et al, 2013, Executive Summary, page vi.
[32] The Study's Executive Summary refers to high-density and low-density scenarios for pool loading. (See: Barto et al, 2013, Executive Summary, page vi.) A person reading only the initial sections of the Study would be unlikely to realize that the allegedly low-density scenario does not involve the use of open-frame racks.

switch to low-density storage would not reduce the potential for a pool fire. In this context, NRC says:[33]

> "The likelihood of a spent fuel pool release [due to a pool fire] was equally low for both high- and low-density fuel loading. This is because high- and low-density fuel loading contains the same amount of new, hotter spent fuel recently moved from the reactor to the spent fuel pool."

(V-6) The preceding NRC statement is highly misleading. As discussed in paragraph III-4, above, if low-density, open-frame racks were used, then water loss from a pool would lead to fuel ignition only in very rare circumstances. NRC does not dispute that fact. Instead, NRC uses the phrase "low density" to refer to a situation in which a substantial fraction of the cells in a high-density, closed-frame rack do not contain fuel. That situation cannot offer the dramatic reduction in pool-fire risk that would come from reverting to low-density, open-frame racks.

(V-7) This one example demonstrates that the initial sections – Foreword, Abstract, and Executive Summary – of NRC's Draft Consequence Study contain a highly misleading statement. Given that the Study is lengthy and complex, many readers will not penetrate beyond these initial sections, as NRC personnel undoubtedly know. Thus, it is reasonable to conclude that NRC made this misleading statement deliberately, in order to serve some purpose.

(V-8) In Section VI, below, I discuss this one example further. I also discuss other instances in which NRC's Draft Consequence Study is misleading, incomplete in its examination of risk, and/or designed to support pre-determined conclusions.

VI. NRC's Draft Consequence Study: Actual Scope, and Credibility

(VI-1) As discussed in Section V, above, NRC's Draft Consequence Study seeks to create the appearance of being a comprehensive assessment of the risk of a pool fire. That image is conveyed by the structure of the Study, by the way the Study is described in its Foreword, Abstract, and Executive Summary, and by unequivocal statements that high-density spent-fuel pools protect public health and safety.[34] In fact, the Study's scope is narrow. As a result, the Study cannot support the broad findings that it presents.

(VI-2) To its credit, the Study does acknowledge the limitations in its scope, to a reader who penetrates to the interior sections of the Study. For example, Section 2 of the Study articulates many of the questionable assumptions and analytic limitations that permeate the Study. Overall, the Study has misleading parts and comparatively honest parts. This internal difference may be attributable to different authorship for different parts.

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[33] Barto et al, 2013, Executive Summary, page vii.
[34] For example, the Study's Executive Summary concludes with the statement: "The NRC continues to believe, based on this study and previous studies that spent fuel pools protect public health and safety." (See: Barto et al, 2013, Executive Summary, page xii.)

(VI-3) As discussed in paragraphs V-5 and V-6, above, the Study claims to compare the respective risks posed by high-density and low-density modes of fuel storage in a pool. In fact, the Study makes no such comparison. Instead, the Study adopts misleading terminology, using the phrase "low density" to refer to a reduced inventory of fuel in a high-density, closed-frame rack. NRC explains its failure to assess the risk implications of reverting to low-density, open-frame racks with the following statement:[35]

> "Re-racking the pool would represent a significant expense, along with additional worker dose, and was not felt to be the likely regulatory approach taken based on consultation with the Office of Nuclear Reactor Regulation. Much of the benefit of low-density racking is achieved by the implementation of a favorable fuel pattern (1x4). Additionally, to get the full benefit of low-density racking, BWR fuel would likely need to have the channel boxes removed."

(VI-4) This statement by NRC is revealing. It shows that, when NRC began the Study, some of its conclusions were pre-determined. In this instance, NRC rejected the option of reverting to low-density, open-frame racks on the basis of no analysis whatsoever. This rejection was done before the Study commenced, on the basis of a "feeling".

(VI-5) As discussed in Section III, above, between 1979 and 2000 NRC's work on pool-fire risk employed the erroneous assumption that complete drainage of a pool would be the most severe case of water loss. This error apparently arose from the failure of NRC personnel to fully understand a 1979 Sandia report that NRC had commissioned. NRC indirectly acknowledged this error in 2000.

(VI-6) Curiously, in light of this history, NRC's Draft Consequence Study focuses exclusively on complete drainage of a pool. The Study examines two cases. In the "moderate" leak case, drainage would be complete after about 6 hours, while in the "small" leak case, drainage would be complete after about 40 hours.[36] Such cases are more useful for pool-fire risk analysis than the assumption of instantaneous, total drainage, which NRC employed in some of its previous studies. However, these two cases do not cover a full range of water-loss scenarios. Notably, they do not cover the Severe Reference scenario and its variants, as discussed in paragraph III-6, above. That scenario, although not necessarily the "worst" case of water loss from a pool, does capture the role of residual water in the pool.

(VI-7) The implications of the presence of residual water for fuel ignition are illustrated by some simple calculations set forth in Section VII, below. These calculations assume a pool loading (see Figure VII-1) and operating cycle phase (OCP4) as used in NRC's Draft Consequence Study. The contrast with that study is that drainage of water would not be complete. Instead, residual water would be present in the pool for an extended period. The calculations yield estimates of the time between fuel exposure and fuel ignition. Here, I refer to that time as "ignition delay time". Results are summarized in

[35] Barto et al, 2013, Table 3, page 23.
[36] Barto et al, 2013, Figures 52 and 54.

paragraph VII-13, below. Assuming an adiabatic situation for exposed fuel yields an ignition delay time of about 5 hours. Extrapolation of NRC's moderate-leakage and small-leakage cases yields ignition delay times of about 7 hours and 20 hours, respectively.

(VI-8) These time estimates provoke two immediate questions. First, how significant for risk is an ignition delay time in the range 5 to 20 hours? Second, how accurate are these time estimates? I address these questions in order, in the following two paragraphs.

(VI-9) During the Fukushima accident in 2011, the Japanese nuclear industry and government struggled unsuccessfully for several days to establish water makeup to spent-fuel pools. Eventually, they established water makeup using the concrete-pumping truck shown in Figure IV-1. Yet, the Fukushima experience was far from a worst case in terms of onsite phenomena, such as radioactive contamination from a reactor core melt accident, that could preclude mitigating actions. Thus, we have ample evidence that water makeup and other mitigating actions could be precluded for a period substantially exceeding 20 hours. Accordingly, if the ignition delay time is 20 hours, or even longer, it is entirely realistic to consider an event sequence involving: (i) an initial rapid exposure of fuel followed by the presence of residual water for an extended time; (ii) no water makeup; (iii) fuel ignition; and (iv) propagation of a pool fire.

(VI-10) As to the accuracy of these time estimates, neither the adiabatic assumption nor the extrapolation from NRC findings is adequate for the purpose of thoroughly investigating pool-fire risk. However, in the absence of better analysis, these estimates are reasonable for illustrative purposes. Appropriate analysis would require sophisticated modeling backed by experiment, done in a scientific manner. NRC has never done such analysis in a pool-fire context.

(VI-11) These illustrative calculations show that a pool fire could occur if water loss occurred during a particular operating cycle phase – OCP4. NRC's Draft Consequence Study finds (see Figures VII-2 and VII-3) that a pool fire would not occur in OCP4 with the same pool loading. That finding reflects NRC's decision to focus its analysis exclusively on water-loss scenarios involving total drainage of water from a pool. By adopting that focus, NRC has ignored a substantial part of the pool-fire risk.

(VI-12) Water could be lost from a pool as a result of an accident or an attack. NRC's Draft Consequence Study dismisses the possibility of an attack by stating:[37] "Note that sabotage events have been excluded from the scope of this study." No further explanation is offered. Thus, NRC arbitrarily excludes a category of events that contributes substantially to pool-fire risk. As discussed in paragraph IV-9, above, an attack causing a reactor core melt and/or pool fire is a credible threat. The probability of an attack with a substantial likelihood of success is at least equal to the probability of the earthquake that NRC does consider (i.e., 1 in 60,000 years).[38] Also, knowledgeable

[37] Barto et al, 2013, page 8.
[38] Barto et al, 2013, Figure ES-2, page x.

attackers could time and shape their attack in a manner that maximizes the potential for radioactive release.

(VI-13) As discussed in paragraphs IV-7 and IV-8, above, risk linkages among pools and reactors at a particular site could be major determinants of pool-fire risk at that site. NRC's Draft Consequence Study actually provides a useful introduction to these linkages – which they term "interplays" – under the rubric of "multi-unit considerations".[39] Having identified this risk-significant issue, the Study goes on to say:[40]

> "To the extent practicable, this study has attempted to qualitatively account for some of these effects. For example, when the reactor and SFP are hydraulically connected (during refueling), the decay heat and water volumes from both sources are considered. The study also explores these effects on mitigation (Section 8), and addresses some aspects of the uncertainty associated with this treatment (Section 9). However, explicitly modeling multiunit effects was not a focus of this study, because of the existing limitations with the available computational tools. An ongoing project described in SECY-11-0089 will attempt to more rigorously address these effects in the framework of a multiunit Level 3 PRA for Vogtle Electric Generating Plant Units 1 and 2."

(VI-14) In other words, NRC recognizes that pool-reactor linkages are significant to risk, says that a future effort will "attempt" to overcome the limitations of relevant analytic tools, but cannot resist the temptation to include a shoddy treatment of these linkages in NRC's Draft Consequence Study. That inclusion adds to the misleading nature of the Study.

(VI-15) Paragraph IV-2, above, discusses the need to consider "collateral conditions" in a thorough investigation of pool-fire phenomena. One such condition would be the presence of debris in a pool. NRC acknowledges the significance of this issue and then proceeds to ignore it, further adding to the misleading nature of the Study. NRC says:[41]

> "The occurrence of a hydrogen combustion event from a concurrent reactor accident has the potential to generate debris which could impair SFP natural circulation air or steam cooling (should the fuel in the SFP become uncovered) for conditions in which the fuel might otherwise be cooled by means of these passive cooling modes. However, this latter situation is inherently tied to the study's lack of a comprehensive treatment of multiunit aspects."

(VI-16) NRC's Draft Consequence Study focuses its attention exclusively on one pool-fire initiating event – an earthquake with a probability of 1 in 60,000 years. At the same time, as discussed above, NRC acknowledges the risk significance of pool-reactor linkages but proceeds to ignore them. Yet, the probability of a reactor core melt is at least equal to the probability of the earthquake that NRC does consider. Generation 2

[39] Barto et al, 2013, Section 2.2, pp 28-29.
[40] Barto et al, 2013, page 29.
[41] Barto et al, 2013, Table 3, page 25.

commercial reactors have accrued about 15,000 reactor-years of operating experience worldwide, and have experienced five core melts.

(VI-17) The feasibility and effectiveness of mitigating actions – such as providing makeup water to a pool – are significant to pool-fire risk. The Study addresses this matter in its Section 8, under the rubric of "human reliability analysis". In the Study, human error probability is equated to mitigation failure probability. The Study acknowledges the limitations of its analysis in this area, saying:[42]

> "Consistent with the limited scope of the SFPS, a limited scope human reliability analysis (HRA) was performed, to develop initial insights into the likelihood of successful operator actions to prevent spent fuel damage for the specific seismic event and consequence scenarios studied. A full scope HRA would primarily be useful as part of a PRA analysis. A PRA would necessarily consider a much broader scope than the SFPS."

(VI-18) Despite this acknowledgment, the Study proceeds to make unequivocal statements about the feasibility of mitigation. For example, in addressing the potential for a boil-off scenario of water loss, the Study says that the probability of mitigation failure extending for 7 days is "negligible".[43] That statement is based on no analysis, and reflects a pre-determined conclusion. NRC ignores, for example, the possibility that radiation fields and other onsite impacts of a reactor core melt could preclude mitigation for an extended period.

(VI-19) NRC's Draft Consequence Study addresses an issue that is significant in terms of public health and safety. This significance is illustrated by one of the Study's findings. In modeling the offsite impacts of a potential pool fire, the Study considers a case in which modeling indicates that 4.1 million people would experience long-term displacement from their homes.[44]

VII. NRC's Use of the MELCOR Code

(VII-1) NRC has adapted the MELCOR code package, version 1.8.6, to examine the physical and chemical phenomena directly associated with a potential pool fire. Section 6 of NRC's Draft Consequence Study describes MELCOR and its use in this instance. Here, I discuss selected points regarding this application of MELCOR. This discussion does not purport to be a comprehensive review, but addresses some important points.

(VII-2) In Section IV, above, I outline a process whereby a code such as MELCOR could be used to address pool-fire issues in a manner consistent with the principles of science. The process would include NRC funding of independent investigators who would have access to MELCOR, and NRC funding of independently-run workshops where NRC investigators and independent investigators could engage in open, scientific

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[42] Barto et al, 2013, page 173.
[43] Barto et al, 2013, page 175.
[44] Barto et al, 2013, Table 33, page 162.

discourse. To my knowledge, NRC's application of MELCOR in the pool-fire context has not employed such a process.

(VII-3) MELCOR was developed to model a reactor core melt. Accordingly, its fuel-behavior module employs a two-dimensional cylindrical geometry. By contrast, a pool, in plan view, is a rectangle within which the racks form a combination of rectangles. In an effort to accommodate this difference, NRC has assumed that spent fuel in a pool would be arranged in "rings" whose boundaries roughly approximate concentric circles, with overlap between some of these boundaries. Figure VII-1 illustrates this assumption. Each ring would be composed of fuel with a particular age and burnup. Also, NRC has added a modeling capability to account for the presence of racks, which are not present in a reactor core.

(VII-4) If NRC's application of MELCOR had employed a scientific process as discussed above, then an independent reviewer could examine the associated documents and form a professional opinion on the validity of NRC's findings. To my knowledge, no such documents exist. Thus, at this time, I do not have a professional opinion on the quality of the MELCOR findings presented by NRC. It is, however, easy to identify issues and questions that should be addressed in a scientific process to examine NRC's findings. Consider, for example, two issues pertaining to the validity of MELCOR in the pool-fire context:

1. MELCOR has no capability to model the deformation of fuel cladding as temperature rises. Yet, NUREG-1738 predicted that cladding would balloon and burst in a temperature range of 700–850°C. That outcome could reduce heat transfer and promote ignition of cladding. NRC says that these effects would not be significant, but rests that claim on secret, unpublished studies.[45]
2. Radiative heat transfer is an important consideration in pool-fire modeling. Yet, MELCOR employs a simplified approach to modeling this mode of heat transfer. In this context, NRC says:[46] "It should be noted that there is a temperature gradient within each ring, and MELCOR attempts to model a multidimensional geometry with a simplified two-surface radiation model."

(VII-5) In addition to questions about the validity of MELCOR, there are questions about NRC's input assumptions. For example, how closely does the pool layout shown in Figure VII-1 correspond with actual practice in the nuclear industry? In that context, there is a puzzling NRC assumption associated with Figure VII-1. That figure shows a total of 284 newly-discharged fuel assemblies. Of these, 88 assemblies are assumed by NRC to produce decay heat at the rate of 10.9 kW per assembly when aged 20 days, while the remaining 196 assemblies produce 6.6 kW per assembly at the same age.[47] If this is typical practice, then licensees are forgoing substantial available burnup of the

[45] Barto et al, 2013, Table 3, page 26.
[46] Barto et al, 2013, footnote 23, page 110.
[47] These decay heat outputs are calculated from data in Table 25 of: Barto et al, 2013. The same data apply to Figure VII-1 in this declaration.

majority of their fuel assemblies, with a resulting economic penalty.[48] As a related matter, Figure VII-1 shows a rather elaborate layout of fuel, whose achievement would involve substantial shuffling of assemblies. NRC says that this layout is comparatively favorable in terms of the risk of a pool fire. Yet, licensees are allowed a period of time, during and perhaps after a refueling outage, to perform the shuffling needed to achieve a favorable layout. The length of that period of time is a secret because, NRC says, this information could be useful to an adversary.[49] Thus, is it appropriate to assume, as a MELCOR input, that a comparatively favorable layout has been achieved before water is lost?

(VII-6) In the Study, NRC has focused its analysis exclusively on water-loss scenarios involving total drainage of water from a pool. By adopting that focus, NRC has ignored a substantial part of the pool-fire risk. Here, I provide some simple calculations that illustrate the implications of NRC's narrow focus. These calculations show how the presence of residual water could affect fuel ignition. One calculation employs the simplifying assumption that, if residual water is present, the exposed portion of a fuel assembly in a high-density rack is in an adiabatic situation. Using that assumption, anyone with technical training can use pencil and paper to calculate the time required for the temperature of the fuel cladding to rise to its ignition point. The other calculations determine that time by extrapolating from NRC's findings using MELCOR. As indicated above, I do not necessarily accept that MELCOR is valid for its application by NRC to the pool-fire problem, or that NRC's input assumptions are appropriate.

(VII-7) These illustrative calculations consider loss of water from the pool considered in NRC's Draft Consequence Study. This event would occur during operating cycle phase 4 (OCP4). According to NRC, OCP4 and higher-risk phases account for 34 percent of the duration of the total operating cycle.[50] Attention is focused here on Ring 1 fuel, as shown in Figure VII-1. The pool would be loaded at high density.

(VII-8) The assumed scenario for water loss is the Severe Reference scenario as articulated in paragraph III-6, above. Initially, water level would fall rapidly to a point between the top and bottom of the racks. Thereafter, residual water would be lost comparatively slowly by evaporation.[51] The presence of residual water would block air flow beneath the racks. The exposed portion of the fuel would gradually increase in temperature.

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[48] Other factors being equal, decay heat output increases with burnup.
[49] Barto et al, 2013, Section 9.3, page 208.
[50] Barto et al, 2013, Table 16, page 78.
[51] The rate of loss of residual water by evaporation can be estimated as follows. The floor of the pool is 12.2 m by 10.8 m (Barto et al, 2013, page 103) and the total decay heat output in OCP4 is 1,868 kW (Barto et al, 2013, Table 25). Let the submerged fraction of the active length of the fuel be F_s and assume uniform output of decay heat along the active length. Assume 60% water content by volume in the lower portion of the pool. Set water density at 960 kg/m^3 and latent heat of evaporation at 2,260 kJ/kg. Then, the rate of fall of the water surface due to evaporation = F_s(1,868)/((2,260)(960)(0.6)(12.2x10.8)) = F_s(1.09E-05) m/s = 0.04F_s m/hr. For comparison, active length of the fuel is about 4 m.

(VII-9) For the first illustrative calculation, assume that the exposed portion of the fuel is in an adiabatic situation. As shown in Table VII-1, it is easy, with this assumption, to calculate the rate at which fuel temperature would rise. According to NRC, Ring 1 fuel in OCP4 has a decay heat output of 26.6 kW per Mg U.[52] From Table VII-1, one sees that fuel temperature would rise at the rate of 170 K per hour.[53]

(VII-10) Now, consider MELCOR outputs for NRC's examination of a moderate-leakage case in OCP4, as shown in Figure VII-2. During the evolution of this case, there would be a period of time when the upper portion of the fuel is exposed and residual water is present. That period would extend from about t = 3 hours to about t = 6 hours. During that 3-hour period, at the "Lev 5" elevation of the fuel, cladding temperature would rise from about 300 K to about 700 K. Thus, the average rate of temperature rise would be about 130 K per hour. This finding indicates that the exposed portion of the fuel at the Lev 5 elevation would be in an approximately adiabatic situation, at least for temperatures up to 700 K.

(VII-11) Now, apply the same process, as in the preceding paragraph, to NRC's examination of a small-leakage case as shown in Figure VII-3. In that case, the period of time when fuel at the Lev 5 elevation is exposed and residual water is present would extend from about t = 28 hours to about t = 40 hours. During that 12-hour period, the temperature of fuel cladding at the Lev 5 elevation would rise from about 350 K to about 900 K. Thus, the average rate of temperature rise would be about 46 K per hour.

(VII-12) The slower average temperature rise in NRC's small-leakage case, compared to the moderate-leakage case, appears to be attributable to a MELCOR finding that heat transfer from exposed fuel would be more effective at temperatures between 700 K and 900 K than it would be at temperatures below 700 K.[54] Radiative heat transfer would be a substantial contributor to that effect.

(VII-13) NRC assumes that zircaloy ignition would occur at a temperature of about 1,200 K. If the initial fall of water level is rapid, then exposed fuel would have an initial temperature of about 300 K. Thus, ignition would require a temperature rise of about 900 K. Accordingly, the three illustrative calculations, as described above, yield a time to ignition, after exposure of fuel, as follows:

1. Adiabatic assumption: Adiabatic heatup would lead to a temperature rise of 170 K per hour. Thus, time to ignition = 900/170 = 5.3 hours
2. Extrapolation of NRC's moderate-leakage case: If temperature rise continued at 130 K per hour, time to ignition = 900/130 = 6.9 hours
3. Extrapolation of NRC's small-leakage case: If temperature rise continued at 46 K per hour, time to ignition = 900/46 = 19.6 hours

[52] NRC says (Barto et al, 2013, Table 25) that 88 Ring 1 fuel assemblies have a combined decay heat output of 422 kW in OCP4. If the mass of one assembly is assumed to be 0.18 Mg U, then decay heat output = (422/88)/(0.18) = 26.6 kW per Mg U.
[53] In Table VII-1, set R = 26.6 kW per Mg U. Then, rate of temperature rise = (26.6)(6.38) = 170 K/hr.
[54] Note the respective shapes of the Lev 5 curves in the temperature-time charts in Figures VII-2 and VII-3.

(VII-14) Extrapolation of NRC's findings is reasonable for illustrative purposes, in the absence of better analysis. However, neither the adiabatic assumption nor the extrapolation used here is adequate for the purpose of thoroughly investigating pool-fire risk. As discussed in Section IV, above, a thorough, comprehensive investigation would begin by establishing a solid technical understanding of phenomena directly related to a potential pool fire, including heat transfer, zircaloy ignition, and fire dynamics. The necessary modeling and experimental work would be done according to scientific principles. That work could yield, for example, scientifically-defensible estimates of ignition delay time in a Severe Reference scenario for water loss. It is far from clear that MELCOR can yield good estimates of this time, given MELCOR's simplified treatment of radiative heat transfer.

(VII-15) If ignition of fuel occurred in a Severe Reference scenario for water loss, the fire would begin as a steam-zircaloy reaction. Progress of the fire would be limited by the amount of steam that would be generated from residual water and rise through each fuel assembly. Note, however, that the flow of steam reaching the exposed portion of a particular assembly would be determined primarily by the decay heat output of that assembly. Thus, for a pool layout as shown in Figure VII-1, Ring 1 fuel would not only be the first fuel in the pool to experience steam-zircaloy ignition, but would also experience the highest flow of steam that could feed a steam-zircaloy fire.

VIII. Conclusions and Recommendations

Conclusions

(VIII-1) Prior to publication of the Draft Consequence Study, NRC's technical credibility on the pool-fire issue was low. Over a period exceeding three decades, NRC had published bad analysis and hidden other analysis behind a veil of secrecy. Moreover, NRC failed to conduct sophisticated modeling and supporting experiments that could have resolved technical issues central to pool-fire risk, despite having an appropriate capability prior to 1990.

(VIII-2) NRC's Draft Consequence Study seeks to create the appearance of being a comprehensive assessment of the risk of a pool fire. That image is conveyed by the structure of the Study, by the way the Study is described in its Foreword, Abstract, and Executive Summary, and by unequivocal statements that high-density spent-fuel pools protect public health and safety. In fact, the Study's scope is narrow. As a result, the Study's examination of pool-fire risk is incomplete, and cannot support the broad, unequivocal findings that the Study presents. This disjunction between the apparent and actual scope of the Study is misleading. Moreover, in specific instances, the Study is misleading and is designed to support pre-determined conclusions. Examples of specific deficiencies in the Study are provided in the following paragraph.

(VIII-3) Some specific instances in which NRC's Draft Consequence Study is incomplete, misleading, and/or designed to support pre-determined conclusions are as follows:

1. Pretence of considering low-density storage: The Study does not consider the risk implications of reverting to low-density, open-frame racks. Instead, NRC misuses the phrase "low density" in order to create a false impression of the Study's scope. This pretence reflects pre-determined conclusions based on a "feeling".
2. Limited consideration of water-loss scenarios: The Study focuses its analysis exclusively on water-loss scenarios involving total drainage. By so doing, the Study ignores a substantial part of the pool-fire risk. For example, the Study makes no effort to determine how the presence of residual water could affect fuel ignition. Extrapolation of Study findings indicates that consideration of this issue would substantially increase the estimated risk.
3. Limited consideration of initiating events: The Study considers only one type of initiating event – an earthquake. That narrow focus reflects a pre-determined conclusion that earthquake is the dominant contributor to the risk of a pool fire.
4. No consideration of attack: The Study ignores the potential for an attack on a pool and/or adjacent reactor to initiate a pool fire. Yet, the probability of an attack with a substantial likelihood of success is at least equal to the probability of the severe earthquake that the Study does consider. Thus, the Study significantly under-estimates pool-fire risk.
5. No analysis of risk linkages among pools and reactors: The Study identifies the potential for risk linkages, but does not properly analyze them. For example, the Study does not analyze a situation in which onsite radioactive contamination and other impacts of a reactor core melt would preclude mitigating actions that might prevent a pool fire. Yet, the probability of a core melt at an adjacent reactor is at least equal to the probability of the severe earthquake that the Study does consider. Thus, the Study significantly under-estimates pool-fire risk.
6. Misleading statements regarding mitigating actions: The Study concedes that its analysis of the feasibility of mitigating actions is very limited. Yet, the Study makes unequivocal statements about this feasibility. Some of those statements are misleading, and reflect pre-determined conclusions.

(VIII-4) In the Study, NRC employs the MELCOR code to model phenomena related to a pool fire – including heat transfer, cladding ignition, and fire dynamics. MELCOR findings are significant to NRC's estimation of pool-fire risk. Yet, the validity of MELCOR in this context, and the appropriateness of NRC's input assumptions, have not been tested through a process of open scientific inquiry. There are significant issues that should be addressed through such a process, including MELCOR's simplified treatment of radiative heat transfer.

(VIII-5) In the Study, NRC has erected an elaborate superstructure of analysis on a weak foundation of basic knowledge about pool-fire phenomena. This superstructure culminates in a regulatory analysis. As discussed in paragraph VIII-2, above, the findings emanating from this superstructure lack scientific credibility and are misleading. Thus, the design of the Study is fundamentally and irredeemably flawed.

(VIII-6) The Study addresses an issue that is significant in terms of public health and safety. This significance is illustrated by the Study's finding that a pool fire could lead to long-term displacement from their homes of more than 4 million people. Thus, citizens deserve a much better analysis of pool-fire risk than the incomplete, misleading work presented in NRC's Draft Consequence Study.

Recommendations

(VIII-7) NRC's Draft Consequence Study should be scrapped.

(VIII-8) In addressing the pool-fire issue, NRC should focus its initial attention exclusively on establishing a solid technical understanding of phenomena directly related to a potential pool fire. To do this, NRC would start with a clean slate and use the best available modeling capability backed up by experiment. This modeling and experimental work would be done according to scientific principles. Further recommendations regarding such work are provided in Section IV, above.

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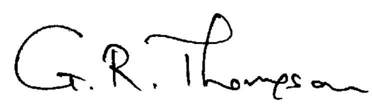
I declare, under penalty of perjury, that the facts set forth in the foregoing narrative, and in the two appendices below, are true and correct to the best of my knowledge and belief, and that the opinions expressed therein are based on my best professional judgment.

Executed on 1 August 2013.



Gordon R. Thompson

APPENDIX A: Tables and Figures

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Table III-1
Potential Sabotage Events at a Spent-Fuel Pool, as Postulated in NRC's August 1979 Generic EIS on Handling and Storage of Spent LWR Fuel

Event Designator	General Description of Event	Additional Details
Mode 1	• Between 1 and 1,000 fuel assemblies undergo extensive damage by high-explosive charges detonated under water • Adversaries commandeer the central control room and hold it for approx. 0.5 hr to prevent the ventilation fans from being turned off	• One adversary can carry 3 charges, each of which can damage 4 fuel assemblies • Damage to 1,000 assemblies (i.e., by 83 adversaries) is a "worst-case bounding estimate"
Mode 2	• Identical to Mode 1 except that, in addition, an adversary enters the ventilation building and removes or ruptures the HEPA filters	
Mode 3	• Identical to Mode 1 within the pool building except that, in addition, adversaries breach two opposite walls of the building by explosives or other means	• Adversaries enter the central control room or ventilation building and turn off or disable the ventilation fans
Mode 4	• Identical to Mode 1 except that, in addition, adversaries use an additional explosive charge or other means to breach the pool liner and 1.5 m-thick concrete floor of the pool	

Notes:
(a) Information in this table is from Appendix J of: NRC, 1979.
(b) The postulated fuel damage ruptures the cladding of each rod in an affected fuel assembly, releasing "contained gases" (gap activity) to the pool water, whereupon the released gases bubble to the water surface and enter the air volume above that surface.

Table IV-1
Some Potential Modes and Instruments of Attack on a Nuclear Power Plant

Attack Mode/Instrument	Characteristics	Present Defenses at US Plants
Commando-style attack	• Could involve heavy weapons and sophisticated tactics • Successful attack would require substantial planning and resources	Alarms, fences and lightly-armed guards, with offsite backup
Land-vehicle bomb	• Readily obtainable • Highly destructive if detonated at target	Vehicle barriers at entry points to Protected Area
Small guided missile (anti-tank, etc.)	• Readily obtainable • Highly destructive at point of impact	None if missile launched from offsite
Commercial aircraft	• More difficult to obtain than pre-9/11 • Can destroy larger, softer targets	None
Explosive-laden smaller aircraft	• Readily obtainable • Can destroy smaller, harder targets	None
10-kilotonne nuclear weapon	• Difficult to obtain • Assured destruction if detonated at target	None

Notes:
(a) This table is adapted from: Thompson, 2007, Table 7-4. Further citations are provided in that table and its supporting narrative. For additional, supporting information of more recent vintage, see: Ahearne et al, 2012, Chapter 5.
(b) Defenses at nuclear power plants around the world are typically no more robust than at US plants.

Table IV-2
The Shaped Charge as a Potential Instrument of Attack

Category of Information	Selected Information in Category
General information	• Shaped charges have many civilian and military applications, and have been used for decades • Applications include human-carried demolition charges or warheads for anti-tank missiles • Construction and use does not require assistance from a government or access to classified information
Use in World War II	• The German MISTEL, designed to be carried in the nose of an un-manned bomber aircraft, is the largest known shaped charge • Japan used a smaller version of this device, the SAKURA bomb, for kamikaze attacks against US warships
A large, contemporary device	• Developed by a US government laboratory for mounting in the nose of a cruise missile • Described in detail in an unclassified, published report (citation is voluntarily withheld here) • Purpose is to penetrate large thicknesses of rock or concrete as the first stage of a "tandem" warhead • Configuration is a cylinder with a diameter of 71 cm and a length of 72 cm • When tested in November 2002, created a hole of 25 cm diameter in tuff rock to a depth of 5.9 m • Device has a mass of 410 kg; would be within the payload capacity of many general-aviation aircraft
A potential delivery vehicle	• A Beechcraft King Air 90 general-aviation aircraft can carry a payload of up to 990 kg at a speed of up to 460 km/hr • The price of a used, operational King Air 90 in the USA can be as low as $0.4 million

Source:
This table is adapted from Table 7-6 of: Thompson, 2009.

Table IV-3
Performance of US Army Shaped Charges, M3 and M2A3

Target Material	Indicator	Value for Stated Type of Shaped Charge	
		Type: M3	**Type: M2A3**
Reinforced concrete	Maximum wall thickness that can be perforated	150 cm	90 cm
	Depth of penetration in thick walls	150 cm	75 cm
	Diameter of hole	• 13 cm at entrance • 5 cm minimum	• 9 cm at entrance • 5 cm minimum
	Depth of hole with second charge placed over first hole	210 cm	110 cm
Armor plate	Perforation	At least 50 cm	30 cm
	Average diameter of hole	6 cm	4 cm

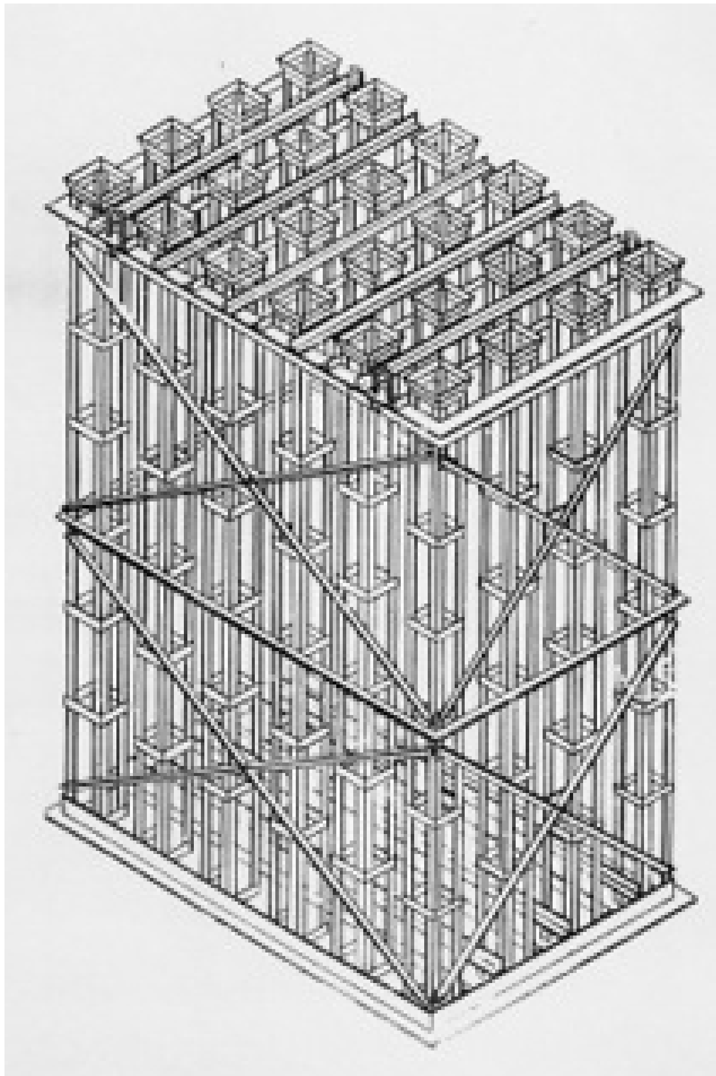
Notes:
(a) Data are from US Army Field Manual FM 5-25: Army, 1967, pp 13-15 and page 100.
(b) The M2A3 charge has a mass of 5 kg, a maximum diameter of 18 cm, and a total length of 38 cm including the standoff ring.
(c) The M3 charge has a mass of 14 kg, a maximum diameter of 23 cm, a charge length of 39 cm, and a standoff pedestal 38 cm long.

Table VII-1
Adiabatic Heatup of a Spent BWR Fuel Assembly

Indicator	Value	
	Zircaloy	UO_2 Pellets
Mass per Mg U of fuel	564 kg (includes cladding, channel box, and grid spacers)	1,130 kg
Specific heat (av., approx.)	400 J/kg/K	300 J/kg/K
Radioactive decay heat	R kW per Mg U (or W per kg U) of fuel	
Rate of temperature (T) rise from decay heat, if pellets and zircaloy are a tightly coupled adiabatic system	T' = R/(400x0.564 + 300x1.13) K/s = R(1.77E-03) K/s (or R(6.38) K/hr)	

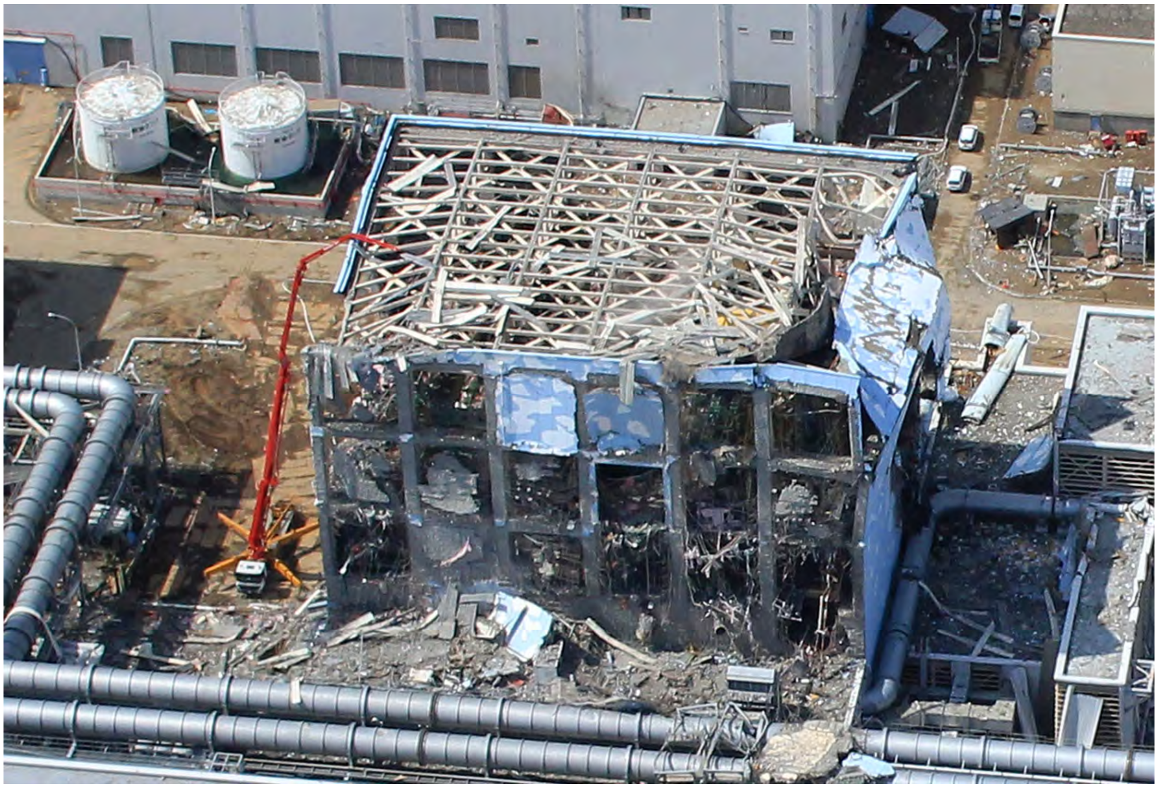
Notes:
(a) Zircaloy mass is from Table 3.2 of: Roddy et al, 1986.
(b) The specific heats shown are averages over the temperature range 100-1,000 °C. For zircaloy, specific heat spikes sharply between about 800 °C and 1,000 °C. (See: IAEA, 1997, Figure 4.2.1.1.) For UO_2, specific heat does not spike until temperature approaches 3,000 K. (See: Popov et al, 2000, Figure 4.2.)
(c) This calculation applies to any portion of the active length of a fuel assembly, provided that decay heat output is uniform along the active length.
(d) The influence of materials other than zircaloy and UO_2 (e.g., fission products) is neglected here. That influence could be examined in a more precise calculation.
(e) No credit is taken here for heat output from exothermic reactions.

Figure III-1
Typical Low-Density, Open-Frame Rack for Pool Storage of PWR Spent Fuel



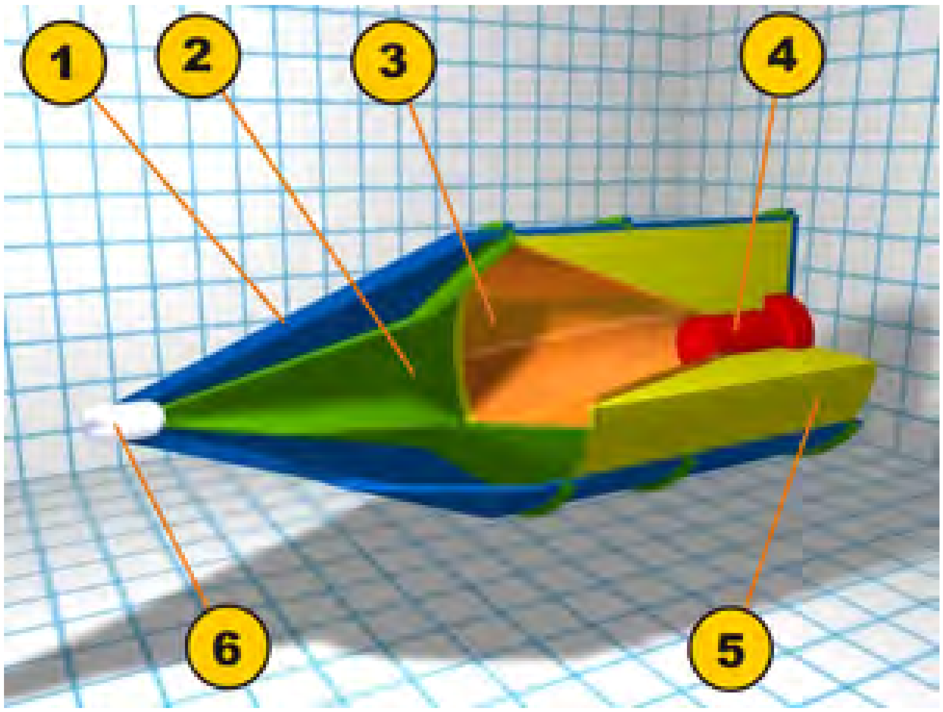
Source:
Adapted from Figure B.2 of: NRC, 1979.

Figure IV-1
Unit 4 at the Fukushima #1 Site During the 2011 Accident



Source:
Accessed on 20 February 2012 from Ria Novosti at: http://en.rian.ru/analysis/20110426/163701909.html; image by Reuters Air Photo Service.

Figure IV-2
Schematic View of a Generic Shaped-Charge Warhead



Notes:
(a) Figure accessed on 4 March 2012 from: http://en.wikipedia.org/wiki/Shaped_charge
(b) Key:

Item 1: Aerodynamic cover
Item 2: Empty cavity
Item 3: Conical liner (typically made of ductile metal)
Item 4: Detonator
Item 5: Explosive
Item 6: Piezo-electric trigger

(c) Upon detonation, a portion of the conical liner would be formed into a high-velocity jet directed toward the target. The remainder of the liner would form a slower-moving slug of material.

Figure IV-3
MISTEL System for Aircraft Delivery of a Shaped Charge, World War II

Notes:
(a) Photograph accessed on 5 March 2012 from:
http://www.historyofwar.org/Pictures/pictures_Ju_88_mistel.html
(b) A shaped-charge warhead can be seen at the nose of the lower (converted bomber) aircraft, replacing the cockpit. The aerodynamic cover in front of the warhead would have a contact fuse at its tip, to detonate the shaped charge at the appropriate standoff distance.
(c) A human pilot in the upper (fighter) aircraft would control the entire rig, and would point it toward the target. Then, the upper aircraft would separate and move away, and the lower aircraft would be guided to the target by an autopilot.

Figure IV-4
January 2008 Test of a Raytheon Shaped Charge, Intended as the Penetration (Precursor) Stage of a Tandem Warhead System

Before Test

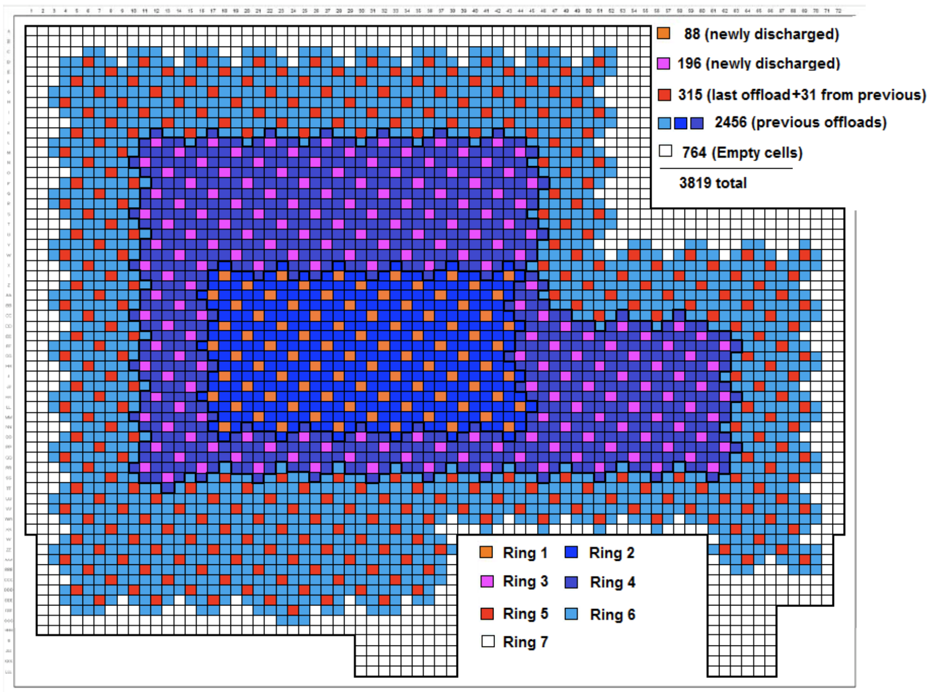
After Test (viewed from the attacked face)

Notes:
(a) These photographs are from: Raytheon, 2008. For additional, supporting information, see: Warwick, 2008.
(b) The shaped-charge jet penetrated about 5.9 m into a steel-reinforced concrete block with a thickness of 6.1 m. Although penetration was incomplete, the block was largely destroyed, as shown. Compressive strength of the concrete was 870 bar.
(c) The shaped charge had a diameter of 61 cm and contained 230 kg of high explosive. It was sized to fit inside the US Air Force's AGM-129 Advanced Cruise Missile.

Figure IV-5
Aftermath of a Small-Aircraft Suicide Attack on an Office Building in Austin, Texas, February 2010

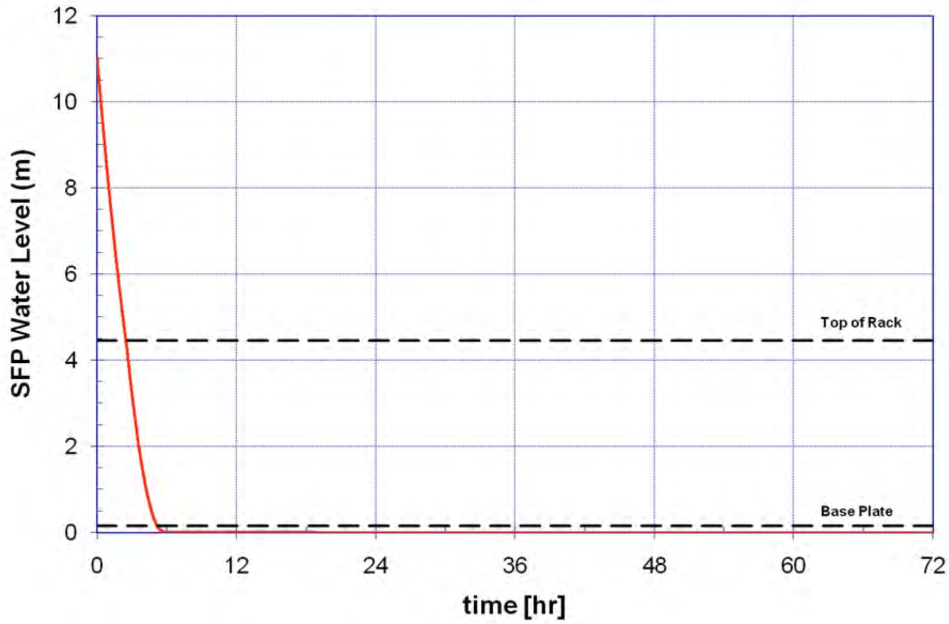
Notes:
(a) Photograph and information in these notes are from: Brick, 2010.
(b) A major tenant of the building was the Internal Revenue Service (IRS).
(c) The aircraft was a single-engine, fixed-wing Piper flown by its owner, Andrew Joseph Stack III, an Austin resident who worked as a computer engineer.
(d) A statement left by Mr Stack indicated that a dispute with IRS had brought him to a point of suicidal rage.

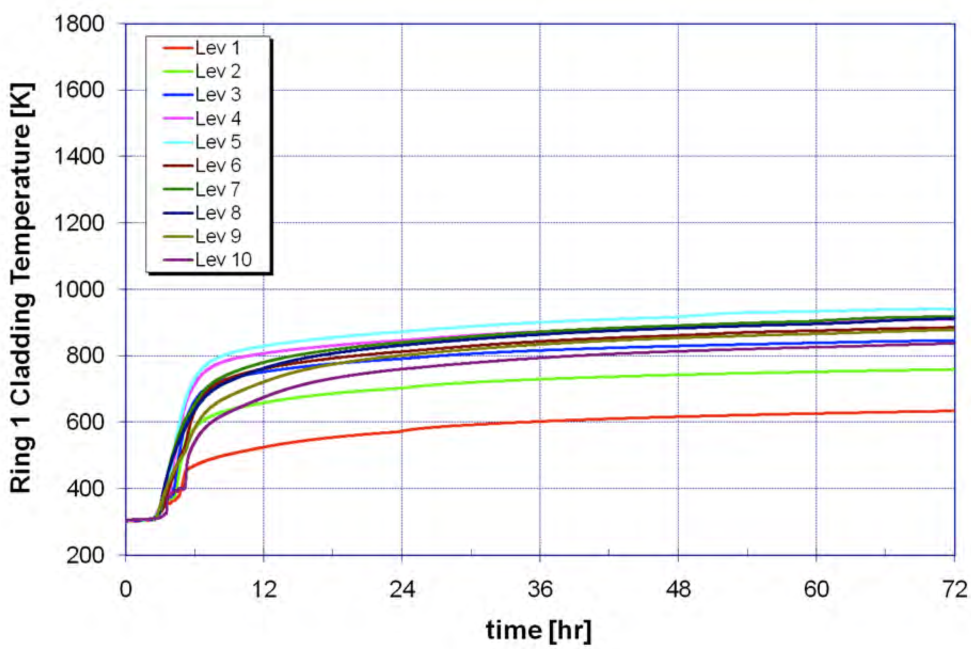
Figure VII-1
One of the Pool Layouts Modeled in NRC's Draft Consequence Study: The OCP2, High-Density, 1x4 Case



Notes:
(a) This figure is a copy of Figure 46 from: Barto et al, 2013.
(b) OCP2 (operating cycle phase 2) is described in Table 25 of: Barto et al, 2013.

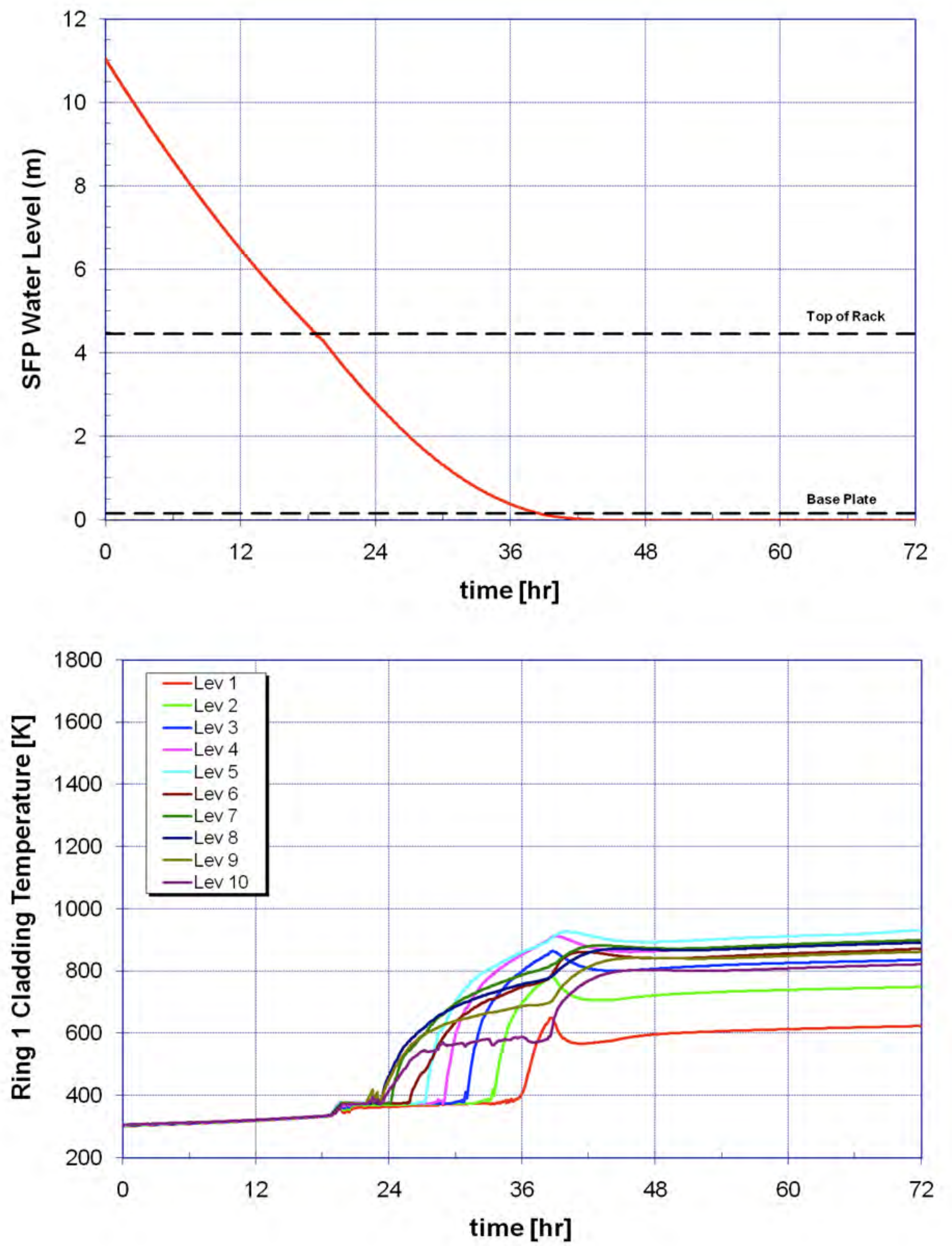
Figure VII-2
Findings from NRC's Draft Consequence Study: Water Level and Ring 1 Cladding Temperature for Unmitigated High-Density Moderate Leak (OCP4)





Notes:
a) These figures are copies of Figures 52 and 53 from: Barto et al, 2013.
(b) OCP4 (operating cycle phase 4) is described in Table 25 of: Barto et al, 2013.
(c) Vertical nodalization (Lev 1, etc.) is shown in Figure 41 of: Barto et al, 2013.
(d) Distribution of fuel (Ring 1, etc.) is shown in Figure 46 of: Barto et al, 2013.

Figure VII-3
Findings from NRC's Draft Consequence Study: Water Level and Ring 1 Cladding Temperature for Unmitigated High-Density Small Leak (OCP4)



Notes:
(a) These figures are copies of Figures 54 and 55 from: Barto et al, 2013.
(b) OCP4 (operating cycle phase 4) is described in Table 25 of: Barto et al, 2013.
(c) Vertical nodalization (Lev 1, etc.) is shown in Figure 41 of: Barto et al, 2013.
(d) Distribution of fuel (Ring 1, etc.) is shown in Figure 46 of: Barto et al, 2013.

APPENDIX B: Bibliography

(Ahearne et al, 2012)
John F. Ahearne and eight other authors, with editing by Charles D. Ferguson and Frank A. Settle, *The Future of Nuclear Power in the United States* (Washington, DC: Federation of American Scientists, and Washington and Lee University, February 2012).

(Albrecht, 1979)
Ernst Albrecht, Minister-President of Lower Saxony, "Declaration of the state government of Lower Saxony concerning the proposed nuclear fuel center at Gorleben" (English translation), May 1979.

(Alvarez et al, 2003)
Robert Alvarez, Jan Beyea, Klaus Janberg, Jungmin Kang, Ed Lyman, Allison Macfarlane, Gordon Thompson, and Frank von Hippel, "Reducing the Hazards from Stored Spent Power-Reactor Fuel in the United States", *Science and Global Security*, Volume 11, 2003, pp 1-51.

(Army, 1967)
Department of the Army, *Explosives and Demolitions, FM 5-25* (Washington, DC: Department of the Army, May 1967).

(Barto et al, 2013)
Andrew Barto and nine other authors, *Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a US Mark I Boiling Water Reactor, Draft Report* (Washington, DC: US Nuclear Regulatory Commission, June 2013).

(Benjamin et al, 1979)
Allan S. Benjamin and three other authors, *Spent Fuel Heatup Following Loss of Water During Storage, NUREG/CR-0649* (Washington, DC: US Nuclear Regulatory Commission, March 1979).

(Beyea et al, 1979)
Jan Beyea, Yves Lenoir, Gene Rochlin, and Gordon Thompson (group chair), "Potential Accidents and Their Effects," Chapter 3 in *Report of the Gorleben International Review*, March 1979. (This chapter was prepared in English and translated into German for submission to the Lower Saxony State Government.)

(Brick, 2010)
Michael Brick, "Man Crashes Plane Into Texas IRS Office", *The New York Times*, 18 February 2010.

(Collins and Hubbard, 2001)
T. E. Collins and G. Hubbard, *Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants, NUREG-1738* (Washington, DC: US Nuclear Regulatory Commission, February 2001). (This document was first released in October 2000.)

(GAO, 2012)
US Government Accountability Office, *Spent Nuclear Fuel: Accumulating Quantities at Commercial Reactors Present Storage and Other Challenges, GAO-12-797* (Washington, DC: GAO, August 2012).

(IAEA, 1997)
International Atomic Energy Agency, *Thermophysical properties of materials for water cooled reactors, IAEA-TECDOC-949* (Vienna: IAEA, June 1997).

(National Research Council, 2006)
National Research Council Committee on the Safety and Security of Commercial Spent Nuclear Fuel Storage (a committee of the Council's Board on Radioactive Waste Management), *Safety and Security of Commercial Spent Nuclear Fuel Storage: Public Report* (Washington, DC: National Academies Press, 2006).

(NRC, 2013)
US Nuclear Regulatory Commission, *Waste Confidence Generic Environmental Impact Statement, NUREG-2157, Draft Report for Comment* (Washington, DC: US Nuclear Regulatory Commission, August 2013).

(NRC, 1990)
US Nuclear Regulatory Commission, *Severe Accident Risks: An Assessment for Five US Nuclear Power Plants, NUREG-1150* (Washington, DC: US Nuclear Regulatory Commission, December 1990).

(NRC, 1979)
US Nuclear Regulatory Commission, *Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel, NUREG-0575* (Washington, DC: Nuclear Regulatory Commission, August 1979).

(Parry et al, 2000)
ASLBP No. 99-762-02-LA, "Affidavit of Gareth W. Parry, Stephen F. LaVie, Robert L. Palla and Christopher Gratton in Support of NRC Staff Brief and Summary of Relevant Facts, Data and Arguments upon which the Staff Proposes to Rely at Oral Argument on Environmental Contention EC-6", 20 November 2000.

(Popov et al, 2000)
S. G. Popov and three other authors, *Thermophysical Properties of MOX and UO_2 Fuels Including the Effects of Irradiation, Report ORNL/TM-2000/351* (Oak Ridge, Tennessee: Oak Ridge National Laboratory, November 2000).

(Raytheon, 2008)
Raytheon Company, "Raytheon Unveils New Record-Breaking Bunker Busting Technology", 12 March 2008, accessed on 7 March 2012 at: http://www.raytheon.com/newsroom/feature/bb_03-10/

(Roddy et al, 1986)
J. W. Roddy and four other authors, *Physical and Decay Characteristics of Commercial LWR Spent Fuel, ORNL/TM-9591/V1&R1* (Oak Ridge, Tennessee: Oak Ridge National Laboratory, January 1986).

(Thompson, 2013)
Gordon R. Thompson, Declaration, "Recommendations for the US Nuclear Regulatory Commission's Consideration of Environmental Impacts of Long-Term, Temporary Storage of Spent Nuclear Fuel or Related High-Level Waste", 2 January 2013.

(Thompson, 2009)
Gordon R. Thompson, *Environmental Impacts of Storing Spent Nuclear Fuel and High-Level Waste from Commercial Nuclear Reactors: A Critique of NRC's Waste Confidence Decision and Environmental Impact Determination* (Cambridge, Massachusetts: Institute for Resource and Security Studies, 6 February 2009).

(Thompson, 2008)
Gordon R. Thompson, "The US Effort to Dispose of High-Level Radioactive Waste", *Energy and Environment*, Volume 19, Nos. 3+4, 2008, pp 391-412.

(Thompson, 2007)
Gordon R. Thompson, *Risk-Related Impacts from Continued Operation of the Indian Point Nuclear Power Plants* (Cambridge, Massachusetts: Institute for Resource and Security Studies, 28 November 2007).

(Thompson, 2005)
Gordon R. Thompson, *Reasonably Foreseeable Security Events: Potential threats to options for long-term management of UK radioactive waste* (Cambridge, Massachusetts: Institute for Resource and Security Studies, 2 November 2005).

(Thompson, 2003)
Gordon Thompson, *Robust Storage of Spent Nuclear Fuel: A Neglected Issue of Homeland Security* (Cambridge, Massachusetts: Institute for Resource and Security Studies, January 2003).

(Wald, 2005)
Matthew L. Wald, "Agencies Fight Over Report on Sensitive Atomic Wastes", *The New York Times*, 30 March 2005.

(Warwick, 2008)
Graham Warwick, "VIDEO: Raytheon tests bunker-busting tandem warhead", *Flightglobal*, 26 February 2008, accessed on 7 March 2012 from: http://www.flightglobal.com/news/articles/video-raytheon-tests-bunker-busting-tandem-warhead-221842/

April 13, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: DRAFT FINAL TECHNICAL STUDY OF SPENT FUEL POOL ACCIDENT RISK AT DECOMMISSIONING NUCLEAR POWER PLANTS

During the 471st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2000, we met with representatives of the NRC staff and discussed the subject document. We also had the benefit of the documents referenced, which include the available stakeholders comments. This report is in response to the Commission's request in the Staff Requirements Memorandum dated December 21, 1999, that the ACRS perform a technical review of the validity of the draft study and risk objectives.

Background

Decommissioning plants are subject to many of the same regulatory requirements as operating nuclear plants. Because of the expectation that the risk will be lower at decommissioning plants, particularly as time progresses to allow additional decay of fission products, some of these requirements may be inappropriate. Exemptions from the regulations are frequently requested by licensees after a nuclear power plant is permanently shut down. To increase the efficiency and effectiveness of decommissioning regulations, the staff has engaged in rulemaking activities that would reduce the need to routinely process exemptions. The staff has undertaken the technical study and risk analysis discussed here to provide a firm technical basis for rulemaking concerning several exemption issues.

In the draft study the staff has concluded that, provided certain industry decommissioning commitments are implemented at the plants, after one year of decay time the risk associated with spent fuel pool fires is sufficiently low that emergency planning requirements can be significantly reduced. It also concluded that after five years the risk of zirconium fires is negligible even if the fuel is uncovered and that requirements intended to ensure spent fuel cooling can be reduced.

Recommendations

1. The integrated rulemaking on decommissioning should be put on hold until the staff provides technical justification for the proposed acceptance criterion for fuel uncovery frequency. In particular, the staff needs to incorporate the effects of enhanced release of ruthenium under air-oxidation conditions and the impact of the

MELCOR Accident Consequence Code System (MACCS) code assumptions on plume-related parameters in view of the results of expert elicitation.
2. The technical basis underlying the zirconium-air interactions and the criteria for ignition needs to be strengthened. In particular, the potential impact of zirconium-hydrides in high burnup fuel and the susceptibility of the clad to breakaway oxidation need to be addressed.
3. Uncertainties in the risk assessment need to be quantified and made part of the decisionmaking process.

Discussion

The staff's conclusion that the risk after one year of decay time is sufficiently low that emergency planning requirements can be reduced is based partially on the assessed value of fuel uncovery frequency (3.4 x 10^{-6} /yr) being less than the Regulatory Guide 1.174 large, early release frequency (LERF) acceptance value ($1x10^{-5}$ /yr). This LERF risk-acceptance value was derived to be a surrogate for the Safety Goal early fatality quantitative health objectives (QHO) *for operating reactors*. The derivation from the QHO is based, however, on the fission product releases that occur under severe accident conditions which are driven by steam oxidation of the zircaloy and the fuel. These releases include only insignificant amounts of ruthenium. Under air-oxidation conditions of spent fuel fires, significant data indicate much enhanced releases of ruthenium as the very volatile oxide. Indications are that, under air oxidation conditions, the release fractions of ruthenium may be equivalent to those for iodine and cesium. In the accident at Chernobyl significant releases of ruthenium were observed and attributed to the interactions of fuel with air.

These findings have significant implications. The ruthenium inventory in spent fuel is substantial. Ruthenium has a biological effectiveness equivalent to that of Iodine-131 and has a relatively long half-life. If there are significant releases of ruthenium, the Regulatory Guide 1.174 LERF value may not be an appropriate surrogate for the prompt fatality QHO. In addition, because of the relatively long half-life of ruthenium-106, it is likely that the early fatality QHO would no longer be the controlling consequence.

In response to our concerns about the effects of substantial ruthenium release, the staff has made additional MACCS calculations in which it assumed 100 percent release of the ruthenium inventory. For a one-year decay time with no evacuation, the prompt fatalities increased by two orders of magnitude over those in the report which did not include ruthenium release, the societal dose doubled and the cancer fatalities increased four-fold.

Our concern is not just with ruthenium. We are concerned with the appropriateness of the entire source term used in the study. There is a known tendency for uranium dioxide in air to decrepitate into fine particles. The decrepitation is caused by lattice strains produced as the dioxide reacts to form U_3O_8. This decrepitation is a bane of thermogravimetric studies of air oxidation of uranium dioxide since it can cause fine particles to be entrained in the flowing air of the apparatus. This suggests that decrepitating fuel would be readily entrained in vigorous natural convection flows produced in an accident at a spent fuel pool. The decrepitation process provides a low-temperature, mechanical, release mechanism for even very refractory radionuclides. The staff did consider the possibility that "fuel fines" could be released from fuel with ruptured cladding. It did not, however, believe these fuel fines could escape the plant site. Nevertheless, the staff considered the effect of a $6x10^{-6}$ release fraction of fines. This minuscule release fraction did not significantly affect the calculated findings. There is no reason to think that such a low release fraction would be encountered with decrepitating fuel.

Consequences of accidents involving a spent fuel pool were analyzed using the MACCS code. The staff has completed an expert opinion elicitation regarding the uncertainties associated with many of the critical features of the MACCS code. The findings of this elicitation seem not to have been considered in the analyses of the spent fuel pool accident. One of the uncertainties in MACCS identified by the experts is associated with the spread of the radioactive plume from a power plant site. The spread expected by the experts is much larger than what is taken as the default spread in the MACCS calculations. There is no indication that the staff took this finding into account in preparing the consequence analyses. In addition, the initial plume energy assumed in the MACCS calculations, which determines the extent of plume rise, was taken to be the same as that of a reactor accident rather than one appropriate for a zirconium fire. We suspect, therefore, that the consequences found by the staff tend to overestimate prompt fatalities and underestimate land contamination and latent fatalities just because of the narrow plume used in the MACCS calculations and the assumed default plume energy.

The staff needs to review the air oxidation fission products release data from Oak Ridge National Laboratory and from Canada that found large releases of cesium, tellurium, and ruthenium at temperatures lower than 1000^{o}C. Based on these release values for ruthenium, and incorporating uncertainties in the MACCS plume dispersal models, the consequence analyses should be redone.

Based on the results of this reevaluation of the consequences, the staff should determine an appropriate LERF for spent fuel fires that properly reflects the prompt fatality QHO and the potential for land contamination and latent fatalities associated with spent fuel pool fires.

In developing risk-acceptance criteria associated with spent fuel fires, the staff should also keep in mind such factors as the relatively small number of decommissioning plants to be expected at any given time and the short time at which they are vulnerable to a spent fuel pool fire.

We also have difficulties with the analysis performed to determine the time at which the risk of zirconium fires becomes negligible. In previous interactions with the staff on this study, we indicated that there were issues associated with the formation of zirconium-hydride precipitates in the cladding of fuel especially when that fuel has been taken to high burnups. Many metal hydrides are spontaneously combustible in air. Spontaneous combustion of zirconium-hydrides would render moot the issue of "ignition" temperature that is the focus of the staff analysis of air interactions with exposed cladding. The staff has neglected the issue of hydrides and suggested that uncertainties in the critical decay heat times and the critical temperatures can be found by sensitivity analyses. Sensitivity analyses with models lacking essential physics and chemistry would be of little use in determining the real uncertainties.

The staff analysis of the interaction of air with cladding has relied on relatively geriatric work. Much more is known now about air interactions with cladding. This greater knowledge has come in no small part from studies being performed as part of a cooperative international program (PHEBUS FP) in which NRC is a partner. Among the findings of this work is that nitrogen from air depleted of oxygen will interact exothermically with zircaloy cladding. The reaction of zirconium with nitrogen is exothermic by about 86,000 calories per mole of zirconium reacted. Because the heat required to raise zirconium from room temperature to melting is only about 18,000 calories per mole, the reaction enthalpy with nitrogen is ample. In air-starved conditions, the reaction of air with zirconium produces a duplex film in which the outer layer is zirconium dioxide (ZrO_2) and the inner layer is the crystallographically different compound zirconium nitride (ZrN). The microscopic strains within this duplex layer can lead to exfoliation of the protective oxide layer and reaction rates that deviate from parabolic rates. These findings may well explain the well-known tendency for zirconium to undergo breakaway oxidation in air whereas no such tendency is encountered in either steam or in pure oxygen. Because of these findings, we do not accept the staff's claim that it has performed "bounding" calculations of the heatup of Zircaloy clad fuel even when it neglects heat losses.

The staff focuses its analysis of the reactions of gases with fuel cladding on a quantity they call an "ignition temperature." The claim is that this is the temperature of self-sustained reaction of gas with the clad. Gases will react with the cladding at all temperatures. In fact, at temperatures well below the "conservative ignition temperature" identified by the staff, air and oxygen will react with the cladding quite smoothly and at rates sufficient to measure. Data in these temperature ranges well below the "ignition" temperature form much of the basis for the correlations of parabolic reaction rates with temperature. We believe that the staff should look for a condition such that the increase with temperature of the heat liberation rate by the reaction of gas with the clad exceeds the increase with temperature of the rate of heat losses by radiation and convection. Finding this condition requires that there be high quality analyses of the heat losses and that the heat of reaction be properly calculated. Since staff has neglected any reaction with nitrogen and did not consider breakaway oxidation (causes for the deviations from parabolic reaction rates), it has not made an appropriate analysis to find this "ignition temperature."

In fact, the search for the ignition temperature may be the wrong criterion for the analysis. The staff should also be looking for the point at which cladding ruptures and fission products can be released. Some fraction of the cladding may be ruptured before any exposure of the fuel to air occurs. Even discounting this, one still arrives at much lower temperature criteria for concern over the possible release of radionuclides.

There are other flaws in the material interactions analyses performed as part of the study. For instance, in examining the effects of aluminum melting, the staff seems to not recognize that there is a very exothermic intermetallic reaction between molten aluminum and stainless steel. Compound formation in the Al-Zr system suggests a strong intermetallic reaction of molten aluminum with fuel cladding as well. The staff focuses on eutectic formations when, in fact, intermetallic reactions are more germane to the issues at hand.

We are concerned about the conservative treatment of seismic issues. Risk-informed decisionmaking regarding the spent fuel pool fire issues should use realistic analysis, including an uncertainty assessment.

Because the accident analysis is dominated by sequences involving human errors and seismic events which involve large uncertainties, the absence of an uncertainty analysis of the frequencies of accidents is unacceptable. The study is inadequate until there is a defensible uncertainty analysis.

The risk posed by fuel uncovery in spent fuel pools for decommissioning plants may indeed be low, however, the technical shortcomings of this study are significant and sufficient for us to recommend that rulemaking be put on hold until the inadequacies discussed herein are addressed by the staff.

Sincerely
/RA/
Dana A. Powers
Chairman

References:

1. Draft For Comment, Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," February 2000.
2. SECY-99-168, dated June 30, 1999, memorandum from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Improving Decommissioning Regulations For Nuclear Power Plants.
3. Memorandum dated December 21, 1999, from Anette L. Vietti-Cook, Secretary of the Commission, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements -SECY-99-168 - Improving Decommissioning Regulations for Nuclear Power Plants.

4. Letter dated November 12, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Spent Fuel Fires Associated With Decommissioning.
5. Letter dated December 16, 1999, from William D. Travers, Executive Director for Operations, NRC, to Dana A. Powers, Chairman, ACRS, Subject: Spent Fuel Fires Associated With Decommissioning.
6. E-mail message dated April 5, 2000, from Alan Nelson, Nuclear Energy Institute, to M. El-Zeftawy, ACRS, transmitting NEI comments on Appendix 2.b, "Structural Integrity Seismic Loads."
7. U. S. Nuclear Regulatory Commission, NUREG/CR-6613, "Code Manual for MACCS2, May 1998.
8. U. S. Department of Commerce, "JANAF Thermochemical Tables," Second Edition, Issued June 1971.
9. U. S. Nuclear Regulatory Commission, NUREG/CP-0149, Vol. 2 "Twenty-Third Water Reactor Safety Information Meeting," October 23-25, 1995, "The Severe Accident Research Programme PHEBUS FP.: First Results and Future Tests," published March 1996.
10. U. S. Nuclear Regulatory Commission, NUREG/CR-6244, Vol. 1, "Probabilistic Accident Consequence Uncertainty Analysis," Dispersion and Deposition Uncertainty Assessment, published January 1995.
11. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.

Page Last Reviewed/Updated Thursday, March 29, 2012

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October 2, 2013

Dr. J. Sam Armijo, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: *ACRS Review of Transfer of Spent Fuel to Dry Cask Storage*

Dear Dr. Armijo:

On behalf of 26 environmental organizations across the United States[1], I am writing to request that you re-open the Advisory Committee on Reactor Safeguards (ACRS) review of whether the Nuclear Regulatory Commission (NRC) should require licensees to transfer spent fuel from high-density storage pools to combined dry storage and pool storage in open-frame, low-density racks. To date, the process used by the ACRS to evaluate technical information regarding the issue of accident risks posed by high-density pool storage of spent fuel has excluded meaningful input from any party other than the NRC Staff, and has failed to consider serious criticisms by outside experts of the NRC Staff's own technical work. Given the enormous safety significance and technical complexity of the issue, the ACRS' failure to consider views outside the NRC is unacceptable.

For instance, on July 9, 2013, the ACRS held a meeting on the NRC Staff's Draft Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a US Mark I Boiling Water Reactor (June 2013) ("Draft Consequence Study"), which forms the principal basis for the NRC Staff's recommendation that expedited transfer of spent fuel from high-density storage pools should not be required because it is not warranted on safety grounds. At that time, the public comment period regarding the Draft Consequence Study had just begun, and no outside expert had been provided with sufficient time to examine the Draft Consequence Study and present an analysis of it to the ACRS. Without even waiting until the August 1 deadline for public comment on the Draft Consequence Study had passed, the ACRS issued a favorable review on July 18, 2013.

Thus, the ACRS' favorable review of the Draft Consequence Study does not reflect any consideration of serious criticisms submitted by non-NRC experts, including Dr. Gordon Thompson of the Institute for Resource and Security Studies and David Lochbaum of the Union

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[1] With Mindy Goldstein of the Turner Environmental Law Clinic, I am counsel in this matter to: Beyond Nuclear, Blue Ridge Environmental Defense League, Center for a Sustainable Coast, Citizens Allied for Safe Energy, Don't Waste Michigan, Ecology Party of Florida, Friends of the Coast, Friends of the Earth, Georgia Women's Action for New Directions, Green States Solutions, Hudson River Sloop Clearwater, Missouri Coalition for the Environment, NC WARN, Nevada Nuclear Waste Task Force, New England Coalition, No Nukes Pennsylvania, Nuclear Energy Information Service, Nuclear Information and Resource Service, Nuclear Watch South, Physicians for Social Responsibility, Public Citizen, Riverkeeper, SEED Coalition, San Luis Obispo Mothers for Peace, Sierra Club Nuclear Free Campaign, and Southern Alliance for Clean Energy.

of Concerned Scientists.[2] In detailed technical comments, these experts argued that the Draft Consequence Study is biased and incomplete.

On September 19, 2013, the ACRS Subcommittee on Materials, Metallurgy, and Reactor Fuels held a meeting that included the topic of the NRC Staff's recommendation against expedited transfer of spent fuel from high-density storage pools. Although the meeting was noticed as open to the public, 78 Fed. Reg. 56,756 (Sept. 13, 2013), the ACRS subsequently decided to close it. Therefore, no members of the public could attend.

The full ACRS is scheduled to meet today, October 2, to once again discuss the NRC Staff's recommendation against expedited transfer of spent fuel from high-density fuel storage pools. And once again, meaningful participation by non-NRC experts is prohibited. In a letter dated September 17, 2013, I requested an opportunity for Dr. Thompson to make a detailed presentation to the ACRS regarding his criticisms of the NRC Staff's technical analysis. Dr. Thompson also sought an opportunity to address the ways in which concerns about spent fuel storage accident risks, expressed by the ACRS' former Chairman Dana Powers in 2000, remain unresolved.[3] I requested that Dr. Thompson be given a full ten-to-fifteen minutes for his presentation, and that the time be dedicated to his presentation rather than squeezed into the public comment period. Mr. Brown of your staff suggested that I get the agreement of the NRC Staff to my request. Therefore I sought agreement from Mr. Kevin Witt, who agreed to it. Mr. Hackett of your staff subsequently agreed to give Dr. Thompson 15 minutes to present his views.

In a telephone conversation with me yesterday, however, you stated that you would not assure Dr. Thompson of a 15-minute opportunity to address the ACRS, and that he would have to share the 15-minute comment period allocated to all members of the public. In addition, you stated that the merits of the Draft Consequence Study were no longer open to any significant discussion before the ACRS. In light of your retraction of your staff's previous commitment, and in light of the fact that the ACRS did not intend to entertain any detailed discussion of the NRC Staff's key technical study underlying its recommendation, Dr. Thompson decided not to travel from his office in Boston to the meeting today.

My clients are extremely concerned that in light of the serious deficiencies in the Draft Consequence Study, it should not be relied upon for any regulatory decisions regarding

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[2] Declaration of 1 August 2013 by Gordon R. Thompson: Comments on the U.S. Nuclear Regulatory Commission's Draft Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a US Mark I Boiling Water Reactor (August 2, 2013) ("Thompson Comments"); letter from David Lochbaum, UCS to Cindy Bladey, NRC re: Draft Report titled Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a US Mark I Boiling Water Reactor (July 18, 2013) ("Lochbaum Comments").

[3] *See* letter from Dana Powers, ACRS, to Richard A. Meserve, re: Draft Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (April 13, 2000) (copy attached).

management of spent fuel. In addition, they are concerned that other studies, on which the NRC Staff claims to rely for its recommendation against expedited transfer of spent fuel storage, are grossly inadequate to support such a recommendation. We think that the ACRS also should be deeply troubled by these criticisms, because the issue of spent fuel storage risks is one of the most important unaddressed safety and environmental issues facing the NRC today, affecting every single operating reactor in the United States.

It is therefore of paramount importance that before making any recommendation to the NRC Commissioners regarding the issue, the ACRS should conduct a thorough review of non-NRC technical criticisms of the Draft Consequence Study and any other technical studies on which the NRC Staff proposes to rely for its recommendation against ordering expedited transfer of spent fuel out of high-density pool storage. Accordingly, we request that you defer making any recommendation regarding the issue of expedited transfer of spent fuel out of high-density storage, until you have taken the following measures:

- Re-open the ACRS review of the Draft Consequence Study, which was closed by the ACRS' letter of July 18, 2013.

- Hold a subcommittee meeting, attended by relevant ACRS experts, including Dr. Powers. At that meeting, Dr. Thompson, Mr. Lochbaum, and other non-NRC technical experts should be given an opportunity to thoroughly present their views regarding the merits of the technical studies on which the NRC Staff relies for its recommendation regarding expedited transfer of spent fuel.

- Hold a full committee meeting at which the committee members hear presentations by the NRC Staff, non-NRC experts, and members of the subcommittee.

The ACRS meetings should not be held until a reasonable period of time after the NRC has published the transcript of a public meeting held by the NRC Staff on September 18, 2013, at which the Staff responded to questions regarding the Draft Consequence Study and other studies on which it relied. That meeting yielded important information: for instance, the Staff's admission that it does not consider the Draft Consequence Study to be a bounding analysis of spent fuel pool risks.

I look forward to hearing from you regarding our request.

Sincerely,

/s/
Diane Curran

Cc: Allison Macfarlane, NRC Chairman
Christopher Brown, ACRS Staff
Edwin Hackett, ACRS Staff
Kevin Witt, NRC Staff

April 13, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: DRAFT FINAL TECHNICAL STUDY OF SPENT FUEL POOL ACCIDENT RISK AT DECOMMISSIONING NUCLEAR POWER PLANTS

During the 471st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2000, we met with representatives of the NRC staff and discussed the subject document. We also had the benefit of the documents referenced, which include the available stakeholders comments. This report is in response to the Commission's request in the Staff Requirements Memorandum dated December 21, 1999, that the ACRS perform a technical review of the validity of the draft study and risk objectives.

Background

Decommissioning plants are subject to many of the same regulatory requirements as operating nuclear plants. Because of the expectation that the risk will be lower at decommissioning plants, particularly as time progresses to allow additional decay of fission products, some of these requirements may be inappropriate. Exemptions from the regulations are frequently requested by licensees after a nuclear power plant is permanently shut down. To increase the efficiency and effectiveness of decommissioning regulations, the staff has engaged in rulemaking activities that would reduce the need to routinely process exemptions. The staff has undertaken the technical study and risk analysis discussed here to provide a firm technical basis for rulemaking concerning several exemption issues.

In the draft study the staff has concluded that, provided certain industry decommissioning commitments are implemented at the plants, after one year of decay time the risk associated with spent fuel pool fires is sufficiently low that emergency planning requirements can be significantly reduced. It also concluded that after five years the risk of zirconium fires is negligible even if the fuel is uncovered and that requirements intended to ensure spent fuel cooling can be reduced.

Recommendations

1. The integrated rulemaking on decommissioning should be put on hold until the staff provides technical justification for the proposed acceptance criterion for fuel uncovery frequency. In particular, the staff needs to incorporate the effects of enhanced release of ruthenium under air-oxidation conditions and the impact of the

MELCOR Accident Consequence Code System (MACCS) code assumptions on plume-related parameters in view of the results of expert elicitation.

2. The technical basis underlying the zirconium-air interactions and the criteria for ignition needs to be strengthened. In particular, the potential impact of zirconium-hydrides in high burnup fuel and the susceptibility of the clad to breakaway oxidation need to be addressed.
3. Uncertainties in the risk assessment need to be quantified and made part of the decisionmaking process.

Discussion

The staff's conclusion that the risk after one year of decay time is sufficiently low that emergency planning requirements can be reduced is based partially on the assessed value of fuel uncovery frequency (3.4×10^{-6} /yr) being less than the Regulatory Guide 1.174 large, early release frequency (LERF) acceptance value (1×10^{-5} /yr). This LERF risk-acceptance value was derived to be a surrogate for the Safety Goal early fatality quantitative health objectives (QHO) *for operating reactors*. The derivation from the QHO is based, however, on the fission product releases that occur under severe accident conditions which are driven by steam oxidation of the zircaloy and the fuel. These releases include only insignificant amounts of ruthenium. Under air-oxidation conditions of spent fuel fires, significant data indicate much enhanced releases of ruthenium as the very volatile oxide. Indications are that, under air oxidation conditions, the release fractions of ruthenium may be equivalent to those for iodine and cesium. In the accident at Chernobyl significant releases of ruthenium were observed and attributed to the interactions of fuel with air.

These findings have significant implications. The ruthenium inventory in spent fuel is substantial. Ruthenium has a biological effectiveness equivalent to that of Iodine-131 and has a relatively long half-life. If there are significant releases of ruthenium, the Regulatory Guide 1.174 LERF value may not be an appropriate surrogate for the prompt fatality QHO. In addition, because of the relatively long half-life of ruthenium-106, it is likely that the early fatality QHO would no longer be the controlling consequence.

In response to our concerns about the effects of substantial ruthenium release, the staff has made additional MACCS calculations in which it assumed 100 percent release of the ruthenium inventory. For a one-year decay time with no evacuation, the prompt fatalities increased by two orders of magnitude over those in the report which did not include ruthenium release, the societal dose doubled and the cancer fatalities increased four-fold.

Our concern is not just with ruthenium. We are concerned with the appropriateness of the entire source term used in the study. There is a known tendency for uranium dioxide in air to decrepitate into fine particles. The decrepitation is caused by lattice strains produced as the dioxide reacts to form U_3O_8. This decrepitation is a bane of thermogravimetric studies of air oxidation of uranium dioxide since it can cause fine particles to be entrained in the flowing air of the apparatus. This suggests that decrepitating fuel would be readily entrained in vigorous natural convection flows produced in an accident at a spent fuel pool. The decrepitation process provides a low-temperature, mechanical, release mechanism for even very refractory radionuclides. The staff did consider the possibility that "fuel fines" could be released from fuel with ruptured cladding. It did not, however, believe these fuel fines could escape the plant site. Nevertheless, the staff considered the effect of a 6×10^{-6} release fraction of fines. This minuscule release fraction did not significantly affect the calculated findings. There is no reason to think that such a low release fraction would be encountered with decrepitating fuel.

Consequences of accidents involving a spent fuel pool were analyzed using the MACCS code. The staff has completed an expert opinion elicitation regarding the uncertainties associated with many of the critical features of the MACCS code. The findings of this elicitation seem not to have been considered in the analyses of the spent fuel pool accident. One of the uncertainties in MACCS identified by the experts is associated with the spread of the radioactive plume from a power plant site. The spread expected by the experts is much larger than what is taken as the default spread in the MACCS calculations. There is no indication that the staff took this finding into account in preparing the consequence analyses. In addition, the initial plume energy assumed in the MACCS calculations, which determines the extent of plume rise, was taken to be the same as that of a reactor accident rather than one appropriate for a zirconium fire. We suspect, therefore, that the consequences found by the staff tend to overestimate prompt fatalities and underestimate land contamination and latent fatalities just because of the narrow plume used in the MACCS calculations and the assumed default plume energy.

The staff needs to review the air oxidation fission products release data from Oak Ridge National Laboratory and from Canada that found large releases of cesium, tellurium, and ruthenium at temperatures lower than 1000^{o}C. Based on these release values for ruthenium, and incorporating uncertainties in the MACCS plume dispersal models, the consequence analyses should be redone.

Based on the results of this reevaluation of the consequences, the staff should determine an appropriate LERF for spent fuel fires that properly reflects the prompt fatality QHO and the potential for land contamination and latent fatalities associated with spent fuel pool fires.

In developing risk-acceptance criteria associated with spent fuel fires, the staff should also keep in mind such factors as the relatively small number of decommissioning plants to be expected at any given time and the short time at which they are vulnerable to a spent fuel pool fire.

We also have difficulties with the analysis performed to determine the time at which the risk of zirconium fires becomes negligible. In previous interactions with the staff on this study, we indicated that there were issues associated with the formation of zirconium-hydride precipitates in the cladding of fuel especially when that fuel has been taken to high burnups. Many metal hydrides are spontaneously combustible in air. Spontaneous combustion of zirconium-hydrides would render moot the issue of "ignition" temperature that is the focus of the staff analysis of air interactions with exposed cladding. The staff has neglected the issue of hydrides and suggested that uncertainties in the critical decay heat times and the critical temperatures can be found by sensitivity analyses. Sensitivity analyses with models lacking essential physics and chemistry would be of little use in determining the real uncertainties.

The staff analysis of the interaction of air with cladding has relied on relatively geriatric work. Much more is known now about air interactions with cladding. This greater knowledge has come in no small part from studies being performed as part of a cooperative international program (PHEBUS FP) in which NRC is a partner. Among the findings of this work is that nitrogen from air depleted of oxygen will interact exothermically with zircaloy cladding. The reaction of zirconium with nitrogen is exothermic by about 86,000 calories per mole of zirconium reacted. Because the heat required to raise zirconium from room temperature to melting is only about 18,000 calories per mole, the reaction enthalpy with nitrogen is ample. In air-starved conditions, the reaction of air with zirconium produces a duplex film in which the outer layer is zirconium dioxide (ZrO_2) and the inner layer is the crystallographically different compound zirconium nitride (ZrN). The microscopic strains within this duplex layer can lead to exfoliation of the protective oxide layer and reaction rates that deviate from parabolic rates. These findings may well explain the well-known tendency for zirconium to undergo breakaway oxidation in air whereas no such tendency is encountered in either steam or in pure oxygen. Because of these findings, we do not accept the staff's claim that it has performed "bounding" calculations of the heatup of Zircaloy clad fuel even when it neglects heat losses.

The staff focuses its analysis of the reactions of gases with fuel cladding on a quantity they call an "ignition temperature." The claim is that this is the temperature of self-sustained reaction of gas with the clad. Gases will react with the cladding at all temperatures. In fact, at temperatures well below the "conservative ignition temperature" identified by the staff, air and oxygen will react with the cladding quite smoothly and at rates sufficient to measure. Data in these temperature ranges well below the "ignition" temperature form much of the basis for the correlations of parabolic reaction rates with temperature. We believe that the staff should look for a condition such that the increase with temperature of the heat liberation rate by the reaction of gas with the clad exceeds the increase with temperature of the rate of heat losses by radiation and convection. Finding this condition requires that there be high quality analyses of the heat losses and that the heat of reaction be properly calculated. Since staff has neglected any reaction with nitrogen and did not consider breakaway oxidation (causes for the deviations from parabolic reaction rates), it has not made an appropriate analysis to find this "ignition temperature."

In fact, the search for the ignition temperature may be the wrong criterion for the analysis. The staff should also be looking for the point at which cladding ruptures and fission products can be released. Some fraction of the cladding may be ruptured before any exposure of the fuel to air occurs. Even discounting this, one still arrives at much lower temperature criteria for concern over the possible release of radionuclides.

There are other flaws in the material interactions analyses performed as part of the study. For instance, in examining the effects of aluminum melting, the staff seems to not recognize that there is a very exothermic intermetallic reaction between molten aluminum and stainless steel. Compound formation in the Al-Zr system suggests a strong intermetallic reaction of molten aluminum with fuel cladding as well. The staff focuses on eutectic formations when, in fact, intermetallic reactions are more germane to the issues at hand.

We are concerned about the conservative treatment of seismic issues. Risk-informed decisionmaking regarding the spent fuel pool fire issues should use realistic analysis, including an uncertainty assessment.

Because the accident analysis is dominated by sequences involving human errors and seismic events which involve large uncertainties, the absence of an uncertainty analysis of the frequencies of accidents is unacceptable. The study is inadequate until there is a defensible uncertainty analysis.

The risk posed by fuel uncovery in spent fuel pools for decommissioning plants may indeed be low, however, the technical shortcomings of this study are significant and sufficient for us to recommend that rulemaking be put on hold until the inadequacies discussed herein are addressed by the staff.

Sincerely
/RA/
Dana A. Powers
Chairman

References:

1. Draft For Comment, Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," February 2000.
2. SECY-99-168, dated June 30, 1999, memorandum from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Improving Decommissioning Regulations For Nuclear Power Plants.
3. Memorandum dated December 21, 1999, from Anette L. Vietti-Cook, Secretary of the Commission, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements -SECY-99-168 - Improving Decommissioning Regulations for Nuclear Power Plants.

4. Letter dated November 12, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Spent Fuel Fires Associated With Decommissioning.
5. Letter dated December 16, 1999, from William D. Travers, Executive Director for Operations, NRC, to Dana A. Powers, Chairman, ACRS, Subject: Spent Fuel Fires Associated With Decommissioning.
6. E-mail message dated April 5, 2000, from Alan Nelson, Nuclear Energy Institute, to M. El-Zeftawy, ACRS, transmitting NEI comments on Appendix 2.b, "Structural Integrity Seismic Loads."
7. U. S. Nuclear Regulatory Commission, NUREG/CR-6613, "Code Manual for MACCS2, May 1998.
8. U. S. Department of Commerce, "JANAF Thermochemical Tables," Second Edition, Issued June 1971.
9. U. S. Nuclear Regulatory Commission, NUREG/CP-0149, Vol. 2 "Twenty-Third Water Reactor Safety Information Meeting," October 23-25, 1995, "The Severe Accident Research Programme PHEBUS FP.: First Results and Future Tests," published March 1996.
10. U. S. Nuclear Regulatory Commission, NUREG/CR-6244, Vol. 1, "Probabilistic Accident Consequence Uncertainty Analysis," Dispersion and Deposition Uncertainty Assessment, published January 1995.
11. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.

Page Last Reviewed/Updated Thursday, March 29, 2012

Remarks by Donald Helton (staff member in the Office of Nuclear Regulatory Research) at the October 2nd, 2013 ACRS Full Committee Meeting on Expedited Fuel Movement:

- The Regulatory Analysis represents a significant amount of work accomplished in a relatively short timeframe, and the NRR staff should be commended for its breadth and complexity.
- These remarks are intended to provide additional emphasis on particular aspects of the Regulatory Analysis that may not resonate with the Committee and the Commission, as currently characterized in the draft Commission paper.
- They represent my views. They do not represent an Office of Nuclear Regulatory Research view.

1. The Regulatory Analysis shows that expedited movement of fuel older than 5 years from spent fuel pools to dry cask storage does not provide a substantial safety enhancement. It is important for the reader to understand that the significance of the safety enhancement has been judged based solely on the risk to individuals living in close proximity to a nuclear power plant. This means that risk to an individual is assumed to be a reasonable surrogate for cumulative human health risk, even though the events in question are known to have widespread effects in the unlikely event they occur.
2. The Regulatory Analysis shows that the studied action is not cost-beneficial when radiological release frequency estimates are biased in favor of a cost/beneficial finding, while total offsite impacts (human health and otherwise) are not comprehensively considered. Specifically, a dated dose conversion factor and a 50-mile distance truncation are employed. The Commission paper acknowledges this, and emphasizes the importance of the sensitivity studies, without informing the reader that:
    a. In many instances this is the difference between a cost/beneficial and non-cost/beneficial determination, and
    b. It makes an order-of-magnitude difference in some results.
3. The staff's work to date does not provide a clear perspective on the cost/beneficial result when both the conservatisms and non-conservatisms are removed. Based on my own investigation (which involved constructing a cumulative distribution function from the low, base, and high cases, using the beyond-50-mile/$4000 per person-rem sensitivities), I expect that the action would not be cost/beneficial for a majority of the fleet but could be cost/beneficial for many plants. Additional work to refine specific simplifying assumptions in the Regulatory Analysis (such as the effect of mitigation in reducing the release frequency), or to perform a simplified plant-by-plant screening based on available information, might alter this conclusion in a more non-cost/beneficial direction.
4. The Regulatory Analysis does not consider related alternatives (e.g., expedited movement of fuel older than ten years, refinement of spent fuel pool heat load management requirements) that might be more cost-beneficial.
5. Since, on the whole, there is no compelling evidence upon which to take generic regulatory action, I agree with the Commission paper's recommendation to close the Japan Lessons Learned Tier 3 item. However, in light of the points raised above, I believe that the staff should advocate for continued staff activity under another appropriate regulatory program to assess whether action would be cost-beneficial for specific plants when simplifying assumptions are refined, or when other contributing factors (such as inadvertent criticality) are considered. This would be in addition to resolving the issue for Western plants (as the Commission paper already envisions). This information would then be provided to the Commission.

6. I believe the staff should also seek Commission direction on the use of quantitative health objectives for an individual as a suitable measure of substantial safety enhancement for classes of accidents known to be low-likelihood, high consequence events, particularly when this determination causes the staff to dismiss cost-beneficial or potentially cost-beneficial alternatives.
7. Since future work is not expected to change the NRC's understanding of the fundamental processes affecting potential environmental consequences of spent fuel pool zirconium fires (beyond the significant state-of-knowledge captured by this Regulatory Analysis, the supporting Spent Fuel Pool Study, and the numerous past investigations of this issue), I believe that activities related to the development of the Environmental Impact Statement and proposed rule for Waste Confidence should proceed unencumbered by the follow-on activities recommended earlier in these remarks.
8. I believe that the characterization of the Regulatory Analysis in the Commission paper needs to be strengthened to capture the importance of these items, such that the Commission paper provides the Commission with a balanced perspective on which to provide direction.

Thank you for your time.

NEI 13-02
Industry Guidance to Implement
EA-13-109

Advisory Committee on Reactor Safeguards

October 2, 2013

General Characterization

- Cooperative effort between industry and NRC
- Numerous public meetings and technical exchanges to develop NEI 13-02
  - Good alignment between industry and NRC on guidance document with two topics currently under discussion
- Industry is working toward common understanding of the elements of the HCVS design
  - A November design workshop is planned

Key NEI 13-02 ISG Topics

- Instrument qualification - **Resolved**
- Accident management (EPG/SAG) - **Addressed**
- Generic Letter 89-16 (Appendix E) - **Addressed**
- Drywell temperature design value – **In Discussion**
- Anticipatory venting – **Use FLEX Resolution**
- DW vent with engineered filter option versus WW vent proposed for Mk I & II plants – **Rulemaking Topic**

ACRS Sub-Committee Follow-up Items

- Industry Engagement
  - NEI 13-02 discussed at BWROG meeting in July for a full day
- Procedural Interaction
  - BWROG Emergency Procedure Committee involved in writing team and key elements provided at committee meetings
- Anticipatory Venting
  - Using JLD Generic Issue Process for NRC endorsement
- CAP
  - Protection for Inadvertent action protects CAP capability when venting is not needed

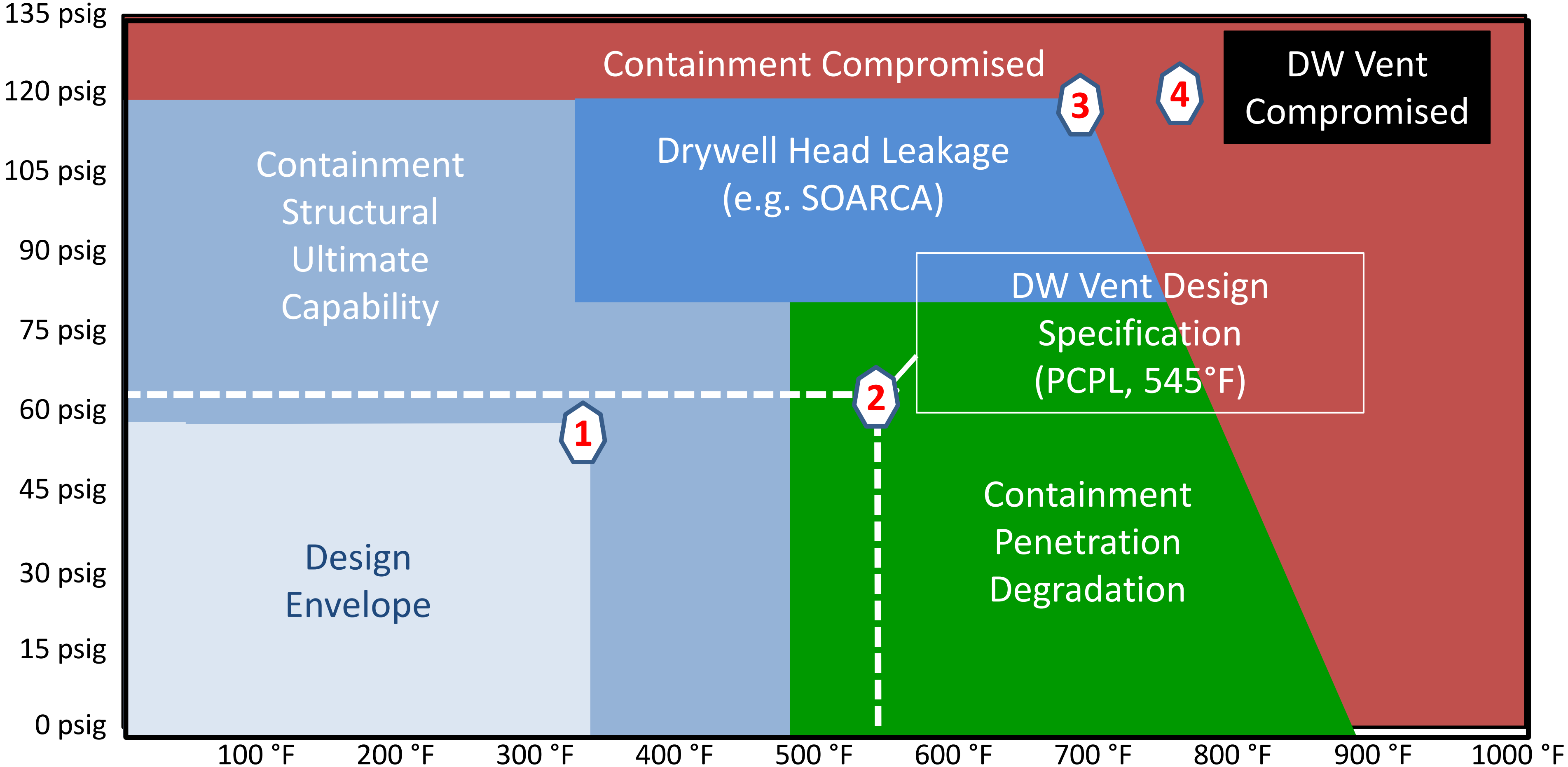
NEI 13-02 Changes from ISG Revision

- Improved Severe Accident definition
- Clarified that Components include Instrumentation
- Corrected Overview Section 1.4
- Discussion on DW Vent Design, Operation and Capability
  - Used the following figure to demonstrate the inherent margin provided by selection of 545°F

HCVS DW Vent Design Temperature

- The Vent system is generally made up of the same type of hard and elastomer components that the containment are made from:
  - Design values yield a higher rupture/failure value
  - Design, Procurement and Vendors work in design value space not rupture/failure space
  - 545F is significantly higher than the design values of the containment components which yield rupture/failure values illustrated on the following diagram
  - No testing of containment or HCVS vent components will be required to demonstrate ultimate capability

HCVS DW Vent Consistent with Containment Capability



Anticipatory BWR Venting

- Anticipatory Venting supports extended RCIC Operation for Mitigating Strategies/FLEX
- Draft whitepaper provided to NRC to address conditions for ELAP/FLEX use
- Preferred choice for Containment/Core Decay heat removal.
  - Maximizes core cooling and containment function reliability
  - Minimizes support systems and operator actions and uses installed equipment
  - Utilizes ≈10 times more efficient method of heat transfer
- CAP
  - Motor Driven ECCS Pumps are not available in an ELAP and Steam driven systems do not rely on CAP
  - CAP is available when venting because the containment will be at a higher pressure that drives the need for venting and the containment pressure is not reduced to zero when venting
- Venting capability will be enhanced with EA-13-109 in BWR MK I & II

Backup Information

Severe accident elements of EA-13-109

- Two phased approach (wetwell and drywell)
- Design vs. capability of system components
  - Hydrogen generation from severe accident
  - Core concrete interaction
  - Temperature and radiation levels

"The HCVS shall be designed to withstand and remain functional during severe accident conditions,... The design is not required to exceed the current capability of the limiting containment components." EA-13-109 criteria 1.2.10

Functional Requirements

- Severe accident capability
- Limit containment pressure
- Vent capability from wetwell and drywell under ELAP conditions
- Control the use of common systems within and between units
- Addresses all venting modes

Design Attributes

- Simplified operator actions with redundant controls
  - Prevention of inadvertant actuation
  - Habitability/accessibility under severe accident conditions
- Prevention of cross flow to buildings/systems/units
- Protection from flammable gas ignition
- Initial 24 hour operation with installed equipment
- Longer term operation to support venting function
- Wetwell design consistent with saturation conditions at containment pressure limits

References on Containment Failure

- "Mark I Containment Severe Accident Analysis." Prepared for the Mark I Owners Group, Chicago, IL: Chicago Bridge & Iron, NA-CON, April 1987
- Grieman, L.G. et al., Reliability Analysis of Steel Containment Strength, U.S. Nuclear Regulatory Commission, Division of Technical Information & Document Control, NUREG/CR-2442, June 1982.
- NUREG/CR-5334, "Severe Accident Testing of Electrical Penetration Assemblies", Clauss, D.B., November 1989
- Wayne Sebrell, The Potential for Containment Leak Paths Through Electrical Penetration Assemblies Under Severe Accident Conditions, NUREG/CR-3234; SAND83-0538, dated July 1983.
- R.F. Kulak et al., "Structural Response of Large Penetrations and Closures for Containment Vessels Subjected to Loadings Beyond Design Basis," NUREG/CR-4064, February, 1985
- Bridges T.L., Containment Penetration Elastomer Seal Leak Rate Tests, NUREG/CR-4944, July 1987.
- Koenig L., "Performance of Seals and Gaskets Under Severe Accident Conditions," DE-ACO4-76DP00789, Sandia National Laboratory, pp. 174-180.

Anticipatory BWR Venting

BWROG Guidance

Allowed to Vent Containment When:

Containment Pressure > Scram Setpoint

AND

Required for core cooling*/lower offsite dose

*Maintain RCIC operation or allow low pressure injection

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

Mark I and Mark II BWRs Containment Venting Systems

Guidance for Order EA-13-109

Briefing to the

Advisory Committee on Reactor Safeguards

October 2, 2013

Japan Lessons Learned

Agenda

- Overview and Schedule
- NRC staff presentation – Interim staff guidance development (JLD-ISG-13-02)
- Questions and comments

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

Overview and Schedule

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

Overview - SRM

- SECY-12-0157 issued November 26, 2012
- SRM issued March 19, 2013
  - Modify Order EA-12-050 to include severe accident conditions
  - Develop technical bases for filtering strategies with drywell filtration and severe accident management of containments
  - Develop proposed and final rules for filtering strategies
  - Seek Commission guidance on use of qualitative factors in regulatory decisions

Overview – Order EA-13-109

- Order EA-13-109 issued June 6, 2013

- Included a phased approach to ensure minimal delays in implementing adequate protection provisions and cost justified safety enhancements of the Order, while allowing possible development of alternate approaches

- Also included a 2-phase implementation of Order with subsequent incorporation of requirements into rulemaking activities, which would also include broader accident management strategies

Phase 1 - Scope

Mark I and II

- Wetwell Venting System

- Requirements from EA-12-050
  - Reliable, hardened containment venting system
  - Adequate protection

- Revised order added Severe Accident Capability
  - Cost Justified Safety Enhancement

Phase 1 - Timeline

- Implementation :
  - no later than startup from the second refueling outage that begins after June 30, 2014, or June 30, 2018, whichever comes first.

- Integrated Plan
  - June 30, 2014

Phase 2 - Scope

Mark I and II

- Drywell Venting System
- Cost Justified Safety Enhancement

<u>Options:</u>

- Installation of severe accident capable drywell vent

Or

- Develop reliable strategy that obviates need for a drywell vent

Phase 2 - Timeline

- Implementation :
  - no later than startup from the first refueling outage that begins after June 30, 2017, or June 30, 2019, whichever comes first
- Integrated Plan
  - December 31, 2015

Schedule - ISG

- ISG issuance endorsing NEI 13-02 – October 2013
- ISG issued for public comment – September 18, 2013
- Public comment period ends – October 18, 2018
- Public and industry interactions – June to Sept. 2013
  - 7 public meetings/webinars
  - Next public meeting – October 2013 (tentative)

NRC Presentation

Draft Interim Staff Guidance

(JLD-ISG-2013-02)

Interim Staff Guidance For Order EA-13-109
Severe Accident Capable Vent
Mark I and Mark II Containments

– **Order EA-13-109**
  Primary Objective

  Prevent containment failure from over-pressure/over-temperature (before and after core damage, including a breach of RPV by molten core debris)

  Severe accident conditions relate to the dominant accident sequences. Most likely (dominant) failure mode of the containment from over-pressure/over-temperature is the failure of the drywell head flange seal as predicted by severe accident calculations and confirmed by Fukushima. Severe accident capable vent shall also be designed to, among other things, prevent failure of the drywell head seal

  Assist in the removal of decay heat

Interim Staff Guidance For Order EA-13-109 Severe Accident Capable Vent Mark I and Mark II Containments

– **Order EA-13-109**

Phase 1 – Wetwell vent

Phase 2 – Drywell vent or reliable venting strategies that make it unlikely that drywell venting is needed. Place holders in NEI 13-02 (Section 3 and Appendix C) for inclusion of guidance at a later date for drywell vent or venting strategies.

Different timelines allow for consideration of the nexus between Phase 2 and rulemaking into a cohesive set of results and requirements.

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Interim Staff Guidance For Order EA-13-109
Severe Accident Capable Vent
Mark I and Mark II Containments

Order EA-13-109

– **HCVS Functional Requirements**

Performance Objectives:

Minimize reliance on operator actions

Minimize plant operators exposure to occupational hazards

Account for radiological conditions that could impede personnel actions

Controls and indications shall be accessible and functional under a range of plant conditions

Interim Staff Guidance For Order EA-13-109 Severe Accident Capable Vent Mark I and Mark II Containments

- **Order EA-13-109**

- **HCVS Functional Requirements**

  Design Features

  Vent Capacity, effluent discharge, minimizing unintended cross flow, capability to operate from main control room or remote location, minimum capability to operate 24 hours by means of permanently installed equipment, means to monitor the status of the vent system, monitor effluent discharge for radioactivity, withstand and remain functional during severe accident conditions, ensure that lower flammability of gases passing through HCVS are not reached or system designed to withstand deflagration and detonation loading, and operation, testing, inspection and maintenance.

Interim Staff Guidance For Order EA-13-109
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Mark I and Mark II Containments

- **Order EA-13-109**

- **HCVS Quality Standards**

  Containment isolation barrier (consistent with the design basis of the plant)

  Beyond the isolation barrier (reliable and rugged performance that ensures HCVS functionality following a seismic event)

- **HCVS Programmatic Requirements**

  Develop, implement and maintain procedures

  Train appropriate personnel in the use of HCVS

Interim Staff Guidance For Order EA-13-109 Severe Accident Capable Vent Mark I and Mark II Containments

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

– **JLD-ISG-2013-02**

Staff endorsement of the guidance in NEI 13-02 is subject to the following clarifications and exceptions:

EPGS/SAGs/ EOPs/SAMGs

NEI 13-02 contains many references to the BWROG generic EPGs/SAGs. Staff's believes the procedural requirements to operate and make use of HCVS including whether a drywell vent is needed during severe accident conditions will depend on Phase 2 evaluations and the related rulemaking. Staff's endorsement of NEI 13-02 is not an endorsement of the BWROG generic EPGs/SAGs or plant-specific EOPs/SAMGs.

NEI 13-02 included a statement at staff's request that the requirements of Order EA-13-109 takes precedence over any design features that may be required of the HCVS to facilitate the PGs/SAGs/EOPs/SAMGs.

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– **JLD-ISG-2013-02**

Anticipatory Venting

References in NEI 13-02 for using HCVS to vent containment at lower pressure to facilitate the use of a low-pressure portable pump or to allow continued use of installed steam-driven equipment is currently being reviewed by staff as part of submittals under Order EA-12-049. Therefore, it is not addressed in this ISG.

Appendix E – Interface with the requirements of GL 89-16

Contains no information related to the design and implementation of the HCVS. Staff did not review Appendix E, as it is not within the scope of the ISG.

Interim Staff Guidance For Order EA-13-109 Severe Accident Capable Vent Mark I and Mark II Containments

– **<u>JLD-ISG-2013-02</u>**

<u>Severe accident conditions – Drywell Temperature</u>

NEI 13-02 states design pressure and temperature for the drywell vent are PCPL and 545°F

<u>Staff position</u>:

Industry proposal is reasonable for Phase 1 decisions concerning the common wetwell drywell vent portions. Drywell head flange seal is the most likely (dominant) failure mode of the containment from over-pressure/over-temperature as predicted by severe accident calculations and confirmed by Fukushima. Therefore, in Phase 2, the drywell vent or alternate filtration strategies shall be developed and operated in a manner to protect the drywell head seal from gross leakage.

Postulated severe accident conditions could exceed the proposed design conditions as depicted on the NEI 13-02, Figure 2-1 and supported by previous severe accident analysis.

Staff believes that ultimate integrity capability values (margin) of the drywell head seal and drywell vent, including vent operation should inform the evaluation of the head seal leakage consideration during Phase 2 and rulemaking evaluations.

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JLD-ISG-2013-02

Instrumentation Reliability and Operating Environment

NEI 13-02 provides a high level approach for describing the instrumentation design considerations. The staff endorses the guidance and plans to continue evaluating the template iterations and providing future input.

Staff position:

To continue evaluating new or existing instrumentation advances and options documenting operational experience in which well-designed instrumentation systems were able to maintain sustainability for hazardous locations.

Interim Staff Guidance For Order EA-13-109 Severe Accident Capable Vent Mark I and Mark II Containments

– **Other Observations**

Section III to Enclosure 1 of the communication of Order EA-13-109 stated that licensees with Mark II containments may resolve concerns about suppression pool bypass by an alternative approach to Phase 1 and Phase 2 requirements by the installation of a containment drywell vent with an installed engineered filter.

The ISG states that the above alternative, in effect, applies to both Mark I and Mark II containments.

Questions & Discussion

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

NRC Presentation
Draft Interim Staff Guidance
(JLD-ISG-2013-02)

Japan Lessons Learned

Interim Staff Guidance For Order EA-13-109
Severe Accident Capable Vent
Mark I and Mark II Containments

– **Order EA-13-109**
  Primary Objective

  Prevent containment failure from over-pressure/over-temperature (before and after core damage, including a breach of RPV by molten core debris)

  Severe accident conditions relate to the dominant accident sequences. Most likely (dominant) failure mode of the containment from over-pressure/over-temperature is the failure of the drywell head flange seal as predicted by severe accident calculations and confirmed by Fukushima. Severe accident capable vent shall also be designed to, among other things, prevent failure of the drywell head seal

  Assist in the removal of decay heat

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– **Order EA-13-109**

Phase 1 – Wetwell vent

Phase 2 – Drywell vent or reliable venting strategies that make it unlikely that drywell venting is needed. Place holders in NEI 13-02 (Section 3 and Appendix C) for inclusion of guidance at a later date for drywell vent or venting strategies.

Different timelines allow for consideration of the nexus between Phase 2 and rulemaking into a cohesive set of results and requirements.

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