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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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FUKUSHIMA

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RELIABILITY AND PRA SUBCOMMITTEES

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TUESDAY

JULY 7, 2015

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ROCKVILLE, MARYLAND

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The Subcommittees met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 8:30 a.m., Stephen P. Schultz, Chairman, presiding.

COMMITTEE MEMBERS:

STEPHEN P. SCHULTZ, Meeting Chairman

RONALD G. BALLINGER, Member

DENNIS C. BLEY, Member

CHARLES H. BROWN, JR. Member

MICHAEL L. CORRADINI, Member

HAROLD B. RAY, Member

JOY L. REMPE, Member

GORDON R. SKILLMAN, Member

JOHN W. STETKAR, Member

DESIGNATED FEDERAL OFFICIALS:

WEIDONG WANG

JOHN LAI

ALSO PRESENT:

EDWIN HACKETT, Executive Director, ACRS

JONATHAN BARR, RES

ROBERT BEALL, NRR

HOSSEIN ESMAILI, RES

ED FULLER, RES

JEFF GABOR, Erin Engineering

PAUL GUNTER, Beyond Nuclear

STEVEN KRAFT, NEI

MARY LAMPERT, Pilgrim Watch*

STUART LEWIS, EPRI

ABY MOHSENI, NRR

WILLIAM RECKLEY, NRR

MARTIN STUTZKE, RES

RANDOLPH L. SULLIVAN, NSIR

RUTH THOMAS, Environmentalists, Inc.*

DOUGLAS TRUE, Erin Engineering

*Present via telephone

A G E N D A

P R O C E E D I N G S

8:30 a.m.

CHAIRMAN SCHULTZ: Good morning. This meeting will now come to order. This is a joint meeting of the Fukushima and the Reliability and PRA Subcommittees, two standing subcommittees of the Advisory Committee on Reactor Safeguards.

I'm Stephen Schultz, the chairman of the Fukushima Subcommittee. ACRS members in attendance are Harold Ray, Gordon Skillman, John Stetkar, Dennis Bley, Ron Ballinger, Charlie Brown, Joy Rempe and Mike Corradini.

In this meeting, the Subcommittee will review a Commission information paper and the draft rulemaking regulatory basis document for containment protection and release reduction for Mark I and Mark II boiling water reactors.

We will hear presentations from the NRC staff describing these documents and their path forward for addressing release reduction and filtering strategies and severe accident management issues for these facilities. We will also hear from industry representatives.

The Subcommittees will gather information, analyze relevant issues and facts, and

formulate proposed positions and actions as appropriate for deliberation by the full Committee. This entire meeting will be open to public attendance and participation.

The meeting is being conducted in accordance with the provisions of the Federal Advisory Committee Act. Rules for the participation in the meeting have been published in the *Federal Register*, as part of the notice for this meeting. Weidong Wang and John Lai of the ACRS staff are the Designated Federal Officials for this meeting.

A transcript of this meeting is being kept and will be made available, as stated in the *Federal Register* notice. Therefore, we request that participants in the meeting use the microphones located throughout the meeting room when addressing the Subcommittee. When recognized, first identify yourself and speak with sufficient clarity and volume so that you may be readily heard.

We have received written comments and requests for time to make oral statements from members of the public regarding today's meetings. Also, there are individuals on the bridge line today who are listening in on today's proceedings. To effectively coordinate their participation in this

meeting, we will be placing the incoming bridge line on mute, so that those individuals may listen in.

At the appropriate time later in the meeting, we will provide the opportunity for public comments from the bridge line and from members of the public in attendance. I remind all of us to turn off our cell phones and communication devices so there is no interruption during the meeting.

We'll now proceed with the meeting, and I'll call on Aby Mohseni of the Office of Nuclear Reactor Regulation to open the presentations today. Aby, welcome.

MR. MOHSENI: Thank you very much Dr. Schultz and distinguished members for the opportunity to address the Committee. The NRC staff is here today to discuss the draft regulatory basis for the containment protection and release reduction rulemaking. The staff previously presented a number of its insights to this Subcommittee on August 22nd and November 19th, 2014, and benefitted from your feedback.

The NRC staff performed a preliminary quantitative risk evaluation using a high level conservative estimate, that determined that any of the potential alternatives evaluated within the CPRR

rulemaking would not be considered a substantial safety enhancement.

The staff's quantitative analysis shows that based on the NRC's safety goal policy statement's quantitative health objectives, there are no expected individual fatalities from a postulated severe accident at a BWR with a Mark I or a Mark II containment, and the individual latent cancer fatality risk is well below the QHO for what is safe enough.

The NRC staff developed the Commission paper SECY-15-0085, and its supporting draft reg basis that provides details of the high level conservative estimate and the recommended path forward. Please note that the Commission will be voting on that SECY paper.

The SECY paper describes the staff's plans to make the requirements of the severe accident capable containment vent order, EA-13-109, generically applicable to Mark I and Mark II BWRs, and that the use of external water during a severe accident is a worthwhile benefit.

The presentations today will also provide the staff's perspective that based on the more refined risk analysis of each alternative evaluated,

a regulatory requirement for engineered filters would not be needed. The staff's presentations this morning will describe the decision rationale to support the recommended path forward.

To start today's presentation, Bob Beall will provide an overview of the CPRR regulatory evaluation. Bob is the lead PM for the CPRR rulemaking in the Office of Reactor Regulation. Next, Marty Stutzke from the Office of Nuclear Regulatory Research will talk about the risk and PRA evaluation performed with a draft CPRR reg basis.

Hossein Esmaili from the Office of Nuclear Reactor Research will then present a summary of the MELCOR results, and the final staff presentation will be Jon Barr, also from the Office of Nuclear Regulatory Research, talking about the use and results of the MACCS code in the CPRR rulemaking.

We also have Bill Reckley here next to me, who will add, as needed, comments to or answer some of the questions that the Committee members may have. I will now turn it over to Bob.

MR. BEALL: Good morning. Okay, thank you. Good morning everyone. I'm Bob Beall. I'm the project manager for the CPRR rulemaking, and I'd like to start off today by going over some background

information of how we got to where we are in the CPRR rulemaking.

Back in November 2012, the NRC staff made a recommendation to the Commission in SECY paper 12-0157, that we issue an order to licensees that required insulation of engineered filters for boiling water reactors with Mark I and Mark II containments. The staff informed the Commission that when qualitative factors were considered in addition to quantitative factors, a recommendation to require the installation of engineered filters was justified.

However, the staff acknowledged that in a comparison of only the quantitative costs and benefits for the proposed engineered filters it considered safety enhancements, they would not demonstrate the benefits to exceed the associated costs of engineered filters.

In the Commission staff requirements memorandum, SRM to the SECY paper, it directed the staff to issue an order to the licensee to install severe accident capable vents in BWRs with Mark I and Mark II containments.

That became Order EA-13-109. In addition, the Commission directed the staff to perform several other actions, to pursue a rulemaking

to fully evaluate the design and installation of engineered filters containment venting systems, and to perform -- do a performance-based strategy, containment strategy for BWR Mark I and Mark II containments.

You also asked us to explore the requirements associated with the measures to enhance the capability to maintain containment integrity and to cool core debris, evaluate a number of performance criteria including things like decontamination factors and equipment and procedure availability, and lastly develop a SECY paper on the use of qualitative factors for regulatory processes and of course raise any policy issues to the Commission if we find any.

Some recent NRC actions that are applicable to the CPRR rulemaking is that in Commission paper -- in the COMSECY-13-0030, the staff evaluated the benefits of expedited spent fuel transfer to dry storage. After considering the analysis results, the staff recommended that further analysis would not support a requirement that reactor licensees expedite the transfer of spent fuel.

In the SRM to the COMSECY, the Commission also determined that no additional regulatory actions were warranted for spent fuel storage. The

Commission confirmed that the risk is low enough that a requirement to expedite the transfer of spent fuel will not rise to a substantial safety enhancement.

As we've shown in a few slides, the risk calculated for the spent fuel transfer analysis are comparable to what we have calculated for the CPRR rulemaking results. Also as directed by the Commission in SRM-12-0157, the staff made a recommendation to the Commission on qualitative factors in SECY Paper 14-0087.

MEMBER CORRADINI: Can I ask a question?

MR. BEALL: Yes sir.

MEMBER CORRADINI: I'm struggling with the first bullet. You're making the comparison why? Because the numbers are the same or because the analysis was the same?

MR. BEALL: The analysis was similar. They did a -- they looked at the QHOs also for the risk factors. We'll be talking about that in a few slides, okay, when we bring that up, okay.

MEMBER CORRADINI: Okay.

MR. BEALL: The staff proposed to update the current cost benefit guidance to include a set of methods that would -- that could be used for the consideration of qualitative factors within a cost

benefit analysis. In the SRM to the SECY paper, the Commission directed updating the guidance, but did not authorize the expansion of use of qualitative factors. The Commission also stated that the appropriate degree of the use of qualitative factors on regulatory decision-making ultimately lies with the Commission.

As part of the CPRR rulemaking, the staff identified four major alternatives and a number of sub-alternatives to evaluate, using filtering strategies and severe accident management. The four major alternatives were number one, take no action. This is basically the Order EA-13-109, as implemented, without any additional regulatory actions.

Alternative 2 was pursue rulemaking to make the Order EA-13-109 generically applicable for the protection of BWR Mark I and Mark II containments against over-pressurization. Alternative 3 is to pursue rulemaking to address the overall BWR Mark I and Mark II containment protection against failure modes, by making the Order generically applicable, and requiring external water addition into the reactor pressure vessel or drywell, to prevent containment failure from overpressurization or liner

melt-through.

And the last Alternative 4 was to pursue rulemaking to address both containment protection against multiple failures and release reduction measures for controlling releases through containment venting systems. This would include making the Order EA-13-109 generically applicable, to require external water addition and licensees were required to reduce the fission product release through the containment by either implementing strategies to maximize the scrubbing or filtering of fission products before venting from containment, and/or installing an engineered filter from the containment vent path.

In this slide, we talk about a high level conservative estimate calculation the staff performed. The high level conservative estimate provides a conservative estimate of the latent cancer fatality risk, which is then compared to the NRC's safety goal policy statement.

This conservative high level estimate uses the highest values from BWRs with Mark I and Mark II containments for extended loss of AC power or ELAP, and an individual latent cancer penalty risk values for more than one plant. So this is like usually the worst case for each of those factors.

There's also assumptions about the FLEX equipment installed as part of the mitigating strategy order, would work only six out of ten times to arrest the accident involving an ELAP. The NRC safety policy statement --

MEMBER SKILLMAN: Excuse me. What is the basis of that six per ten times? Where did that come from?

MR. BEALL: Marty's going to go into that in a few minutes when he arrives. But it was based on some industry data that we have from -- I think it was from walkdowns Bill, and --

MR. RECKLEY: But it's -- again, Marty will get into it a little more in detail. But it's a combination of factors between equipment failure, primarily RCIC in this case, and then also assumptions on human errors in the control room and in the case of FLEX, outside of the control room the assumption is human errors go up.

So it's basically an assumption. We don't want to come across as saying this is our best estimate for the reliability or functionality of this mitigating strategy. It's just a PRA assumption, so that we could carry through as part of this conservative estimate, as Bob's described it.

MEMBER SKILLMAN: All right, thank you.

MR. BEALL: So the NRC safety policy goal statement --

MEMBER STETKAR: Bob?

MR. BEALL: Yes sir.

MEMBER STETKAR: I was just going to ask this, and this may be a Marty question too, but I kind of got confused when I saw this figure. I understand how the high level conservative estimate is, was quantified, okay. I understand that. This figure, I don't understand how all of the estimates with the uncertainty bounds relate to that thing.

I mean this figure forces me to compare the QHO, this high level estimate, and then all of that detail down in the bottom, and understand what they mean relative to one another. And I for one couldn't figure out how all the detail relates to the high level, what you're characterizing as the high level conservative estimate.

MR. BEALL: Well, the high level conservative estimate --

MEMBER STETKAR: I know how that was calculated.

MR. BEALL: Right.

MEMBER STETKAR: You took the highest

ELAP frequency and you multiplied it by the highest conditional individual latent cancer fatality.

MR. BEALL: Right.

MEMBER STETKAR: Right?

MR. BEALL: Right.

MALE PARTICIPANT: From all this area.

MEMBER STETKAR: Okay. No, from all of the sites.

MR. BEALL: All the sites, all the sites.

MALE PARTICIPANT: Okay.

MEMBER STETKAR: I know how that was calculated. The bottom things are an amalgam of ostensibly all of the plants, except it uses Peach Bottom for the MELCOR and the MACCS analyses, and all right. Peach Bottom has the second highest population density. I don't know which site has the highest.

But I don't know -- I don't know what is the basis for the factor of 30 difference, on average, between the high level conservative estimate and the -- all of the stuff down below?

MR. BEALL: Right. Well Marty's here now, so you can --

MEMBER STETKAR: Oh, hi Marty.

MR. STUTZKE: Good morning. It's Marty

Stutzke from the Office of Research. In my presentation, I'll try to clarify where the magic numbers come from.

MEMBER STETKAR: Okay, okay. I actually understand where all the numbers come from. I just don't know how to interpret this particular figure. I understand how to interpret the figure without all of the noise in the bottom, and I understand how to interpret all of the noise on a stand-alone basis.

I don't understand how to interpret the noise compared to that high level conservative estimate. So if you want to think about that before you come up.

MR. STUTZKE: Okay.

MR. BEALL: So as it also -- as the chart also shows, the high level conservative estimate is more than an order of magnitude below the QHO, and it's also close to the COMSECY risk value for the expedited spent fuel transfer, which is the red triangle. Which is kind of going back to your question.

We're trying to show that this is another data point that the Commission showed that the risk was low enough relative to QHO, showing it compared to what the numbers we're coming up with, to say that

okay, as from a regulatory aspect, we have -- there's, you know, we're very low on -- you have a lot of margin in order of magnitude below the QHO.

MEMBER CORRADINI: Okay. But so since you brought up the red triangle, is there red triangle mitigated or unmitigated? What does the red triangle represent, an unmitigated event at the same Peach Bottom site? So I'm comparing -- I look at the red triangle and I look at all these bars, and they should be a one to one similarity in calculational procedure? Because I don't think so. Could somebody out there on the staff?

MR. ESMAILI: This is Esmaili. The red triangle was the highest, was the highest release and the highest consequence that we found for all the plants. This was not for Peach Bottom. This was --

MEMBER CORRADINI: So this was the 0130?

MR. ESMAILI: This was SECY-13-0030, that we biased all the results to highest releases, you know.

MEMBER CORRADINI: Okay.

MR. BEALL: So it's meant to be a comparison to where that analysis showed, to where we are now. Okay.

CHAIRMAN SCHULTZ: So Bob, is the point

-- is this provided to do a comparison of the decision-making process that was used for the spent fuel pool evaluation?

MR. BEALL: That's correct, that's correct. It shows you relative to what the Commission said that this is -- this is safe enough from this aspect, and where we line up in comparison to that.

CHAIRMAN SCHULTZ: And when we get to Marty's presentation, we're going to hear more about the analysis results? I'll call them below the line, which appears to demonstrate that even the status quo, without any improvements, which we have in place going forward with the rulemaking that's already ongoing and the orders that have been issued, that the status quo already demonstrates that we are below the decision-making criteria that we used for the -- the decision-making process evaluation that was used for the spent fuel pool.

MR. BEALL: Yes sir, that's correct.

CHAIRMAN SCHULTZ: That is to the far left of the curve?

MR. BEALL: That's correct.

CHAIRMAN SCHULTZ: Thank you.

MR. BEALL: So the higher conservative

estimate reaffirms the NRC's safety policy goal statement that determined that the risk was low enough and it did not rise to a substantial safety enhancement.

MEMBER CORRADINI: So again, just so not looking at the bunch of bands, I want to look at the dashed line, which is the high level conservative estimate, and that's a -- just simply a product of the ILCF and the ELAP frequency, not necessarily at the same site but the biggest of one times the biggest of the other?

MR. BEALL: Right, times the FLEX availability. That's like 60 percent.

MEMBER CORRADINI: Okay, okay. All right, thank you.

MR. BEALL: Okay.

MEMBER STETKAR: Oh, I'm sorry. Can I -- one last thing.

MR. BEALL: Sure.

MEMBER STETKAR: That's an as-is calculation? That's without a hardened vent necessarily at that site, or is it with the Order in place? It's not with filtering, it's not with water addition. That I've got, but it's with the hardened vent in place, or what is it? I'm still trying to

understand the comparison.

MR. STUTZKE: Yeah, this is Marty Stutzke. It doesn't make any difference, in the sense it's not taking any credit for operator actions post-core damage.

MEMBER STETKAR: Oh, okay.

MR. STUTZKE: So as core damage is released --

MEMBER STETKAR: I'm sorry it is. Without operator actions, FLEX is guaranteed to fail. So it does.

MR. STUTZKE: Operator actions post-core damage. Prior to core damage --

MEMBER STETKAR: We'll get into more of that.

MR. STUTZKE: We'll get into that.

MR. BEALL: Okay. On this slide, the industry approach to comply with Phase 2 of the Order is to -- is for the addition of water during a severe accident as part of the actions needed to support venting and help prevent overpressurization of the Mark I and Mark II containments.

Proposed severe accident water addition measures designed to limit the maximum containment temperatures, and also to make unlikely for plants to

need a severe accident drywell vent by maintaining the capability through the vent from the containment well, the containment wetwell, excuse me.

The proposed inclusion of external water addition has the additional benefit of providing capabilities to address other containment integrity failure modes, such as leak through the drywell head and liner melt-through. In addition, venting from the containment through the wetwell reduces the release of radioactive materials, because the water in the suppression pool is scrubbed just prior to its release.

For release reduction, the staff also assessed in the draft regulatory basis the requirement to reduce or filter planned releases during containment venting operations following core damage. While the installation of engineered filters do show some advantages when evaluated against additional off site consequences measures, like the number of people that would be subject to long-term protective actions, the results of the CPRR analysis show that the potential benefit of regulatory actions to reduce the amount of radioactive materials released using an engineered filter during severe accidents do not meet the quantitative criteria for

providing a substantial safety improvement, and the cost would far exceed the calculated benefits.

CHAIRMAN SCHULTZ: Bob, a question. With regard to the evaluation that the staff has done for the large engineered filter and the small engineered filter, have there been technical evaluations done on the capability of an engineered filter, large or small?

MR. BEALL: You're talking about the DF factor, decontamination factor?

CHAIRMAN SCHULTZ: Yes, focusing on a DF factor under a variety of scenarios that the filter might be put into -- put into place to address.

MR. BEALL: Yeah, the staff looked at that, I think --

CHAIRMAN SCHULTZ: If we're going to discuss is later, that's fine.

MR. ESMAILI: We're going to look at the operation of the filter. Just like what we did in 12-0157, we assumed the DF based on the fact that the fission products are scrubbed in the pool. What is coming out of the pool is probably, you know, small particles. But again, we looked a DF of 10, assuming that there is, you know, that the scrubbing is very effective in the pool, to a DF of 1,000.

Marty will show you that there is not much difference in terms of whether we assume a DF of 10 or 1,000, because -- because of that unavailability of the FLEX of .6. So in a lot of cases, you are going to lead to a containment failure, and those are not scrubbed anyway. So that engineered filter -- so it's -- things are going to be dominated by the unfiltered, you know, from the containment failure phase.

CHAIRMAN SCHULTZ: I just wanted to understand how much was done to evaluate the capability of a filtration system, in that when we say oh, we'll use a factor of 10 or a factor of 100, we certainly get the impression that those are achievable, and we also get the impression to at least some that that's achievable for all scenarios.

MR. RECKLEY: This is Bill Reckley. One way to look at this was in the analytical, we can just make assumptions. If we had found that a particular DF would have made a big difference, the next step in the process would have been to evaluate whether the things that are available in the marketplace could have actually delivered that kind of DF.

But because we didn't get there, we

didn't go that next step to actually say whether such a filter would be available. Now we are familiar with the discussions and those that are marketing filters and the claims that they make, experiments that have been done in Europe. So that was discussed in a little bit of detail in the SECY paper in 2012.

But for this effort, we didn't go back to that, because no matter what DF -- as Hossein said, no matter what DF we assumed, you didn't get there from a regulatory standpoint. So we didn't go back and check the actual availability of the hardware.

CHAIRMAN SCHULTZ: Thank you.

MEMBER REMPE: So I was going to ask this later, but just out of curiosity, the industry talks about a large and a small filter, versus what you assumed, and were those comparable assumptions?

MR. ESMAILI: I think in our definition, the large and small -- I think what the industry's talking about is the loading, the capacity of the filter. What Marty was looking at again was the -- how much decontamination you get out of this filter, whether it's 10 or 1,000. So it was a large --

MEMBER CORRADINI: Regardless of capacity.

MR. ESMAILI: Regardless of capacity,

and those figures that you show, those boards at the bottom, one of those boards is for a DF of 10, and one of these boards is for a DF of 1,000. What he's going to show you later is that it doesn't make that much of a difference, because it is dominated with what you cannot filter. It's coming out of the, you know, it's either liner melt-through or the -- or the head leakage that you cannot filter.

So even if you filter everything else, you know, even if your DF is ten million, you're still outbounded by what you cannot filter.

MR. BEALL: So for the regulatory evaluation, the proposed CPRR rulemaking would make generically applicable in Part 50 requirements of the Order EA-13-109, with the additional requirement for the use of external water. To the extent that the proposed CPRR rulemaking would make the requirements of the Order.

Oh, excuse me. The staff finds that Alternative 3 has sufficient regulatory basis to propose with this rulemaking. Finally, supports that the analysis that the staff initially proposed or performed with the Mark I/Mark II containments show that the combined benefits of severe accident water capable events and external water addition enhance

the capability to maintain containment integrity and to cool core debris.

The added water cools the molten fuel and can arrest its progression and prevent a loss of containment functions like liner melt-through, containment pressurization and drywell head leak. The external water addition requirement had minimal cost to the industry, since it's the industry's proposed response to Phase 2 of the Order EA-13-109.

Also making the water addition generically applicable would not rely on regulatory guidance documents like Alternative 2 would.

MEMBER CORRADINI: Can I ask a question there? So you're looking at back to Phase 2 of the Order. But as I remember, it's still unclear as to the guidance whether there would be water addition into the vessel which leaks into the drywell, whether it's injection directly into the drywell, or whether it's a spray into the drywell. It's still unclear. Guidance has not been determined in term of that. Is that correct?

MR. BEALL: That's correct. We also did a preliminary backfit assessment, and for the CPRR rulemaking, the staff has determined that the -- to make generically applicable the requirements for the

Order 13-109 would not -- will not cut the backfit. The additional requirement for the external water capability could constitute a backfit.

The staff will determine during the normal rulemaking process if a complete backfit analysis is required, and ensure that the proposed rulemaking is in compliance with the requirements of C.F.R. 50.109.

MEMBER CORRADINI: Bob, I'm sorry. You go ahead, I'm sorry.

MEMBER STETKAR: When I read this, it kind of bothered me, because what this basically says if everybody decides to go ahead with Alternative 3, the staff can then determine that Alternative 3 is not valid, because it doesn't meet the backfit requirements. Is that correct?

MR. BEALL: That's correct, because we haven't -- we have not done a full backfit analysis.

MEMBER STETKAR: So why didn't you at least do a comparison between Alternative 2 and Alternative 3, to give the public and the Commission some basic idea of what the delta costs and the delta -- I mean we know what the delta benefits are between 2 and 3.

MR. BEALL: Right.

MEMBER STETKAR: We don't have any information about the delta costs between 2 and 3. So now we have people making comments on this draft basis, and we have Commissioners voting on apparently this draft SECY paper, and all of their votes and all the comments may be useless, because we don't have all the information.

MR. BEALL: Well --

MEMBER STETKAR: How do I make a decision on Alternative 2 to Alternative 3? If Alternative 2's going to cost me a $1.98, pretty easy decision. If Alternative 3 is going to cost -- I may have misspoken. If Alternative 3 will cost $1.98, easy decision. But if it's going to cost $198 billion, it may be a different decision.

MR. BEALL: Well Alternative 2 is codifying the Order.

MEMBER STETKAR: I understand that.

MR. BEALL: Okay. So Phase 2 gives licensees two paths. They can do a severe accident hardened vent or find a way to -- some other type of strategies to mitigate the issue. So what Alternative 3 removes the severe accident hardened vent option. It just has --

MEMBER STETKAR: It removes the severe

accident hardened vent without water addition option. But it has a severe accident --

(Simultaneous speaking.)

MR. BEALL: We don't have the additional cost for a high temperature vent.

MR. RECKLEY: This is Bill Reckley again. Really, we see no difference in the cost between options 2 and option 3. It's basically the difference between option 2 and option 3 is the regulatory treatment of the severe accident water addition, and whether the protection of liner melt-through in option 2 remains a collateral benefit from the Order.

The Order was for overpressure protection. So they add the means of complying with the Order brought in water addition, in support of the venting operations, right. Option 3 changes the rationale to also codify that the water addition is for the protection of liner melt-through and over temperature over pressure conditions.

What would be done by the licensees for options 2 and option 3 are the same. It is basis for the rule and what the rule captures. So it ends up being a regulatory treatment question that we didn't see affecting or causing there to be a different cost

between the two, other than some administrative rulemaking cost with option 3, plus the scope changes. But for what's done in the field, it would be the same.

MEMBER CORRADINI: So can I ask John's question differently? The way you explained it didn't help me. So I'm interpreting that if the staff feels that option 3, which is a hardened vent, a severe accident capable hardened vent along with water addition and management is potentially a doable option, it seems like they go together.

So the way I read this is you could actually determine if something was too expensive they wouldn't go together. Am I misunderstanding?

MR. RECKLEY: No. All this is saying is from a --

MEMBER CORRADINI: You can say yes, I'm misunderstanding.

MR. RECKLEY: Again, I think the easiest way to look at the options 2 and option 3 is to think of what the rule would change, the rule or the supporting language of the rule would say. Option 2 would say licensees shall provide over pressure protection for Mark I and II containments.

MEMBER CORRADINI: Okay.

MR. RECKLEY: The way licensees are going to do that, they're going to include severe accident water addition as part of Phase 2 of that Order. That's fine; that's the flexibility the Order gave.

MEMBER CORRADINI: Okay.

MR. RECKLEY: Option 3 would say licensees shall protection against the dominant failure mechanisms for Mark I and II containment, over pressure, over temperature and liner melt-through. What the licensees will do is the same, because they're already going to do severe accident water addition in support of the Order.

MEMBER CORRADINI: But the way -- okay, all right.

MR. RECKLEY: Now the fact that the Order only required over pressure protection means that from a regulatory standpoint, when we do the reg analysis for option -- for Alternative 3, we would have to do a backfit, because the addition of those other requirements for protection of the dominant failure mechanisms was not addressed in the Order.

MEMBER CORRADINI: But I think what John's asking, which I guess he and I were teeing off the same third versus the fourth bullet part of this is it just makes common sense. You already must

have some sort of estimate that this is a doable thing at a reasonable cost.

MR. RECKLEY: Well it's -- it is -- well, from our standpoint, it will have already been done in order to implement the Order.

MEMBER CORRADINI: Well but -- okay.

MEMBER STETKAR: You know Bill, I read those snippets in there, where it says "the staff believes," "the staff thinks." I heard this morning "we think," "we sort of know," "we kind of think." But when it finally comes down to the words in the SECY, the words say what is presented on this slide, that if people decide that Alternative 3 is the path forward, then we need to do a backfit analysis.

MR. RECKLEY: Yes.

MEMBER STETKAR: Which may determine that Alternative 3 compared to Alternative 2 is not justified. Regardless of what you say, all of these speculations of we think, we kind of know, we don't think that this is going to be an additional burden except for the regulatory cost to me is all speculation, because it doesn't sound like you've actually thought through the cost to the licensees of compliance with a rule that requires the severe accident water addition.

Which is different than people saying well, the industry has said yeah, you know, it makes a good -- we think we're going to go down this path. We don't think anybody is going to put in only the vents without water addition. The industry hasn't committed to that on the docket. Maybe somebody's going to do that.

MR. RECKLEY: We'll see in December when they submit their plans.

MEMBER STETKAR: Okay.

MR. BEALL: Which will be ahead of the rulemaking.

CHAIRMAN SCHULTZ: Right. But again, to go back to John's -- one of John's first points, and that is you do not see a value and providing a quantification related to the evaluation of Alternative 2 versus Alternative 3 at this point, rather than to say it appears that it's going to be done anyway, so there's no -- there's no delta cost to industry.

That doesn't give either the Commission or the public information that could be valuable in terms of what's being invested and what is the benefit of moving from Alternative 2 to Alternative 3.

MR. BEALL: We do have some cost data in

the reg basis for what we feel, you know, for the cost of the filters and costs of --

MEMBER STETKAR: That's 4 versus 3.

MR. BEALL: Right. I'm looking at 2 versus 3.

MEMBER STETKAR: That's not 3 versus 2.

CHAIRMAN SCHULTZ: Because even though all of these things come together at once, if one is thinking about the CPRR aspects of this particular project versus something else that has been done on hardened vents, that for this one, for this in particular, it seems as if valuable information is not being communicated, in terms of the benefits of -- what are the benefits and what are the costs of alternative, moving from Alternative 2 to Alternative 3.

MR. BEALL: Well I think like Bill said is that there's not much difference between 2 and 3 for cost, because the industry's going to be doing the same thing. It's just --

CHAIRMAN SCHULTZ: No, but that's -- but that's -- you're missing my point, which is -- which is there is a cost with what the industry is doing, and there is a benefit that happens to accrue to this particular issue, independent of the previous issue,

which was do you harden and vent and not, and how do you handle that overall approach.

We learned a lot from the evaluations that the staff and industry has done there. But here, there's another piece of information that could be valuable in communicating what you achieve by going from Alternative 2 to Alternative 3, and it also helps reinforce perhaps, if the decision is not to recommend, and that's what I see coming from here, not to recommend moving to Alternative 4.

It provides additional perspective on what has been achieved and what the cost was or it will be, and then compare that to the parameters of Alternative 4. You come away with a better understanding of the overall evaluation of all the alternatives.

It seems as if okay, now we're going to do this in retrospect, which is interesting. But we're going into a period of public comment, discussion, preparation for rulemaking and to postpone it to rulemaking leaves the decision-making information off the table at a very important time.

MR. BEALL: Well right now this is -- I guess this is the draft regulatory basis. So your comments taken here, we can incorporate that when the

final document comes out, okay.

CHAIRMAN SCHULTZ: Thank you.

MR. BEALL: In conclusion, right now the staff is proposing to go forward with Alternative 3, which is develop a proposed rulemaking to implement the protection of the BWR Mark I and Mark II containments, to make generically applicable the containment protection measures imposed by the Order EA-13-109, including the proposed implementation of the Phase 2 that uses external water addition.

The staff will not include in the rulemaking a requirement for engineered filters. The staff analysis determined that engineered filters do not represent a substantial increase in the overall protection of health and safety of the public. The Commission will be voting on the CPRR SECY paper, and will be writing direction to the staff later in 2015.

MEMBER BLEY: I've got a comment. Something's still bothering me from something you said earlier, and I'll wait 'til your later presentations to understand this better. But in some ways, thinking of an assumption or estimate of the failure to implement FLEX being fairly high, 40 percent, that might be thought of as conservative.

However, from what I'm hearing, what also

is providing the basis for which all our results saying the filters aren't any good or aren't helpful, is embedded there. If the FLEX failure were more like 1 in 100, which I know we're going to see people claim on some other aspects of it, I want to know what would happen to our calculations on the value of the filters.

MEMBER RAY: There's another part of that too.

CHAIRMAN SCHULTZ: Your button. Go green.

MEMBER RAY: Quite right. There's another part of that, which is the cost assumption associated with FLEX. How do you treat that as an increment, when you're counting it here? If you just say well there's no cost increase given that we're counting it here, because we've assumed only a 60 percent reliability, that's also a stretch, it seems to me, or at least that's something -- it's a new way to think about it.

MEMBER STETKAR: In truth, I've thought a little bit about this. In truth, that affects a lot of the stuff that I was calling the noise on that filter, on that figure. It affects a lot of the stuff on the noise. It doesn't have a big --

If you look at that figure and take their high level conservative estimate and increase it by a factor of one and a half or so, so you make it 1E to the minus 7-ish roughly, it's still an order of magnitude more or less below the quantitative health objective.

So even without that .6, which they factored in there, that comparison doesn't make any substantive difference. It would, however, down in the noise, and it would when you compare basically Alternative 3 versus Alternative 4, that the delta benefits from this certainly. And that's --

CHAIRMAN SCHULTZ: Got to think a little more.

MEMBER STETKAR: And that's a real concern.

MR. BEALL: Okay. That concludes my presentation. Now I'd like to have Marty talk about PRA evaluation.

MR. STUTZKE: Okay. Again, I'm Marty Stutzke from the Office of Research. I'm Marty Stutzke from the Office of Research, and I want to talk to you to summarize the results of the risk evaluation that's been done. To begin with, we can look at -- it's my Slide No. 3, which is the CPRR

alternatives.

The point of the slide is that the alternatives in the old SECY paper, SECY 12-0157 -- yeah, I believe the Committee my slides -- those alternatives are different than the alternatives in the new SECY paper 15-0085. So this table tries to map one to the other, and the reality is it's not a perfect fit like that.

But I kind of wanted to show you some of the implementation alternatives, which greatly affect the logic model and the quantification for them. For example, you had asked the question does it make a difference whether we inject water into the wetwell directly, or into the reactor pressure vessel? So we looked at those alternatives. Does it matter if we vent the drywell before we vent the wetwell? So that sort of thing.

Does it matter if venting is a manual action versus we put a rupture disk on it. Early in the project, we thought about a purely passive rupture disk on the drywell like this, say under the scenario, you know, perhaps everybody's incapacitated sort of thing. So select, let it go.

Later on, we introduced the notion, along with industry, of this what I'll call injection

control, the SAWA, the severe accident water addition that says hook the portable pump up and let it rip. Fill that containment full of water.

SAW-M says hook it up and control the flow, so that you do not submerge the wetwell vent, so that you always have that capability like that. Later on, then we worried about this vent operation under manual conditions, of a strategy that's known as open and leave open.

Once the vent's open, just leave it open, totally depressurize it, versus a vent cycling strategy that says reclose it after it boils down. I think we used 10 psi. That one's interesting. In one of the earlier versions of the EPG scenario. So back in '83 in the safety evaluation report, I found that vent cycling was one of the interesting things, because it was believed it would help retain radioactivity inside the containment better, like this.

Then we had already talked about filter capacity versus large versus small in size. So the result of this was that it generated 20 regulatory sub-alternatives, and you can find those in my backup slides starting on pages 20 and 21, that lists all the combinations that we went into to try to capture

them.

MEMBER STETKAR: Marty?

MR. STUTZKE: Yeah.

MEMBER STETKAR: I have no one to ask it, so I'll ask it now. Go back to your Slide 4. Slide 4. There we go. You included credit for FLEX water addition prior to core damage, did you not?

MR. STUTZKE: Yes, we did.

MEMBER STETKAR: And why did you do that?

MR. STUTZKE: Because we knew from the mitigation or beyond design basis strategies that FLEX would be like --

MEMBER STETKAR: No, because in mitigation and beyond design basis strategies, the industry has assumed that RCIC remains operating forever, and in NEI 13-02, the industry has stated that they only need to provide power for the FLEX water addition pathway into the reactor vessel prior to eight hours after the loss of all AC power.

So the industry cannot get water into the reactor vessel before core damage occurs. So why do you take credit for it, and how much difference does it make to your overall conclusions if you eliminate that possibility from the core damage event trees?

MR. STUTZKE: Well, I have a slide on the

delta that we can get to in just a minute. But --

MR. FULLER: May I respond to the question? Is this on?

CHAIRMAN SCHULTZ: Yes. Just your name please.

MR. FULLER: This is Ed Fuller from the Office of Research. In fact, if you look at the OIPs for the various reactions to Order EA-12-049, you find that they would be lining up to inject water at the time that RCIC would fail and furthermore, when the RCIC would fail, there is an action that they would probably be doing to reduce the pressure in the vessel, to the point where you can put water into it and prevent core damage.

And I believe a lot of that is reflected in Marty's PRA. Correct me if I'm wrong, Marty.

MR. STUTZKE: Oh absolutely.

MEMBER STETKAR: Marty does reflect all of that, Ed. I will give you quotes. I didn't want you to force me to do this, but in the current copy of NEI 13-02, Revision 1, dated April 20, 2015, and I won't quote the section numbers because they're long numbers, it says that "The following requirement applies to permanently installed portions of the severe accident water addition flow path. An ability

to provide motor force to any power or pneumatically operated valve from the water addition point to the reactor pressure vessel or containment before SAWA is needed to support use of the severe accident drywell vent or severe accident water management."

And in another part following on, it says "In lieu of Order Element 1.2.6, the motor force needed to establish the SAWA flow path is expected to be achieved in the same manner to meet NEI 12-06 guidance, compliance with Order EA-12-049, except that it needs to be established prior to eight hours, as indicated in Appendix I."

Appendix I says as long as you get it within eight hours, you're okay for the purposes of severe accident mitigation. Now that tells me that if I'm designing this thing, I can store all of my power recovery stuff maybe five, six, eight, ten miles from the site, as long as I can justify that I can get it there within eight hours.

I don't care what people say they'll be thinking about fixing to get ready to maybe kind of do something that might be sorta kinda good to do. They haven't committed to do anything other than what it says in writing in this guidance. It says "within eight hours." It doesn't say within the time before

we expect RCIC to fail.

MR. STUTZKE: Well if you do an analysis of this though, you find that in fact you most of the time have a lot more than eight hours before you're in trouble. There are certain scenarios where it is less than eight hours, by the way.

MEMBER STETKAR: Yeah, but my point is -- my point is that the industry is committing to be able to get water into the vessel to mitigate core damage. They are not committing to get water into the vessel to prevent core damage, and if the industry has a different commitment, I'd like to hear from the industry, where we've had this discussion.

They are committing to get water into the vessel to prevent core damage, I'm sorry, to mitigate core damage. I'm too emotionally involved here, and yet your analysis presumes that it is always available to prevent core damage, regardless of when RCIC fails and regardless of when DC power fails. They could fail at Time T zero.

MR. STUTZKE: That would be a truly flexible system. The logic structure takes no credit for FLEX for the first four hours, okay, and in fact that's one of the dominant failure mechanisms you see come out. If the RCIC pump fails, they're in this

Phase 1 stage. You're going accordingly.

MEMBER STETKAR: You're right, you're right.

MR. STUTZKE: And probabilistically, the difference between four and eight hours is not important. What fails RCIC is the earthquake. It fails on demand.

MEMBER STETKAR: Yeah, and the earthquake never fails FLEX does it?

MR. STUTZKE: No, that's not in there. But in other words, you know, the failure to run of the RCIC pump is the difference between four and eight hours. That's the point I'm trying to make.

Okay. Interesting pop-up message. So there are in order to implement or to look at the 20 sub-alternatives, there are a variety of core damage event trees and accident progression event trees to try to capture these sorts of things.

Looking at the ELAP frequencies, we've looked at the so-called internal initiating events, the classic plant-centered, switchyard-centered, grid related, weather-related LOOPs. Those weather-related LOOPs are failure only of the offsite power lines. They don't worry about collateral damage like turbine or tornado missiles going into the site, that

sort of thing, and seismic events like that.

I have in order to look at the reliability of the onsite emergency AC sources, I turned to NUREG CR 5500, which is updated annually by the Office of Research. This is a SPAR model that's been re-quantified. So it's not just the diesel reliability, but it's all the parts that support the diesels. It includes the common cause failures, things like that.

And it generates numbers for different site configurations, depending on the number of emergency AC sources, as well as the plants SBO coping durations.

MEMBER STETKAR: Marty, you really only have five SPAR models for the electric power systems.

MR. STUTZKE: Five in which way?

MEMBER STETKAR: Well, because they're only five different internal ELAP frequencies. You earlier said well, that was just based on I have some offsite power configuration and some number of diesels.

MR. STUTZKE: I believe every SPAR model will have the actual plant.

MEMBER STETKAR: Okay. Why didn't you use the actual SPAR models then?

MR. STUTZKE: Then I would be doing PRAs on every individual BWR Mark I.

MEMBER STETKAR: You used the seismic failures for every one?

MR. STUTZKE: I used the seismic hazard curve.

MEMBER STETKAR: Okay.

MR. STUTZKE: Not necessarily the seismic model.

MEMBER STETKAR: Okay.

MR. STUTZKE: I mean and it's deeper than the onsite emergency systems. The interesting part is that FLEX is in fact flexible, and every site, every licensee has a different way of implementing it. Actually, when we started this we built a little matrix, trying to get a feel for all the different ways that FLEX could be implemented.

So what you're looking at is a stylized, I won't say worst case or whatever, but it's a stylized plant, which is pretty typical in the way that we've done reg analysis in the past like that.

MEMBER STETKAR: Marty, why did the -- we dug into this quite a bit back August of last year, and I went and compared this stuff.

Why did the seismic ELAP frequencies

increase uniformly? They increased for every single site. They didn't increase uniformly for every site. The smallest increase was a factor of about 10, and the largest increase was almost a factor of 1,000. Why did that happen between August of last year and now?

MR. STUTZKE: Well, the one thing that happened following our meeting in August was that I did revise the seismic fragilities, based on your comments.

MEMBER STETKAR: Okay.

MR. STUTZKE: I had gone through IPEEE information and aggregated it together to create that. The other thing is to realize is that the seismic model assumes 100 percent coupling. So it no longer cares about the number of onsite sources. Failure of one is a failure of all of the sources, blackout.

MEMBER STETKAR: So wasn't that the case last year though?

MR. STUTZKE: That was the case last year.

MEMBER STETKAR: Okay. So that's the reason for the delta.

MR. STUTZKE: In other words, the 100

percent seismic coupling, the notion of seismic coupling is that pieces of similar equipment that have similar orientations, similar elevations in the plant that see similar loads, okay.

MEMBER STETKAR: Oh okay, okay.

MR. STUTZKE: Okay. So if one fails --

MEMBER STETKAR: They all fail.

MR. STUTZKE: That's the assumption. Now people will argue in seismic PRA well it's only 80 percent coupled or 60 percent, and I don't have that information available to us. So I just assumed if one failed, they all fail.

MEMBER STETKAR: But again, that's not a reason for the difference between August and now, because that same assumption is in both, right?

MR. STUTZKE: That is correct.

MEMBER STETKAR: So was it simply the fragilities that made that difference?

MR. STUTZKE: That's what I would have -- I would have to look into to confirm.

MEMBER STETKAR: Because I'm -- as I said, the smallest one is about a factor of 10. The biggest and it's a real outlier is about a factor of 1,000. Most of them are a delta of about a factor of roughly 20 to 50 times.

I mean that's a pretty large change, and it also changes the entire notion, because in August, most of the ELAP frequencies were driven by internal events, and now they're by far dominated 95 percent or more for most sites from seismic events.

MR. STUTZKE: Well, we can look at it, look at the next slide. This is the breakdown of contributions.

MEMBER CORRADINI: Where is the flooding?

MEMBER STETKAR: Flooding isn't there.

MEMBER CORRADINI: At all?

MR. STUTZKE: There's no consideration of floods.

MEMBER STETKAR: Fires are not there, flooding is not there.

MR. STUTZKE: I've got probabilistic flood hazard assessment, to really go after floods, plus the locations.

MEMBER BLEY: There's a couple where it's the other way, but not many. Oyster Creek I think --

MEMBER STETKAR: There's two or three where the internal events are still higher.

MR. STUTZKE: I think actually -- well,

I've gotta find my notes. There were, I think, four if I recall. But where they're higher, they're fairly comparable. The smallest seismic is about 35 percent, and that is in fact for Nine Mile, where the internal is about 7 minus 6 and the seismic is about 4 minus 6.

MEMBER STETKAR: The reason I make this point is that the message now -- regardless of the reason for the change, this is now going forward, the characterization of the contributors to the ELAP frequency, which honestly to me intuitively sounds better than what we saw a year ago.

But it's important to note that for most sites, by and large these events are being driven by earthquakes that are beyond the design basis for that site, and you do have some sensitivity -- well, you have results that show what fraction for each of the seismic bins, and most of them come from .3 to about point, I don't know, seven, .75 G. But that's beyond the design basis for I believe all of these sites.

MR. STUTZKE: That's correct.

MEMBER STETKAR: I want to make that point later when we get to the MACCS analyses.

MR. STUTZKE: Yeah. I am interested to see when we complete the seismic reevaluations, how

close these estimates are. It doesn't matter. Okay. This will be a difficult slide for us to get through.

We've had difficulties in assessing human error probabilities from August. You remember at that time, our intent was to do a more detailed human reliability analysis, to get a better sense of the actual human error probabilities like this, and came ultimately to the conclusion that the best we could do within our current schedule and our current budget was to stay with the scoping human error probabilities like that.

I will point out that the alternatives for the various strategies are conceptual. They're not currently in the EPG SAGs or in the licensee's training programs. We had approached industry to see if we could go and talk to them and were rebuffed, for this reason.

MEMBER CORRADINI: Say that again?

MR. STUTZKE: In other words, what point is there to talk to an operator about how he would operate a hypothetical system? So everything being at the conceptual design stage, it's difficult to get a feel, you know, and the classic HRA method, to go do the task analysis, talk to the operators, come up with some sense of how well they're trained, how good

the procedural basis are, that sort of thing.

None of that exists in this study, because CPRR does not implement it, okay. To make it worse, as I pointed out, FLEX implementation is flexible. So you can't go to the one site. You go to multiple sites and get a cast of different viewpoints on how to do this.

Even if we had that information available to us, the fact of the matter is we don't have a very good staff-developed or any staff-developed HRA method that addresses post-core damage accidents like this. Our go-to methodology tends to be SPAR-H. It by itself has, we'll say has been soundly criticized by certain point as to what it does and it doesn't do.

But its use has been primarily towards and control them prior to core damage types of actions like this. I would point out previous subcommittee meetings, one on the Site Level 3, where we talked about the -- the staff talked about their plans for doing the Level 2 portion of that.

They've assigned screening values that are very representative of the scoping values that I've used in here, the same order of magnitude sort of thing like that. We had originally intended in

the CPRR analysis to use IDHEAS or an expansion of IDHEAS, which is now called the general methodology.

Just several months ago, those methodologies were presented and it's clear that they're not ready to go to support this type of regulatory decision. So staying with that, I felt that it was more direct to say, you know, I've assumed these scoping error probabilities.

I will point out they're not screening. Screening implies that we're using them to distinguish one from another like that. There is no screening in the quantification technique. There's no truncation frequency, there's nothing. So the different probabilities affect the, if you will, the split fractions as to what MELCOR and MACCS bins ultimately the results match too.

MEMBER STETKAR: Yeah, which affect the overall results of this study.

MR. STUTZKE: Yeah.

MEMBER STETKAR: Now --

MR. STUTZKE: Well, the other thing is the amount of effort one should put into the analysis needs to be commensurate with its importance in the decision-making process. So I did sensitivity studies that I have --

MEMBER STETKAR: Marty, if you -- before you get to the sensitivity studies, let me just say something. I read everything that you wrote. There's a lot of "well, we don't have methods; we don't know how to do this. We don't -- it's too difficult. It's very."

But there was no even -- there was not even an attempt made to say that under conditions, for example, where I have to go locally in the plant, with no DC power available, after a very, very large earthquake I used the same .3 in my models for core damage, which are procedure-driven, Level 1 PRA type actions, and I used the same .3 for post-core damage, no DC power available out in the plant.

One would think that it ought to be higher. I don't know whether it's a factor of 2.7625 precisely higher, but there was not even an attempt to say well, in these cases we think it ought to be higher, so we'll use a higher scoping value. They're all the same.

MEMBER BLEY: I'd chime in just two things. You know, I think the qualitative nature of things John just described seem like the sorts of things one could do. From my own experience, when you get to a spot where there aren't accepted

procedures and training and all of that as yet, there is one method that's set up to deal with that, and there are a lot of problems with it.

It was the old HEART method. It's now been extended to something called NARA, which isn't fully available. But there are enough papers on it that you can actually gin it out. But it matches the kind of qualitative descriptive stuff John just went through, and that's why I'd say yeah, it seems that some accounting for where things are clearly going to be tougher would be reasonable.

MEMBER REMPE: The other thing I was struggling with is it seems like with the water management, that there's a need to rely on a lot of instrumentation, and it's not clear to me that that instrumentation would be there. I was -- it just seems like it's kind of a pretty high success assumption, and I just was wondering, I mean you're assuming that instrumentation's going to work in this type of analysis, right?

MR. STUTZKE: That's correct.

MEMBER REMPE: And so that -- I'm not clear what actions are being taken to ensure that that assumption's valid.

MR. STUTZKE: It's a post-TMI regulation

that says --

(Simultaneous speaking.)

MEMBER STETKAR: With no DC power, with no DC power available.

MR. STUTZKE: This says it will be available.

MEMBER REMPE: And what is being done to ensure that with that post-TMI regulation? I did see that comment in the MELCOR or in the document, and what actions are being done to ensure that at this time?

MR. STUTZKE: I imagine it's a compliance action.

MEMBER REMPE: Is that being done right now? All of those water level measurements are being inspected for those type of conditions?

MR. STUTZKE: That's my understanding. But I would defer to NRR.

MR. RECKLEY: Yeah. This is Bill Reckley again. For those licensees that would be pursuing severe accident water management, that would then fall under the scope of the Order.

So they would have to describe in their overall integrated plan for the water, the instrumentation they're going to use, and then the

NEI guidance document that we endorsed in there and staff guidance calls for the requirements that would be placed on all of the equipment for severe accident water management, including the instrumentation.

MEMBER REMPE: So there will be additional regulation, because right now what's out there in the regulation just says "an accident with core damage." It doesn't give any details about --

MR. RECKLEY: Well, we would look at the specifics, but I don't want to oversell this. This is largely going to be Reg Guide 1.97 instrumentation, and that's what the guidance calls for; that's what the staff accepted so --

MEMBER REMPE: Okay.

MR. STUTZKE: Okay. So one way to look at or to get a sense of the operator actions, this is sensitivity of core damage frequency to the in control room and out of control rooms. You'll see there's three circles on the graph. The one in the middle is using the so-called scoping numbers; the one on the lower right corner is taking no credit whatsoever. So that says ELAP occurs and core damage occurs with probability 1.

The one in the upper left corner takes full credit for the operator. So now you're looking

only at equipment reliability types of issues like this. What is interesting about this, it took me a while to put my finger on what's interesting about it, is you don't see a large variation from the highest number to the lowest number.

No credit to the operators is roughly 2 times 10 to the minus 5; full credit for the operators is 5 times 10 to the minus 6. Why? The hardware reliability is not so reliable. It's seismic failures of other equipment that you need to be operable like this, like the seismic failure of the RCIC pump, etcetera, etcetera, like this. So --

MEMBER CORRADINI: So can I say that a different way, just so I -- so you're saying that even if the operators were perfectly reliable, there's nothing there to operate?

MR. STUTZKE: That's correct.

MEMBER CORRADINI: Is that what you're saying?

MR. STUTZKE: That's correct.

MEMBER CORRADINI: The instrumentation, I think, is a red herring, because the instrumentation you need for SAWA is in a cool, moist area and it was cool and moist before and after.

MEMBER STETKAR: I'm sorry. This has

nothing to do with SAWA. Be careful.

MEMBER CORRADINI: John, don't go in that direction yet, quite yet. Let me ask you, Marty. This is really compelling in the context of core damage frequency. The core damage frequency is not the relevant measure of merit for everything that we're talking about today.

MR. STUTZKE: I understand.

MEMBER STETKAR: Is the release category frequencies. Why didn't you do a similar evaluation for release category frequencies, so we could understand the sensitivity of the operator actions through the most important release categories that contribute to ILCF? And now you can talk about SAWA, because SAWA affects the release categories --

MEMBER CORRADINI: Because it's post-damage?

MEMBER STETKAR: Because it's post-damage.

MEMBER CORRADINI: Okay.

MEMBER STETKAR: I wanted to get to that point, because this is -- this is nice within the context of core damage. The reason you don't see a big difference is the conditional core damage probability is about .47. So you only have roughly a

factor of 2 delta between perfectly good and perfectly failed operator actions anyway, which is what this shows.

MR. STUTZKE: Flip to the comparison. I tried to --

MEMBER STETKAR: But again, this is core damage.

MR. STUTZKE: This is core damage. But let me try to explain the --

MEMBER STETKAR: Why didn't you do it from release categories?

MEMBER CORRADINI: Or did you?

MEMBER STETKAR: Or did you?

MR. STUTZKE: No, I didn't, and the answer is technically it's a little involved, because of the plant damage states. Look at the comparison of core damage frequencies, because it tries to explain this notion of how much effort is necessary to reach a regulatory decision.

Having come back from the August meeting, we all realize that we needed to do something about the human reliability, like what could be done in the time frame that we were working on. So the first effort at a high level conservative effort said you know, if I just take the maximum ELAP frequency and

multiply it times the biggest conditional risk that we got out of MACCS, I'm clearly below the safety goal, okay? I won't tell you what I was saying at the time I was doing this, but I was thinking of you John.

Then it's like okay look, I've bounded it, okay. I've absolutely bounded it. Then we went through and we did some more MACCS work to raise that frequency, because the numbers that I originally were using were assuming evacuation, etcetera, etcetera. So suppose people just sit there and do nothing, and we came up with the new conditional consequence numbers like that.

Some discussion went by. We wanted to credit FLEX in some way like this, and so the 60 percent success probability number evolved. That generates the second dot, the red dot on there, okay, and what that dot represents is not just the frequency of core damage, but the frequency of release.

So it's like a PRA that has only one release category, but that frequency and this high conditional consequence coming up, that's the basis of the conservative high level estimate. In contrast, look at the yellow box in there. The baseline using these scoping HEPs numbers for core

damage frequency is sitting at about 9 times 10 to the minus 6.

From the sensitivity study, it could be a size about 2 times 10 to the minus 5, or 4 times 10 to the minus 5 on this. I would point out that when we reach these regulatory decisions, we're not supposed to pick the maximum; we're supposed to use the fleet average. We understand some sites have worst weather, and that explains the difference between the baseline CDF and this 7.4 max ELAP frequency like this.

Then for comparison, to try to segregate all the noise, so to speak, was that I plotted the parametric uncertainty distribution for the baseline fleet average CDF, and that's shown on the kind of orange box bar graph, and the mean values were reasonably close. The upper percentile is comparable to not crediting the operators at all, etcetera, etcetera.

The intent is to show you that the high level conservative estimate is in fact conservative, based on margin.

MEMBER STETKAR: Marty, for the record, because not all of the numbers may have dug into the details that I did. It came up earlier, with this

assumption of .4 failure, .6 success for FLEX, which explicitly is included in what is shown as the red dot on this figure; is that correct?

MR. STUTZKE: That's correct.

MEMBER STETKAR: That .4 and .6 is not included in what's shown in the yellow box or what's shown to the left; is that correct? Because those boxes include actual failure rates estimated for the FLEX pump, the flow paths and the associated .1 to .3 human error probabilities. But the overall FLEX, I didn't go try to quantify it. But it wasn't assumed to fail .400 percent of the time?

MR. STUTZKE: No, no.

MEMBER STETKAR: In the yellow or the left-hand side?

MR. STUTZKE: I mean the .4 basically comes from taking the baseline of 8.9 times 10 to the minus 6, and divide it by the 1.9 times 10 to the minus 5, and you get roughly 60 percent.

MEMBER STETKAR: But it was -- that .4 for the purposes of that red dot or that conservative high estimate evaluation essentially was derived as mixed value with human error probabilities, plus the hardware probabilities from the detailed analysis.

MR. STUTZKE: That is correct.

MEMBER STETKAR: Okay, thanks.

MR. STUTZKE: That is correct. Another draft that's interesting is to identify significant contributors to "damage," and again that's only been done for core damage, not all the way out to release, is to look at identification of significant contributors by using risk achievement worth, like the ratio. If you're saying the components totally failed divided by the baseline CDF, and on the Y axis is the plot plus the vessel importance or the percent contribution to the overall risk.

Generally, we say a component or an operator action is important if it's raw values figured in two, or its vessel is bigger than 0.005. So you end with this quadrant system, and I've tried to indicate what is important to write down. Again, what you notice is the raw values don't get above about three or so, because the reliability is already set.

There's not much room to move there with that. None of the operator actions from the baseline and in some of the baseline numbers are -- show up as being significant with respect to raw for that reason, again because they're already set reasonably high. So it gave me some confidence that I was there.

Now what you do see as important are the seismic failures of the diesel, seismic failures of the batteries, things like that.

MEMBER STETKAR: I mentioned it before, but it's worth asking. Why is the FLEX equipment guaranteed to survive any possible earthquake? In particular, the structures that has the FLEX equipment.

MR. STUTZKE: I can't answer that.

MEMBER STETKAR: Okay.

MR. STUTZKE: Okay. Looking at the alternatives, again the sub-alternatives actually have this rather cryptic notation you'll see going along the X axis there. So we talk about Alternatives 1 and 2 alpha, 3 alpha, etcetera, etcetera. That labeling scheme actually got changed midway during our project. So I've tried to label the draft to give you an idea of the different conditions.

Let me walk you through that. Status quo is just the order is what it is. Alternative 2A is to codify the existing order like this. Coming down there, then we look at combinations of severe accident water addition or severe accident water management. We can look at open and leave open

venting versus open -- versus them cycling like this.

The arrows at the top indicate the point of injection, either directly to the reactor vessel or into the drywell like this, and then overall wetwell forced venting strategy versus drywell forced venting strategy. The bars that are shown in crosshatch are all the filtered cases, and so you will always see them in pairs. So either a decontamination factor of 10 or 1,000 like that.

MEMBER CORRADINI: Can I ask just -- so this is maybe something I actually understand. So the 4Bi(1) and the 4Ci(1) are water addition and associated management, and then wetwell venting through the hardened vent?

MR. STUTZKE: Yes.

MEMBER CORRADINI: And the difference between those crosshatched ones and the ones that are not crosshatched to the left is what?

MR. STUTZKE: No filter.

MEMBER CORRADINI: No filter.

MR. STUTZKE: Every crosshatch bar assumes the presence of a filter.

MEMBER CORRADINI: Oh the drywell filter?

MR. STUTZKE: Or a wetwell filter --

MEMBER CORRADINI: Okay, okay. Either external to the wetwell itself --

MR. STUTZKE: An engineered filter, correct.

MEMBER CORRADINI: An engineered filter, okay. That's what I didn't understand.

MR. STUTZKE: So you see when the analysis was done, that the status quo assumes the hardened vent system in place, but it does not assume there's any water addition, because it's not explicitly required by the Order like this. So you can see a benefit or reduction in the risk just because of the water addition to come in. That's because it's preventing liner melt-through like this.

Then you see another further drop when you get over to 3-47 down there. You see another reduction because of the presence of the filters.

MEMBER CORRADINI: But there is a filtering action by the wetwell itself, which is --

MR. STUTZKE: Of course.

MEMBER CORRADINI: Which is modeled and buried inside of the 3A-4A(1), 4A(2), etcetera?

MR. STUTZKE: Correct.

MEMBER CORRADINI: And what is that? That's variable depending upon temperature? That's

actually physics-based?

MR. STUTZKE: Yes.

MEMBER CORRADINI: Sorry, once in a while --

MR. STUTZKE: Yes. That's part of the MELCOR work.

MEMBER CORRADINI: Okay.

MR. STUTZKE: The other thing you need bear in mind is that the logic structure assumes that with respect to over pressure protection, the wetwell vent and the drywell vent are equal. So they're redundant systems, and so the event tree will ask oh, the wetwell vent didn't open; try the drywell vent.

Now there is filtering prior to core damage through the drywell vent, right. The discharge path prior to core damage is through the safety relief valves.

MEMBER STETKAR: What wetwell? Filtering through the wetwell vent prior to core damage.

MR. STUTZKE: Well, and also true for the drywell, because the discharge path is from the core through the safety relief valves into the pool. Then either the wetwell vent is opened to relieve the over pressure, or the drywell vent is opened and it comes

back through the SAM lines.

So you see once core damage has occurred, you don't get the benefits if you're using the drywell vent.

MEMBER CORRADINI: Say that again? I'm sorry again? Just repeat that.

MR. STUTZKE: Once core damage has occurred and specifically once the vessel is breached, there is a big difference between using the wetwell versus the drywell.

MEMBER CORRADINI: Okay. One of the things, by the way, when I read this, I try to read both the SECY paper and the regulatory basis as an outsider, in my opinion, the effects of the filtration don't come across to a general reader. In other words, Alternative 4 is characterized as the release reduction alternative, as if the only way you get reduction and release is from putting a filter on the vent.

Whereas indeed Alternative 3 in fact gets you a lot of that, and that really doesn't come across. It comes across if you know all of this detail stuff, if you can sort through the alphabet soup and all of the detailed analyses. But it really -- at a high level, it really doesn't come across, at

least it didn't to me.

MEMBER BLEY: It comes across if you already have it in your head, I guess.

MEMBER STETKAR: Well yeah. I mean once I knew what all of this meant, I knew that. But at the high level, at the SECY paper level, where the alternatives are presented and the reasons for why you don't get a lot of additional benefit from adding the filter, Alternative 4, is not readily apparent. At least it wasn't to me. You may just want to rethink that.

MEMBER CORRADINI: So can I just make sure I understand. So we're now back to stick with the wetwell first venting strategy and the crosshatch is with an external filter and no crosshatches is not. So the fleshy-colored stuff is with and without, and the difference of the 40 percent or 30 percent difference in the risk is because I'm still overpressurizing and leaking through the drywell head?

What is the reason that I only get that little of a reduction? Hossein said you were going to explain this, so this is your chance. Sorry.

MR. STUTZKE: Okay. What you're looking at is, of course, the risk over all possible -- well,

a large number of accident scenarios, and there are scenarios here where no venting occurs, either because the operator fails to make the vent like this.

MEMBER STETKAR: Or there's no water because they failed to do that.

MR. STUTZKE: There's no water to do that. And so, you know, it's the sum of the frequency of all those sequences times their respective consequences added up that generates this risk number.

MEMBER STETKAR: So if I were to take -- so okay. So I thought that's what you were going to tell me. So now if I were to take the question that Dennis asked, where you assumed .1 and .3, and I would have said that geez, .3 seems kind of low. It's a coin toss. It's whether or not the operator's going to do things right under these extreme conditions, how would that difference qualitatively?

Would the spread get bigger or get smaller? Meaning the crosshatched and the not crosshatched.

MR. STUTZKE: If I take no credit at all for the operator post-core damage, right, it's going to look like the solid black lines or worse.

MEMBER STETKAR: But if I --

MR. STUTZKE: Right. No water addition post-core damage.

MEMBER STETKAR: But if I at least admit qualitatively, the chances of the operator doing the right thing becomes less possible, less probable as I get more severe environments. Is the difference between -- I'm looking at the difference between the crosshatched and not crosshatched. Do they get bigger or smaller?

MR. STUTZKE: It will get smaller.

MEMBER STETKAR: Okay.

MR. STUTZKE: Because the crosshatched --

MEMBER STETKAR: That's what I thought.

MR. STUTZKE: Right. Everything will approach the left-hand side of the figure eventually.

MEMBER BROWN: Can I ask a question?

MR. STUTZKE: Yes.

MEMBER BROWN: I mean as I look at this, I'm trying to contrast the EPRI analysis tech basis with your presentation, and if you'll look at the table that they provided of all their 24 cases, the difference between the water management, water addition, venting, cycling, whatever and the filtered versions were almost negligibly different, whereas

your changes from the management to the filtering is not negligibly different. It's a .4 or .5 times 10 to the minus 9th up and down.

Did you -- I was just -- all I did was wonder why in the world your analysis shows some bigger effect, if the filtering were, as their analysis shows, virtually no effect in the analysis, in their analysis? I'm looking at their Table 3.49. We're talking about -- I'm looking at latent cancer. I'm trying to look at the data column. It's the same as your board one.

MR. ESMAILI: Let me try it and see whether that answers your question. I think one of the differences between our analysis and the industry's analysis is that we do, and I have it in one of my backup slides, is that, as Marty said, we do vent before lower head failure, before we actually start injecting.

And so if you actually had a filter, you can filter dose. So there is some benefit to filtering even without water, because you are venting before you inject water. So I think this is reflected in the differences between the cases where I think the hashed and solid cases, correct?

MEMBER BROWN: Yeah. I'm just thinking

between the orange -- with the pink, and I'm just looking at the, what is it, 4A, 3A, 4A(1), ii(1), iii(1), etcetera, and then the two filter cases there. I'm just looking at the water management versus whatever it is plus the filters after that.

So that's why I was trying to get a -- I'm not sure I understand that answer about before or after.

MR. ESMAILI: Well, you know, unavailability of the FLEX is reflected in all of these.

MEMBER BROWN: Yeah.

MR. ESMAILI: Okay, so in all of these bars that you see, even though it says SAWA, that doesn't mean that water injection or water addition is available.

MEMBER BROWN: Guaranteed.

MR. ESMAILI: Is guaranteed. So when you go from the -- that's pink. Let's focus on the RPV injection. So when you go from those four bars, those four bars have some contribution from the fact that you have a liner melt-through and you have an uncontrolled release, as I was saying before.

When you go to the filtered one, the hash lines, okay, some of that -- some of the release comes

from going to the vent, and some of it comes from going through the, you know, containment failure. The one that comes from the containment failure you cannot, you know, you cannot filter. You are not -- you don't have to filter at the end of that because it's just released.

The one that comes from the containment venting you can filter. So that difference, that difference, which is not much because, you know, 50 percent of the time in your 40 or 60 year, you know, your FLEX is not there, to put water in there. So you are going to get a liner melt-through.

So that difference is because you can filter some of this stuff, because the filtering is occurring at the time of core damage before lower head failure. But the fact that the FLEX was unavailable, okay, that difference is not great, and I think we had the question before, you know.

That difference if putting the filter is not that great, for the same reason that the filtering from 10 to 1,000 is not that great, because you always have that contribution that you cannot filter. It's going through the liner, melt-through to the upper head or what have you.

MEMBER BLEY: Let me try something Marty,

and I may be completely wrong on this. I thought in the EPRI studies, they were doing essentially the studies say what happens if you have a vent and it works, if you have a filter and it works, so that these numbers include the chance of equipment failure and of operator failure, albeit stylized to some extent.

I thought their plots were assuming everything worked. Is that wrong?

MR. STUTZKE: They actually have an event tree structure.

MEMBER BLEY: Okay. They do the same thing, okay.

MALE PARTICIPANT: They have a what structure?

MEMBER BLEY: But with different numbers.

MR. STUTZKE: An event tree. They have a logic and all that structure.

MEMBER BROWN: Yeah. Well they went through, what do they call it, an event tree structure for both, whatever. I've forgotten what the APET and then the core damage eventually, and then the accident progression eventually, to get this window down. I'll forget it next week. It works for today.

CHAIRMAN SCHULTZ: Charlie, we could wait and hear from the industry.

(Simultaneous speaking.)

MEMBER BROWN: That's fine. I just don't understand. That just seemed to be almost a wash in the EPRI analyses, as opposed to what --

CHAIRMAN SCHULTZ: That's the same question.

MEMBER BROWN: Okay.

MEMBER STETKAR: I mean these are not very big differences.

MEMBER BROWN: I know that, but there was -- they're even smaller in the EPRI analysis, in terms of going into the filters, between the water management and the filtered sections.

MEMBER BLEY: Were there linear scales over there too? These are linear scales.

MEMBER STETKAR: These are linear scales. I mean the difference between 1.8 and 1.25 --

MEMBER BROWN: I know how to read the graphs.

MEMBER STETKAR: Okay.

MEMBER BROWN: There's a fractional difference --

(Simultaneous speaking.)

MEMBER BROWN: It's not an order of magnitude, and I thought that the EPRI was sort of showing about the same difference. I'm just looking at the same type of numbers and the spread is still bigger than they are -- on this chart than they are in their table.

MEMBER STETKAR: Which table?

MEMBER BROWN: 3.49, 3-49.

MEMBER STETKAR: 3-49?

MEMBER BROWN: On page 89.

MEMBER REMPE: 3-47.

MEMBER BROWN: On 89. No, no.

MEMBER STETKAR: Okay. That's fine.

MEMBER BROWN: 3-49. I probably don't have my glasses on, 49.

(Simultaneous speaking.)

MEMBER BROWN: I'm looking at the MACCS table of their summary of all their --

(Off mic comments.)

MEMBER BROWN: It looked like a wash. I mean to me this looks like a feed and bleed event. You feed and bleed on to keep it cool and prevent hot stuff from getting out.

MR. ESMAILI: This might help. I think

our differences are a little bit more pronounced in the industry, and because of precisely what I told you is that we do have venting at some point before lower head failure, where the industry on the other hand, I will defer to the industry rep.

MEMBER BROWN: But that point, that nuance I didn't get out of other.

MR. ESMAILI: So in other words, even if you have no water, even the cases that are dry, no water, because you are venting at some point, you can filter that release.

MR. STUTZKE: Okay, we've got to move on. Okay. The next graph is showing you conditional containment failure probabilities for the various cases like this, where I've defined containment failure. These are the red bars. So it's either over pressure failure or liner melt-through. Realize there are sequences in the risk evaluation where both failure mechanisms have occurred, over pressure followed by liner melt-through like this.

I say it's lowest for the cases where you have RPV injection like this. Moving on to the next one is where does the actual core debris end up? It's another dimension. Looking at it, the drain being in the vessel itself like this. Again, for RPV

injection you have that option. For drywell injection, there's no chance of retaining debris inside the reactor vessel because it's not cooled there. It's being cooled out in the drywell like this, so forth and so on.

Then something that's been new since our August meeting is the parametric uncertainty analysis, which I guess on a personal note the math is really fun, but I don't know if it's worth the effort like this, but we can talk about it like that. But I've tried to include the uncertainty into the seismic hazard curve.

So it actually picks a random seismic hazard curve according to the various fractals that have been provided like this. It goes through the seismic fragility evaluation, all of that, the ELAP frequencies, human error probabilities done with constrained non-informative prior distributions to really flatten them out as much as I possibly could.

Last and not least, a simple attempt, I'm just going to call it crude and it probably is, to look at the conditional consequences. That is information that I was able to glean from SOARCA in their uncertainty work like this. So they worry about all the various parameters inside the MACCS

models, etcetera, etcetera.

It's all rolled up into a single basically log normal type bound with a conditional consequence.

MEMBER SKILLMAN: Marty, I would like to ask about the human error probabilities. About a half an hour ago, 45 minutes ago you mentioned that this was a challenge, trying to really sort through the HEP and HRA, and you assigned, it appeared to be one chance in ten making an error in the control room, three chances in ten outside the control room.

My question is, is there other information available regarding human error probabilities, either from the military or from industry, where individuals have watched behavior where there is a highly refined or highly proceduralized process, and people make decisions and make errors or don't make errors.

In other cases, when people are off script, how they intuitively behave in making errors. I'm just wondering, is there other information that is available that would be useful?

MR. STUTZKE: The answer is yes, there is. In fact, you know, people I guess, HRA analysts have tried to encapsulate this type of information as

long as I've been in the industry, which is some 35 years like that. In my view, the issue comes down to fidelity of the simulation.

For example, I think it was back in the 80's where they did the simulated exercises, and you tried to observe people's behavior in the control room. Again, those are highly proceduralized actions, etcetera, etcetera. What we're dealing with here is something that's not so easily simulated like this. You know, I would throw in back in my Naval career, I used to train operators, and it's very hard to simulate fires, things like this, the actual conditions that people find themselves in like that.

But that's probably as far as I want to go. Obviously, I'm not an HRA expert. Danny's got people. Yes, they've tried to do this. We're dealing here, we're trying to assess the probability of the operator to react to events that don't occur every often. In fact, fortunately not.

MEMBER BALLINGER: What about aircraft simulators nowadays? Those are very, very sophisticated. They're able to simulate every abnormal, usual conditions.

MR. STUTZKE: Most plant simulators, to my knowledge, stop to core damage.

MEMBER BALLINGER: I'm talking about aircraft.

MR. STUTZKE: But I understand that.

MEMBER BALLINGER: Yeah.

MEMBER CORRADINI: But they don't simulate just to put the plane home. I've seen these. You go to Boeing. They don't simulate two engines stopping. They simulate one engine stopping and you train for that. When the two engines stop, it's a very bad day.

(Off mic comments.)

MEMBER BALLINGER: Not military, commercial.

MR. STUTZKE: Sure. It's not that it's unsimulable, if that's a word, can't be simulated. It's just nobody has, you know, and again, how would you simulate the entire plant? Most of these FLEX actions, for example, rigging the portable pump is to carry a very heavy piece of equipment from the storage warehouse across the yard, connecting the hose up. How do you simulate that?

MALE PARTICIPANT: You do it.

MEMBER BLEY: And when you walk through it, you do it right.

MR. STUTZKE: You can walk through it,

sure, and the problem with walk-through is they're not time stressed necessarily.

MEMBER BLEY: Or any other stressed, yeah.

MR. STUTZKE: Right, and there's lots of people to help you. The other thing that I would point out, that makes this somewhat of a challenge is we don't worry, or at least we don't attempt to quantify accessibility to where the actions need to occur.

I mean one of my favorite points is what's the fragility of the stairwell or the garage door that has to be opened? What if those things can't be opened? Now where do they go? And we don't have much information on this type of scenario. So to try to get a little bit back on schedule --

MEMBER STETKAR: Marty just, you said something that I actually didn't get out of reading the report. You said that you did try to characterize uncertainty in the conditional consequence evaluations that's folded into this plot; is that correct?

MR. STUTZKE: That's correct.

MEMBER STETKAR: That really doesn't come -- you should make sure that that comes out.

Regardless of how approximate you may think it is, because reading everything that I did, I thought that these uncertainty bounds were limited to only the seismic hazard and the propagation of parametric uncertainties through CDET and APET models.

MR. STUTZKE: Right, right.

MEMBER STETKAR: So what -- and I think that would help people, because one of my -- one of my comments reading through this was well, this doesn't seem to account at all for uncertainties in the consequence analysis, I think.

MR. STUTZKE: It does, in a simple way. We tried.

MEMBER STETKAR: But even just saying that it does is important.

MR. STUTZKE: Yeah, yeah. Okay. So when you look at the results, the other thing that doesn't show up in what you've seen yet, and by the way, I should point out the staff's current intention was to write a NUREG to document all of the analyses that would be done, as well as to provide some sort of a historical perspective, knowledge management on containment venting. We've found interesting history since that.

But one of the things, to return to this

uncertainty work is I mean you can see the uncertainty bands propagated through like this. I selectively turned off the uncertainty to find out what's driving the answer, and the answer is the seismic hazard curves.

MEMBER STETKAR: Sure.

MR. STUTZKE: I mean by a long shot. In other words, I have the ability just to use the mean hazard curve and so I suppose the only uncertainty is in the RCIC pump failed to run number. What does that buy me? The answer is --

MEMBER STETKAR: Unless you used a huge uncertainty distribution for the condition consequences.

MR. STUTZKE: Yes.

MEMBER BLEY: And just to say, and that's true for even the handful of cases where the risk was pretty balanced between -- certainly yeah.

MR. STUTZKE: That's correct.

MEMBER BALLINGER: This does not include anything related to flooding, right?

MR. STUTZKE: That's correct.

MEMBER BALLINGER: So if the seismic component is dominant, what do you think in your judgment the flooding hazard component is going to

be?

MR. STUTZKE: Well, I don't know. I mean some sites are going to be more vulnerable to floods. In coastal locations, I worry about the hurricanes and things like that. My personal bugaboo, having grown up in the upper south, is tornadoes, and we don't look at those either very well.

But you know, it's fair to say, I mean it's not going to raise the risk by a factor of 10, just my judgment. So to wrap up a little bit --

MEMBER STETKAR: Marty, before you get to this. One last thing, and I said I'd -- I mentioned it early. I actually -- I understand this plot, right, and I kind of like this plot. I didn't understand that the uncertainties also included the conditional consequences, which I already mentioned.

What confused me and continues to confuse me is when this plot is superimposed on the other thing that is characterized as the whatever you call it, the extremely conservative --

MR. STUTZKE: I call it conservative.

MEMBER STETKAR: Because I don't know how, I still don't know how to relate that to this, and putting them on the same plot together forces one to try to understand how they relate to one another.

I think I understand how it is, but there's a danger of people trying to infer why these numbers are a factor of 30 below that high value and what that means.

It confuses this .6 successive FLEX with what is the actual value of FLEX in each of these things. I just found it really, really confusing. It's just a way of presenting the results. I personally would have kept those plots separate and not superimpose them, because I think they're trying to tell different stories.

MR. STUTZKE: Yeah.

MEMBER STETKAR: That's just a comment.

MR. STUTZKE: Yeah, I understand. It's -- well the fact is it is confusing at certain --

MEMBER STETKAR: It isn't if you know all of the details that went into every one of the calculations. But very few people who read the SECY paper, in fact I believe very few people who read the draft regulatory basis -- in fact, just reading the draft regulatory basis, you can't understand that level of detail anyway.

You have to dig into the models that will eventually be documented in the NUREG, and that's a heck of a lot of detail for people who are trying to

understand it at a fairly informed high level, the technical reasons for why I'm seeing differences in these plots.

MR. STUTZKE: Yeah. I agree. We can always improve the communication of the answer.

MEMBER STETKAR: Well, but some of this stuff is really important, especially if you go out for public comments. The communication is in many cases more important than the minutiae of the technical details.

MR. STUTZKE: Okay. Last slide. Some insights from this. In general, the water addition and the venting are effective strategies. You can demonstrate that the risk is reduced; that you preserve the containment structural integrity. The venting alone is needed to prevent the overpressurization mode. The liner melt-throughs is not only venting but water addition.

So you've got to have both, right, because liner melt-through is at decay heat removal. So the water comes in and cools it. It steams off and you've got to vent it to rid of it like that. To get really important risk reductions, you need both strategies to work like this.

As we talked before, engineered filters

can be effective in reducing the risk, but if liner melt-through occurs, part of the release becomes unfilterable. It's not passing through the filter when you notice the benefit goes down similarly.

CHAIRMAN SCHULTZ: The last bullet is talking in a relative sense. That is, the engineering filter's benefit, in comparison to Alternative 2, Alternative 3 and then comes Alternative 4, in a relative sense it could further reduce the risk.

MR. STUTZKE: That is correct.

MEMBER CORRADINI: And the differences are driven by the assumed operator action failure probability?

MR. STUTZKE: In part, yeah.

MEMBER CORRADINI: And the other part is what again?

MR. STUTZKE: The hardware reliability itself.

MEMBER CORRADINI: Simply because what's not available because of the initiating event?

MR. STUTZKE: Right.

MEMBER CORRADINI: Okay.

MR. STUTZKE: Now for example, if the FLEX, the portable FLEX pumps fail early on, you'll

go to core damage and then they're no longer available to provide post-core damage water injection. You've already used the resources that you have available.

MEMBER BROWN: For the uninitiated or for the harder to understand like myself, this assumes, and my understanding is for this whole process is it assumes we have core damage, severe accident. Let me finish Mike, and you need -- you want to prevent overpressurization and liner melt-through post-core damage; is that correct?

MR. STUTZKE: That's correct.

MEMBER BROWN: Okay, thank you. Now you can shake your head up and down like this, as opposed to the --

MEMBER STETKAR: For clarification, for my help, the whole analysis does not assume you have core damage. Marty's analysis starts with a loss of all AC power.

MEMBER BROWN: I understand that.

MEMBER STETKAR: And it evaluates whether you get core damage, and then if you do have core damage, what you said.

MEMBER BROWN: Yes, I got that. I remember the loss of power part of it. But I mean in terms of -- putting aside the quantification part

of it, I'm trying to get, wrap my head around why the water -- why both of them go together. The water addition seems to be a key ingredient, my words, I think it was a key ingredient in order to really minimize downstream consequences along with the venting.

You have both tools available to deal with it. The only -- I'm still not convinced that only Alternative 2, which is the venting only, is even though it looks quantitatively like not a whole lot of difference, why you wouldn't want to have the other tools available as well. That's a personal opinion.

MR. STUTZKE: Oh, it's also as Bill Reckley had said earlier. It's a matter of regulatory perspective. Sure, it's possible to design a really high temperature drywell vent. In fact, we could put one heck of a piece of pipe and valve and vented it thousands of degrees if we choose to do that, without any water addition. Practically, they're not going to do that. It's easier to cool it down.

MEMBER BROWN: Yeah. Cooling is always a nice thing to be able to do.

MR. STUTZKE: I agree.

MEMBER BROWN: It seems to me. So that's --

MR. STUTZKE: I agree. And so in order to cool that down, that's also preventing a liner melt-through like this.

MEMBER BROWN: Right. Okay. You've helped me out with the answer and clarification here also. Thank you.

CHAIRMAN SCHULTZ: Other questions for Marty?

MEMBER BLEY: Yes. I'm sorry.

CHAIRMAN SCHULTZ: Go ahead.

MEMBER BLEY: I'm back to the one where you guys said now the industry had event trees too. Yeah, they did, but when I look at them, I can't tell for sure what they used for human error probabilities. I can see functional failures in the event trees that are affected by human error.

They did run the sensitivity study, not quite as extreme as you did, on the latent effects, and instead of running from zero to one, they ran from .01 to about 1.0, with essentially no difference in those. So I'll wait until I hear from them, but I just wanted to mention that.

I'm still not sure if that might not in

fact -- what they use have some effect on why those things are --

MR. STUTZKE: Well, I will let industry answer. But it's my understanding that --

MEMBER BLEY: Yeah. I don't want you to answer for them.

CHAIRMAN SCHULTZ: Ron.

MEMBER BALLINGER: Yeah. I really appreciate the risk insights. I'm sure they're true. But if you take, and I also very much appreciate the uncertainty slide. But if you take Slide 15 and superimpose it on Slide 11, which I'm trying to mentally, it looks to me like nothing makes any difference. They're all within the -- with the exception of the two black bars, they're all within the margin of error.

MALE PARTICIPANT: Well nothing makes much difference.

MEMBER BALLINGER: Nothing makes much difference.

MR. STUTZKE: That's the other message. They discussed the -- the parametric uncertainty is larger than the differences among the various sub-alternatives. That being said, we're supposed to make regulatory decisions based on the mean values,

which are the black boxes that should correspond roughly to the heights of those bars there, and "consider" the uncertainty. But that's a difficult job to do.

CHAIRMAN SCHULTZ: On schedule. We're behind a presentation and a break. I'm going to call the break at this point and ask people to come back in ten minutes, so we can move forward with the next presentation. So please be back at 10:40.

(Whereupon, the above-entitled matter went off the record at 10:31 a.m. and resumed at 10:43 a.m.)

CHAIRMAN SCHULTZ: Bring the meeting back into session.

Hossein, resume with your presentation, please?

MR. ESMAILI: All right, thank you. This is Hossein Esmaili from Office of Research.

Slide 2. I'm just going to talk about the overview of the MELCOR calculations that we did. We performed rather detailed accident progression analysis for representative BWR with a Mark I and representative BWR with a Mark II containment. The main objective was to predict the source term that we were going to pass on to Jon to his consequence

analysis with MACCS.

The source term is, of course, impacted by the effectiveness of mitigation, so we looked at the reactor pressure control, containment venting in terms of timing, and location of the venting, wetwell, drywell, and different water management strategies.

For the Mark I, we used the SOARCA model. We converted to the latest version of the MELCOR. We had to build in some of the boundary conditions in terms of venting and water addition, etcetera.

For the Mark II model, we had an earlier model and we converted it to MELCOR 2.1. We upgraded it to be more consistent with the Mark I in terms of the boundary conditions that we're studying under these conditions.

Next slide.

Okay, so this is a rather busy slide and I have no intention of discussing this in detail. I just want to point out a few things, that for the Mark I we did about 50 runs, so the runs made 50 calculations. And we looked at various combinations of the boundary conditions, both pre- and post-core damage.

So one of the things you see RPV pressure

controls that we looked at the availability of power and ability of the operators to control the pressure, both pre- and post-core damage. The second thing is in terms of pre-core damage we looked at the operation of the RCIC and anticipatory venting. This is an early venting tool to potentially prolong the operation of the RCIC.

For the baseline calculation we assume that the suction source from the suppression pool so the condensate storage tank was gone. But we did sensitivity to CST two of the runs and we assume that the RCIC would fail at high suppression pool water temperature.

We also looked at the SRV opening, pre-core damage, the time that the operators open the SRV after RCIC failure and post-core damage if the SRV gets stuck open.

In the case of a post-core damage boundary conditions, water injection, as Marty was talking about, is either into the RPV or to the drywell. And the aim is not only to preserve the containment, but potentially reduce the source term. I think this thing will come up a little better in our path forward.

In terms of the containment venting the

preferred path is the wetwell venting. If water injection is not controlled, if it's not managed, at some point I will show that later, at some point you will need to go and open the drywell vent and vent through the drywell when the water level becomes too high.

Next slide.

So this is a little bit better. This is the run matrix for the Mark II containment for the condensed version of the Mark I. So we learned some stuff from the Mark I containment and so we reduced the number of actual runs that we did, but here because of the variations in the containment is amongst various Mark IIs, we did perform some scoping calculations to see what effect the containment design had.

So in the base model that we have, the lower cavity, this is the part just below right next to the wetwell, the lower cavity is partially filled dry volume. So in the sensitivity cases, what we did was that we removed that and we put in the water to compare to other containment designs that have water in the lower cavity design.

In addition, Dr. Powers mentioned this in several of the briefings we had, because of the

importance of suppression pool bypass and by bypass, I mean what you see on the red arrow. This is when the debris upper level had failure, the debris comes there and accumulates on the drywell pedestal. At some point the drain line failed, so there's a direct path now from the drywell to the wetwell. So at that point it's a bypass of the suppression pool.

Because of that, we did sensitivity. What we did was we looked at the previous studies, assumed as a baseline that there's 20 minutes delay from the time the debris comes out until the bypass occurs. We reduced that to zero minutes. As soon as the debris comes out, it's one of the sensitivities that you see at the bottom. As soon as the debris comes out, there's a bypass. So it fails.

And we also prolonged it. We did calculation. We assumed that the debris comes out. We're just going to wait until we completely ablate that few feet of concrete between the drywell and the wetwell and see what happens in terms of the release.

I just want to mention that there is actually one plant that there is no drain line. There is no downcomer in the end pedestal region and there's no drain line. So the timing of the bypass could be

prolonged.

Next slide.

So we talk a little about severe accident water addition and severe accident water management. The way we modeled it, you can see the figure on the right. So the lower head failure, of course, for these cases that I'm describing here, at about 23 hours. At this point, we start injecting water.

MEMBER STETKAR: Hossein?

MR. ESMAILI: Yes?

MEMBER STETKAR: Why is it always guaranteed that the reactor vessel will be pressurized sufficiently for FLEX injection when you get the onset of vessel breach? In other words, can't you have vessel breaches that eject material that have pressure in the vessel higher than the FLEX shutoff head?

MR. ESMAILI: What in reality happens is that what we experience is that you have a lot of --

MEMBER STETKAR: We've had a lot of these melts.

MR. ESMAILI: A lot of this molten material that is coming in.

MEMBER STETKAR: No, no, no. You said in experience what you see, so we've run these tests,

so we know that indeed the pressure will always blow the whole --

MR. ESMAILI: The experiments, which is actually -- there have been some Sandia experiments on the lower head failure experiments that showed what the --

MEMBER STETKAR: Were you trying to get lower head failure to maximize the debris release into the containment in those experiments?

MEMBER REMPE: And did they really simulate the geometry for a BWR vessel, those Sandia tests?

MR. ESMAILI: No, they did not, but this is the point I'm trying to make is that during the in-vessel melt progression, a lot of material eventually makes it into the lower head. There it's full of water. At some point you are going to vaporize that water. So this debris has to heat up. Up until that time, there is no lower head failure. So because of the large amount of material that is in there, there is going to be a rather extreme terminal loading of the vessel wall. And based on the experiments we have seen, at some point this lower head is going to fail, either by creep rupture or the penetration failure and if you have material that is

sitting there, molten material that is sitting there, they're going to come out of the vessel. They're not just going to -- they're going to ablate the vessel wall more and more so by the time that the molten material has come out of the vessel, there is enough hole, let's say in the lower head, to completely depressurize the vessel. This is our understanding. We cannot have a situation where you would fail the lower half and you would not depressurize the vessel.

MEMBER STETKAR: It's not my area of expertise, but you make these absolute statements like you cannot have.

MEMBER CORRADINI: There's a transition period where there will be a pressure difference, but the pressure difference is not going to last for hours and hours. It will last from seconds to minutes.

MEMBER BLEY: And that won't make any difference.

MEMBER STETKAR: Seconds to minutes don't make any difference. Hours and hours might.

MEMBER CORRADINI: Right. I think that's what -- I think that was Hossein's --

MR. ESMAILI: It's not -- this hot material is not going to stay there without failing the lower half and once this lower head fails, either

-- even if it's a two centimeter instrument failure, if molten material goes to that, it ablates the wall. So by the time -- and we have seen this in the direct containment heating issue resolution. Those experiments that were done showed that -- and the models there that even if you start with a two centimeter hole, by the time that the debris comes out and the gas blowthrough occurs, you are ending up with a 40 centimeter hole. A 40 centimeter hole is big enough to completely depressurize the vessel.

Regardless of that, what we are assuming here is that we are assuming that we are injecting after the debris has come out of the vessel.

MEMBER CORRADINI: I guess I wanted to ask a different question. What is your concern that there is a path whereby you just sit inside and you can't get water in?

MEMBER STETKAR: That you can't get water through the vessel. They presume for the reactor vessel injection it comes through the vessel out through the hole and then you can cool the debris in the drywell within a sufficient amount of time after the drywell attack.

MR. ESMAILI: After the lower head has failed.

MEMBER STETKAR: After the lower head has failed. And I'm questioning whether or not that scenario always works, that indeed you get sufficient injection.

MEMBER CORRADINI: That's the reason I asked the question earlier that it's still not clear from a guidance standpoint whether you want to do a combination of in-vessel injection as well as --

MEMBER STETKAR: B- and I'm not arguing about the drywell injection to cool the debris because obviously that -- whether it's spray or whether it's pumped water into the drywell, I don't care how they get it there. I'm just questioning the scenario because the industry has said that they should prefer to have vessel injection. That's the whole reason for my questioning about getting power to the valves to get water from outside the reactor building to the reactor vessel.

And the models say that -- Marty's models allow you to have in-vessel retention if you can be pressurized between the time of onset of core damage to vessel breach. His models allow you to do that, requires active depressurization. But even if that fails, his models have ex-vessel retention in the drywell presuming that you have injection through the

vessel through the penetrations in the bottle of the vessel with sufficient water to cool the debris for only in-vessel injection, not any water addition -- any other water addition to the drywell.

So my question was framed around those scenarios where it's presumed that you always have sufficient depressurization post-vessel breach to allow enough water to get in from the low head FLEX pumps through the reactor vessel through the hole down into the drywell to clinch the debris before you start to get drywell liner attacked.

MEMBER BALLINGER: So there's no chance that the vessel just becomes another steam generator in the sense that you pump the water in there and it just flashes the steam because there's other stuff in there and you get steam out the bottom as opposed to water.

MEMBER STETKAR: You can't pump the water in if the pressure is off. This is a low pressure pump.

MEMBER BALLINGER: I could still be steam at lower pressure.

MR. ESMAILI: That would require that the lower head has already failed.

MEMBER BALLINGER: Right.

MR. ESMAILI: And the scenario that you're describing said the lower head had failed, but somehow you have kept pressure inside the reactor. And based on all that we know, experiments, modeling, etcetera and this is not -- we believe that the vessel is going to be depressurized.

MEMBER CORRADINI: Your question is it re-seals itself. Somehow it breaks and it --

MEMBER STETKAR: Or it's --

MEMBER CORRADINI: Have we thought about possible geometries where --

MEMBER STETKAR: I'm kind of -- what you're saying on this, in every way, shape or form, once I start bringing stuff down to the bottom, it's not going to sit there --

MEMBER CORRADINI: I'm just questioning whether all of those analyses and experiments that have been run have been done for the purpose of maximizing core material relocation into the drywell which would set parameters in ways to get the stuff into the drywell as fast as possible compared to the way physics might work which might not be that way if you wanted to hold it up in the vessel at relatively high pressure.

MEMBER BLEY: And you start maybe

injecting water just as that process begins.

MEMBER REMPE: So for a long time industry like that visualization where you had the separated debris with the ceramic beneath the metallic layer and then you had a side failure and only the metal would go out of the vessel because that was advantageous for other reasons with AP1000 or other plants. So then you had like a small size failure. I don't think MELCOR models that type of failure. They assume a bottom center failure.

MR. ESMAILI: It can have stratification. But what I'm --

MEMBER REMPE: You have a limited size though. You don't have a limited size -- would that not affect this scenario what John's trying to say about that there's still going to be ceramic and debris down there because not all of it is going to go out.

MR. ESMAILI: It's still going to depressurize.

MEMBER REMPE: It will depressurize.

MR. ESMAILI: So we just have to agree on the fact that once you have this couple of thousand degrees debris that is sitting, this vessel wall, whether it's going to be quick rupture of the vessel

wall or if it's going to be some instrument to transition or it's going to be some focusing effect, this thing is going to give. It cannot hold that much material and once that opens, right, once the lower head -- that initial hole is going to be sufficient to depressurize. So that's the only thing --

MEMBER REMPE: As water is going off to the side or whatever.

MR. ESMAILI: No, no. And I think we all agree that that's what's going to happen, then we say yes, we can have -- we can put water into the pressure vessel.

MEMBER SKILLMAN: Hossein, what if it doesn't happen the way you theorize? I mean I know of an example of one where we had 5,000 dripping corium on the lower head, the head held. But if there had been --

MR. ESMAILI: But that was the recovered one. Here, we are just waiting -- here, we are waiting until the lower head fails. We're actually waiting hours and hours until this material comes in.

MEMBER SKILLMAN: You're depending on vessel failure for depressurization to ensure you can pass water through that vessel. So in a way, you're

kind of depending on a small break LOCA going to a large break LOCA in order to achieve cooling?

MR. ESMAILI: No. We are -- our assumption is that post-core damage lower injection whether into the drywell or RPV is after lower head failure. Okay?

So by the time the debris comes into the lower -- before that, there's substantial amount of water. It takes maybe four to six hours to even vaporize this water that is sitting there. And then the debris heats up. By the time the debris heats up, this is not TMI, this is we don't have injection at that point. It just heats up, heats up, heats up. The vessel is going to give. You can have an instrument failure or you can have creep rupture. I cannot think of any scenario that you can actually keep that much material in the vessel without cooling it.

MEMBER STETKAR: Yes, but the models do account for in-vessel retention. I mean there is some likelihood that indeed you can arrest the core damage before vessel failure.

MR. ESMAILI: That's right.

MEMBER STETKAR: Just for the record, it was --

MR. ESMAILI: Right, and we did one calculation that we did retain. Here, we are talking about the case where the lower head failed. Once that fails, you can either inject into the drywell or the RPV because the system is going to be depressurized, if has not been depressurized before.

MEMBER REMPE: On another question just quickly, the main document that was submitted indicated that the MELCOR model were informed from insights from Fukushima as well as SOARCA. Were any model changes implemented because -- do we really have enough information from Fukushima to make modeling changes?

MR. ESMAILI: What we are talking is the operation of RCIC, for example.

MEMBER REMPE: Okay.

MR. ESMAILI: We just prolonged it. If you look at SOARCA, we assume four hours of RCIC time. Here we allow about 16 hours. But then it fails at about 9.6 hours.

MEMBER REMPE: Thanks.

MR. ESMAILI: I was under the impression there was not going to be any questions.

MEMBER REMPE: That was a follow-up question.

MR. ESMAILI: Okay, so just to finish my thought. So at 23 hours we are going to have injection. The red line is what we call water SAWA, water addition. So we're going to go with 500 gpm for the next two days. And at some point here we have to open the drywell.

MEMBER CORRADINI: And then as to where it goes at this point, the blue line --

MR. ESMAILI: The blue line is the water management.

MEMBER CORRADINI: I'm sorry, the red line then it's going in with vessel addition?

MR. ESMAILI: It doesn't matter. In this case, in the single cases, I'm showing cases 9 and 10, it's RPV addition. So I'm just showing you how we modeled this water management. We can do other things. This is a stylize assumption that we make.

The point I wanted to make is that if you look at the blue line, you still have to go for about 16 or 18 hours before you get to that 21 feet level. So going back to the -- at that point, then you can reduce and then you reduce the water levels and instead of 500 gpm, you have less than 100 gpm and we can remove the decay.

So going back to the table, these are the

differences between Case 1 which is a no water; Case 9 and 10 which is water management and water addition. And you can see there are some differences in terms of pre-core damage in terms of boundary conditions. But what happens is that in about 12 hours you have core uncovery.

And shortly after that, this is what I was talking about, shortly after that it depends, you may, maybe about an hour to two and a half hours you have to open the vent. And the vessel failure occurs at about 23 hours. So this initial venting that I was talking to you about is going to happen maybe four to six hours before the lower head failure.

If you have no water, you are eventually going to have a drywell liner melt-through. It's not going to be immediate, so when the lower head fails at 23 hours, liner melt-through occurs at 31 hours. This is because you have some residual water. This amount of residual water needs to vaporize because of the recirculation line break, etcetera before you can get to the drywell liner.

If you have water addition, whether it's management or addition it doesn't matter. You're not going to fail the containment, but what you can see in Case 10, the last column, is that you have to open

-- you have to close the wetwell vent line at 42 hours. So that's almost 20 hours, 19 hours after you started injecting. And then close it at that point and then open the drywell venting at 54 hours. So there's a substantial amount of time in which this action can take place.

Next slide.

Okay, so in terms of the magnitude of release, I have shown this figure before, there is no probability associated with it. We provided this information to Marty and then Marty appropriately put it into his PRA.

And so the thing I wanted to mention here is that highest release, these are the main steam line creep rupture. You can see all those top ones. So these are like really bounding because what happens is that at around the time of core damage, you have a main steam line creep rupture. This is a small containment for a Mark I. The upper head is going to open and you're going to essentially bypass the suppression. So a lot of the material is going to go out through that path.

The next is the cases with no water. Here you can see at the bottom it says 1/2A. And here you can see the releases are of the order of a

few percent.

Then you do water management, water addition, etcetera, and you can -- in some cases you can drop these releases below one percent, maybe .7, depending on what the boundary conditions were.

What we found out is that whether there's water addition or water management, it doesn't really matter that much. The releases are comparable. And even RPV and drywell injection, when you're looking at this -- it's difficult to look at the big differences here.

MEMBER CORRADINI: What is 3B? I'm sorry that I don't remember.

MR. ESMAILI: 3B is water addition into the drywell. So I had these things at the bottom. So it's SAWA, severe accident water addition, the 500 gpm and you open the drywell at some point and you inject into the drywell.

MEMBER CORRADINI: How are you injecting that I'm getting over pressurization? That's what I don't understand. I'm still looking at 3B. So 3B is water addition via drywell injection and I'm getting -- and so it's drywell vent only with water addition? That's what I'm still struggling with. I'm sorry.

MR. ESMAILI: No, no, no. Some of these cases are only -- so the green ones that you see is the preferred path. So we open the wetwell. We open the wetwell. If we need to, we go to the drywell. But it happens much, much later as I showed you in the previous.

So red lines are drywell only. I never open the wetwell. The wetwell is not there. I only open the drywell and I did some calculations to have -- to correspond to Marty as the release category.

MEMBER CORRADINI: Thank you.

MR. ESMAILI: Next slide. Okay, so now we advance to the Mark II. For the Mark II, again, this is a condensed version, but I'm just showing you a comparison. So all these solid lines are the Mark II. The hashed are the Mark I. And I just show that as a comparison for the actual runs, you know how things matter.

So what we notice is that for the Mark II, the releases are sometimes compatible, but lower in general than the Mark I. Now you have to know that the Mark II, the drywell by itself is about 30 percent more volume than the Mark I that we studied. So it has more volume. So all of the cases like, for example, Run 3 and Run 52 that resulted in a main

steam line creep rupture in a Mark I does not do so in a Mark II. So I don't get a main steam line creep. Even though the probability of having a main steam line creep rupture, if you appropriately do pressure control is very, very low.

Then in Run 1, we did all those sensitivity calculations by just different cavity designs, by having water instead of having just all type of calculations and you see those releases starting from the green going all the way to the purple and that shows that, you know, if you have the bypass at zero, if the bypass is immediate, as soon as it becomes it creates a bypass, you have the highest releases and if you allow the bypass some time, then the releases are going to down lower.

Next slide.

So this one is -- I've shown this before. This was part of the order action, so used some of these calculations for the order and this just shows you that it doesn't matter what type of water addition you have whether it's into the drywell, into the RPV, addition management, etcetera, for all those cases your temperature, the maximum drywell structure temperatures remain below that 545 that is part of the order.

So whether addition not only prevents containment failure, but also cools the containment so you are not getting very, very hot temperatures.

Yes?

MEMBER REMPE: Before you leave this slide, so maybe I interrupted too soon, but there's a figure 4.10 in your main report that has the Mark I containment gas temperature for Case 1 and I think you have it as one of your backup slides and the peak gas temperature in the drywell pedestal area is real close to 1,900K. And it's in degrees F here which always bothers me when it goes back and forth. I think to think about it.

If you had those kind of high temperatures wouldn't you -- if the concrete -- maybe the gas temperature just goes by in a hurry, but when I see temperatures that are like where concrete is degrading and stuff like that, do people think about that? I don't think they do in MELCOR.

MR. ESMAILI: To get to those temperatures, I mean if you go back to Slide 8 -- let me actually take you to -- okay, Slide 18. I think you are referring to Slide 18, right?

MEMBER REMPE: It's one of your backup slides I noticed when I was looking at the slides

ahead of time. The main report is where I first saw it.

MR. ESMAILI: Here is Slide 18. So this is no water injection. This is the containment atmosphere temperature. The highest temperature we see in the pedestal region, this is a very small volume, this is where the debris is, but it can go up to about 2500, 2700, right?

But the rest of the drywell you see at the lower power, the middle portion, the upper, maximum we can get is about 1200 to 1500 Fahrenheit. These are all in Fahrenheit.

MEMBER REMPE: Again, the gas temperature that's so high, I mean it's the concrete that's exposed to those kind of temperatures at certain locations, you start seeing it decompose and all sorts of things like that and I don't believe the code ever accounts for it. It doesn't heat up enough.

MR. ESMAILI: You mean other failure in the --

MEMBER REMPE: Yes, there's a bunch of things that would happen if you got to those kind of temperatures and it was just something I was going to mention.

MR. ESMAILI: We are venting anyway.

MEMBER REMPE: Yes, well --

MR. ESMAILI: Even if the concrete stays intact all these cases have the vents open so these are going out.

MEMBER REMPE: Well, I was just wondering when you see temperatures that aren't possible.

MR. ESMAILI: But with water injection, you see the temperatures are quite low and this is even the gas temperature, the structure temperatures remain somewhat cooler than this because there's a lag between the two.

MEMBER REMPE: Okay.

MR. ESMAILI: Okay, so Slide 9. So as Marty said a combination of venting and water addition is required to prevent containment failure. This is also a beneficial strategy for mitigating radiological releases. As soon as you start injecting, you are stabilizing the releases, you are not getting any post in the lower head failure or revaporization, etcetera.

Anticipatory venting, this is before core damage. This is also beneficial to reduce the containment pressure. So it increases the time from when you need to -- from the time that you have core damage until the time that you have to post-core

damage venting.

This has been discussed before, containment venting is efficient in purging hydrogen and non-condensables. So one of the slides that I have in the backup shows that once you open the vent you are purging all of the non-condensables and hydrogen.

The highest calculated releases goes from the main steam line creep rupture, but this was one of the least likely variations. And this is, I think, what Marty has taken. So we had to work really hard to induce the main steam line creep rupture in this scenario.

For the Mark II, the releases are generally comparable to or lower than those in the Mark I. And we also did scoping calculations with different cavity designs. We didn't do all the plants, but it was sufficient scoping calculations to tell us where the releases are. And certainly below the Mark I that Marty was talking about.

CHAIRMAN SCHULTZ: So that statement means that even when you look at variations with regard to the configurations, the results were comparable with what you had had previously.

MR. ESMAILI: Yes, because there's lots

of uncertainties about how these -- how these drain lines fail, how these piping fail.

CHAIRMAN SCHULTZ: But you're not expecting dramatically different results.

MR. ESMAILI: I am not expecting dramatically different -- yes, because it's really driven by what's happened inside the vessel and we do vary the timing of this bypass.

CHAIRMAN SCHULTZ: Thank you.

MR. BARR: My name is Jon Barr and I'll be discussing the MACCS offsite consequence analysis supporting the CPRR rulemaking. I work in research in the Accident Analysis Branch and I'll cover the modeling approach used and then just the results and conclusions. I plan for this to take about 20 minutes. I don't know where we are on schedule, but I'll try to go a little faster.

I also want to acknowledge the support and feedback that I've gotten on this from other MACCS staff, people in the working group, Tina Ghosh, Keith Compton, A.J. Nosek, Randy Sullivan, and NSIR was very helpful for the evacuation part of the calculation and also Sandia provided some support, Nathan Bixler and Joe Jones.

And then finally, I want to also

acknowledge my branch chief, Pat Santiago, for her support and feedback on this and giving me the opportunity to work on such an interesting project.

Okay, so now for the modeling approach used here, this rulemaking is supposed to apply to or designed to apply to the entire fleet of Mark I and Mark II BWR sites across the U.S. Given that they lie across the country with different site characteristics, the approach used here was to develop one reference MACCS site-specific model for a reference Mark I site and another for a Mark II site and then to consider some of the site-to-site variability through some sensitivity calculations.

Peach Bottom was the reference site for the Mark I calculations. We've studied it heavily in SOARCA and SECY-12-0157 and so therefore had a lot of the necessary information readily available. And we chose Limerick for the reference Mark II site given that it has the highest population among -- actually all Mark I and Mark II sites. And thought it would be better to be able to extrapolate from a high population site to a lower population rather the opposite direction.

Sensitivity calculations were done to look at some of the important parameters across

various parts of the code include the emergency phase, the intermediate phase, and the long-term phase, as well as population values. And then also the approach was to use the most current data sources available for these models.

So for example, like for the Peach Bottom model, we reviewed it and looked at all different areas, looked for sources of new data to use. And one example is that the licensees submitted evacuation time estimate reports in 2014 which have updated information about how the people around the site would evacuate and that uses more recent census information, more detailed -- more recent modeling.

MEMBER STETKAR: Jon, considering the fact that these events are determined almost entirely by earthquakes that are larger than the design basis earthquake of the nuclear power plant, how do those evacuation time estimates account for the fact that we know that we've had a very, very large earthquake?

MR. BARR: The evacuation time estimates look at a range of conditions and times --

MEMBER STETKAR: But this is not an average event.

MR. BARR: Right, right.

MEMBER STETKAR: We know that we've had

a very bad earthquake here.

MR. BARR: Yes. So the evacuation time estimates do not consider events such as these seismic events.

MEMBER STETKAR: So how have you considered it in your MACCS analyses?

MR. BARR: So initially when I was developing the MACCS model, I talked with the team and we said that essentially we're not just going to focus on seismic. We're going to look at internal events as well.

MEMBER STETKAR: But I'm sorry, that was back in August of 2014. The frequencies were driven by those internal events which indeed you could justify the average evacuation because it's a sunny day. Now we know that the ELAP frequencies are -- except for three or four sites, predominantly driven by very severe earthquakes. When I say very severe, I'm not talking of Fukushima-size earthquake, but I'm talking of earthquake larger than is the design basis of the nuclear power plant which would exceed the normal structural design bases for things like houses and hospitals and factories.

MR. BARR: So right, there were some sensitivity calculations done. One of them looked at

a delay of one hour for the entire evacuating cohorts. Another one looked at one specific source term and then looked at various, longer evacuation delays of three hours and six hours and also assumed no evacuation and no emergency phase relocation so the entire EPZ was just sheltering in place. That was only for one source term, but it helps us in that kind of hypothetical situation.

MEMBER STETKAR: Everybody is going to shelter themselves in place. They're not going to run outside and get concerned about buildings falling down as people do in earthquakes.

MR. SULLIVAN: If I can chime in, Randy Sullivan. We happened to do -- is this on?

We actually have looked at earthquake damage at Peach Bottom. As it turns out in the SOARCA report, it turns out to be a hard rock site and there's very little impact to the evacuation routes. However, there would be impact to communication system and it would disrupt emergency response and all that sort of thing.

The sirens themselves are battery backup at these sites, so they would sound. And of course, radios in cars would still work. But we've done some analysis of the aftermath of a serious earthquake and

there are impacts. It does delay evacuation. But the basic fabric of response would remain although degraded. Thank you.

MEMBER BLEY: Just following up on Jon on the case where you assume they don't evacuate and shelter in place, the earthquake people and the structures people are going to tell people not to go back in their structures, so it seems an odd combination of assumptions.

MEMBER RAY: I want to take the opportunity of Jon's comment to comment on something I wanted to say at some point along here. One, but first a reminder that yes, earthquakes dominate, but on the other hand that's a reflection of the fact that we've considered earthquakes. We haven't considered flooding, for example. I don't know whether there are sites that would have an even larger effect due to flooding. But more importantly, we do benefit from evacuation whether it's impeded by the after effects of an earthquake or not in terms of the quantitative health objective being met.

I continue -- I've made this comment before so I feel I can make it again here, when it comes to the difference between alternative 3 and 4, for example, the impact on land usage following an

event isn't quantified, isn't monetized. And I am not convinced that doesn't have a relevance to whether you have a benefit from a filter or not because we don't view it. Yes, the people are evacuated by the time the event actually results in an on-site release that's substantial, so the health objective is benefitted by that evacuation. But then there's the loss of the land usage that is not part, presently, part of our standards for evaluating the benefits of features like a filter. I just want to make note of that as well as the fact that we don't consider flooding which I think has already been stated.

MR. BARR: On the point of the seismic as the initiating event, I just want to mention two more points. One is that when we look at the source terms that were calculated with MELCOR, most if not all of them provide a start or release to the environment that sufficiently delayed that there is some margin when we look at the evacuated time estimate and the time at which we think the people will be able to get out and then compare it to when the actual releases would start from the plant. There is, in most cases, a substantial margin there.

Another thing to consider also is that the earthquakes would provide advanced notification

for some people that there is actually an event going on and in some cases people might have a little extra time in the beginning to be aware that there is a situation and start monitoring immediately, rather than as we model some of the people would be continuing to go about their business. Then they finally learn about the event and etcetera.

MEMBER STETKAR: All of those qualitative discussions may register, but the fact remains is that you took average evacuation time estimates for your analyses. You didn't try to model what was actually happening based on the front end of the process which says this is going to be a big earthquake. So all of your published results are skewed by the fact that your baseline analyses use average evacuation times. They use average assumptions about what fraction of people will shelter in place, what fraction of people will evacuate as a function of time after they're alerted which is fine for a sunny day event. But this isn't a sunny day event.

The other things are presented in the context of sensitivity studies or as you said a particular release category with particular assumptions about either delays or assumptions about

everyone sheltered in place. But those are assumptions and they're not the basic analysis.

MEMBER RAY: Again, not to belabor this given the time we shouldn't be taking it, I think, but similarly, a site which we've looked at here that is affected by a tsunami that originates a long ways away can make the limited egress paths seriously impacted.

MEMBER STETKAR: But again, we brought up floods. In fairness to the staff, they did what they did. They didn't do what they didn't do. They didn't look at floods and they didn't look at internal fires. They didn't look at a bunch of stuff. They looked at two cases of loss of all AC power. And currently, the current snapshot says that that condition is for most sites determined by a severe seismic event. And that's fine. That doesn't affect the MELCOR analyses in particular, but I'm questioning how it might affect the MACCS analyses.

MR. SULLIVAN: Well, it's an interesting discussion of what's -- this is Randy Sullivan again -- of what's the proper quantitative health objective to use in a severe accident such as a way behind design basis earthquake that has affected the populace. Would you use the steady state, sunny day,

quantitative health objective also? Or would you modify that in some way? That's something that we've wrestled with for a long time and I'm not sure we came to a good resolution.

MR. BARR: On this slide, I'll talk about some of the technical elements of the MACCS models that are very high level. For the site data that's used in each population comes from the 2010 census and then it's scaled forward to a project target year of 2013 for consistency.

The economic values are scaled using a CPI to 2013 and then therefore the offsite costs in dollars are reported in 2013 dollars.

For the meteorology data, the onsite meteorological tower data was submitted from Exelon, a licensee of those two sites. We already had it from SOARCA for Peach Bottom, but it was submitted for Limerick, too. And the process was then to review the raw weather data, look at the missing data, look at data recovery and then to convert it to the appropriate format for MACCS. And then of the two years available, the one with higher data recovery was used. It contains hourly data points for wind speed, wind direction, precipitation and stability. And that weather data was also compared to data from

other years to assess how representative it is of a larger time span.

Another part of the meteorology was that each calculation uses about 1,000 weather trials and then the mean value across the weather trials is reported as the end result for that calculation.

The MACCS uses a straight line Gaussian plume segment model for adversary transport and deposition. And the approached used here was very consistent with past studies.

The emergency phase covers the first week of the accident and models the sheltering, evacuation, relocation. I mentioned before a lot of the data came from or was informed by the ETEs that were recently submitted.

Following the emergency phase is the intermediate phase which is modeled to last for three months in all calculations. This is a time period in which essentially all the authorities would be planning the long-term clean up before the decontamination would actually begin and then is the long-term phase which would last for many decades, out to essentially 50 years when all of the cleanup activities would go on.

One of the important framers for the

long-term phase is the habitability criterion which essentially is a dose per time period that defines whether or not the population would be allowed to live on the land and continue the activity there or whether or not relocation and interdiction would be triggered. Pennsylvania uses a 500 millirem per year habitability criterion and that was used in the base calculations. The EPA protective action guides use a 2 rem number in the first year so that was modeled in the sensitivity study.

The dosimetry and health effects are identical to in the past study, SECY-12-0157. Health effects used the linear no threshold dose response model. And radionuclide release I'll cover on the next slide.

Slide 4, please.

So Hossein showed you the big table with all the different MELCOR source terms, about 41 for the Mark I analysis and 12 for the Mark II analysis. We decided to look at a range of possible decontamination factors and applied each of these to each source term case. We took 10, 100, and 1,000 and applied them only to the release pathways that would be vented.

This led to about 164 Mark I source terms

and 48 for Mark II, so a binning strategy was used for some efficiency there. Also, a lot of the source terms became very similar to each other. The binning was based on cesium and iodine magnitude, cesium being the most important radionuclide for long-term consequences and iodine for early consequences.

The timing of release was also considered, but in all cases it sufficiently delayed relative to evacuation so that was not considered in part of the bidding process.

Slide 5.

On this slide and the next slide I have the source term bins for Mark I here, they're 18. And then for the Mark II on the next slide, there are nine. These show the range of cesium and iodine and then in the fourth column is the representative case. This was selected to approximate the average overall of the source terms with any given bin. And the column on the far right shows the start of release to the environment. They were almost all in the 10 to 20 hour range which is sufficiently after the evacuation, if guided by the ETE.

Slide 6.

These are the Mark II source term bins. There were fewer MELCOR calculations so fewer source

term bins were needed here. Again, the start of release was delayed.

Limerick has a higher population in the ETE compared to Peach Bottom so the evacuation would take longer, but not substantially.

So essentially using the 18 source terms on the last slide and the 9 in this slide, I ran MACCS for all of those and got essentially what was a look up table of all the condition of consequence results that could then be mapped back to the PRA to associate with the different CPRR alternatives.

MEMBER SKILLMAN: Jon, what drove the variability in the start of release?

MR. BARR: A lot of the assumptions regarding the accident progression.

MEMBER SKILLMAN: Look at 5 and 6, what drives the start of release? Just the type of accident?

MR. ESMAILI: Yes, the different boundary conditions. Sometimes in some of the cases, the venting occurs at about maybe 14 hours. This is pretty consistent. In some cases, the venting does not occur until, in the Mark II, until the lower head fails. In the case of a Mark I, it's a little bit different.

MEMBER SKILLMAN: Thank you.

MR. BARR: Slide 7. So here's some sensitivity calculations that were run. These were all run essentially one at a time for three different source terms. I looked at low source term, a medium source team and a high source term.

So the first was a delay of evacuation of one hour. Next, in each MACCS model, I divided up the population into different cohorts and modeled one cohort in each group that would not evacuate, either because they didn't receive messaging or they refused. This percentage of population was increased to five percent to look at the effect of that result.

In the intermediate phase, three months was used for the duration of the base case runs. I modeled this also with zero and also one year.

In the long-term phase, the Pennsylvania State specific guidance was 500 millirem per year was used in the base case runs, so 2 rem per year was used in the sensitivities to approximate the EPA's protective action guides.

And then also population density. For that one, Peach Bottom and Limerick were the high sites for Mark I and Mark II. I looked at the 50-mile population for the rest of the Mark I and II

sites and it shows plants that approximated sort of the low area and a medium population to see that impact in the results.

Slide 8.

MEMBER RAY: So how did the sensitivity on habitability affect the result? Where do we see that?

MR. BARR: So on this slide, I'll talk about some of the sensitivities, some high-level conclusions and then I can touch on that one.

A lot of the times, it really depended on the actual source term, given that some were very small, one was large. It would really depend. But in general, with a higher habitability criterion, we would essentially allow a higher dose to be accrued by people before we trigger relocation. So therefore, the health risks would be slightly higher in some cases.

Okay, so for Slide 8, looking all of the base calculations as well as the 100+ sensitivities, these are some conclusions for the QHOs. So first as was mentioned before there is zero early fatality risk for all of the source terms in all of the cases. This is partly due to the release magnitude and release timing and the assumption that the protective

actions in the emergency phase will be affected.

And for the other QHO for latent cancer fatality risk, the conditional latent cancer fatality risk was sufficiently low in all cases that when we considered the accident frequency, we're still orders of magnitude below the NRC safety goal and that goes for the status quo situation, even without external water addition and external filter.

The conditional latent cancer fatality risk is dominated by the exposures in the long-term phase which would be below the habitability criterion, partly due to the effective emergency phase protective actions and the release source term timing and magnitude.

The habitability criterion really acts as a backstop to the dose that can be accrued by the population and so this consequence measure tends to be a little bit less sensitive to some of the modeling inputs and assumptions, whereas the other types of consequence measures which are triggered as a result of relocating people tend to be more sensitive.

Slide 9.

So one way to look at the conditional consequences was to group them by the associated CPRR alternative. These are shown for the Mark I runs in

the top and Mark II in the bottom. And these are five of the six offsite consequence measures that we focused on, the sixth being the early fatality risk which was zero for all cases so it's not shown here.

And these show that if we start with the status quo case and then look at a successful external water addition and then on top of that the combination of external water addition with an external filter with a DF of 10, we see continuous reductions and consequences. Some measures showing more sensitivities than others.

MEMBER STETKAR: Jon, before you leave this, in the interest of time, I don't have enough time to go into details, but sometimes the details make a difference. These numbers, the conditional values under the individual latent cancer fatality column, I know how they were calculated. I don't know why they were calculated from the particular MACCS and MELCOR results. And I'll just for the record, and you guys can look at it later, is that the calculations that were done to produce those numbers did not include MACCS bins 13, 14, 16, 17, and 18. They were not included in those calculations. And those five bins individually each show the highest conditional latent cancer fatality

consequences. They're all higher than any of the others that you did include. So I was curious why you didn't include those?

MR. BARR: I believe those were the bins that were corresponding to the main steam line creep rupture scenarios which in discussing with the team we judged to be less likely than the other than the ones that were used which were the finer melt-through.

MEMBER STETKAR: But all of these are groups of scenarios. It's too much detail to go into now because of our time constraints. I'll just make that observation that they do not include those cases and it's really curious because some of the elements of those cases seem to affect qualitatively conclusions like there's no difference between water addition and water management. If you start looking at some of the subtleties where you make gross conclusions within the context of what you quantified, you might draw that conclusion. If you include the things that you left out, you might not draw that same conclusion.

I don't know because I haven't done the analyses. I get curious when things are left out and it's not justified why they're left out.

MR. BARR: Slide 10. These two charts

essentially show the same data from the previous slide, but now they are shown relative to the status quo case so that we can look at it on one chart. And essentially, these illustrate the potential reductions in consequences for the different types of alternatives. For example, if external water addition is successful, and then if we have both successful external water addition and an external filter providing a decontamination factor of 10.

And the five consequence metrics from the previous slide are shown with different colors here and we see is that the reductions given that the accident happens, can be notable and even significant for some types of consequence measures. And the population is subject to long-term protective actions tend to be more sensitive. for example, on the far right in the third grouping, the light blue bar is barely visible in the Mark II case. So some of the offsite consequence measures are affected more than others here.

Slide 11.

You've seen this chart before, but I just wanted to provide it to balance out essentially the conclusions that could be drawn from the previous slide showing that if the accident happens these

alternatives provide essentially significant reductions and consequences. But again, if we look at things on an absolute sense considering the frequency, we see we're well below the QHO. We're within the uncertainty bounds and on a relative standpoint there's not as much difference between the cases here.

Slide 12.

So at our GLD Steering Committee meeting we had some questions about essentially given that the delayed nature of the source terms that we've modeled here, what if we had a much faster source term, would that then show that there was a dramatic improvement and a potential benefit of an external filter. We looked at the fastest release MELCOR case which is Case 49. It's a short term station blackout and it starts to release to the environment at 7.3 hours. And essentially looked at different conditions, different evacuation delays which would approximate sort of the effect of having a faster source term.

And each of these have shown for two groupings, the first on the left is external water addition and the second is both external water addition and an external filter. And if you look at

the first grouping it shows that the difference is about a factor of 3. So adding the filter reduces the conditional individual latent cancer fatality risk by about a factor of 3.

And then using more extreme assumptions related to evacuations, we can see that the actual benefit of an external filter increases, but only up to a factor, a relative factor of about 6. So essentially, this is showing us that for this specific source term, there is more benefit to a filter if we have more extreme evacuation assumptions or if we have an earlier release, but the difference is not so dramatic to make us question this part of the analysis essentially.

MEMBER BLEY: Did you -- to follow up on what Mr. Ray asked earlier. Did you do the same kind of a look at bound contamination?

MR. BARR: In this case, where this is looking at just adjusting the evacuation assumptions so the long-term consequences would be unchanged.

MEMBER STETKAR: Jon, you did this for that particular earliest release category. I don't have a sense of how these things work. Do you think that this relative conclusion would apply across all of the release categories?

MR. BARR: I think that if we had this accident timing with a larger source term, we would see a larger benefit of the filter.

MEMBER STETKAR: No, no. I'm not talking about the filter. I'm talking about the relative height without the filter of the conditional individual latent cancer fatality risk. In other words, we see a factor of roughly 2.5 increased from the base case model --

MR. BARR: Right, right.

MEMBER STETKAR: -- to no evacuation, sheltering in place with increasing contributions from emergency phase. Do you feel, do you know, do you have a sense would that apply across all of the release categories you evaluated? Or don't you know? And it's fair to say you don't know.

MR. BARR: I think that even with larger source terms, the cancer fatality risk would still be dominated by the long-term exposure and so I don't think it would be terribly sensitive.

MEMBER STETKAR: But they're not here. I mean once you get to sheltering in place, the cancer fatalities are dominated, if you use that word. The pink is a lot bigger than the blue. Whereas in the base case models and with at least up to about three

hours, maybe five hours, the blue clearly outweighs the pink.

I'm asking whether the relative scale in other words from the base case analyses and increase in the conditional latent fatality risk of about a factor of pick a number, 2.5, 3, somewhere in that ballpark would apply across the full range of release categories that you evaluated. They have larger releases, but delayed more in time to some extent than the particular case that you took.

MR. BARR: I'd have to think about that.

MEMBER STETKAR: Okay. I don't do these kind of analyses, so I don't have any sense or intuitive feel for it, but I understand what you did, too.

MR. BARR: And this is just one source term, so we definitely don't want to take this out of proportion.

MEMBER STETKAR: That's right, right. Okay, thanks.

MR. BEALL: That's our presentation for today.

CHAIRMAN SCHULTZ: Thank you very much. Any more questions of the staff at this point?

We'll then move to the industry

presentation. While they're coming up, I'd like to thank the staff for the presentations this morning to each of you. I appreciate the work that's gone in and preparation and the delivery of the presentations to the committee this morning.

We'll move right to the industry presentation.

You are such a familiar group, you don't need name plates. We all know each other.

MR. KRAFT: They're coming. They're on their.

CHAIRMAN SCHULTZ: We're going to abide by the niceties here.

MR. KRAFT: Mr. Chairman --

CHAIRMAN SCHULTZ: Just check your microphone, Steve. This is our new system where -- just press the button on and then off when you're done.

MR. KRAFT: Got it. Thank you very much.

So Mr. Chairman, inquiry, how much time do we have?

CHAIRMAN SCHULTZ: You can just move forward in your presentations and we're not cutting you short based on the clock.

MR. KRAFT: Oh, okay. Thank you for that

kindness.

Once again, I'm Steven Kraft from the Nuclear Energy Institute. Thank you very much to the committee for inviting us to talk about this important rulemaking. I think you're familiar with the gentleman to my right here, Doug True from Erin Engineering; Stuart Lewis from EPRI; and Jeff Gabor from Erin Engineering. And we also have some members of the industry sitting at the table over there, Randy Bunt from Southern, and Pat Fallon from DTE, who would be available to answer questions as needs might be.

I will have a couple of opening remarks and then we'll turn the bulk of the presentation over to Stuart and Doug and Jeff. Our primary purpose today is to go over the technical work that we did that you've seen before. You saw a lot of the work we were here talking about, Phase 2 of the order implementation and we'll focus on that.

But I wanted to make a couple of opening remarks with regard to our initial views on the work done to support SECY-15-0085. First of all, we thought it was an excellent body of work in response to the Commission direction after they reviewed the original SECY-12-0057 that recommended ordering a filter for the plants that are affected by the order.

And we think that at the end of the day, these results are very consistent with industry's. There are differences and I think Doug and Stuart and Jeff can talk about those and why the heights of the bars look a little different in the charts and the differences look a little different.

But it is my purpose to sort of talk for a second about the regulatory aspect of all of this and there are three options, four options as you know in the SECY. I will simply say that in reviewing those options thus far since we saw the paper about was a week and a half now, we think we can achieve alignment with the staff on an option for going forward which may not be any of the options described to the paper. It may be a combination. It may be an alternate. It may be a different way of looking at the analysis. We haven't yet concluded. And frankly, we have not had any formal engagement with the staff to talk about it.

The second thing that we're concerned about in reading through predominantly Option 3, the staff's preferred option is we're sensing some potential for scope creep over and above what is in the order. The option says we want to make the order generally applicable, but when we read some of the

words we wonder about how that's going to be accomplished when it's turned into a rule.

The term, generally applicable, bothers me. I'm told it's a legal nicety compared to codify. My own lawyer could not explain it to me to where I would understand it. But I wonder if generally applicable to what. There are only 29 units affected. No one is building another Mark I or Mark II. So I kind of wonder about that. But it may be that legally that's not much.

But there are other things in that section that bother us about scope creep. And we look forward to engaging the NRC staff.

I don't want to delay the discussion any further. I'm happy to answer questions, but I would like to turn it over to Stuart to pick up the presentation.

MR. LEWIS: Thanks, Steve. Can you hear me? My name is Stuart Lewis. I'm the program manager for the Risk and Safety Management Program at EPRI. I'm really here on behalf of Rick Wachowiak who couldn't be here in person. He is on the phone up there somewhere, so presumably if you have a question for him, there's a way to tie him in. And he and Doug and Jeff and their team really did all

the technical work in this effort. So I'll try to keep my remarks fairly brief so you can get into the details with them.

But just as a bit of background, if we can go into the next slide, all of these analyses really were performed in the context of trying to understand what the important accident scenarios were that would allow under to understand the role of mitigating strategies of filtering and venting strategies. And they focused on the role of accident management, in particular, both the capabilities that exist now and those that are being implemented with the new venting and potentially with filtering strategies. And I think that understanding of accident management, whether it's to cool the core debris after there's been significant core damage to remove decay heat in an attempt to preserve the containment integrity, or to mitigate any releases that might result. That's really been a key part of this entire effort.

Let's go to the next slide.

So in terms of more specifically what the objectives were, a lot of work had to do with understanding the role of FLEX and responding to EPAP scenarios and you've already heard a lot about how

that was done in the NRC studies. These guys talk about what was done in our work here. Again, this was done in the context of what we were expected to be the dominant accident scenarios and especially those that might be relevant to this kind of discussion.

Ongoing, I think there was really a good level of interaction with the NRC staff to understand the intent of the work that was being done. Many of the technical details were shared along the way so that I think both sense of analyses could at least be done in a clear way and both sides could understand the work that was being done. And there was a lot of interaction also with the -- boiling water reactor owners' group to ensure that the emergency procedure guidelines and severe accident guidelines were properly reflected in the work also to provide insights back to the owners' group so that they could think about what might be done in the future to enhance the guidelines.

Ultimately, of course, there's a lot of input being provided to industry decision makers regarding the cost-benefit considerations related to especially the filtering strategies, but also the other strategies.

We have published a high-level report that describes the work and focuses on the conclusions and I know some of you by your comments and then all of you had a chance to look in detail at that report. There is a more extensive volume that we're in the process of publishing right now that we'll have all the technical details. That will be out in September, so it's nearing completion and for those of you who want to dig into the nitty-gritty, you'll have a lot more effort to do that, Jon, in the future.

Just as an overview of this process, very similar to the NRC's approach, it starts with a containment on core damage of entry to lay out the accident sequences of interest. In this case, there were 13 core damage scenarios that we've identified and evaluated. For each of those scenarios, the accident progression was considered in the context of the accident progression of entry, an APET which had 39 outcomes. So for each one of the 13 core damage scenarios, 39 possible outcomes were considered outside 507 different scenarios that were evaluated. For each one of those, a set of MAAP calculations to look at the physical progression of the accident and a WinMACCS calculation to look at the offsite consequences was performed. So we took advantage of

our performance computer at EPRI to be able to do a larger number of calculations than would typically be done in this kind of context because this was all done for all 24 of the alternatives. So that's a lot of different sets of calculations to try to capture the range of what needed to be considered.

And with that, I'd like to turn it over to Doug.

MR. TRUE: I'm Doug True from Erin Engineering. I'm not going to spend a great deal of time on this chart. We showed it to you before. It just articulates at a high level the 24 different scenarios or alternatives that we looked at starting with the base case. It looks like the staff did -- the staff had 24. I didn't go back and figure out exactly what the differences were, but they were -- they overlap substantially and certainly focused on the same key endpoints. The color scheme just sort of helps identify whether it's a case with water or not and whether the water is going into the RPV or the drywell.

So when we put this framework together we had an observed endpoint in mind that we wanted to look at which was a set of risk metrics and a set of other considerations that we felt that gave us insight

into the different alternatives.

Basically, the same kinds of risk metrics that the staff looked at with core damage frequency, conditional failure probability of containment and latent cancer risk and we also looked a little bit at financial consequence risk. We didn't take it all the way to cost benefit or anything because that's not really EPRI's purview, but we put the information together to understand what the financial consequences were.

MEMBER RAY: Say a little bit more. I know we're in a hurry, but financial consequence risk.

MR. TRUE: It's basically the present value of the offsite consequences associated with the accident which includes the dose, transition of dose to dollars as well as contaminated land, indiction, loss of use, relocation of people.

MEMBER RAY: Okay, well, that's what I was asking about earlier. So it's part of your study, at least in some form.

MR. TRUE: Yes. It's part of our study in the financial consequence side of things.

MEMBER STETKAR: But only the metrics that are quantified by MACCS. Is that correct?

MR. TRUE: Correct, based on --

MEMBER RAY: Explain what that means.

MR. TRUE: I know we included the onsite contamination. We basically replicated what 12-0157 did for the industry side of the cost. We didn't look at the accident side of the cost. We didn't look at regulatory costs and those kind of things. Those were usually much, much smaller anyway and they're not really our purview, but we looked at the onsite cost, the loss of asset and offsite consequences.

MEMBER REMPE: But you didn't really look at offsite costs, did you? If I look at page 25, it says metrics such as offsite population does offsite land contamination and population requiring relocations were considered, but they were already included in the model. I didn't quite understand that.

MR. TRUE: It's separate metrics. It's separate metrics. We didn't pull those out as separate metrics because they were already accounted for in that financial --

MEMBER REMPE: And so the tables in your report are just the differential costs because I thought somehow or other reading this that you were saying ah, the costs are the same for this for all

the offsite stuff.

MR. TRUE: Yes, so the financial consequences are all the accident-related costs, onsite and offsite. And for each alternative where we call a MACR which is a maximum averted cost risk. It's the total cost risk associated with having that alternative. And so we have a base case alternative and then we have enhancement alternatives. And the difference between the MACR, as we call it, the delta MACR is basically the financial benefit gained by that alternative, but we put it in terms of this maximum averted cost risk.

The statement you read was saying when we thought about what metrics we wanted to look at, we didn't pull those out because it would effectively be double counting since they were already accounted for in the financial consequence risk metric whereas latent cancers are not accounted for in the financial consequences with doses, but not latent cancers. Initial containment failure probability is implicit, but it's not explicitly called out. So we thought these were a set that provided the whole landscape of the accident, but not double counting.

MEMBER RAY: Doug, let me ask a simple question. We've got to move on here, but would your

analysis produce a different result at a site that was surrounded by high-value land, residential areas, what not, as opposed to one that's out in the middle of nowhere?

MR. TRUE: Sure. The method that we use, if you do it on a plant-specific basis, in fact, Jeff could talk for a little bit about this, when we do SAMA analyses for license renewal, you have to do it on a site-specific basis and you have to look at the land values around that particular site. And Indian Point has a lot higher value than South Texas, for example, or Cooper.

MEMBER RAY: San Onofre versus Palo Verde, for example.

MR. TRUE: Same kind -- those would be accounted for.

MR. KRAFT: Friends of mine who live there would object to being told they lived in the middle of nowhere, Harold.

MEMBER RAY: Well, let's not go there, but nevertheless, I'm trying to find two things. So often the results are the same regardless of the site and I'm just trying to find out if you're --

MR. TRUE: We did it for a site, for the Peach Bottom site for the Mark I. We did do some

population variances. We didn't do offsite property value variances. It wouldn't be hard to extrapolate from those. Part of the issue is we got to the point where the margins were so large on latent cancers that we, too, sort of thought spending a whole lot of time on the financial consequence side --

MEMBER RAY: Well, I just have a problem when the answer is always the same regardless of the site and when I instinctively think that well, maybe it would be different at one site than at another site. That's all. Let's leave it and go on.

MR. TRUE: The other considerations were things we addressed more qualitatively and each of the alternatives tried to look at what does this alternative do to defense-in-depth, what does it do to helping us control containment temperature, because that was a big issue when we were doing this work in parallel with the order. We were trying to make sure we were influencing the order properly. What role the human actions played. What's required for instrumentation and release control and hydrogen perspectives as well.

MEMBER BALLINGER: This may not be the right time to ask this question, but along with Harold was coming up with at Fukushima, has anybody done a

check to see, a calibration point on the evacuated area when you consider people -- you do a calculation on dose, but then there's an issue of the people themselves not wanting to come back which increases the area it affects beyond the area where you would get a dose that would mean something. So there's some human factor going on here. Has anybody ever done a sort of calibration based on what happened at Fukushima?

MR. TRUE: On the dose side of it or the financial --

MEMBER BALLINGER: The populations are based on -- latent cancer risks are based on dose calculations. And they're established in laws and things like that, the numbers that you use. But in the case of Fukushima, there's a plume where people are evacuated.

MR. TRUE: Right.

MEMBER BALLINGER: But has anybody considered the fact that there may be a larger area or that area may be habitable in terms of a dose, but uninhabitable because the people simply don't want to come back.

MR. TRUE: Refuse to come back. I don't know of anybody investigating that significantly.

MEMBER STETKAR: Part of the staff is in the back end, I don't remember what section, I think section 5, addresses the notion in the sense of a cost. They say well, the project costs of the Fukushima accident may be -- I don't recall, five or six times higher, but the staff hasn't dug into why that might be, whether it's more people moved. They don't know. They at least mentioned that there is some anomaly at least between the government's estimated cost for Fukushima, long-term costs, versus the metrics that you quantify from the staff's evaluation.

MR. TRUE: That's true. There is a gap.

MEMBER STETKAR: Why that is I'm not sure anybody knows exactly.

MR. TRUE: Yes. It's come up on a number of other reports. NAS brought it up and ASME did too.

We've shown these before. I don't know in the interest of time whether this is worth a whole lot of time, but basically this just tracks where all the risk goes because once you start with core damage, it all ends up somewhere in a release path and works its way through our core damage of entry and accident progression of entry.

The key things here are that we have a pretty substantial contribution and the length of the bar is basically the proportion of the risk that's coming from that. That's going from early RCIC failures. Those are failures where RCIC curves prior to the time when FLEX could be deployed to successfully prevent core damage.

MEMBER CORRADINI: And when you choose that it's based on just temperature? What is the B-

MR. TRUE: It could be a failure of RCIC itself because it was out of service, it didn't start, it didn't run, it was damaged by the seismic events, whatever the contributions are.

MEMBER BALLINGER: Oh, by the way, the fact that those folks left, it's not because they didn't like you, there's another meeting they have to go to that they're obligated to.

MR. TRUE: Okay. I'm sure John digested the reported anyway.

CHAIRMAN SCHULTZ: We will continue here.

MR. TRUE: But the most important thing and this came out of the NRC, if you don't have water in the base case, then you're going to end up in a liner melt or over temperature situation. When we go and add water to the mix through either RPV or drywell

injection, then we see a proportion of more of the release is going out the vent. In this case, initially, the wetwell vent. It might be followed by a drywell vent later.

We still have a little fraction of over pressure failures where the vent didn't work for whatever reason, a passive event might help address that little piece that has an over pressure failure. But it's a small fraction and then it's still a substantial fraction of liner melt scenarios coming through the end state.

So if you talk then about NRC's alternative for it, it's only this fraction the WW fraction that actually the filter does any good for. And that's why you see in the risk results you see it come down and sort of asymptotically stop is that this other chunk down here where the vent didn't work or the liner melted, those are still there and the filter isn't doing any good or the suppressible is not doing any good because the source term is going out a different pathway.

The more conservative you get in the way you treat human actions and in addition to water and use of FLEX, the smaller this wetwell vents piece becomes because if you're saying the operators aren't

going to put the water in, then you're going to go to liner melt. The operator is not going to attach the pump. You're going to go to liner melt. All those reasons push you in the direction of there's no water and you're going to have a high temperature failure like we had in the base case.

So we tried to balance that by having reasonable kind of failure probability for the human events in the regime where we had the equipment available. There were certain events where we knew we couldn't mitigate because the damage was sufficient or the timing was sufficient that we wouldn't be able to use the equipment, but we thought by sort of using more reasonable value around point 1 for the human actions to line the water, actually would show a larger benefit from the different alternatives rather than a lesser benefit.

I hope that explains the strategy of human events. I'm not going to get into all the details of this anyway.

So this result looks at the conditional containment failure probability and basically from that last chart all these bars down here that remain as containment failures, those are the liner melt and over pressure failures. The water addition brings it

down and these portions that drop are where the vent is effective in protecting the containment.

We do see some variability, but it is a pretty small variability from case to case. The case with the passive filter only down here actually improves the conditional containment failure probability because it is a passive event. We're no longer relying on the operator to open the vent. However, with this metric, because it's conditional on core damage, doesn't show is that that the core damage frequency actually went up because if you have no manual event and it's only a passive vent system, then you don't have anticipatory venting and you're going to lose RCIC earlier which is going to make you more dependent upon the portable pumps to prevent core damage.

It wasn't necessarily an obvious insight that came out of it, but when we went to go model, we realized that we would be driving ourselves to the portable pumps more commonly in those cases than when we didn't.

This one is just similar to the one that the staff presented in various forms. It shows the latent cancer risk, the QHO value. Our results are shown down here. Again, not a lot of variability in

those results. We can stop and talk about why the filters don't show a significant difference in a second here. But then I kind of overlayed on top of here what the NRC's 95th and 5th percentile values were from their analysis. And basically shows we're right in the same range as the NRC staff, even though we used different methods and different assumptions about equipment and humans and other contributors we ended up essentially the same basic region for our results.

So let's talk about the differences for a second and why, I think Hossein actually explained it right, but we'll just reiterate that if you're looking at the filter cases, you're basically comparing a 2A case here which is RPV SAWA, so water into the vessel, to a case over here, 4A, which is water going to the RPV with a small filter or a large filter which is the 5A, alpha, case.

So there's a little bit of a difference there. It's a fraction of a -9 difference. Calculationally, in the source term, the reason that we don't see as big an increase -- or decrease with the filter is that the suppression pool, the way MAAP models the core melt progression, the suppression pool captures more of the fission products and the

large amount of the hydrogen are generated by the melt core analysis during the end vessel phase drive them to vent sooner. That's what Hossein was sort of explaining was that with melt core they generate more hydrogen. You have to vent the containment sooner in certain scenarios and that causes more releases out the vent pathway as well as the suppression pool not being quite as effective because the temperatures or gas coming out of the SRVs are also higher.

Jeff, is there anything you want to add? Does that make any sense? You were the one that was asking about that difference.

This is another one that we put together that basically shows that when we look at the pre-Fukushima condition we didn't have any FLEX capability versus the latent cancer fatality risk essentially what becomes our base case now with the benefit of FLEX we get a substantial reduction.

If we added a filter on just the FLEX, it really doesn't give us any benefit because the containment is going to fail due to the liner melt or high temperature failures and it's going to be bypassed.

We get incremental benefits associated with the SAWA capable event which is the staff's

alternative 3, I guess, arguably 2, too, but 3. And then if we add the filter on top of that we get a very small incremental benefit out of that as we just showed. So basically, what's happening is we're reaching this point of diminishing returns that FLEX was a substantial safety benefit which is what we all believed it was going to do for us and then the severe accident water is another fairly substantial, but beyond that there's not a whole lot because we're going to left with these scenarios that basically have no means to mitigate. At some point the damage is sufficient enough that you're not going to be able to mitigate the scenario.

MEMBER RAY: As measured by relative LCF.

MR. TRUE: Measured by latent cancer risk.

MEMBER RAY: Right.

MR. TRUE: Financial consequences, so this is the MACR term that basically the present value of the averted cost weighted by the frequency of all the scenarios, and you see a drop with the water, and then there's some variation across each of the cases associated with the different alternatives. But again, the biggest benefit is by protecting the containment, providing the water to keep the

temperatures down, and protect the liner. After that, the influences are smaller. So this is the absolute value of the MACR; if we look then at the delta, you would compare the applicable base case, and we have sort of two base cases because of the order of our two possible outs end states. One was no dry wall vent, and one with a dry wall vent. Then the delta, because the next slide that shows as compared to the base case, what's the change in financial, so sort of the value of the different alternatives. And again, those deltas are all pretty much in the same range around the less than $1 million, and whether we add a filter or not, in our analysis it doesn't really significantly change that. How we manage the accident in terms of vent cycling and water management versus no water management, it's all kind of noise in the overall equation.

We did do a pretty large number of some study cases, and Jeff and his guys ran something like 12,000 MAAP and MACCS runs just to do the base case, instead of running through 24 scenarios and 509 end states, but then we went back and said okay, we should probably look at the plant to plant variability, because we know that Peach Bottom was not necessarily representative of all the Mark I's. Some

phenomenological sensitivities, and then there are some benefit model sensitivities that affect the overall--the cost, and the cancer metrics.

So I'm not going to go through all these; they're in the report. We basically tried to summarize for each of these different sensitivities. Whether we did it qualitatively or quantitatively, which if we did it quantitatively, which metrics were used, which alternatives were looked at, we probably only focused on the base case, the severe accident water to the RPV, and severe accident water to the RPV with a filter, and both of those also with a vent of course. So we were focused on just seeing for those particular cases, did we see big shifts, and it allowed us to look at the filter implications as well as the water implications case by case or sensitivity by sensitivity.

So in this slide, the deposition rate is something that changed in the recent years from pre-SOARCA to SOARCA where deposition rate of aerosols was modified by 10 percent impact overall, and they looked at that for the base case RPV injection of severe accident water, dry wall injection is 3 Alpha, and then 5 Bravo is the dry wall and filter. And we looked at latent cancer risk, the MACR and the delta

MACR to see if that had any significant effect. There are all kinds of charts and stuff that go through each of those and explain this.

The bottom line was that we didn't really see that any of the sensitivity cases changed the fact that there were substantial margins, whether you're looking at it from a latent cancer risk perspective, or even financially had effects of maybe a factor of two or something, but given the margins, we were substantially below those. But we did confirm many of the things that the staff previously talked about today, that you know, the operators are essential in managing access, whether it's operating the vent or pressurizing the vessel or lining up the severe accident water or surrounding the power supply with portable power device effects, the operators are always going to have an important role in accident management. Water addition was shown of course to be the biggest factor, just like the staff showed.

There is an incremental benefit to engineered filters for that set of scenarios where the filter is effective, but there's also a residual portion where engineered filters aren't going to be effective, so they're not a panacea. And the idea of a completely passive system that can contain an event

by itself and the operators didn't have to do anything as failsafe just wasn't really actually true; it actually increased the likelihood of having a core damage event and had no significant impact on the overall latent cancer implications.

MEMBER RAY: Well Doug, the possibility of a self-actuating vent and an operator manual vent, I mean you have one or the other but not both is what you're modeling here, right?

MR. TRUE: Yes, well we actually analyzed all the different flavors because we had an error for the operator not closing the vent again after they'd opened it, so those were built into it, but not -- we did not do -- oh wait.

MR. KRAFT: We did. We looked at a bypass.

MEMBER RAY: Well just the way you were saying it, again, I don't want to divert things here because of the time, but the way you were saying it made it sound like you either had a manual or an operator-controlled vent or a self--

MR. TRUE: We had--we did have a case where we had a manual pre-core damage. So prior to core damage, the operator did it, and then after core damage it was passive. We did look at that.

MEMBER RAY: Which should be some parallel arrangement

MR. TRUE: Yes. Yes. And that would have been this, so it actually--we'll do it in latent cancer maybe. So it basically shows up the same. All of their cases came out pretty much the same, irrespective of how we managed the vent.

MEMBER RAY: Well I wasn't talking about filter, I was just talking about manual versus--

MR. TRUE: Yes. Well we looked at the passive when we had a filter. We didn't look at a passive without the filter.

MR. KRAFT: The reason that it became very important to look at this because in the SECY 1201.7 evaluation, the staff focused on core passive filter, and we couldn't get a true understanding of what they meant, because we couldn't figure out when you looked at the systems that were installed in European units, is that operators have to close the vent, they have to drain the filter, you know, it's a complicated system; what do you mean by passive? What we ultimately extracted was they meant there's a rupture disk, and when you over-pressurize, the rupture disk blows. But operators don't walk off the site and leave it alone; they still perform their

function. And so that's what was finally modeled, and what we were looking at--because what was lost I thought--we thought in 1201.7, and it's actually a non-intuitive result. Is this--the total passive event shows an increase in the CDF because you lose the ability to do the anticipatory venting, which is part of the problem.

MEMBER RAY: Yes, I understand what you're saying, Steve, I just don't know that it's that clear, but that may be just because we're rushing here, and I haven't studied this thoroughly enough. But anyway, the upshot of it is we only have this dual arrangement associated with a filter?

MR. LEWIS: With a filter, yes, for RK 6 Bravo.

CHAIRMAN SCHULTZ: Doug, before you go to the next slide conclusions, what I'm gathering is that the--all the work that was done here reinforced what had already been learned with regard to water addition and its benefits. Any particular insights that amplified the overall approach for water addition, water management based upon the work in the last four or five months?

MR. KRAFT: Steve, I think it was the other way around. I think what happened was if I

remember the way we did this in the industry, it was as we talked about in the last meeting on the vent order, the vent order was in its pure sense, you know, so much mechanical/electrical engineering. While EPRI was doing what we used to call filtering strategies assessments, what came out of that is the value of water. So we more or less said hey, what if we put the water in, and that's kind of the way the industry saw it. So I think the--forgive the pun--flowed backwards in the process. So I don't think there were additional insights, it was the original insights that prompted us to suggest to go into the vent water.

MR. TRUE: Yes. We had the original EPRI work from, when was that, 2013 or something that showed the filtering strategies. Then when we started down this rulemaking path, actually industry sort of got out in front on the probabilistic model and by late 2013, we had a beginning of a probabilistic model that we shared with the staff. Marty took that, built his model which is much more complicated than our model and more detailed, and did a great job at extending that method. But in that process of doing our initial analysis with the probabilistic model, we realized most of these same

insights were available at that point. We went back to the industry and said you know, this really looks like water is the thing when there's a probabilistic case to show that this is probably going to be the best thing for us to do. So then it got fed back into the order, and then we came on later and published our report that showed that, so timing was sort of--

MR. KRAFT: And to a question that John was raising about when he was questioning the staff on their analysis in terms of we think the issue is going to do X, we have verbal -- an understanding with the CNOs and the BWR Owner's Group that yes, water addition is definitely going to be done. They understand the way we wrote the guidance, and that water management is definitely going to be done at all the units. But again, you won't see that officially until the OIPs are submitted at the end of the year. But we have high confidence that's going to happen, in fact, I was at a BWR Owner's Group meeting a couple of weeks ago, and we have formed like an assist program where we will have members of the Owner's Group and a couple of other people meeting regionally with the plants, and they'll present to us here's what we're thinking of doing, and if they're

going to start deviating from that guidance, we're going to want to know why, because their CNOs have made a commitment and we think it's important that these plants install what is actually needed and operating in the proper way. So we're kind of trying to make sure there's a little bit of discipline in the industry on how we go about doing this.

CHAIRMAN SCHULTZ: Good time for my question, and that is from what you've just said, you don't anticipate any deviations from the OIPs that the proposals that are going to be coming in at the end of the year based upon the discussions that are ongoing in this area?

MR. KRAFT: Okay, so officially, I am bound to say our guidance is only one way to do it, and industry, members of the industry, licensees have to come to the NRC if they want to do something different. They've got--the burden is on them to explain it, but I know the NRC staff has made clear they're--well wait a minute, what's wrong with the guidance and why don't you want to do that. Where you might see deviations might be in some of the details, how you do a rad calc or how you make an assumption about an operator doing something; it won't be in the overall concept, you know, water

addition, water management. I don't think you'll see deviations there; that's my guess. But again, you know, can't really say 100 percent until we see those--the best I can tell you, Steve, is in my expectation, everyone will do what the guidance says. That's the best I can tell you right now.

MR. LEWIS: Well just to wrap up this discussion--

MEMBER BROWN: Before you do that, if I could, let me ask the question another way. If the rule came out not option three, alternative, only if you did with alternative two with the effect of generically incorporating 109, do you still feel--I mean your guidance would still be out there, but yet the rule would not be requiring it, and therefore there would have been backfit analysis that goes along with it I guess from the staff or NRC standpoint. Well, the worry is that if you don't do that, then owners will say well, they're not asking for that, so we'll just do the one thing only. That's the concern that I think John was trying to express. He's not here, so I'm just trying to--

MR. KRAFT: And reading in between the lines of the SECY, I think that's a staff concern as well. The answer is that they do a rule--if you're

to do option two, there would have to be guidance. So we would assume, right along with option two coming forward as a rule, there would be a draft guide much like they did in the mitigating strategies we're making where the draft guides were re-endorsements of NEI guidance. So we'll see a re-endorsement of NEI 13-02, some combination of the two ISGs that would lead to the same thing happening as we're doing in the vent order as through the rulemaking. So that could be pathway to accomplish exactly the same thing. And frankly, eliminate the burden on everyone of doing a backfit analysis on something that well, is it going to work, not work, what are you basing it on, you know. To me, what we're stuck in here is a situation where everyone understands what the good idea is, now how do we get through the wickets of what amounts to a regulatory process to get there really is what the--I think the question before the house is.

MEMBER BROWN: I just wanted to ask the question--

MR. KRAFT: Good question, Charlie. Absolutely.

MEMBER BROWN: Coming through the back door here.

MR. KRAFT: Coming through the back door.

MR. LEWIS: These conclusions ought to look fairly familiar, because they're very similar to the ones that Marty had earlier today, so even though there are differences, many differences in the details of the two sets of studies and MELCOR map predict somewhat different accident progression for core damage accidents and so forth, the bottom line really is very similar, and that is that providing the ability to put water into the containment on the debris, whether it's through the rec pressure vessel or directly into the dry well and venting the containment gives you the best option for preserving containment integrity in the longer term and preventing the really serious releases. And if you do that, then the incremental benefit of engineered filters is really fairly small.

The one thing that our study perhaps focuses on a little bit more is the significance of the manual actions that will need to be taken to respond to these accidents effectively. You will need manual response to make water available to the core debris and to manage the venting systems, but overall I think you'll see the conclusions are very similar at that level at least.

MEMBER RAY: These are risk-based

conclusions, aren't they? I mean we've always been talking in risk-based when we talk about something being negligible and so on?

MR. LEWIS: Well, I think these guys who did the calculations can speak to them better than I can, but I think that the--it's supported both on an engineering qualitative basis and a risk-informed basis. You can look at it in a variety of ways, and you come to essentially the same conclusion. What it doesn't account for perhaps is a public perception or a policy relating to the--

MEMBER RAY: Well okay, we'll leave it, but I mean I could go back and look at some of the tables here and say how the heck do you conclude that's a risk or that's a negligible difference? The way you do it I think is by saying that it's risk-based; on that basis it's negligible but in an absolute amount, it's trillions of dollars for example in that one table. That's not negligible by any stretch, but I'm just saying, and I think Doug you'll agree with this, we're talking a risk-based judgment then it's negligible.

MR. TRUE: Yes, it's consistent with the backfit analysis approach of risk basing the monetization.

MEMBER RAY: I just want to make clear that if we're saying something is negligible that, you know, without consideration of risk appears huge that the reason that we say it's negligible is that it's based on a risk assessment.

CHAIRMAN SCHULTZ: Any other questions for industry? Thank you. At this point, appreciate the presentations and the discussion we've had today. At this point, I'm going to turn to public comments, and I did want to note that we, as I did at the beginning of the meeting today, that we have received written comments and attachment packages from two members of the public, from Mary Lampert and from Paul Gunter, and those comments and packages will be part of the record for this meeting. But I would like to provide opportunity for public comments here and now. I'd like to have the phone line open; it's open now, but I'd first like to ask for comments coming from the audience in the room here today. Paul, please introduce yourself and--

MR. GUNTER: Thank you very much. Paul Gunter with Beyond Nuclear. I really appreciate your patience--

CHAIRMAN SCHULTZ: And yours, too.

MR. GUNTER: --giving the extension here.

You know, I think that I have to start out by saying that we categorically disagree that the difference in having a filtered vent is negligible. Given the broad uncertainties that have been addressed here, but more particularly what our concern is that we're missing an opportunity as the public to address through a formal public rulemaking our differences through our experts and through our comments and frankly, we're being denied standing more importantly. So in our view, this is sort of like a game of keep away between industry and staff, where the public sees this bouncing around but no real opportunity, no real standing to address a lot of uncertainties, more particularly as has been outlined with human reliability issue. You know, that's just one point.

You know, we wanted to argue for the inclusion of a qualitative analysis. That--our opportunity would have been afforded in a rulemaking process, but again, we're being cut out of that part of the proceeding, and I hope that the ACRS can provide some reconsideration to the importance of a more deliberate, independent and inclusive process for these Mark Is and Mark IIs which have a broad history where the public has been denied either by 10

CFR 50.59, or other ways that this has been a game of keep away so far. You know, I provided the ACRS with an NRDC report, Natural Resources Defense Council, regarding hydrogen explosions and severe nuclear accidents, unresolved safety issues involving hydrogen generation and mitigation.

The fact that contained protection and release reduction has relegated hydrogen control to a tier three is a real concern given that what the NRDC report documents, there are national lab studies, the IAEA has demonstrated that the NRC is under-predicting hydrogen gas generation, particularly with regard and concern for these large inventories of zirconium in the reactor, the BWR cores that hydrogen generation is initiated by water and steam introduction. But that is a big concern that that's not, again, that's not being addressed.

It's things like this and others that affect design parameters for these hardened vents, and it's our concern that as we proceed, we're going to wait and see, but it's a real concern that we're going to be looking at a wide variance of design reliability, and basically a repetition of Generic Letter 89-16, where you have some plants where the-- one in particular where the success path is an

explosion in the auxiliary building at the standby gas treatment system that vents at ground level, and that's considered a success path right now before the U.S. NRC. That does not carry public confidence. Thank you.

CHAIRMAN SCHULTZ: Thank you, Paul. Any other comments from the audience in the room? Seeing none here, I'd like to go to the phone line and ask if there's a member of the public on the phone line who would like to make a comment, please--

MS. LAMPERT: Yes, I would.

CHAIRMAN SCHULTZ: --yes, thank you. Please introduce yourself for us.

MS. LAMPERT: Yes, this is Mary Lampert, Director of Pilgrim Watch, watching nuclear power plants. My comment is this, the staff issued the May 15 draft analysis in an effort to reverse course and justify not adding a filter. Even pretending for a minute that the staff's bogus analysis is without flaws, their own analysis shows adding a filter before provides the most bang for the buck.

The staff estimated that a filter costs between $11 million and $64 million in the May 15 draft. So go to the middle, call it $30 million. The savings according to the staff in offsite costs

would be $3.51 billion; that's 100 to one. Any honest cost benefit analysis that used computer code, that had assumptions post-Fukushima, would come up with 1000 to one. Adding 100--admitting 100 to one and not recommending a filter is mind-boggling. With 1000 to one or any over 100 to one is criminal. How would an honest assessment bring it way about 100 to one? Here are a couple of examples.

Where the assumptions in the analyses and the assumptions in the outdated computer codes used, MELCOR, MACCS and SOARCA are bogus, serving only to assure that licensees do not have to pay to put a filter in, because they already said they're going to go along with water. First, the probability of an accident is underestimated in consequence codes, and that is important because probability is multiplied by the consequences, and if the probability is unrealistically too low as it is, then it trivializes costs. Fukushima raised the probability of an accident's baseline more than 10 times, from one event per 31,000 reactor years to one event per 2,900 reactor years. Fukushima raised the number of actual core damage events in the last 36 years to five; that would mean one every seven years.

The probability is further affected by

the staff's assumptions that SOARCA will work six out of 10 times. I sent an attachment about Entergy's plans at the Pilgrim Station, which is known as the Rube Goldberg fix. Never to be relied upon, highly unlikely to work. The probability is further influenced by underestimating hydrogen explosions, as Paul Gunter pointed out. The PRA minimized health costs and over-estimated evacuation times by relying on bogus evacuation time estimates. I will send you a 2.206 that we filed showing the ETEs, at least for the Pilgrim Station, are worthless. If there was a change in the value of life that NRC uses to coincide with other agencies at $5 million to $9 million, as opposed to the $3 million, that could make a huge difference. Health impacts ignored, wrongly do not look at cancer incidence, they don't look at the other health effects from exposure in a severe radiological event, outlined by the National Academies in their 7.

Recent studies were ignored, such as the studies by Cardiff and the Techno River studies, which shows a greater susceptibility to lower doses. Indirect health costs were ignored, and there's no justification for the 2,000 person-rem conversion rate. Clean up and decontamination are the elephants in the room. This was hinted to in the conversations

before, which talked about well, what about the cost of land contamination? In the analyses, they came up with a very small number, I believe it was $12.9 billion out to five miles in the clean up and decontamination costs. However, the National Academies--relying on the Japan estimates. However, the National Academies came up with an overall cost of $250 billion. ASME has come up with $500 billion, and nobody seems to appreciate the fact that there's nothing that can be done with the waste, so how do you put a cost on that? I have three acres here, as an example.

Further, they limit the radioactive release concentrations by using a straight line Gaussian plume instead of meteorological models for complex terrains. If you consider the impact of the sea breeze effect or the impacts that you find along river valleys, there will be a larger area that will be impacted than simply using the straight line Gaussian plume. And further, by relying on the meteorological equipment on site, which tells where the plume is going on site, but it does not tell what happens to it when it gets off site. Further, they ignore additional economic consequences such as the economic infrastructure, which would be the clean up

of roads and public lands, they ignore the multiplier effect, averaging that is used is the mean and not the 95th percentile, which would be sensible.

Other offsite consequences ignored are aqueous discharges, they limit short term to cesium-137 and not a multiple of others. I could go on and on. The fellow David Shannon at Sandia who wrote the Fortran for MACCS2 is on record with the NRC saying unequivocally if you want to determine economic costs, don't use this code. It is not quality assured, despite the fact in the analysis the staff tried to whittle around and say it was when it's not, et cetera.

So what I'm saying is, there is--by going to simply a quantitative analysis, ignoring qualitative, which is raw, and doing a quantitative analysis using computer codes, MACCS2, MELCOR, SOARCA, that have pre-Fukushima assumptions and assumptions that they have such as embedding the Gaussian plume, et cetera, is going to come up with the answer apparently, at least the majority of the Commission wants and industry wants, not that they're distinguishable. And if the NRC thinks for one minute that they're fooling the public, they are wrong. And if you think this brings about confidence

in the regulatory agency that we're supposed to depend upon, it's wrong. It just feeds what we're seeing and hearing, the captured--the regulator captured by the industry.

And so let's turn this around. It is nonsensical to say that a filter is not going to be protective of public health and reduce offsite consequences. It makes no sense. It's apparent. That's why you have filters on normal releases, so not to have it on an accident release, coming from containment, boggles the mind. And so what is done is a game with numbers; however, all the assumptions lead only to the answer you want, the answer that defies common sense. I mean I really, I couldn't stand listening to this much longer. Thank you very much.

CHAIRMAN SCHULTZ: Mary, thank you for your comments. Others on the line, telephone line that would like to make a comment?

MS. THOMAS: Yes, this is Ruth Thomas.

CHAIRMAN SCHULTZ: Hello, Ruth. Please make your comment.

MS. THOMAS: Yes, I support everything that Mary said, and it's a melody of manipulating numbers and facts and what facts, just manipulating

to come out with a conclusion that the NRC and the industry--I know she said that a are one and the same seems like, and not in the public's interest. And this is violating the National Environmental Quality Act, it's violating the charter of the Nuclear Regulatory Commission, and as with Mary, it's so overwhelming to listen to this going on, and realize that this is happening in America. This is happening, and worse is going to happen, and it's not scientific, it's--I don't know whether the people making the decisions on this are going to create a Barry Commoner, but it's the idea that everything is connected to everything. Everything has to go someplace. Nature is always right, and the materials that are being used are not in nature, they're man-made. And it's--all I can say.

Things are so insecure, I pray that the people making the decisions will realize what they're doing; they're bringing us closer and closer to destruction. I'm with an organization Environmentalists, Inc.; we've been studying and trying to communicate with the officials making these decisions and with the Nuclear Regulatory Commission, Department of Energy for over 40 years, as have many other organizations and scientists and doctors, and

I just pray that this will be turned around because we're going in the wrong direction with it. Thank you.

CHAIRMAN SCHULTZ: Thank you for your comments, Ruth. Are there others on the line who would like to make a comment? If so, please introduce yourself. Hearing none, I'll close the phone line and ask members of the Committee if there are any comments that they would like to make for the record at this time? One of the things that I want to bring to consideration of the full Committee is our schedule. Right now, based upon what we had set as of our last meeting, we are scheduled to meet again with the staff in October.

The papers and information that we have received, the SECY document and the attachments as well have been provided now as you've heard, they've been provided to the public, they're out for public comment and our approach is that we are wanting to wait for the end of the public comment period, for the staff to receive those public comments, for the staff to develop the responses to the public comments, and they would be able to bring that information to us, the comments as well as their responses and any changes they would make to the overall approach

recommended to us in October. And then we would write a letter to the Commission in our full Committee meeting in November. So that's the current schedule, and the question is would anyone recommend or we can certainly discuss at the full Committee meeting this week as to whether that should change. Are there other comments from the Committee related to the presentations today? No?

MEMBER RAY: None other than what I've made earlier which I would reiterate, but there's no reason to.

CHAIRMAN SCHULTZ: Dick?

MEMBER SKILLMAN: None, thank you.

CHAIRMAN SCHULTZ: Okay. John?

MEMBER STETKAR: Nothing on this, thanks.

CHAIRMAN SCHULTZ: Okay. Ron? Charlie?

MEMBER BROWN: No further comment.

CHAIRMAN SCHULTZ: And Joy?

MEMBER REMPE: No further comment, but I appreciate this time to--

CHAIRMAN SCHULTZ: And again as I mentioned previously, I appreciate the presentations today by the staff, as well as from the industry, and appreciate the public comments as well. Mary, if you're still on the line, I did want to mention to

you that you indicated that you'd like to send in an additional document, and you would send that along the same pathway that you did previously. We received that, and all the information that's been provided by Paul Gunter and Mary Lampert have been provided already to the Committee. So if you'll please send that to the emails that are provided in the agenda for the meeting, that information will be included in the record if it comes in soon, that is in the next day, and if it does not, it will certainly be provided to the members of the subcommittees. So thank you very much. All right, with that then, I will close this meeting.

MEMBER STETKAR: Okay, one point we've decided that the next subcommittee meeting is starting at 1:30, so members be back for that subcommittee meeting at 1:30. Sandra, I will be informed of that.

(Whereupon, the proceedings were concluded at 12:59 p.m.)

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ACRS Subcommittee Meeting: Containment Protection and Release Reduction (CPRR) Rulemaking

July 7, 2015

NRC Staff Presentations

- Introductions of NRC Staff
- Staff Overview of CPRR Regulatory Evaluation
- Staff Overview of CPRR Risk Evaluation
- Staff Overview of CPRR MELCOR Analysis
- Staff Overview of CPRR MACCS Analysis

CPRR Draft Regulatory Basis: Regulatory Evaluation

Robert Beall
Office of Nuclear Reactor Regulation
July 7, 2015

Background

- Commission Paper SECY-12-0157
  – Recommended filters based on qualitative considerations.
- Commission Direction in SRM-SECY-12-0157
  – Installation of severe accident capable vents (Order EA-13-109).
  – Ensure that performance and risk of filtering strategies and filters are fully evaluated.
  – Fully explore requirements associated with measures to enhance the capability to maintain containment integrity and to cool core debris.
  – Examine multiple performance criteria.
  – Any policy issues should be raised to Commission.
  – Developed a separate paper on the use of qualitative considerations.

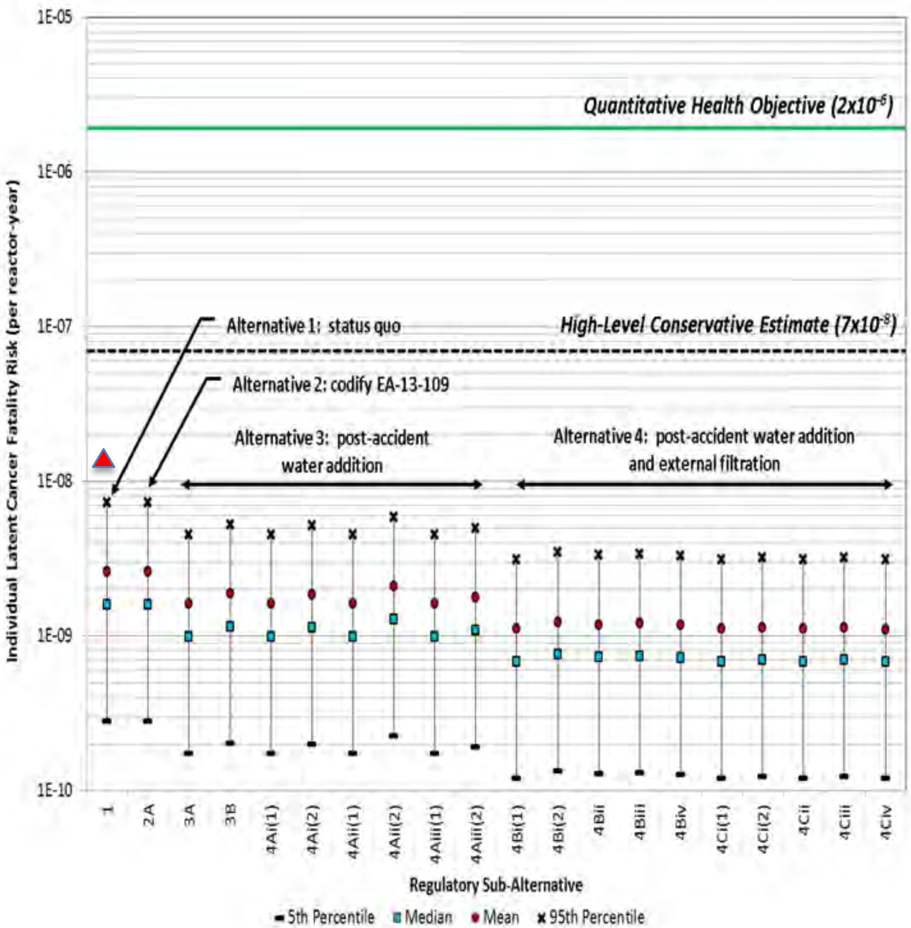
Recent NRC Actions

- The staff's analysis and the Commission decision on COMSECY-13-0030, "Expedited Spent Fuel Pool Transfer" reaffirmed the Safety Goal Policy Statement on Quantitative Health Objective (QHO).
  – Risk was low enough (i.e., less than QHO) that it did not rise to a "substantial safety enhancement".
- The staff issued SECY-14-0087 on Qualitative Consideration.
  – The Commission SRM did not authorize an expansion of the consideration of qualitative factors in regulatory analyses and backfit analyses.

Alternatives Evaluated

1. No action/Status Quo (i.e., no rulemaking)
2. Over-pressurization Measures
   – Codify Order EA-13-109
3. Additional Containment Protection (CP) Measures
   – External water addition via reactor pressure vessel or drywell.
4. Additional Release Reduction + CP Measures
   – Filtration Strategies
   – Small engineered filter
   – Large engineered filter

High-Level Conservative Estimate Calculation



= Expedited Spent Fuel Pool (conservative estimate)

- Evaluated all BWRs with Mark I & II containments by:
  - Using the highest ELAP frequency value (7.4x10-5 per year),
  - Using the highest conditional ILCF risk factor (2.2x10-3 given an initiating event), and
  - Using a FLEX equipment success probability of 0.6 per demand following core melt.
- This yields a conservative high estimate of 7x10-8 cancer fatalities per year.
- No credit taken for any of the CPRR alternatives.

Containment Protection

- The industry's approach to complying with Phase 2 of Order EA-13-109 is to incorporate the addition of external water during a severe accident.
  - External water reduces the likelihood of containment failure from over-pressure.
  - External water also reduces the likelihood of failure from over-temperature or liner melt-through.
  - External water may also make the need for a drywell vent unlikely by maintaining the capability to vent from the containment wetwell (SAWM).
  - Venting from the containment through the wetwell reduces the release of radioactive materials because the water in the suppression pool scrubs or filters the release.

Release Reduction

- The staff assessed possible requirements to reduce or filter planned releases during containment venting following core damage.
  – Filtering strategies using containment wetwells and existing plant features.
  – A large engineered filter.
  – A small engineered filter.
- Engineered filters show advantages when evaluated against the offsite consequence measures.
- The potential benefits of using engineered filters to reduce the release of radioactive materials during containment venting does not meet the quantitative criteria for providing a substantial safety improvement.
  – The cost of the engineered filters far exceed the calculated benefits.

Regulatory Evaluation

- CPRR alternative 3 would have the advantage of:
  - Preventing failure of containment structures and resultant uncontrolled release of radioactive material.
  - Make it unlikely the need for a drywell vent by maintaining capabilities to vent from the containment wetwell if using SAWM.
  - Addresses other containment failure modes (i.e., drywell head leakage and liner melt-through).
  - Has minimal costs because it is being implemented as part of the industry’s response to Order EA-13-109.
  - Codifies the external water addition requirement instead of relying on regulatory guidance like alternative 2 would need.
- Alternative 3 would provide the benefits of Order EA-13-109 for over-pressure protection (alternative 2) with the worthwhile benefits of capturing the over-temperature protection from external water addition and release reduction without the significant cost of engineered filters (alternative 4).

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Backfit Assessment

- The proposed CPRR rulemaking would codify:
  - The requirements in Order EA-13-109.
  - Add requirements for the use of water addition.
- The proposed rule codifying Order EA-13-109 would not constitute a backfit.
- The requirement for water addition capability could constitute a backfit.
- The staff will determine if a backfit analysis is required and ensure that any proposed rule is in compliance with the requirements of 10 CFR 50.109 during the proposed rule phase.

CPRR Regulatory Conclusions

- NRC staff has generated a Commission paper with preliminary conclusions, path forward, and a draft regulatory basis document.
  - Proceed with the CPRR rulemaking to make requirements of EA-13-109 (i.e., severe accident capable containment vents) generically applicable.
  - Require severe accident water addition as part of the response to Phase 2 of EA-13-109.
  - Discontinue other aspects of the CPRR rulemaking (i.e., engineered filters).
- The Commission has converted the CPRR information paper to a Notation Vote paper.
  - The staff expects an SRM later this summer.

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Backup Slides

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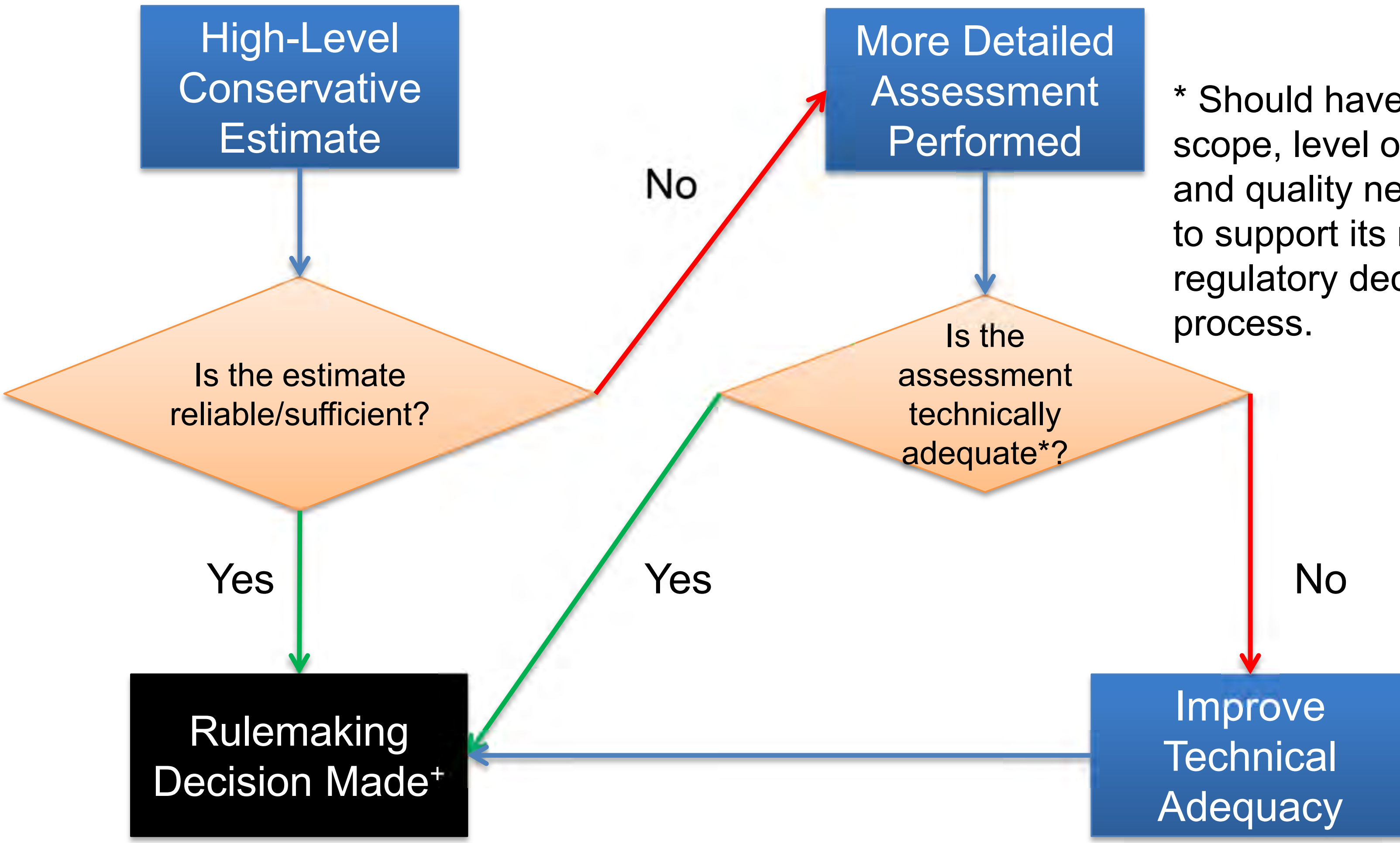
Containment Failure Modes by Alternatives

Reactor Conditions	Containment Failure Mode	Alternative 1 (No Action)	Alternative 2 (Rulemaking EA-13-109)	Alternative 3 (Rulemaking EA-13-109 with SAWA)	Alternative 4 (Release Reduction)
No Core Damage	Over-Pressure	Required	Required	Required	Required
Core Damage (Severe Accident)	Over-Pressure	Required	Required	Required	Required
	Over-Temperature	Not Required	Not Required	Required	Required
	Liner Melt Through	Not Required	Not Required	Required	Required
	Release via Controlled Venting	Not Required	Not Required	Not Required	Required

While not required, SAWA implemented as part of over-pressure protection provides collateral benefits of additional containment protection for over-temperature and liner melt through.

Process for CPRR Rulemaking Options for Disposition



* Should have the scope, level of detail, and quality necessary to support its role in the regulatory decision process.

+ Staff provides quantitative and qualitative information to Commission for consideration in their rulemaking decision.

CPRR Risk Evaluation

M. Stutzke, RES/DRA

Outline

- Definition of alternatives
- Logic model development
- ELAP frequencies
- Human reliability analysis
- Results
- Risk insights

CPRR Alternatives

SECY-15-0085	SECY-12-0157	Objective	Implementation Alternatives
1	Alternatives 1 and 2 - Provide capability for post-core-damage containment venting.	Prevent containment failure due to over-pressurization	• Location: ○ Wetwell ○ Drywell • Actuation: ○ Manual (operator action) ○ Passive (rupture disk)
2	Alternative 3 - Provide capability to cool core debris. Requires water injection pathway and containment venting.	Prevent containment failure due to liner melt-through	• Injection Pathway: ○ Reactor pressure vessel ○ Drywell • Injection Control: ○ SAWA (severe accident water addition) ○ SAWM • Venting – Same as Alternatives 1 and 2
3	Alternative 3 - Operate post-core-damage containment vents and water injection in a manner than enhances the containment's ability to retain and scrub radioactive releases.	Reduce the quantity of radioactivity released to the environment following a core-damage accident	• Vent operation: ○ OLO (open-and-leave-open) ○ VC (vent cycling) • Injection control: ○ SAWA ○ SAWM
4	Alternative 4 - Install engineered filters on the containment vents.	Reduce the quantity of radioactivity released to the environment following a core-damage accident	• Filter Capacity: ○ Large ○ Small

Logic Model Development

- Defined 20 regulatory sub-alternatives that capture variations within the 4 alternatives
- Logic model represents a generic BWR Mark I with a RCIC system
  - FLEX implementation varies by site and licensee
- One initiating event: extended loss of ac power (ELAP)
- Event tree models
  - Three core-damage event trees (CDETs) that address MBDBE strategies
    - Phase 1 – use of installed plant equipment
    - Phase 2 – use of onsite portable equipment
  - Six accident progression event trees (APETs) that address CPRR strategies
    - Post-core-damage containment venting
    - Post-core-damage water addition
- Note: engineered filters affect consequences, not sequence delineation or quantification

ELAP Frequencies

- ELAP frequency = frequency of SBOs (station blackouts) with duration > SBO coping time required by 10 CFR 50.63 (the SBO rule)
- Two contributors included:
  - Internal events as defined by NUREG/CR-6890: plant-centered, switchyard-centered, grid-related and weather-related LOOPs (loss of offsite power events)
  - Seismic events
- Site-specific information:
  - SBO coping duration: NUREG-1776
  - Number of onsite emergency AC sources: NUREG/CR-5500, Vol. 5
  - Seismic hazard: NTTF Rec. 2.1 information request (Columbia – IPEEE)
- Generic information:
  - EPS random and common-cause failures: NUREG/CR-5500, Vol. 5 (based on SPAR fault tree models)
  - Offsite power recovery curves for internal events: NUREG/CR-6890 (statistical analysis of historical data – used in convolution)
  - Seismic fragilities: aggregation (linear opinion pool) of IPEEE information
  - Note: Assumed 100% seismic coupling among redundant equipment

Estimated ELAP Frequencies

Site	Containment Type	EPS Class	SBO Coping Time (h)	ELAP Frequency (per year)		
				Internal Events	Seismic	Total
Browns Ferry	Mark I (RCIC)	4	4	4.1E-07	2.2E-05	2.3E-05
Brunswick	Mark I (RCIC)	2	4	6.9E-06	1.1E-05	1.8E-05
Columbia	Mark II	2	4	6.9E-06	5.0E-05	5.7E-05
Cooper	Mark I (RCIC)	2	4	6.9E-06	4.9E-06	1.2E-05
Dresden	Mark I (IC)	4	4	4.1E-07	1.7E-05	1.7E-05
Duane Arnold	Mark I (RCIC)	2	4	6.9E-06	2.4E-06	9.2E-06
Fermi	Mark I (RCIC)	4	4	4.1E-07	8.8E-06	9.2E-06
FitzPatrick	Mark I (RCIC)	4	4	4.1E-07	3.5E-06	3.9E-06
Hatch	Mark I (RCIC)	3	4	3.4E-06	9.6E-06	1.3E-05
Hope Creek	Mark I (RCIC)	3	4	3.4E-06	7.6E-06	1.1E-05
La Salle	Mark II	3	4	3.4E-06	3.4E-05	3.8E-05
Limerick	Mark II	4	4	4.1E-07	1.0E-05	1.0E-05
Monticello	Mark I (RCIC)	2	4	6.9E-06	6.7E-06	1.4E-05
Nine Mile Point 1	Mark I (IC)	2	4	6.9E-06	3.8E-06	1.1E-05
Nine Mile Point 2	Mark II					
Oyster Creek	Mark I (IC)	2	4	6.9E-06	8.3E-06	1.5E-05
Peach Bottom	Mark I (RCIC)	3	8	2.0E-06	3.8E-05	4.0E-05
Pilgrim	Mark I (RCIC)	2	8	4.1E-06	7.0E-05	7.4E-05
Quad Cities	Mark I (RCIC)	4	4	4.1E-07	7.0E-06	7.4E-06
Susquehanna	Mark II	3	4	3.4E-06	4.6E-06	8.0E-06

Challenges in Assessing Human Error Probabilities

- CPRR alternatives are conceptual - not currently incorporated in the EPG/SAGs or licensee programs
- FLEX implementation is flexible
- Lack of staff-developed HRA method that addresses post-core-damage operator actions at the conceptual design stage
  - SPAR-H is a HEP quantification approach that has been primarily used to estimate the probabilities of in-control-room, pre-core-damage actions
  - ACRS subcommittee meeting 10/15/2014 on the site Level 3 PRA project
  - ACRS subcommittee meeting 4/24/2015 on IDHEAS and the general methodology
- Scoping human error probabilities
  - In-control-room actions: 0.1
  - Ex-control-room actions: 0.3
  - Decision to use scoping HEPs is consistent with RG 1.174, RG 1.200, and NUREG-1855

Sensitivity of Core-Damage Frequency to Human Error Probabilities

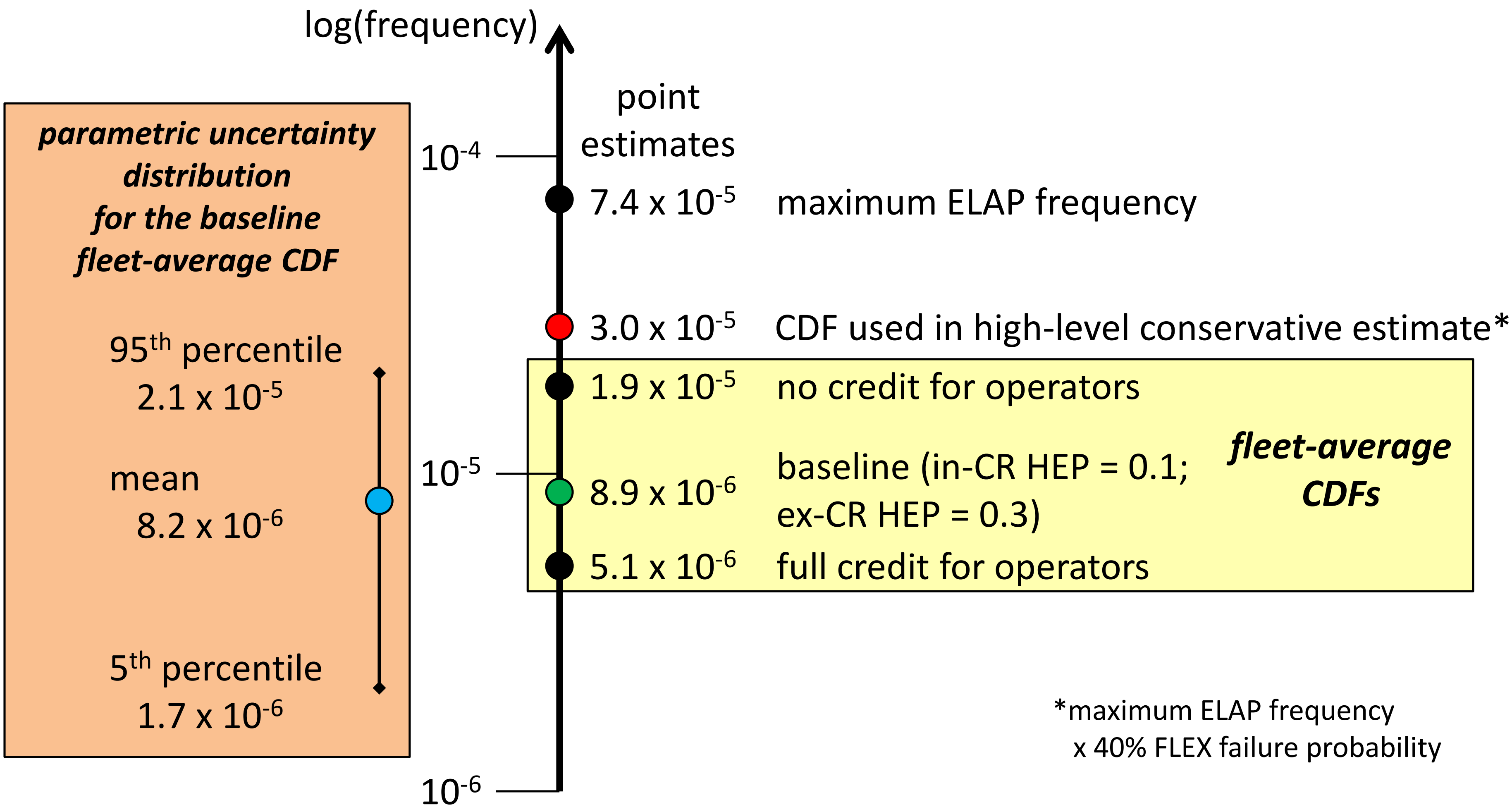
In-Control-Room HEP	Ex-Control-Room HEP													
	0	0.001	0.002	0.003	0.006	0.01	0.02	0.03	0.06	0.1	0.2	0.3	0.6	1
0	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.7E-06	8.5E-06	1.5E-05	1.9E-05
0.001	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.7E-06	8.5E-06	1.5E-05	1.9E-05
0.002	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.7E-06	8.5E-06	1.5E-05	1.9E-05
0.003	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.7E-06	8.5E-06	1.5E-05	1.9E-05
0.006	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.8E-06	8.5E-06	1.5E-05	1.9E-05
0.01	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.8E-06	8.5E-06	1.5E-05	1.9E-05
0.02	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.8E-06	8.6E-06	1.5E-05	1.9E-05
0.03	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.9E-06	8.6E-06	1.5E-05	1.9E-05
0.06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.2E-06	5.2E-06	5.4E-06	5.7E-06	7.0E-06	8.7E-06	1.5E-05	1.9E-05
0.1	5.2E-06	5.2E-06	5.2E-06	5.2E-06	5.2E-06	5.2E-06	5.3E-06	5.3E-06	5.5E-06	5.9E-06	7.2E-06	8.9E-06	1.5E-05	1.9E-05
0.2	5.5E-06	5.5E-06	5.5E-06	5.5E-06	5.5E-06	5.5E-06	5.6E-06	5.7E-06	5.9E-06	6.3E-06	7.6E-06	9.3E-06	1.5E-05	1.9E-05
0.3	6.0E-06	6.0E-06	6.0E-06	6.0E-06	6.0E-06	6.0E-06	6.1E-06	6.2E-06	6.4E-06	6.8E-06	8.1E-06	9.8E-06	1.6E-05	1.9E-05
0.6	8.7E-06	8.7E-06	8.7E-06	8.7E-06	8.7E-06	8.7E-06	8.7E-06	8.8E-06	8.9E-06	9.2E-06	1.0E-05	1.1E-05	1.6E-05	1.9E-05
1	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.4E-05	1.4E-05	1.5E-05	1.7E-05	1.9E-05

Only equipment failures:
CDF = 5.1E-6/y

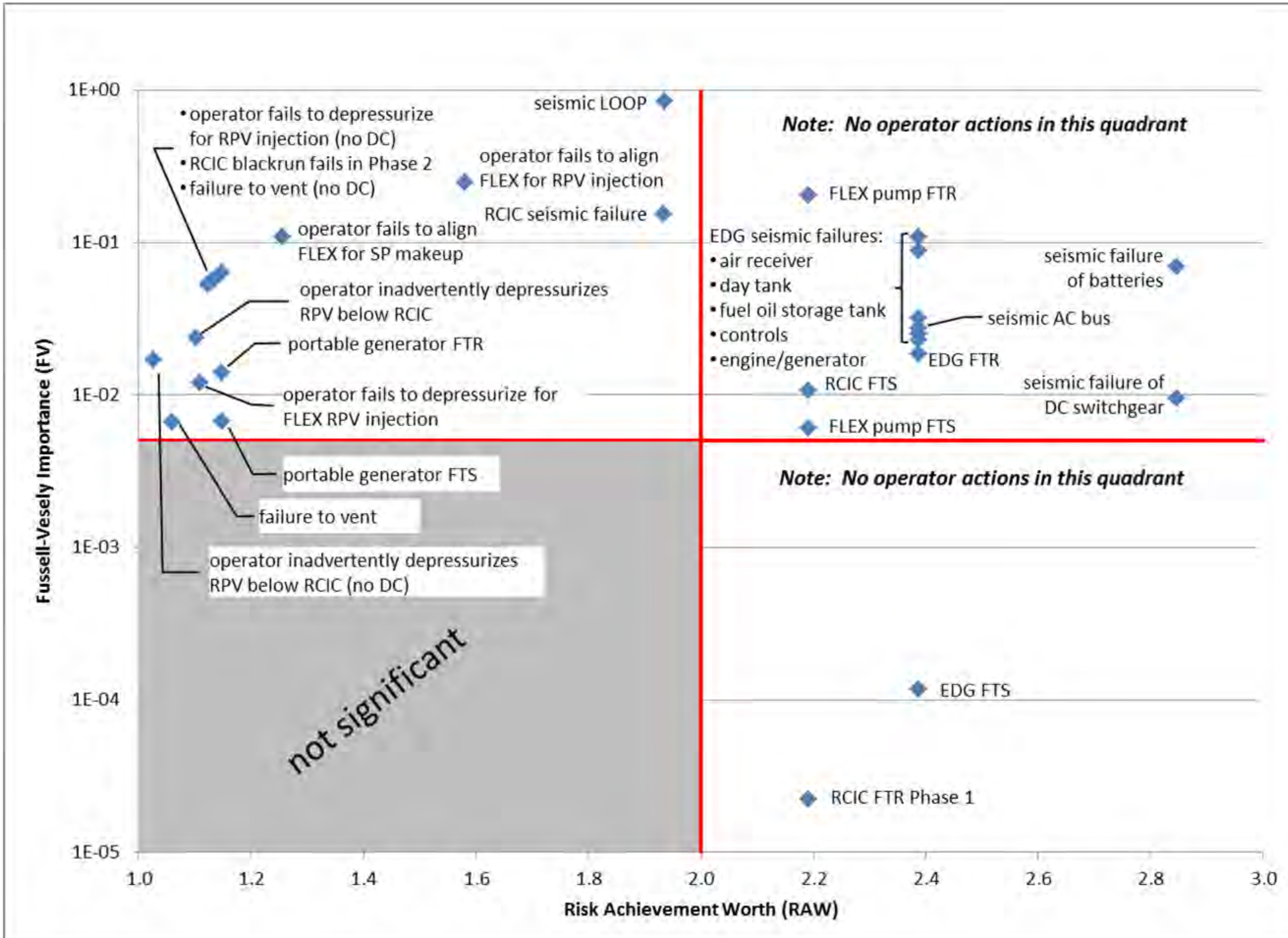
Scoping HEPs:
CDF = 8.9E-6/y

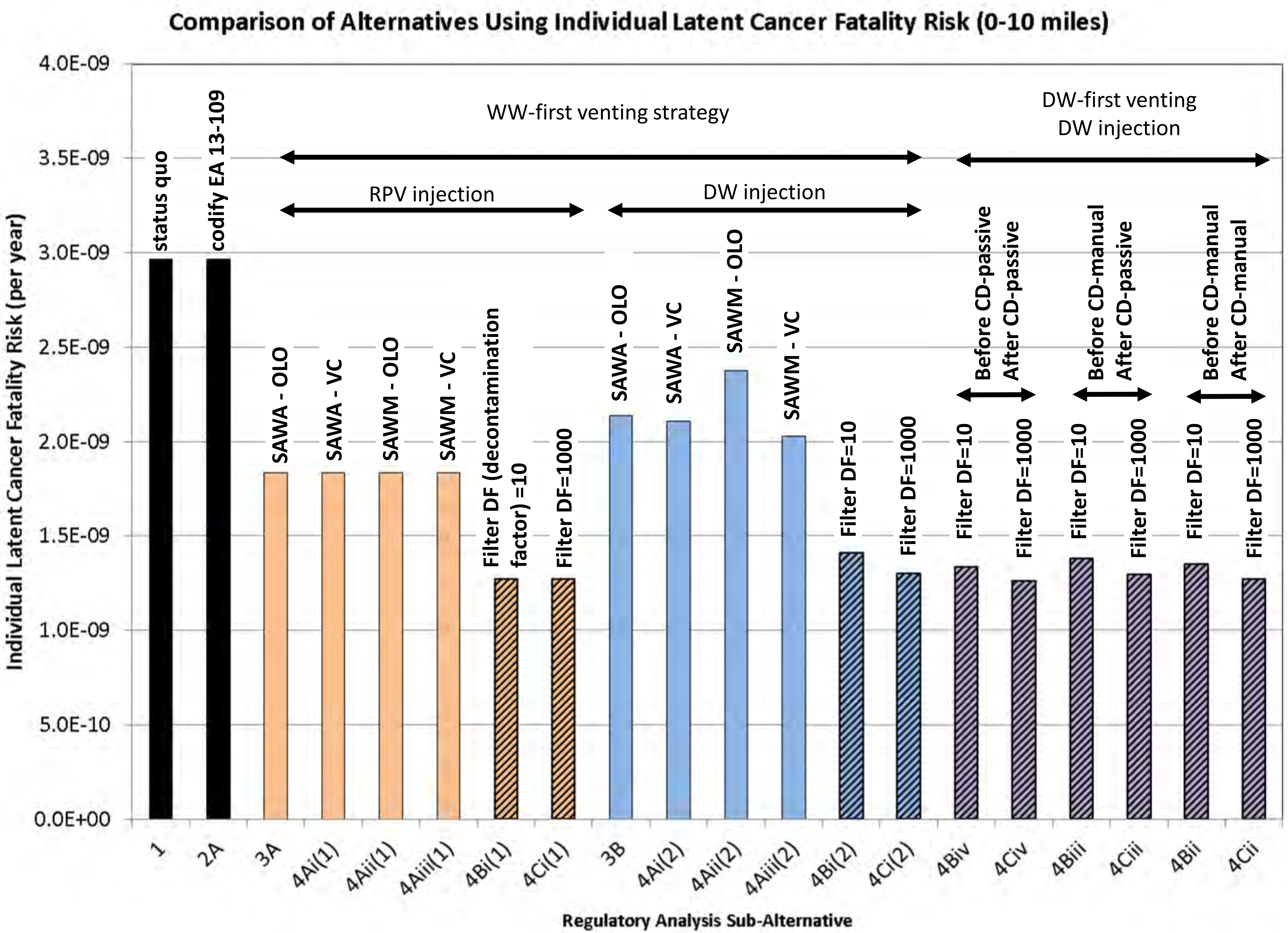
No credit for operators:
CDF = 1.9E-5/y

Comparison of Core-Damage Frequencies

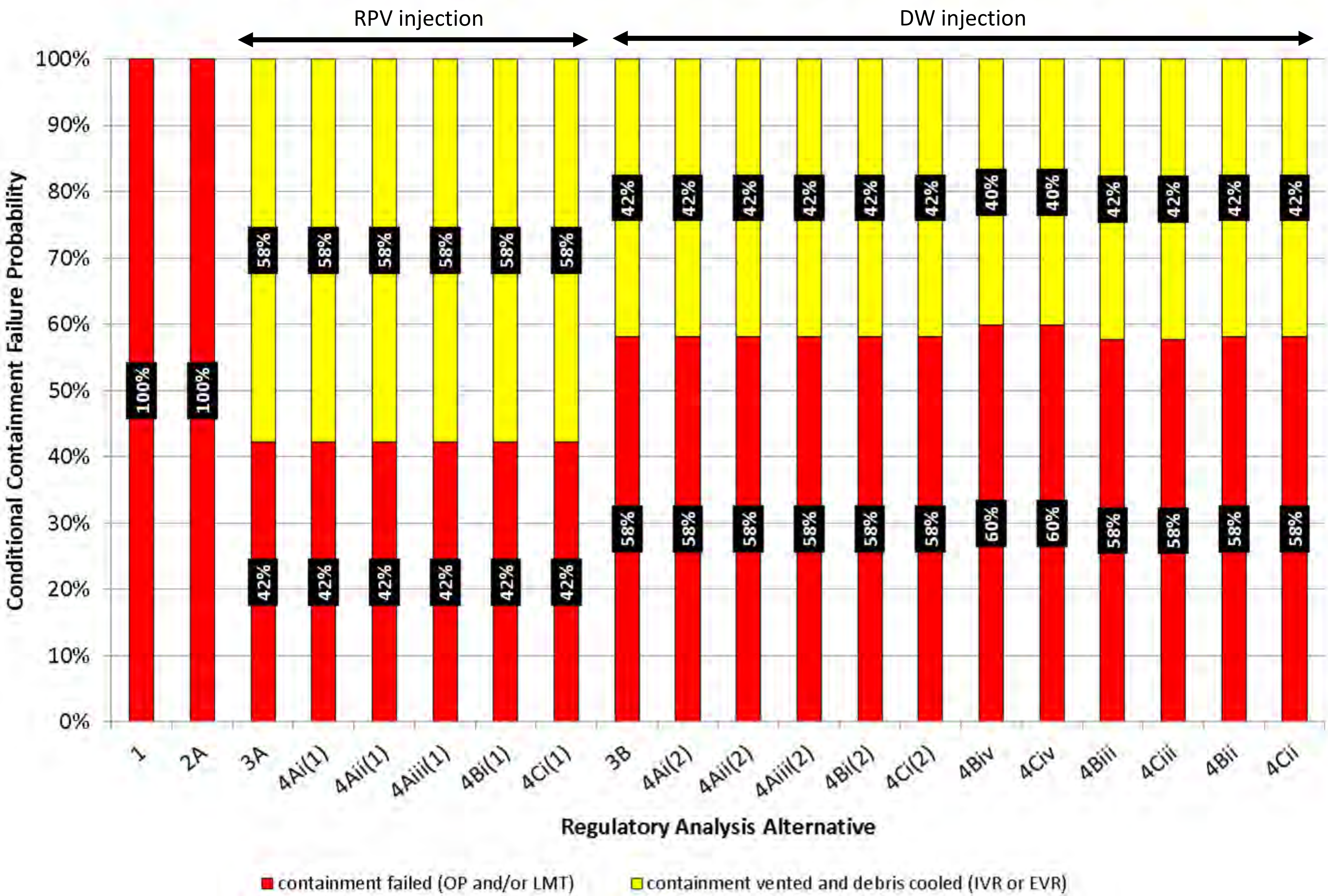


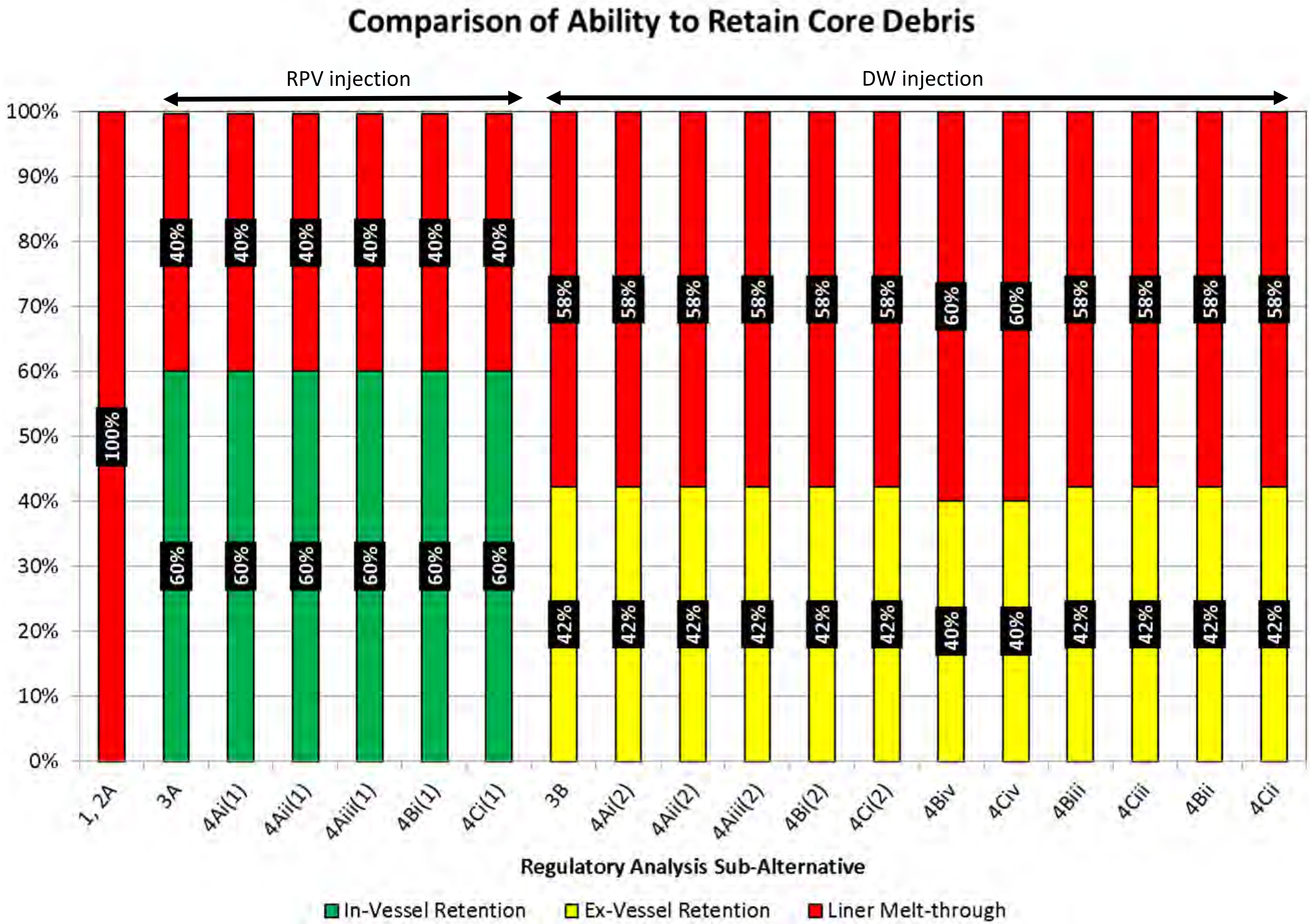
Importance Measures for Core Damage




Comparison of Alternatives Using Individual Latent Cancer Fatality Risk (0-10 miles)
Individual Latent Cancer Fatality Risk (per year)
4.0E-09
3.5E-09
3.0E-09
2.5E-09
2.0E-09
1.5E-09
1.0E-09
5.0E-10
0.0E+00
WW-first venting strategy
DW-first venting
DW injection
RPV injection
DW injection
status quo
codify EA 13-109
SAWA - OLO
SAWA - VC
SAWM - OLO
SAWM - VC
Filter DF (decontamination factor) =10
Filter DF=1000
SAWA - OLO
SAWA - VC
SAWM - OLO
SAWM - VC
Filter DF=10
Filter DF=1000
Before CD-passive
After CD-passive
Before CD-manual
After CD-passive
Before CD-manual
After CD-manual
Filter DF=10
Filter DF=1000
Filter DF=10
Filter DF=1000
Filter DF=10
Filter DF=1000
1
2A
3A
4Ai(1)
4Aii(1)
4Aiii(1)
4Bi(1)
4Ci(1)
3B
4Ai(2)
4Aii(2)
4Aiii(2)
4Bi(2)
4Ci(2)
4Biv
4Civ
4Biii
4Ciii
4Bii
4Cii
Regulatory Analysis Sub-Alternative

Comparison of Conditional Containment Failure Probability



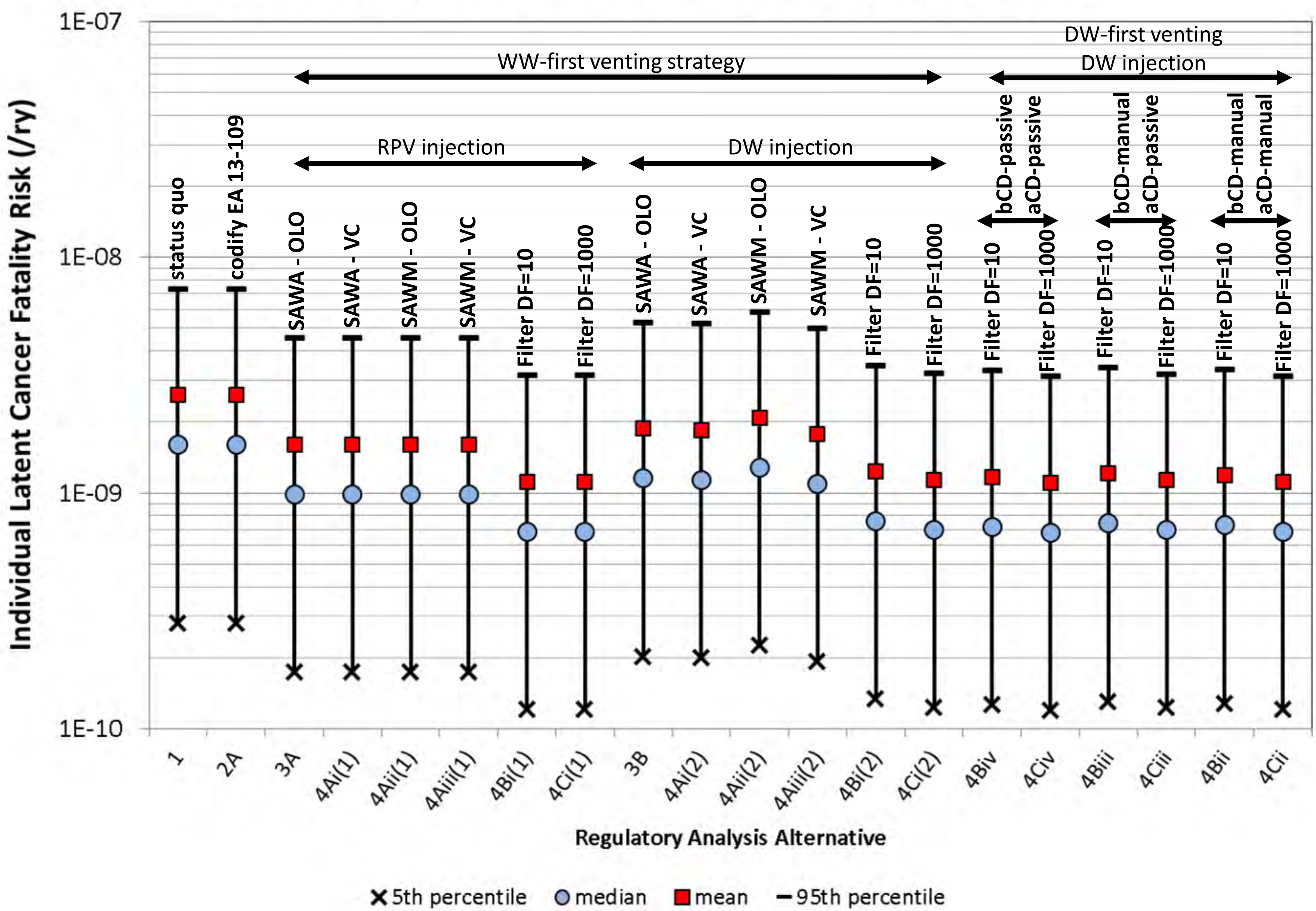


Comparison of Ability to Retain Core Debris
RPV injection
DW injection
100%
90%
80%
70%
60%
50%
40%
30%
20%
10%
0%
100%
40%
60%
58%
42%
60%
40%
1, 2A
3A
4Ai(1)
4Aii(1)
4Aiii(1)
4Bi(1)
4Ci(1)
3B
4Ai(2)
4Aii(2)
4Aiii(2)
4Bi(2)
4Ci(2)
4Biv
4Civ
4Biii
4Ciii
4Bii
4Cii
Regulatory Analysis Sub-Alternative
In-Vessel Retention
Ex-Vessel Retention
Liner Melt-through

Monte Carlo Parametric Uncertainty Analysis

- Parameters included
  - Seismic hazard curves
  - Seismic fragilities
  - ELAP frequencies (internal events)
  - Human error probabilities
  - Equipment failure rates and failure-on-demand probabilities
  - Conditional consequences (informed by SOARCA)

Approximate Uncertainty Bounds for Individual Latent Cancer Fatality Risk



Risk Insights

- Containment venting and water addition are effective strategies for preserving containment structural integrity and reducing severe accident risks
  - Venting is needed to prevent containment overpressurization
  - Venting and water addition are needed to prevent liner melt-through
  - Neither strategy is sufficient by itself; both strategies are necessary to reduce risk
- An engineered filter on the containment vent is effective in reducing severe accident risks; however, its benefit is notably reduced if liner melt-through occurs.

BACKUP VIEWGRAPHS

Acronyms and Initialisms

AC	alternating current
AV	anticipatory venting
APET	accident progression event tree
BWR	boiling water reactor
CDET	core damage event tree
CDF	core-damage frequency
CPRR	containment protection and release reduction
DW	drywell
DWF	drywell first venting strategy
ELAP	extended loss of AC power
EPS	emergency power system
HEP	human error probability
HRA	human reliability analysis
IC	isolation condenser
LOOP	loss of offsite power
MACCS	MELCOR accident consequence code system
OLO	open-and-leave-open venting strategy
PDS	plant damage state
PRA	probabilistic risk assessment
RC	release category
RCIC	reactor core isolation cooling
RPV	reactor pressure vessel
SAWA	severe accident water addition
SAWM	severe accident water management
SBO	station blackout
VC	vent cycling strategy
WW	wetwell
WWF	wetwell first venting strategy

Purpose of the Risk Evaluation

- Provide inputs to the regulatory analysis
  - Cost-benefit analysis
  - Safety Goal evaluation
- Develop risk insights for CPRR strategies

Regulatory Basis Sub-Alternatives

Number	Regulatory Basis Sub-Alternatives	Before Core Damage				After Core Damage					
		Vent Priority	Actuation	Operation Mode	Reclose Vent if Core Damage is Imminent	Post-Accident Injection Location	Post-Accident Injection Operating Mode	Vent Priority	Actuation	Operation Mode	Filter Size
1	1	WWF	M	AV	yes	no		WWF	M	OLO	no
2	2A	WWF	M	AV	yes	no		WWF	M	OLO	no
3	3A	WWF	M	AV	yes	RPV	SAWA	WWF	M	OLO	no
4	3B	WWF	M	AV	yes	DW	SAWA	WWF	M	OLO	no
5	4Ai(1)	WWF	M	AV	yes	RPV	SAWA	WWF	M	VC	no
6	4Ai(2)	WWF	M	AV	yes	DW	SAWA	WWF	M	VC	no
7	4Aii(1)	WWF	M	AV	yes	RPV	SAWM	WWF	M	OLO	no
8	4Aii(2)	WWF	M	AV	yes	DW	SAWM	WWF	M	OLO	no
9	4Aiii(1)	WWF	M	AV	yes	RPV	SAWM	WWF	M	VC	no
10	4Aiii(2)	WWF	M	AV	yes	DW	SAWM	WWF	M	VC	no

Venting Priority
DWF - drywell-first venting strategy
WWF - wetwell-first venting strategy
Venting Actuation
M - manual venting
P - passive venting (rupture disk)
Venting Operation Mode
AV - anticipatory venting; open at 15 psig and leave open
OLO - open at PCPL and leave open
VC - vent cycling at PCPL with 10 psig band

Post-Accident Injection Location
DW- drywell via external connection
RPV- reactor pressure vessel via external connection
Post-Accident Injection Operating Mode
SAWA - severe accident water addition
SAWM - severe accident water management
Filter Size
L- large filter
S - small filter

Regulatory Basis Sub-Alternatives (con't.)

Number	Regulatory Basis Sub-Alternatives	Before Core Damage				After Core Damage					
		Vent Priority	Actuation	Operation Mode	Reclose Vent if Core Damage is Imminent	Post-Accident Injection Location	Post-Accident Injection Operating Mode	Vent Priority	Actuation	Operation Mode	Filter Size
11	4Bi(1)	WWF	M	AV	yes	RPV	SAWA	WWF	M	OLO	S
12	4Bi(2)	WWF	M	AV	yes	DW	SAWA	WWF	M	OLO	S
13	4Bii	WWF	M	AV	yes	DW	SAWA	DWF	M	OLO	S
14	4Biii	WWF	M	AV	yes	DW	SAWA	DWF	P	OLO	S
15	4Biv	DWF	P	OLO	no	DW	SAWA	DWF	P	OLO	S
16	4Ci(1)	WWF	M	AV	yes	RPV	SAWA	WWF	M	OLO	L
17	4Ci(2)	WWF	M	AV	yes	DW	SAWA	WWF	M	OLO	L
18	4Cii	WWF	M	AV	yes	DW	SAWA	DWF	M	OLO	L
19	4Ciii	WWF	M	AV	yes	DW	SAWA	DWF	P	OLO	L
20	4Civ	DWF	P	OLO	no	DW	SAWA	DWF	P	OLO	L

Venting Priority
DWF - drywell-first venting strategy
WWF - wetwell-first venting strategy
Venting Actuation
M - manual venting
P - passive venting (rupture disk)
Venting Operation Mode
AV - anticipatory venting; open at 15 psig and leave open
OLO - open at PCPL and leave open
VC - vent cycling at PCPL with 10 psig band

Post-Accident Injection Location
DW- drywell via external connection
RPV- reactor pressure vessel via external connection
Post-Accident Injection Operating Mode
SAWA - severe accident water addition
SAWM - severe accident water management
Filter Size
L- large filter
S - small filter

Approach

- Define regulatory analysis alternatives and sub-alternatives
- Develop logic models:
  - Three core-damage event trees (CDETs):
    - Expands approach used in SECY-12-0157
    - Credits MDBDE strategies (e.g., FLEX)
    - Informs the definition of MELCOR cases
    - Results grouped by plant damage states (PDSs)
  - Six accident progression event trees (APETs)
    - Expands the approach used in SECY-12-0157
    - Consideration of CPRR strategies according to sub-alternatives
    - Results grouped by release categories (RCs)
  - Implemented by spreadsheets
    - Complete probabilistic treatment of success paths
    - No fault trees developed → no truncation issues
- Logic model quantification:
  - Matrix multiplication (NUREG-1150 approach)
  - APET branch probabilities depend on the input PDS (account for dependencies)
- Risk quantification
  - Mapping RCs to MACCS calculations
  - Sensitivity and uncertainty analyses

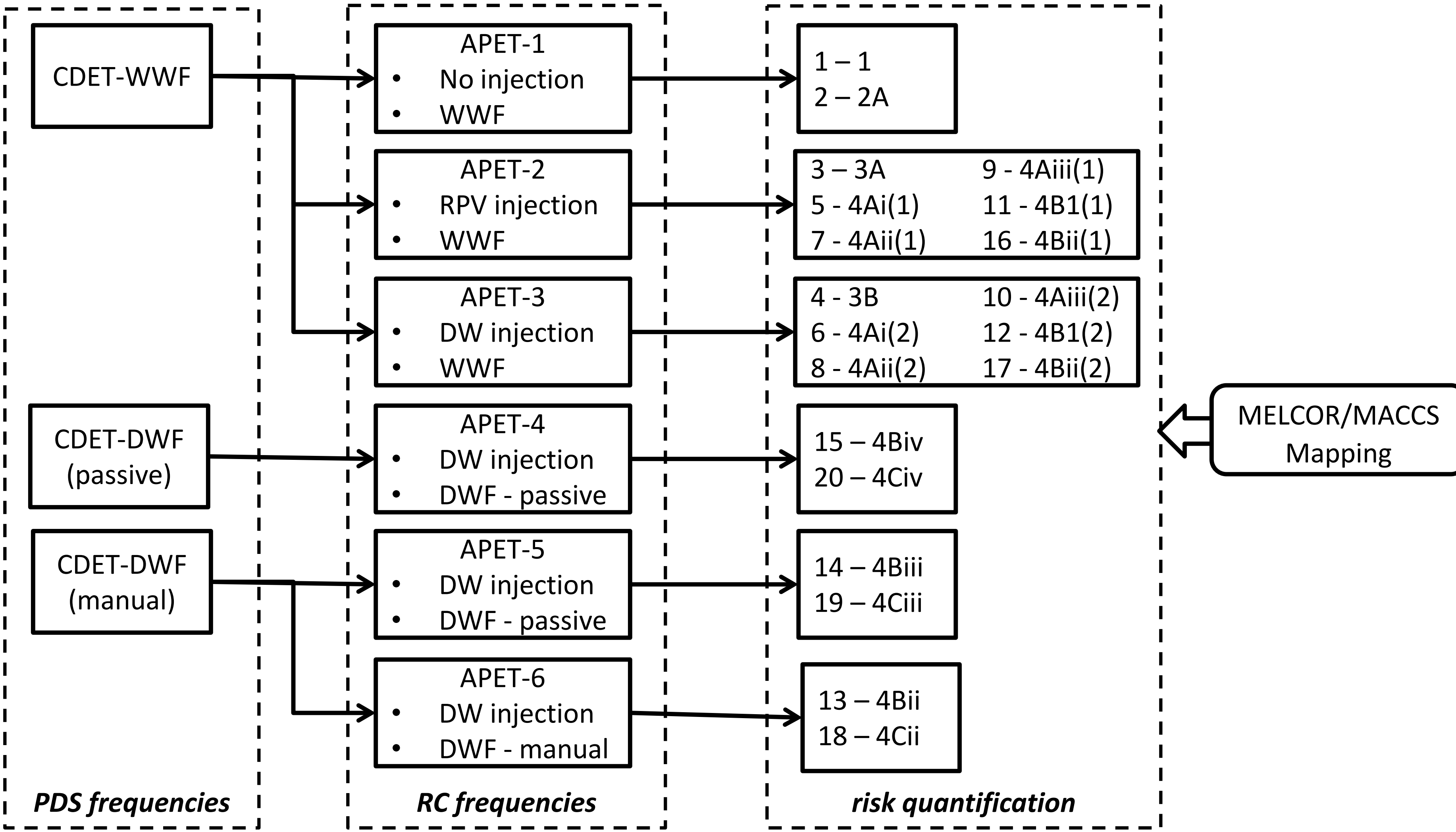
Logic Model Scope and Assumptions

- Generic BWR plant having a RCIC system and a Mark I containment
- Phase 1 – use of installed plant equipment
  - Duration of 4 hours
  - No credit for use of the condensate storage tank (CST); assumed to be non-seismically qualified)
  - Credit for RCIC blackstart and blackrun if dc power fails
  - No credit for portable equipment
  - No need to vent containment
  - Operators will attempt to depressurize the RPV to 200-400 psig in order to minimize SRV cycling and suppression pool heatup
  - CDETs credit local manual SRV operation, if dc power fails
- Phase 2 – use of onsite portable equipment
  - Duration of 68 hours
  - Operators will initiate anticipatory containment venting at 15 psig in order to minimize heatup of the suppression pool and prolong RCIC operation except for sub-alternatives that consider passive DW venting
  - WW and DW vents are redundant
  - Credit for local manual operation of the containment vent valves if dc power fails
  - Assumed there is no need to provide RCIC pump room ventilation
  - Portable generator must be aligned to provide dc power within 4 hours
  - If the RCIC pump fails, core cooling can be provided by aligning the portable FLEX pump for RPV injection and depressurizing the RPV below the portable FLEX pump’s shutoff head
  - In the CDETs, it is assumed that the operators will attempt to reclose the containment vent valves, in accordance with the BWR Owners’ Group Emergency Procedure and Severe Accident Guidelines (EPG/SAGs), if they recognize that core damage is occurring

Logic Model Scope and Assumptions (Con’t.)

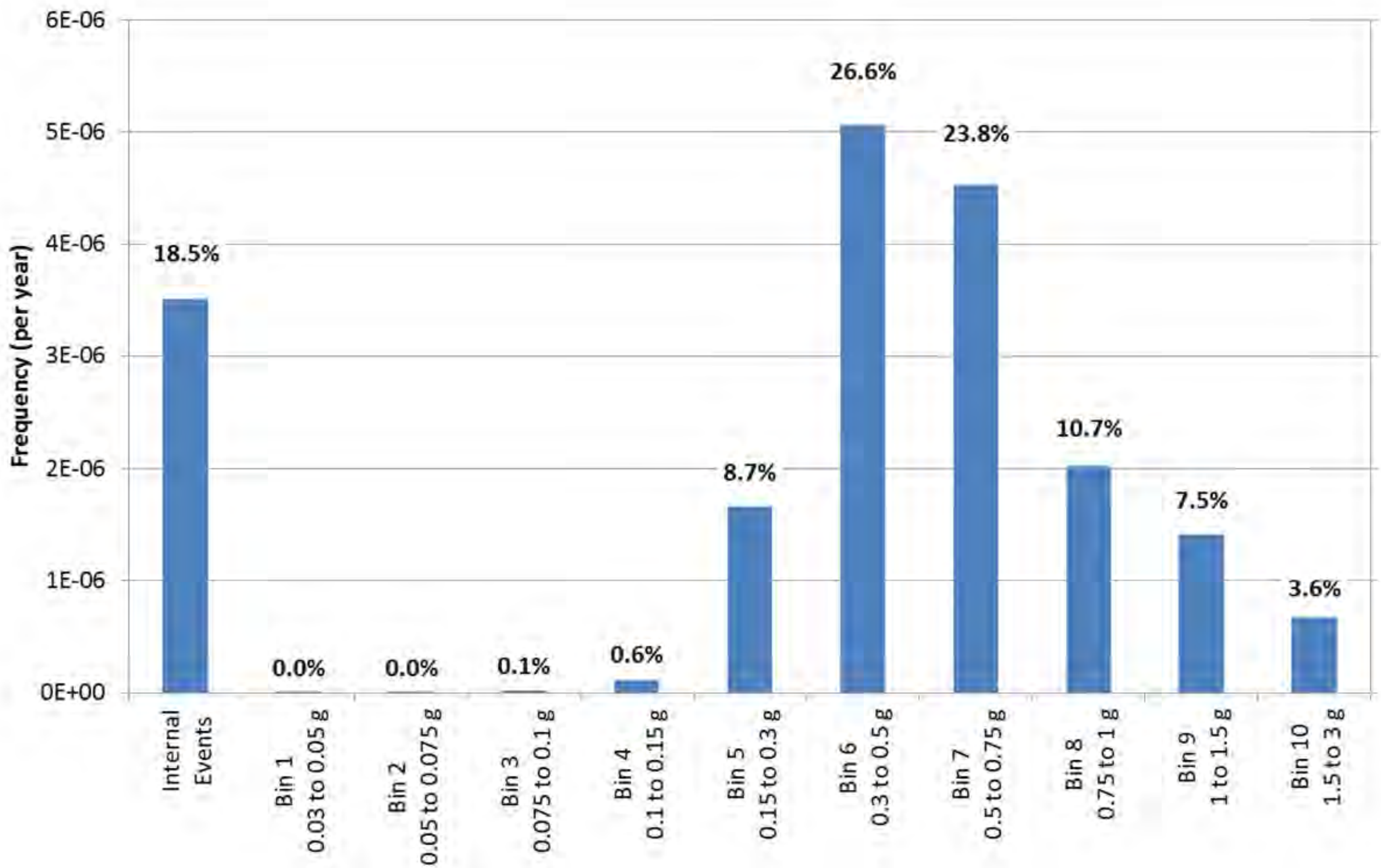
- Phase 2 – use of onsite portable equipment
  - Containment over-pressurization failure is prevented by opening the containment vent valves:
    - The WW and DW vents are redundant (i.e., the DW vent can be used to provide anticipatory venting if the WW vent fails closed, and vice versa).
    - The CDETs credit local manual operation of the containment vent valves if dc power fails.
    - MELCOR calculations indicate that the containment must be vented before vessel breach because of the build-up of non-condensable gases generated during fuel-clad oxidation.
    - successful post-core-damage containment venting is a controlled release of radioactive materials to the environment, which is allowable under 10 CFR Part 50, Appendix A, General Design Criterion 16, “Containment design,” and consistent with the Commission’s direction in the SRM to SECY-12-0157, Option 2, to assume the installation of severe accident capable hardened venting system.
  - In the APETs, post-core-damage water injection into the RPV will prevent vessel breach if it is initiated prior to core relocation and the RPV is depressurized below the shutoff head of the portable FLEX pump using the SRVs. In contrast, the regulatory basis alternatives involving post-core-damage DW injection cannot prevent vessel breach.

Logic Model Links



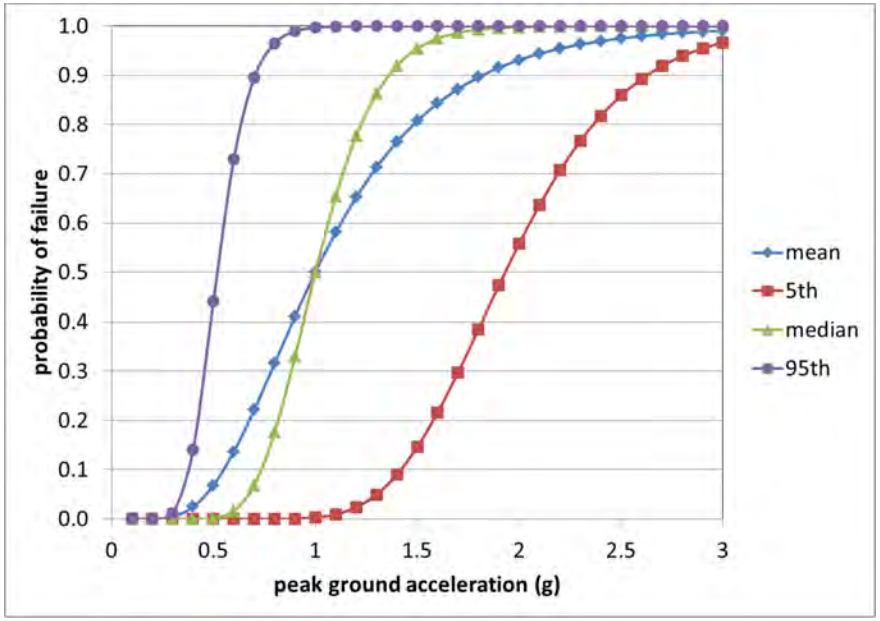
Contributions to ELAP Frequency

Average of Point Estimates for BWR Mark I Plants with RCIC Systems

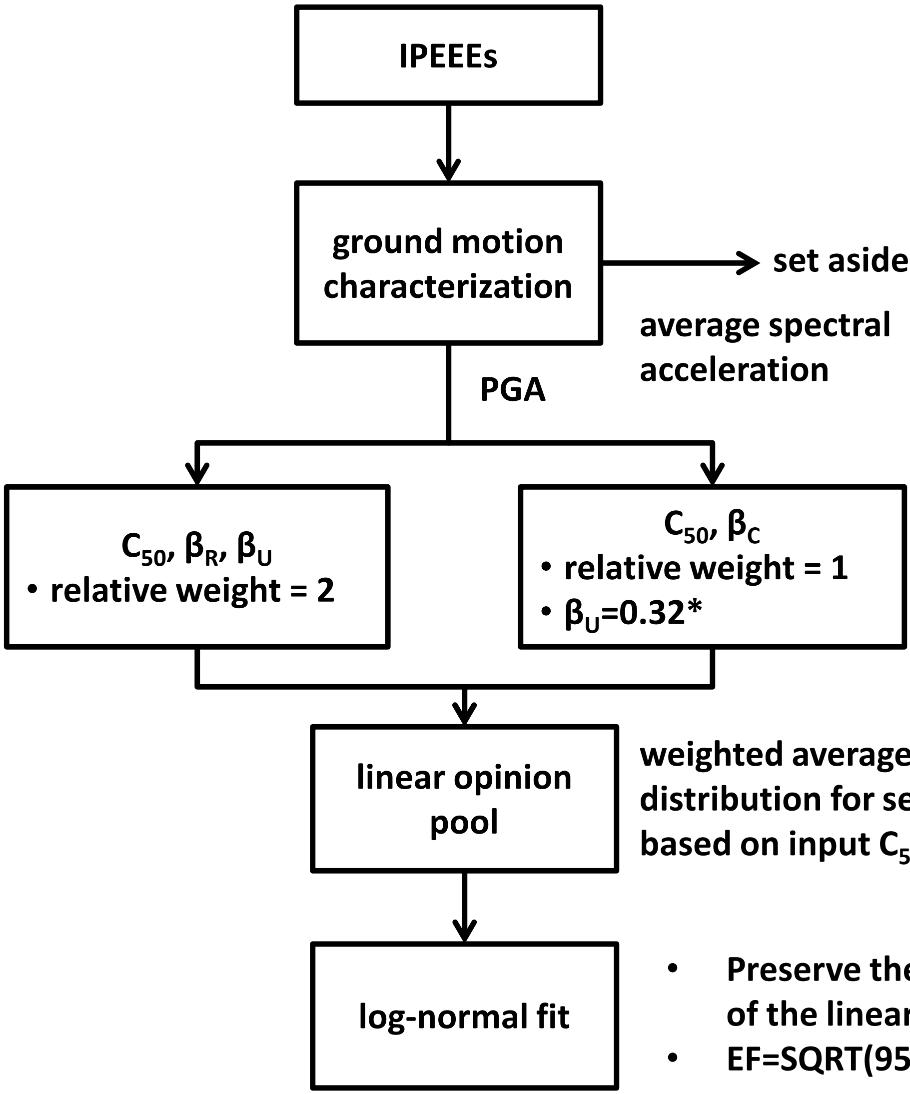


Seismic Fragility

- C_{50}: median capacity
- β_R: logarithmic standard deviation for randomness (e.g., ground motion variability)
- β_U: logarithmic standard deviation for uncertainty (e.g., material properties)



Review and Use of IPEEE Seismic Fragilities

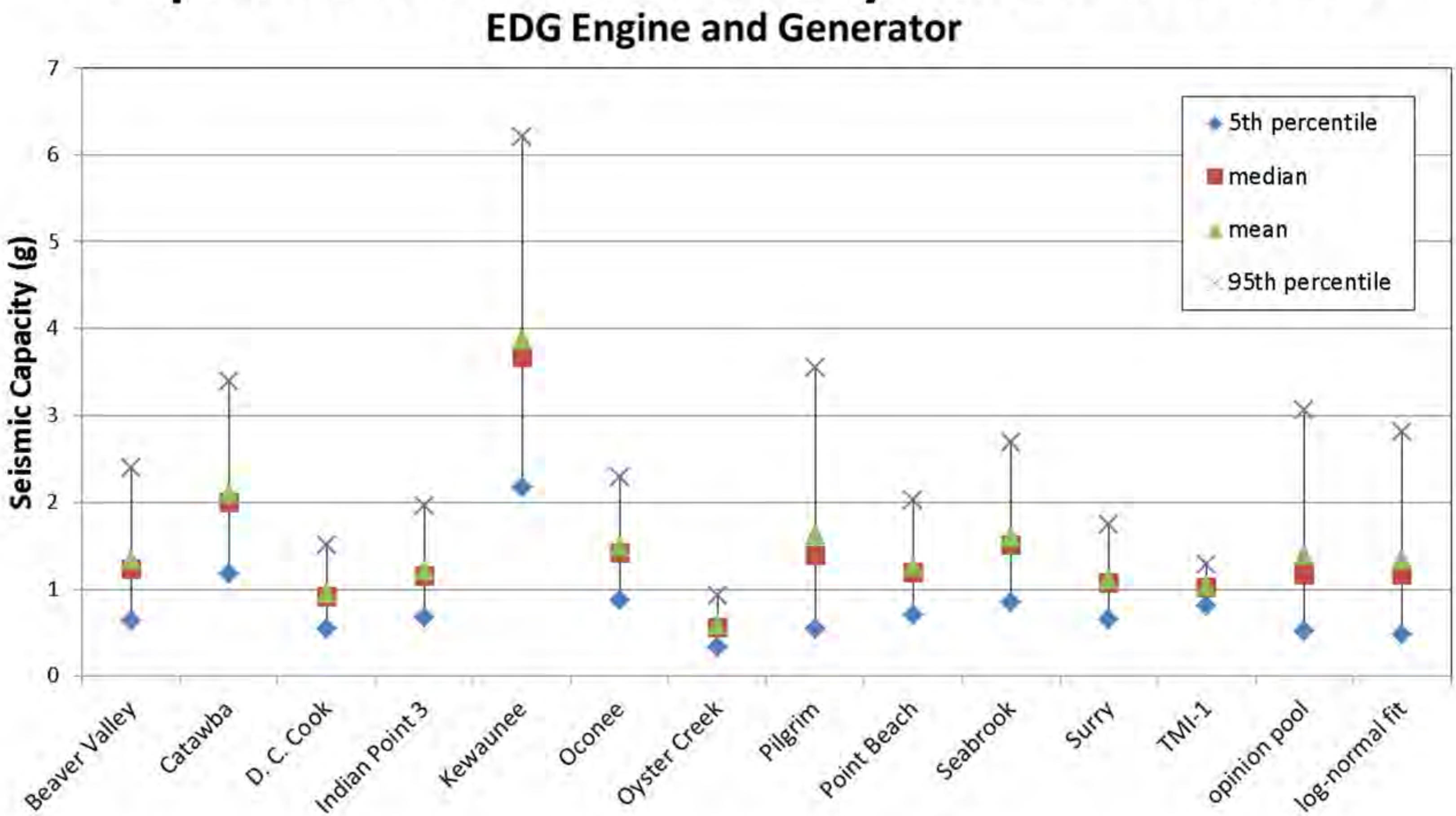


*Recommended by the SPID ("Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," EPRI Report 1025287, November 2012

weighted average uncertainty distribution for seismic capacity, based on input C_{50} and β_U values

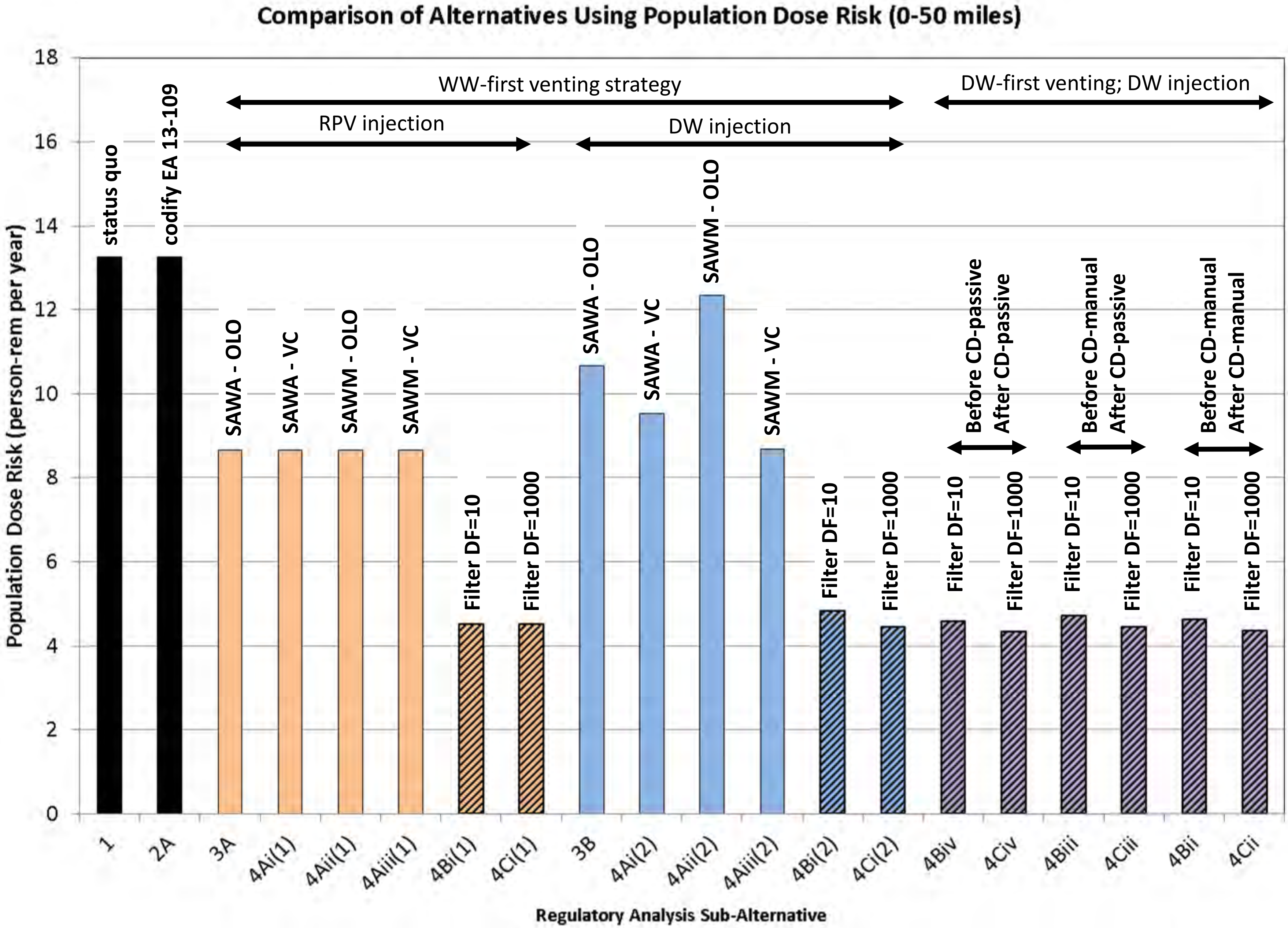
- Preserve the median and the error factor of the linear opinion pool
- EF=SQRT(95th percentile/5th percentile)

Seismic Fragility Population Variability Distribution

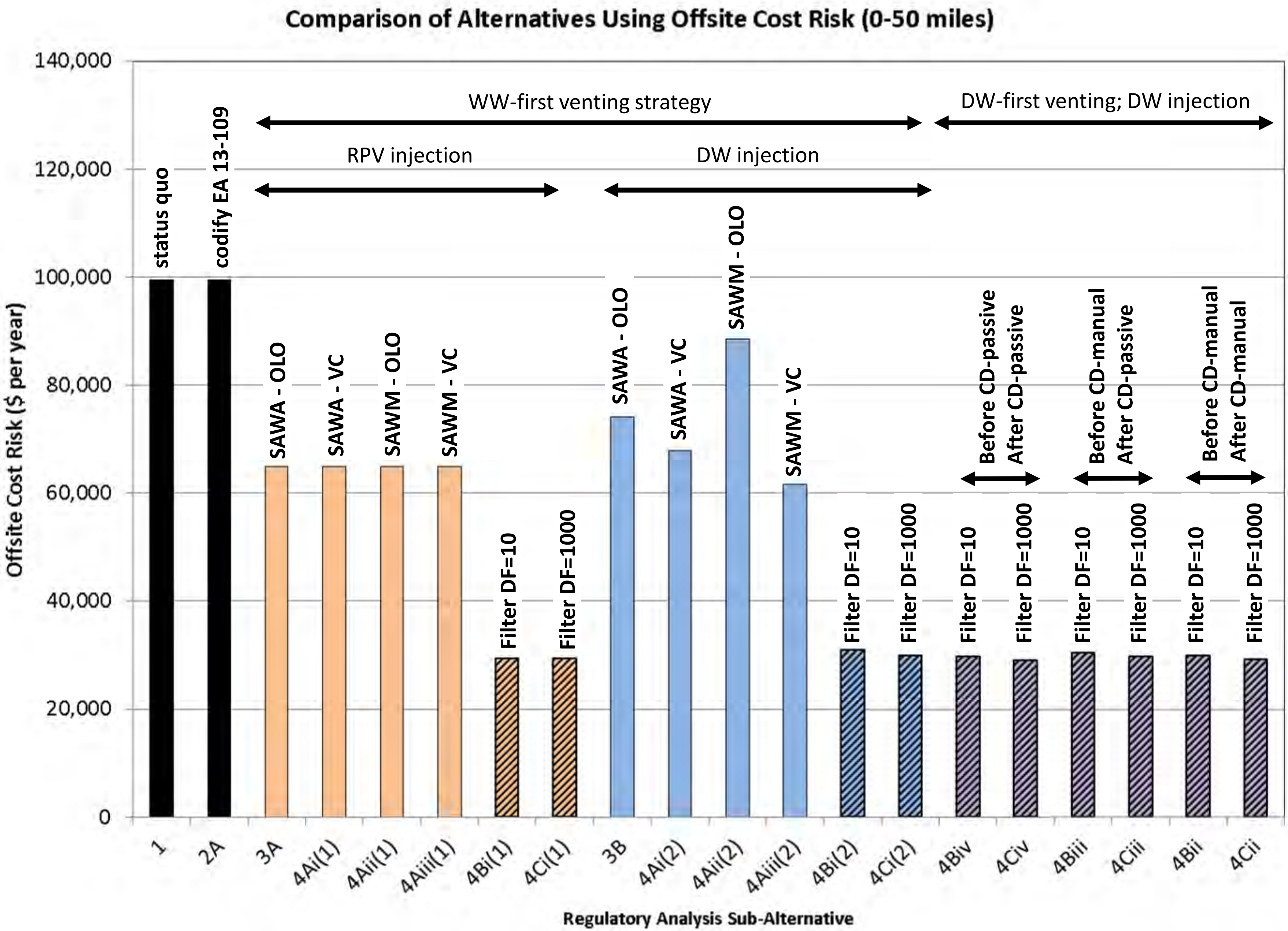


Importance Measures

- Identify significant risk contributors:
  - Fussell-Vesely (FV) importance > 0.005
  - Risk achievement worth (RAW) > 2
- Fussell-Vesely importance is the relative contribution of a specific basic event (operator action, equipment failure) to the calculated risk metric.
- Risk achievement worth is the increase in risk if a specific basic event is assumed to be failed or was assumed to be always unavailable (defined as a ratio in this risk evaluation).

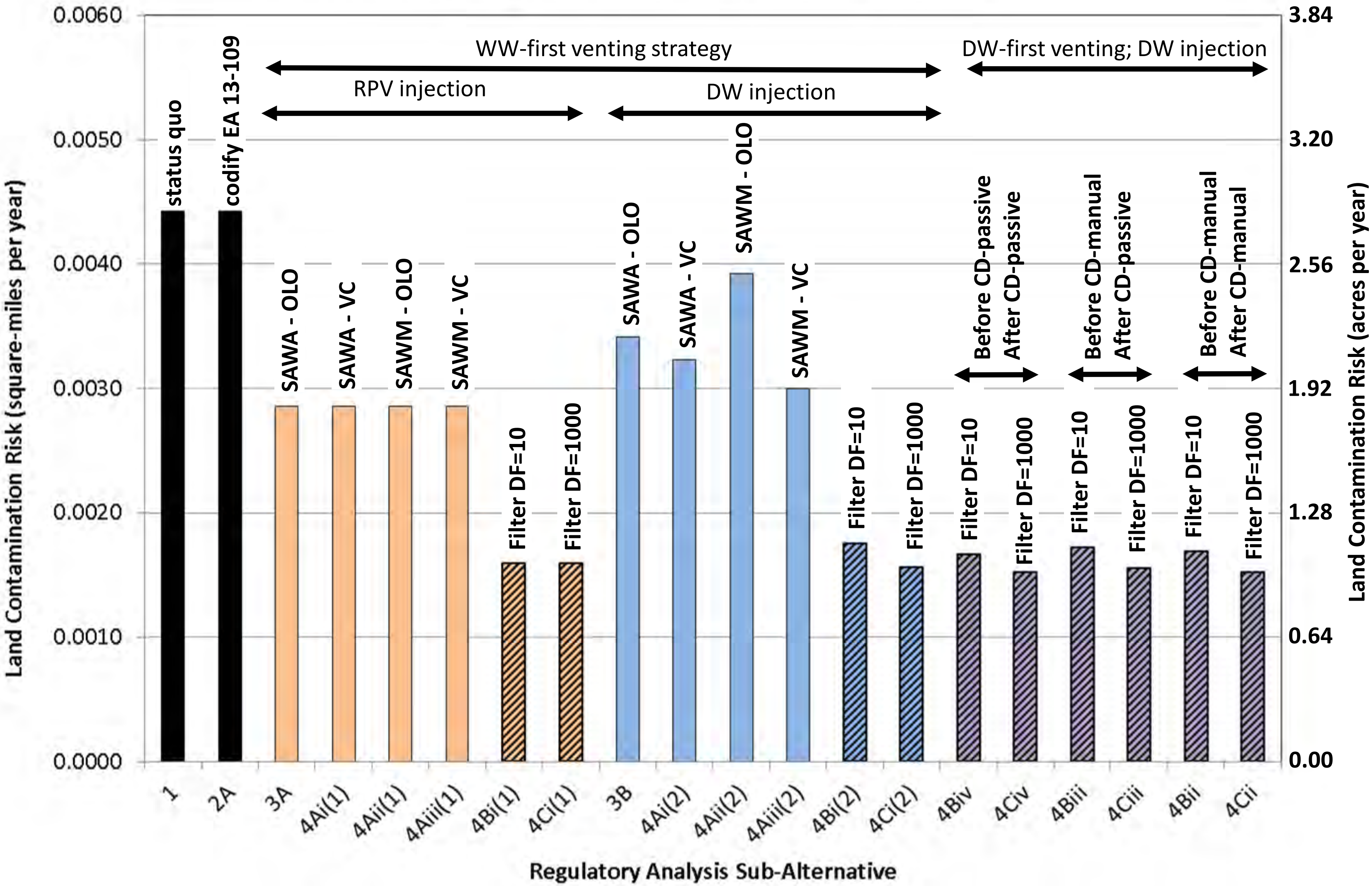


Comparison of Alternatives Using Population Dose Risk (0-50 miles)
Population Dose Risk (person-rem per year)
0
2
4
6
8
10
12
14
16
18
WW-first venting strategy
DW-first venting; DW injection
RPV injection
DW injection
status quo
codify EA 13-109
SAWA - OLO
SAWA - VC
SAWM - OLO
SAWM - VC
Filter DF=10
Filter DF=1000
SAWA - OLO
SAWA - VC
SAWM - OLO
SAWM - VC
Filter DF=10
Filter DF=1000
Before CD-passive
After CD-passive
Before CD-manual
After CD-passive
Before CD-manual
After CD-manual
Filter DF=10
Filter DF=1000
Filter DF=10
Filter DF=1000
Filter DF=10
Filter DF=1000
1
2A
3A
4Ai(1)
4Aii(1)
4Aiii(1)
4Bi(1)
4Ci(1)
3B
4Ai(2)
4Aii(2)
4Aiii(2)
4Bi(2)
4Ci(2)
4Biv
4Civ
4Biii
4Ciii
4Bii
4Cii
Regulatory Analysis Sub-Alternative

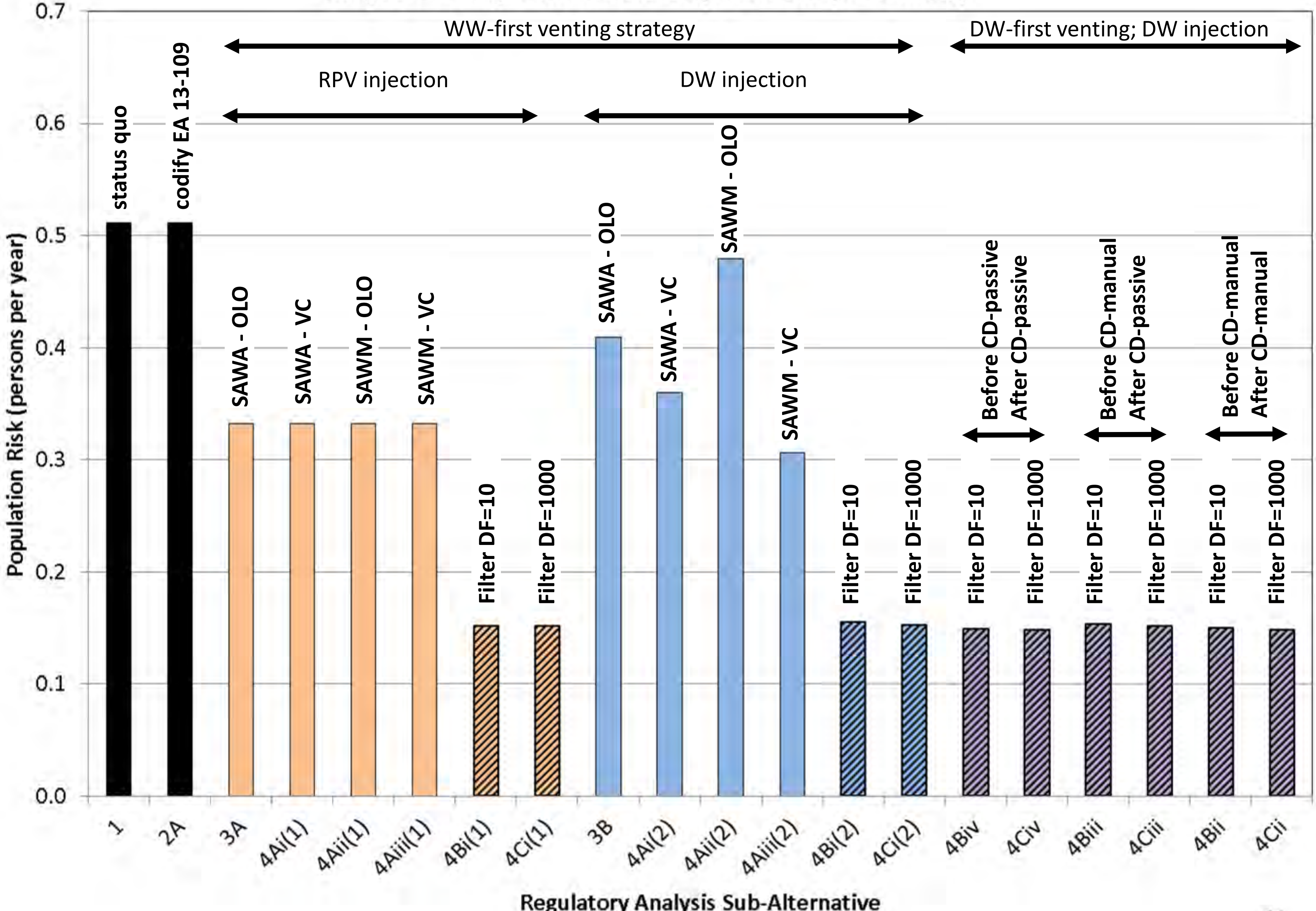


Comparison of Alternatives Using Offsite Cost Risk (0-50 miles)
Offsite Cost Risk ($ per year)
140,000
120,000
100,000
80,000
60,000
40,000
20,000
0
WW-first venting strategy
DW-first venting; DW injection
RPV injection
DW injection
status quo
codify EA 13-109
SAWA - OLO
SAWA - VC
SAWM - OLO
SAWM - VC
Filter DF=10
Filter DF=1000
SAWA - OLO
SAWA - VC
SAWM - OLO
SAWM - VC
Filter DF=10
Filter DF=1000
Before CD-passive
After CD-passive
Before CD-manual
After CD-passive
Before CD-manual
After CD-manual
Filter DF=10
Filter DF=1000
Filter DF=10
Filter DF=1000
Filter DF=10
Filter DF=1000
1
2A
3A
4Ai(1)
4Aii(1)
4Aiii(1)
4Bi(1)
4Ci(1)
3B
4Ai(2)
4Aii(2)
4Aiii(2)
4Bi(2)
4Ci(2)
4Biv
4Civ
4Biii
4Ciii
4Bii
4Cii
Regulatory Analysis Sub-Alternative

Comparison of Alternatives Using Area of Land Exceeding Long-Term Habitability Criterion (0-50 miles)



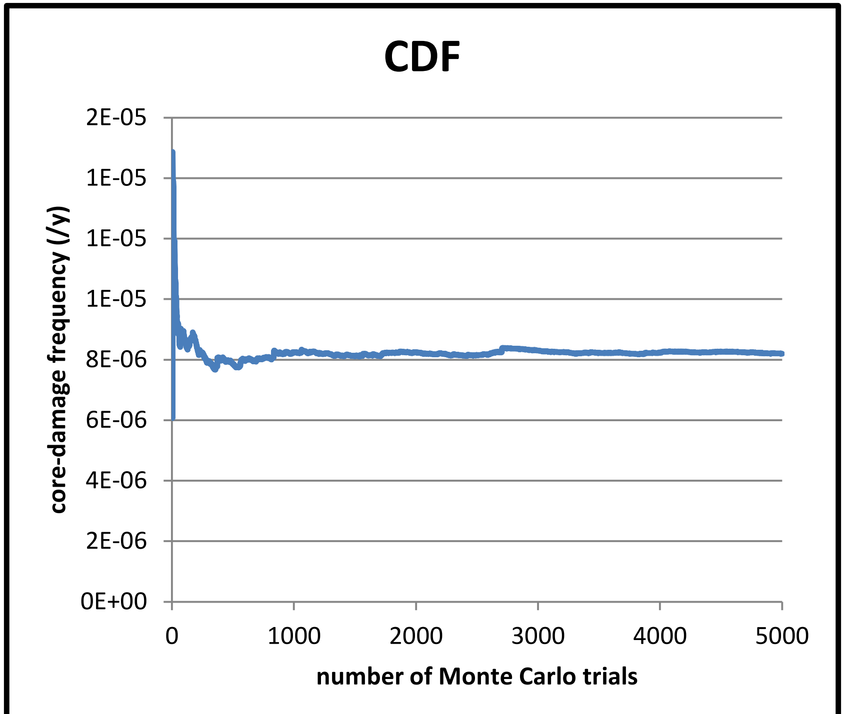
Comparison of Alternatives Using Population Subject to Long-Term Habitability Criterion (0-50 miles)

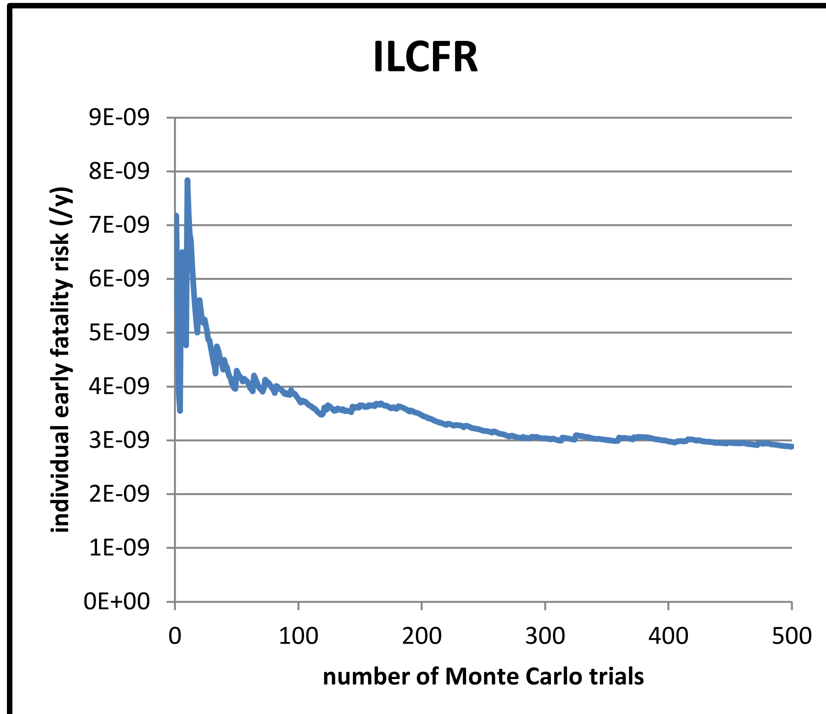


Convergence of Monte Carlo Sampling

CDF	approximate 90% confidence bound for the mean	
Sample Mean	Lower Bound	Upper Bound
8.2E-6	8.0E-6	8.4E-6

ILCFR	approximate 90% confidence bound for the mean	
Sample Mean	Lower Bound	Upper Bound
3.0E-9	2.6E-9	3.3E-9





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Protecting People and the Environment

MELCOR ANALYSIS IN SUPPORT OF CPRR RULEMAKING

Hossein Esmaili
Office of Nuclear Regulatory Research
Division of Systems Analysis

July 7, 2015

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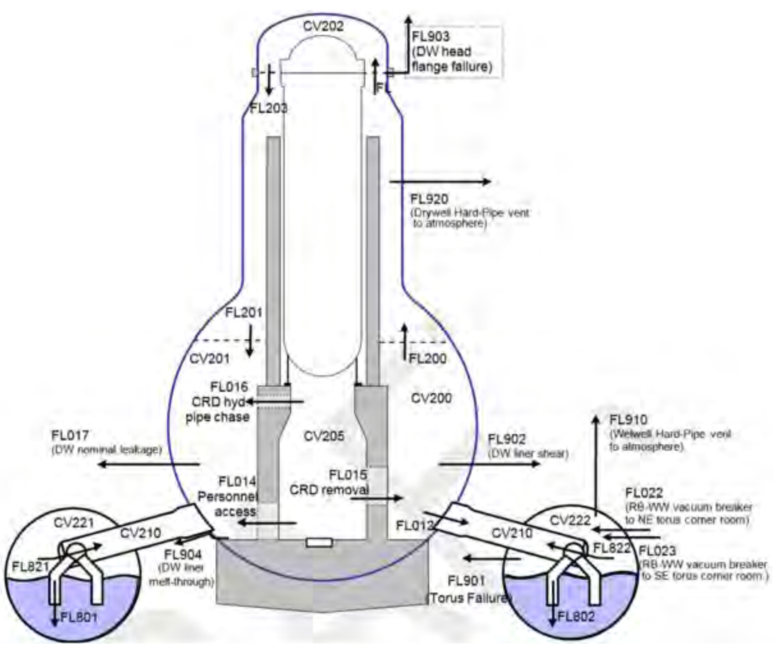
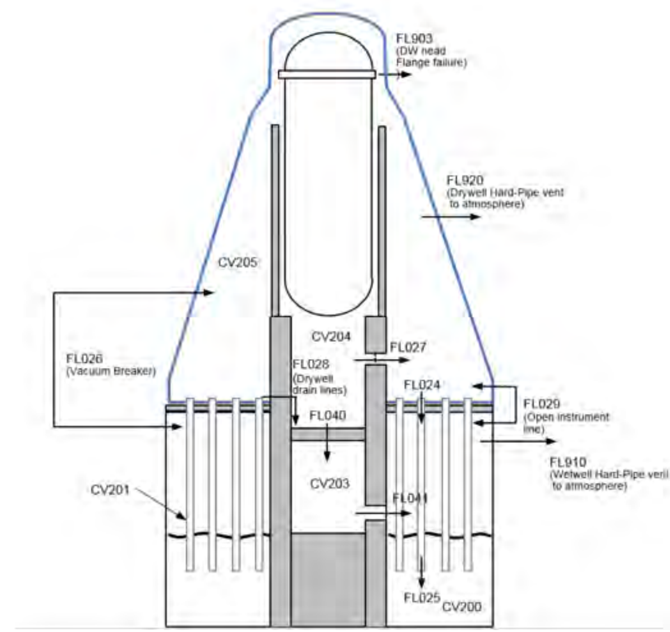
Technical Approach

- MELCOR accident progression analysis for representative BWR Mark I and Mark II plants
  - fission product release characteristics
  - effectiveness of mitigation (RPV pressure control, containment venting, and core/containment water injection strategies including severe accident water addition [SAWA] and severe accident water management [SAWM]).
- SOARCA Mark I model converted to MELCOR 2.1
- Mark II model (NUREG/CR-5305) converted to MELCOR 2.1





MELCOR Run Matrix for Mark I

		Pre Core Damage						Post Core Damage				
		RPV Pressure control	RCIC Operation				Anticipatory Venting	Flex Operation		SRV Operation	Venting	
		Availability (hr)	RCIC Availability (hr)	RCIC Suction	Failure Temp (F)	Open SRV after RCIC fails	Setpoint (psig)	Injection @ LH failure	WW Level Control Injection @ 21' (gpm)	Allow SRV stuck open failure?	Location	Setpoint (psig)
Option	Case											
1/2A	1	72	16	SP	230	N	15	-	-	Y	WW	PCPL
1/2A	1S1	72	16	SP	230	N	5	-	-	Y	WW	PCPL
1/2A	2	72	16	SP	230	N	15	-	-	Y	WW	PCPL
1/2A	3	4	4	SP	230	N	15	-	-	N	WW	PCPL
1/2A	4	72	16	SP	240	N	15	-	-	Y	WW	PCPL
1/2A	5	72	16	CST	230	N	15	-	-	Y	WW	PCPL
1/2A	6	72	16	SP	230	N	15	-	-	Y	WW	PSP
3A	7	72	16	SP	230	N	15	RPV	0	Y	WW	PCPL
	7dw	72	16	SP	230	N	15	RPV	0	Y	DW	PCPL
3A	10	72	16	SP	230	N	15	RPV	500	Y	WW/DW	PCPL
3A	11	72	16	SP	230	Y	15	RPV	500	Y	WW/DW	PCPL
4Aii(1)	8	72	16	SP	230	N	15	RPV	throttle	Y	WW	PCPL
4Aii(1)	9	72	16	SP	230	Y	15	RPV	throttle	Y	WW	PCPL
4Aii(1)	12	72	16	SP	230	N	15	RPV	throttle	Y	WW	PSP
4Aii(1)	13	72	16	CST	230	N	15	RPV	throttle	Y	WW	PCPL
4Ai(1)	14	72	16	SP	230	N	15	RPV	0	Y	WW	PCPL
4Aiii(1)	15	72	16	SP	230	N	15	RPV	throttle	Y	WW	PCPL
4Ai(1)	18	72	16	SP	230	N	15	RPV	500	Y	WW/DW	PCPL
4Ai(1)	16	72	16	SP	230	N	15	RPV	500	Y	WW/DW	PCPL
3B	21	72	16	SP	230	N	15	DW	0	Y	WW	PCPL
3B	24	72	16	SP	230	N	15	DW	500	Y	WW/DW	PCPL
	24dw	72	16	SP	230	N	15	DW	500	Y	DW	PCPL
4Aii(2)	22	72	16	SP	230	N	15	DW	throttle	Y (50%)	WW	PCPL
	22dw	72	16	SP	230	N	15	DW	throttle	Y	DW	PCPL
4Aii(2)	23	72	16	SP	230	N	15	DW	throttle	Y	WW	PCPL
4Aii(2)	25	72	16	SP	230	Y	15	DW	throttle	Y	WW	PCPL
3B	26	72	16	SP	230	Y	15	DW	500	Y	WW	PCPL
4Ai(2)	27	72	16	SP	230	N	15	DW	0	Y	WW	PCPL
4Aiii(2)	28	72	16	SP	230	N	15	DW	throttle	Y	WW	PCPL
	28dw	72	16	SP	230	N	15	DW	throttle	Y	DW	PCPL
4Ai(2)	32	72	16	SP	230	N	15	DW	500	Y	WW/DW	PCPL
4Ai(2)	30	72	16	SP	230	N	15	DW	500	Y	WW/DW	PCPL
	30dw	72	16	SP	230	N	15	DW	500	Y	DW	PCPL
4Aiii(2)	29	72	16	SP	230	Y	15	DW	throttle	Y	WW	PCPL
	29dw	72	16	SP	230	Y	15	DW	throttle	Y	DW	PCPL
4Ai(2)	31	72	16	SP	230	Y	15	DW	500	Y	WW/DW	PCPL
	31dw	72	16	SP	230	Y	15	DW	500	Y	DW	PCPL
3A	41	4	4	SP	230	N	15	RPV	0	N	WW	PCPL
3B	43	4	4	SP	230	N	15	DW	0	N	WW	PCPL
3A	42	4	4	SP	230	N	15	RPV	500	N	WW/DW	PCPL
3B	44	4	4	SP	230	N	15	DW	500	N	WW/DW	PCPL
4Aii(1)	47	4	4	SP	230	N	15	RPV	throttle	Y	WW	PCPL
4Aii(2)	48	4	4	SP	230	N	15	DW	throttle	Y	WW	PCPL
3B	45	-	16	SP	230	-	-	DW	500	Y	DW	PCPL
3B	46	-	16	SP	230	-	-	DW	500	Y	WW/DW	PCPL
3B	49	-	0	-	-	-	-	DW	500	Y	WW/DW	PCPL
4Ai(2)	50	-	0	-	-	-	-	DW	500	Y	WW/DW	PCPL
3B	51	-	16	SP	230	-	15	DW	500	Y	DW	15
3B	52	-	16	SP	230	-	15	DW	500	N	DW	15
3B	53	-	16	SP	230	-	15	DW	500	Y	DW	15

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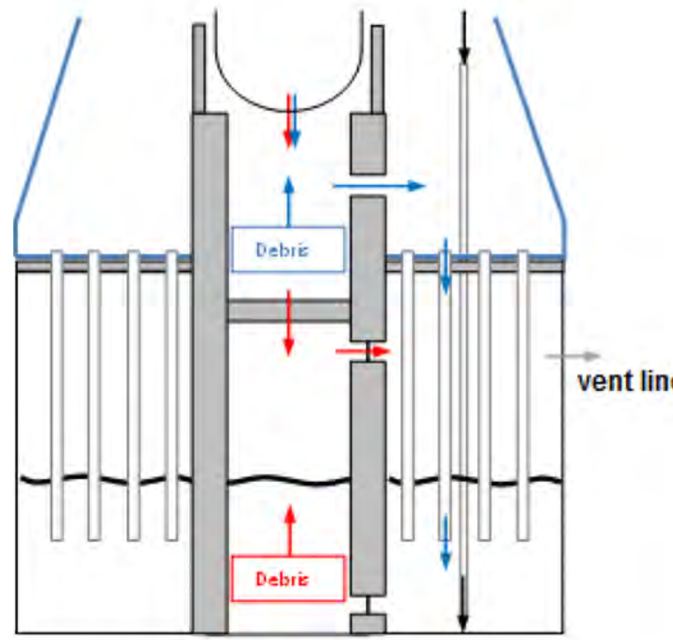
MELCOR Run Matrix for Mark II

		Pre Core Damage						Post Core Damage				
		RPV Pressure control	RCIC Operation				Anticipatory Venting	Flex Operation		SRV Operation	Venting	
		Availability (hr)	RCIC Availability (hr)	RCIC Suction	Failure Temp (F)	Open SRV after RCIC fails	Setpoint (psig)	Injection @ LH failure	SAWA Injection rate (gpm)	Allow SRV stuck open failure?	Location	Setpoint (psig)
Option	Case											
1/2A	1	72	16	SP	230	N	15	-	-	Y	WW	60
	1p1	72	16	SP	230	N	15	-	-	Y	WW	45
1/2A	3	4	4	SP	230	N	15	-	-	N	WW	60
1/2A	5	72	16	CST	230	N	15	-	-	Y	WW	60
1/2A	6	72	16	SP	230	N	15	-	-	Y	WW	30
3A	10	72	16	SP	230	N	15	RPV	500	Y	WW/DW	60
	10p1	72	16	SP	230	N	15	RPV	500	Y	WW/DW	45
3A	11	72	16	SP	230	Y	15	RPV	500	Y	WW/DW	60
	11p1	72	16	SP	230	Y	15	RPV	500	Y	WW/DW	45
3B	24	72	16	SP	230	N	15	DW	500	Y	WW/DW	60
	24p1	72	16	SP	230	N	15	DW	500	Y	WW/DW	45
3A	42	4	4	SP	230	N	15	RPV	500	N	WW/DW	60
3B	44	4	4	SP	230	N	15	DW	500	N	WW/DW	60
3B	45	-	16	SP	230	-	-	DW	500	Y	DW	60
3B	49	-	0	-	-	-	-	DW	500	Y	WW/DW	60
3B	51	-	16	SP	230	-	15	DW	500	Y	DW	15
3B	52	-	16	SP	230	-	15	DW	500	N	DW	15
		Sensitivity to DW-WW bypass and lower reactor cavity (LRC) pool										
1	1a1	Same as case 1 but assuming suppression pool bypass is delayed until upper reactor cavity (URC) floor is completely ablated										
1	1b1	Same as case 1 but assuming lower reactor cavity is filled with water										
1	1b2	Same as case 1b1 but assuming immediate suppression pool bypass (0 min delay)										
1	1b3	Same as case 1b1 but assuming suppression pool bypass is delayed until URC floor is completely ablated										

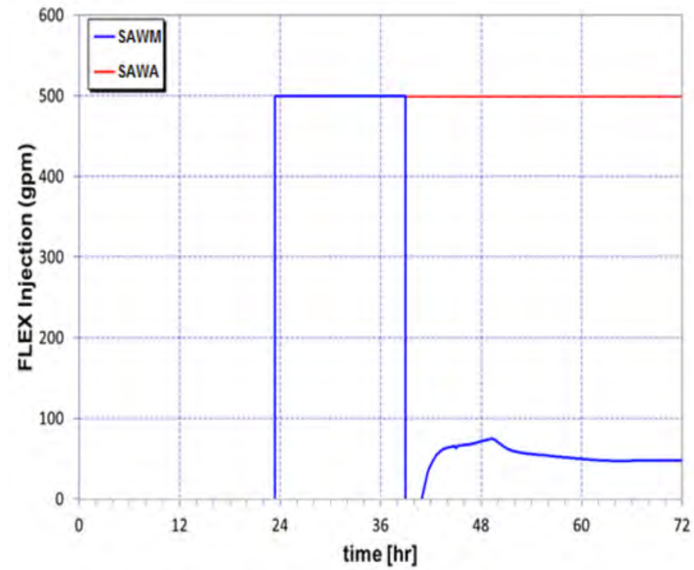


RPV intact / SRV & SP scrubbing
Lower head failure / DCV & SP scrubbing
Suppression pool bypass / MCCI & overlying water scrubbing

MELCOR Results for Mark I

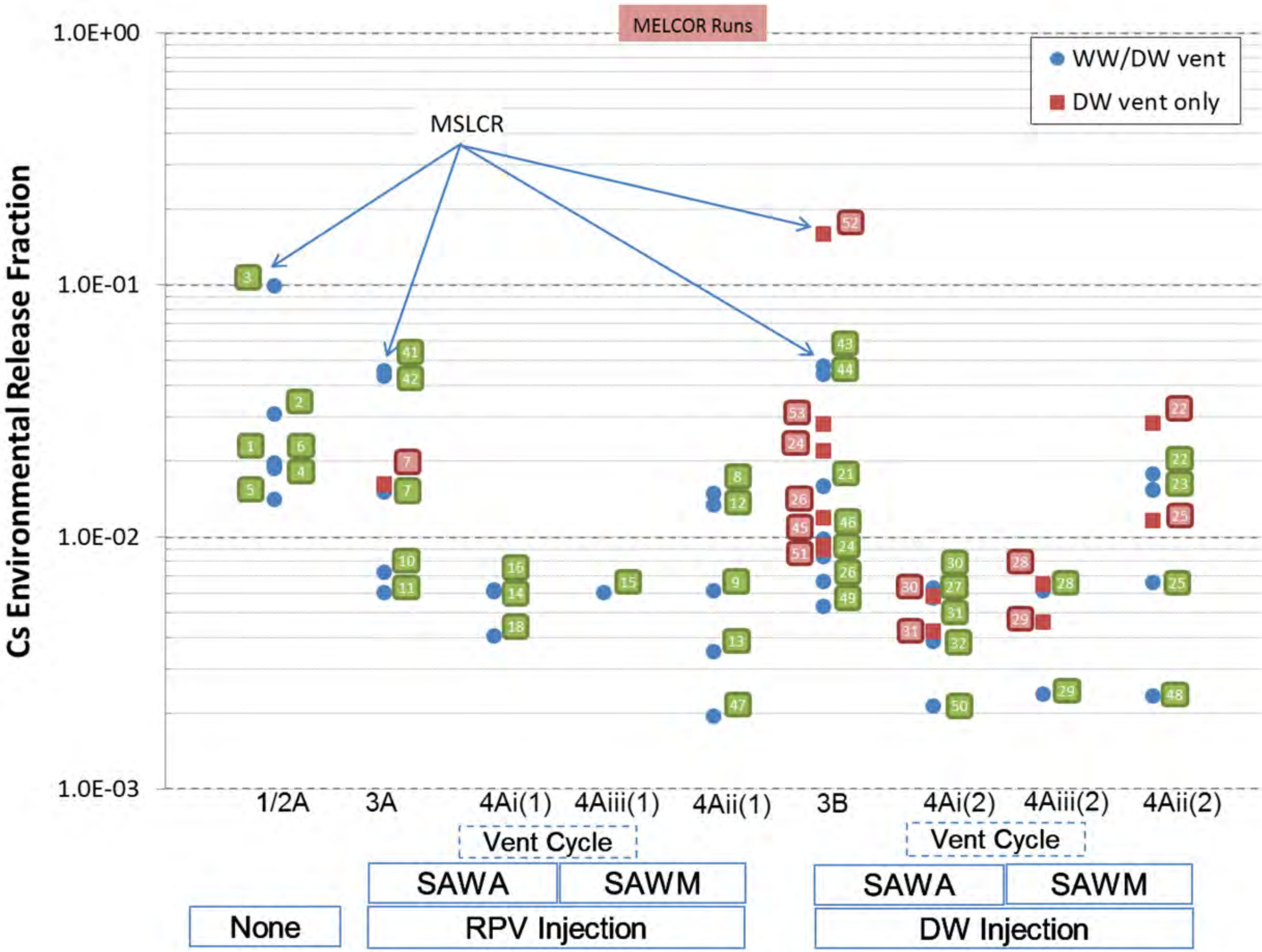
(Timing of key events for selected scenarios)

Event Timing (hr)	Case 1 (no water)	Case 9 (SAWM)	Case 10 (SAWA)
Start of ELAP	0.0	0.0	0.0
Operators first open SRV to control pressure	0.17	0.17	0.17
Low-level 2 and RCIC actuation signal	0.18	0.18	0.18
Operators open SRV to control pressure (200-400 psig)	1.0	1.0	1.0
RCIC flow terminates	9.6	9.6	9.6
SRV sticks open or operators open SRV after RCIC fails	16.0	9.6	16.0
Water level reaches TAF	12.4	11.9	11.9
First hydrogen production	13.7	13.2	13.7
First fuel-cladding gap release	13.7	13.2	13.7
Start of containment venting (WW) at 60 psig	14.9	14.4	16.3
Relocation of core debris to lower plenum	15.6	15.5	15.5
RPV lower head dries out	18.1	18.2	18.9
RPV lower head fails grossly	23.0	23.4	23.1
Drywell head flange leakage	27.1	-	-
Hydrogen burn in reactor building refueling bay	28.8	-	-
Drywell liner melt-through	31.4	-	-
WW vent line closed (high pool water level)	-	-	42.2
Start of containment venting (DW) at 60 psig	-	-	54.3
Calculation terminated	72	72	72
Selected MELCOR Results	**Case 1**	**Case 9**	**Case 10**
Debris mass ejected (1000 kg)	292	280	287
In-vessel hydrogen generated (kg)	1195	1032	1232
Iodine release fraction at 72 hr	2.28E-01	7.86E-02	8.10E-02
Cesium release fraction at 72 hr	1.94E-02	6.12E-03	7.26E-03



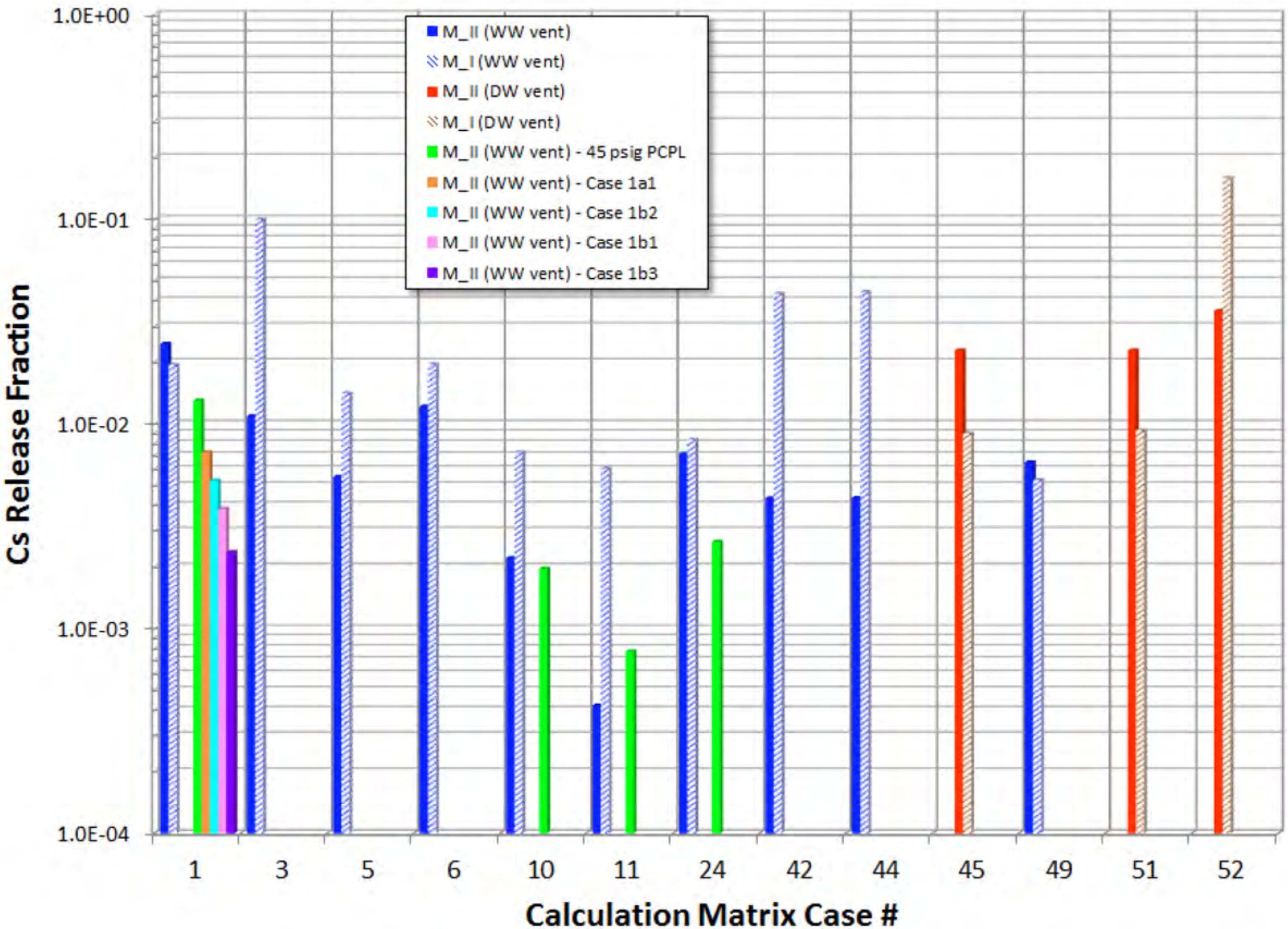
MELCOR Results for Mark I

(Cs release fraction to environment)



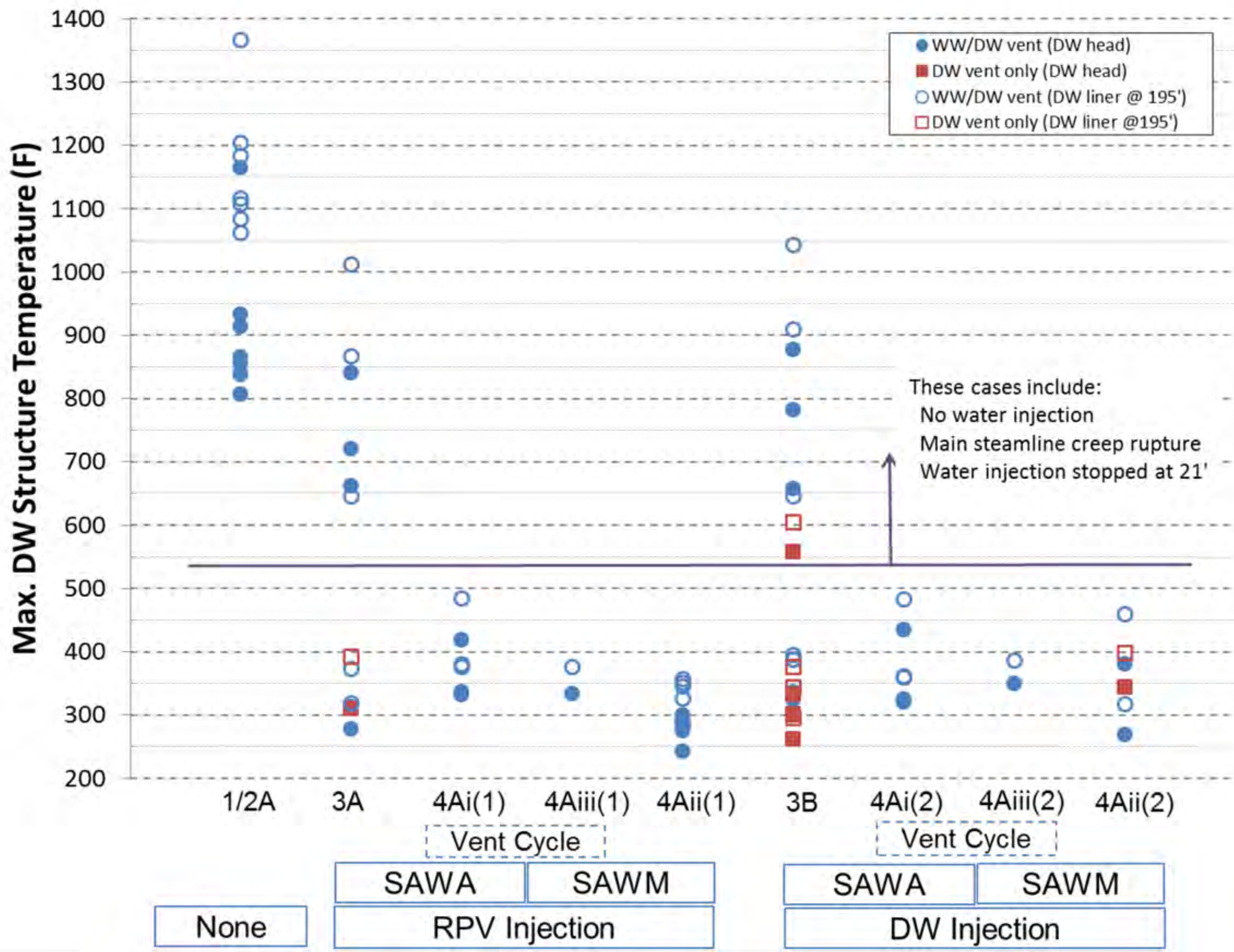
MELCOR Results for Mark II

(Cs release fraction to environment & comparison to selected Mark I scenarios)



MELCOR Results for Mark I

(containment structure temperature)



Conclusions

- A combination of venting and water addition is required to prevent containment failure and is a beneficial strategy for mitigating radiological releases. Containment venting generally results in a puff release to the environment which can occur shortly after core damage and before water addition.
- Anticipatory venting (before core damage) is beneficial to reduce the containment pressure and delay the radionuclide release to the environment.
- Containment venting is efficient in purging hydrogen and non-condensables. Water injection is also helpful in maintaining a steam-inerted atmosphere which can preclude an energetic hydrogen combustion.
- The highest calculated releases to the environment result from a main steam line creep rupture scenario, which is one of the least likely variations.
- The releases to the environment for the Mark II analysis are generally comparable to or lower than those in the Mark I analysis.
- For the Mark II analysis, scoping analyses were performed to investigate different lower cavity configurations. The environmental releases are within the range of source terms predicted based on the variations in the scenario boundary conditions.

Acronyms

BWR	Boiling water reactor
DCV	Downcomer vent
DW	Drywell
MCCI	Molten core concrete interaction
MSLCR	Main steam line creep rupture
PCPL	Primary containment pressure limit
PSP	Pressure suppression pressure
RCIC	Reactor core isolation cooling
RPV	Reactor pressure vessel
SAWA	Severe accident water addition
SAWM	Severe accident water management
SOARCA	State-of-the-art reactor consequence analysis
SP	Suppression pool
SRV	Safety/Relief valve
WW	Wetwell

BACKUP SLIDES

MELCOR Analysis Assumptions

- All MELCOR transients start with an ELAP
- All transients are 72 hour in duration
- Industry (BWROG) EPG/SAG Rev. 3 is in place
- FLEX is in place both pre- and post-core damage
  - 500 gpm injection into <u>RPV</u> or <u>Drywell</u> from external source at vessel breach
  - Provision for both Severe Accident Water Addition (SAWA) or Severe Accident Water Management (SAWM)
- Initial buildup of water in the drywell from nominal leakage
- Possible end states of accident progression include
  - Liner melt-through (LMT) – Mark I only
  - Main steam line creep rupture (MSLCR)
  - Drywell head flange leakage by overpressure and overtemperature

MELCOR Analysis Assumptions

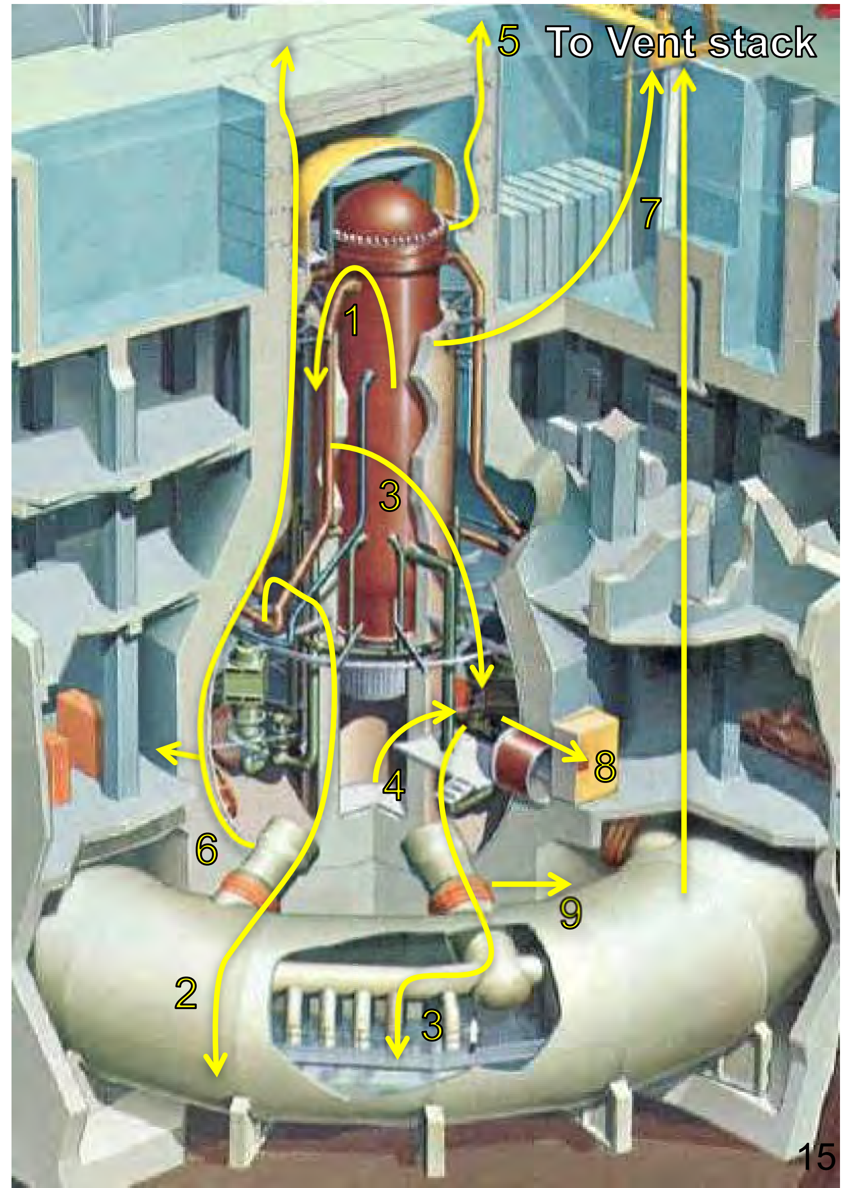
- **RCIC operation**
  - Suction from SP (option for suction from CST/SP)
  - Flow rate 600 gpm
  - RPV level control via throttling of RCIC
- **RPV pressure control**
  - Initial pressure control in 800 – 1000 psig band after 10 min
  - Controlled depressurization after one hour
  - Subsequent pressure control in 200 – 400 psig band for continued RCIC operation

MELCOR Analysis Assumptions

- **Containment venting**
  - Anticipatory venting prior to core damage (15 psig)
    - Not performed if RCIC already failed
  - Upon entry into SAG vent closed; reopens at PCPL (60 psig) – option to reopen at PSP considered
  - Transition from WW to DW venting at SP high water level (21’ above bottom of torus for Mark I and 50’ for Mark II)
  - Vent cycling in (PCPL)/(PCPL-10) band; option with PSP considered
  - Vent sizing consistent with industry assumptions

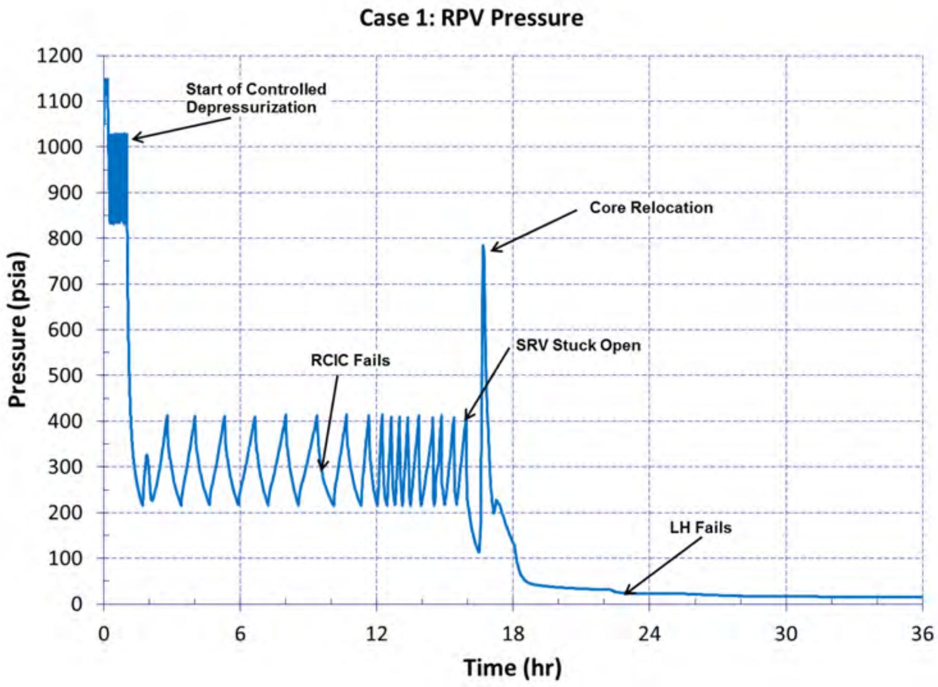
Mark I Fission Product Pathways

- From degrading core
- 1. Through upper RPV internals and MSL
- To Suppression Pool
  2. SRVs
  – 3. DW and Main vents if RCS ruptures
- If vessel ruptures
  – 4. Core Concrete Interaction
- To environment
  – 5. Drywell head leakage
  – 6. Liner melt through
  – 7. Venting
  – 8. Other penetration leaks
  – 9. Bellows ruptures

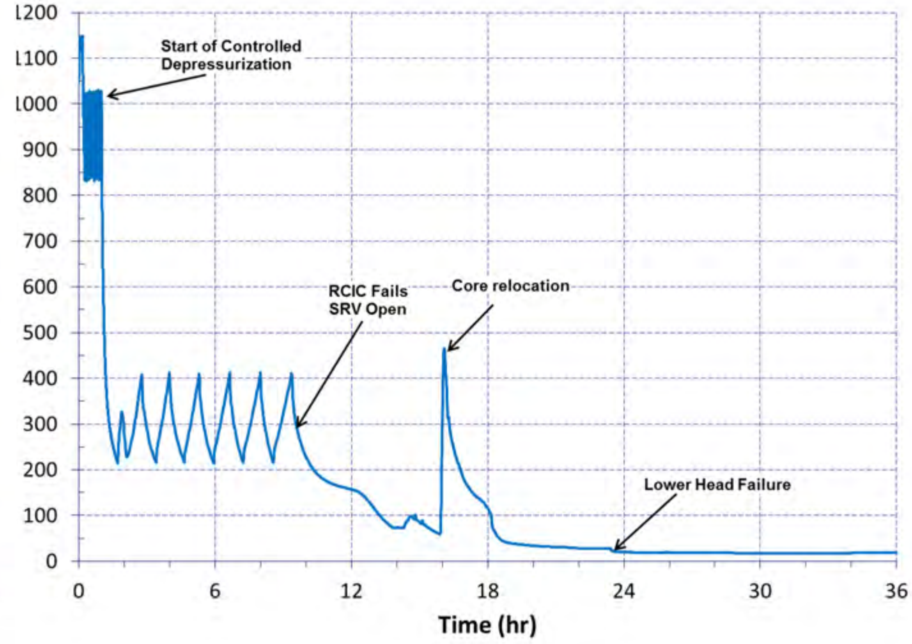


MELCOR Results for Mark I
(RPV pressure response)

Case 1: No injection



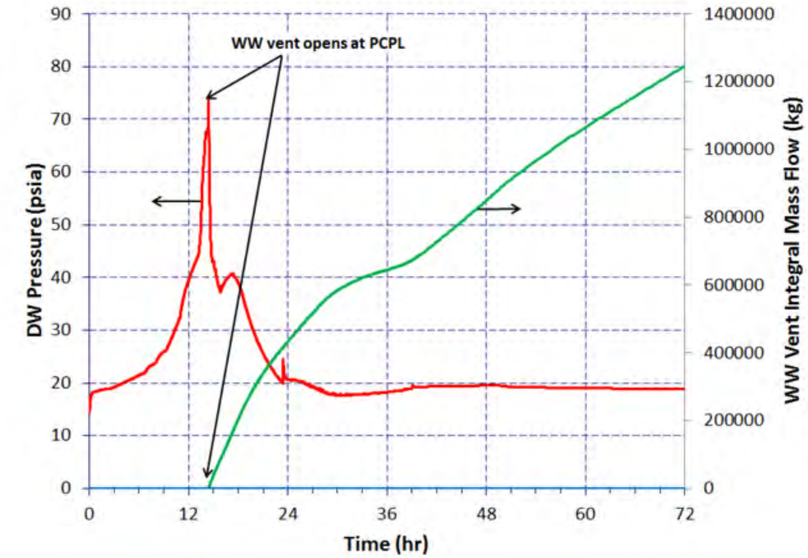
Case 9: SAWM

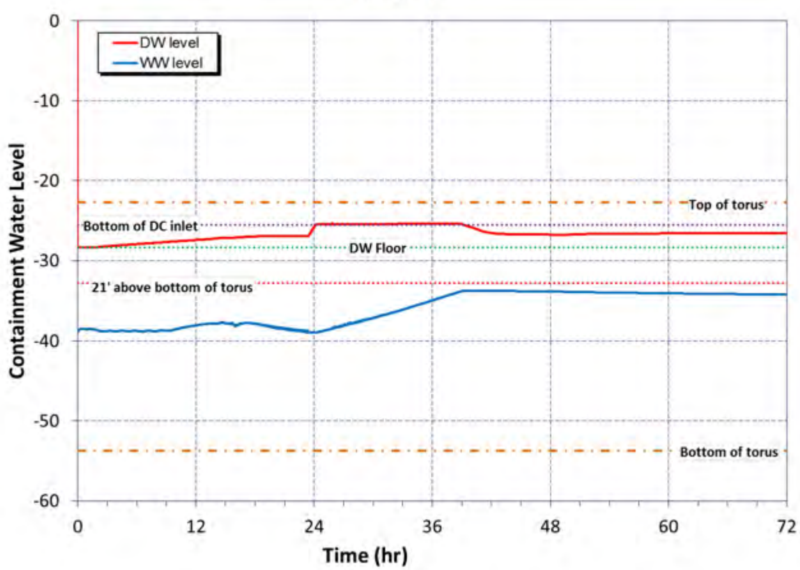


MELCOR Results for Mark I

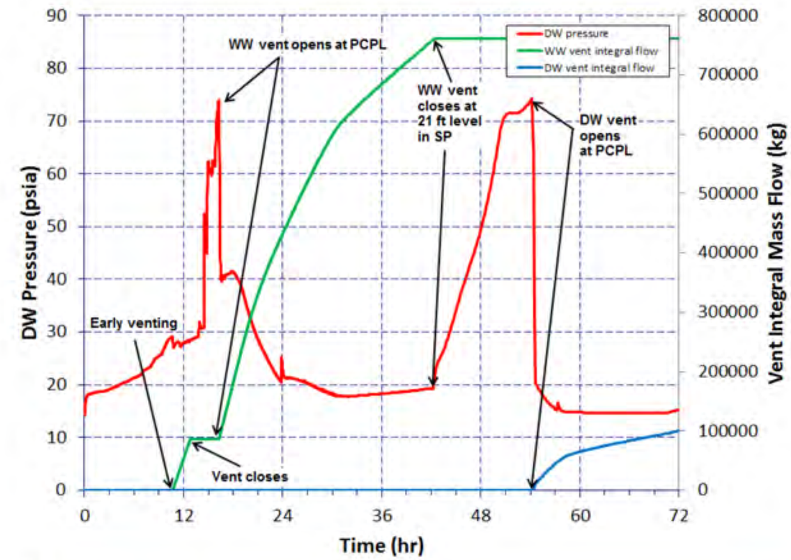
(containment pressure & water level)

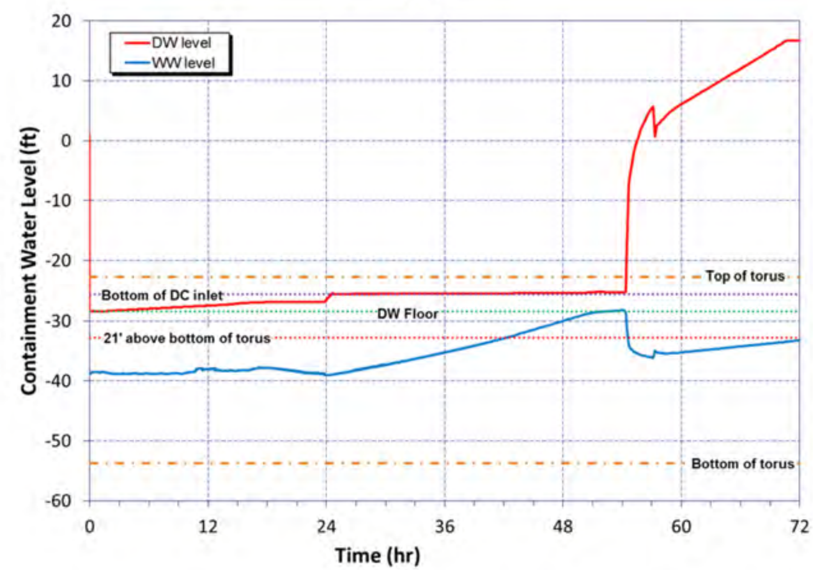
Case 9: SAWM





Case 10: SAWA

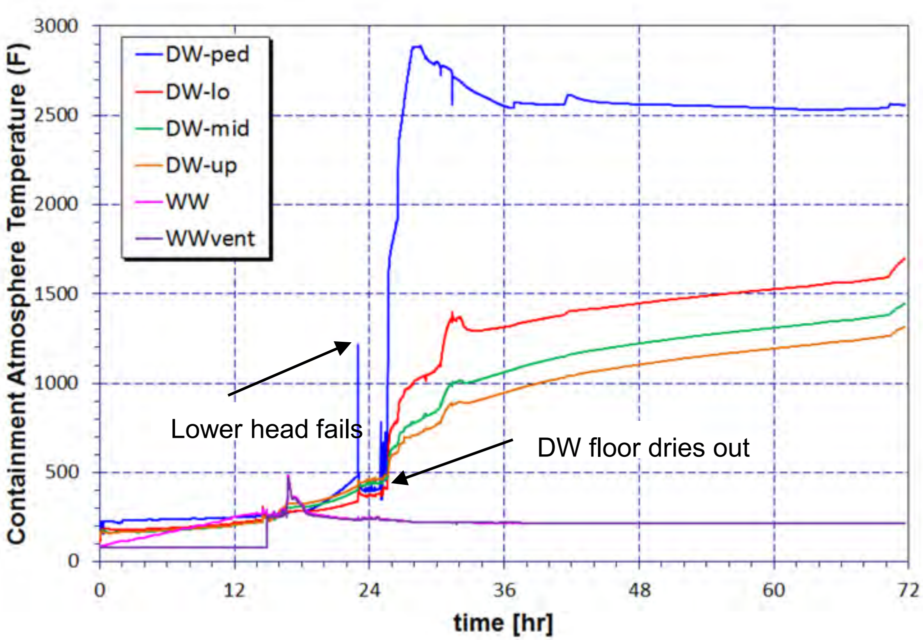




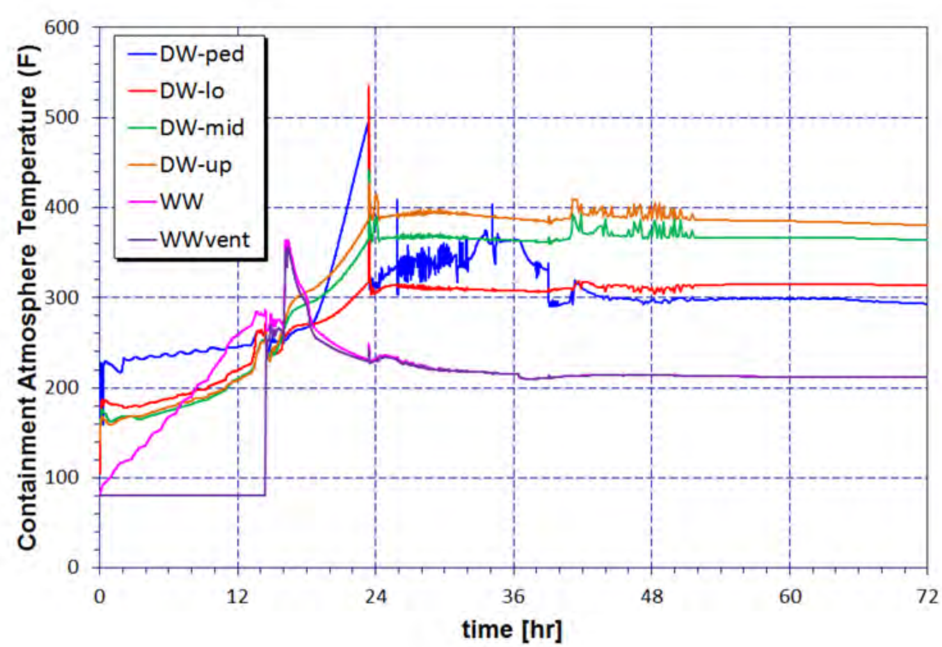
MELCOR Results for Mark I

(containment temperature response)

No water injection



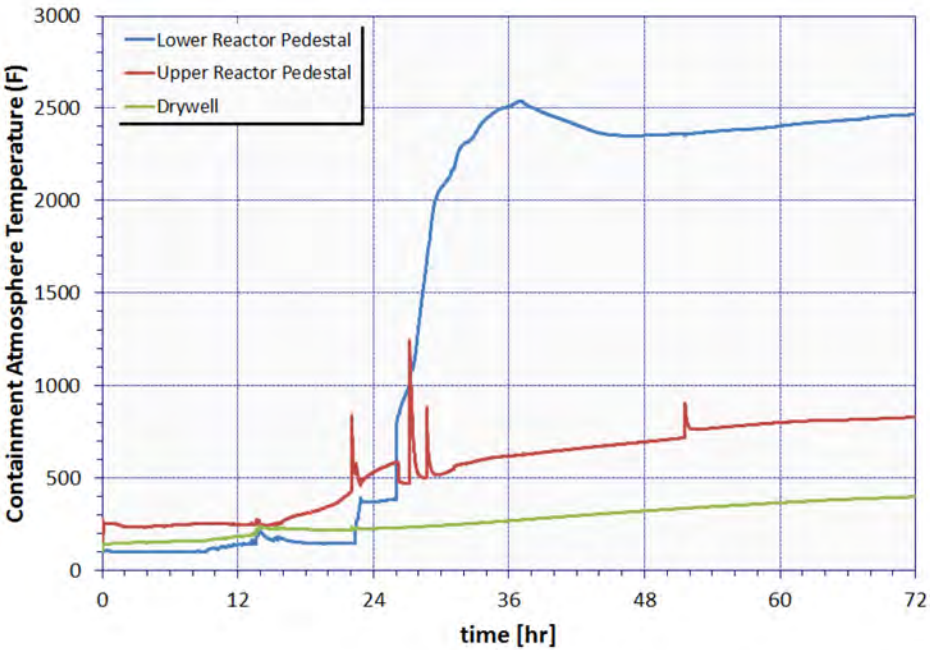
With water injection



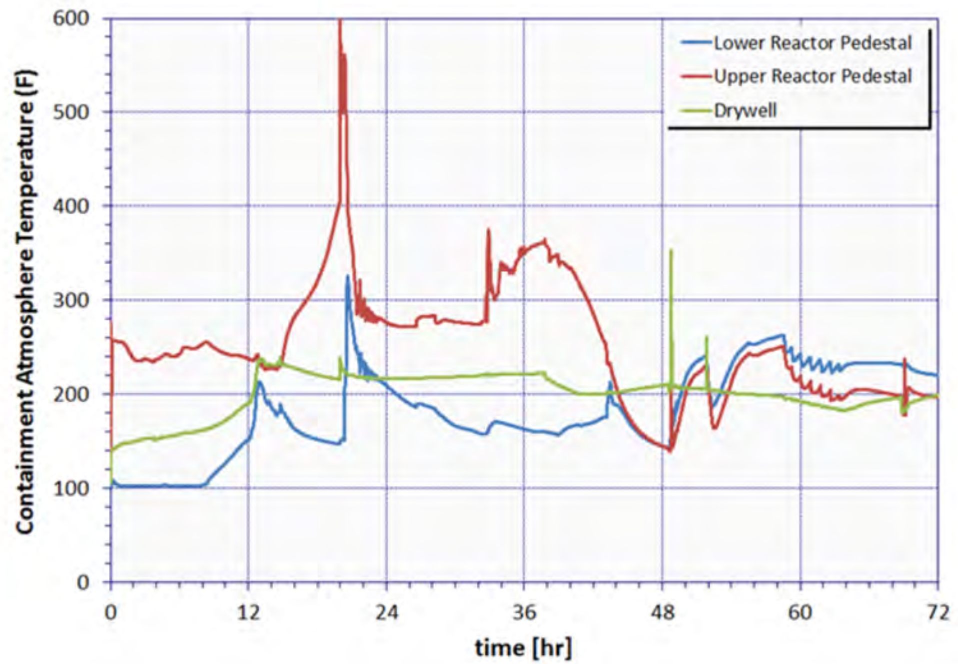
MELCOR Results for Mark II

(containment temperature response)

No water injection

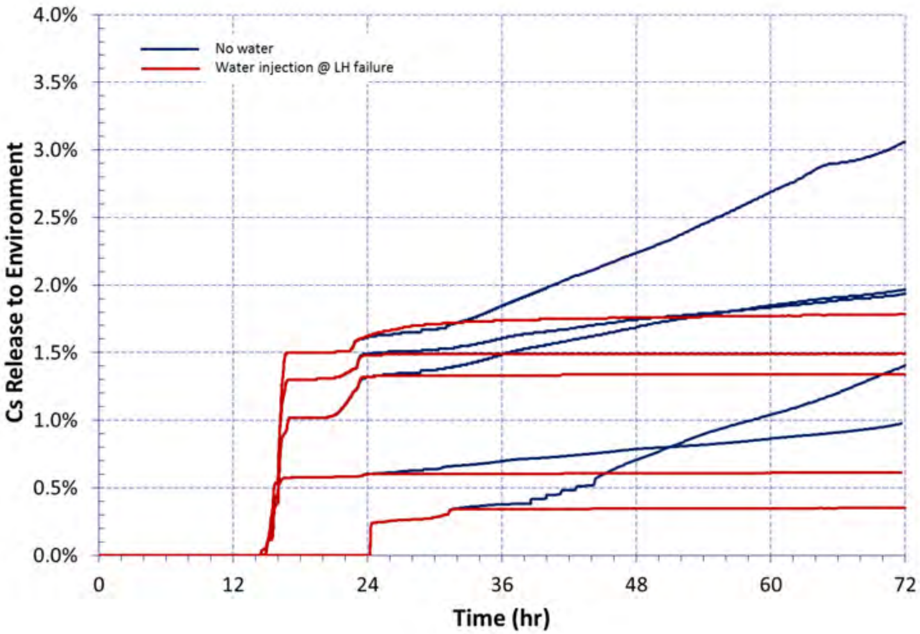


With water injection



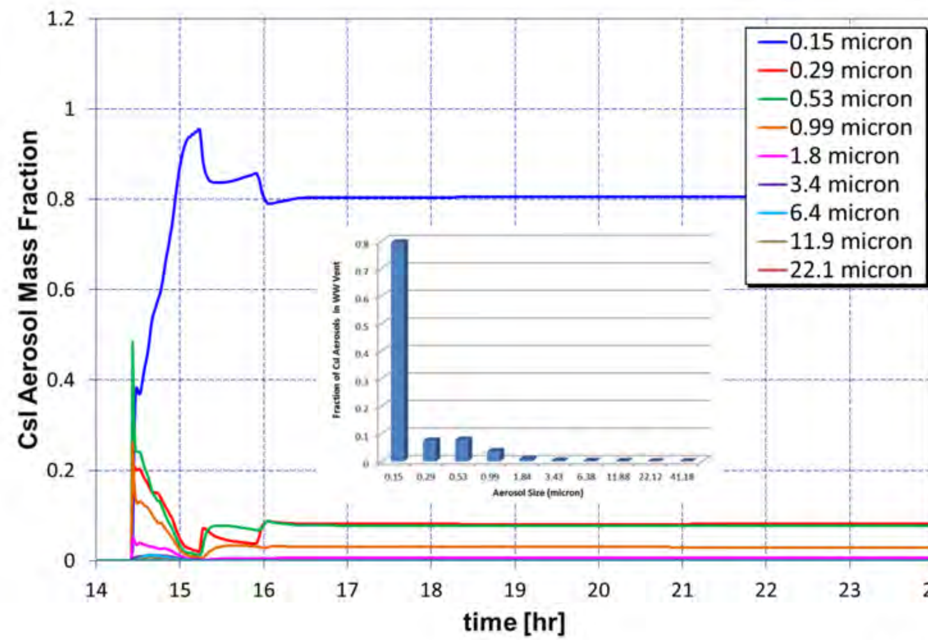
MELCOR Results for Mark I

(Cs release fraction to environment)



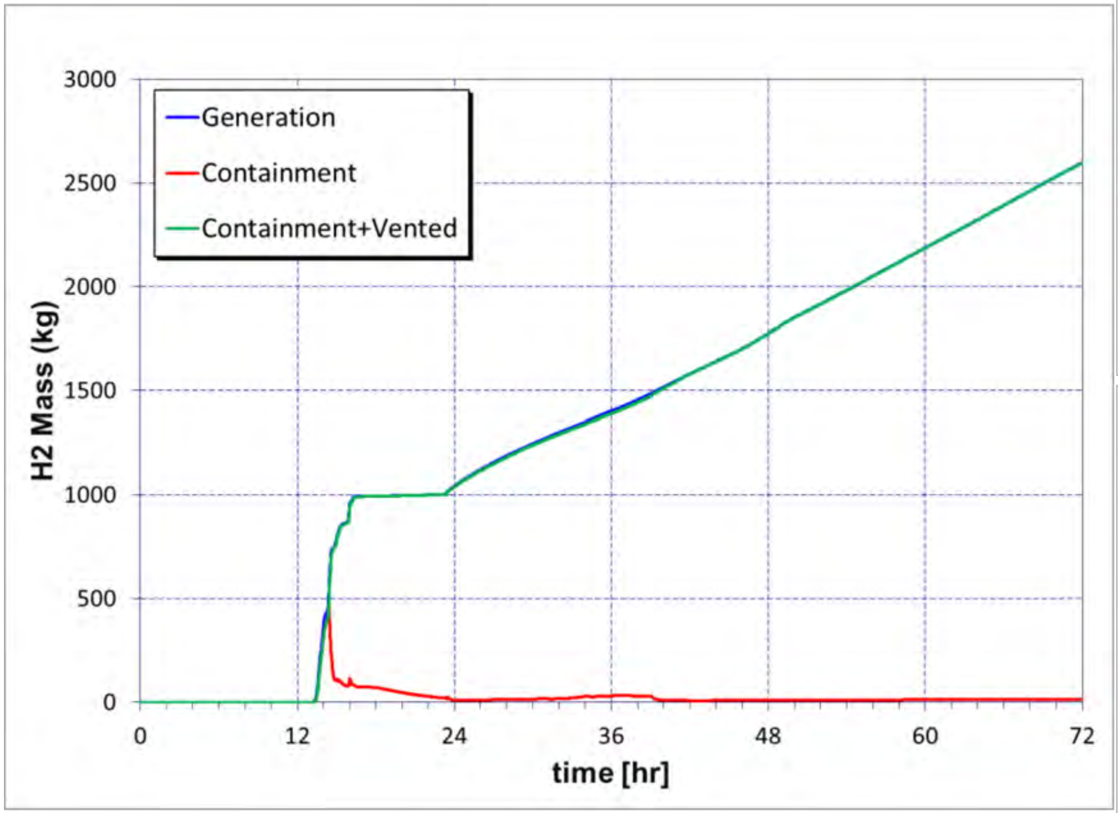
Water addition at lower head failure has the benefit of mitigating further release, but does not affect the release at the time of venting

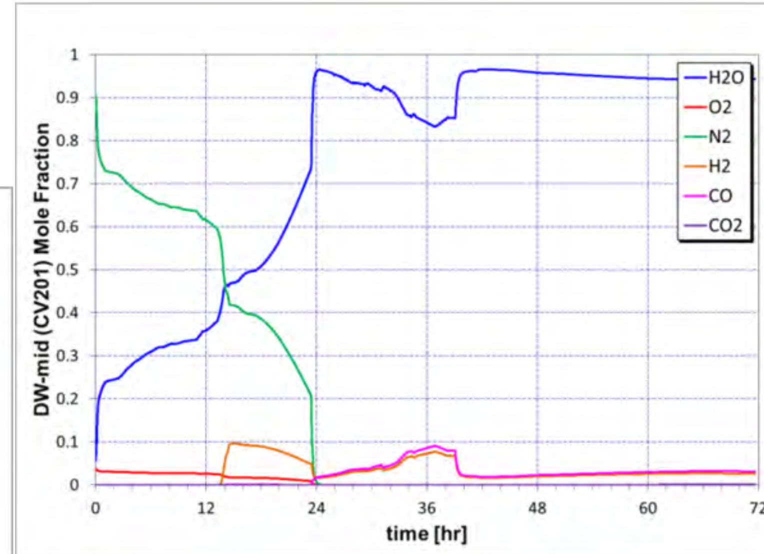
Particle size distribution dominated by very small aerosols at the time of venting

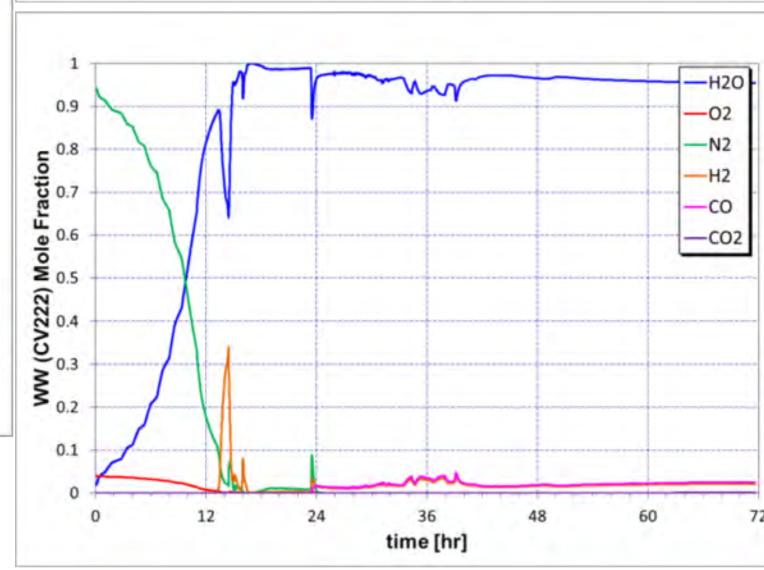


MELCOR Results for Mark I

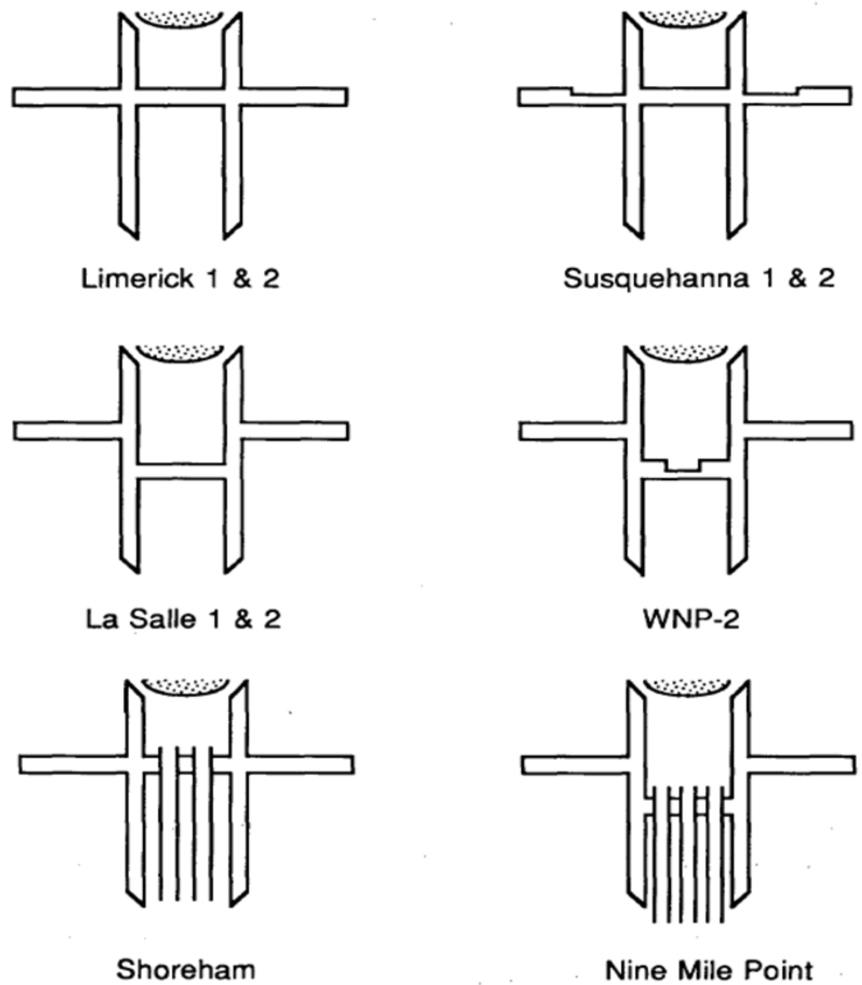
(Hydrogen generation and transport)







BWR Mark II Containment Designs



U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

MACCS Offsite Consequence Analysis in Support of CPRR Rulemaking

Jonathan Barr

Office of Nuclear Regulatory Research

Accident Analysis Branch

July 7, 2015

ACRS Subcommittee Meeting

MACCS Modeling Approach

- Develop site-specific MACCS models for
  - reference BWR Mark I site (Peach Bottom)
  - reference BWR Mark II site (Limerick)

- Sensitivity calculations to explore impact of selected input parameters

- Incorporate most current data sources available and consequence modeling best practices

Technical Elements

- Site Data
- Meteorology
- Atmospheric Transport and Deposition
- Emergency Phase Protective Actions
- Intermediate & Long-Term Phase Protective Actions
- Dosimetry & Health Effects
- Radionuclide Release

Radionuclide Release

- 41 MELCOR source terms for Mark I, 12 for Mark II
- Range of decontamination factors applied to source terms for releases through vented pathways: 10, 100, and 1000
- 164 total Mark I source terms, 48 for Mark II
- Binning strategy used for efficiency of consequence analysis
- Binning based on cesium and iodine magnitude
- 18 source term bins for Mark I, 9 for Mark II

BWR Mark I Source Term Bins

Bin	Bin Cs Range (%)	Bin I Range (%)	Representative Case	Rep Case Cs (%)	Rep Case I (%)	Start of Release (hrs)
1	0.0002 - 0.001	0.001 - 0.01	28DF1000	0.0006%	0.006%	15.9
2	0.001 - 0.003	0.01 - 0.03	48DF100	0.002%	0.02%	11.4
3	0.003 - 0.01	0.03 - 0.1	10DF100	0.01%	0.08%	16.3
4	0.01 - 0.03	0.1 - 0.3	7DF1000	0.02%	0.26%	14.9
5	0.03 - 0.1	0.3 - 1.0	11DF10	0.06%	0.78%	14.4
6	0.1 - 0.3	1.0 - 3.0	48	0.23%	1.69%	11.4
7	0.3 - 1.0	3.0 - 10.0	15	0.60%	5.85%	15.9
8	0.3 - 1.0	10.0 - 20.0	46	0.98%	11.01%	14.8
9	1.0 - 2.0	2.0 - 4.0	5DF10	1.05%	2.89%	24.2
10	1.0 - 2.0	4.0 - 10.0	5	1.39%	6.46%	24.2
11	1.0 - 2.0	10.0 - 20.0	8	1.49%	19.25%	14.9
12	1.0 - 2.0	20.0 - 40.0	1	1.93%	22.68%	14.9
13	2.0 - 4.0	3.0 - 10.0	41DF1000	3.40%	7.65%	9.8
14	2.0 - 4.0	10.0 - 20.0	22dw	2.82%	18.64%	15.9
15	2.0 - 4.0	20.0 - 40.0	53	2.79%	29.05%	18.4
16	4.0 - 10.0	10.0 - 20.0	41	4.54%	14.10%	9.8
17	4.0 - 10.0	20.0 - 40.0	3DF10	8.85%	24.65%	9.8
18	10.0 - 20.0	20.0 - 40.0	52	15.90%	34.32%	18.4

BWR Mark II Source Term Bins

Bin	Bin Cs Range (%)	Bin I Range (%)	Representative Case	Rep Case Cs (%)	Rep Case I (%)	Start of Release (hrs)
1	0.00001 - 0.0001	0.0001 - 0.001	11DF1000	0.00004%	0.0005%	20.3
2	0.0001 - 0.001	0.001 - 0.01	5DF1000	0.0006%	0.005%	32.2
3	0.001 - 0.01	0.01 - 0.1	42DF100	0.0043%	0.037%	14.3
4	0.01 - 0.1	0.1 - 1.0	11	0.042%	0.45%	20.3
5	0.1 - 0.4	1.0 - 3.0	51DF10	0.23%	2.01%	17.6
6	0.4 - 1.0	3.0 - 10.0	5	0.55%	4.94%	32.2
7	1.0 - 2.0	~ 10.0	3	1.09%	10.26%	14.3
8	2.0 - 3.0	~ 20.0	1	2.46%	19.81%	22.8
9	3.0 - 4.0	~ 30.0	52	3.57%	28.67%	17.6

Sensitivity Calculations

- Delay to start evacuation (1 hr)
- Increase in nonevacuating cohort fraction (5%)
- Intermediate phase duration (0, 3 months, 1 yr)
- Long-term phase habitability criterion (500 mrem/yr, 2 rem/yr)
- Population density (low, medium, high)

MACCS Results and Conclusions Related to Quantitative Health Objectives (QHOs)

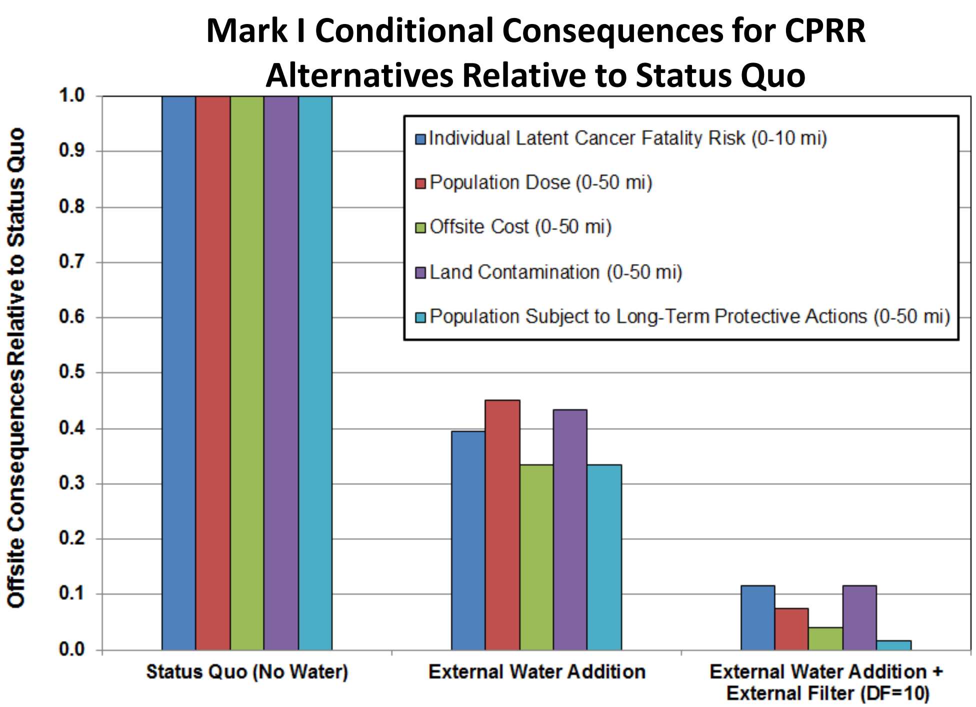
- For source terms including 100+ sensitivity calculations, there is zero early fatality risk
- Frequency-weighted individual latent cancer fatality (LCF) risk is orders of magnitude below the NRC Safety Goal for all alternatives including the status quo (no water) case even for the highest conditional LCF risk among all 100+ sensitivities
- Conditional LCF risk (per event) is dominated by long-term phase low level exposures
  - Because of habitability criterion, LCF risk is relatively insensitive to changes in input parameters
  - Offsite cost, land contamination, and population subject to long-term protective actions are often more sensitive

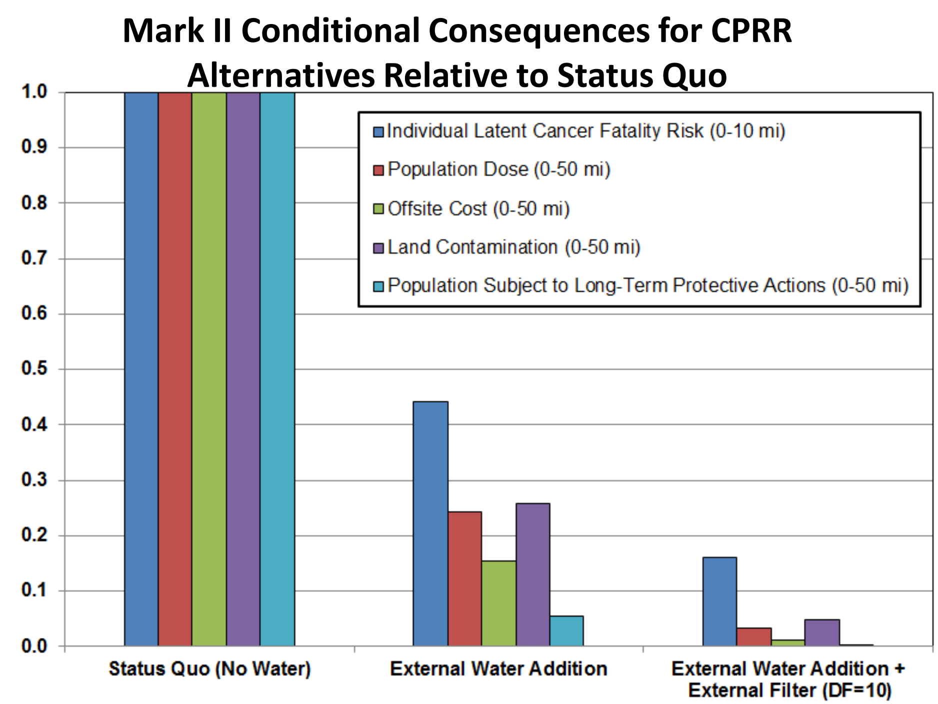
Conditional Offsite Consequence Results Grouped by CPRR Alternative (1)

	Offsite Consequences (per event) / CPRR Alternative	Individual Latent Cancer Fatality Risk (0-10 mi)	Population Dose (0-50 mi) (person-rem)	Offsite Cost (0-50 mi) ($ 2013)	Land Contamination (0-50 mi) (square miles)	Population Subject to Long-Term Protective Actions (0-50 mi)
Mark I	Status Quo (No Water)	3.3E-04	1,600,000	12,000,000,000	530	69,000
Mark I	External Water Addition Successful	1.3E-04	720,000	4,000,000,000	230	23,000
Mark I	External Water Addition Successful and External Filter with DF=10	3.8E-05	120,000	490,000,000	62	1,100
Mark II	Status Quo (No Water)	3.4E-04	4,100,000	63,000,000,000	620	470,000
Mark II	External Water Addition Successful	1.5E-04	1,000,000	9,700,000,000	160	26,000
Mark II	External Water Addition Successful and External Filter with DF=10	5.5E-05	140,000	690,000,000	30	690

Conditional Offsite Consequence Results Grouped by CPRR Alternative (2)

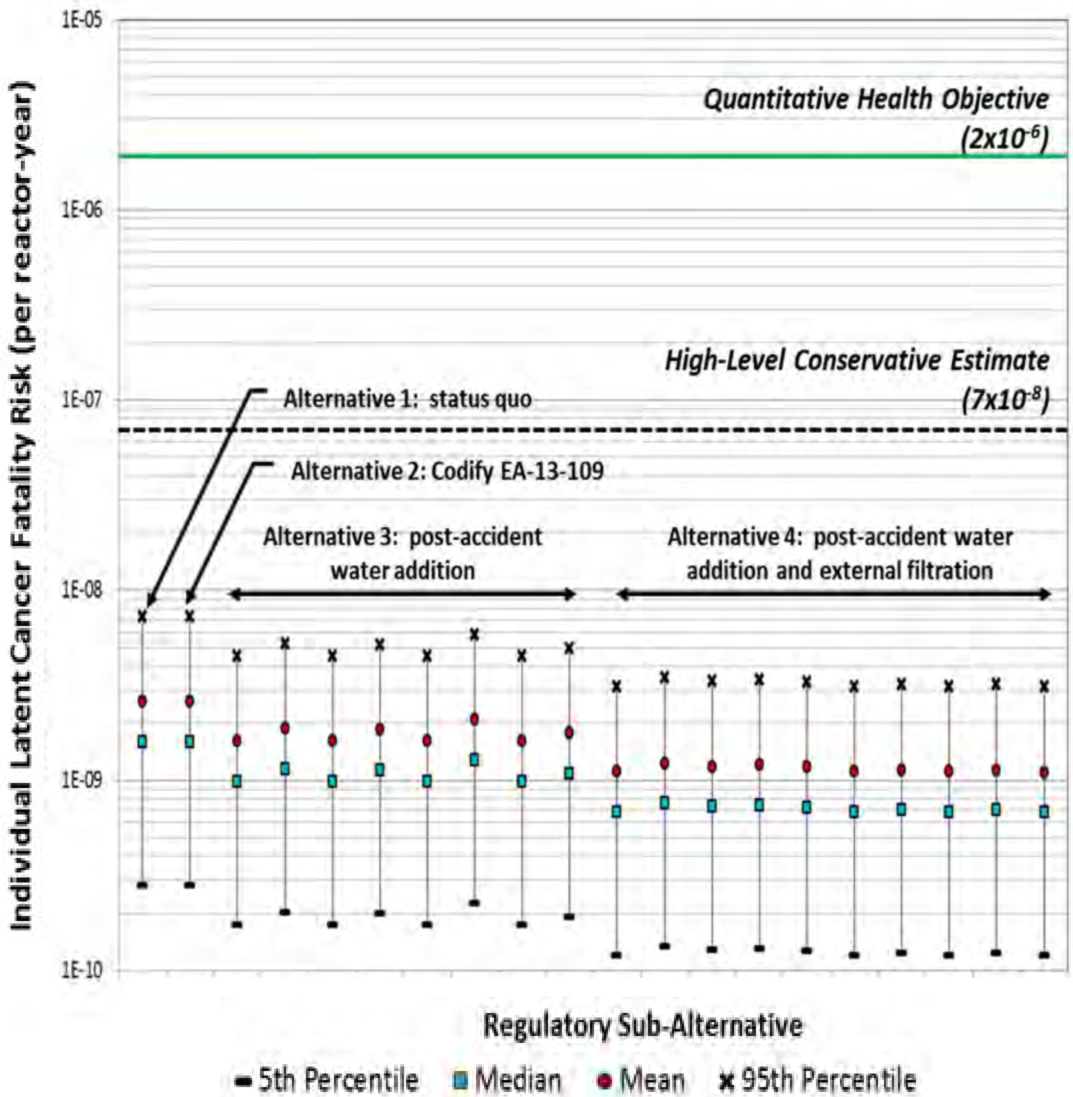
- Regulatory alternatives result in notable reductions in offsite consequences in the event the accident occurs and external water addition is successful
- For both the Mark I and Mark II analysis, the greatest consequence reduction is for the number of people that would be subject to long-term protective actions





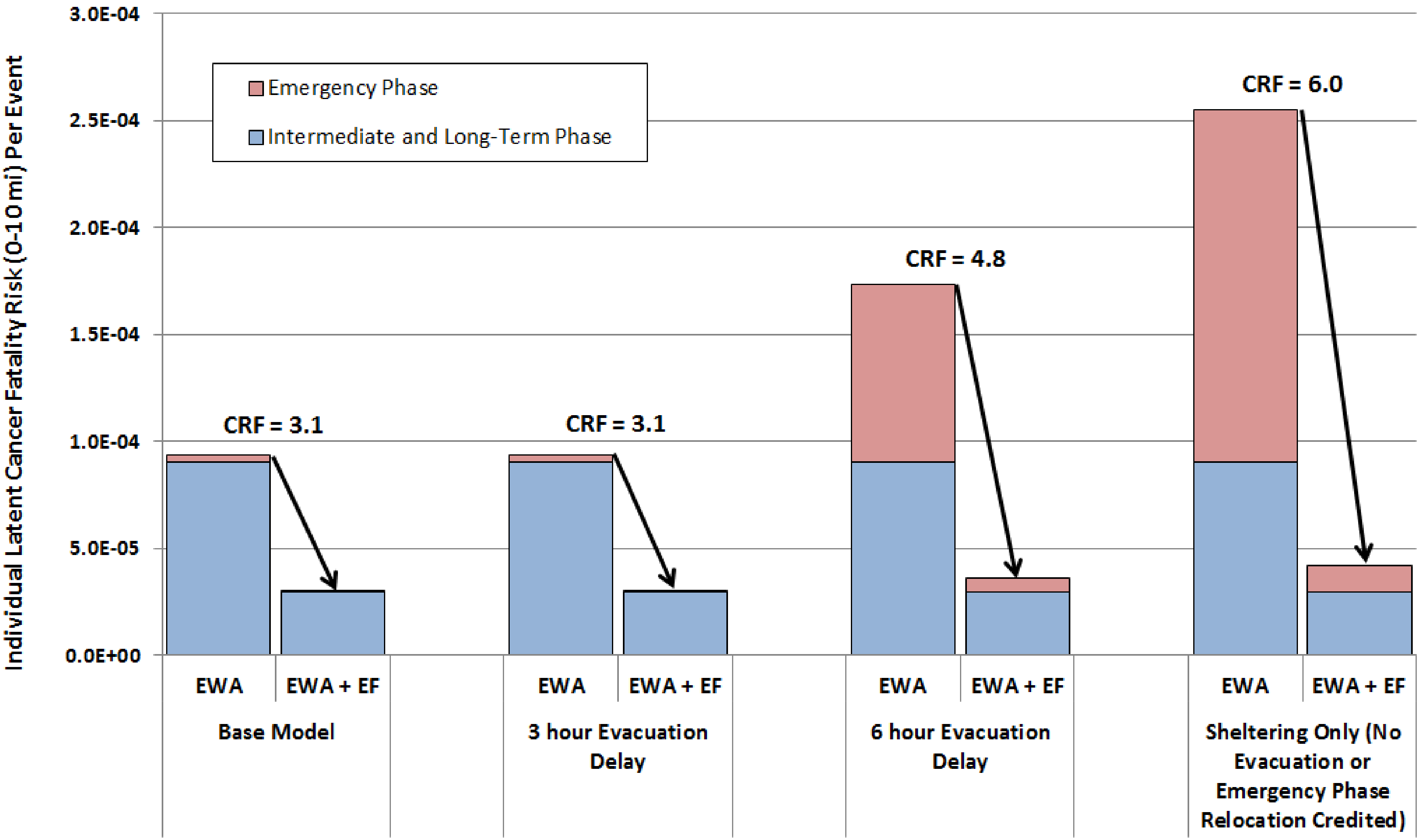
Comparison to NRC Safety Goal

- Frequency-weighted individual LCF risk is orders of magnitude below the NRC Safety Goal QHO
- High-level conservative estimate using highest ELAP frequency and highest conditional LCF risk about 30 times below QHO
- Risk reduction from external water addition and an external filter are within uncertainty bounds
- Protective actions are designed primarily to reduce health risk so individual LCF risk is relatively insensitive to modeling changes



Evacuation Sensitivity Calculations for the Fastest Release MELCOR Source Term

(*MELCOR Mark I Case 49: 7.3 hrs, 0.5% Cs, 1.7% I*)

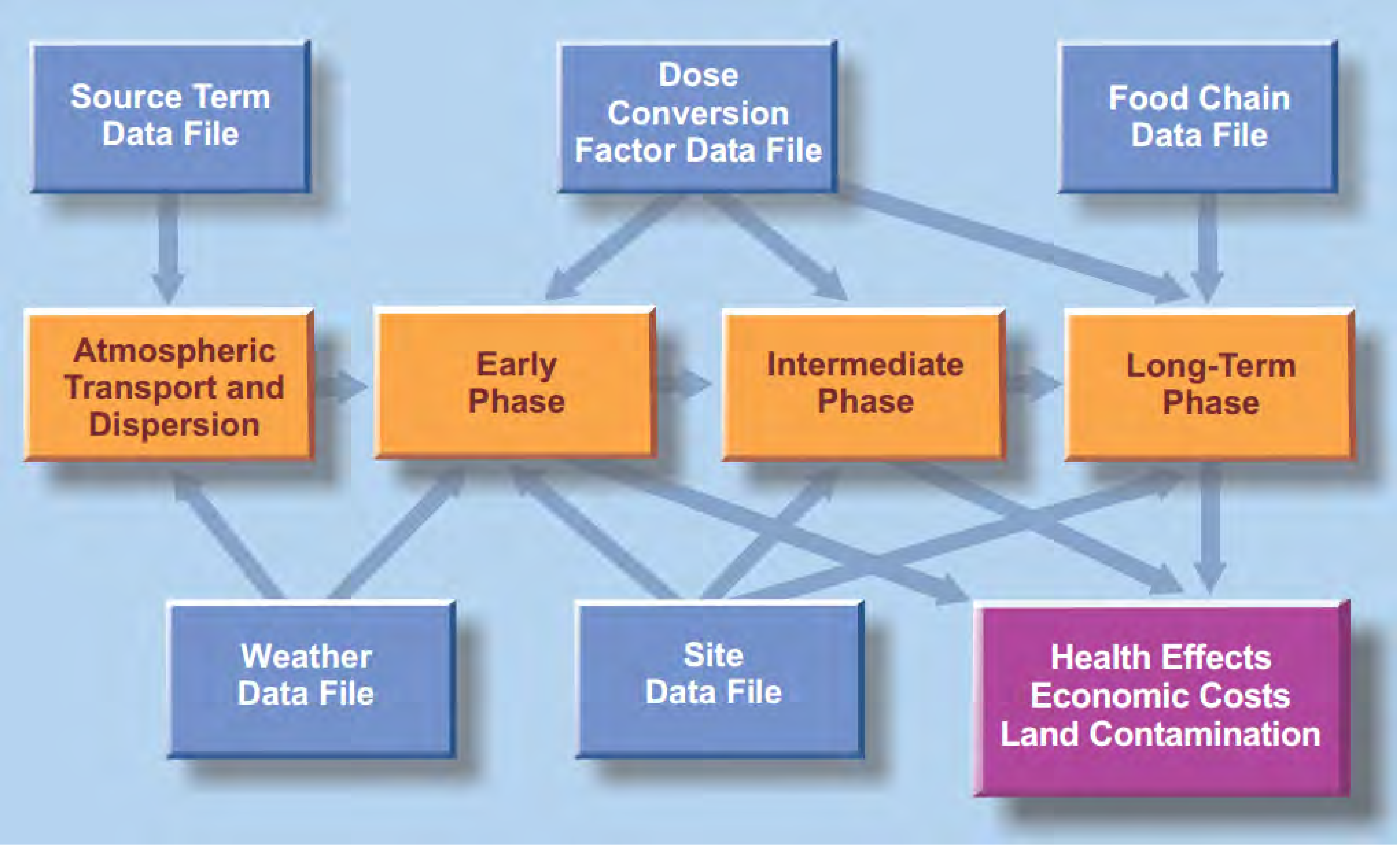


EWA: External Water Addition (500 gpm from FLEX pump) into RPV/Drywell upon lower head failure

EF: External Filter (assumed decontamination factor = 10)

Backup Slides

MACCS Overview

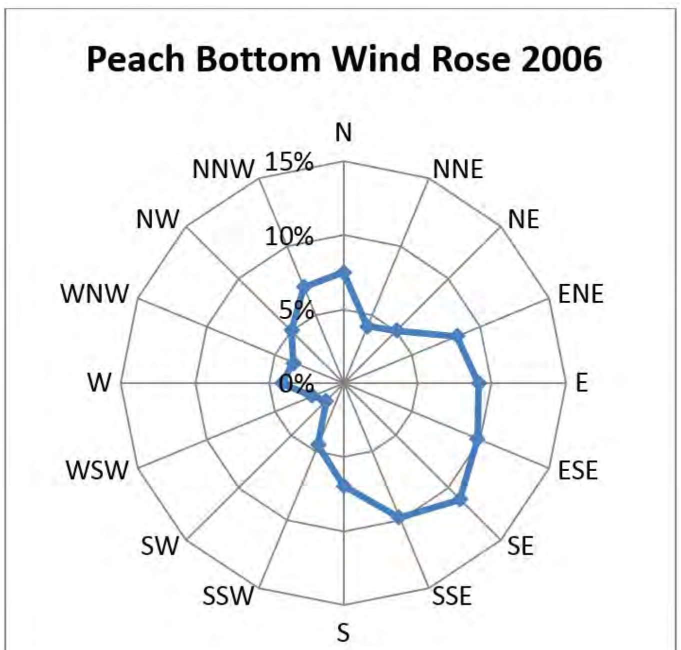


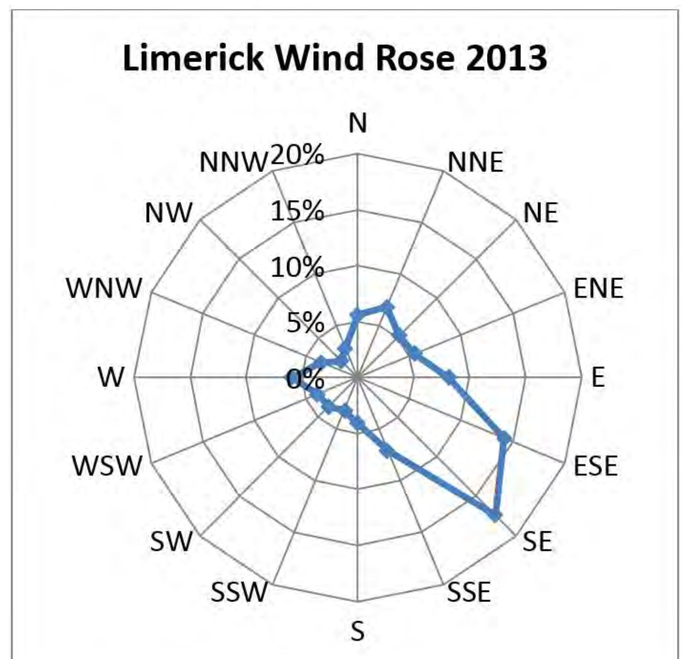
Phases of the MACCS Conceptual Model

	Early (Emergency) Phase	Intermediate Phase	Long-Term Phase
Primary Offsite Accident Response Objective(s)	• Protect public from plume exposures	• Protect public from exposures to deposited materials • Plan for long-term cleanup and recovery activities	• Protect public exposures to deposited materials • Conduct long-term cleanup and recovery activities
Typical Duration and Time Frame	~ 1 week, starting at the time of the accident's initiating event	Weeks to years, starting at the end of the early phase	Months to decades, starting at the end of the intermediate phase
Exposure Pathways	• Inhalation • Skin deposition • Cloudshine • Groundshine	• Groundshine • Inhalation of resuspended materials	• Groundshine • Inhalation of resuspended materials • Food and water ingestion
Protective Actions	• Sheltering • KI ingestion • Evacuation • Relocation	• Relocation	• Interdiction • Decontamination • Condemnation

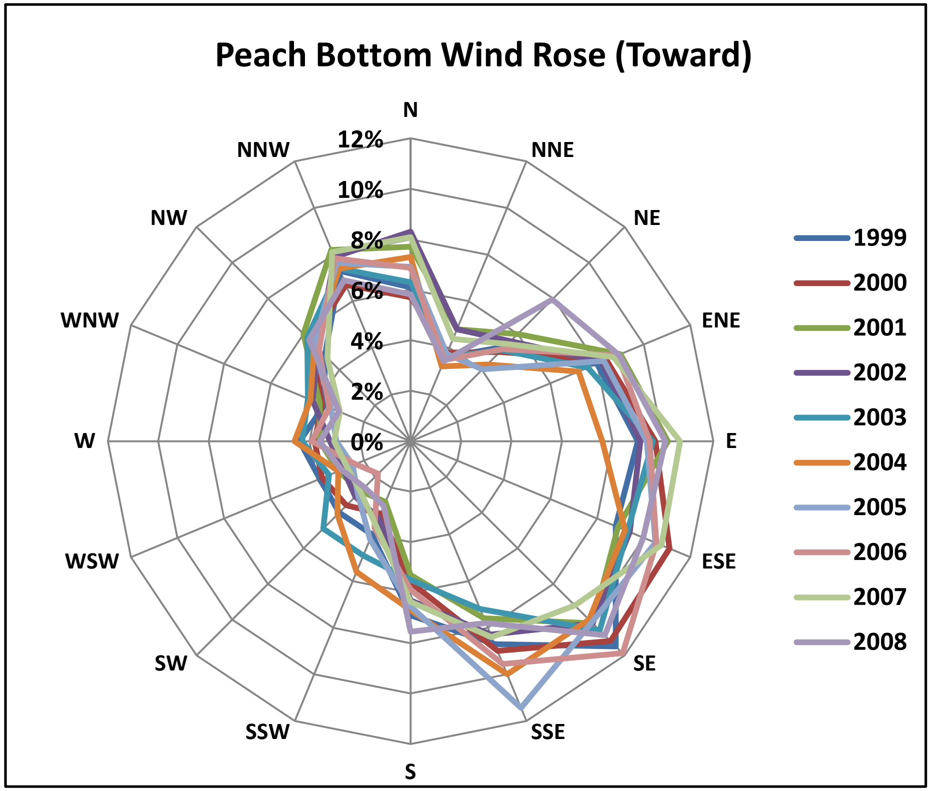
Summary of Meteorological Data

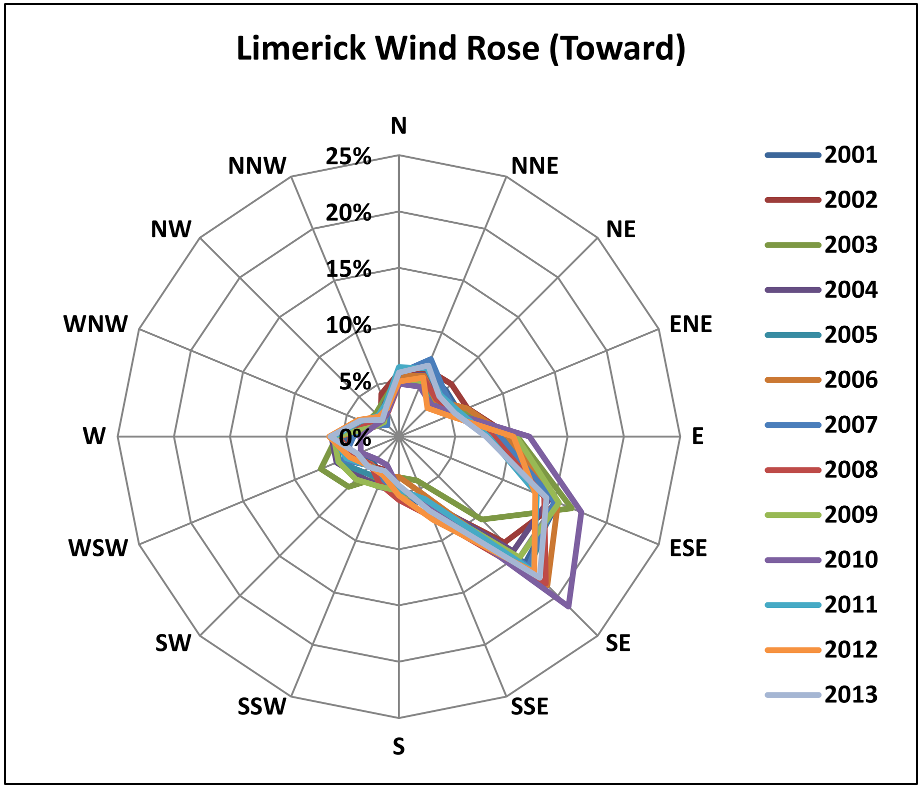
		Peach Bottom Year 2006	Limerick Year 2013
Average Wind Speed (m/s)		2.12	2.36
Precipitation	Total (in)	44.42	44.92
	Hours	602	650
	Frequency (%)	6.87%	7.42%
Stability Class Frequency (%)	Unstable	17.75%	7.33%
	Neutral	24.57%	47.91%
	Stable	57.68%	44.76%
Joint Data Recovery (%)		99.25%	95.19%

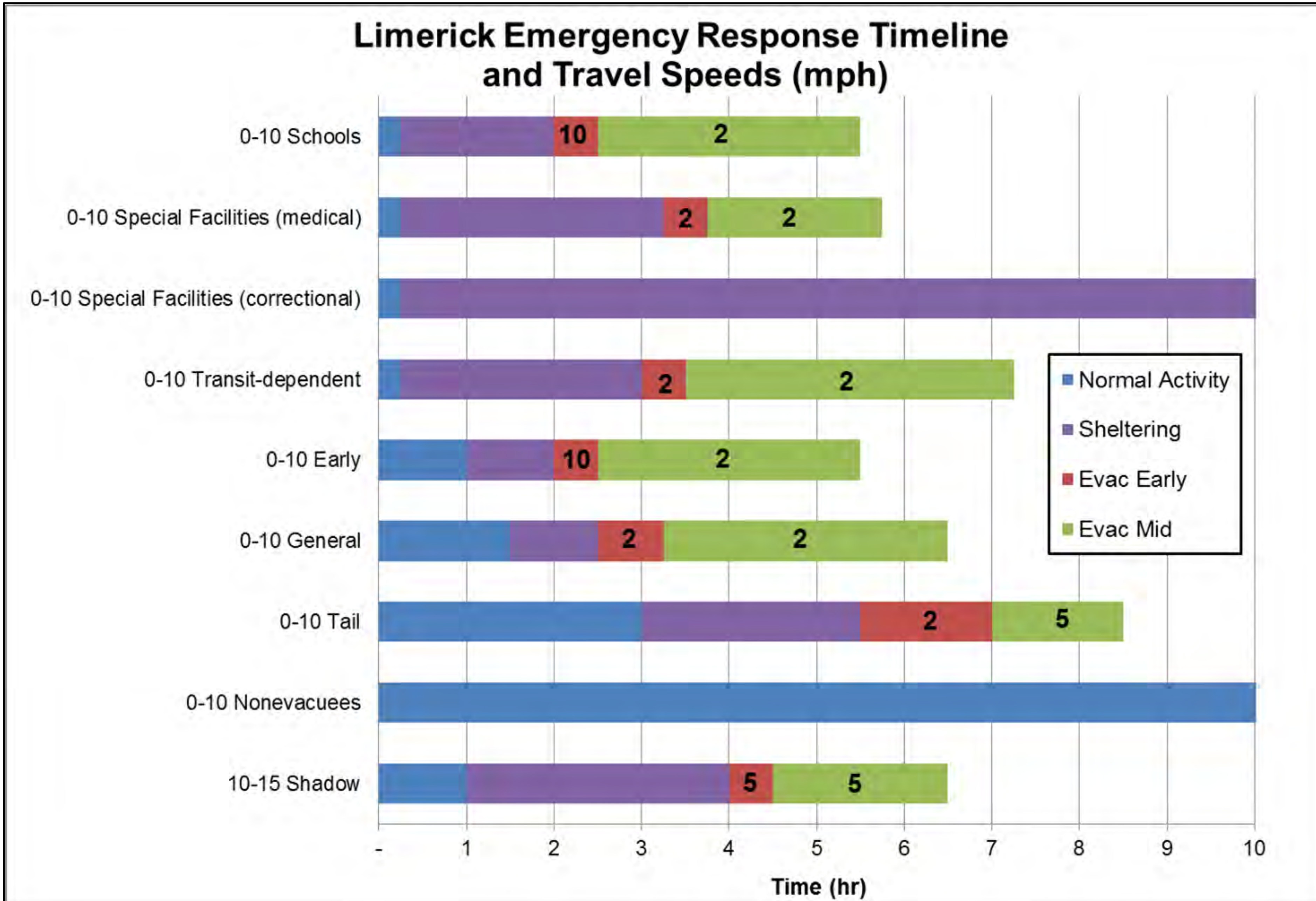




Analysis of Meteorology Data






Limerick Emergency Response Timeline
and Travel Speeds (mph)
0-10 Schools
0-10 Special Facilities (medical)
0-10 Special Facilities (correctional)
0-10 Transit-dependent
0-10 Early
0-10 General
0-10 Tail
0-10 Nonevacuees
10-15 Shadow
10
2
2
2
2
2
10
2
2
2
2
5
5
5
Normal Activity
Sheltering
Evac Early
Evac Mid
-
1
2
3
4
5
6
7
8
9
10
Time (hr)

Potential External Filter Effectiveness for Three Example Cases

CPRR Alternatives	MELCOR Mark I Case and External Filter DF	Percent of Source Term Released Through Vented Pathway		Total Source Term Released to Environment		MACCS Source Term Bin	Description of External Filter Effectiveness
		Cesium	Iodine	Cesium	Iodine		
Status Quo (No External Water Addition) resulting in drywell liner melt-through	1	78.2%	85.5%	1.93%	22.70%	12	External filter has a notable effect on reducing environmental release for DF=10 but smaller incremental benefit for higher DF
	1DF10			0.57%	5.24%	7	
	1DF100			0.44%	3.49%	7	
	1DF1000			0.42%	3.32%	7	
Status Quo (No External Water Addition) resulting in main steam line creep rupture	3	11.5%	21.6%	9.88%	30.20%	17	External filter has an insignificant effect on reducing environmental release
	3DF10			8.85%	24.32%	17	
	3DF100			8.75%	23.74%	17	
	3DF1000			8.74%	23.68%	17	
External Water Addition Successful	10	100.0%	100.0%	0.72%	8.04%	7	External filter reduces environmental release
	10DF10			0.07%	0.80%	5	
	10DF100			0.007%	0.08%	3	
	10DF1000			0.0007%	0.008%	1	

* Among the many MELCOR cases resulting in DW LMT, on average about half the cesium is released through a vent pathway and half escapes through other non-vented pathways.

Mark I Baseline Results by Source Term Bin

Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Individiual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)	
						0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	28DF1000	0.0006%	0.006%	14.9	7	0	4.65E-07	4.57E-08	2.06E-08	1,620	2,380
2	48DF100	0.002%	0.02%	11.4	8	0	1.90E-06	1.90E-07	8.69E-08	5,480	8,260
3	10DF100	0.01%	0.08%	16.3	6	0	6.25E-06	7.16E-07	3.21E-07	16,500	27,300
4	7DF1000	0.02%	0.26%	14.9	20	0	1.72E-05	2.35E-06	1.01E-06	48,400	77,600
5	11DF10	0.06%	0.78%	14.4	4	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
6	48	0.23%	1.69%	11.4	8	0	7.95E-05	1.61E-05	7.79E-06	253,000	450,000
7	15	0.60%	5.85%	14.9	7	0	1.21E-04	3.28E-05	1.64E-05	524,000	932,000
8	46	0.98%	11.01%	14.8	17	0	1.53E-04	4.59E-05	2.34E-05	790,000	1,410,000
9	5DF10	1.05%	2.89%	24.2	34	0	3.55E-04	7.50E-05	3.35E-05	1,040,000	1,720,000
10	5	1.39%	6.46%	24.2	41	0	4.06E-04	9.78E-05	4.51E-05	1,360,000	2,290,000
11	8	1.49%	19.25%	14.9	5	0	1.35E-04	6.41E-05	3.43E-05	1,110,000	2,030,000
12	1	1.93%	22.68%	14.9	22	0	2.91E-04	1.01E-04	5.23E-05	1,720,000	3,090,000
13	41DF1000	3.40%	7.65%	9.8	17	0	5.22E-04	1.49E-04	7.89E-05	1,900,000	3,610,000
14	22dw	2.82%	18.64%	14.9	27	0	4.27E-04	1.28E-04	6.57E-05	1,830,000	3,320,000
15	53	2.79%	29.05%	17.4	13	0	2.59E-04	1.19E-04	6.96E-05	1,740,000	3,520,000
16	41	4.54%	14.10%	9.8	16	0	5.57E-04	1.75E-04	9.82E-05	2,300,000	4,520,000
17	3DF10	8.85%	24.65%	9.8	63	0	7.10E-04	2.95E-04	1.68E-04	3,830,000	7,720,000
18	52	15.90%	34.32%	17.4	11	0	5.39E-04	2.23E-04	1.50E-04	3,080,000	6,870,000

Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
						0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	28DF1000	0.0006%	0.006%	14.9	7	78,900,000	78,900,000	0	0	-	-
2	48DF100	0.002%	0.02%	11.4	8	79,700,000	79,700,000	1	1	0	0
3	10DF100	0.01%	0.08%	16.3	6	98,100,000	98,700,000	10	11	1	1
4	7DF1000	0.02%	0.26%	14.9	20	141,000,000	141,000,000	23	23	7	7
5	11DF10	0.06%	0.78%	14.4	4	220,000,000	240,000,000	41	65	118	118
6	48	0.23%	1.69%	11.4	8	1,150,000,000	1,390,000,000	116	175	3,440	3,440
7	15	0.60%	5.85%	14.9	7	2,740,000,000	3,690,000,000	190	361	15,000	16,600
8	46	0.98%	11.01%	14.8	17	3,760,000,000	5,220,000,000	242	506	20,700	27,400
9	5DF10	1.05%	2.89%	24.2	34	7,290,000,000	8,600,000,000	351	429	35,200	35,200
10	5	1.39%	6.46%	24.2	41	9,900,000,000	12,000,000,000	479	715	51,400	51,500
11	8	1.49%	19.25%	14.9	5	5,960,000,000	9,720,000,000	286	673	40,500	55,800
12	1	1.93%	22.68%	14.9	22	13,000,000,000	17,400,000,000	549	1,040	64,500	79,700
13	41DF1000	3.40%	7.65%	9.8	17	19,400,000,000	24,700,000,000	783	1,170	168,000	190,000
14	22dw	2.82%	18.64%	14.9	27	12,900,000,000	18,300,000,000	544	1,010	93,700	114,000
15	53	2.79%	29.05%	17.4	13	15,700,000,000	26,500,000,000	573	1,290	111,000	142,000
16	41	4.54%	14.10%	9.8	16	25,500,000,000	35,400,000,000	904	1,500	235,000	281,000
17	3DF10	8.85%	24.65%	9.8	63	47,000,000,000	68,100,000,000	1,360	2,470	417,000	504,000
18	52	15.90%	34.32%	17.4	11	46,500,000,000	87,700,000,000	987	2,170	467,000	873,000

Mark II Baseline Results by Source Term Bin

Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)	
						0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	11DF1000	0.00004%	0.0005%	20.3	20	0	9.72E-08	1.03E-08	3.45E-09	282	345
2	5DF1000	0.0006%	0.005%	32.2	20	0	1.15E-06	1.81E-07	6.35E-08	4,340	5,440
3	42DF100	0.0043%	0.037%	14.3	13	0	6.58E-06	8.67E-07	3.02E-07	20,700	26,700
4	11	0.042%	0.45%	20.3	20	0	7.90E-05	9.68E-06	3.27E-06	202,000	261,000
5	51DF10	0.23%	2.01%	16.6	9	0	1.35E-04	3.39E-05	1.21E-05	689,000	888,000
6	5	0.55%	4.94%	32.2	20	0	2.29E-04	1.05E-04	4.01E-05	2,160,000	2,900,000
7	3	1.09%	10.26%	14.3	20	0	3.08E-04	1.88E-04	7.43E-05	4,140,000	5,580,000
8	1	2.46%	19.81%	22.8	25	0	4.70E-04	3.17E-04	1.25E-04	6,110,000	8,260,000
9	52	3.57%	28.67%	16.6	10	0	4.03E-04	2.46E-04	1.01E-04	5,430,000	7,440,000

Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
						0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	11DF1000	0.00004%	0.0005%	20.3	20	381,000,000	381,000,000	-	-	-	-
2	5DF1000	0.0006%	0.005%	32.2	20	381,000,000	381,000,000	0	0	-	-
3	42DF100	0.0043%	0.037%	14.3	13	393,000,000	393,000,000	2	2	0	0
4	11	0.042%	0.45%	20.3	20	844,000,000	846,000,000	44	47	1,030	1,030
5	51DF10	0.23%	2.01%	16.6	9	4,250,000,000	4,380,000,000	130	221	15,400	15,400
6	5	0.55%	4.94%	32.2	20	24,000,000,000	28,000,000,000	303	551	62,400	62,400
7	3	1.09%	10.26%	14.3	20	80,800,000,000	105,400,000,000	698	1,200	619,000	649,000
8	1	2.46%	19.81%	22.8	25	85,500,000,000	109,300,000,000	854	1,680	721,000	741,000
9	52	3.57%	28.67%	16.6	10	53,600,000,000	63,800,000,000	618	1,400	414,000	449,000

Note: # Hrs with Significant Cs Release is the number of hours in which at least 0.5% of the source term's total Cs is being released.

Population Information

Mark I	Site	Population within 50 mi	Population Density (per sq mi)	Rank
	Dresden	7,374,320	939	1
HIGH*	**Peach Bottom**	**5,645,811**	**719**	**2**
	Hope Creek	5,633,411	717	3
	Pilgrim	4,851,642	618	4
	Fermi	4,808,370	612	5
	Oyster Creek	4,567,689	582	6
	Monticello	3,052,698	389	7
MEDIUM*	**Vermont Yankee**	**1,536,793**	**196**	**8**
	Browns Ferry	997,194	127	9
	Fitzpatrick & Nine Mile Point	923,614	118	10
	Duane Arnold	673,752	86	11
	Quad Cities	653,780	83	12
	Brunswick	479,743	61	13
LOW*	**Hatch**	**453,404**	**58**	**14**
	Cooper	159,946	20	15

Mark II	Site	Population within 50 mi	Population Density (per sq mi)	Rank
HIGH*	**Limerick**	**8,108,436**	**1,032**	**1**
	LaSalle	1,909,500	243	2
MEDIUM*	**Susquehanna**	**1,790,924**	**228**	**3**
	Nine Mile Point	923,614	118	4
LOW*	**Columbia**	**464,310**	**59**	**5**

* Site selected for population sensitivity calculations

Containment Protection and Release Reduction Rulemaking Regulatory Evaluation

ACRS Fukushima Subcommittee

July 7, 2015

BWR OWNERS' GROUP

NEI
NUCLEAR ENERGY INSTITUTE

SECY-15-0085

- Excellent work in response to SRM-SECY-12-0157
- Overall results consistent with industry's
- Regulatory options
  - Can achieve alignment with Staff on option for going forward
  - Concerned with potential for scope creep when order is written into a rule
- Look forward to engaging with NRC Staff

Industry Technical Evaluations Supporting CPRR

Stuart Lewis (EPRI): EPRI Program Manager

Rick Wachowiak (EPRI): EPRI Project Manager

Jeff Gabor (ERIN): Investigator

Doug True (ERIN): Investigator

Advisory Committee on Reactor Safeguards
July 7, 2015

CPRR Rulemaking

- Evaluation of the residual benefits of filtering strategies should be made in the context of an effective accident management capability and focused on the dominant accident scenarios
- Industry has always viewed the CPRR Rulemaking in the context of accident management
  - Response to postulated severe accidents like the accidents at Fukushima requires operator action
- Accident management involves:
  - Cooling core debris
  - Managing decay heat
  - Mitigating releases

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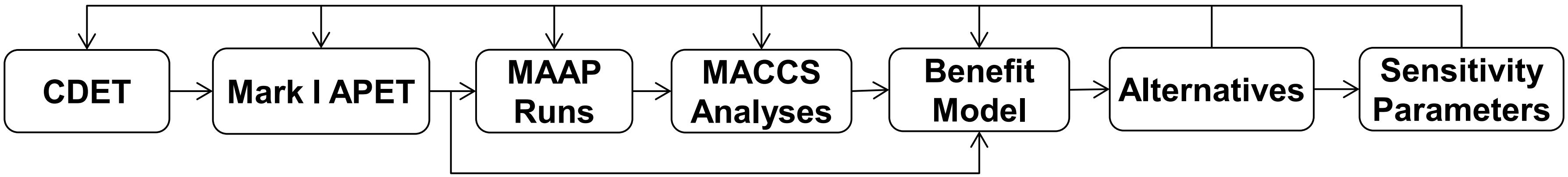
Objectives of EPRI Evaluation

- Understand the role FLEX plays in ELAP mitigation
- Understand dominant severe accident scenarios
- Develop clear, manageable analysis of filtering strategy alternatives
- Support open dialog with NRC staff on assumptions, technical issues, dominant scenarios, and insights
- Inform the implementation of EA 13-109 (to the extent feasible)
- Providing insights to BWROG on EPG/SAGs
- Support industry decision-makers on the cost-benefit considerations

Technical Basis for Severe Accident Mitigating Strategies: Volume 1. EPRI, Palo Alto, CA: 2015. 3002003301.

EPRI Analysis Framework



- CDET → 13 core damage scenarios
- APET → 39 release scenarios (per CDET sequence)
- Total = 507 release scenarios
- Specific MAAP & WinMACCS runs for each 507
- All done 24 times to address the alternatives

Summary of Alternative Cases

Alternative	SAC WW Vent	DW Vent	SAWA	SAWM	Vent Control	Filter	Filter Path
New Base	Yes	None	No	No	No	None	Manual
1A	Yes	SACV	No	No	No	None	Manual
2A	Yes	None	RPV	No	No	None	Manual
2B	Yes	None	RPV	**Yes**	No	None	Manual
2C	Yes	None	RPV	No	**Yes**	None	Manual
2D	Yes	None	RPV	**Yes**	**Yes**	None	Manual
2E	Yes	SACV	RPV	No	No	None	Manual
2F	Yes	SACV	RPV	**Yes**	No	None	Manual
2G	Yes	SACV	RPV	No	**Yes**	None	Manual
2H	Yes	SACV	RPV	**Yes**	**Yes**	None	Manual
3A	Yes	None	DW	No	No	None	Manual
3B	Yes	None	DW	**Yes**	No	None	Manual
3C	Yes	None	DW	No	**Yes**	None	Manual
3D	Yes	None	DW	**Yes**	**Yes**	None	Manual
3E	Yes	SACV	DW	No	No	None	Manual
3F	Yes	SACV	DW	**Yes**	No	None	Manual
3G	Yes	SACV	DW	No	**Yes**	None	Manual
3H	Yes	SACV	DW	**Yes**	**Yes**	None	Manual
4A	Yes	SACV	RPV	No	No	**Small**	Manual
4B	Yes	SACV	DW	No	No	**Small**	Manual
5A	Yes	SACV	RPV	No	No	**Large**	Manual
5B	Yes	SACV	DW	No	No	**Large**	Manual
6A	Yes	SACV	DW	No	No	**Large**	**All Passive**
6B	Yes	SACV	DW	No	No	**Large**	**Manual Pre-CD**

Legend
No SAWA
RPV SAWA
DW SAWA

Considerations in Assessing Alternatives

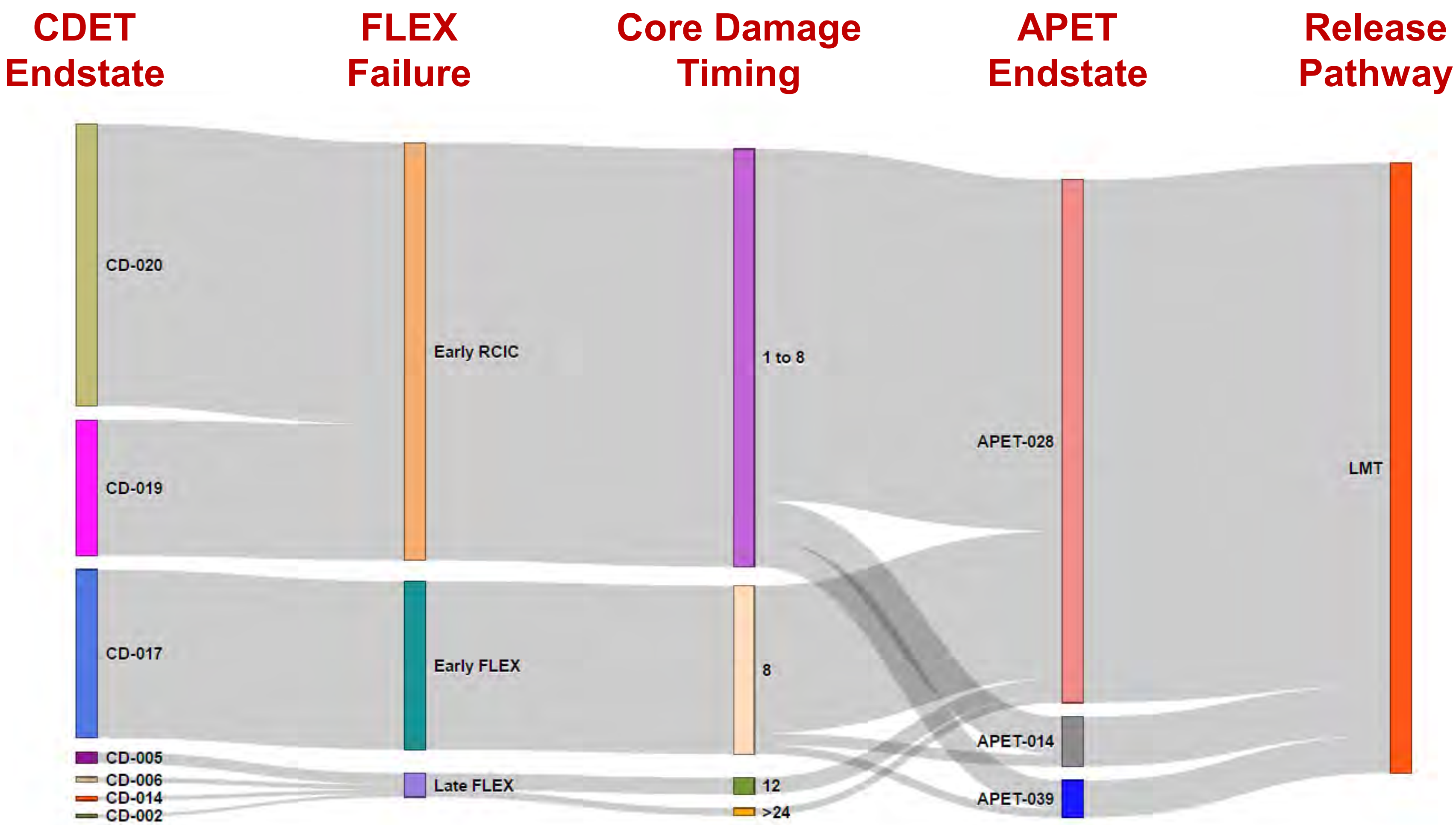
Risk Metrics

- Core Damage Frequency (CDF)
- Conditional Containment Failure Probability (CCFP)
- Latent Cancer Fatality (LCF) Risk
- Financial Consequence Risk

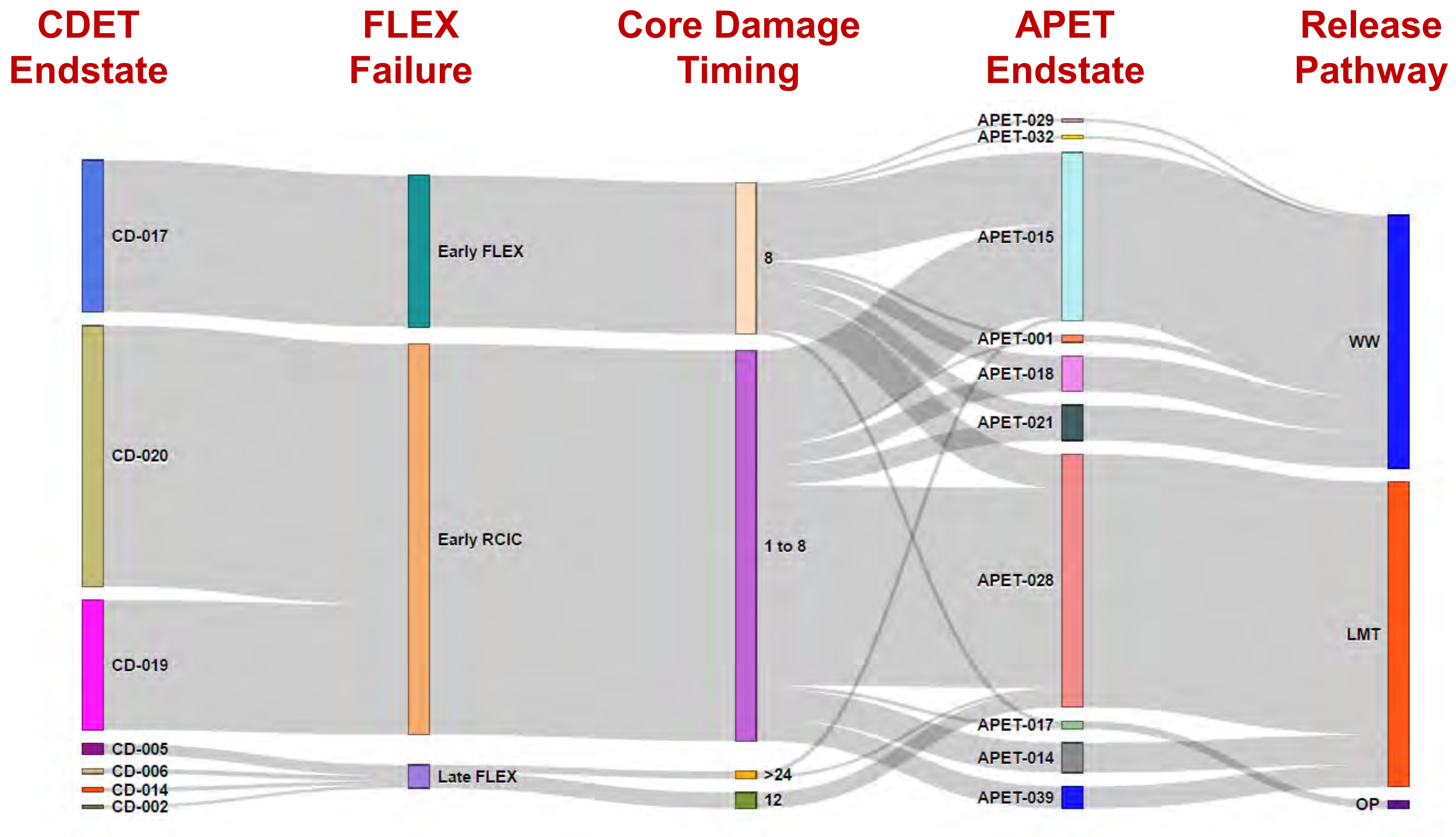
Other Considerations

- Defense-in-depth
- Containment temperature control
- Reliance on human actions
- Instrumentation requirements
- Release reduction
- Hydrogen control

Baseline Results Visualization

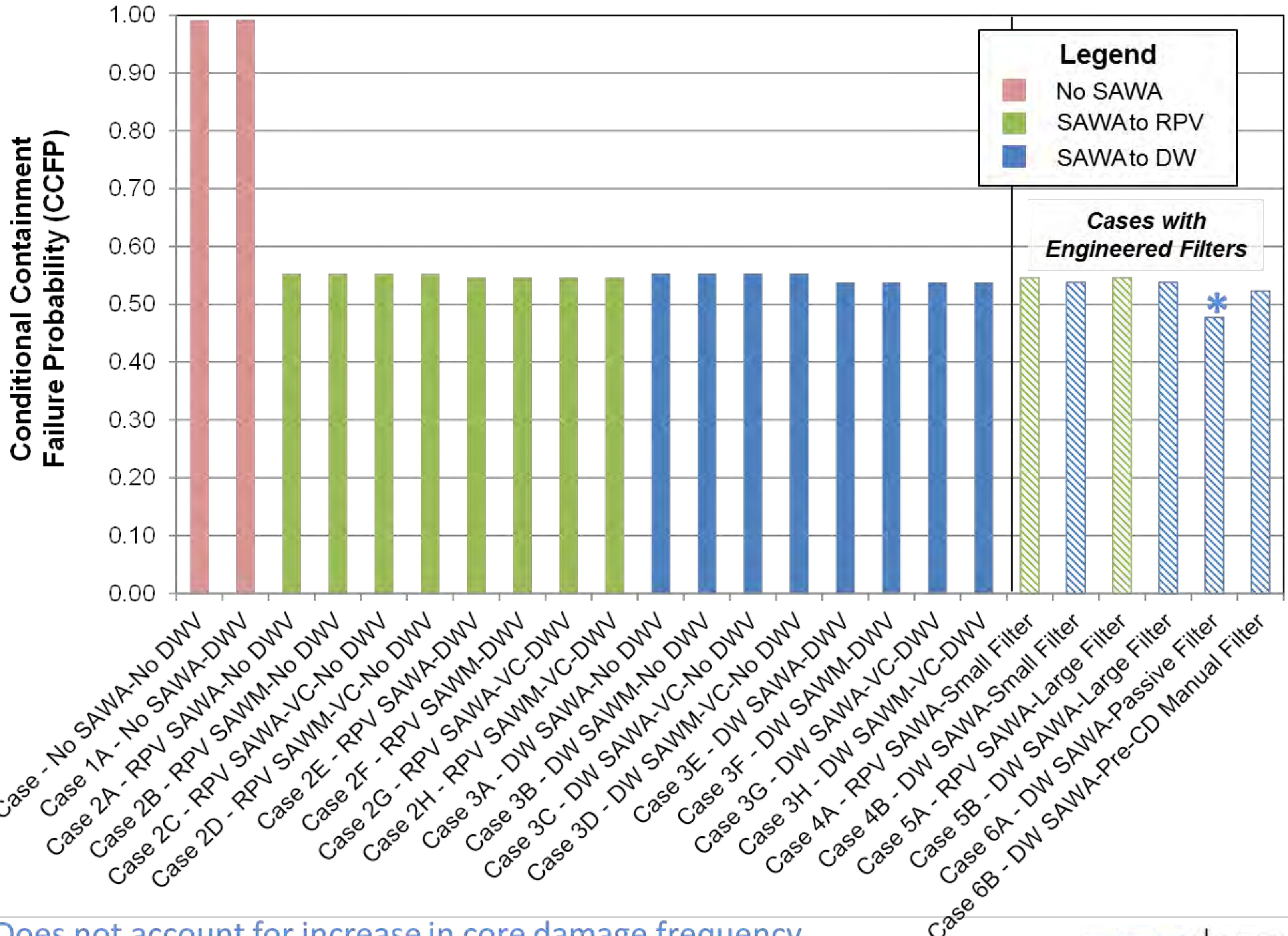


Cases with Water Addition Results Visualization



High-level Results for Mark I Containments

Conditional Containment Failure Probability



* - Does not account for increase in core damage frequency

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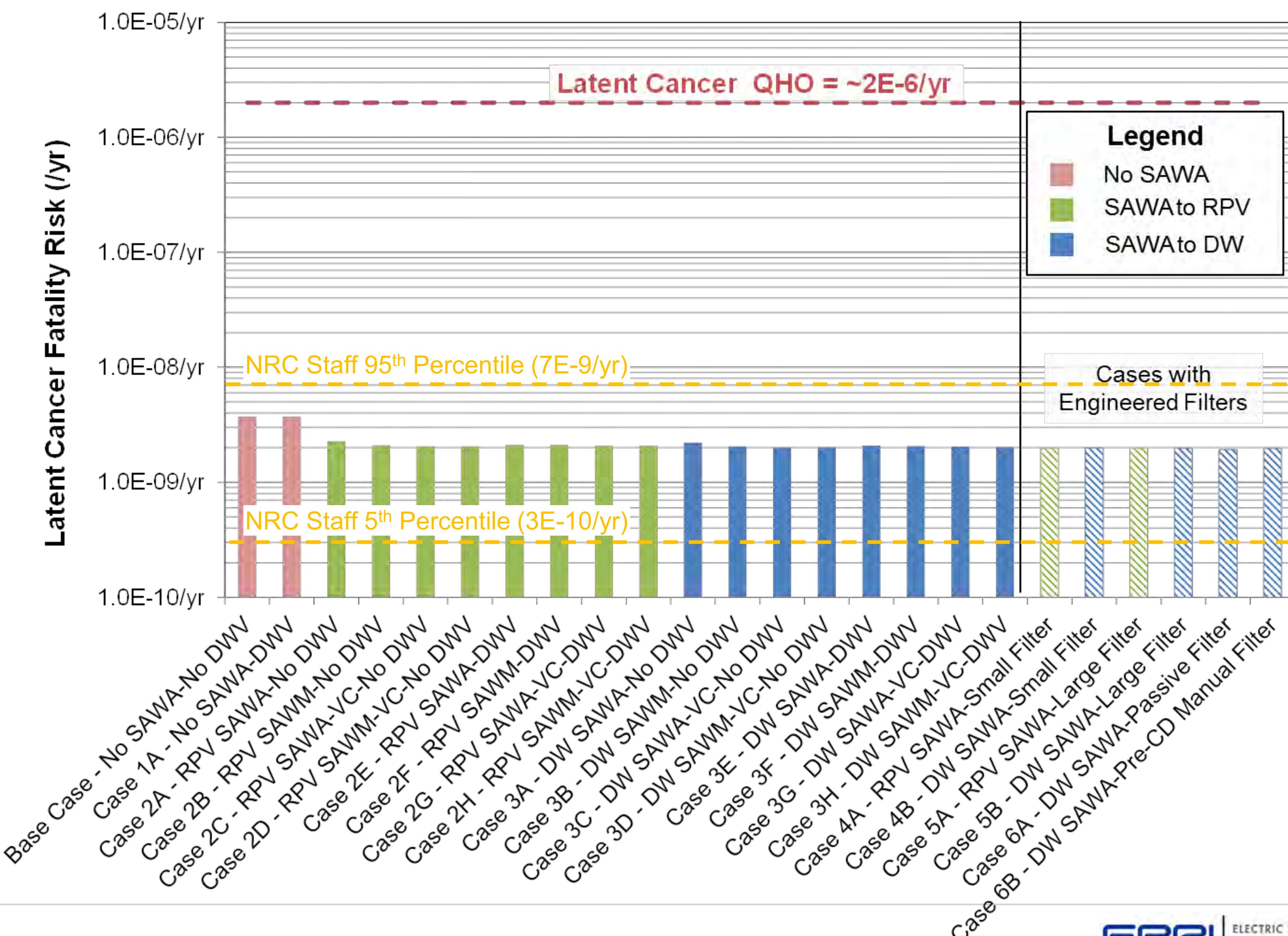
Latent Cancer Fatality Risk



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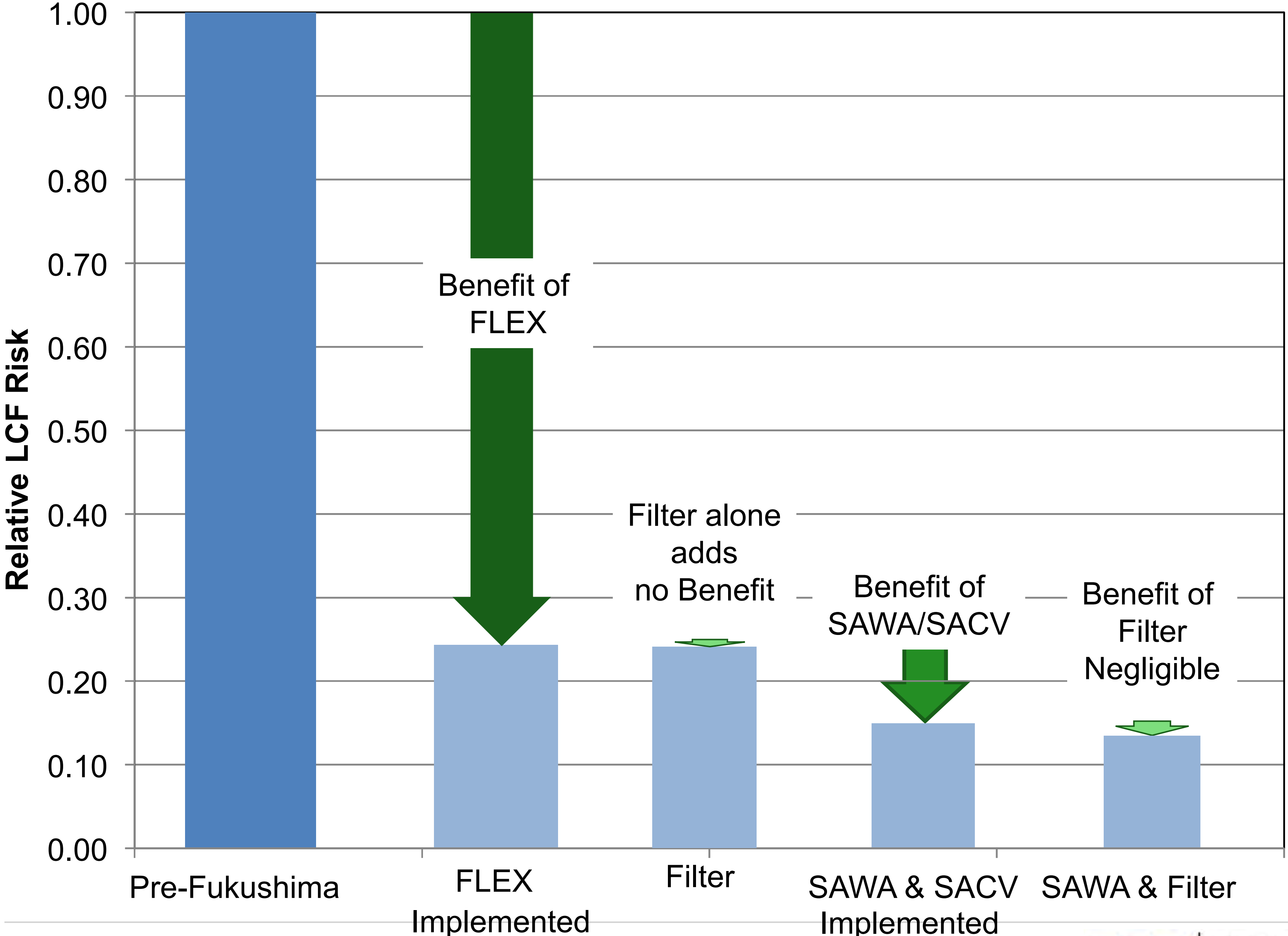
Latent Cancer Fatality Benefits



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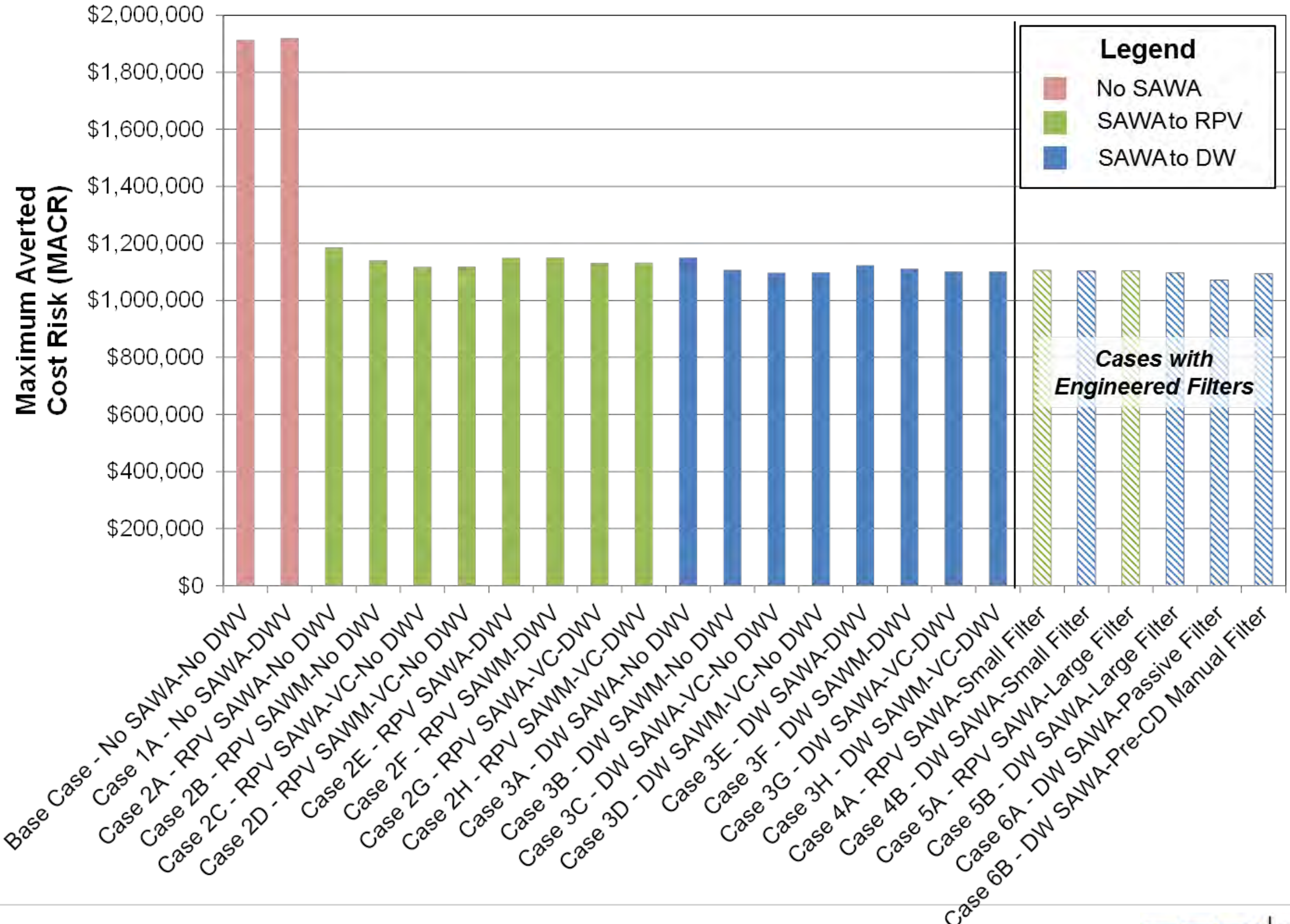
Financial Consequence Risk



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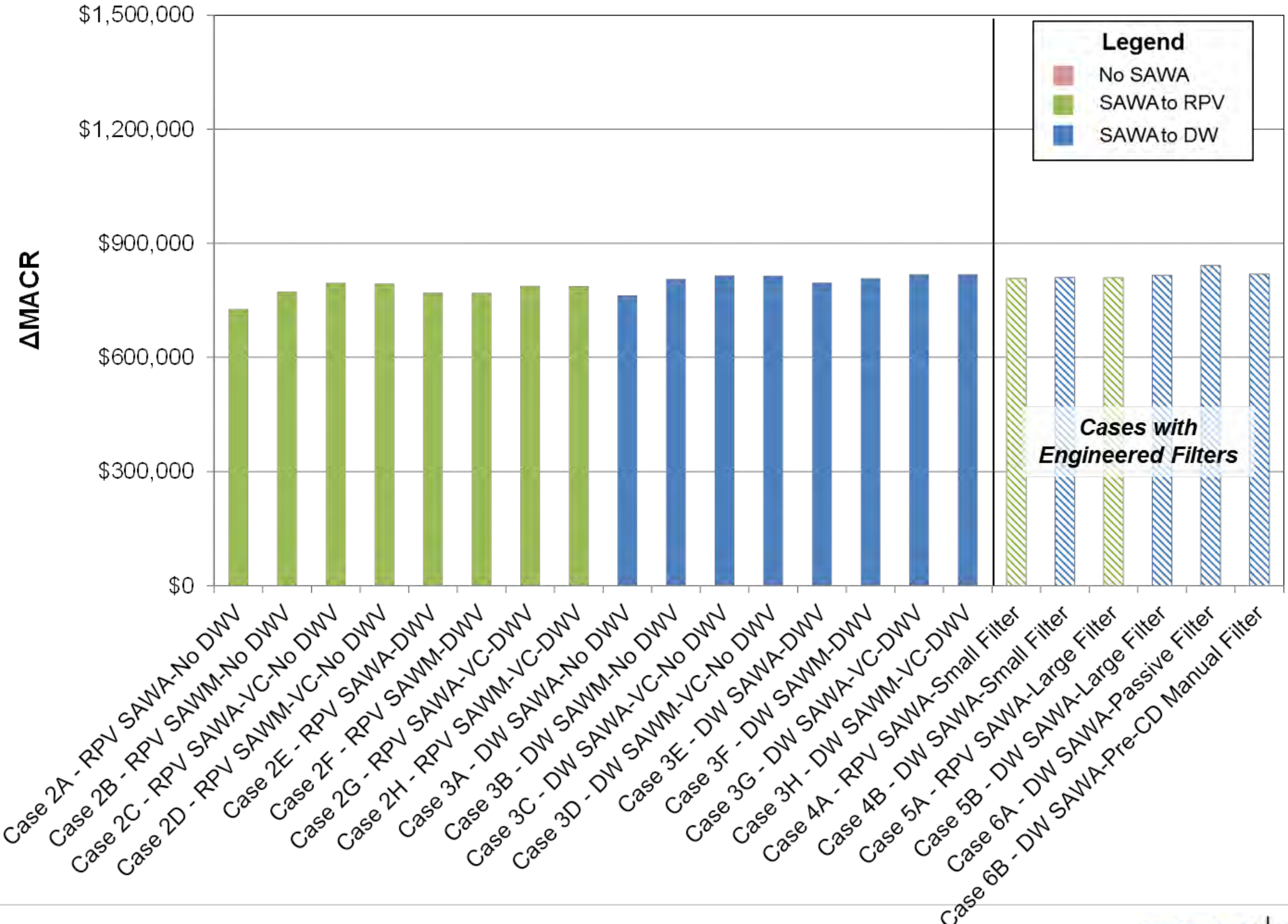
Change in Financial Consequence



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Sensitivity Cases

Plant to Plant Variability

- Containment Heat Capacity
- Torus Freeboard Volume
- DW to WW Spillover Height
- Population
- Probabilistic Logic Model
- ELAP frequency
- Human error rates for water addition

Phenomenological

- SRV Seizure
- In-Vessel Retention
- LMT Timing

Benefit Model

- Deposition Rate
- Evacuation Effectiveness
- $/Person-rem
- Discount Rate
- Combination – Discount Rate & $/Person-rem

Plant to Plant Variability

Sensitivity Parameter	Approach	Alternatives	Metrics	Conclusions
Containment Heat Capacity	Qualitative	All	N/A	• Reference plant has most limiting containment heat capacity of US fleet
Torus Freeboard Volume	Qualitative	All	N/A	• Reference plant is one of the most limiting with respect to torus freeboard volume. • Ample time exists for operating staff and emergency response organization to implement SAWM.
DW to WW Spillover Height	Qualitative	All	N/A	• Water addition is the most significant factor in providing debris cooling and controlling drywell temperatures.
Population	Quantitative	2A	LCF	• Reference plant represents the second largest population site (50 mi). Sensitivity to address largest population site reveals no significant impact.

Sensitivity to Consequence/Benefit Inputs

Sensitivity Parameter	Approach	Alternatives	Metrics	Conclusions
Deposition Rate	Quantitative	Base, 2A, 3A, 5B	LCF, MACR, ΔMACR	• Adopting the aerosol deposition rate from NUREG-1150 increases the economic risk in the Base Case by over 20%, but has only a very small impact (<10%) on the alternatives that provide severe accident water addition.
Evacuation Effectiveness	Quantitative	Base, 2A, 3A, 5B	LCF, MACR, ΔMACR	• Computed risks are relatively insensitive to evacuation effectiveness from 95% to 100%. • Cases with no evacuation showed much higher latent cancer risks than cases with relatively effective evacuation, but negligible impact on overall financial risks.
$/Person-rem	Quantitative	Base, 2A, 3A, 5B	MACR, ΔMACR	• Changing the value of a person-rem from $2,000 to $5,200 increases the overall financial consequences. However, this change only translates to a ~30% increase in the financial risks.
Discount Rate	Quantitative	Base, 2A, 3A, 5B	MACR, ΔMACR	• A bounding assumption of no discount on the present value of property (i.e., no property depreciation) increased the overall financial results by approximately a factor of 2.
Combination – Discount Rate & $/Person-rem	Quantitative	Base, 2A, 3A, 5B	MACR, ΔMACR	• The combined impact of a lower discount rate (3%) and a higher value of a person-rem ($5,200) increases the overall financial risks by approximately a factor of 2.

Insights

- Essential role of the operators
- Importance of water addition
- Incremental benefit of engineered filters
- Totally passive vent shown to increase CDF
- Sensitivity cases confirmed that the margins identified in the base results are not challenged by uncertainties

Conclusions

- Adoption of severe accident water addition strategies provides the greatest overall safety benefit, both in terms of protecting containment and reducing releases
- Manual actions would be required to manage the severe accident for all strategies
- Other alternatives, including installation of engineered filters, provide negligible additional benefit to public health and safety

Together…Shaping the Future of Electricity

Wang, Weidong

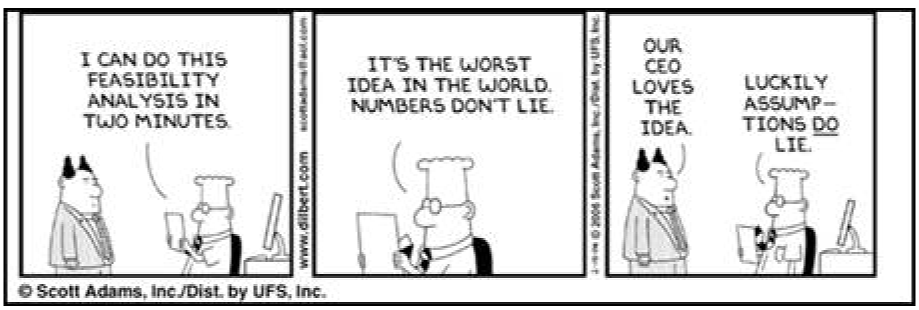
From: Mary Lampert <mary.lampert@comcast.net>
Sent: Monday, July 06, 2015 12:45 PM
To: Wang, Weidong
Subject: [External_Sender] ACRS Meeting Vents July 7
Attachments: lochbaum NRC PRESNTATION PILGRIM-20140731.pdf; EA.12-049 MOORING PLAN IN INTAKE-PW COMMENT.pdf

To the Members of the ACRS:

Regarding Filters for the DTV:

The Staff's conclusion to recommend Option 3 (SAWA/SAWM) and not Option 4 (addition of filters for the vent) rests on flawed assumptions and use of outdated consequence codes- MELCOR, MACCS, SOARCA.

The numbers in the NRC staff's consequence analysis may add up correctly but the answer is wrong because the assumptions underlying the Staff's PRA are incorrect.



The weaknesses of the consequence codes will be discussed in a later paper. I wanted to draw your attention before tomorrow's meeting to Staff's Draft Regulatory Basis for Containment Protection and Release Reduction for Mark I and Mark II Boiling Water Reactors (10 CFR Part 50), May 2015 assumption that SAWA is successful 6 out 10 times. (at 105)

SAWA plans are site specific. Pilgrim Watch showed that Pilgrim's Waterways Project to add supplemental water is unlikely to work, attached. In addition, David Lochbaum (Union of Concerned Scientist) paper on FLEX used Pilgrim Station as one example to show the weakness in the FLEX program, attached.

Therefore the assumption that SAWA will be successful 6 out of 10 times in not tenable increasing the probability of offsite releases and consequent offsite costs and latent cancer fatality risk -and other health consequences ignored by the Staff. If Staff used a realistic analysis of SAWA, it would indeed support Option 4. Further if Staff did not persist in using flawed consequence codes, the offsite consequences would unquestionably justify filters.

In fact, Staff's own May 15 flawed analysis shows that adding a filter (Option 4) provides the most bang for the buck.

Staff estimated a filter costs between $11 and $64 million dollars (May 15 draft, pg., 27) and Table 4.23 (May 15 draft, pg., 91) shows that by adding a filter along with SAWA (that industry is committed to installing for $3 million anyway) will save $4.51 billion. A filter even at $64 million saves $3 and one-half billion. Not bad. With an honest analysis using an updated PRA, think what the saving actually would be.

Thank you,

Mary Lampert
Pilgrim Watch
148 Washington Street
Duxbury, MA 02332
Tel. 781.934-0389

Waterways Application, No. W14-4147
Cape Cod Bay, Plymouth, Plymouth County
Ch 91 Application of Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station

PILGRIM WATCH PUBLIC COMMENT (JULY 17, 2014)

Pilgrim Watch (Hereinafter "PW") is a grassroots organization that serves the public interest in issues regarding the Pilgrim Nuclear Power Station in Plymouth. We are located in Duxbury Massachusetts, a community directly across the bay from the Pilgrim Nuclear Power Station and within Pilgrim's 10-mile Emergency Planning Zone.

PW finds, contrary to Entergy's contractor, *Beals and Thomas*, (Project Narrative, 2.1.1) that the project serves no public purpose; in fact it is harmful to the public because once this "Rube Goldberg" designed project is installed, it will mean that Entergy will never install mitigation that would actually work and mitigate during a beyond design basis external event.

The application simply describes the bare bones of project; it fails to include information and analysis required for DEP to properly review and subsequently deny the application.

Project Narrative

Beals and Thomas describe <u>the Outhaul & Mooring System Project</u> in its Project Narrative, 2.0 (May 9, 2014). It says the following:

2.1 Introduction: The project is being undertaken to comply with NRC Order EA-12-049 that requires additional mitigation requirements in the event of a beyond-design basis external event.

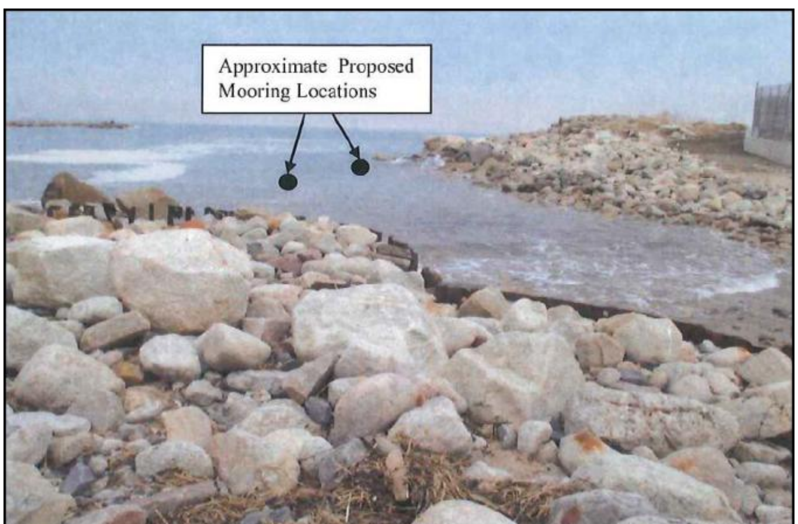
<u>PW: The narrative fails to describe what external events</u> it is designed to protect. Entergy describes the external events in its *Pilgrim Nuclear Power Station Overall Integrated Plan For Diverse And Flexible Coping Strategies (Flex) For Requirements For Mitigation Strategies For Beyond-Design-Basis External Events*[1](February 2013) that *Beals and Thomas* fail to cite. The events include: seismic, high wind, snow, ice and extreme cold, and extreme high temperature. (Entergy, 1) NRC required the licensee to update the risk from these events; they have not completed the review. Therefore the project is designed to mitigate events not properly analyzed.

[1] NRC Electronic Library, Adams, Accession Number ML13225A587

We know that climate change will impact Pilgrim Station with an increased risk for flooding, and more frequent and severe storms so that any project must take the projections into consideration. Also NRC ranked Pilgrim as second among all the nation's nuclear reactors for seismic risk. Importantly, absent from the analysis for external events are acts of malice.

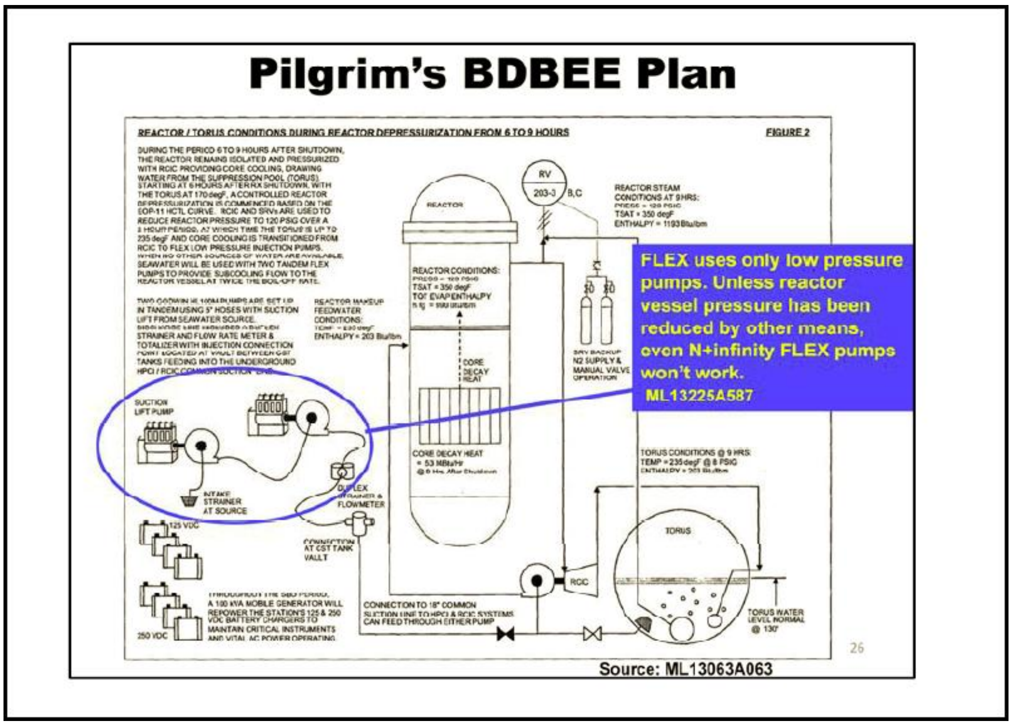
2.3.1 Description Work

- In order to deploy the intake pipes of portable emergency pumping equipment, an outhaul and mooring system will be constructed within the existing barge landing. Two Eco-Mooring Systems will be installed within the intake embayment just beyond the mean low water line. Two helical pile moorings will be driven into the sand by an auger.
- Consistent with a typical mooring, a float and Grainger snatch block pulley will be anchored to each mooring with a line to the surface, and connected to the shore via the outhaul system.
- The outhaul system consists of a snatch block pulley mounted with beam brackets on the foundation wall of the outer security fence at the barge landing area, connecting to the floating pulleys with anchor line.
- The mooring and outhaul system are designed to withstand 12,000 lbs of loading, which is well beyond forces exerted by wind and wave action.
- In the event of an emergency that requires a significant heat sink, two floating strainers connected to semi-rigid suction pipe will be deployed with the outhaul system, and anchored to the mooring.
- The suction pipe will then be connected to a centrifugal pump temporarily deployed by a truck at the Mean High Water Line, which will feed into the 6" stainless steel buried pipe, providing cooling water to the facility.



Proposed mooring locations/barge landing area looking east

Barge landing area looking south



2.1.1 Proper Public Purpose – Pilgrim Watch Dispute: Project serves no public purpose; it has a very high probability of failure

Beals and Thomas find that the project serves a proper public purpose, "as it is an improvement within the intake embayment required emergency cooling water withdrawal associated with an existing energy facility."

PW: *Beals and Thomas* fail to properly analyze the project in terms of its probability for failure, especially in adverse weather conditions when an accident requiring emergency cooling is most likely needed and at the same time weather conditions are likely to impede both workers and equipment from getting to location to perform the tasks required.

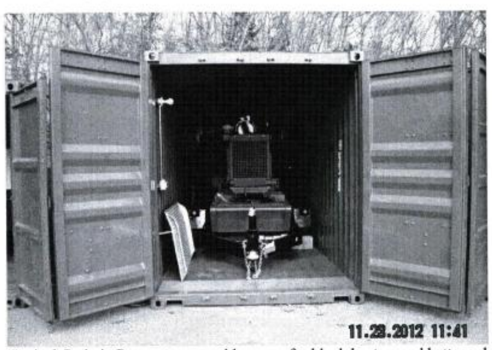
Examples:

1. **Truck/Trailer/Pump**: The project depends on a truck bringing a pump and we presume flexible hose to the Barge landing area. <u>What can go wrong?</u>

The truck, trailer, and pump are housed in storage shed(s). From Entergy's photographs below they look like they are out of the 1920'3 and not robust to withstand extreme conditions.

February 2013 FLEX Integrated Plan

During receipt period, generators mounted on trailers, hose racks since installed.



Typical Godwin Pump storage, with power for block heaters and battery charging.

Entergy, 83

Entergy says in the Integrated Flex Plan that, "The single debris removal equipment identified may not be able to move debris to enable transport of equipment within the 6-9 hour time restriction.[2]" Therefore there is no certainty the trucks will get out of the sheds in, any extreme

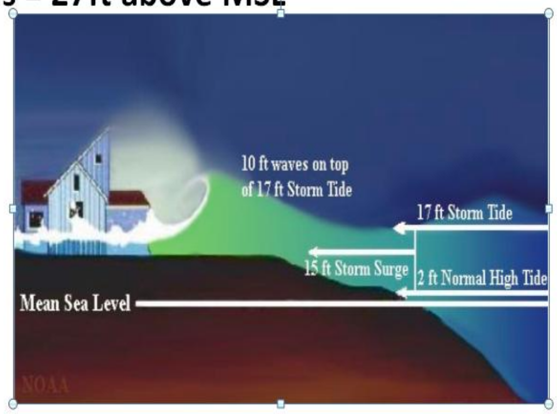
[2] NRC Electronic Library, Adams, Accession No. ML13225A587

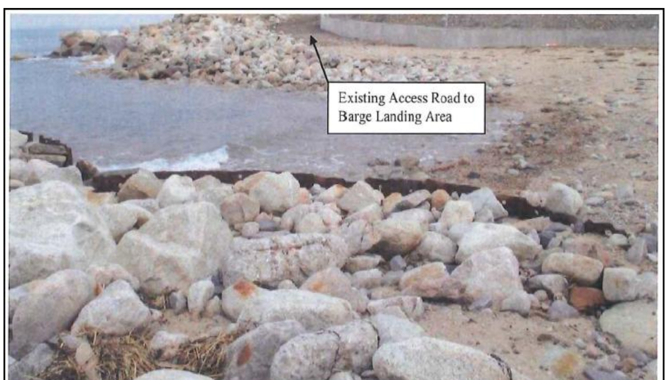
natural event that it is designed to operate. Entergy admits at 79 that none of its FLEX[3] equipment is designed to operate in a tornado.

Equipment heaters protect FLEX equipment from cold weather damage before the beyond-design-basis event. NRC Bulletin 79-24 discussed events at nuclear plants where safety-related systems were disabled by cold weather. Therefore the question raised is will that equipment housed in the shed be damaged before the need arises? There is no regulatory requirement to monitor the storage shed heaters or to fix them within some timeframe if Entergy happens to notice that they are broken. Monitored and tested equipment has been disabled by cold weather, so who can assume that unmonitored and untested equipment will work?

Getting to Location: Now assume that the vehicle, trailer and pump get out of the shed, the plan non-conservatively assumes that it can get to the Barge Landing Area-despite the possibility of high waves on top of a storm surge at high tide or over deep snow and ice.

15 ft surge on 2 ft tide= 17ft storm tide + 10 ft waves = 27ft above MSL





Barge landing area looking south

[3] FLEX stands for Diverse and Flexible Mitigation Equipment

We can picture both the equipment and workers swept into the intake canal; if in fact they arrive at the landing area in the first place considering NRC's Technical Evaluation Report [4] that says, "The single debris removal equipment identified may not be able to move debris to enable transport of equipment."

2. Snatch Block Pulley: The outhaul system consists of a snatch block pulley mounted with beam brackets on the foundation wall of the outer security fence at the barge landing area, connecting to the floating pulleys with anchor line.

PW gives it high probability of snagging from seaweed and debris caught on the lines,

3. Floating Strainer: The plan calls for two floating strainers to be connected to a semi-rigid suction pipe that will be deployed with the outhaul system, and anchored to the mooring.

PW: Mr. William Mauer called the manufacturer of the floating strainer on 7/16. He asked them how much wave action this piece of equipment was designed to function in. The response was that this and all of their floating strainers are designed to work in <u>still water</u> ponds and lakes....they are not designed or evaluated for use in water bodies with any kind of wave action. The key word is still water an unlikely occurrence during an extreme external event that the project is designed to handle.

4. Centrifugal Pump: The plan says that the suction pipe will then be connected to a centrifugal pump temporarily deployed by a truck at the Mean High Water Line, which will feed into the 6" stainless steel buried pipe, providing cooling water to the facility.

PW: The plan fails to say that it is a low pressure pump. In order for the pump to provide cooling water to the facility, the reactor pressure must be reduced for the pumps to work. The plan is non-conservative to assume that the reactor pressure vessel pressure gets lowered enough to let the FLEX's low pressure pump to provide makeup flow.

5. Testing: There is no mention of testing the project that should occur during a severe nor'easter in the dead of winter with a lot of snow on the ground. As we know from years of experience living here that is when the plant meets its greatest challenge.

Conclusion

Pilgrim Watch has shown that the project serves no public purpose. For example: The storage shed(s) are vulnerable to common-mode losses (debris, heater malfunction); the Barge Landing area is vulnerable to storm action risking that the equipment and workers will not arrive there or if they do will end up in the intake canal; the pulley system is vulnerable to jams; the strainer is designed for still water; the pump operate only at low pressure non-conservatively depending on

[4] NRC Electronic Library, Adams, Accession No. ML13225A587

the reactor reaching low pressure in a timely manner; the system mentions no plan to test it in an external events that the system is supposed to mitigate. Therefore absent serving a public purpose, we believe that DEP should deny the application.

Thank you for your consideration.

Mary Lampert
Pilgrim Watch, Director
148 Washington Street, Duxbury, MA 02332
July17, 2014

Union of Concerned Scientists

Status of Fukushima Lessons

July 31, 2014

David Lochbaum
Director, Nuclear Safety Project
Union of Concerned Scientists
www.ucsusa.org

Status to date

Good*

My focus today will be on the mitigating strategies order. Many themes are applicable to other Fukushima lessons.

On the Good Side

Station blackout rule assumed that alternating current power would be restored within the plant-specific coping duration (typically 4 or 8 hours)

Mitigating strategies order seeks to provide core, containment and spent fuel cooling for an infinite period.

On the Caveat Side

Original assumption that Fukushima invalidated has been replaced by the assumption that FLEX equipment can be placed and operated in time.

Is this assumption also invalid?

On the Good Side

February 2013 FLEX Integrated Plan

Equipment / Event (Note 1)	Qty	Seismic	External Floods	Hurricane / Tornado	Snow, Ice, Cold	High Temps
Duplex Strainer Trailer 400 GPM 1/8 Inch Size Req'd N = 1						
FLEX-North	1	1	1	1 or 0	1	1
FLEX-South	1	1	1	1 or 0	1	1
Total Available	N+1	N+1	N+1	N	N+1	N+1
Resin Demin Skid 60 cu-ft Mixed Bed Req'd N = 2						
FLEX-North	2	2	2	2 or 0	2	2
FLEX-South	2	2	2	2 or 0	2	2
Total Available	N+2	N+2	N+2	N	N+2	N+2
Frac or Bladder Tank Req'd N = 1						
On-Site North - Frac	1	1	1	1 or 0	1	1
FLEX-North - Bladder	1	1	1	1 or 0	1	1
FLEX-South - Bladder	1	1	1	1 or 0	1	1
Total Available	N+2	N+2	N+2	N	N+2	N+2
Air-Powered Diaphragm Pumps Req'd N = 2						
FLEX-North	2	2	2	2 or 0	2	2
FLEX-South	2	2	2	2 or 0	2	2
Refuel Floor	1	1	1	1 or 0	1	1
Total Available	N+3	N+3	N+3	N	N+3	N+3
Battery Room Fans						

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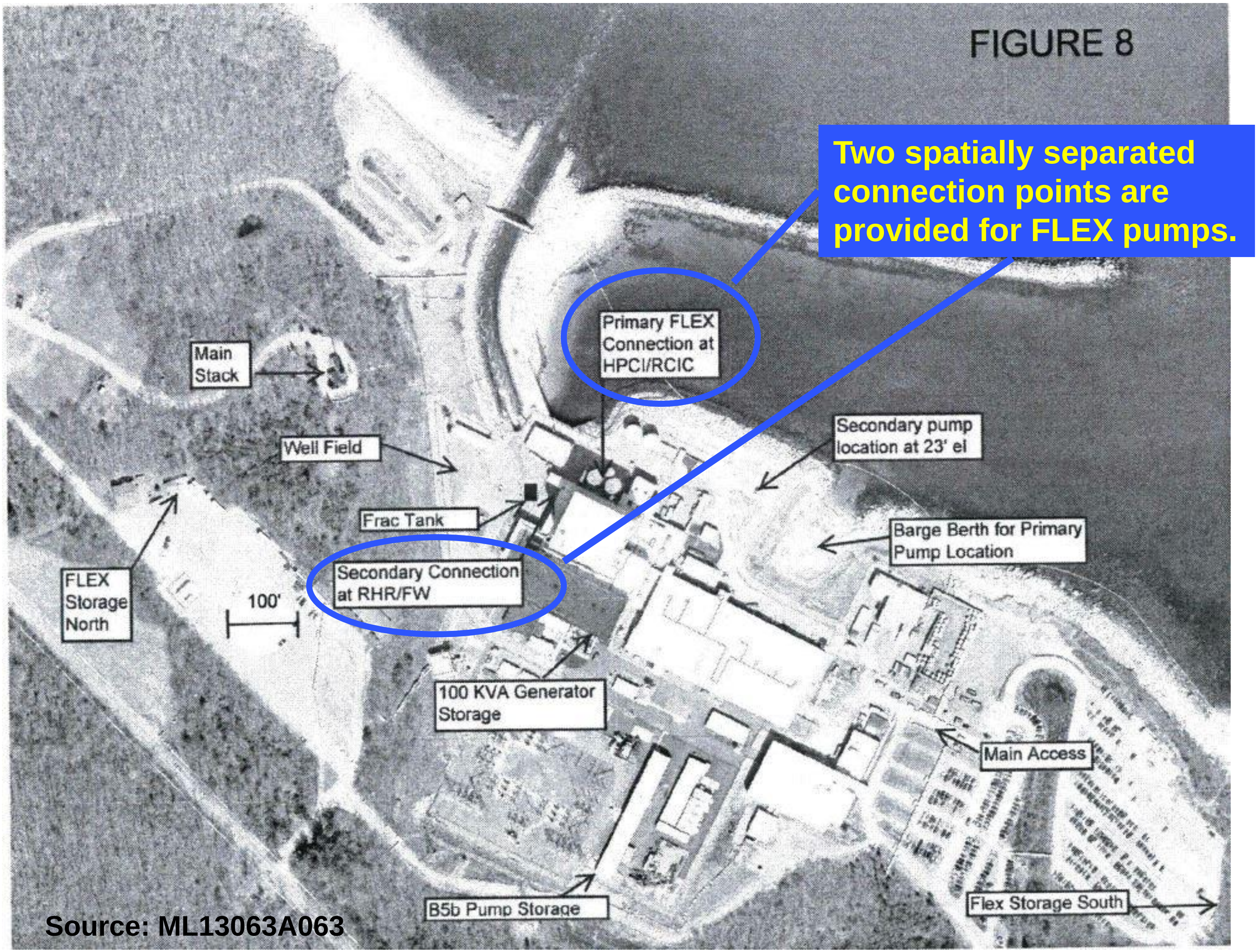
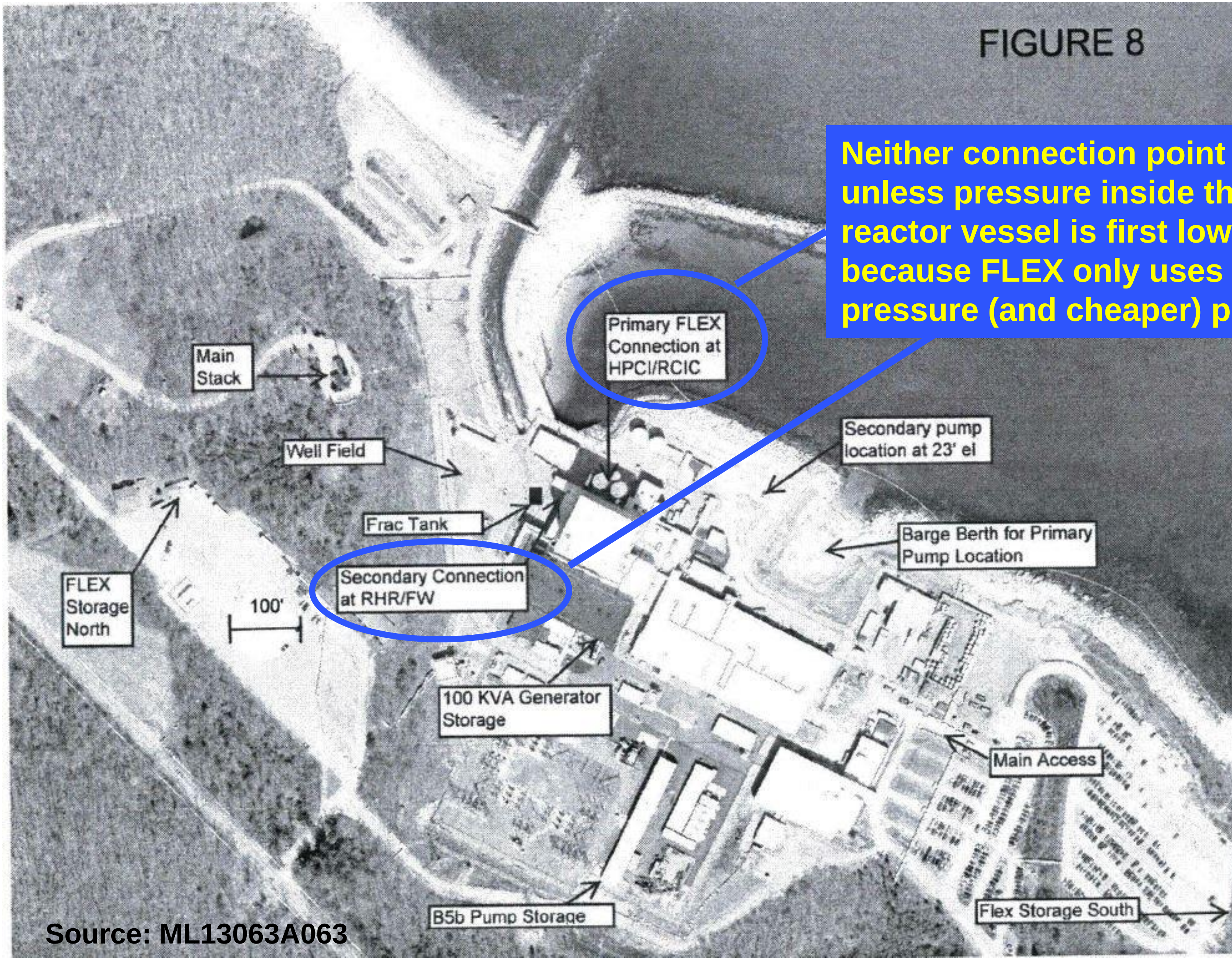
In general, FLEX provides at least N+1 widgets or connections when N is required for success.

Source: ML13063A063

On the Good Side



On the Caveat Side



Neither connection point works unless pressure inside the reactor vessel is first lowered because FLEX only uses low pressure (and cheaper) pumps.

Source: ML13063A063

On the Caveat Side

February 2013 FLEX Integrated Plan

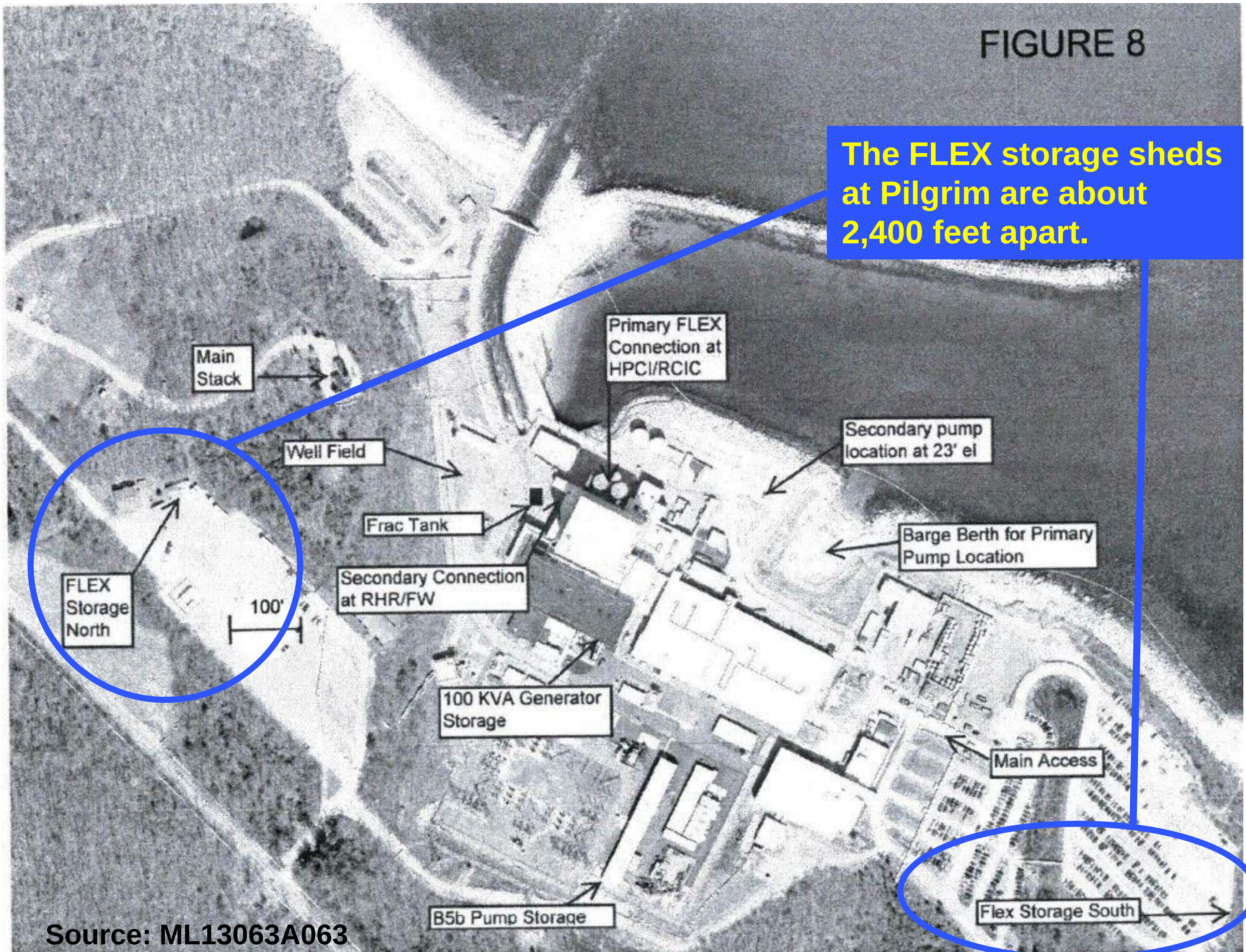
Equipment / Event (Note 1)	Qty	Seismic	External Floods	Hurricane / Tornado	Snow, Ice, Cold	High Temps
Duplex Strainer Trailer 400 GPM 1/8 Inch Size Req'd N = 1						
FLEX-North	1	1	1	1 or 0	1	1
FLEX-South	1	1	1	1 or 0	1	1
Total Available	N+1	N+1	N+1	N	N+1	N+1
Resin Demin Skid 60 cu-ft Mixed Bed Req'd N = 2						
FLEX-North	2	2	2	2 or 0	2	2
FLEX-South	2	2	2	2 or 0	2	2
Total Available	N+2	N+2	N+2	N	N+2	N+2
Frac or Bladder Tank Req'd N = 1						
On-Site North - Frac	1	1	1	1 or 0	1	1
FLEX-North - Bladder	1	1	1	1 or 0	1	1
FLEX-South - Bladder	1	1	1	1 or 0	1	1
Total Available	N+2	N+2	N+2	N	N+2	N+2
Air-Powered Diaphragm Pumps Req'd N = 2						
FLEX-North	2	2	2	2 or 0	2	2
FLEX-South	2	2	2	2 or 0	2	2
Refuel Floor	1	1	1	1 or 0	1	1
Total Available	N+3	N+3	N+3	N	N+3	N+3
Battery Room Fans						

Page 79 of 83

0 < N and nature rather than the NRC's assessment determines the outcome.

Source: ML13063A063

On the Good Side



On the Caveat Side

NRC *assumes* that "Should one storage area be lost, the surviving storage area has adequate equipment."

Tornado that devastated Moore, OK must not have been aware of the 2,400 foot rule.

Disaster Picture: FEMA

Source: ML13225A587

On the Caveat Side

February 2013 FLEX Integrated Plan

Equipment / Event (Note 1)	Qty	Seismic	External Floods	Hurricane / Tornado	Snow, Ice, Cold	High Temps
(Note 5) Req'd N = 2						
FLEX-North	1	1	1	1 or 0	1	1
FLEX-South	1	1	1	1 or 0	1	1
Batt Room, Staged	2	2	2	2	2	2
Total Available	N+2	N+2	N+2	N+1	N+2	N+2
Small Diesel Generator, 120/240 VAC 1-PH Req'd N = 1 - 12 kW Req'd N = 2 - 6 kW						
FLEX-North	3	3	3	3 or 0	3	3
FLEX-South	3	3	3	3 or 0	3	3
Total Available	N+3	N+3	N+3	N	N+3	N+3
Debris Removal Wheel Loader On-Site, Req'd N = 1	N	N	N	N	N	N

Notes:

1. The Tornado Event is the most limiting and potentially results in only "N" FLEX Equipment available, including the loss of the B.5.b Pump, but this event has no potential to drain the SFP, which is the basis for the primary SFP Spray capability of the B.5.b Pump in accordance with 10 CFR 50.54(hh) for Security-Related Events. All other events will result in at least "N+2" FLEX Pumps available, each of which has the same capability as the B.5.b Pump and can provide SFP Spray at the same flow rates and conditions. The B.5.b requirement includes a SFP makeup rate of at least 500 GPM and SFP Spray requirement of 250 GPM and is not based on a particular leakage or boil-off makeup rate, it is the required spray flow needed to prevent exposed spent fuel from reaching the oxidation temperature after a SFP draindown. This B.5.b capability is not compromised in any way by the simultaneous deployment of FLEX Equipment. For all ELAP and LUHS Events, "N" FLEX Pumps provide the required capacity for Core Cooling, Containment Heat Removal, and SFP Makeup Water.

N+3 = N when only one debris remover is provided, unless events are "tidy" and only deposit debris in designated places.

Source: ML13063A063

On the Caveat Side

NRC technical evaluation report:

“The single debris removal equipment identified may not be able to move debris to enable transport of equipment within the 6-9 hour time restriction for the pumps and generators.”

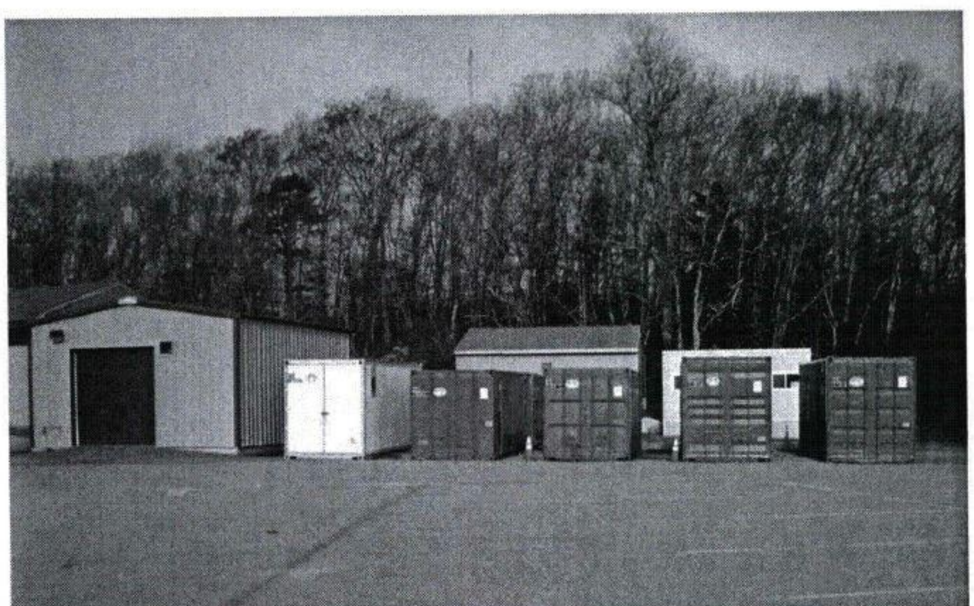
Source: ML13225A587

On the Good Side

February 2013 FLEX Integrated Plan

Attachment 5
PNPS FLEX Equipment Storage Sea Vans

PNPS will be storing FLEX equipment in Sea Vans at two separate locations at the opposite extremes of the Owner Controlled Area (approximately 1800ft geographically separated). The locations are also at the higher elevations on the site, a minimum of 30ft above mean sea level. The North Storage Area is partially established and is as shown in the photos below. The Sea Vans are supplied with AC power for equipment heaters and lighting, one Sea Van is environmentally controlled, and the others ventilated. The site storage is located and arranged to also support equipment testing, operability, and provide for rapid deployment.



FLEX Storage North; lighting and power is provided to each Sea Van.

Equipment over and above that provided for B.5.b is now onsite.

Source: ML13063A063

On the Caveat Side

Equipment heaters protect FLEX equipment from cold weather damage <u>before</u> the BDBEE.

NRC Bulletin 79-24 discussed events at nuclear plants where safety-related systems were disabled by cold weather. These systems were monitored and surveilled, yet failed.

On the Caveat Side

NRC requires that workers periodically check air inlet and outlet ventilation ports for dry casks for blockage, but not for FLEX storage pods.

Source: NRC Flickr Gallery

Pilgrim's BDBEE Plan

A simplified description of the Pilgrim Integrated Plan to mitigate the postulated extended loss of ac power (ELAP) event is that the licensee will initially remove the core decay heat by using the Reactor Core Isolation Cooling (RCIC) system. The steam-driven RCIC pump will initially supply water to the reactor from the condensate storage tank, or the suppression pool, depending on availability. Steam from the reactor will then be vented through the safety relief valves to the suppression pool in the torus to gradually cool down the reactor pressure vessel (RPV). RPV depressurization will be stopped at a pressure of about 120 pounds per square inch gauge (psig) to ensure sufficient steam pressure for continued RCIC operation. Once FLEX pumps are deployed, with suction aligned to Cape Cod Bay, the RCIC turbine will be shut down and the FLEX pumps will be used to inject seawater into the RPV. Water will fill the RPV and flow out the SRVs to the suppression pool. Before the suppression pool temperature exceeds 281 degrees Fahrenheit, the suppression pool (torus) will be vented to atmosphere using the hardened vents to release heat and stop the temperature increase. In the long term, the licensee will fill a tank with fresh water from wells at the site, and then inject fresh water into the RPV and establish a stable water level with heat removal by boiling. The licensee's analysis shows that the suppression pool will not overfill during this event.

Source: ML13225A587

The plan non-conservatively *assumes* that the reactor vessel pressure gets lowered enough to let FLEX's low pressure pump(s) provide makeup flow.

Pilgrim's BDBEE Plan

A simplified description of the Pilgrim Integrated Plan to mitigate the postulated extended loss of ac power (ELAP) event is that the licensee will initially remove the core decay heat by using the Reactor Core Isolation Cooling (RCIC) system. The steam-driven RCIC pump will initially supply water to the reactor from the condensate storage tank, or the suppression pool, depending on availability. Steam from the reactor will then be vented through the safety relief valves to the suppression pool in the torus to gradually cool down the reactor pressure vessel (RPV). RPV depressurization will be stopped at a pressure of about 120 pounds per square inch gauge (psig) to ensure sufficient steam pressure for continued RCIC operation. Once FLEX pumps are deployed, with suction aligned to Cape Cod Bay, the RCIC turbine will be shut down and the FLEX pumps will be used to inject seawater into the RPV. Water will fill the RPV and flow out the SRVs to the suppression pool. Before the suppression pool temperature exceeds 281 degrees Fahrenheit, the suppression pool (torus) will be vented to atmosphere using the hardened vents to release heat and stop the temperature increase. In the long term, the licensee will fill a tank with fresh water from wells at the site, and then inject fresh water into the RPV and establish a stable water level with heat removal by boiling. The licensee's analysis shows that the suppression pool will not overfill during this event.

Source: ML13225A587

The plan non-conservatively *assumes* that instrumentation not covered by post-Fukushima orders will guide operators into taking proper and timely actions.

Pilgrim's BDBEE Plan

A simplified description of the Pilgrim Integrated Plan to mitigate the postulated extended loss of ac power (ELAP) event is that the licensee will initially remove the core decay heat by using the Reactor Core Isolation Cooling (RCIC) system. The steam-driven RCIC pump will initially supply water to the reactor from the condensate storage tank, or the suppression pool, depending on availability. Steam from the reactor will then be vented through the safety relief valves to the suppression pool in the torus to gradually cool down the reactor pressure vessel (RPV). RPV depressurization will be stopped at a pressure of about 120 pounds per square inch gauge (psig) to ensure sufficient steam pressure for continued RCIC operation. Once FLEX pumps are deployed, with suction aligned to Cape Cod Bay, the RCIC turbine will be shut down and the FLEX pumps will be used to inject seawater into the RPV. Water will fill the RPV and flow out the SRVs to the suppression pool. Before the suppression pool temperature exceeds 281 degrees Fahrenheit, the suppression pool (torus) will be vented to atmosphere using the hardened vents to release heat and stop the temperature increase. In the long term, the licensee will fill a tank with fresh water from wells at the site, and then inject fresh water into the RPV and establish a stable water level with heat removal by boiling. The licensee's analysis shows that the suppression pool will not overfill during this event.

Source: ML13225A587

The plan non-conservatively *assumes* that RCIC takes suction from the suppression pool. When RCIC takes suction from its normal and usual source, the suppression pool fills more.

* * * * *

The caveat would shrink if:

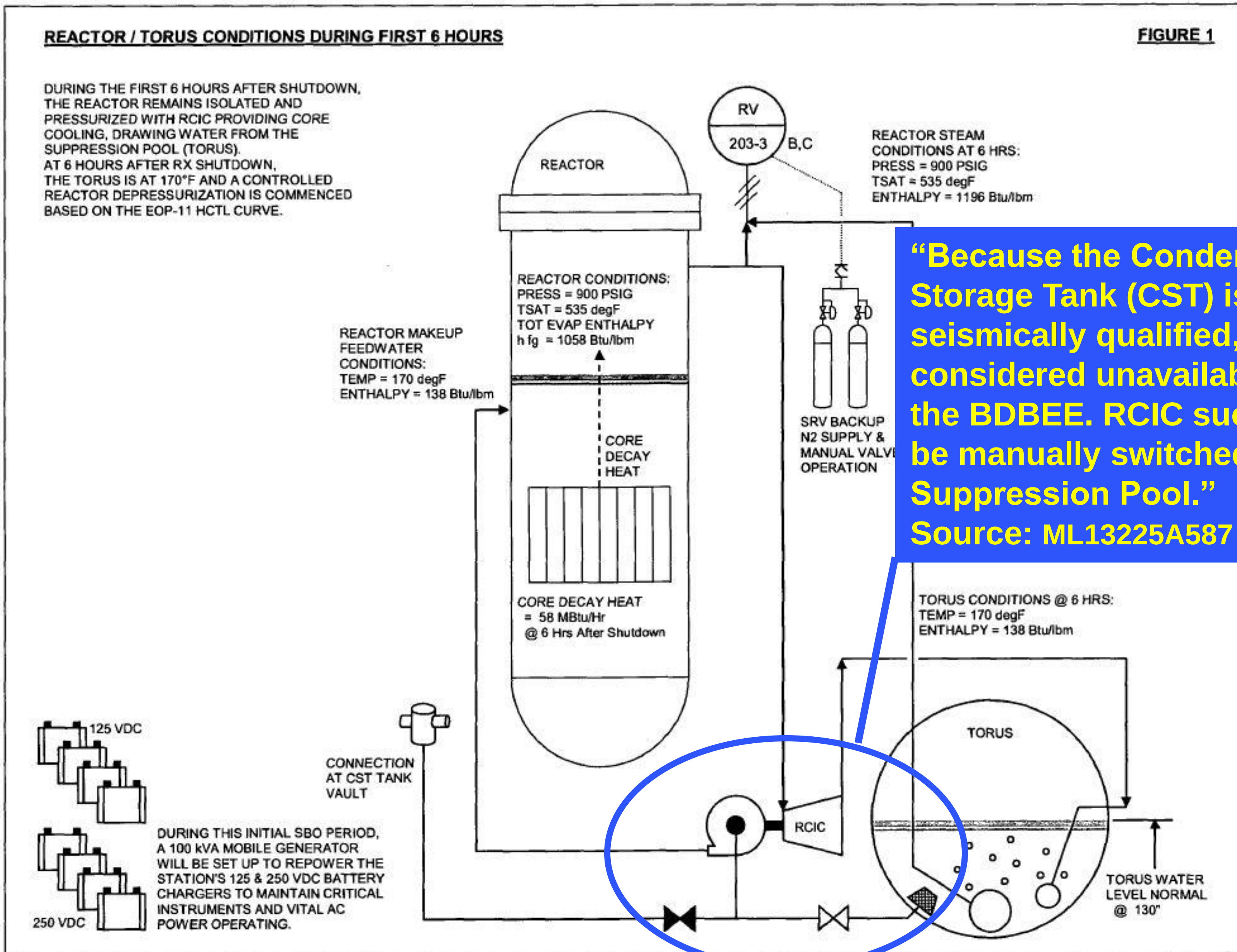
- FLEX employed both high and low pressure pumps
- FLEX storage sheds were less vulnerable to common-mode losses
- Regulatory requirements governed FLEX equipment while in storage
- Non-conservative assumptions that transform BDBEE into BBDBEE were eliminated

Acronym List

BDBEE – one acronym too many in the series of Class 9, severe accident, and Beyond Design Basis External Event labels for bad days
FLEX – Diverse and Flexible Mitigation Capability
NRC – Nuclear Regulatory Commission
RCIC – Reactor core isolation cooling
SBO – station blackout where all AC power is unavailable

Backup Slides

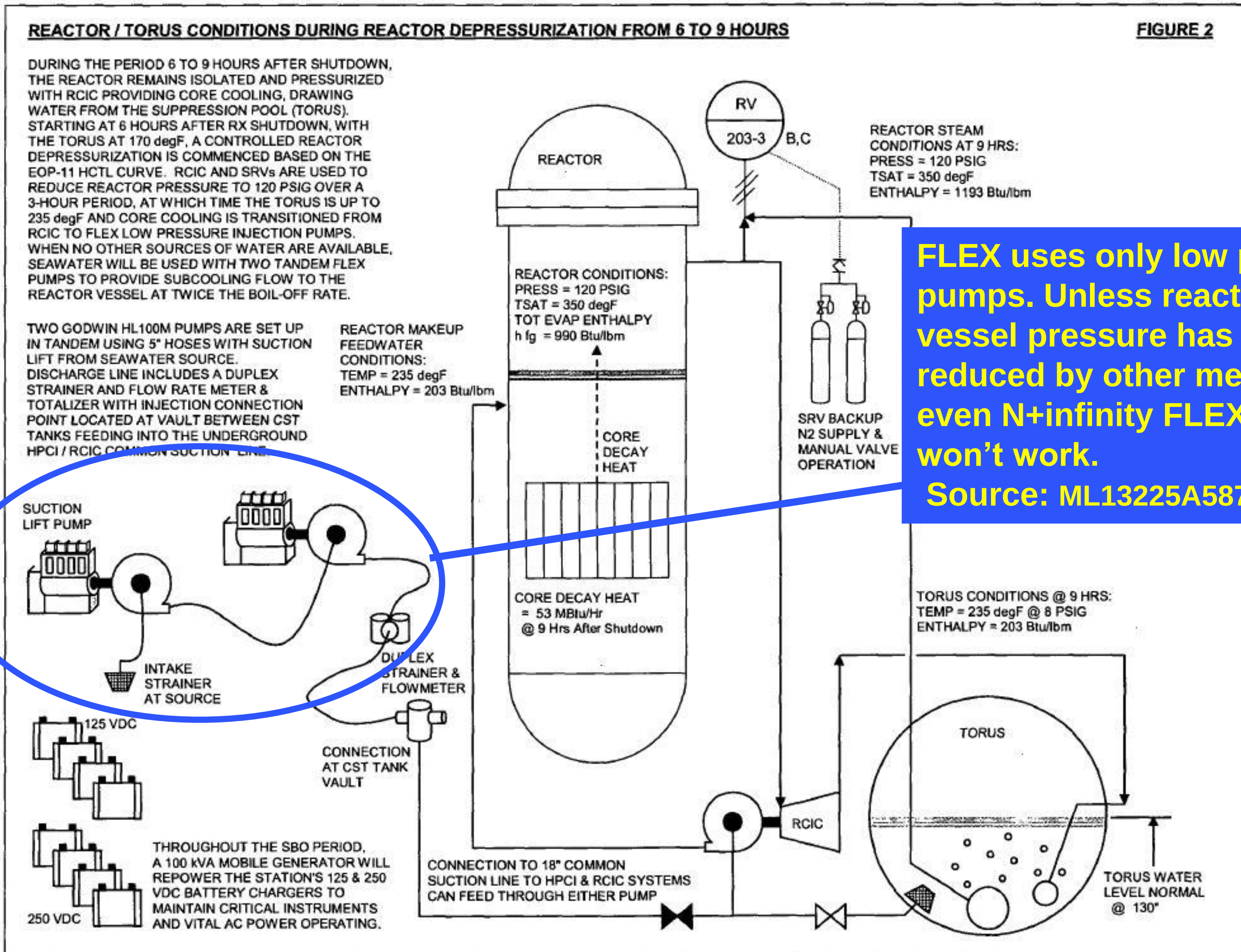
Pilgrim’s BDBEE Plan



“Because the Condensate Storage Tank (CST) is not seismically qualified, it is considered unavailable for the BDBEE. RCIC suction will be manually switched to the Suppression Pool.”
Source: ML13225A587

Source: ML13063A063

Pilgrim's BDBEE Plan

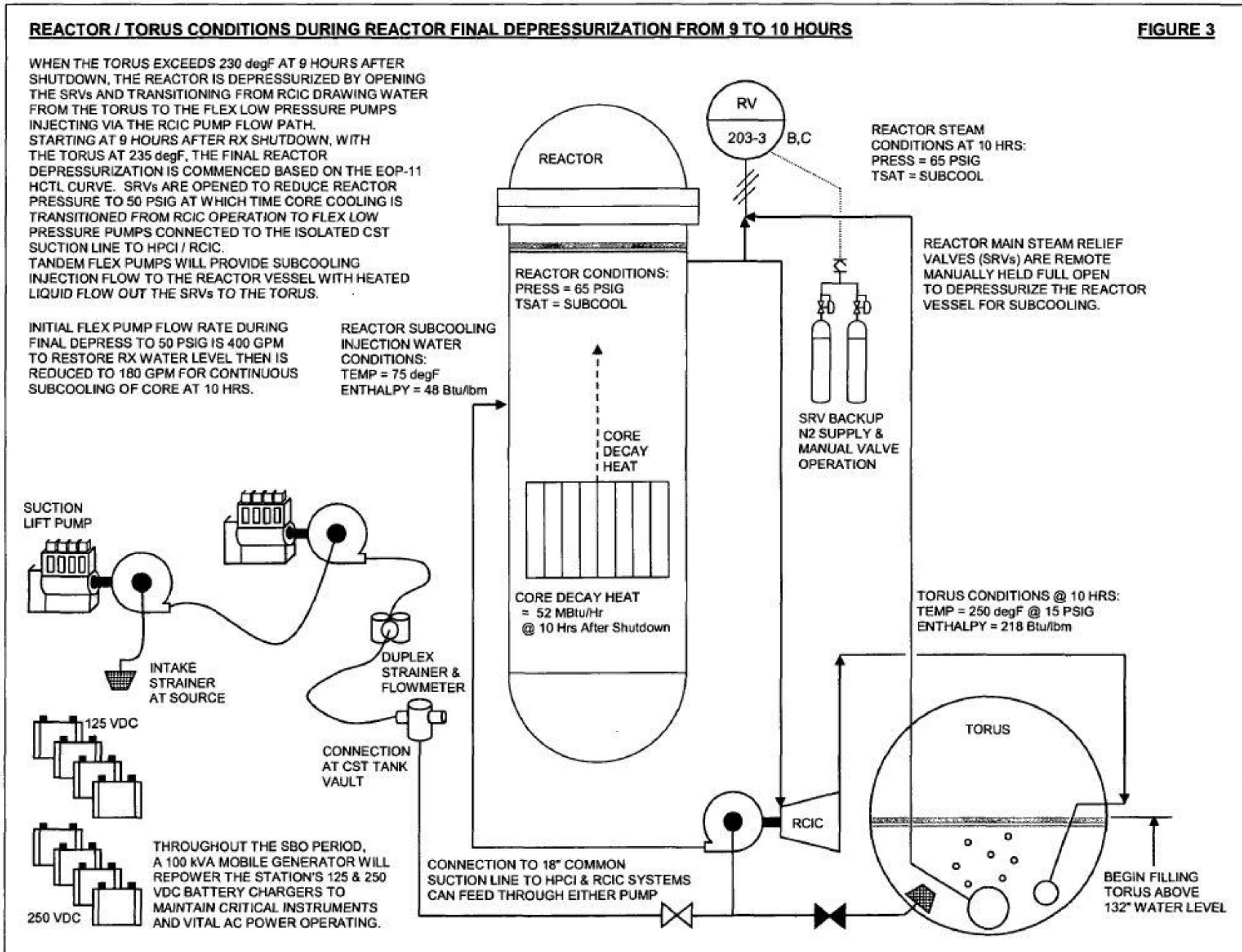


FLEX uses only low pressure pumps. Unless reactor vessel pressure has been reduced by other means, even N+infinity FLEX pumps won't work.

Source: ML13225A587

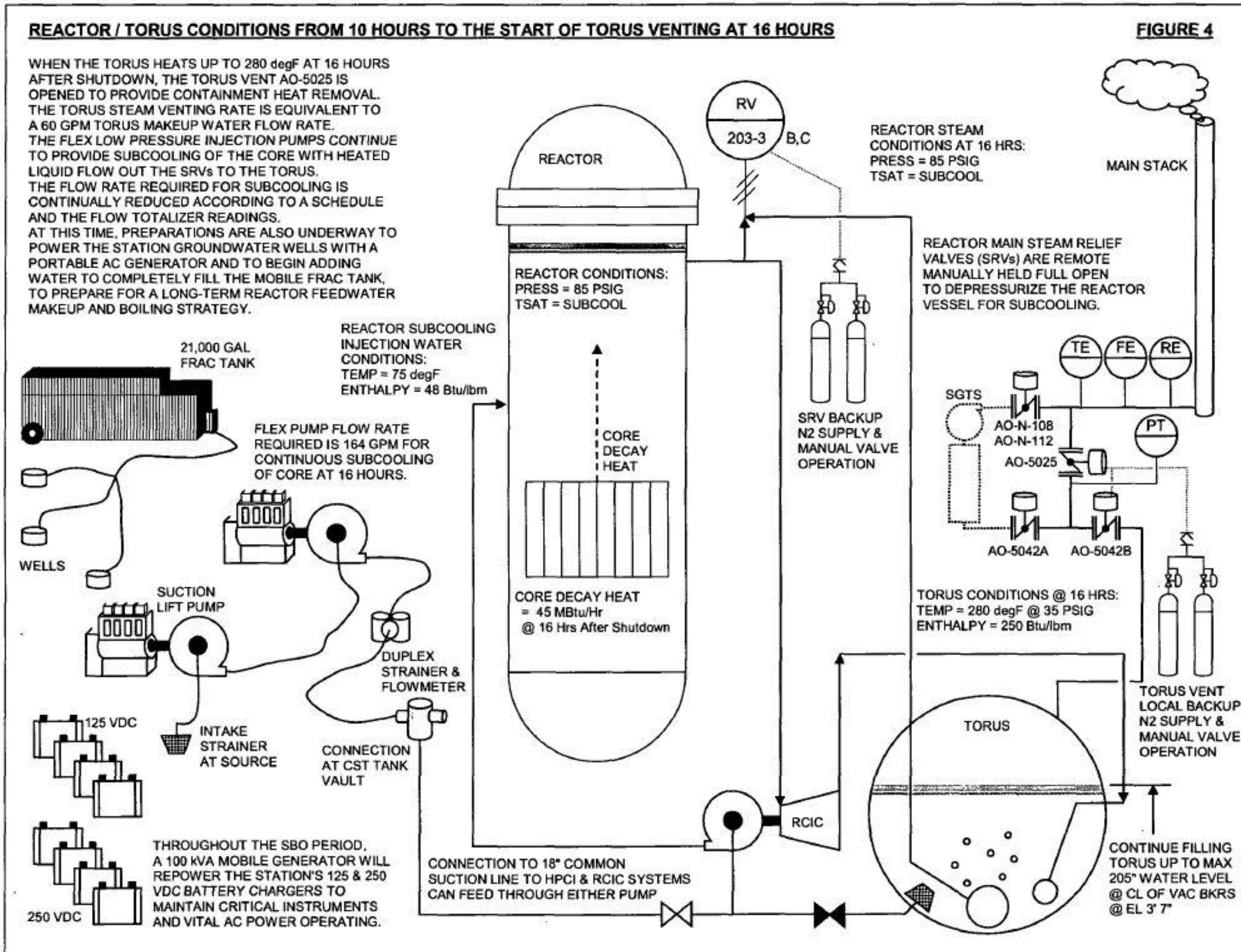
Source: ML13063A063

Pilgrim's BDBEE Plan



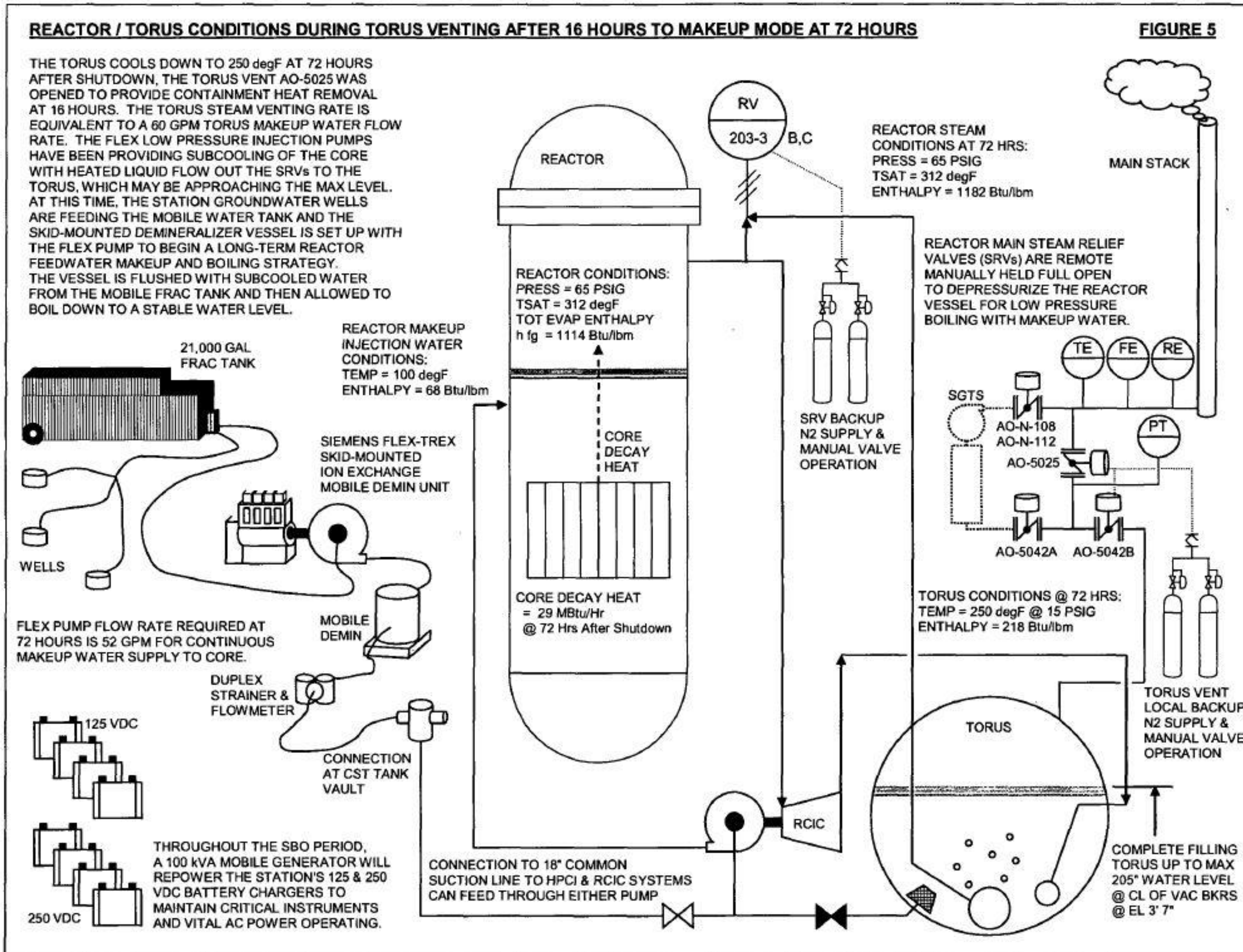
Source: ML13063A063

Pilgrim's BDBEE Plan



Source: ML13063A063

Pilgrim's BDBEE Plan



Source: ML13063A063

Wang, Weidong

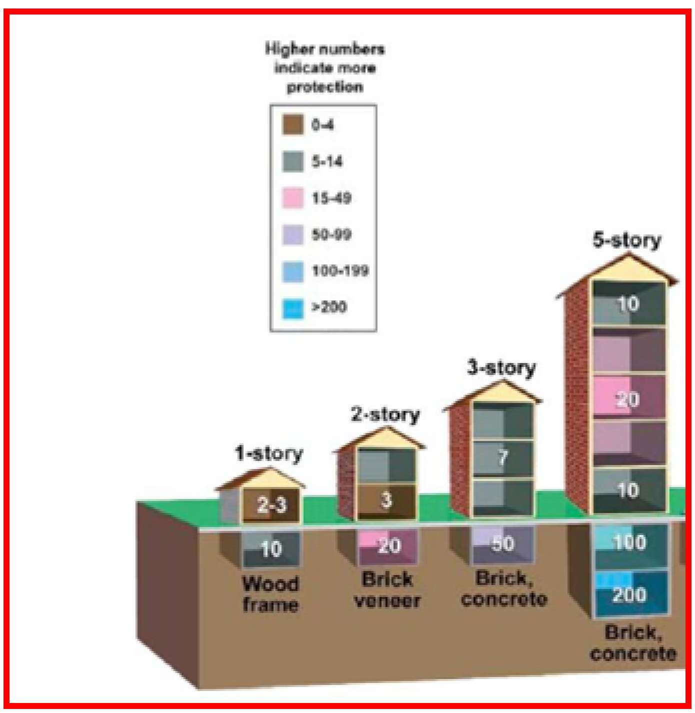
From: Mary Lampert <mary.lampert@comcast.net>
Sent: Tuesday, July 07, 2015 1:14 PM
To: Wang, Weidong
Subject: [External_Sender] Evacuation Time Estimates
Attachments: 2.206 PW KLD ETE 08.30.13.pdf; 01.22.15 REJECTION KLD ML14318A418.pdf

To the Members of the ACRS:

The May Staff Draft relies on ETE's in its analysis; the attached 2.206 demonstrates that the ETE relies on assumptions that underestimate evacuation times. The NRC rejected the petition saying simply the ETE followed the recipe for doing ETE's – no attention to the merits of the argument brought forward. The decision is attached- highlighted and comments made by in the text

As pointed out by an ACRS member during the July 7 meeting, if scenarios in addition to earthquakes had been modeled, evacuation delays would be likely to increase. During winter storm Juno in SE Massachusetts January 2015, evacuation was not possible for 4 days, according to the EMD for the Town of Duxbury. Additionally there is no assurance releases will not occur far sooner than modeled.

As for sheltering:
Source: EPA Protective Action Guidance for Radiological Incidents, March 13, 2013



I am happy to provide any additional information. I have reviewed for the Town of Duxbury annually the town's Pilgrim Radiological Emergency Plan and Procedures for over two decades.
Thank you for your attention to this important public safety issue that is being approached by staff with a scarcity of commonsense.

Respectfully,

Mary

August 30, 2013

Mr. James Borchardt
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
By Mail & Email: NrcExecSec@nrc.gov

PILGRIM WATCH'S 2.206 PETITION TO MODIFY, SUSPEND, OR TAKE ANY OTHER ACTION TO THE OPERATING LICENSE OF PILGRIM STATION UNTIL THE NRC CAN ASSURE EMERGENCY PREPAREDNESS PLANS ARE IN PLACE TO PROVIDE REASONABLE ASSURANCE PUBLIC HEALTH & SAFETY ARE PROTECTED IN THE EVENT OF A RADIOLOGICAL EMERGENCY

I. INTRODUCTION

Pursuant to §2.206 of Title 10 in the Code of Federal Regulations, Pilgrim Watch (Hereinafter "PW") on behalf of its members and members of the Pilgrim Coalition, Project for Entergy Accountability, Cape Cod Bay Watch, EcoLaw, Beyond Nuclear, Greenpeace, and others request that the Nuclear Regulatory Commission (NRC) to institute a proceeding to modify, suspend or take any other action[1] as may be proper to the operating license of Pilgrim Station in order that the NRC can assure Pilgrim's Radiological Emergency Plan and Standard Operating Procedures/Guidelines are based on accurate and credible Evacuation Time Estimates (ETEs).

ETEs provide information for use in the formulation of a licensee's protective action recommendation and the ORO's protective action decisions. It is important that the time required

[1] NRC Enforcement actions include: notices of violation, civil penalties, orders, notice of nonconformance, confirmatory action letters, letters of reprimand, and demand for action.

to evacuate the public is both clearly understood and reliable to ensure appropriate protective action is implemented.

Because Pilgrim's ETE underestimates evacuation times there is no reasonable assurance "to ensure appropriate protective action is implemented;" the population will achieve a timely evacuation; that public health and safety will be protected in the event of a radiological emergency; or that the NRC can satisfy its statutory requirement to protect public health and safety.

The primary basis for this petition is two recent documents prepared by KLD for Entergy: The *KLD Pilgrim Evacuation Estimate December 12, 2012 Final Report KLD-TR-510*[2] (Hereinafter, "ETE") and the attached *KLD MEMO to John Giarrusso (MEMA) from Chris Chaffee (KLD) Regarding the Cape Cod Telephone Survey Results*, July 25, 2013, attached (Hereinafter, "Cape Survey"); and the attached August 16, 2013 Letter from Senator Markey and Senator Warren regarding Pilgrim's ETE and Cape Telephone Survey to Leo Denault (Entergy) and forwarded to Chairman MacFarlane's Office.

These documents show that Entergy's Evacuation Time Estimates (ETEs) for Pilgrim Station are based on inaccurate assumptions and simply are not credible. The ETE's fundamental flawed assumptions and data explain the ETE's absurd conclusion that even in the worst case scenario everyone in the EPZ will be evacuated in about six hours.

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[2] NRC Electronic Library, Accession Number ML13023A031

Inaccurate Assumptions Underestimate Demand - Total Number People & Vehicles Evacuating

1. ETE, unlike the Cape Survey, relied on a Telephone Survey that did not inform survey respondents that the questions related to a nuclear emergency, and thus significantly underestimated how many would evacuate.

2. The ETE's Shadow Evacuation assumptions incorrectly assume that only 20% of those instructed not to evacuate will voluntarily evacuate anyway.

3. The ETE incorrectly assumes that those in the EPZ will follow a staged keyhole evacuation. (ETE, 7.2)

4. The KLD ETE underestimated demand by failing to take proper account of the Summer Transient Population.

5. The ETE Study underestimated employees, thus Lowering Demand Estimates

6 Evacuation of the school population & transportation dependent at nursing/group homes were underestimated.

Inaccurate Assumption/Estimates Regarding Road Capacity

7. The ETE fails to account for chronically heavy traffic over Summer weekends & special events that significantly increases travel times.

8. ETE assumptions about traffic flow during inclement weather & peak commuter/holiday traffic are not credible.

9. The ETE's estimates for specific roadway capacity are not credible

10. Emergency Personnel: The ETE assumes, absent factual support, that emergency personnel will be available in sufficent number to assure timely traffic flow.

Inaccurate Assumptions Regarding Trip Generation Times

11. Trip generation time relied on flawed telephone survey & assumptions.

12. The ETE incorrectly assumed a rapidly escalating accident, and that mobilization of the general population will commence within 15 minutes after siren notification.

13. KLD failed to consider the impact of delayed staffing traffic control points on the ETE.

14. The ETE incorrectly assumed that 25% of the EPZ households will await the return of a commuter prior to evacuating underestimating vehicles.

15. The ETE incorrectly assumes that 50% of the transportation dependent population will rideshare.

16. The ETE incorrectly assumes timely evacuation of transportation dependent.

17. The ETE assumptions about mobilization times for school population & special facilities are not credible.

18. The ETE assumptions about trip generation for populations on boats are not credible.

19. The ETE ignores the impact of voluntary evacuations from Cape Cod that would have a large impact on traffic in the EPZ; and ignores the effect of voluntary evacuations within the EPZ and shadow evacuation that would slow EPZ evacuation times.

II. FACTUAL BASIS

The ETE covers demand estimation, estimation of roadway capacity, and development of evacuation time estimates for various subgroups – estimation of trip mobilization time and trip generation time, and evacuation time estimates. Every section of the ETE is flawed and accounts for the absurd conclusion that an evacuation in a radiological emergency at Pilgrim will be accomplished in six hours.

A. DEMAND ESTIMATES UNDERESTIMATED

The assessment of demand estimation provides the total number of people and vehicles to be evacuated for each of the population groups. Both the ETEs and the Cape Survey underestimated demand.

1. ETE, Unlike the Cape Survey, Relied On a Telephone Survey That Did Not Inform Survey Respondents That the Questions Related to a Nuclear Emergency thusSignificantly Underestimating How Many Would Evacuate

The EPZ Telephone Survey sampled only those within the EPZ. By design, its questions never used the words "nuclear" or "radiological." They simply refer to “an emergency.”

The ETE (ETE Attachment A, F-14) interviewer instructions refers to “emergency planning,” not to a nuclear or radiological emergency.

Telephone Survey Instrument

Hello, my name is ___________ and I'm conducting a survey for the Emergency Management Agencies of Carver, Duxbury, Kingston, Marshfield and Plymouth municipalities. The information you provide will be used for emergency planning to enhance local response plans. Emergency planning for some hazards may require evacuation. Your answers to my questions will greatly contribute to this effort. I will not ask for your name.

COL. 1	Unused
COL. 2	Unused
COL. 3	Unused
COL. 4	Unused
COL. 5	Unused
Sex	COL. 8
	1 Male
	2 Female

The only ETE questions relating to this Petition are Questions 13A and 13 B (ETE Attachment A, F-21) and likewise they do not refer to a radiological or nuclear emergency, simply an emergency.

13A.	Emergency officials advise you to take shelter at home in an emergency. Would you: (READ ANSWERS) A. SHELTER; or B. EVACUATE	COL. 52 1 A 2 B X DON'T KNOW/REFUSED
13B.	Emergency officials advise you to take shelter at home now in an emergency and possibly evacuate later while people in other areas are advised to evacuate now. Would you: (READ ANSWERS) A. SHELTER; or B. EVACUATE	COL. 53 1 A 2 B X DON'T KNOW/REFUSED

The Cape Telephone Survey sampled residents throughout the Cape, not simply those in the 10-15 mile zone. Unlike the ETE, the Cape Telephone Survey asked the respondent what s/he would do if there was "an incident at the Pilgrim Nuclear Power Station, " rather than saying nothing about the type of "emergency" involved.

The difference in the results of the two telephone surveys clearly demonstrated that any telephone survey designed to obtain reliable information from respondents must tell the respondent upfront that the survey is for an accident at the nuclear power plant.

The EPZ Telephone Survey failure to tell respondents that the survey was for a nuclear accident, was designed to confirm KLD's and federal guidance assumption that only 20% of the population would self evacuate and they would follow a segmented evacuation. Question 13A (EPZ Survey F-10) asked:

> *"Emergency officials advise you to take shelter at home in an emergency. Would you?"* This question is designed to elicit information regarding compliance with instructions to shelter in place. The results indicate that 81 percent of households who are advised to shelter in place would do so; the remaining 19 percent would choose to evacuate the area. Note the baseline ETE study assumes 20 percent of households will not comply with the shelter advisory, as per Section 2.5.2 of NUREG/CR-7002. Thus, the data obtained above is in good agreement with the federal guidance.

The Cape Cod Telephone Survey could not more clearly show that if potentially affected respondents were asked "would you evacuate" "if they were an incident at the Pilgrim Nuclear Power Station," 70% (not the 20% assumed by the NRC or the 19% of the ETE) would do so.

"Suppose there were an incident at the Pilgrim Nuclear Power Station (PNPS) and you were informed that people in the Emergency Planning Zone were advised to evacuate, would you evacuate?" Approximately 70% percent of Cape Cod residents indicated they would evacuate due to a nuclear incident at PNPS.

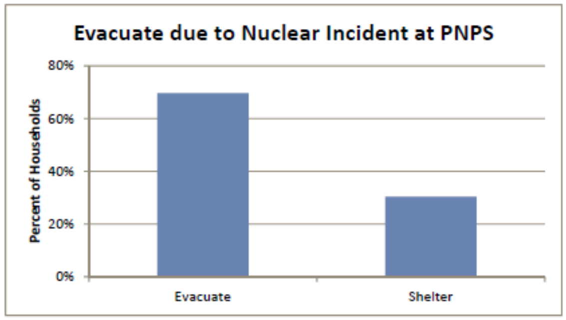

Figure 2. Evacuation Due to an Incident at PNPS

Cape Cod Traffic Study | 3 | KLD Engineering, P.C. Rev. 0

"If you were told that Cape Cod is not in the Emergency Planning Zone for the Pilgrim Nuclear Power Station, would you still evacuate?" Approximately 50% percent of Cape Cod residents indicated they would evacuate due to a nuclear incident at PNPS, even knowing they are not in the Emergency Planning Zone (EPZ).

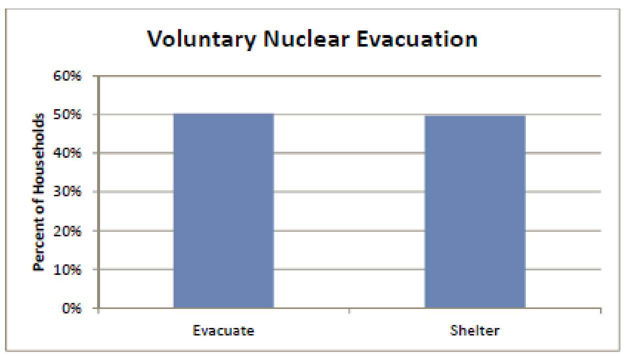

Figure 3. Voluntary Evacuation Due to an Incident at PNPS

It is hardly surprising that many more respondents said they would evacuate when they were told that that there had been a nuclear incident at Pilgrim than if neither Pilgrim, nor nuclear, nor radiological were even mentioned. The Cape Telephone Survey clearly demonstrates that the ETE Telephone Survey, that intentionally did not tell respondents upfront that the question refers to what they would do if there was an accident at the nuclear power plant, is not credible and cannot provide any basis for Pilgrim's evacuation time estimates.

This indisputable conclusion is completely consistent with the previous experience and studies[3] that equally clearly show that people view a nuclear accident very differently than a weather-related evacuation order; and they evacuate in far greater numbers and with less regard for official instructions.

2. The ETE's Shadow Evacuation Assumption is Wrong. It incorrectly assumes that only 20% of those not instructed to evacuate will voluntarily evacuate anyway.

The ETE's Study Methodological Assumption 5 says: "As indicated in Figure 2-2 of NUREG/CR-7002, 100% of the people in the impacted keyhole evacuate. 20% of those within the EPZ, not within the impacted keyhole, will voluntarily evacuate. 20% of those people within the Shadow Evacuation will voluntarily evacuate." (ETE, 2-2)

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[3] Studies regarding "shadow evacuation" inside and outside the EPZ indicate that the public will respond once they become aware. Examples: Three Mile Island: the Pennsylvania Governor issued an evacuation advisory (note, it was not an order). It was expected to have precipitated the flight of only 3,400 people (pregnant women and pre-school children within five miles of the plant); instead, a total of 144,000 people (a government figure) evacuated the surrounding region. Subsequent surveys in New York by Dr. Zeigler indicated that the public outside the 10-mile EPZ would evacuate once they heard there was a nuclear emergency. Recognizing that the public has a greater fear of radiation than natural disasters, a shadow evacuation occurred during Hurricane Floyd in 1999 and Hurricanes Katrina and Rita. Again in a chemical accident, the shadow evacuation was studied and documented in the Graniteville South Carolina chlorine spill in 2005. (Zeigler, Donald, Johnson, James, Jr., "Evacuation Behavior In Response To Nuclear Power Plant Accidents," The Professional Geographer, May, 1984; Zeigler, Donald, Testimony Prepared for Westchester County Legislature, Dec 13, 2001, http://www.closeindianpoint.org/evacuation_testimonial.htm; Witt, James Associates, "Review of Emergency Preparedness of Areas Adjacent to Indian Point and Millstone," James Lee Witt Associates, March 2002, http://www.wittassociates.com/index.xml http://www.nirs.org/reactorwatch/emergency/epwittrpt2003.pdf; Seminole County Division of Emergency Management, Evacuation Plans, http://www.seminolecountyfl.gov/dps/em/emprep_evacuation.asp; Duhe,Duke, Evacuation Behavior in Response to the Graniteville, South Carolina, Chlorine Spill, Hazards Research Lab, University South Carolina, 2005, http://www.colorado.edu/hazards/research/qr/qrl178/qr178.html)
Third, it is essential for planning that the public trust the authorities in order for there to be some assurance that the public will follow directions. If the authorities only inform some of the population, irrespective of intentions, they will lose all credibility, increasing the likelihood of a chaotic response.

Figure 2-1. Voluntary Evacuation Methodology

ETE's assumption is based on NRC's NUREG/CR-7002; and was confirmed by the ETE Telephone Survey only when that survey never told respondents the questions pertained to a nuclear incident at Pilgrim.

In the ETE Telephone Survey relatively few respondents said they would not follow "emergency officials" advice "to shelter" when they were <u>told nothing</u> about what the supposed emergency involved. ETE, F-10:

> ***"Emergency officials advise you to take shelter at home in an emergency. Would you?"*** This question is designed to elicit information regarding compliance with instructions to shelter in place. The results indicate that 81 percent of households who are advised to shelter in place would do so; the remaining 19 percent would choose to evacuate the area. Note the baseline ETE study assumes 20 percent of households will not comply with the shelter advisory, as per Section 2.5.2 of NUREG/CR-7002. Thus, the data obtained above is in good agreement with the federal guidance.
>
> ***"Emergency officials advise you to take shelter at home now in an emergency and possibly evacuate later while people in other areas are advised to evacuate now. Would you?"*** This question is designed to elicit information specifically related to the possibility of a staged evacuation. That is, asking a population to shelter in place now and then to evacuate after a specified period of time. Results indicate that 71 percent of households would follow instructions and delay the start of evacuation until so advised, while the balance of 29 percent would choose to begin evacuating immediately.

The Cape Survey, in stark contrast, told respondents the purpose of the survey and because it did the shadow evacuation estimates were very large. The Cape Survey specifically asked respondents, "If you were told that Cape Cod is not in the Emergency Planning Zone for the Pilgrim Nuclear Power Station, would you still evacuate? (Cape Survey Question 3B), "Approximately 50 percent of the Cape Cod residents indicated that they would evacuate due to a nuclear incident at PNPS, even knowing they were not in the Emergency Planning Zone (EPZ (Cape Survey, 4)

"Suppose there were an incident at the Pilgrim Nuclear Power Station (PNPS) and you were informed that people in the Emergency Planning Zone were advised to evacuate, would you evacuate?" Approximately 70% percent of Cape Cod residents indicated they would evacuate due to a nuclear incident at PNPS.

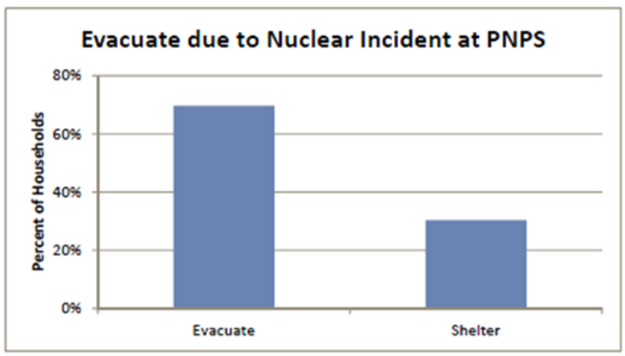

Figure 2. Evacuation Due to an Incident at PNPS

Cape Cod Traffic Study | 3 | KLD Engineering, P.C. Rev. 0

The results of the Cape Survey show, at a 95% confidence level, that the ETE's assumption that no more than 20% will evacuate is not just wrong; it is ludicrous. It cannot properly be used to determine Pilgrim's evacuation time estimates.

The Cape Survey also showed that the ETE's assumption that the "shadow evacuation" includes only the 10-15 mile region is incorrect. The Cape survey included resident respondents throughout the Cape, out to 25 miles. It showed at the 95% confidence level that approximately half of those within 25 miles of Pilgrim would evacuate, even if they knew that they were not in Pilgrim's Emergency Planning Zone.

The Cape Survey demonstrates that, in determining evacuation demand and evacuation time estimates, those living more than 15 miles from Pilgrim cannot be ignored. Those living up to 25 miles away must be expected to evacuate. And, consistent with the previous studies and experience cited above, the percentage of those in the "shadow region" that must be expected to evacuate is at least 2.5 times that assumed by KLD's ETE, the NRC, and the current evacuation time estimates. Half, not "20 percent," "of households will not comply with the shelter advisory."

Comparing the Cape Telephone Survey (and experience and studies) with the ETE Telephone Survey (and NUREG/CR-7200) proves that none of the ETE's 20% estimates of how many would evacuate, and none of Pilgrim's evacuation demand or evacuation time estimates are valid. They also show that the NUREG and ETE estimates of how many would evacuate in the event of a nuclear incident or emergency are indisputably wrong - and they are wrong because they are based on surveys that were intentionally designed to not to tell any respondent what type of emergency was really at issue, and to provide the answer that the industry wanted, rather than any real answer. At a 95% confidence level and with essentially identical possible sampling errors, the Cape Telephone Survey shows that any honest and realistic time estimates must assume that that the number of people who will evacuate within the EPZ is more than three (3) times what the NRC and ETE assumed, and that "shadow evacuation" outside the EPZ will be more than two and a half times the NRC's and ETE's unrealistic assumptions.

There can be no doubt that a 250% to 300% increase in the number of evacuees from within the EPZ will have a dramatic increase in traffic density and speed, and itself will dramatically increase the time necessary to evacuate. There also can be no doubt that a large scale evacuation from Cape Cod will also dramatically increase KLD's faulty evacuation time estimates; traffic

from Cape Code has nowhere to go except onto the evacuation routes for the EPZ. (See 19, below)

3. The ETE Incorrectly Assumes That Those In The EPZ Will Follow A Staged Keyhole Evacuation (ETE, 7.2)

A Staged Evacuation is where one area is told to evacuate and other areas are told to shelter-in-place until directed to evacuate. (NUREG/CR-70002, 1.31) The ETE study (ETE, 7.6) showed that "the staged evacuation option provides no benefits and adversely impacts many evacuees located beyond 2 miles from PNPS."

> **7.6 Staged Evacuation Results**
>
> Table 7-3 and Table 7-4 present a comparison of the ETE compiled for the concurrent (un-staged) and staged evacuation studies. Note that Regions R22 through R27 are the same geographic areas as Regions R02 and R04 through R08, respectively.
>
> To determine whether the staged evacuation strategy is worthy of consideration, one must show that the ETE for the 2 Mile region can be reduced without significantly affecting the region between 2 miles and 5 miles. In all cases, as shown in these tables, the ETE for the 2 mile region is unchanged when a staged evacuation is implemented. The reason for this is that the congestion within the 5-mile area does not extend upstream to the extent that it penetrates to

> within 2 miles of the PNPS. Consequently, the impedance, due to this congestion within the 5-mile area, to evacuees from within the 2-mile area is not sufficient to materially influence the 90th percentile ETE for the 2-mile area. Therefore, staging the evacuation to sharply reduce congestion within the 5-mile area, provides no benefits to evacuees from within the 2 mile region and unnecessarily delays the evacuation of those beyond 2 miles.
>
> While failing to provide assistance to evacuees from within 2 miles of the PNPS, staging produces a negative impact on the ETE for those evacuating from within the 5-mile area. A comparison of ETE between Regions, R22 and R02; R23 and R04; R24 and R05; R25 and R06; R26 and R07; and R27 and R08 reveals that staging retards the 90th percentile evacuation time for those in the 2 to 5-mile area by up to 40 minutes for non-snow scenarios and 1 hour and 20 minutes for snow scenarios (see Table 7-1). This extending of ETE is due to the delay in beginning the evacuation trip, experienced by those who shelter, plus the effect of the trip-generation "spike" (significant volume of traffic beginning the evacuation trip at the same time) that follows their eventual ATE, in creating congestion within the EPZ area beyond 2 miles.
>
> In summary, the staged evacuation option provides no benefits and adversely impacts many evacuees located beyond 2 miles from the PNPS.

The Cape Telephone Survey's finding that 50% of the population would self-evacuate even if they were told that they were not in the EPZ shows that the population will not follow a staged evacuation; far larger numbers will evacuate and traffic estimates considerably slowed.

The ETE's findings have broad significance for emergency planning. The Staged Evacuation concept appears to be NRC's and the licensee's solution to the problem that population has dramatically increased since Pilgrim was licensed in 1972 and the infrastructure is inadequate to support a large evacuation in a timely manner. For example, Plymouth's population has increased three-fold since Pilgrim was constructed – from 18,606 into 56,132 in 2012[4].

4. The KLD ETE Underestimated Demand by Failing to Take Proper Account of the Summer Transient Population

In estimating how many summer transients would evacuate, the KLD ETE inaccurately estimated the size of the summer transient population, and incorrectly assumed that the percentage of summer transients that would choose to self evacuate would be the same as the percentage of year-round residents.

a. <u>The ETE Underestimates Summer Transient Population</u>:

The EPZ ETE section 3.3.1 Seasonal Transient Population explains that: "It is assumed that seasonal residents will be renting homes near the shoreline. Using only those Census blocks that are within half a mile of the waterways, the number of seasonal homes was calculated by determining the percentage of vacant households and subtracting out the average vacant household percentages (24%) within the EPZ. An average household size of 2.5 persons per household is used to determine the seasonal transient population, and the 1.37 transient vehicles. These numbers are adapted from the telephone survey results." (see Appendix F)

The methodology significantly underestimates the transient population. It ignored for example, that summer rentals are not limited to ½ mile from the shore where rental rates are highest and that many summer transients are home owners that want to use their property, not rent it, during the summer.

[4] www.mass.gov/dhcd/iprofile/239.pdf.

Research shows that transients have high levels of spontaneous evacuation and will prepare to evacuate more quickly than residents[5]. The New Jersey Hurricane Evacuation Study, for example, found that "it is reasonable to assume that 90% to 95% of vacationers will evacuate their accommodations if evacuation orders are issued … "90% of vacationers will return home when they evacuate … (and) more than 95% of vacationers….drive from homes (and) [t]hey will use their own vehicles when evacuating."

The 2004 KLD estimated transients within the EPZ at 42,215; the 2012 KLD inexplicably estimated only 20,745. (ETE, 1-10) The overall population has increased in Massachusetts, as have the number of visitors. Neither has decreased by more than half. And, Marshfield's population data alone shows that KLD's estimated are less than half what they should be. KLd estimated Marshfield summer transient population to be 6,102 (ETE, Table 3-4, pg., 3-11); the Boston Globe reported that Marshfield's summer transient population was 12,000,[6] twice KLD's estimate.

b. <u>The Cape Telephone Survey was limited to residents and ignored the large number of transients on Cape Cod and its effect on the ETE</u>

Senators Markey and Warren's letter pointed out that "Cape Cod is a unique geographical area, withover 200,000 permanent residents and as many as 300,000 vacationers in the summer." (Letter at 1-2) In other words, the kld ignores 60% of th summer population. The only routes off the Cape cross the Sagamore and Bourne bridges, which consequently take evacuees onto roadways used by evacuating residents of the Emergency Plannign Zone.

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[5] New Jersey Hurricane Evacuation Study Transportation Analysis, Technical Memoranda, Prepared for US Army Corps of Engineers Philadelphia District, by PBS&J Tallahassee FLA, June 2007 (http://www.ready.nj.gov/plan/pdf/maps/hurrevacution_study.pdf)

[6] Boston Globe, Globe South, *Don't Love that Dirty Water*, Jessica Bartlett, August15, 2013, pg., 6

Based on the telephone Survey of Cape residents, it is safe to predict over 70% of the vacationers will evacuate if they leave in a Pilgrim emergency and more than 50% will evacuate even if told not to do so. The reason that their numbers are likely to be higher than residents is that transients away from their residence are likely to elect to evacuate to their home on the mainland. Home is associated with safety.

5. The ETE Study Underestimated Employees, thus Lowering Demand Estimates

The ETE only accounts for non-residents who work in the EPZ for larger employers (15 employers and the schools); that resulted in only 1,146 employees and 1,092 vehicles.[7] The estimate fails to account for the many smaller employers in the EPZ who employ non-EPZ residents. While each business may only employ a few non-EPZ residents, there are many small businesses and these potential evacuees add up and need to be accounted for. The numbers are largest in the summer months to service the tourism industry and those employees are lilkey to commute in their own vehicles adding to demand data and traffic congestion.

The Cape Suvey also does not include non-resident employees, a number that increases dramatically over the summer months to service the tourism industry. Larger numbers of evacuees will slow evacuation of the EPZ sharing and slowing the same mainland evacuation routes.

Many of the seasonal and low-wage jobs are filled by students and temporary foreign workers, who migrate to the Cape during the tourism season, specifically for temporary employment (http://www.sustaincapecod.org/indicators/Business)

[7] ETE, Table 3-5, pg., 3-15, *Summary of Non-EPZ resident Employees and Employee Vehicles* total EPZ

6 Evacuation of School Population & Transportation Dependent at Nursing/Group Homes - Underestimated

The ETE incorrectly assumes that parents will not attempt to pick up their children from schools, and that students will be evacuated by bus in an obvious effort to lower the number of vehicles evacuating. It ignores that high school students with cars will self evacuate, with younger siblings. It ignores that family members are likely to go to elderly housing complexes, nursing homes, and group homes to gather their loved ones. As a consequence, a far larger number of vehicles will be on roadways than modeled.

For example, the Town of Duxbury recognized that parents will go to schools in the event of an emergency and before a general emergency to pick up. Therefore a protocol for picking up students is in the School Department's Standard Operating Procedure (SOP).[8]

B. <u>INACCURATE ASSUMPTIONS /ESTIMATES ROAD CAPACITY (ETE, Ch., 4)</u>

Roadway capacity is defined as the maximum rate at which vehicles can be expected to traverse a secton of roadway during a given time period under prevailing roadway traffic and control conditions. Roadway capacity influences evacuation travel time particularly as traffic demand approaches or exceeds capacity-such as in a nuclear disaster. Capacity is impacted by, for example: structural characteristics of the roads, adverse weather, and intersection control. (NUREG/CR-70002)

7. The ETE Fails To Account For Chronically Bad Traffic over Summer Week- Ends & Special Events that increases travel times

The ETE fails to consider the truly bad traffic jams that occur in the region. During the 2013 July 4th week-end, for example, traffic backed up on Cape Cod for 25 miles ahead of the Sagamore Bridge, and it took as long as eight hours to drive from the Cape to Boston, traveling

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[8] http://www.town.duxbury.ma.us/Public_Documents/DuxburyMA_EMA/Dux-06%20School%20Department_July2010.pdf

over the major evacuation routes for the EPZ.[9] If a nuclear accident occurred during such traffic congestion, all the traffic would need to be rerouted over alternative roads that are not included in the current estimate of evacuation times. During July and August of 2012, inbound traffic over the Sagamore and Bourne Bridges on Friday, Saturday and Sundays averaged 230,000 vehicles on a typical summer week-end and 255,000 on the July 4th weekend[10]. Those same vehicles exit the Cape over evacuation routes for the EPZ. The ETE simply modeled one special event, Plymouth's 4th of July celebration. Even there, the ETE underestimated demand by assuming most family members would drive together to the celebration in one car; it is more likely that teenage and young adult family members would drive in separate cars, adding to traffic volume.

8. ETE Assumptions about Traffic Flow during Inclement Weather & Peak Commuter/Holiday Traffic Are Not Credible

The ETE Evacuation Scenarios included:

Table 6-2. Evacuation Scenario Definitions

Scenario	Season[1]	Day of Week	Time of Day	Weather	Special
1	Summer	Midweek	Midday	Good	None
2	Summer	Midweek	Midday	Rain	None
3	Summer	Weekend	Midday	Good	None
4	Summer	Weekend	Midday	Rain	None
5	Summer	Midweek, Weekend	Evening	Good	None
6	Winter	Midweek	Midday	Good	None
7	Winter	Midweek	Midday	Rain	None
8	Winter	Midweek	Midday	Snow	None
9	Winter	Weekend	Midday	Good	None
10	Winter	Weekend	Midday	Rain	None
11	Winter	Weekend	Midday	Snow	None
12	Winter	Midweek, Weekend	Evening	Good	None
13	Summer	Weekend	Evening	Good	Plymouth 4th of July Fireworks
14	Summer	Midweek	Midday	Good	Roadway Impact – Lane Closure on Rt. 3 NB

[1] Winter assumes that school is in session (also applies to spring and autumn). Summer assumes that school is not in session.

[9] http://www.bostonglobe.com/metro/2013/07/08/cape-going-nowhere-holiday-traffic-nightmare-spills-over-into-monday/gRG9bQkdv0h7B4E8Chs13N/story.html
[10] http://www.capecodtransit.org/downloads/CapeFLYER.pdf

a. Inclement Weather: The EPZ ETE assumes that roads are passable and that "approriate agencies are plowing roads as they would normally" (ETE, 2.2) so that area roads used in an evacuation would be able to handle 80% of the good weather highway capacity in the event of snow and 90% in the event of rain. The report claims that, "it is reasonable to assume that the highway system will remain passable - albeit at a lower capacity-under the vast majority of snow conditions;" and that snow plow crews would be avialable and the clearing efforts would be highly effective. In the February 8-9, 2013 blizzard road conditions were so severe that the Massachusetts Governor placed a ban on driving. [11] During that storm Duxbury Beach was overtopped and the beach road used for evacuation by Gurnet-Saquish and Duxbury beach residences were impassable. During Hurricane Sandy in late 2012, storm surge overtopped Plymouth Beach and led to the closure of 3A, one of the evacuation routes from Plymouth.[12] Severe weather conditions are one of the triggers of a nuclear accident. Last, it is very likely that snow operators will not appear for duty, but instead will evacuate with their family. They have not been surveyed to determine their response. It should be done, and anonomously.

Further, as Table 6-2 shows, evacuation scenarios modeled traffic flow during rain and snow midday. KLD avoided peak traffic periods and chose a time period when it is more likely that snow plow crews were at work and best able to clear roads. Also, the ETE failed to account for fog in Pilgrim's coastal region.

b. Peak Travel Times Avoided:

The ETE fails to precisely define "Time of day." From the general description, it is clear that peak travel times for commuters and summer travelers are avoided in its estimates. The Pilgrim area is a tourist magnet for visitors to its beaches, ponds/lakes, forests and historic sites.

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[11] http://www.gazettenet.com/home/4359892-95/ban-inches-road-snow

[12] http://www.wickedlocal.corn!plymouth/topstories/x1272748828/HURRICANE-SANDY-Dodged-that-bullet#axzz2MKf6CMsI

Summer visitors get an early morning start. Midday traffic is the lightest and that is precisely when the ETE estimates were made. Summer evening traffic is at its peak in early evening, before or very shortly after an early dinner; but the ETE fails to say when in the evening they modeled traffic. In "Winter" or non-summer seasons, midday is modeled, avoiding peak commuter traffic; and again "evening" is not defined by providing the hour.

9. The ETE's Estimates For Specific Roadway Capacity Are Not Credible

The ability of the road network to service the demand is a major factor in determining how rapidly the population can evacuate. The ETE estimates are not credible. For example:

a. Two-Lane Roads: The ETE assumption that on rural roads, narrow lanes and shoulders will not interrupt the free flow of traffic is absurd. It overlooks that rural 2-lane roads have numerous smaller roads and driveways feeding into them that will slow traffic.

b. Multi-Lane Highways: Route 3 North is the main evacuation route for Duxbury Beach, Saquish Neck, Gurnet Point, Clark's Island (sub-area 4); Duxbury (sub-area 9) and Marshfield, subarea 10. Route 3 south is the major evacaution route for Plymouth subareas 1,2,3,5 and 6. When route 3 was completed in 1963 it was designed to carry 76,000 cars daily; it is way over capacity now.[13] The population evacuating over that route in a nuclear disaster will far exceed the design capacity.

c. Choke Points, Not Established: Roadways have choke points under a variety of conditions. The ETE fails to establish and record the specific choke point capacity for each roadway used in a radiological emergency at Pilgrim Station.

---

[13] *Route 3 widening project is back on track: Weymouth-Duxbury stretch in Romney's transportation plan*, Patriot Ledger, Tom Benner, March 11, 2005

10. Emergency Personnel: The ETE Assumes, Absent Factual Support That Emergency Personnel Will be Available In Sufficent Number To Assure Timely Traffic Flow

Availability of emergency personnel are important for intersection control and in general to assure traffic flow. The ETE provides no basis to support that emergency personnel will be available in sufficient number to assure the timely movement of traffic in an evacuation during a radiological disaster at Piglrim Station. An anonomous survey of respondents is required to provide reasonable assurance that sufficient personnel would be available. Recognizing the effect of federal, state and local budget cuts on personnel, it also is necessary to see an actual list, a real total count, of emergency personnel available in the pertinent departments.

C. INACCURATE ASSUMPTIONS TRIP GENERATION TIMES (ETE, 5)

Development of ETEs (NUREG/CR-70003, Ch. 4) includes trip generation time, evacuation modeling, and estimates of evacuation times. Pilgrim's ETE underestimated each.

11. Trip Generation Time Relied On Flawed Telephone Survey & Assumptions

The ETE followed Federal Guidelines (NUREG/CR-70002) to estimate the elapsed time the public will take to get ready to evacuate. ETE's estimates are not credible because: KLD based its data on the telephone survey of only EPZ residents, and failed to tell even resident respondents that the questions were for a nuclear emergency at Pilgrim, and made a number of incorrect assumptions. Incorrect assumptions and data artificially resulted in the not credible conclusion that a complete evacuation of the EPZ would occur in six hours.

12. The KLD incorrectly assumed a rapidly escalating accident and that mobilization of the general population will commence within 15 minutes after siren notification. (ETE, 2-5, 5-1)

This ignores provisions in the EPZ Radiological Emergency Plan and Standard Operating Procedures that notifies segments of the general public at the Alert and/or Site Area stage of the

emergency, prior to the General Emergency. See, for example, the duties of the Harbormaster at the Alert in the Town of Duxbury's procedures where beaches are closed and boaters advised to come ashore.[14] It is highly probable that information from these advisories will spread to other members of the public with today's readily available rapid communication systems, and that mobilization will begin earlier than the General Emergency. Unplanned early mobilization of the population is likely to lead to a chaotic and unplanned evacuation of the population resulting in accidents and overall time delays, acerbated by unmanned traffic control points until after a General Emergency called.

13. The ETE Failed to Consider Impact Delayed Staffing Traffic Control Points on ETE

"Traffic Control Points (TCP) within the EPZ will be staffed over time, beginning at the Advisory to Evacuate." (ETE, 2-5) Therefore TCPs are assumed to be not in place when actual evacuations begin prior to the advisory to evacuate, which will not be until a General Emergency. The function of the TCPs is to "facilitate the movement of all (mostly evacuating) vehicles at the location." Their absence when mobilization occurs before the General Emergency is assured to delay evacuation times.

14. The ETE Incorrectly Assumed That 25% Of The EPZ Households Will Await The Return Of A Commuter Prior To Evacuating Underestimating Vehicles

Flawed data on mobilization times that resulted from the telephone survey included, for example, assumptions about commuters. "The ETE assumed that 65% of the households in the EPZ have at least (1) commuter; 38% of those households with commuters will await the return of a commuter, prior to beginning their evacuation trip. Therefore 25% of EPZ households will await the return of a commuter, prior to beginning their evacuation trip." (2-5) It should have been obvious to KLD that in a radiological emergency households with commuters are not going to delay evacuation, until a parent gets home, especially considering the lengthy commute times many workers experience daily under normal traffic conditions. It defies reason, for example, to assume a husband or wife would drive more than 35 miles back to Duxbury from Boston towards the "eye of the storm" to evacuate together with the family. Reception Centers, outside the EPZ,

[14] http://www.town.duxbury.ma.us/Public_Documents/DuxburyMA_EMA/Dux-11%20Harbormaster_July2010.pdf

function in part is to reunite family members. This ludicrous assumption incorrectly reduces vehicle use, spreads out the number of vehicles on the evacuation routes at one time to make the ETEs appear timely.

15. The ETE Incorrectly Assumes That 50% of the Transportation Dependent Population Will Rideshare (ETE, 2-6)

Based on the telephone survey that did not tell respondent that it was about a nuclear disaster, the ETE incorrectly concluded that 50% of the transportation dependent (those without vehicles at the time of the evacuation) would rideshare, again underestimating traffic load. It is not realistic to assume 50% will rideshare because that 50% figure does not account for the facts that neighbors may not be at home in the event of an emergency to be able to offer a ride; it does not consider that evacuees will fill their vehicle with family, pets and some household items so that there would not be space for others. It does not consider the population's natural motivation is a radiological disaster, especially post Fukushima, is to get out as soon as possible without surveying neighbors in need of assistance; and it does not consider an overloaded phone system where it would not be possible to call a neighbor for a ride. There will be more traffic, congestion, on the roads because fewer than 50% are likely to rideshare. Those needing a ride will have to wait for busses to arrive from outside the EPZ increasing the overall ETE.

16. The ETE Incorrectly Assumes Timely Evacuation of Transportation Dependent

The ETE acknowledges that a <u>second wave of bus drivers</u> will be required to transport the schools and special facilties. The model for the second-wave for Duxbury; for example, assumes that the bus heads back from the Reception Center after 15 mintes, returns to the EPZ, and completes the second trip in 79 minutes (ETE, pg., 8-38) The times underestimate what will occur in reality. They ignore the time required to decontaminate the driver and bus; time to find

substitute drivers, if even possible in a nuclear disaster; time to find substitute busses and their mobilization times; and the willingness of driver to return to a contaminated area in a nuclear disaster. Bus drivers, like snow plow and tow truck operators and emergency workers should be anonomously surveyed to determine what fraction who will choose to stay with their families in a nuclear disaster and thus not be available in an emergency. Absent such a survey, there is no reasonable basis to assume that all will show up and will go back for a second trip in a nuclear emergency.

17. The ETE Assumptions About Mobilization Times for School Population & Special Facilities are Not Credible

School Population

(1) The ETE incorrectly assumes that parents will not attempt to pick up children from schools and that instead the student population will be transported by busses and met by family/guardians at the Host Facilities. The assumption is not supported by Pilgrim's Radiological Emergency Procedures. Those procedures recognize that parents will in fact try to pick up their children and students with vehicls will evacuate themselves with siblings, if appropriate. The Duxbury School Standard Operating Procedure, for example, provides explicit procedures for parent pick-up in a radiolgoical emergency. More vehicles than estimated will be on the roadways, slowing ETEs.

(2) The estimation of trip generation for the School population is ludicrous. The ETE for Duxbury assumes that the average speed of school buses from the EPZ to the Reception Center is 40:35 in the rain, and 30 minutes in the snow assuming that 40 mph is the speed limit on state roads. It is absurd to suggest that in a radiological disaster the speed limit would be achievable; KLD arrives at these times by making a host of equally ridiculous assumptions reviewed in this petition.

(3) The ETE acknowledges that a second wave of bus drivers will be required to transport the schools and special facilties, discussed above at 11, e.

Special facility populations- hospitals, nursing homes, group homes (ETE, 8-10)

(1) The estimation of trip generation for these populations are equally ludicrous. In the ETE's Duxbury data for medical facilities, for example, the assumed load time for patients is *one* minute per patient; and the estimated travel time is only to the EPZ boundary, quite unlike the school population that estimates travel time to the host facility/reception center. If the time were modeled to the host medical facility, as it should, ETEs would escalate.

(2) The analysis for the second wave of drivers is flawed as discussed above at 16.

18. The EPZ ETE Assumptions About Trip Generation For Populations On Boats Are Not Credible (ETE, 5-18)

The ETE incorrectly assumes that boaters will return to marinas within the mobilization time for transients in the EPZ (15 minutes). This ignores the time required for sail boats without motors to get back to their moorigns and ashore and the effect of low tides. KLD 2004 (section 5-11) in contrast found at 15 minutes only 15% of those on boats were notified; and at 15 minutes only 17% of the baoters were ready to evacuate. It took 60 minutes for 100% to be ready to evacuate. The boating population has increased substantially since 2004; it makes no sense that the times to evacuate gone down.

19. The ETE ignores the impact of voluntary evacuations from Cape Cod that would have a large impact on traffic in the EPZ; and ignores the effect of voluntary evacuations within the EPZ and shadow evacuation that would slow EPZ evacuation times.

The ETE assumes a rapidly escalating emergency, where a General Emergency evacuation order is the first advisory issued. The ETE acknowledges that in a more slowly

developing emergency many residents may voluntarily choose to evacuate earlier at the Alert or Site Area emergency. This is likely because at the Alert or Site Area stage, public parks and beaches are closed at the Site Area and boaters advised to get off the water. This will result in the publci knowing about serious problems at Pilgrim and likely to choose to "Get out of Dodge."

The ETE fails to consider the impact of residents on Cape Cod evacuating voluntarily in that situation. Such an early shadow evacuation that necessarily must use the Sagamore or Bourne Bridge to exit the Cape and would add to the congestion faced by residents evacuating from the EPZ. Portions of Plymouth (Subareas 1,2,3, or 5) evacuation route crosses where Cape traffic arrives on the mainland from the Sagamore and Bourne Bridges[15]. Likewise an early voluntary and shadow evacuation of residents inside the EPZ and a larger percent of the population outide the EPZ will clog the evacuation routes upstream meaning that those downstream or most a risk will experience delays in evacuation times – much like a cork placed in a bottle.

III. CONCLUSION

Faulty assumptions in KLD's ETE for Pilgrim Station show that it is not credible and that there would be much higher levels of congestion, and much longer evacuation times due to far larger voluntary and shadow evacuations, higher transient and worker population,; poorer road conditions in inclement weather than modeled, slower trip generation estimates; and likely fewer emergency workers than KLD estimated. These would lead to a significant lengthening of the time required for the EPZ to evacuate. Absent an honest and credible ETE, the population does

[15] http://www.mass.gov/eopss/docs/mema/nuclear/2013-pilgrim-nuclear-calendar.pdf

not have reasonable assurance. Pilgrim should not be operating until a new ETE with realistic evacuation time estimates based on credible assumptions, including a telephone survey informing repondents that it is for a radiological emergency at Pilgrim Station, is developed and reviewed by the EPZ Emergency Management Agencies, MEMA and the public.

Several months ago, Judge Rosenthal of the ASLB accurately said that, with one possible exception, the NRC had not granted a section 2.206 petitioner the substantive relief it sought for at least 37 years. Judge Rosenthal concluded that, "where truly substantive relief is being sought (i.e., some affirmative administrative action taken with respect to the licensee or license), there should be no room for a belief on the requester's part that the pursuit of such a course is either being encouraged by Commission officialdom or has a fair chance of success."[16]

We truly hope that Judge Rosenthal will be proven wrong and this petition will be granted.

Respectfully submitted on behalf of the Petitioners,

Mary Lampert
Pilgrim Watch, Director
148 Washington Street
Duxbury, MA 02332
Tel 781-934-0389
Email: mary.lampert@comcast.net
August 30, 2013

[16] Memorandum And Order (Denying Petitions For Hearing), LBP-12-14, July 10, 2012, Additional Comments of Judge Rosenthal (See NRC's EHD Docket EA-12-05-/12-51)

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APPENDICES

KLD ENGINEERING, P.C.

Memo

To: John Giarrusso

From: Chris Chaffee

CC: Jack Priest, Mike Slobodien, Kevin Weinisch

Date: July 25, 2013

Re: Cape Cod Telephone Survey Results

TELEPHONE SURVEY

1. Introduction

At the request of the Massachusetts Emergency Management Agency and Entergy, KLD has conducted a telephone survey to obtain demographic information about Cape Cod residents regarding emergency planning. This memo documents the telephone survey results.

The survey was designed to elicit information from the public concerning household demographics and reactions during emergencies. Information will be included in the Cape Cod Traffic Study final report and is encouraged to be used by emergency planners.

2. Survey Instrument and Sampling Plan

A draft of the survey instrument was submitted to stakeholders. After receiving comments, it was modified accordingly prior to conducting the survey. Attachment A presents the final survey instrument used in this study.

The survey sampling plan was developed by taking representative samples from each zip code in Cape Cod, proportional to the zip code's population. The population estimate and number of households in each area were determined by overlaying Census data and Cape Cod's boundary using GIS software. The proportional number of desired completed survey interviews for each area was identified, as shown in Table 1. The sample size of 500 **completed** survey forms yields results with a sampling error of approximately ±4.4% at the 95% confidence level.

The completed survey adhered to the sampling plan.

EDWARD J. MARKEY
MASSACHUSETTS

COMMITTEES:

COMMERCE, SCIENCE, AND TRANSPORTATION
SMALL BUSINESS AND ENTREPRENEURSHIP
FOREIGN RELATIONS

CHAIRMAN

SUBCOMMITTEE ON INTERNATIONAL DEVELOPMENT AND FOREIGN ASSISTANCE, ECONOMIC AFFAIRS, INTERNATIONAL ENVIRONMENTAL PROTECTION, AND PEACE CORPS

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August 16, 2013

Leo P. Denault
Chief Executive Officer
Entergy Corp.
639 Loyola Avenue
New Orleans, LA 70112

Dear Mr. Denault:

We are writing to raise serious concerns regarding Entergy's draft estimate of the time it would take to evacuate residents around Plymouth, Mass. in the event of a nuclear accident at the Pilgrim Nuclear Power Station, and to urge you to direct that all necessary revisions be made before finalizing this document. We are also concerned that the current draft completely ignores the potential need for residents of Cape Cod to evacuate and makes highly unrealistic assumptions about how quickly people living nearest the reactor could evacuate. Entergy's document, as it is currently drafted, is simply non-credible and inadequate, and fails to sufficiently protect those living near the reactor or those located on Cape Cod, who could find themselves trapped and unable to evacuate at all for hours or longer by traffic congestion and/or bridge closures. Entergy should go back to the drawing board and create a plan that properly takes into account the realities of weather, traffic, human behavior and other factors into account. Right now, such a real-world plan does not appear to exist.

In the event of a nuclear accident, it is critical to quickly evacuate people in the immediate vicinity of the nuclear power plant who could potentially be exposed to radiological materials in the air they breathe. Accordingly, the Nuclear Regulatory Commission (NRC) requires detailed evacuation plans be established for the Emergency Planning Zone (EPZ), which extends ten miles from the nuclear power plant. It may also be necessary to evacuate people from a larger area to prevent their exposure to radiological materials that have settled onto the ground or are present in food and water. This larger area is typically estimated as up to 50 miles from the nuclear plant, depending on wind, the severity of the accident, and other factors. After the nuclear melt-downs at the Fukushima Daiichi Nuclear Power Plant in Japan, the NRC recommended American citizens located within 50 miles of the plant evacuate[1].

Cape Cod is a unique geographical area, with over 200,000 permanent residents and as many as 300,000 vacationers in the summer. While none of the Cape falls within the 10 mile

[1] http://articles.washingtonpost.com/2011-03-16/national/35207282_1_fukushima-daiichi-fuel-rods-spent-fuel-pool

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 21, 2015

Mary Lampert
Pilgrim Watch, Director
148 Washington Street
Duxbury, MA 02332

Dear Ms. Lampert:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am responding to the petition that you submitted pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 2.206 "Requests for action under this subpart." Your petition dated August 30, 2013[1], as revised on November 22, 2013[2], requested the NRC to take enforcement-related action against the Pilgrim Nuclear Power Station (Pilgrim) renewed operating license to assure that (1) Pilgrim's Radiological Emergency Plan and Standard Operating Procedures/Guidelines are based on accurate and credible evacuation time estimates (ETEs), and (2) Entergy Nuclear Operations, Inc. (Entergy, the licensee) has the means to provide early notification and clear instruction to the populace within the plume exposure pathway emergency planning zone (EPZ).

You stated that this request is primarily based on the following documents:

(1) Pilgrim "Development of Evacuation Time Estimates" Final Report dated December 18, 2012[3], prepared by KLD Engineering, P.C. (KLD);

(2) A July 25, 2013, memo from KLD to the Massachusetts Emergency Management Agency regarding the Cape Cod residents demographics and reactions during emergencies telephone survey results;

(3) An August 16, 2013, letter from Senator Markley to Entergy regarding the draft ETE for residents around Plymouth, MA in the event of an accident at Pilgrim; and

(4) The Town of Duxbury, MA Emergency Management Agency telephone survey regarding an emergency siren test.

You stated that the ETE's fundamental assumptions and data were flawed, which explained the ETE's conclusion that even in the worst case scenario, everyone in the plume exposure pathway EPZ will be evacuated in about six hours. You discussed the results of the Cape Code survey and maintain that it was not properly used to determine Pilgrim's ETE. You also discussed the results of the Town of Duxbury Emergency Management Agency telephone survey and asserted that the survey shows that Entergy does not have the means to provide early notification and clear instruction to the populace within the plume exposure pathway EPZ.

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1 Agencywide Documents Access and Management System (ADAMS) Accession No. ML13267A234.
2 ADAMS Accession No. ML14223A090.
3 ADAMS Accession No. ML13023A031.

Background

On November 19, 2013, you addressed the NRC Petition Review Board (PRB) via teleconference. The official transcript of the teleconference is publicly available in ADAMS[4]. During the teleconference, you alleged that Mr. Thomas White of Entergy made a statement regarding the quality of the Pilgrim ETE. You submitted a written statement to this effect on January 21, 2014[5].

On February 20, 2014, the PRB met internally to review the petition and make an initial recommendation. The PRB's initial recommendation was to reject the petition, in accordance with MD 8.11 Handbook Part III, paragraph C.2, "Criteria for Rejecting Petitions Under 10 CFR 2.206," because the petitioner raised "issues that have already been the subject of NRC staff review and evaluation either on that facility, other similar facilities, or on a generic basis, for which a resolution has been achieved, the issues have been resolved, and the resolution is applicable to the facility in question."

On April 9, 2014[6], you were informed of the PRB's initial recommendation and offered a second opportunity to address the PRB.

On April 18, 2014[7], you requested a copy of the NRC's request to the Southwest Research Institute – Center for Nuclear Waste Regulatory Analyses (Center) for review of the Pilgrim updated ETE (December 2012) and the comments provided by the Center in response to the NRC request.

On April 25, 2014[8], the NRC informed you that those documents were internal correspondences and contained pre-decisional information used as input for consideration in the NRC staff's final decision. Therefore, the documents were profiled as non-public records.

On June 12, 2014, you addressed the PRB to discuss the PRB's initial recommendation via teleconference. The official transcript of the teleconference is publicly available in ADAMS[9].

On June 12[10], August 11[11], and September 3, 2014[12], you provided supplements to the petition.

On September 10, 2014, the PRB met internally to review the supplements to the petition and determined a final recommendation.

On December 3, 2014[13], you provided an additional supplement to the petition, which was reviewed by the PRB.

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4 ADAMS Accession No. ML14141A087.
5 ADAMS Accession No. ML14056A312.
6 ADAMS Accession No. ML14129A245.
7 ADAMS Accession No. ML14120A016.
8 ADAMS Accession No. ML14129A200
9 ADAMS Accession No. ML14223A088.
10 ADAMS Accession No. ML14167A079.
11 ADAMS Accession No. ML14224A568.
12 ADAMS Accession No. ML14251A068.
13 ADAMS Accession No. ML14338A180.

Discussion

An ETE is one of the inputs used by the licensee to inform the development of protective action recommendation strategies within the plume exposure pathway EPZ, which is an area with a radius of about 10 miles around a nuclear power plant. State and local emergency management officials may also use the ETE as an input for developing traffic management plans to support an evacuation. The NRC and FEMA have not established regulatory requirements for actual evacuation times. As stated above, the ETE is used as an input for planning an evacuation, and it is not a surrogate for what minimum evacuation time must be achieved, in the highly unlikely event that an evacuation was required.

The NRC approves a licensee's initial ETE during the licensing process. Appendix E to 10 CFR Part 50 requires licensees to submit subsequent ETE updates to the NRC, but does not require NRC approval as a licensing action. Per the Statement of Consideration for the Emergency Preparedness Rule issued in November 2011, the NRC's review of a submitted ETE update will be limited to a "completeness review," performed as an inspection activity to verify consistent application of the ETE guidance contained in NUREG/CR-7002, "Criteria for Development of Evacuation Time Estimate Studies." This review does not constitute formal NRC approval, and the updated ETE remains subject to future NRC inspection.

In December 2012, Entergy submitted an ETE update for Pilgrim. The NRC staff contracted the Southwest Research Institute – Center for Nuclear Waste Regulatory Analyses (Center) to conduct the completeness review of the ETE update. The completeness review performed by the Center served as a technical feeder/input into the NRC's emergency preparedness inspection activities. The NRC staff reviewed the comments provided by the Center and found that no further actions were required. The results were documented in Section 1EP4 of the NRC's Inspection Report 05000293/2013002[14], dated May 8, 2013.

As stated in the Executive Summary to NUREG/CR-7002, "It is important to use the information found in approved emergency plans when developing an ETE study to ensure that the results represent the expected response from authorities." Per the Memorandum of Understanding between the Federal Emergency Management Agency (FEMA) and the NRC, contained in Appendix A to 44 CFR 353, FEMA has responsibility for determining the adequacy and capability of implementing offsite plans and communicating those findings and determinations to the NRC. Based on FEMA's review of State and local plans for a radiological emergency at Pilgrim and its evaluation of an exercise that was conducted on November 16-17, 2010, FEMA has determined that State and local preparedness remains adequate to protect the health and safety of the public living in the vicinity of Pilgrim and that appropriate measures can be taken offsite in the event of a radiological emergency at Pilgrim. The results were documented in the Final Report for the Pilgrim Plume and Ingestion Exercise[15].

The FEMA Radiological Emergency Preparedness (REP) Program Manual provides instructions for assessing activities associated with alerting and notifying the public in order to ensure that the capability exists to notify the public in a timely manner following a protective action decision by authorized offsite emergency officials. The ability to promptly alert and notify the public is evaluated by FEMA during biennial exercises. The ETEs are developed using existing State

14 ADAMS Accession No. ML13129A212.
15 ADAMS Accession No. ML11223A279.

and local radiological emergency plans and are not intended to validate the notification criteria. As such, questions relating to any validation of alert and notification criteria are outside the purview of an ETE.

In the Statements of Consideration for the 1980 Emergency Preparedness Rule [45 Federal Register 55407, August 19, 1980], the Commission described its rationale for the public alerting and notification system requirement. In that discussion, the Commission stated, in part, "The Commission recognizes that not every individual would necessarily be reached by the actual operation of such a [public alerting and notification] system under all conditions of system use." Given the inter-connected nature of society today, it is reasonable to assume that the population will avail themselves of any battery powered communication capability (e.g., smartphones, tablet computers, etc.) available to the home to obtain news and information and in this manner may also be alerted to the nuclear power event. In many communities, residents with smartphones can register for emergency alert notifications. It is also reasonable to assume that persons who received the direct notification will call, text, or e-mail nearby friends and relatives, further extending the direct notification informally. As such, the survey data cited by Pilgrim Watch does not invalidate the assumptions and findings of the Pilgrim ETE.

A number of issues discussed in the petition, including control of traffic in and out of the EPZ, transient populations, and shadow evacuations, are addressed in the ETE, and are used to inform ETEs from the EPZ. The NRC reviewed the petition information and the Pilgrim ETE, and found the ETE was based on multiple inputs including, FEMA-approved State and local radiological emergency plans. Additionally, the PRB determined that the assumptions used by the licensee were consistent with the guidance contained in NUREG/CR-7002.

Summary

The PRB reviewed the information you provided in the petition, in its supplements, and during the November 19, 2013, and June 12, 2014 teleconferences. The PRB also conducted a subsequent completeness review of the December 2012 updated ETE for Pilgrim. As a result of this review, it was found that the Pilgrim ETE was developed using FEMA-approved State and local REP plans and was consistent with the guidance contained in NUREG/CR-7002.

The assumptions for the development of an ETE are addressed by NRC's guidance, NUREG/CR-7002, and in Supplement 3, "Guidance for Protective Action Strategies," to NUREG-0654/FEMA-REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," prepared by FEMA and the NRC. As a result, the PRB's final determination is the same as its initial recommendation. In accordance with NRC Management Directive 8.11, Part III, C.2, "Criteria for Rejecting Petitions Under 10 CFR 2.206," the PRB's final determination is to reject your petition because the issues that you raised have already been reviewed, evaluated, and resolved by the NRC. Additionally, attachments 1, 2, and 3 of the June 12, 2014, supplement will be reviewed and evaluated within a different NRC process.

Thank you for bringing these issues to the attention of the NRC.

Sincerely,

Allen D. Howe

Allen Howe, Deputy Director
Division of Inspection and Regional Support
Office of Nuclear Reactor Regulation

Docket No. 50-293

cc: Licensee (w/copy of incoming 2.206 request)
Distribution via Listserv

Thank you for bringing these issues to the attention of the NRC.

Sincerely,

/RA/

Allen Howe, Deputy Director
Division of Inspection and Regional Support
Office of Nuclear Reactor Regulation

Docket No. 50-293

cc: Licensee (w/copy of incoming 2.206 request)
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ADAMS Accession No. ML14318A418 ***Via email**

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DATE	12/15/2014	11/20/2014	11/20/2014	12/17/2014
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Wang, Weidong

From: Paul Gunter <paul@beyondnuclear.org>
Sent: Monday, July 06, 2015 2:53 PM
To: Wang, Weidong
Subject: [External_Sender] Document in support of public comments on CPRR and ACRS Subcommittee on Fukushima
Attachments: nrdc_hydrogen-generation-safety-report_2014.pdf

Hello Mr. Wang,

Please find attached the Natural Resource Defense Committee report "Preventing Hydrogen Explosions in Severe Nuclear Accidents: Unresolved Safety Issues Involving Hydrogen Generation and Mitigation" in support of my brief comments before the ACRS subcommittee on the Fukushima Daiichi nuclear accident.

Thank you,
Paul
--
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NRDC REPORT

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Preventing Hydrogen Explosions In Severe Nuclear Accidents:

Unresolved Safety Issues Involving Hydrogen Generation And Mitigation

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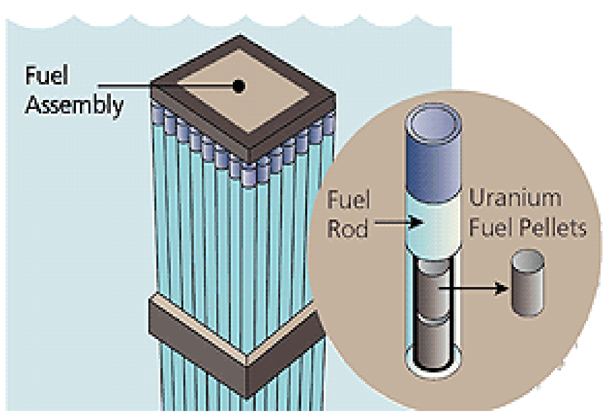
I. EXECUTIVE SUMMARY

As demonstrated during the March 2011 severe nuclear accident in Fukushima, Japan, accumulation and subsequent detonation of hydrogen gas produced by an overheated nuclear core reacting with steam can breach a reactor's containment structures and result in widespread radioactive contamination.[1] The gas is initially generated by the rapid oxidation of the zirconium alloy tubes ("fuel cladding") that surround the low-enriched uranium fuel pellets in commercial power reactors (Figure 1).

When the fuel cladding enters a certain temperature range well above its typical operating temperature, the zirconium-steam reaction becomes "autocatalytic," meaning that it propagates via self-heating from the chemical reaction itself. This produces large quantities of hydrogen in a brief period. This intense reaction also causes the fuel cladding to erode and breach, which releases harmful levels of radionuclides into the reactor vessel. The fuel cladding is the first line of defense among multiple barriers—the reactor vessel, a steel and/or reinforced concrete "containment," and a further, secondary containment in some designs[2]—that are intended to prevent release to the environment of the biologically hazardous radionuclides produced by nuclear fission (see Figure 2). In some accident scenarios, over-pressurization of the reactor vessel can be exacerbated by the buildup of hydrogen from the zirconium-steam reaction, causing seals at the multiple penetrations of the vessel required for reactor monitoring and control to leak hydrogen into the containment.

Figure 1: Structure of a Uranium Fuel Assembly



Source: NRC

To protect the integrity of the reactor's cooling system, pressure relief valves are designed to open automatically, resulting in discharge of radioactively contaminated steam and hydrogen gas into the containment. In older boiling water reactor (BWR) designs, this discharge is initially into the "pressure suppression pool" or "wetwell" portion of the primary containment.[3]

In the March 2011 Fukushima Daiichi accident—in which the cores of three GE-designed boiling water reactors lost all cooling and melted down—hydrogen leaked from the primary containments into the reactor buildings. The hydrogen accumulated in the reactor buildings and detonated, causing large releases of harmful radionuclides that contaminated a wide area and prompted the evacuation of some 90,000 people. A smaller hydrogen explosion also occurred in the March 1979 Three Mile Island Unit 2 (TMI-2) accident—a partial core meltdown of a pressurized water reactor (PWR)—that did not breach the containment.

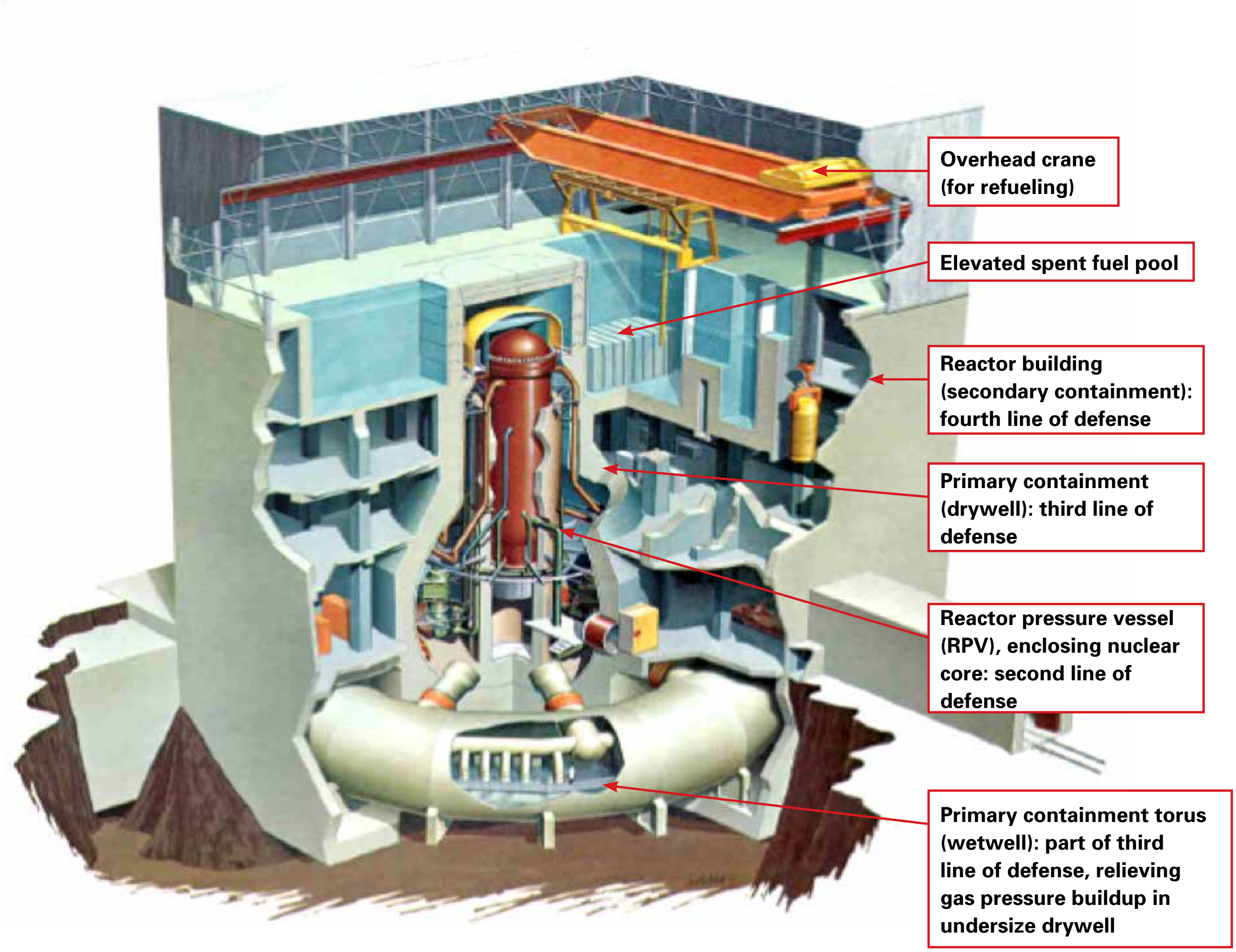
The U.S. Nuclear Regulatory Commission (NRC) has a checkered history when it comes to requiring measures that would effectively reduce the risk of hydrogen explosions in the event of a severe accident at a U.S. nuclear power plant. This regulatory lapse is rooted in the history of the development of commercial nuclear power in the United States, when the NRC's predecessor agency, the Atomic Energy Commission (AEC), had a dual mandate: both to *promote* and to *regulate* commercial nuclear power.

As a consequence of this internal conflict of interest, rather than consult independent scientific and technical institutions, the AEC entrusted two companies that designed nuclear reactors—Westinghouse and General Electric (GE)—with the mission of demonstrating that in a large-pipe-break loss-of-coolant accident (LOCA), the emergency core-cooling systems for their respective reactor designs would in fact prevent overheating of the core, and hence prevent the generation of large quantities of explosive hydrogen gas.

In response to the TMI-2 partial meltdown in 1979, the NRC revised its regulations regarding the control of hydrogen in an effort to help prevent hydrogen explosions in severe nuclear accidents. In 1981, the NRC issued a requirement that GE-BWRs with the small-volume Mark I and somewhat larger Mark II containments operate with their atmospheres inerted with nitrogen, to minimize the risk of hydrogen combustion. In 1985, the NRC required installation of hydrogen igniters—systems to burn off leaked hydrogen before it accumulates

Figure 2: Cutaway View of a GE Mark I Boiling Water Reactor (BWR)

This is the design that exploded at Fukushima Daiichi, Japan, in March 2011.
Twenty-two units of this design are still operational in the U.S.



Source: NRC Reactor Concepts Manual, Rev. 0200

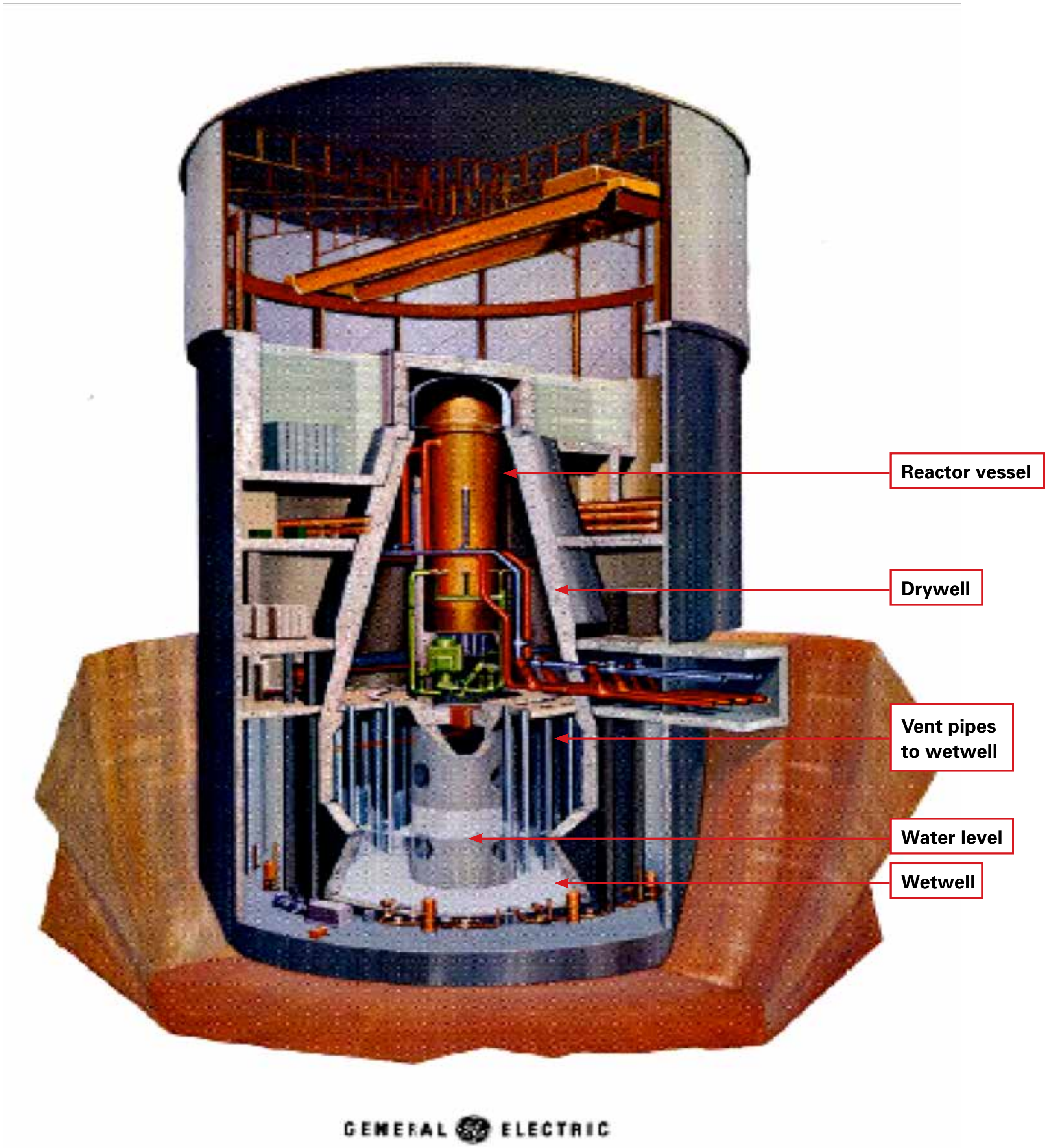
to explosive concentrations—in pressurized water reactor (PWR) "ice condenser" containments and GE-BWR Mark III containments.

By contrast, after Fukushima Daiichi's three devastating hydrogen explosions, the NRC decided to relegate investigating severe accident hydrogen safety issues to the *lowest-priority* and least proactive stage (Tier 3) of its post–Fukushima Daiichi accident response. Hence, beyond ensuring reliable containment pressure relief vents are added to obsolescent Fukushima-type reactors, it could take many years, or even decades, before the U.S. nuclear industry implements further hydrogen control measures.

Multiple technical pathways exist for minimizing the risk of hydrogen explosions in severe nuclear accidents. However, in the aftermath of the Fukushima Daiichi accident, the NRC has merely declared that severe nuclear accidents are vanishingly rare events that can be either prevented or sharply limited in scope, thereby avoiding any significant buildup of hydrogen and attendant explosion risk. The reality, however, is that merely waving a rhetorical magic wand over

Figure 3: Cutaway View of a GE Mark II BWR with Unified Concrete Drywell/Wetwell Primary Containment Design

This design is deployed at Limerick Units 1 and 2, Susquehanna 1 and 2, and Nine Mile Point 2.
The primary containment volume is only slightly larger than that of the Mark I.



Source: Containment Integrity Research at Sandia National Laboratories - An Overview, NUREG/CR-6906

the problem of hydrogen explosion risk flies in the face of a number of unresolved safety issues, including:

- experimental evidence that current reactor computer safety models do not accurately predict the onset of rapid hydrogen generation in severe nuclear accidents, and that they under-predict the rates of hydrogen generation that occur in such accidents;
- an aging fleet of U.S. reactors that will increasingly operate beyond the 40-year term of their initial licenses while facing severe competitive pressures from other electricity generation technologies, creating a perilous tradeoff between economic viability and public safety;
- the compromised ability of 40-year old containments to prevent hydrogen leakage (for example, at the seals of pipe and cable penetrations) under the elevated-pressure conditions that are expected to occur in severe accidents;
- the apparent willingness of the NRC to accede to licensee requests to relax and defer requirements for periodic containment pressurization and leak rate testing; and
- the lack of technical readiness of U.S. power reactor owners to detect and control dangerous concentrations of hydrogen in all the places where it could migrate and explode in a nuclear power plant.

We conclude that the NRC is failing to meet the statutory standard of "adequate protection" of the public against the hazard of hydrogen explosions in a severe reactor accident. Our reasons are summarized below and set forth in more detail in the body of this report.

1. NRC computer safety models underpredict the rates of hydrogen generation that have occurred in experiments simulating severe nuclear accidents.

Reports from the Oak Ridge National Laboratory (1997), the OECD Nuclear Energy Agency (2001), and the International Atomic Energy Agency (IAEA) (2011) support the conclusion that current computer safety models underpredict the rates of hydrogen generation that may occur in severe accidents when zirconium fuel cladding and other core components react with steam, especially during a re-flooding of an overheated reactor core. Unfortunately, the NRC's 2011 *Recommendations for Enhancing Reactor Safety in the 21st Century: Near-Term Task Force Review of Insights from the Fukushima Daiichi Accident* and subsequent Fukushima safety review documents do not discuss the fact that the NRC's computer safety models—such as the widely used MELCOR code developed by Sandia National Laboratories—underpredict the hydrogen generation rates that occur in severe accidents. By overlooking the deficiencies of computer safety models, the NRC undermines its own philosophy of defense-in-depth, which requires the application of conservative models. When hydrogen generation rates are underpredicted, hydrogen mitigation systems are not likely to be designed so that they can handle the hydrogen gas generation rates that would occur in actual severe accidents.

2. BWR Mark I and Mark II primary containments are especially vulnerable to overpressurization and hydrogen leaks.

In 1972, the chief nuclear safety analyst for the AEC recommended discouraging further use of the type of primary containments used in the GE-BWR Mark I and Mark II designs, claiming they were susceptible to overpressurization. One reason these containments are vulnerable is that their volumes are relatively small: typically about one-ninth and one-sixth the volume, respectively, of PWR large dry containments. In September 1989, the NRC publicly acknowledged that BWR Mark I primary containments might not be able to withstand the internal gas pressures that would build up in severe accidents. However, at the time, the NRC merely issued guidance that was not legally binding, recommending that owners of BWR Mark I designs "on their own initiative" install a "hardened vent" to the external environment for each reactor unit's doughnut-shaped wetwell—to reduce the internal gas pressure and remove decay heat in the event of a severe accident.

In the United States, the vents currently installed in each BWR Mark I wetwell (see Figure 1) do not have a standardized design, are not outfitted with high-capacity filters to prevent the release of harmful radionuclides in accidents, are not subject to NRC inspection for proper maintenance and continuing operability, and do not have an independent train of backup power sources to help ensure remote operation during a station blackout (i.e., a total loss of both grid-connected and backup alternating current power at a nuclear power plant).

As overall leak-rate tests demonstrate, GE-BWR Mark I and Mark II primary containments are not designed to prevent hydrogen leakage in accidents. These tests are legally required at U.S. nuclear power plants for determining how much radiation would be released from the containment in a "design-basis accident" (i.e., an anticipated accident in which, by design, a core melt would be prevented). In overall leak rate tests—conducted below their nominal design pressures—BWR Mark I and Mark II primary containments have been shown to leak hundreds of pounds of air per day. For example, in 1999, tests conducted at Nine Mile Point Unit 1 (a BWR Mark I) and at Limerick Unit 2 (a BWR Mark II) found that overall leakage rates at both units exceeded 350 pounds of air per day, an amount that is less than the maximum allowed leak rates. This means that in a severe accident, even if there were no damage to a primary containment, hydrogen would leak into the secondary containment (reactor building). Leak rates would increase as the internal pressure increased, and they would become even greater if the seals at the various piping and cable penetrations were damaged. (Typical BWR containments have 175 penetrations, almost twice as many as typical PWR containments.)

Figure 4: Internet Images of the Fukushima Daiichi Nuclear Power Station from the Ocean Side Before and After the March 2011 Tsunami and Hydrogen Explosions Destroyed (from Right) Units 1, 3, and 4

A plume is visible coming from a blown-out shield building panel in the side of the Unit 2 reactor, which, while still intact, also experienced a core melt.

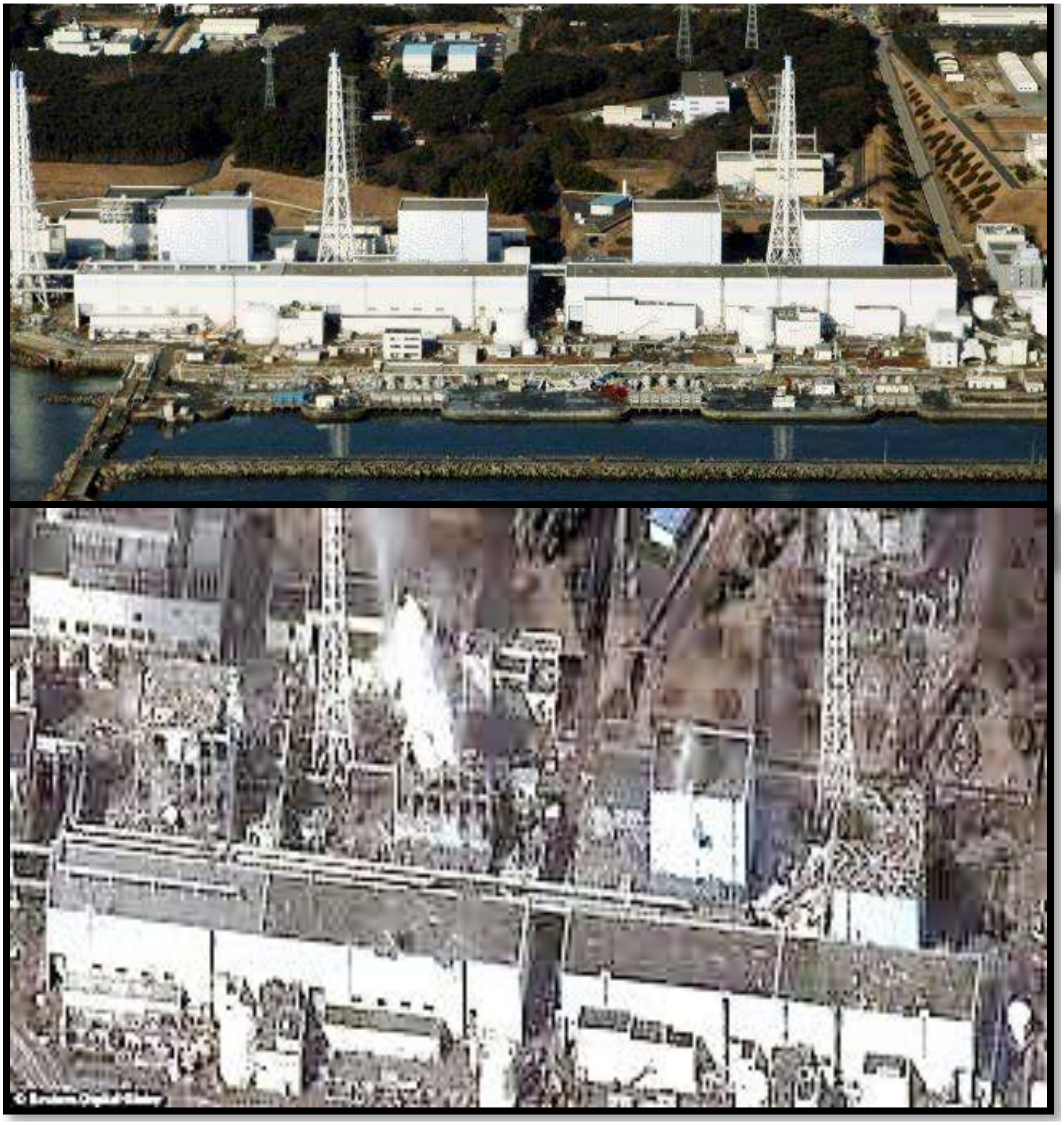


Photo credits: top, unknown; bottom, Digital Globe

3. GE-BWR Mark I and II containments perform poorly in leak rate tests, yet the NRC is planning to further relax requirements for leak rate testing.

BWR Mark I primary containments have failed a number of overall leak rate tests; for example, Oyster Creek—the oldest operating commercial reactor in the United States, which is considered to be quite similar to Fukushima Daiichi Unit 1—has failed at least five tests. In one test, Oyster Creek's primary containment leaked at a rate 18 times greater than its design leak rate; if this test was conducted at the same pressure as subsequent Oyster Creek tests, which seems likely, the primary containment leaked more than 6800 pounds of air per day. Such results raise the questions: What were the observed pre-accident leak rates—*below design pressure*—of the three primary containments that leaked hydrogen at Fukushima Daiichi? Could there have been excessive hydrogen leakage at one or more of the primary containments, without it becoming overpressurized?

Since the Fukushima Daiichi accident, the problem of hydrogen leakage from primary containments has still not been adequately addressed. Mark II primary containments must also be assessed as likely to incur hydrogen leaks in severe accidents. Nevertheless, the NRC is currently preparing to extend the intervals at which overall and local leak rate tests must be conducted to once every 15 years (from the current 10 years) and once every 75 months (from the current five years), respectively. This will only further decrease the safety margin of BWR Mark I and Mark II designs. In its safety analyses to assess extending the test intervals, the NRC overlooked the fact that BWR Mark I and Mark II primary containments are particularly vulnerable to hydrogen leakage.

In a severe accident, BWR Mark I primary containments that leak excessively in tests *conducted below their design pressure* would leak dangerous quantities of explosive hydrogen gas into secondary containments; however, the NRC does not seem concerned about these excessive leakage rates. A 1995 NRC report, NUREG-1493, concluded that "increasing allowable leakage rates by 10 to 100 times results in a *marginal risk increase*, while reducing costs by about 10 percent" [emphasis added]. And a 1990 NRC report, NUREG-1150, concluded that even if there is leakage equivalent to 100 percent of the contained gas volume per day, "the calculated individual latent cancer fatality risk is below the NRC's safety goal." But this safety goal clearly would not be achieved if leaking hydrogen were to detonate in the reactor buildings, as it did at Fukushima Daiichi.

In March 2013, the NRC asserted that "[s]ensitivity analyses in NUREG-1493 and other studies show that *light water reactor accident risk is relatively insensitive to the containment leakage rate* because the risk is dominated by accident sequences that result in failure or bypass of containment" [emphasis added]. In reality, the progression of the Fukushima Daiichi accident was indeed affected by the leakage of hydrogen gas. The evidence suggests that Unit 3's primary containment *did not fail* before hydrogen leaked into the Unit 3 reactor building and detonated. The internal pressure of Unit 3's primary containment actually *increased* after the hydrogen explosion occurred.

In a nuclear power plant accident, a mixture of hydrogen, nitrogen, and steam could leak from the primary containment; as internal pressures increase and the accident progresses, the concentration of hydrogen in the leaking mixture would increase. If there were *no damage* to the primary containment, the quantity of hydrogen that leaked (by weight) would be relatively small, because hydrogen is about one-fourteenth as dense as air. However, a secondary containment could be breached if, for example, only 20 to 40 pounds of hydrogen were to leak into it, accumulate locally, and explode.

4. Large-volume PWR dry containments, made of reinforced concrete with a steel liner, are a prominent safety feature of many U.S. nuclear power plants; however, they are not necessarily invulnerable to the effects of hydrogen explosions.

The NRC mistakenly claims that the large containment volumes of most PWRs—a reactor design found in about two-thirds of the U.S. nuclear fleet—would keep the pressure spikes from potential hydrogen explosions within their design pressures. But this claim is predicated on an uncertain and therefore misplaced assumption that hydrogen combustion would occur in the form of a "deflagration," a combustion wave traveling at a subsonic speed relative to the unburned gas.

However, when local hydrogen concentrations are greater than about 10 percent by volume, it is possible for a deflagration to transition into a "detonation," a combustion wave traveling at a supersonic speed relative to the unburned gas. Unfortunately, in a severe accident, a hydrogen detonation could occur within a PWR large dry containment if there were elevated local hydrogen concentrations, especially in the presence of carbon monoxide and high temperatures; this could cause internal pressure spikes to exceed twice the containment's design pressure.

Furthermore, a local hydrogen explosion occurring inside the containment could propel debris, such as concrete blocks from internal walls, into the containment structure at high velocities. The impact of such internally generated missiles could damage essential safety systems and severely crack a PWR's containment.

According to a 2011 IAEA report on the mitigation of hydrogen hazards in severe nuclear accidents, "no analysis ever has been made on the damage potential of flying objects generated in an explosion" of hydrogen. Yet we know from the Fukushima Daiichi accident that debris propelled from hydrogen detonations caused extensive damage to backup emergency power supplies and hoses that were intended to inject seawater into overheated reactors. Some of the debris dispersed around the site by explosions was highly radioactive, exposing personnel to higher dose rates and setting back their efforts to control the accident.

As nuclear safety expert David Lochbaum has noted, "During design basis accidents, the response of operators and workers is primarily passive—verifying that automatic equipment actions have occurred. In essence, workers are observers during design basis accidents. During severe accidents, workers get off the bench and into the game.

The keystone of [the U.S. nuclear] industry's response to Fukushima is 'FLEX,' an array of portable components moved into place by workers. Inadequate hydrogen control during a severe accident would seem to render FLEX virtually useless."[4]

5. In the presence of the quantities of hydrogen generated in severe accidents, untimely ignitions from currently installed devices for controlling the buildup of hydrogen inside some U.S. nuclear reactor containments could cause hydrogen detonations.

Hydrogen "recombiners" are devices that eliminate hydrogen by combining it with oxygen, a reaction that produces steam and heat. There are two types of hydrogen recombiners: passive autocatalytic recombiners (PARs), which operate without electric power, utilizing catalytic surfaces to facilitate the combining of hydrogen and oxygen molecules; and thermal recombiners, which are electrically powered.

In September 2003, the NRC rescinded its requirement that most types of PWRs operate with hydrogen recombiners installed in their containments, because it decided that the quantity of hydrogen that would be released in design-basis accidents is not risk-significant. Indian Point on the Hudson River near New York City is the only nuclear power plant in the United States that currently operates with PARs. The new Westinghouse AP1000 design, under construction in Georgia, South Carolina, and China, is intended to operate with only two PARs installed in its containment. The hydrogen removal capacity of a single recombiner unit is only several grams per second whereas hydrogen generation in a severe accident could range from 100 to 5,000 grams per second.

If a PWR still operates with hydrogen recombiners, there are typically only two units installed in its containment, their mission being to reduce the quantity of hydrogen generated in a design basis accident. By contrast, European PWR containments typically have 30 to 60 such devices installed, with the mission of reducing the quantity of hydrogen generated in a severe accident.

Clearly, just two recombiners would not be capable of eliminating, in timely fashion, the quantity of hydrogen generated in a severe accident. But this is not their only limitation. When hydrogen recombiners are exposed to the elevated hydrogen concentrations that occur in severe accidents, they have a tendency to malfunction and incur ignitions, which could cause a hydrogen detonation that compromised the containment. Hence, it seems that maintaining the token capacity of two recombiners actually presents a *net safety hazard*. This is especially a problem with PARs, which operators would not be able to deactivate; at least electrically powered thermal recombiners could be switched off when a hydrogen concentration reached a level at which the recombiner could incur ignitions.

The NRC requires that hydrogen igniters be installed in reactor containments that are neither inerted nor designed to withstand high internal pressures—PWR ice condenser and BWR Mark III containments. Igniters are intended to burn off hydrogen as it is generated in an accident, before it can reach concentrations at which combustion would threaten the integrity of the less sturdy containment. In a severe accident, to safely actuate hydrogen igniters, operators would need to know the local concentration of hydrogen in the vicinity of each igniter; if igniters were actuated too late—after local detonable concentrations of hydrogen built up—they could actually cause a hydrogen detonation that breached the containment.

6. The NRC has insufficient requirements for monitoring the quantities of hydrogen generated in severe accidents.

NRC rules state that in nuclear accidents, hydrogen monitors must begin to function within 90 minutes of the emergency injection of coolant water into the reactor vessel. Ninety minutes could be too late in a fast-moving accident scenario. In 2003, the NRC took the odd step of reclassifying both hydrogen and oxygen monitors (required for BWR primary containments that operate with nitrogen-inerted atmospheres) as non-safety-related equipment, meaning that the equipment does not need to have redundancy, seismic resistance, or an independent train of onsite standby power.

Furthermore, GE-BWR Mark I and Mark II designs operate with hydrogen monitors installed only in their inerted primary containments, not in their reactor buildings. In the Fukushima Daiichi accident, hydrogen from three nuclear units leaked into these buildings and exploded.

7. Operators of PWRs lack a sufficient capability to monitor the onset and progression of core degradation in the event of an accident.

This insufficient capability limits operator knowledge of when to transition from emergency operating procedures (EOPs)—intended to *prevent* core damage—to severe accident management guidelines (SAMGs)—intended to stabilize a damaged reactor core with auxiliary ad-hoc cooling measures while preventing significant off-site releases of radionuclide contamination. The operating measures appropriate to preventing core damage early in an accident are obviously not the same as those intended to contain the consequences of core damage that has already occurred while forestalling further compounding events, such as hydrogen explosions, that could result in a significant loss of containment. Not knowing which regime one is operating in could have severe consequences.

In PWRs, core-exit thermocouples—temperature measuring devices—are the primary equipment that would be used to detect inadequate core-cooling and to signal the point at which operators should transition from EOPs to SAMGs. However, data from experiments demonstrate that core-exit temperature measurements are neither an accurate nor a timely indicator of *maximum fuel-cladding temperatures* in the core, and hence an unreliable indicator of the likelihood of significant hydrogen production. In the most realistic severe accident experiment ever conducted—in which an actual reactor core was heated with decay heat

before melting down—core-exit temperatures were measured at approximately 800°F when maximum in-core fuel-cladding temperatures exceeded 3300°F.

In a severe accident, plant operators are supposed to implement SAMGs before the onset of the rapid zirconium-steam reaction, which leads to thermal runaway in the reactor core. Clearly, using core-exit thermocouple measurements in order to detect inadequate core cooling or uncovering of the core is neither reliable nor safe. For example, PWR operators could end up re-flooding an overheated core simply because they do not know the actual condition of the core. Unintentionally re-flooding an overheated core could generate hydrogen, at a rate as high as 5,000 grams per second, and the containment could be compromised if large quantities of that hydrogen were to detonate, as occurred at Fukushima.

NRDC'S RECOMMENDATIONS FOR REDUCING THE RISK OF HYDROGEN EXPLOSIONS IN SEVERE NUCLEAR ACCIDENTS

A. The NRC should develop and experimentally validate computer safety models that can conservatively predict rates of hydrogen generation in severe accidents.

The NRC needs to acknowledge that its existing computer safety models underpredict the rates of hydrogen generation that occur in severe accidents. The NRC should conduct a series of experiments with multi-rod bundles of zirconium alloy fuel rod simulators and/or actual fuel rods as well as study the full set of existing experimental data. The NRC's objective in this effort should be to develop models capable of predicting with greater accuracy the rates of hydrogen generation that occur in severe accidents.

B. The safety of existing hydrogen recombiners should be assessed, with the use of PARs potentially discontinued until technical improvements are developed and certified.

Experimentation and research should be conducted in order to improve the performance of PARs so that they will not malfunction and incur ignitions in the elevated hydrogen concentrations that occur in severe accidents. The NRC and European regulators should perform safety analyses to determine if existing PARs should be removed from plant containments—and, if so, whether they should be replaced with electrically powered thermal hydrogen recombiners that have their own independent train of emergency power. The latter course would require operators to have instrumentation capable of providing timely information on the local hydrogen concentrations throughout the containment, so they could deactivate the thermal recombiners when hydrogen concentrations reached the levels at which the recombiners malfunction and incur ignitions.

C. Existing oxygen and hydrogen monitoring instrumentation should be significantly improved.

In line with the conclusions of the NRC's own Advisory Committee on Reactor Safeguards (ACRS), the NRC should reclassify oxygen and hydrogen monitors as safety-related equipment that must undergo full qualification (including seismic qualification), have redundancy, and have has its own independent train of emergency electrical power.

The current NRC requirement that hydrogen monitors be functional within 90 minutes of emergency cooling water injection into the reactor vessel is clearly inadequate for protecting public and plant worker safety. Following onset of an accident, NRC regulations should require that hydrogen monitors be functional within a timeframe that enables immediate detection of quantities of hydrogen indicative of core damage and a potential threat to containment integrity.

The NRC should also require hydrogen monitoring instrumentation to be installed in:

1. BWR Mark I and Mark II secondary containments;
2. fuel-handling buildings of PWRs and BWR Mark IIIs; and
3. any plant structure where it would be possible for hydrogen to enter.[5]

D. Current core diagnostic capabilities require upgrading to provide plant operators a better signal for when to transition from emergency operating procedures to severe accident management guidelines.

The NRC should require plants to use thermocouples placed at different elevations and radial positions throughout the reactor core to enable plant operators to accurately measure a wide range of temperatures inside the core under both typical and accident conditions. In the event of a severe accident, in-core thermocouples would provide plant operators with crucial information to help them track the progression of core damage and manage the accident, indicating, in particular, the correct time to transition from EOPs to implementing SAMGs.

E. The NRC should require all nuclear power plants to control the total quantity of hydrogen that could be generated in a severe accident.

The NRC should require all nuclear power plants to operate with systems for combustible gas control that would effectively and safely control the total quantity of hydrogen that could potentially be generated in different severe accident scenarios; and to have strategies for venting gas from the inerted primary BWR Mark I and Mark II containments without causing significant radiological releases. The NRC should also require nuclear power plants to operate with systems for combustible gas control that are capable of preventing local concentrations of hydrogen in the containment from reaching concentrations that could support explosions powerful enough to breach the containment, or damage other essential accident-mitigating

features. Hydrogen explosions are not expected to occur inside the primary BWR Mark I and Mark II containments, which operate with inerted atmospheres, unless somehow oxygen is present.

The NRC should require licensees who operate nuclear power plants with hydrogen igniter systems to perform analyses demonstrating that these systems would effectively and safely mitigate hydrogen in different severe accident scenarios. Licensees unable to do so would be ordered to upgrade their systems to adequate levels of performance.

F. The NRC should require that data from leak rate tests be used to help predict the hydrogen leak rates of the primary containment of each BWR Mark I and Mark II licensed by the NRC in different severe accident scenarios.
The NRC should require that data from overall leak rate tests and local leak rate tests—already required by Appendix J to Part 50 for determining how much radiation would be released from the containment in a design basis accident—also be used to help predict hydrogen leak rates for a range of severe accident scenarios involving the primary containments of each GE-BWR Mark I and Mark II licensed by the NRC. If data from an individual leak rate test were to indicate that dangerous quantities of explosive hydrogen gas would leak from a primary containment in a severe accident, the plant owner should be required to repair the containment.

The rationale for this requirement is obvious: Hydrogen explosions, or hydrogen concentrations in the reactor building that pose a detonation risk, can severely inhibit emergency response actions essential to containing the accident. Or even worse, emergency response actions themselves, such as hooking up portable power equipment, could actually provide the spark for hydrogen explosions in critical areas of the plant.

The NRC should also end its practice of allowing repairs to be made immediately before leak rate tests are conducted to evaluate potential leakage paths, such as containment welds, valves, fittings, and other components that penetrate containment. This “repair before test” practice obviously defeats the nuclear safety objective of providing an accurate statistical sample of actual pre-existing containment leak rates.

Finally, the NRC should reconsider its plan to extend the intervals of overall and local leak rate tests to once every 15 years and 75 months, respectively. The NRC needs to conduct safety analyses that consider BWR Mark I and Mark II primary containments are vulnerable to hydrogen leakage. It also seems probable that as old reactors are kept in service beyond their original licensed lifetimes, the intervals between leak rate tests should be shortened rather than extended.

II. HYDROGEN GENERATION IN NUCLEAR POWER PLANT ACCIDENTS

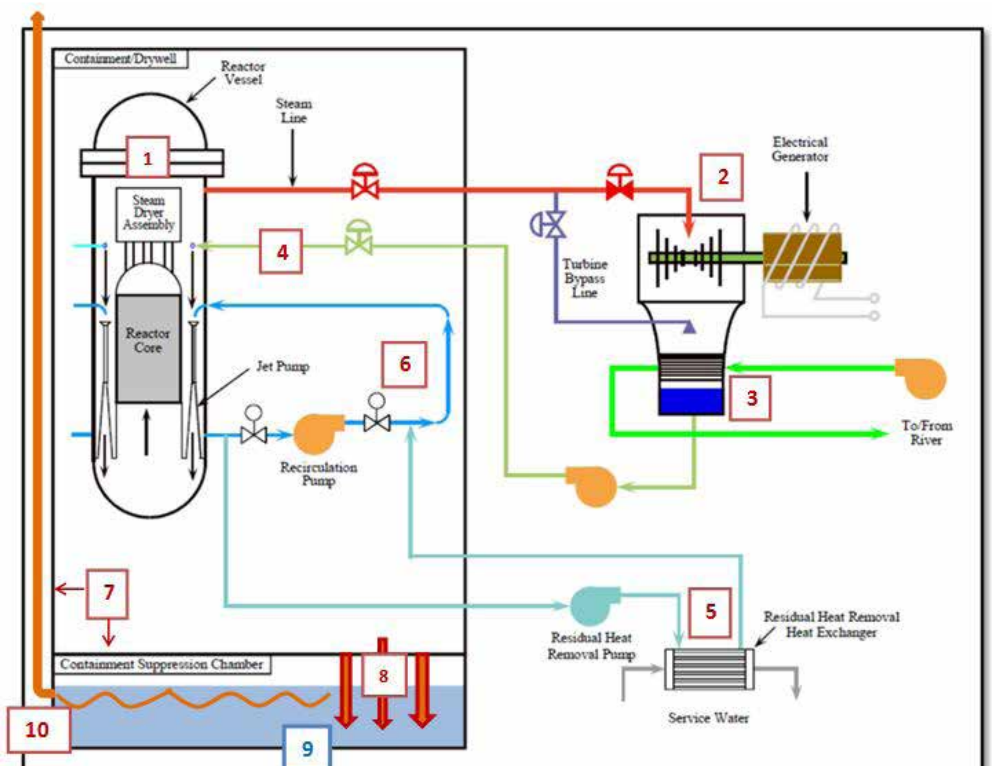
A. TECHNICAL BACKGROUND: DESIGN BASIS ACCIDENTS AND THE ZIRCONIUM-STEAM REACTION

In typical operating conditions at a nuclear power plant, highly pressurized coolant[6] water is pumped through the reactor coolant system[7] piping into the reactor pressure vessel where it flows between the fuel rods, carrying away heat produced by the fission (splitting) of uranium (^{235}U) atoms in the fuel. The coolant water's temperature exceeds 500°F; nonetheless, it still provides cooling for the fuel rods located in the reactor core as long as a sufficient flow of coolant is maintained.[8]

U.S. nuclear power plants are referred to as light water reactors because they use ordinary water (H_2O), as opposed to heavy water (^2H_2O or D_2O), as a coolant. In a boiling water reactor like those that suffered hydrogen explosions at Fukushima, the coolant exits the reactor core as a steam-water mixture. Water droplets are removed in a steam dryer located above the core, and then the steam passes through the steam line to the main turbine, which powers an electric generator, and is condensed back into water before reentering the core (see Figure 5).

Figure 5: Schematic Diagram of Heat Removal from a Boiling Water Reactor (BWR)

Heat is removed during normal operation by generating steam, which rises to the top of the reactor vessel **(1)**, and is then used directly **(red line)** to drive a turbine **(2)** that spins an electrical generator. When a reactor shuts down, however, the core continues to produce heat from radioactive decay. This decay heat is removed initially by bypassing the turbine and delivering the steam directly to the condenser **(3)**, which is cooled by water pumped from lakes, rivers, or ocean **(green)**, with the condensed steam **(blue)** returning to the reactor as coolant **(4)**. When steam pressure drops to approximately 50 pounds per square inch, the residual heat removal (RHR) system **(5)** is used to complete the cool-down process. Water in the normal coolant recirculation loop **(6)** is diverted from the recirculation pump to the RHR pump which sends it through a supplementary heat exchanger and back to the reactor. Multiple electrically controlled pumps and valves are dependent on external sources of electricity for safe operation in the critical period following reactor shutdown. In a severe accident, "drywell" containment **(7)** is designed to vent **(8)** excess radioactive steam pressure into a "wetwell" suppression chamber **(9)** half filled with water, which operators, in turn, can vent to the atmosphere through "Reliable Hardened Vents" **(10)** to relieve excess pressure. Currently, such vents do not filter radioactive aerosols and gases.



Source: NRC Reactor Concepts Manual, Rev. 0200, pages 3-7, with additional explanatory features by NDRC

In a pressurized water reactor the coolant typically circulates to and from the reactor in two to four closed "primary loops," where it is maintained at a pressure high enough to prevent the water from boiling. Each primary loop has a steam generator (heat exchanger) where the coolant heats and boils water circulating through a secondary loop—maintained at a lower pressure than the primary loop—producing pressurized steam to spin the main turbine and generate electricity (see Figures 6 and 7).

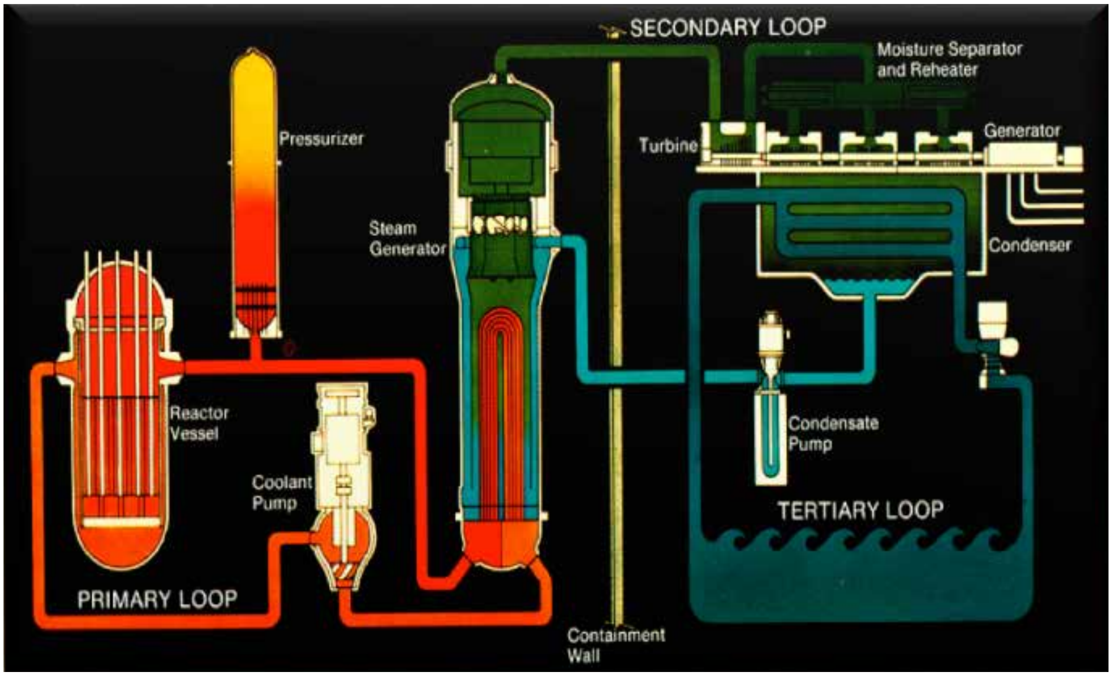
Both reactor types have main condensers to condense the steam back into water after it exits the turbines; this water is pumped back to the reactor pressure vessel (in a BWR) or steam generator (in a PWR). The main condensers of both BWRs and PWRs rely on vast amounts of water, drawn from a local water body such as a lake, river, or ocean. This water may be returned directly to the local water body at elevated temperatures, sometimes damaging the local ecology; alternately, cooling towers may be deployed to remove heat from this water. Roughly two-thirds of the thermal energy produced by a nuclear reactor is not converted into electricity but rather is discharged to the environment as waste heat.

Reactor cores have tens of thousands of uranium fuel rods, bundled together into "fuel assemblies." For example, each reactor at Indian Point Energy Center near New York City has 87 metric tons of fuel contained in 193 fuel assemblies (each with 204 fuel rods), or almost 40,000 fuel rods. The cladding of the fuel rods is made of zirconium alloy.[9] The fuel cladding is a thin tube, typically with a diameter of less than half an inch, sheathing small cylindrical uranium-dioxide fuel pellets stacked one on top of the other. The active fuel region of the fuel rods (the length of the cladding containing the fuel pellets) is approximately 12 feet long.

In sum, a reactor core contains large amounts of zirconium metal that can react with steam at high temperatures to produce vast quantities of hydrogen gas. In the event of a design basis accident,[10] BWR and PWR emergency core cooling systems are designed to inject and circulate water through the reactor core to prevent the fuel rods from overheating when the normal reactor cooling system ceases to function. The respective emergency core cooling systems are required to mitigate a number of postulated design-basis accidents, including the worst-case scenario envisioned

Figure 6: Simplified Schematic Diagram of a Westinghouse Pressurized Water Reactor (PWR) with Three Intersecting Heat Transfer Heat Loops

PWR designs typically have two to four primary loops and a corresponding number of steam generators and main coolant pumps. Water in the primary loop is maintained by the pressurizer at around 2250 pounds per square inch, about twice the pressure of a BWR. Weak points in this system from a radiation containment perspective are the numerous valves and penetrations of the reactor vessel required to control and cool the reactor; the seals of the main coolant pumps, which must be actively cooled and are prone to leakage; and the thousands of small-diameter, thin-walled primary loop "steam tubes" in the steam generators, which are prone to erosion and leakage into the secondary loop. The tertiary loop can be "open," returning heated water from the turbine condenser directly to a local river, lake, or bay; or "closed," utilizing one or more "wet" (evaporative) or "dry" (fan-driven air) cooling towers (not shown) to recycle the tertiary coolant in a "semiclosed" loop (makeup water must be added to the system due to evaporative losses).



Source: The Westinghouse Pressurized Water Reactor Nuclear Power Plant, page 4

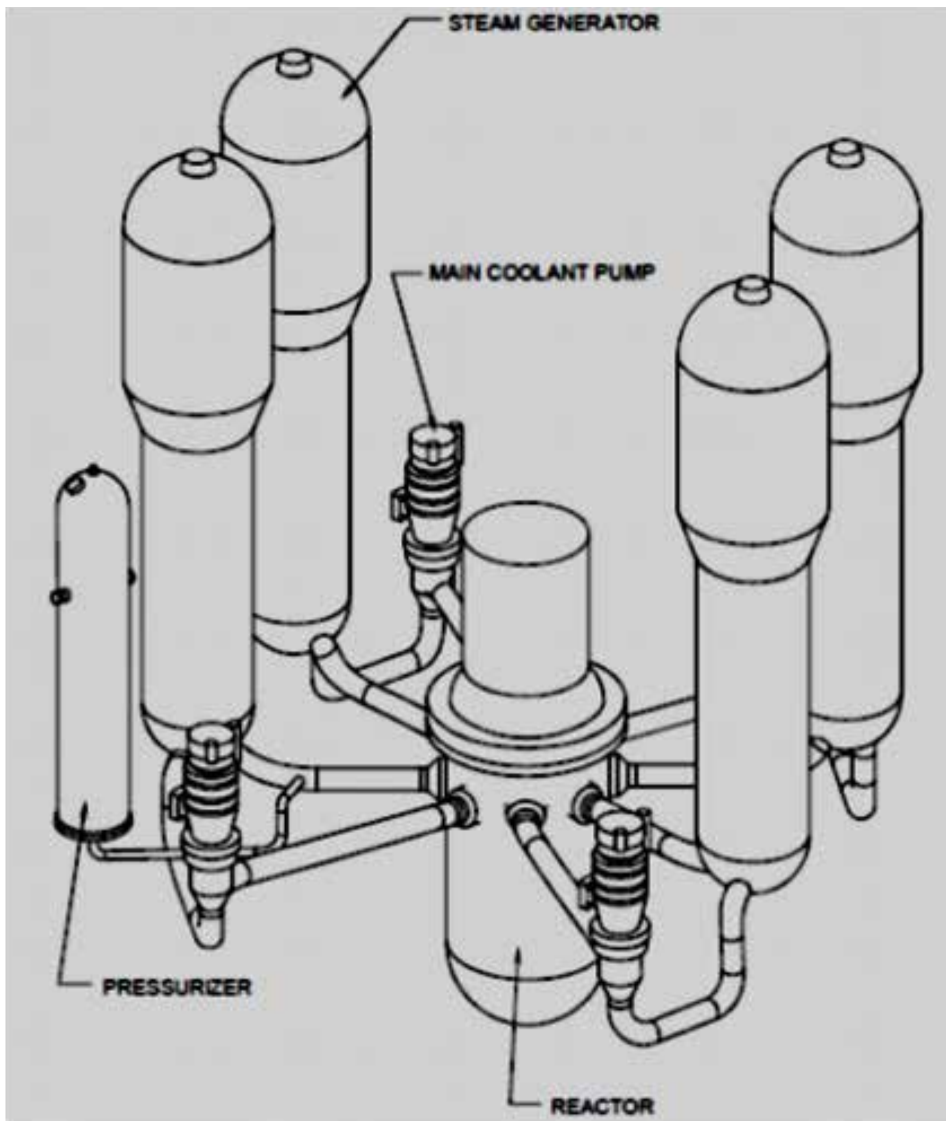
by regulators: a large-pipe-break loss-of-coolant accident (LOCA). Note that the March 2011 Fukushima Daiichi accident in Japan is considered a "beyond design basis" accident[11] or a "severe" accident that exceeded the design parameters of the plant.

In a hypothetical large-pipe-break LOCA at a PWR, the largest pipe in the reactor coolant system would break, causing a rapid discharge of coolant; the core would be either partly or completely emptied of water. The reactor's power would shut down within seconds, because the absence of the coolant, which is also a neutron moderator,[12] and the rapid insertion of control rods would stop the fission chain reaction. A control rod is a rod, plate, or tube containing a neutron-absorbing material used to control the power of a nuclear reactor by preventing further fissions. However, the maximum local temperature of the fuel cladding would increase—from approximately 600°F to more than 1000°F within 60 seconds[13] due to the absence of coolant. The fuel cladding would be heated by the residual heat in the fuel and by decay heating (the radioactive decay of fission products), which at the beginning of an accident would generate about 7 percent of the thermal power produced during normal operation. The decay heat decreases as the accident progresses yet remains a significant heat source for the duration of the accident.

If local fuel-cladding temperatures were to approach 1800°F, the cladding would incur additional heating from the exothermic (heat-generating) reaction of its zirconium content with the steam present in the reactor core. This chemical reaction is variously referred to as a "metal-water reaction," "zirconium-steam reaction," or "zirconium oxidation." The latter term is used because the zirconium-steam reaction produces zirconium dioxide (ZrO_2), in addition to hydrogen and heat.[14]

Figure 7: Layout of a Westinghouse Four-Loop Pressurized Water Reactor (PWR)

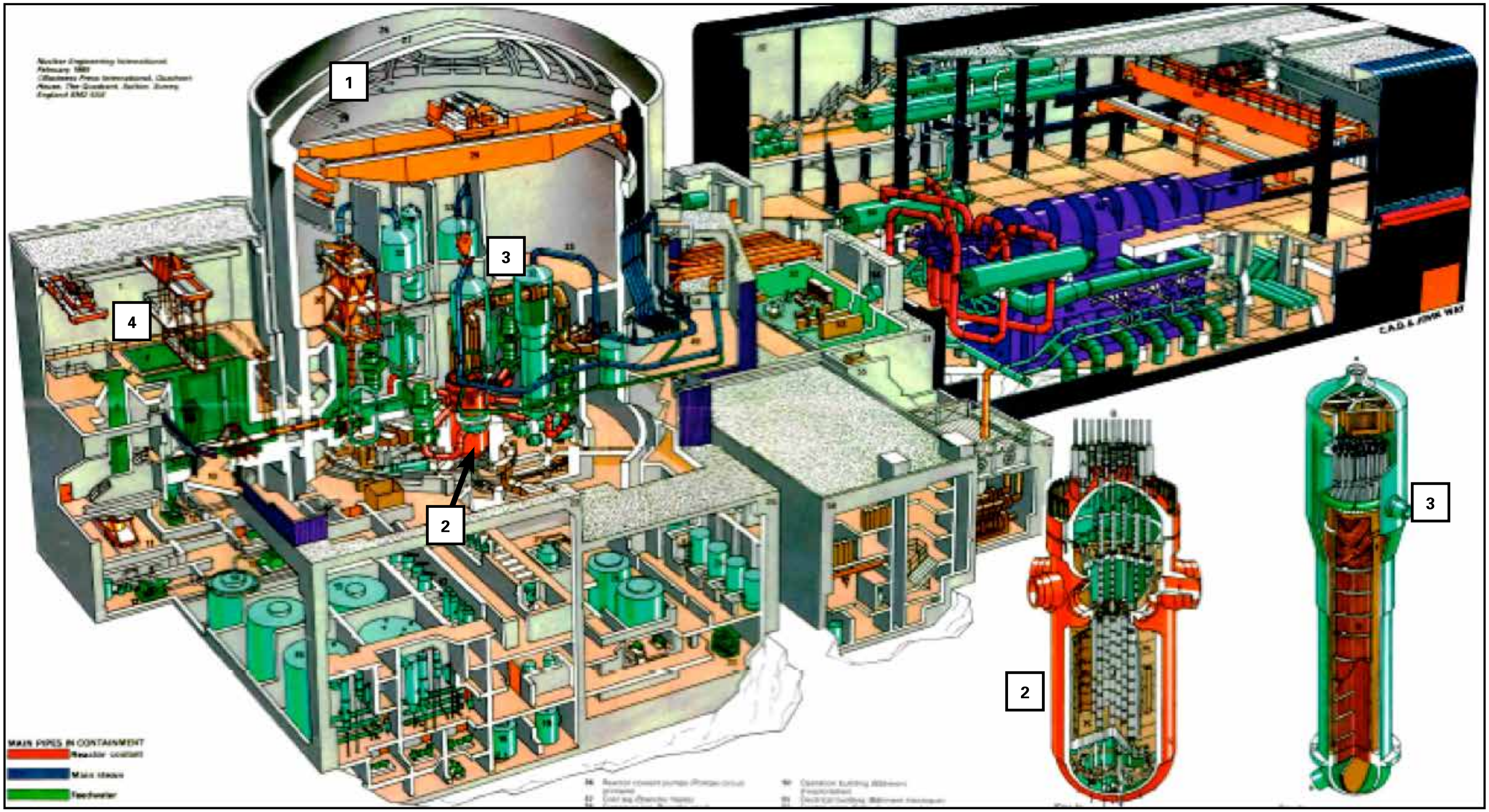
The reactor has four steam generators and four main coolant pumps (the fourth pump is hidden by the perspective of the drawing). All these components are massive. To set the scale, the interior of the reactor vessel is about 15 feet wide by 40 feet high. U.S. examples include Indian Point Units 2 and 3 (New York), Vogtle Units 1 and 2 (Georgia), Comanche Peak Units 1 and 2 (Texas) and Diablo Canyon Units 1 and 2 (California).



Source: NRC Reactor Concepts Training Manual, Pressurized Water Reactor Systems, Section 4-1

Figure 8: Cutaway View of French N4 Standardized PWR Design, Based on Westinghouse Technology but with a Double-Walled Primary Containment Structure (1)

Reactor pressure vessel **(2)** and primary coolant loop piping are shown in **red**; main steam lines (in **blue**) are shown coming from the top of the steam generators **(3)**, shown in **light green**. These are supplied by the feedwater system (**dark green** piping), which also cools the spent fuel pool (4) and main coolant pump seals (dark green). The turbine building (5) encloses a steam-driven turbine generator unit (in **purple**) with a rated output of 1500 MWe. The tertiary cooling loop for the turbine steam condenser is not shown.



Source: University of New Mexico Libraries Exhibition Nuclear Engineering Wall Charts

The NRC's 2011 Near-Term Task Force Review of Insights from the Fukushima Daiichi Accident states that an important aspect of the NRC's approach to safety through defense-in-depth is the mitigation of the consequences of severe accidents, including the mitigation of the hydrogen that would be generated in such an accident. However, the Near-Term Task Force report discusses neither the rates of hydrogen generation that could occur nor the total quantity of hydrogen that could be generated in severe accidents. Given that in the Fukushima Daiichi accident, hydrogen explosions caused large radiological releases, this must be considered a major weakness in the NRC's report and its continuing regulatory response to the lessons learned from the Fukushima accident.

If the emergency core cooling system is to prevent the fuel cladding from overheating in a large-break LOCA, it must overcome the heat from three primary sources: 1) the residual heat stored in the fuel, 2) the heat from radioactive decay, and 3) the heat generated by the zirconium-steam reaction.

B. SEVERE ACCIDENTS AND THE HEAT PRODUCED BY THE ZIRCONIUM-STEAM REACTION

Practically speaking… [zirconium] oxidation runaway comes in…due to the heat of the oxidation reaction increasing generally faster than heat losses from other mechanisms.… [I]f peak [fuel-cladding] temperatures remain below 1000°C [1832°F], you will probably escape the runaway [oxidation], but if you get to 1200°C [2192°F], you will probably see the oxidation "light up" like a 4th of July sparkler (literally that's what it looks like) as it looks like) as it goes into the "rapid oxidation" regime.[15]
—Randall O. Gauntt, Sandia National Laboratories

The Three Mile Island Unit 2 (TMI-2) accident, which occurred in March 1979, was a small-break LOCA[16] that transitioned into a severe accident—a partial meltdown—because there was inadequate cooling of the core. Decay heating caused local fuel-cladding temperatures to increase up to the point at which the cladding began to rapidly react with the steam present in the reactor core, which in turn produced more heat.

Robert E. Henry—an Argonne National Laboratory nuclear safety expert,[17] suggested that in the TMI-2 accident, when local fuel-cladding temperatures reached about 1832°F (1000°C), the heat produced by the zirconium-steam reaction was approximately equal to the heat produced by radioactive decay,[18] and that "from [that] point on, the core was in a thermal runaway state."[19, 20] Henry stated that "[t]he [zirconium] oxidation rate increase[d] with increasing temperature, which [led] to an escalating core heatup rate. Therefore, *the core damage was generally caused by the [zirconium] cladding oxidation*" [emphasis added].[21]

Once thermal runaway (runaway zirconium oxidation) commences in a severe accident, maximum local fuel-cladding temperatures increase rapidly—tens of degrees Fahrenheit per second. Thermal runaway is what leads to a partial or complete meltdown. After thermal runaway commenced in the TMI-2 accident (plausibly at about 1832°F [1000°C]), within a few minutes, maximum local fuel-cladding temperatures would have reached the melting point of zirconium, which exceeds 3300°F.[22]

In the March 2011 Fukushima Daiichi accident, the respective reactor cooling systems of Units 1, 2, and 3 reportedly survived the earthquake more or less intact. However, the plant incurred a loss-of-offsite power, then flooding from the tsunami caused its backup diesel generators to fail, and backup batteries were depleted within about eight hours. The latter were insufficient in any case to power emergency core-cooling pumps once the steam-driven backup pumps became inoperative. Hence, the three units lost the ability to remove their reactors' decay heat. This caused the coolant water to boil away and uncover the fuel rods in the cores of the three units, exposing them to steam. Once the fuel rods were uncovered, decay heating caused cladding temperatures to increase to the point at which their zirconium content rapidly reacted with the steam and generated large quantities of hydrogen gas.

The NRC needs to consider that not all severe accidents would be relatively slow-moving "station-blackout" accidents caused by natural disasters, like the Fukushima Daiichi accident. Fast-moving accidents could also occur; for example, a large-pipe-break LOCA could rapidly transition into a severe accident, because of thermal runaway. A meltdown could commence within 10 minutes of the onset of such an accident.[23]

C. HYDROGEN GENERATION IN ACCIDENTS: RATES AND QUANTITIES

It should be noted that in an unmitigated BWR severe accident the entire Zircaloy inventory of the reactor would eventually oxidize (either in the reactor vessel or on the drywell floor), generating as much as 6000 [pounds] (2722 kg) of hydrogen (plant specific value).[24]
—Sherrell R. Greene of Oak Ridge National Laboratory

In a reactor accident, fuel-cladding temperatures, plant operator actions, and other factors would affect hydrogen generation rates and the total quantity generated.

In a PWR accident in which the maximum fuel-cladding temperature at any point in the core does not exceed 2200°F (the regulatory fuel-cladding temperature limit for design basis accidents[25]), hydrogen generation is predicted to occur at rates from 1 to 50 grams per second;[26] similar rates would occur in a BWR design basis accident. A safety analysis conducted for Indian Point Unit 3 (a large PWR) found, reassuringly, that after a design basis LOCA, it would take a total of 23 days for the hydrogen concentration in the containment to reach 4 percent of the containment's volume (the lower flammability limit).[27]

However, in a severe PWR accident, the picture changes dramatically: hydrogen generation could occur at rates from 100 to 5,000 grams per second[28] (two orders of magnitude greater than in a design basis accident), and similar rates would occur in a severe BWR accident. An OECD Nuclear Energy Agency report states, a "rapid initial [hydrogen]-source occurs in practically all severe accident scenarios because the large chemical heat release of the [zirconium]-steam reaction causes a fast self-accelerating temperature excursion during which initially large surfaces and masses of reaction partners are available."[29]

If an overheated reactor core were re-flooded with water, up to 300,000 grams of hydrogen could be generated in 60 seconds.[30] In this scenario, according to one report, between 5,000 and 10,000 grams of hydrogen could be generated per second.[31] (In the TMI-2 accident, re-flooding of the uncovered reactor core by the emergency core cooling system caused a spike in the hydrogen generation rates; it has been estimated that approximately 33 percent of all the hydrogen produced occurred during re-flooding.[32])

The total quantity of hydrogen that could be generated in a severe accident is different for PWRs and BWRs. Considering hydrogen generated only from the oxidation of zirconium: if the total amount of the zirconium in a typical PWR core, approximately 26,000 kilograms (kg), were to chemically react with steam, this would generate approximately 1150 kg of hydrogen; if the total amount of zirconium in a typical BWR's core, approximately 76,000 kg, were to chemically react with steam, this would produce about 3360 kg of hydrogen.[33]

Large BWR cores typically have about a 58-percent greater initial uranium mass than large PWR cores,[34] and this larger mass is divided into approximately 45 percent more fuel rods than in a PWR. However, these differences alone do not account for the fact that BWR cores have almost three times the mass of zirconium in their cores than PWRs.[35,36] BWR cores have significantly more zirconium mainly because, unlike PWRs, BWR fuel assemblies have "channel boxes" surrounding the fuel rods. The mass of each BWR assembly channel box is greater than 100 kg.[37] Thus a BWR core with 800 fuel assemblies would actually have more than the 76,000 kg of zirconium cited by the IAEA as typically present in a BWR core.)

The total quantity of hydrogen generated in a severe accident can vary widely: The Fukushima Daiichi accident, which resulted in three meltdowns, most likely generated more than 3,000 kg of hydrogen per affected unit; the amount produced in the TMI-2 accident is estimated at about 500 kg.[38] In a severe accident, hydrogen would also be generated within the reactor vessel from the oxidation of non-zirconium materials: metallic structures and boron carbide (in BWR cores).[39] In the TMI-2 accident, the oxidation of steel accounted for approximately 10 percent to 15 percent of the total hydrogen generation.[40] In a case in which the molten core penetrated the reactor vessel, hydrogen would be generated from the oxidation of metallic material (chromium, iron, and any remaining zirconium) during direct containment heating and also from interaction of the molten core with concrete (out of which containment floors are made).[41] A safety study for the PWRs at Indian Point discusses a case in which interaction of a molten core with a concrete containment floor would generate more than 2721.5 kg of hydrogen.[42]

If a molten core interacted with concrete, carbon monoxide (which, like hydrogen, is a combustible gas) would also be generated. Depending on different accident scenarios, concrete types, and geometrical factors affecting the molten core-concrete interaction, the quantities of carbon monoxide generated could vary greatly; concentrations could differ by up to several volume percent in the containment.[43, 44]

D. NRC MODELS UNDERPREDICT SEVERE ACCIDENT HYDROGEN GENERATION RATES

A 2001 OECD Nuclear Energy Agency report advises that high hydrogen generation rates "must be taken into account in risk analysis and in the design of hydrogen mitigation systems." However, the same report notes that computer safety models used by regulators underpredicted the actual rates of hydrogen generation that occurred in two sets of experiments simulating severe accidents: the CORA tests and LOFT LP-FP-2.[45] (The CORA and LOFT LP-FP-2 experiments were conducted to investigate accidents that lead to a meltdown of the reactor core. LOFT LP-FP-2 was conducted with an actual nuclear reactor, 1/50th the volume of a full-size PWR, "designed to represent the major component and system response of a commercial PWR." LOFT LP-FP-2 was an actual core meltdown—the most realistic severe accident experiment conducted to date; it combined decay heating, severe fuel damage, and the quenching of zirconium fuel cladding with water.[46]) Computer safety models also failed to predict hydrogen generation in the initial QUENCH facility experiments.[47] This indicates that computer safety models also underpredict the hydrogen generation rates that would occur in severe accidents.[48]

A 1997 Oak Ridge National Laboratory (ORNL) report states that hydrogen generation in severe accidents can be divided into two separate phases: 1) a phase that runs from when the fuel cladding is still intact through the initial melting of the fuel cladding, which accounts for about 25 percent of the total hydrogen produced; and 2) a phase after the initial melting of the fuel cladding, in which there is additional melting, relocation, and the formation of uranium-zirconium-oxygen blockages, which accounts for about 75 percent of the total hydrogen generated (as indicated in analyses of the BWR CORA-28 and -33 tests).[49]

According to the 1997 ORNL report, computer safety models predict hydrogen generation rates "reasonably well" for the first phase, in which the fuel cladding remains intact, but predict hydrogen generation rates for the second phase "much less robustly." The 1997 ORNL report stresses that it is obvious that computer safety models need to accurately predict hydrogen generation rates when the fuel cladding is no longer intact, especially because most of the hydrogen

generation occurs in that phase.[50]

A 2011 International Atomic Energy Agency (IAEA) report states that computer safety models underpredict the rates of hydrogen generation that would occur during a re-flooding of an overheated reactor core.[51] The report cautions that, in different scenarios, re-flooding could cause hydrogen generation rates to vary to a large degree and that predictions need to consider the possible range of outcomes in order to help prepare for severe accident hydrogen risk. In the BWR CORA-17 test, which simulated the re-flooding and quenching of an overheated core, approximately 90 percent of the hydrogen generation occurred during re-flooding.[52]

Unfortunately, recent reports do not explicitly state the extent that computer safety models under-predict hydrogen generation rates during the re-flooding and quenching of an overheated core—i.e., a percentage value of the under-prediction has not been provided. However, presentation slides from a 2008 European meeting state that the "total amount of hydrogen under reflooding remains *highly underestimated* in [the] CORA-13 and LOFT LP-FP-2 experiments" [emphasis added]. In fact, regarding recent computer simulations of LOFT LP-FP-2, the same presentation slides state: "High temperature excursions with extended core degradation and enhanced hydrogen release observed in the test during reflood was not reproduced *due to the lack of adequate modeling*"[53] [emphasis added].

Despite these reports dating back to 1997, the NRC's 2011 Near-Term Task Force report on insights from the Fukushima Daiichi accident failed to mention, much less discuss, the fact that the NRC's computer safety models—such as the widely used MELCOR code developed by Sandia National Laboratories—underpredict the hydrogen generation rates that occur in severe accidents. By overlooking the deficiencies of computer safety models, the NRC undermines its own philosophy of defense-in-depth, which requires the application of conservative models.[54] When hydrogen generation rates are underpredicted, hydrogen mitigation systems are not likely to be designed so that they could handle the generation rates that would occur in actual severe accidents.

E. AN ATTEMPT TO ELIMINATE HYDROGEN RISK: DEVELOPING NON-ZIRCONIUM FUEL CLADDING

Perhaps the most effective way to help prevent hydrogen explosions in severe accidents would be to develop fuel cladding that does not generate large quantities of hydrogen when the core overheats in such accidents. Zirconium alloy cladding could possibly be replaced with silicon carbide, molybdenum alloys, molybdenum-zirconium alloys, or iron-chromium-aluminum alloys.[55] Silicon carbide is perhaps the most promising alternate; in the design basis accident temperature range—below 2200°F—silicon carbide is far less reactive than zirconium with steam,[56] generating much less hydrogen.

In 2010, according to an article in *Nuclear Engineering International*, a type of silicon carbide fuel cladding with a triplex design[57] was "still in the early stages of development and testing" the article opines that developing such cladding is "a high-risk, but potentially high-payoff"[58] venture. It remains to be seen if triplex silicon carbide would be a suitable replacement for zirconium alloy as a fuel-cladding material; there are a number of problems with silicon carbide cladding that still need to be resolved.

One problem is that during typical reactor operation the fuel pellets in silicon carbide cladding would have higher temperatures than they do when sheathed in zirconium. This would occur for two reasons: First, after extended irradiation, silicon carbide has a lower thermal conductivity than zirconium alloy,[59] meaning less of the fuel's heat would pass through the cladding and into the coolant. Second, the thin gap between the fuel pellets and the cladding would not be closed early in the first fuel cycle as occurs when zirconium cladding is used.[60] Both of these phenomena would prevent the pressurized water from cooling the fuel pellets in silicon carbide cladding as effectively as it does when the fuel pellets are sheathed in zirconium cladding.

A second problem is that an effective means of hermetically sealing the ends of silicon carbide fuel-cladding rods has not yet been developed.[61] If the fuel-cladding rods were not hermetically sealed during reactor operation, fission products would escape from the fuel rods and enter the coolant water.

A June 2012 Nuclear Energy Advisory Committee report lists additional problems with silicon carbide fuel cladding, such as a lack of ductility (the ability to bend, expand or contract without breaking) compared with currently used cladding types. The report also speculates that within four years further research and experimentation should confirm whether or not such problems can be resolved. If the problems are resolved, in-reactor testing of silicon carbide fuel cladding could take an additional 10 to 20 years.[62] Hence, even if all were to go well, it could take more than two decades before silicon carbide fuel cladding is ready for commercial use. There is certainly no reason to expect that zirconium alloy fuel cladding will ever be widely replaced in the aging U.S. fleet of nuclear power plants, which are facing obsolescence in the 2025-2050 timeframe.

III. SEVERE ACCIDENT HYDROGEN EXPLOSIONS: AN UNRESOLVED SAFETY ISSUE

In the Fukushima Daiichi accident, hydrogen detonated in—and seriously damaged—the reactor buildings housing Units 1, 3, and 4, causing large radiological releases. The hydrogen explosion that occurred in the Unit 1 reactor building also "caused a blowout panel in the Unit 2 reactor building to open, which resulted in a loss of secondary containment integrity."[63] Actually, from a strict technical perspective, "secondary containment integrity" was lost the moment the flooded emergency diesel generators failed to supply backup power. Maintaining secondary containment integrity requires (a) an intact reactor building structure, and (b) a standby gas treatment system to filter releases from the intact structure to the atmosphere and maintain the structure at a lower pressure than ambient pressure (thus ensuring, in the case of small leaks, that outside air leaks *in* rather than inside air leaking *out*). Flooding of the emergency diesel generators by the tsunami took away (b) hours before the explosion took away (a).[64]

As discussed in the preceding sections, the zirconium-steam reaction will generate large quantities of hydrogen in severe accidents. When it reaches a sufficient local concentration inside the containment, this hydrogen will explode if exposed to an ignition source, of which there are many, given the amount of electrical equipment and wiring located inside the containment. In the TMI-2 accident, a hydrogen explosion—probably initiated by an electric spark[65]—occurred in the containment (a PWR large dry containment). The TMI-2 accident explosion did not breach the containment; however, the integrity of either a PWR ice condenser containment or a BWR Mark III containment could be compromised by an explosion of the quantity of hydrogen generated in the TMI-2 accident, because such containments have substantially smaller volumes and lower design pressures than PWR large dry containments.[66,67]

The fact that a hydrogen explosion did not breach TMI-2's containment does not preclude the possibility that if a meltdown were to occur at another PWR with a large dry containment, a hydrogen explosion could breach the containment, exposing the public to a large radiological release. Nonetheless, the NRC 2011 Near-Term Task Force report on insights from the Fukushima Daiichi accident claims that the pressure spike of potential hydrogen explosions would remain within the design pressure of PWR large dry containments.[68] However, according to NRC safety analyses,[69] conducted a decade ago, hydrogen explosions inside PWR large dry containments—of the quantity of hydrogen generated from zirconium-steam reactions of 100 percent of the active fuel-cladding length—could cause pressure spikes as high as 114 pounds per square inch (psi)[70] to 135 psi[71]—over twice the design pressure of a typical PWR large dry containment.

Such extreme pressure spikes could cause a PWR large dry containment to fail. There are also other safety analyses with worrisome results. For example, analyses conducted for Indian Point Units 2 and 3 about three decades ago found that peak pressures caused by hydrogen explosions could exceed the estimated failure pressure of Indian Point's containments—approximately 126 pounds per square inch gauge[72] (psig) or 141 pounds per square inch absolute[73] (psia).[74] For certain severe accident scenarios, peak pressure spikes were predicted to be 160 psia, 169 psia, about 157 psia, and 180 psia or greater.[75] (Some nuclear safety experts believe the accuracy of containment failure pressure estimates is questionable; according to one, "Experimental data on the ultimate potential strength of containment buildings and their failure modes are lacking."[76])

A. THE POTENTIAL DAMAGE OF MISSILES PROPELLED BY HYDROGEN EXPLOSIONS

In a severe accident, a local hydrogen explosion within the containment could propel debris, such as concrete blocks from disintegrated compartment walls, at extremely high speeds. The impact of such debris ("internally-generated missiles") could compromise essential safety systems and even breach the containment, especially if it were made of steel.[77] If a PWR large dry containment made of reinforced concrete with a steel liner[78] were struck by a missile propelled by a hydrogen explosion, the containment would be more likely to incur cracks than to experience gross failure. Yet this is mere speculation: According to a 2011 IAEA report, "no analysis ever has been made on the damage potential of flying objects, generated in [a hydrogen]-explosion."[79]

An Institute of Nuclear Power Operations (INPO) report, published in November 2011 thoroughly documents how in the Fukushima Daiichi accident, internally generated missiles and missiles from secondary containments, propelled by hydrogen explosions, caused a considerable amount of damage and set back efforts to control the accident.[80] The report states:

> [D]ebris from the explosion struck and damaged the cables and mobile generator that had been installed to provide power to the standby liquid control pumps. The debris also damaged the hoses that had been staged to inject seawater into Unit 1 and Unit 2. ... Some of the debris was also highly contaminated, resulting in elevated dose rates and contamination levels around the site. As a result, workers were now required to wear additional protective clothing, and stay times in the field were limited. The explosion significantly altered the response to the event and contributed to complications in stabilizing the units.[81]

B. HYDROGEN EXPLOSIONS: DEFLAGRATIONS AND DETONATIONS

In a severe accident, water pumped into the reactor core to cool the fuel rods would heat up and produce thousands of kilograms of steam, which would enter the containment through pressure relief valves or a break in the cooling system circuit. At different points in an accident the presence of large quantities of steam in the containment would have an inerting effect, either helping to prevent or completely preventing hydrogen combustion if the steam concentration were 55 volume percent[82] or greater. (If hydrogen combustion were to occur, the presence of steam would help reduce its intensity.[83]) However, after enough steam condensed and this would be inevitable at some point in an accident, either naturally or by the use of containment spray systems[84]—either local or global hydrogen combustion could occur.

In a dry atmosphere of hydrogen and air, the lower flammability limit of hydrogen is a concentration of 4.1 volume percent.[85] If hydrogen concentrations were from 4.1 to about 8.0 volume percent, hydrogen combustion would be in the form of a deflagration with a relatively slow flame speed.[86] A deflagration is a combustion wave traveling at a subsonic speed relative to the unburned gas. (In the TMI-2 accident, a hydrogen deflagration occurred when the hydrogen concentration was 8.1 volume percent[87] causing a rapid pressure increase of approximately 28 psi in the containment.[88]) A famous instance of a hydrogen deflagration occurred on May 6, 1937, when the hydrogen-filled dirigible Hindenburg ignited while landing at Lakehurst, NJ and collapsed into a smoldering mass of twisted wreckage on the ground within a matter of seconds.

In a severe reactor accident, hydrogen could randomly deflagrate when its concentrations were at 8.0 volume percent or lower, because only a small quantity of energy is required for igniting hydrogen; sources of random ignition include electric sparks from equipment and static electric charges.[89] It has been postulated that in the TMI-2 accident, the hydrogen deflagration was initiated by a ringing telephone[90] and in the case of the Hindenburg, by the buildup of a static electric charge on its specially-coated outer skin.

In one sense, random or in some instances deliberate ignition of hydrogen at relatively low concentrations is beneficial, in that it can prevent the hydrogen from building up to more dangerous detonable concentrations. Unfortunately, in a severe accident, the average hydrogen concentration in the containment could reach 7.0 to 16.0 volume percent, or higher; local concentrations could be much higher. In a dry atmosphere of hydrogen and air, with hydrogen concentrations above about 10.0 volume percent,

Table 1: Calculated Hydrogen (H2) production Due to 75% Zirconium-Water Reaction

Note that all the predicted containment hydrogen concentrations (far right-hand column) are above the combustion threshold of 4.1 volume percent, and most are above temperature-dependent detonation thresholds of 11.6 and 9.4 volume percent hydrogen, at 68°F and 212°F, respectively.

Plant Name	Fuel Design ID	75% Wt. of Zr (lb)	Vol. of H_2 STP (ft^3)*	Containment Vol. (ft^3)	H_2 Conc. in Dry Air (%)
Arkansas-1	B&W B-2,3,4	31382	268432	1.78×10^6	13.10
Bellefonte 1,2	B&W Mark C	35793	306159	3.00×10^6	9.26
Millstone-2	CE 14x14	31111	266109	1.90×10^6	12.29
Palisades	CE 15x15	35415	302928	1.64×10^6	15.59
Arkansas-2	CE 16x16	30958	264807	1.78×10^6	12.95
Point Beach 1,2	W 14x14	16686	142728	1.00×10^6	12.49
Turkey Pt. 3,4	W 15x15	24674	211057	1.55×10^6	11.98
Zion 1,2	W 15x15	30332	259452	2.60×10^6	9.07
Trojan	W 17x17	32124	274775	2.00×10^6	12.08
Fort Calhoun	Exxon CE-14	22460	192116	1.10×10^6	14.87
Palisades	Exxon CE-14	34450	294674	1.64×10^6	15.23
Maine Yankee	Exxon CE-14	36645	313452	1.80×10^6	14.83
Fort Calhoun	Exxon CE-15	22713	194283	1.10×10^6	15.01
Palisades	Exxon CE-15	34839	297998	1.64×10^6	15.38
Maine Yankee	Exxon CE-15	37059	316988	1.80×10^6	14.97
Ginna	Exxon W-15	23259	198948	0.997×10^6	16.64
Robinson-2	Exxon W-15	30179	258139	2.10×10^6	10.95
Ginna	Exxon W-17	21327	182424	0.997×10^6	15.47
Robinson-2	Exxon W-17	27672	236699	2.10×10^6	10.13

*1 lbm of Zr will generate 0.044 lbm of H_2 and density of H_2 at STP = 5.144×10^{-3} lbm/ft^3

Source: D. W. Stamps et al., Sandia National Laboratories, *Hydrogen-Air-Diluent Detonation Study for Nuclear Reactor Safety Analyses*, NUREG/CR-5525, January 1991, available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML071700388)

flames can accelerate up to and beyond the speed of sound: this phenomenon is termed "deflagration-to-detonation transition."[91] A "detonation" is a combustion wave traveling at a supersonic speed (greater than the speed of sound) relative to the unburned gas. Hydrogen combustion in the form of detonations occurred in the Fukushima Daiichi accident.

Higher temperatures and/or the presence of carbon monoxide could increase the likelihood of a deflagration-to-detonation transition. In a dry hydrogen-air mixture, the lower concentration limits at which deflagration-to-detonation transition can occur is 11.6 volume percent at temperature of 68°F; at 212°F, the lower concentration limit falls to 9.4 volume percent.[92] And in the presence of 5.0 volume percent of carbon monoxide (generated if a molten core interacts with a containment's concrete floor), 10.0 volume percent of hydrogen can detonate at approximately 68°F.[93]

One safety expert has concluded that within the large geometries of PWR-containments a slow laminar deflagration would be very unlikely. In most cases, highly efficient combustion modes must be expected."[94] In a small-break LOCA, large quantities of steam could enter the containment well before hundreds of kilograms of hydrogen were released into the containment. In such a scenario, thermal stratification could prevent the hydrogen from mixing with the steam.[95] In scenarios in which large quantities of steam were present in the containment, the hydrogen could reach high concentrations because the inerting effect of the steam could prevent the hydrogen from igniting at lower concentrations. After the steam condensed, a deflagration could transition into a etonation.

Table 2: Release Paths in LWR Containments

Rupture of Containment wall or shell
Leakage past Sealing Surfaces (Operable Penetrations) ○ **Pressure Seating Equipment Hatches** ○ **Pressure Unseating Hatches (Drywell Heads, Equipment Hatches)** ○ **Personnel Air Locks**
Leakage past Purge and Vent Valves
Leakage from Electrical Penetration Assemblies
Leakage due to Failure of a Bellows (Expansion Joint)

Source: *Containment Integrity Research at Sandia National Laboratories: An Overview*, Sandia National Laboratories, NUREG/CR-6906/ SAND2006-2274P, July 2006

C. LIMITATIONS OF COMPUTER SAFETY MODELS TO PREDICT HYDROGEN DISTRIBUTION IN THE CONTAINMENT AND HYDROGEN DEFLAGRATION-TO-DETONATION TRANSITION

In a September 2011 meeting of the Advisory Committee on Reactor Safeguards (ACRS), Dana Powers, senior scientist at Sandia National Laboratories, expressed concern over the fact that hydrogen detonations occurred in the Fukushima Daiichi accident and stated that in experiments, "detonations are…extraordinarily hard to get."[96,97,98] Consequently, computer safety models (codes) derived from these experiments have limitations in predicting the hydrogen distribution and steam condensation that would occur in the containment in different severe accident scenarios.

A 2007 OECD Nuclear Energy Agency report states, "Further work in code development…and code user training, supported by suitable complex experiments, is necessary to achieve more accurate predictive capabilities for containment thermal hydraulics and atmospheric gas/steam distribution. As a result of the code assessment, the modeling of the following three phenomena appeared to be the major issues: *condensation, gas density stratification, and jet injection*" [emphasis added].[99]

Computer safety models also have limitations in predicting the phenomenon of hydrogen deflagrations transitioning into detonations; as well as the maximum pressure loads the containment would incur from detonations, in different scenarios. Westinghouse's probabilistic risk assessment for its new and supposedly "passively safe" AP1000 reactor design, under construction in Georgia and South Carolina, observes that the phenomenon of hydrogen "deflagration-to-detonation transition is complex and not completely understood" and that the maximum pressure loads from detonations are difficult to calculate."[100]

The Fukushima Daiichi accident demonstrated that the NRC needs to conduct more realistic hydrogen combustion experiments—perhaps in facilities on the same scale as actual reactor containments, at elevated temperatures and with the large quantities of hydrogen that are produced in severe accidents.

IV. SEVERE ACCIDENT HYDROGEN MITIGATION

A. HYDROGEN-MITIGATION STRATEGIES FOR DIFFERENT CONTAINMENT DESIGNS

Over the course of six decades, the NRC and its predecessor agency, the Atomic Energy Commission, have licensed six basic types of reactor containments (see Table 3), but within each type there are numerous design and construction differences (see Table 4) that translate into a wide and highly uncertain range of capacities to contain a severe reactor accident.

Table 3: U.S. Power Reactor Containment Structures, by Type

Containment Type	Number
PWR Large Dry	53
PWR Subatmospheric	7
PWR Ice Condenser	9
Total PWRs	69
BWR Mark I	22
BWR Mark II	8
BWR Mark III	4
Total BWRs	34
US Total	103

Source: NUREG/CR-6906/SAND2006-2274P, July 2006

PWRs with Large Dry Containments and PWRs with Subatmospheric Containents

The NRC does not require the owners of PWRs with large dry containments (52 out of 53 such units are currently operational in the U.S.), or the owners of PWRs with sub-atmospheric containments, maintained at an internal pressure below atmospheric pressure (five out of seven such units are currently operational in the U.S.) to mitigate the hydrogen that would be generated in severe accidents. The agency assumes that the large containment volumes of such PWRs are sufficient to keep the pressure spikes of potential hydrogen deflagrations within the design pressures of the structures.[101]

One hydrogen mitigation strategy for these types of containments would be to mix the hydrogen entering the containment using its fan coolers; this would reduce local hydrogen concentrations and mix the hydrogen with steam, which has an inerting effect.[102] A second hydrogen mitigation strategy for such PWRs would be to use hydrogen recombiners, safety devices that eliminate hydrogen in an accident by recombining hydrogen with oxygen—a reaction that produces steam and heat. There are two types of recombiners: passive autocatalytic recombiners (PAR), which operate without electric power, and electrically powered thermal recombiners. The hydrogen removal capacity for one hydrogen recombiner unit is only several grams per second.[103]

In September 2003, the NRC likewise rescinded its requirement that PWRs with large dry containments and PWRs with sub-atmospheric containments operate with hydrogen recombiners installed in their containments. It decided that the quantity of hydrogen produced in design-basis accidents would not be risk-significant and that hydrogen recombiners would be ineffective at mitigating the quantity of hydrogen produced in severe accidents[104] when hydrogen generation could occur at rates as high as 5.0 kg per second.[105]

In the United States, if such PWRs still have hydrogen recombiners, there are typically two of them in each containment, to mitigate the quantity of hydrogen produced in a design basis accident. For example, Indian Point's containments each have two hydrogen recombiner units.[106] To help mitigate hydrogen in a wide range of severe accident scenarios, a group of European nuclear safety experts have recommended that such PWRs have from 30 to 60 hydrogen recombiner units distributed in their containments.[107] However, even 60 hydrogen recombiner units would not be capable of eliminating all of the hydrogen generated in some severe accident scenarios within the timeframe required to prevent a hydrogen explosion.

Table 4: U.S. PWRs Classified by Containment Construction Type

Primary Containment	Construction	Construction Detail	Plants
Large Dry Primary Containment	Steel Cylinder	Steel Cylinder with Reinforced Concrete Shield Building	Kewaunee Prarie Island 1 Prarie Island 2 Davis-Besse St. Lucie 1 St. Lucie 2 Waterford 3
Large Dry Primary Containment	Reinforced Concrete Cylinder with Steel Liner	Reinforced Concrete Cylinder with Steel Liner	Comanche Peak 1 Comanche Peak 2 Diablo Canyon 1 Diablo Canyon 2 Indian Point 2 Indian Point 3 Salem 1 Salem 2 Shearon Harris 1
Large Dry Primary Containment	Reinforced Concrete Cylinder with Steel Liner	Reinforced Concrete Cylinder with Steel Liner and Secondary Containment	Seabrook 1
Large Dry Primary Containment	Posttensioned Concrete Cylinder with Steel Liner	1-D Vertical Posttensioned Concrete Cylinder with Steel Liner	Ginna HB Robinson
Large Dry Primary Containment	Posttensioned Concrete Cylinder with Steel Liner	Diagonal Posttensioned Concrete Cylinder with Steel Liner	Fort Calhoun
Large Dry Primary Containment	Posttensioned Concrete Cylinder with Steel Liner	3-D Posttensioned Concrete Cylinder with Steel Liner	Arkansas 1 Arkansas 2 Oconee 1 Oconee 2 Oconee 3 Crystal River 3 Three Mile Island 1 Calvert Cliffs 1 Calvert Cliffs 2 Palisades Palo Verde 1 Palo Verde 2 Palo Verde 3 San Onofre 2 San Onofre 3 Braidwood 1 Braidwood 2 Byron 1 Byron 2 Callaway Farley 1 Farley 2 Point Beach 1 Point Beach 2 South Texas 1 South Texas 2 Summer Turkey Point 3 Turkey Point 4 Vogtle 1 Vogtle 2 Wolf Creek
Large Dry Primary Containment	Posttensioned Concrete Cylinder with Steel Liner	3-D Posttensioned Concrete Cylinder with Steel Liner and Secondary Containment	Millstone 2
Subatmospheric Primary Containment		Reinforced Concrete Cylinder with Steel Liner	Beaver Valley 1 Beaver Valley 2 North Anna 1 North Anna 2 Surry 1 Surry 2
Subatmospheric Primary Containment		Reinforced Concrete Cylinder with Steel Liner	Millstone 3
Ice Condenser Primary Containment		Steel Cylinder with Reinforced Concrete Shield Building	Sequoyah 1 Sequoyah 2 Watts Bar 1 Catawba 1 Catawba 2 McGuire 1 McGuire 2
Ice Condenser Primary Containment		Reinforced Concrete Cylinder with Steel Liner	DC Cook 1 DC Cook 2

Source: NUREG/CR-6906/SAND2006-2274P, July 2006

Figure 9: Typical PWR Large Dry Containment Designs

Left: Large dry steel primary containment with reinforced-concrete shield. Right: Containment constructed with post-tensioned concrete with steel liner (e.g., Palisades).

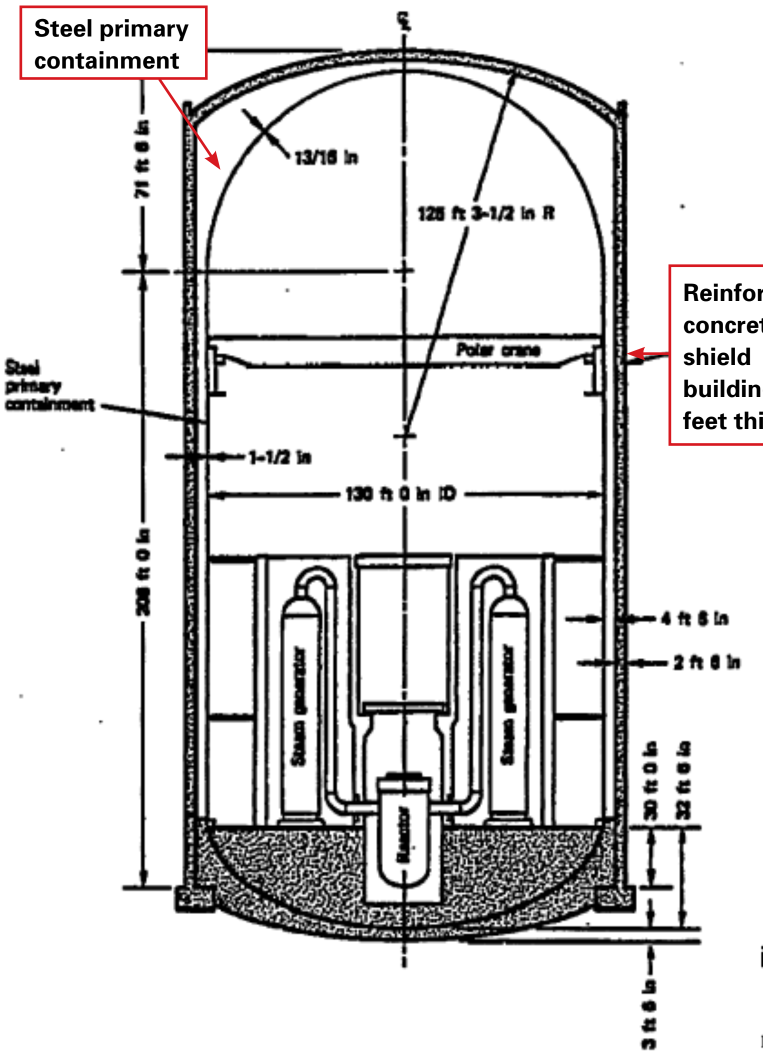


Figure 2 Large Dry Steel Containment with Reinforced Concrete Shield Buildin
(e.g. Davis-Besse)

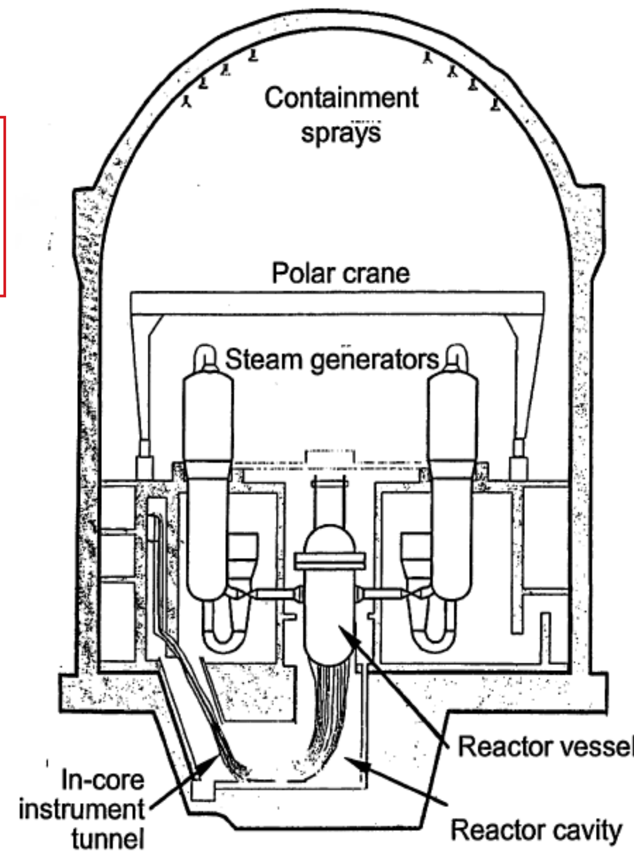


Figure 1 Typical PWR Large Dry Containment with Prestressed Containment (e.g. Palisades)

PWRs with Ice Condenser Containments and BWR Mark III

The NRC requires that PWRs with ice condenser containments (nine such units are currently operational in the U.S.) and BWR Mark IIIs (four are currently operational in the United States) operate with hydrogen igniters installed in their containments in order to mitigate the hydrogen that would be generated in the event of a severe accident.[108] Hydrogen igniters are intended to burn off hydrogen as it is generated in an accident, before it reaches concentrations at which combustion would threaten the integrity of the containment. Hydrogen mitigation is essential for PWRs with ice condenser containments and BWR Mark IIIs *because their containments have relatively low design pressures,*[109] *which makes them more vulnerable to hydrogen exlosions*.

Such containments could be compromised by an explosion of the quantity of hydrogen that was generated in the TMI-2 accident.[110] Hydrogen igniters are intended to manage the quantity of hydrogen that would be generated by a zirconium-steam reaction of 75 percent of the fuel-cladding's active length,[111] which is considerably less than the quantity of hydrogen generated at each melted-down unit at Fukushima-Daiichi.

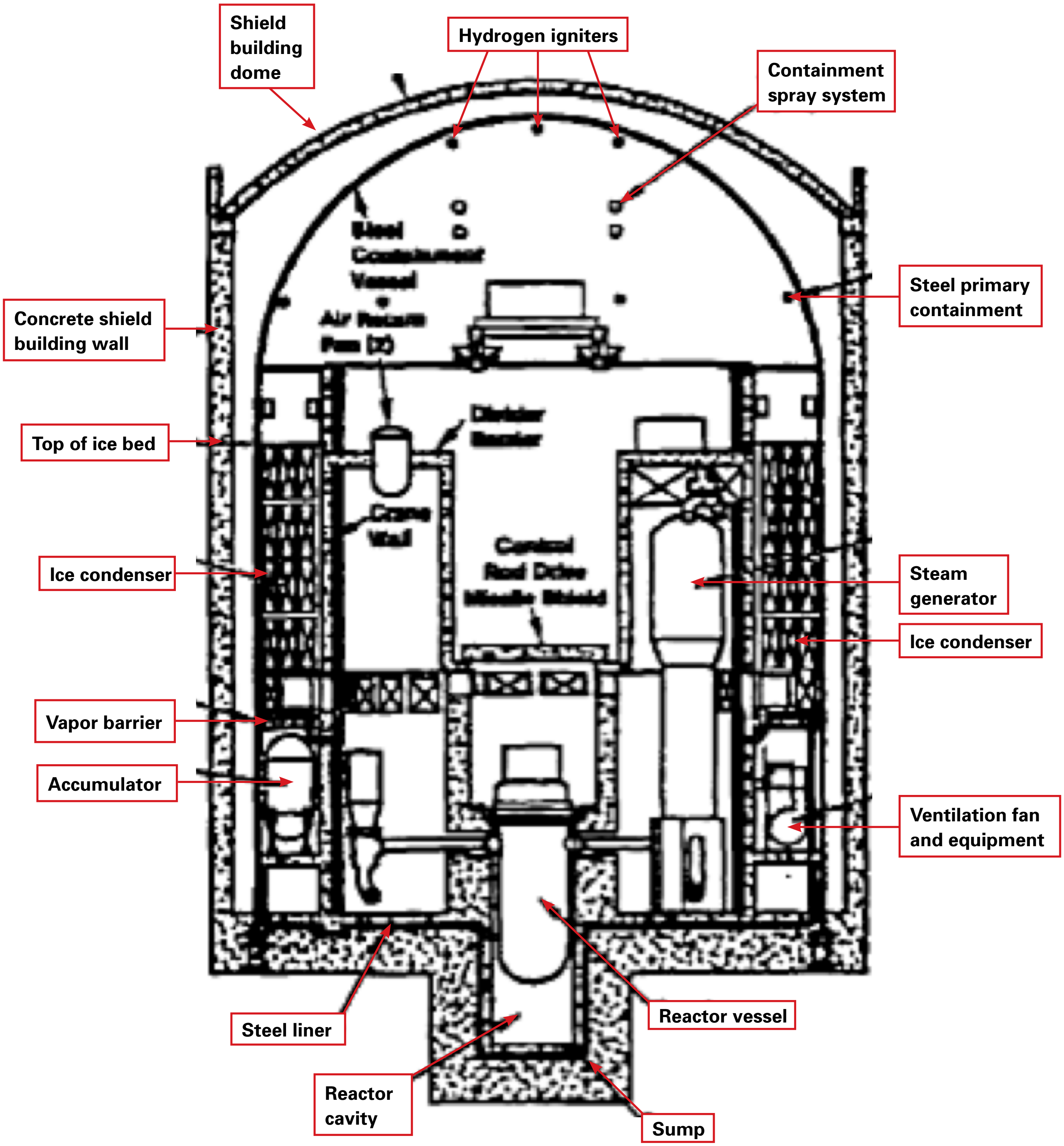
Table 5: U.S. BWRs by Containment Construction Type

A Mark I plant, Vermont Yankee, is missing from the NRC's compilation.

Free Standing Steel Primary Containment	Mark I Steel Drywell & Wetwell	Nine Mile Point 1 Oyster Creek Dresden 2 Dresden 3 Monticello Pilgrim 1 Quad Cities 1 Quad Cities 2 Browns Ferry 1 Browns Ferry 2 Browns Ferry 3 Cooper Duane Arnold Fermi 2 Fitzpatrick Hatch 1 Hatch 2 Hope Creek 1 Peach Bottom 2 Peach Bottom 3
	Mark II Steel Drywell & Wetwell	Columbia
	Mark III Reinforced Concrete Drywell Steel Wetwell	Perry 1 Riverbend 1
Reinforced Concrete Primary Containment with Steel Liner	Mark I Reinforced Concrete Drywell & Wetwell	Brunswick 1 Brunswick 2
	Mark II Reinforced Concrete Drywell & Wetwell	Limerick 1 Limerick 2 Susquehanna 1 Susquehanna 2 Nine Mile Point 2
	Mark III Reinforced Concrete Drywell & Wetwell	Clinton 1 Grand Gulf 1
Post-tensioned Concrete Primary Containment with Steel Liner	Mark II Reinforced Concrete Drywell Posttensioned Wetwell	LaSalle 1 LaSalle 2

Source: NNUREG/CR-6906/ SAND2006-2274P, July 2006

Figure 10: Typical PWR Ice Condenser Steel Containment with Concrete Shield Building (e.g., Sequoyah)



Source: NUREG/CR-6906/SAND2006-2274P, July 2006

BWR Mark I and BWR Mark II

The NRC requires that BWR Mark Is (23 such units are currently operational in the U.S.) and BWR Mark IIs (eight such units are currently operational in the U.S.) operate with primary containments that have an inerted atmosphere[112]—to help prevent hydrogen combustion. An inerted containment atmosphere is defined as having less than 4.0 percent oxygen by volume.[113]

Nitrogen is used to inert BWR Mark I and Mark II primary containments because nitrogen is inexpensive and nontoxic. Such containments are relatively small, so deinerting and inerting for outages between fuel cycles can be achieved within hours; these processes are also inexpensive.[114]

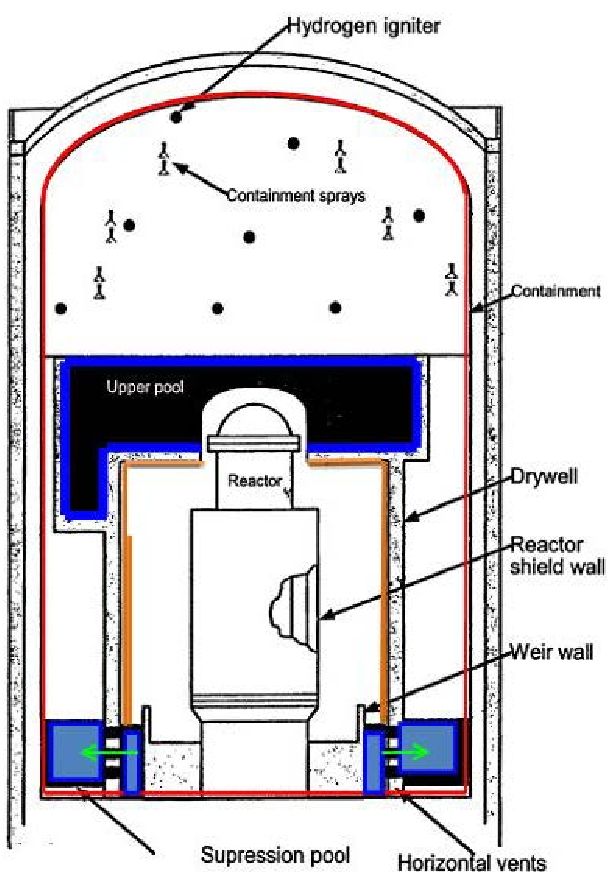
If BWR Mark I and Mark II primary containments were not inerted, they would be extremely vulnerable to hydrogen explosions in severe accidents, because of their relatively small volumes.[115]

Such containments, if not inerted, could easily be compromised by an explosion of the quantity of hydrogen generated in the TMI 2 accident. A year after the Fukushima accident, in March 2012, the NRC ordered the installation of reliable hardened vents in BWR Mark I and Mark II containments by December 31, 2016.[116] A hardened vent could help control hydrogen in a severe accident but its primary purposes are to remove heat from and depressurize BWR Mark I and Mark II containments, which due to their small volumes are more susceptible than other containment designs to failure from overpressurization in an acident.

In September 1989, the NRC issued non-legally binding guidance to all owners of BWR Mark I facilities, recommending[117] that hardened vents be installed.[118] The NRC does not require that hydrogen be mitigated in the secondary containments of BWR Mark I and Mark II units.

Figure 11: Cross-section View of a Typical BWR Mark III Containment (e.g., Perry, Riverbend)

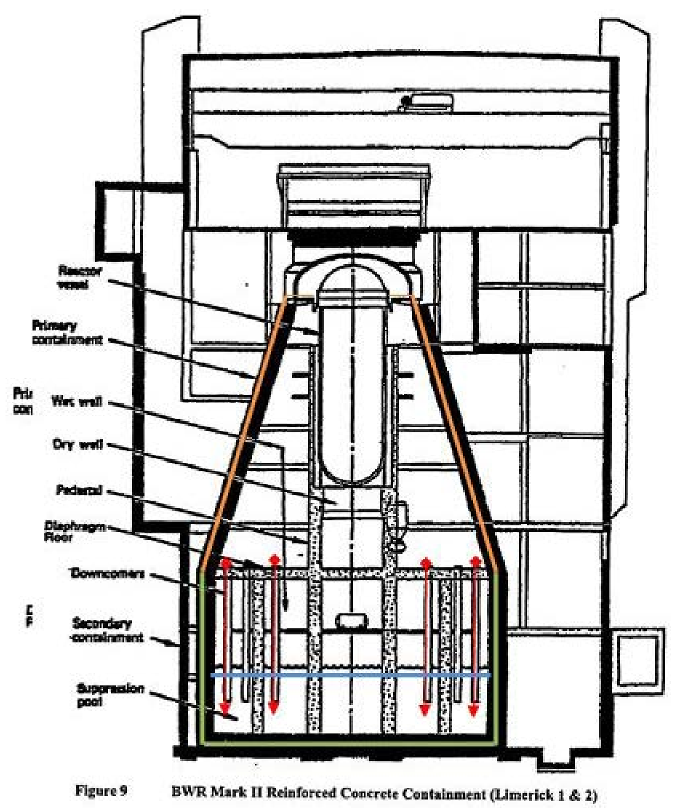
Freestanding steel primary containment **(red)** with lower suppression pool **(blue)** and concrete shield building) has a low design pressure rating (15 psig), requiring that credit be given to the use of hydrogen igniters and containment sprays to meet containment requirements.



Source: NNUREG/CR-6906/SAND2006-2274P, July 2006

Figure 12: BWR Mark II Reinforced Concrete Containment (Limerick Units 1 and 2)

Drywell inerted with nitrogen **(orange)** is connected by pressure relief pipes **(red)** to wetwell **(green)**. Waterline is in **blue**.



Source: NNUREG/CR-6906/SAND2006-2274P, July 2006

CASE STUDY: Hydrogen Risks in Westinghouse's Probabilistic Risk Assessment for the AP1000 and Plans for Managing an AP1000 Severe Accident

Currently four Toshiba-Westinghouse AP1000 units are under construction in South Carolina and Georgia. The NRC purports to have more stringent safety requirements for the AP1000, that "reflect the Commission's expectation that future designs will achieve a higher standard of severe accident performance" than currently operating light water reactors.[119] And Westinghouse has touted the AP1000 as having, in the event of a severe accident, a far lower probability of breaching its containment than currently operating nuclear power plants. However, Westinghouse's probabilistic risk assessment (PRA) for the AP1000 erroneously claims that it would not be possible for a hydrogen detonation to occur in the AP1000's containment if the hydrogen concentration were less than 10.0 volume percent. A hydrogen detonation could compromise the containment and thus cause a large radioactive release. In fact, Westinghouse's PRA assumes that the containment would fail "in all cases," in which hydrogen deflagrations transitioned into detonations.[120]

Westinghouse's PRA for the AP1000 states that "[s]ince the lowest hydrogen concentration for which deflagration-to-detonation transition has been observed in the intermediate-scale FLAME facility at Sandia [National Laboratories] is 15 percent,[121] and [NRC regulation] 10 CFR 50.44 limits hydrogen concentration to less than 10 percent, the likelihood of deflagration-to-detonation transition is assumed to be zero if the hydrogen concentration is less than 10 percent."[122]

Westinghouse does not consider that the lower concentration limits at which deflagration-to-detonation transition can occur, at temperatures of 68°F and 212°F, are 11.6 and 9.4 volume percent of hydrogen, respectively.[123] According to a 1998 Brookhaven National Laboratory report: "Most postulated severe accident scenarios are characterized by containment atmospheres of about 373K [212°F]… However, calculations have shown that under certain accident scenarios local compartment temperatures in excess of 373K [212°F] are predicted."[124]

It is perplexing that Westinghouse's PRA for the AP1000 as well as the NRC's regulations for "future water-cooled reactors" rely on *outdated* assumptions that the phenomenon of hydrogen deflagration-to-detonation transition cannot occur below hydrogen concentrations of 10.0 volume percent: in 1991, Sandia National Laboratories reported that, in an experiment, deflagration-to-detonation transition occurred at 9.4 volume percent of hydrogen.[125] The previous year, the same information was reported at the NRC's Eighteenth Water Reactor Safety Information Meeting.[126]

In a September 2011 Advisory Committee on Reactor Safeguards meeting, Dana Powers, a senior scientist at Sandia National Laboratories, expressed concern over the fact that hydrogen detonations occurred in the Fukushima Daiichi accident and stated that in experiments, "detonations are…extraordinarily hard to get."[127] However, neglecting to reassess hydrogen-combustion safety issues for the AP1000 after Fukushima, the NRC went ahead and issued licenses for two AP1000s in February 2012.

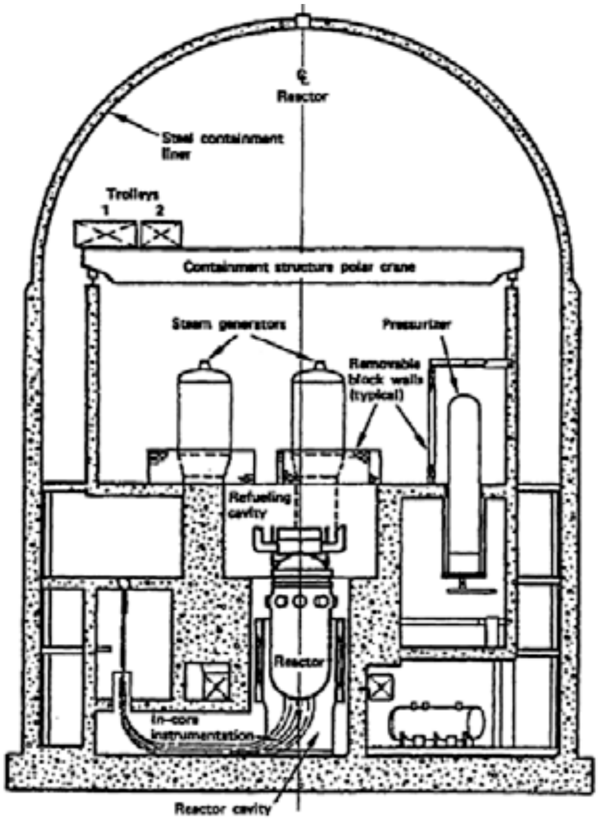
Paradoxically, two of the AP1000 containment's safety devices—hydrogen igniters, and passive autocatalytic hydrogen recombiner (PAR) units when they malfunction and behave like igniters—provide ignition sources that are capable of causing hydrogen detonations. In a severe accident, hydrogen igniters must be actuated at the correct time, because, as Peter Hoffman wrote in the *Journal on Nuclear Materials*: "[t]he concentration of hydrogen in the containment may be combustible for only a short time before detonation limits are reached."[128]

If AP1000 operators were to actuate the hydrogen igniters in an untimely fashion—after a local detonable concentration of hydrogen developed in the containment—it could cause a detonation. This especially could occur because Westinghouse's emergency response guidelines for the AP1000 are flawed: Operators are instructed to actuate hydrogen igniters when the core-exit gas temperature exceeds 1200°F. Westinghouse maintains that the core-exit temperature would reach 1200°F before the onset of the rapid zirconium-steam reaction of the fuel cladding,[129] which leads to thermal runaway in the reactor core; however, experimental data demonstrates that this would not necessarily be the case.

Westinghouse and the NRC, which approved the AP1000 design, both overlooked data—available for more than a quarter century—from the most realistic severe accident experiment conducted to date (LOFT LP-FP-2), in which core-exit temperatures were measured at approximately 800°F when maximum in-core fuel-cladding temperatures exceeded 3300°F. In LOFT LP-FP-2, when core-exit temperatures were 800°F, the rapid zirconium-steam reaction of the fuel cladding had already occurred and the reactor core had started melting down. Hence, relying on core-exit temperature measurements in an AP1000 severe accident could be unsafe: In a scenario in which operators re-flooded an overheated core simply because they did not know the actual condition of the core, hydrogen could be generated at rates as high as 5.0 kg per second. If operators were to actuate hydrogen igniters in such a scenario, it could cause a hydrogen detonation.

Westinghouse's general description of the AP1000 states that "[PARs] control hydrogen concentration following design basis events."[130] However, in the elevated hydrogen concentrations that occur in severe accidents, PARs are prone to malfunctioning and behaving like hydrogen igniters. This is a problem: AP1000 operators would not be able to switch off PARs, because they operate without electrical power. If the AP1000 containment's PAR units malfunctioned and incurred ignitions after a detonable concentration of hydrogen developed in the containment, it could cause a detonation.[131] This could occur in a number of severe accident scenarios, especially those in which the AP1000 containment's hydrogen igniter system was not operational,[132] enabling local detonable concentrations of hydrogen to develop in the containment.

Figure 13: Typical PWR Subatmospheric Reinforced Concrete Containment with Steel Liner (e.g., Diablo Canyon, North Anna, Surrey, Beaver Valley)



Source: NUREG/CR-6906/SAND2006-2274P, July 2006

The Uncertain Performance of Different Containment Designs in a Severe Accident Is Likely to Vary Widely

Figure 14 compares the calculated design pressure (in pounds per square inch above sea level atmospheric pressure, or "psig") of the six main types of U.S. commercial reactor containments with their net free volume in millions of cubic feet. BWR Mark I and II have a nominally strong pressure rating, due to their use of pressure-suppression pools, but very low free volume. The BWR Mark III and PWR ice condenser designs have the lowest design pressures of the group as well as moderate volumes, while the two other PWR containment designs have the largest volumes along with comparatively high design pressures.

The actual safety situation is more complex than reflected in this figure. In reality, no two reactor containments, even at the same facility, are exactly alike, and units of the same type can vary widely in their design and construction details. Predictions of local failure mechanisms, which could lead to significant leakage in an accident even before overall design pressures are exceeded, depend on the availability of accurate as- built information (geometry and material properties) at structural discontinuities (e.g., near containment doors or pipe and cable penetrations). "Even if this information is available (not typical for actual containments), the prediction, a priori, of local failures is at best an uncertain proposition.... Any evaluation of the capacity of an actual containment must be based on the entire system, including mechanical and electrical penetrations and other potential leak paths."

Source: NUREG/CR-6906/SAND2006-2274P, July 2006, p. xvii

B. PROBLEMS WITH CURRENT HYDROGEN-MITIGATION STRATEGIES FOR RESPECTIVE REACTOR DESIGNS

PWRs with Large Dry Containments and PWRs with Subatmospheric Containments

As noted above, the NRC does not require owners of PWRs with large dry containments and PWRs with sub-atmospheric containments to mitigate the hydrogen that would be generated in severe accidents; however, in severe accidents, it would be possible for the pressure spikes of hydrogen explosions to exceed the design pressures of such containments. The NRC has reported that hydrogen detonations could occur in PWRs with large dry containments and PWRs with sub-atmospheric containments. For example, a 1990 NRC letter to plant owners states that in severe accidents, local and global hydrogen detonations could occur in PWRs with large dry or sub-atmospheric containments.[133]

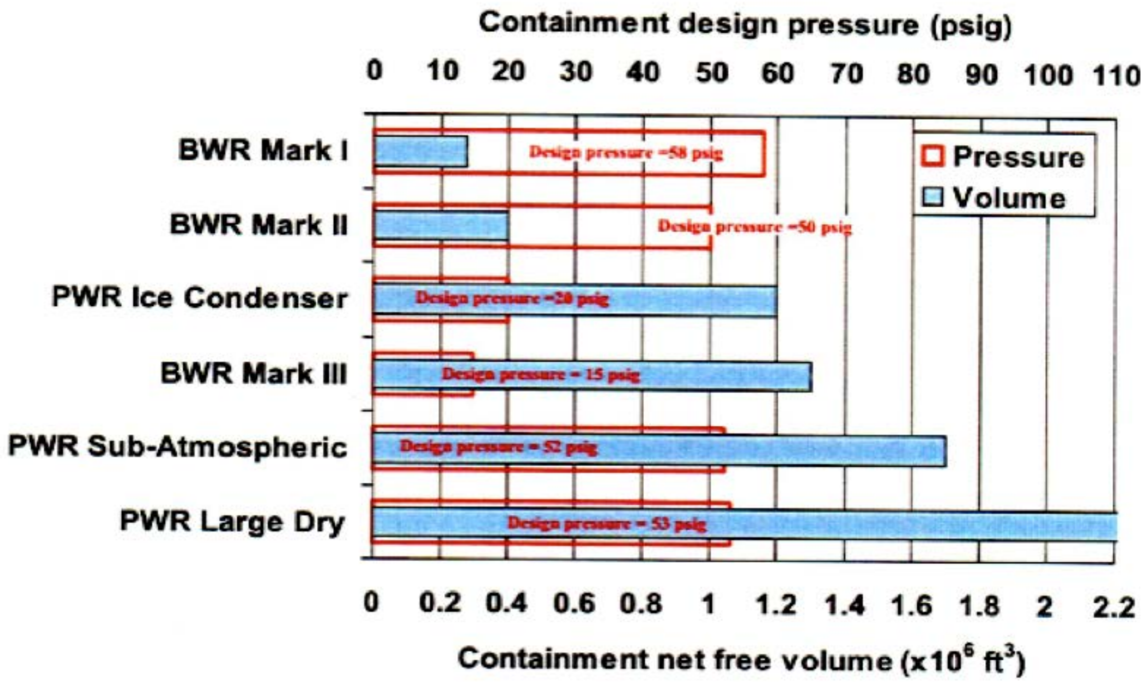
Furthermore, a 1991 report by Sandia National Laboratories cautions that in severe accidents, in which 75 percent of the fuel-cladding active length oxidized, detonable concentrations of hydrogen could develop in dry hydrogen-air mixtures in such containments. The report states that in a severe accident, steam typically would be present in the containment, yet the quantity of steam would be unpredictable because of condensation, which would be facilitated by containment spray systems. Detonations would most likely be initiated through deflagration-to-detonation transition, yet direct detonations could perhaps be possible at higher temperatures.[134]

Hydrogen recombiners would be prone to malfunctioning by incurring ignitions in the elevated concentrations that occur in severe accidents. This would be a serious problem: A recombiner's unintended ignition could cause a detonation.[135]

PARs could be advantageous in station-blackout accidents—a complete loss of grid-supplied and backup on-site alternating current power—because they operate without either external power or plant operator actuation; however, there is no way to prevent such recombiners from self-actuating or to shut them off in elevated hydrogen concentrations. Plant operators would be able to control the operation of electrically powered thermal hydrogen recombiners; yet operators should be cautious about actuating thermal recombiners in an accident. Plant operators should actuate thermal recombiners only if hydrogen concentrations are low and should deactivate them

Figure 14: Typical Containment Volume and Design Pressure for U.S. Nuclear Plants

As a general rule, low volumes make it more likely that design basis pressures will be exceeded in a severe accident.



Source: NUREG/CR-6906/SAND2006-2274P, July 2006

if hydrogen concentrations increase to dangerous levels. Of course, to soundly make such decisions, operators would need to ascertain local hydrogen concentrations throughout the containment, which would be especially difficult in the course of a fast-moving and/or chaotic accident scenario.

Among the PWRs in the United States that still have hydrogen recombiners installed, only one has PARs (Indian Point Unit 2); the others have thermal recombiners—typically two units in each containment. In Europe, some PWRs have from 30 to 60 PARs installed and distributed in their containments to help mitigate hydrogen in the event of a severe accident.[136] This is puzzling, given that such recombiners would be prone to behaving like igniters—malfunctioning by incurring ignitions—in elevated hydrogen concentrations.[137]

After intensive deliberation, European regulators decided not to require igniters in PWRs (those without ice condenser containments) because "[u]ncertainties were identified with respect to, among other aspects, hydrogen distribution and combustion behavior."[138] In line with the reasoning behind this decision, it seems that European regulators should also be hesitant about allowing PWRs to operate with PARs installed in their containments, because unintended ignitions from such recombiners would be neither predictable nor preventable in a severe accident.

Another problem with hydrogen recombiners is that in a severe accident, cesium iodide particles transported through them could be converted into volatile iodine, producing an additional source term of radiation exposure.[139]

PWRs with Ice Condenser Containments and BWR Mark III Containments

The NRC requires that PWRs with ice condenser containments and BWR Mark IIIs operate with hydrogen igniters installed in their containments in order to mitigate the hydrogen that would be generated in the event of a severe accident.[140] However, hydrogen igniters should be used only in cases where the effects of their use are entirely predictable, and predictions must indicate that the containment would not be threatened by any potential deflagrations arising from the deliberate ignition of hydrogen.[141]

Safety experts have questioned the safety of using igniters to mitigate hydrogen at certain times in some severe accident scenarios. For example, an OECD Nuclear Energy Agency report published in August 2000 states, "The main question in the application of the igniter concept is its safety orientation. The use of igniters should reduce the overall risk to the containment and should not create new additional hazards such as a local detonation."[142]

Another paper, published in 2006, states that "[w]ith early ignition, the hydrogen will be eliminated by slow combustion without high thermal and temperature loads, but with late ignition, hydrogen detonation transition will quickly occur with high local thermal and pressure loads which will threaten the integrity of the containment."[143]

A 1990 NRC letter to plant owners cautions that hydrogen igniters would be prevented from operating in station blackouts at PWRs with ice condenser containments and

Figure 15: Prestressed concrete containment vessel (PCCV) at the Ohi Unit 3 reactor in Japan

A 1:4 scale model of a prestressed concrete containment vessel (PCCV) at the Ohi Unit 3 reactor in Japan, undergoes a massive rupture in a 2001 Sandia Laboratory test at 3.63 times its design pressure (Pd), or 206.4 psig. The pressurized model had experienced leak rates in earlier tests, indicating functional failure at 2.4 times Pd.

Source: NNUREG/CR-6906/SAND2006-2274P, July 2006

BWR Mark IIIs. If hydrogen were not burned off, it could reach detonable concentrations; if power were then restored, the igniters could cause a hydrogen detonation.[144]

BWR Mark I and BWR Mark II Containments

Hydrogen generation is a serious problem for the small-volume, inerted BWR Mark I primary containment, because hydrogen is non-condensable at the temperatures expected in a nuclear power plant.[145] In a BWR severe accident, hundreds of kilograms of non-condensable hydrogen gas would be generated (potentially exceeding 3,000 kg[146]) at rates as high as 5,000 to 10,000 grams per second if there were a re-flooding of an overheated reactor core.[147] This would increase the internal pressure of the primary containment. If enough hydrogen were generated, the containment would likely first leak excessively before failing catastrophically from overpressurization.

A BWR Mark I primary containment is made up of a drywell shaped like an inverted lightbulb, which contains the reactor vessel, and a steel wetwell (also called a torus) shaped like a doughnut, which surrounds the base of the drywell. The drywell and wetwell are connected by large pipes. The wetwell is half filled with water (typically about 790,000 gallons[148])—and is sometimes referred to as a suppression pool. A BWR Mark II primary containment also has a drywell and wetwell (concrete), but these are shaped and oriented from their BWR Mark I counterparts.

In a severe accident, water already present or pumped into the reactor core to cool the fuel rods would heat up and produce thousands of kilograms of steam, which would enter the drywell of the primary containment. The water in the wetwell's suppression pool is intended to condense the steam and help absorb the heat released by the accident to reduce the pressure in the primary containment; as the steam pressure builds up in the drywell, steam vents downward into the wetwell through pipes, which terminate underwater in the suppression pool. (Without the condensation of the steam in the suppression pool, the relatively small primary containments of BWR Mark Is and Mark II units (often termed pressure suppression containments) would quickly fail from overpressurization.

However, the generation of sufficiently large quantities of non-condensable hydrogen gas in a severe accident could overwhelm the capacity of the primary containment. For example, there could be a severe accident scenario at a BWR Mark I in which there is a rapid accumulation of steam in the drywell and non-condensable gas (nitrogen[149] and hydrogen) in the wetwell; in such a scenario, the primary containment's pressure could rapidly increase "up to the venting and failure levels."[150]

Early BWRs Perform Poorly in Containment Leak-Rate Tests, Even When Liberal Test Protocols Allow Pretest Repairs to Supposedly "As Found" Condition of Seals and Valves

BWR Mark I and Mark II primary containments are designed to limit—not prevent—hydrogen leakage in accidents. In overall leak rate tests[151]—conducted below design pressure—such containments leak hundreds of pounds of air per day. For example, in 1999, tests conducted at Nine Mile Point Unit 1, a BWR Mark I, and Limerick Unit 2, a BWR Mark II, found that overall leakage rates at both units were in excess of 350 pounds of air per day,[152, 153] which is actually less than the maximum *allowed* leak rates.

This means that in a severe accident even if there were *no damage* to a primary containment, hydrogen would leak into the secondary containment (the reactor building); leak rates would increase as the internal pressure increased and would become even greater if the seals at the various piping and cable penetrations were damaged. (Typical BWR containments have 175 penetrations, almost twice as many as typical PWR containments.)[154]

Regarding reactor containments and hydrogen leakage, a 2011 IAEA report states:

> [N]o containment is fully leak tight, [hydrogen] will leak to the surrounding areas, which often have the function of secondary containment. … Hence, there is a certain risk that combustion may occur outside the primary containment. This may lead to combustion loads exerted on the containment from outside. Usually, containments have considerable margin against loads from inside, as they are in principle designed to carry the pressure loads from a large break LOCA. The pressure bearing capability for loads from outside can be substantially less…"[155]

In an accident, a mixture of hydrogen, nitrogen, and steam would leak from a BWR primary containment; as internal pressures increased and the accident progressed, the concentration of hydrogen in the leaking mixture would increase. If there were no damage to the primary containment, the quantity of hydrogen that leaked (by weight) would be relatively small, because hydrogen is about 14 times less dense than air.[156] However, a BWR secondary containment—which has a design pressure of approximately 3.0 psig[157]—could be breached if, for example, between 20 to 40 pounds of hydrogen were to leak into it, accumulate locally, and explode.

In a severe accident, it is highly probable that the seals at the penetrations of BWR Mark I and Mark II primary containments would become degraded (of course, some penetration-seals could already be degraded by material aging before the accident occurred.) A 1984 report from Brookhaven National Laboratory advises that severe accident risk estimates should consider "[t]he potential for containment leakage through penetrations prior to reaching" estimated containment failure pressures. The report further notes it is highly probable that the leakage of BWR Mark I and Mark II primary containments would "prevent overpressurization," and that "[f]ailure of non-metallic seals for containment penetrations (primarily equipment hatches, drywell heads, and purge valves) are the most significant sources of containment leakage."[158] BWR drywell heads, which have diameters between 30 to 40 feet, would most likely incur the highest leak rates in the containment as internal pressures increased.[159]

Containments have had leaks, exceeding allowable leakage rates, that lasted for many months—"primarily from large penetrations, such as the purge and vent valves, [main steam isolation valves, for BWRs only], and valves inadvertently left open."[160] In fact, BWR Mark I primary containments have failed a number of overall leak rate tests; for example, Oyster Creek—the oldest operating commercial reactor in the U.S., which is considered to be quite similar to Fukushima Daiichi Unit 1—has failed at least five tests.[161]

In one test, Oyster Creek's primary containment leaked at a rate that was 18 times greater than its design leak rate;[162] if this test was conducted at 35 psig, the same pressure as subsequent Oyster Creek tests,[163] which seems likely, the primary containment leaked at a rate in excess of 6800 pounds of air per day.[164] Such results beg the question: what were the pre-accident leak rates—*below design pressure*—of the three primary containments that leaked hydrogen at Fukushima Daiichi?

Since the Fukushima Daiichi accident, the problem of hydrogen leakage from primary containments has not been adequately addressed. (Mark II primary containments would also incur hydrogen leaks in severe accidents.) In fact, the NRC is currently preparing to reduce the frequency of both local and overall leak rate testing from once every five and once every 10 years, respectively, to once every 75 months and 15 years, respectively.

Remarkably, the current 10-year requirement already represents a loosening of the original leak-test intervals, which stood at 2.0 to 3.3 years prior to 1995, depending on the particular nature of the test.[165] In its safety analyses to assess extending the test intervals, the NRC has simply overlooked the fact that BWR Mark I and Mark II primary containments are vulnerable to hydrogen leakage. Moreover, as reactors approach and exceed their originally-licensed lifetimes of 40 years, one might intuitively conclude that the need for containment leak rate testing is actually increasing, not diminishing, in order to gauge the impact of aging penetration seals and isolation valves on containment integrity under a range of accident scenarios, including severe accidents.

Local leak rate tests of containment penetrations are *supposed* to be conducted as "as-found tests," meaning that the penetrations are not supposed to be repaired immediately before testing; however, NUREG/CR-4220 reports that all of the "NRC Senior Inspectors for containment systems [who] were contacted and asked to relate their experience with containment isolation system performance."[166] They stated that:

> [R]eported leakage rates often do not represent true leakage rates. Utilities are generally allowed to perform some minor repair on a valve prior to recording its "as-found" condition for a leakage test. Similarly, major repair (such as completely rebuilding a valve) is permitted prior to recording a valve's "as-left" condition at the end of its leakage test.[167]

Hence, around 1985 when NUREG/CR-4220 was published, it was a *common practice* for utilities to make minor repairs on valves immediately before recording their "as-found" leak rates. The local leak rate tests that are intended to measure leakage rates at containment isolation valves are termed "Type C tests." In September 1995, the NRC extended Type C test intervals from two years to five years. Interestingly, the failure rates of Type C "as-found" tests have decreased by about *one order of magnitude* since the test intervals for such tests were increased in 1995.[168] Such significant improvements beg the question: since 1995, to what degree have valves been repaired immediately before recording their "as-found" leak rates?

NUREG/CR-4220 states that one of the NRC Senior Inspectors "indicated that Types B and C tests [local leak rate tests] are performed before Type A [overall leak rate test], enabling repairs to be made sothat the Type A test can be passed easily."[169]

In a March 2013 ACRS meeting, an ACRS member similarly observed that "[i]f they did all their preparations perfectly, they would never fail."[170] It is clear that overall leak rate tests and local leak rate tests would provide a far more accurate assessment of pre-existing containment leak rates if repairs were not allowed to be made immediately before testing.

A report from the Electric Power Research Institute's (EPRI), "Risk Impact Assessment of Extended Integrated

Table 6: Historical Reactor Containment Integrated Leak Rate Test (ILRT) Failures Even with Test Protocol Allowing Pre-Test Repairs (circa 1985)

Reactor Name	Type	Test No.	Reactor Name	Type	Test No.
Beaver Valley-1	PWR	1	Oconee-1	PWR	1
Beaver Valley-1	PWR	2	Oconee-2	PWR	2
Big Rock Point	BWR	7	Oconee-3	PWR	1
Browns Ferry-1	BWR	1	Oyster Creek	BWR	1
Browns Ferry-2	BWR	2	Oyster Creek	BWR	3
Brunswick-2	BWR	1	Oyster Creek	BWR	4
Brunswick-2	BWR	2	Oyster Creek	BWR	5
Cal.Cliffs-1	PWR	1	Oyster Creek	BWR	6
Cook-1	PWR	2	Palisades	PWR	1
Dresden-1	BWR	1	Palisades	PWR	2
Dresden-1	BWR	6	Peach Bottom-2	BWR	1
Dresden-1	BWR	7	Peach Bottom-3	BWR	1
Dresden-3	BWR	2	Peach Bottom-3	BWR	2
Duane Arnold	BWR	1	Pilgrim-1	BWR	3
Fitzpatrick	BWR	1	Pilgrim-1	BWR	4
Fort Calhoun	PWR	2	Quad Cities-1	BWR	2
Ginna	PWR	2	Rancho Seco	PWR	1
Hatch-1	BWR	1	Robinson-2	PWR	2
Indian Point-1*	PWR	1	San Onofre-1	PWR	1
Indian Point-3	PWR	1	San Onofre-1	PWR	4
LaCrosse	BWR	1	Surry-2	PWR	1
LaCrosse	BWR	2	Surry-2	PWR	2
LaCrosse	BWR	6	Surry-2	PWR	3
LaCrosse	BWR	7	Three Mile Island-1	PWR	1
Millstone-1	BWR	2	Three Mile Island-1	PWR	2
Millstone-1	BWR	3	Turkey Point-4	PWR	1
Monticello	BWR	2	Turkey Point-4	PWR	2
Monticello	BWR	3	Vermont Yankee	BWR	1
Nine Mile Point-1	BWR	3	Vermont Yankee	BWR	2
Nine Mile Point-1	BWR	4			

* Note two failed ILRTs reported so count twice

Source: P.J. Pelto et al., *Reliability Analysis of Containment Isolation Systems*, Pacific Northwest Laboratory, NUREG/CR 4220, June 1985, available at: NRC's ADAMS Documents, Accession Number: ML103050471

Leak Rate Testing Intervals,"[171] has been used by the NRC to help justify the extension of testing intervals.[172] However, this report overlooked the fact that in severe accidents, BWR Mark I and Mark II primary containments leak explosive hydrogen gas into secondary containments. A second major problem with EPRI's report is that its list of overall leak rate test failures does not include the majority of test failures reported in NUREG/CR-4220. NUREG/CR-4220 lists a total of 60 overall (integrated) leak rate tests that failed before March 1985;[173] in fact, NUREG/CR-4220 also reports that when considering the results of local leak rate tests that failed with excessive leakage rates, the number of overall leak rate tests that failed is a total of 109.[174]

By contrast, EPRI's report lists *a total of nine* "containment leakage or degradation events" that occurred before March 1985.[175] Regarding its methodology for assessing the risk impact of extended test intervals, EPRI's report states "The first step is to obtain current containment leak rate testing performance information. … *This information is used to develop the probability of a pre-existing leak in the containment* using the Jeffreys Non-Informative Prior statistical method" [emphasis added].[176] Clearly, the NRC needs to review a large portion of the existing data that EPRI overlooked and reassess the risk impact of extended test intervals.

In a severe accident, any primary containment in a condition that would cause it to fail a leak-rate test would leak dangerous quantities of explosive hydrogen gas into a reactor building, *even at below design pressure*; however, the NRC does not seem concerned about excessive leakage rates. A 1995 NRC report[177] "concluded that…increasing allowable leakage rates by 10 to 100 times results in a *marginal risk increase*, while reducing costs by about 10 percent"[178] [emphasis added]. And a 1989/1990 NRC report[179] concluded that even if there is a containment leakage of 100 percent per day, "the calculated individual latent cancer fatality risk is below the NRC's safety goal."[180] Clearly, this safety goal would not be achieved if leaking hydrogen were to detonate in secondary containments, as it did at Fukushima Daiichi.

In March 2013, the NRC stated that "[s]ensitivity analyses in NUREG-1493 and other studies show that *light water reactor accident risk is relatively insensitive to the containment leakage rate* because the risk is dominated by accident sequences that result in failure or bypass of containment"[181] [emphasis added]. The progression of the Fukushima Daiichi accident was certainly affected by the leakage of hydrogen gas. In fact, it is possible that Unit 3's primary containment *did not fail* before hydrogen leaked into the Unit 3 secondary containment and detonated.

The internal pressure of Unit 3's primary containment actually *increased* after the hydrogen explosion occurred. The explosion occurred on March 14 at 11:01 am, then at 12:00 pm the primary containment's pressure started increasing from 52.2 psia to 53.7 psia, at 4:40 pm the pressure started decreasing from 69.6 psia, and at 8:30 pm the pressure started increasing from 52.2 psia.[182] In the Fukushima Daiichi accident, the BWR Mark I primary containments of Units 1, 2, and 3 incurred internal pressures that exceeded the loads they were designed to sustain. According to an INPO report published in November 2011, the highest recorded internal pressures in the primary containments of Units 1, 2, and 3 were approximately 1.7, 1.7, and 1.4 times greater than their design pressures, respectively.[183] (In the accident, hydrogen leaked from the primary containments—according to INPO: "most probably" at the penetrations[184]—of Units 1, 2, and 3 and detonated in the secondary containments of Units 1, 3, and 4.) The NRC has stated that in the circumstances of the Fukushima Daiichi accident, it is reasonable to conclude that BWR Mark IIs would also incur devastating consequences, "because Mark II containment designs are only slightly larger in volume than Mark I containment designs[185] and also use wetwell pressure suppression."[186]

Reliable Hardened Vents

In an attempt to resolve the problems of BWR Mark I and Mark II primary containment overpressurization and decay heat removal, in March 2012, the NRC ordered that reliable hardened vents be installed in BWR Mark Is and Mark IIs by December 31, 2016.[187] (As stated above, in September 1989, the NRC had tried to solve the same problems by issuing non-legally binding guidance to all the owners of BWR Mark Is, *recommending*[188] that hardened vents be installed in Mark Is.[189]) The NRC's order stipulates a number of performance objectives and features that a new design of a hardened vent must have; for example, "shall include a means to prevent inadvertent actuation."[190]

It could be difficult to design a hardened vent that would perform well in scenarios in which there were rapid containment-pressure increases. A 1988 report by the Committee on the Safety of Nuclear Installations report states that "[f]iltered venting is less feasible for those sequences resulting in early over-temperature or overpressure conditions. This is because the relatively early rapid increase in containment pressure requires large containment penetrations for successful venting."[191] This indicates that a reliable hardened vent's piping will likely need a diameter and thickness greater than what has been voluntarily installed at BWR Mark I containments in the United States.[192]

If a hardened vent were designed for passive operation by means of a rupture disk, in place of a remotely or manually actuated valve, venting would occur if a predetermined threshold pressure were reached. A reliable passive venting capability could be beneficial in severe accident scenarios that have rapid containment pressure increases. However, a 1983 Sandia National Laboratories manual cautions that "it may be difficult to design vents that can handle the rapid transients involved" in a severe accident.[193]

It is important to consider that in the Fukushima Daiichi accident, the particular design of the installed vents may have caused the accident to be worse than it would have been without their use: The INPO report of November 2011 states that "it is postulated that the hydrogen explosion in the Unit 4 reactor building was caused by hydrogen from Unit 3."[194] Unit 3 and Unit 4's containment vent exhaust piping was interconnected, so hydrogen may have been vented from Unit 3 to Unit 4's secondary containment,[195] where it detonated.

In severe accidents, spent fuel pools are vulnerable to the hydrogen explosions that can occur in BWR Mark I and Mark II secondary containments. Spent fuel pools, which store fuel assemblies after they are discharged from the reactor core, are located in the secondary containment of these designs, elevated about 70 to 80 feet above ground level. If a spent fuel pool were compromised by a hydrogen explosion, it could cause large radiological releases.

Some thought initially that the explosion that occurred in Fukushima Daiichi Unit 4 at 6:00 am on March 15, 2011—3.63 days after the March 11, 2011 earthquake—could have been caused by the detonation of hydrogen gas generated by the reaction of steam with the zirconium cladding of fuel rods stored in the spent fuel pool. Subsequent investigations indicated that this was not the case.

However, according to a 2012 ORNL paper, the hydrogen that detonated *could* have come from the Unit 4 pool's fuel assemblies reacting with steam: If there were a loss of spent fuel pool cooling, the water in the pool would be heated by the fuel rods' decay heat until it reached the boiling point; then the water would boil away, uncovering the fuel rods. ORNL computer analyses found that in this scenario, a total of 1,800 kg to 2,050 kg of hydrogen could have been generated. The analyses also found that 150 kg of hydrogen—an amount that could have caused the Unit 4 explosion—would have been generated 3.63 days after the accident commenced if the initial water level in the pool were 4.02 meters (at the top of the active length of the fuel rods).[196]

The NRC does not require that hydrogen be mitigated in the secondary containments of BWR Mark I and Mark II sites in severe accidents. This is a problem, because hydrogen could leak into secondary containments and explode, as occurred in the Fukushima Daiichi accident. The Fukushima Daiichi accident demonstrated that BWR Mark I secondary containments—essentially ordinary industrial buildings with design pressures of approximately 3.0 psig[197]—cannot withstand hydrogen explosions. (BWR Mark II secondary containments also have low design pressures.) In line with the NRC's approach to safety through defense-in-depth,[198] the Fukushima Daiichi accident scenario of hydrogen leaking from overpressurized primary containments and/or hardened vent systems should be considered as likely to occur again, in the event of a severe accident at either a BWR Mark I or BWR Mark II.

C. MONITORING CORE DEGRADATION AND HYDROGEN GENERATION IN SEVERE ACCIDENTS

In a severe accident, plant operators would need equipment that effectively monitored evolving conditions; information, such as temperatures in the reactor core and hydrogen concentrations in the containment, would help them manage an accident and implement hydrogen mitigation. Without accurate and prompt core and containment diagnostics, plant operators would not be able to properly manage an accident. Unfortunately, some of the current methods of monitoring core and containment diagnostics are inadequate.

Monitoring Core Degradation

In a severe accident involving a PWR, the primary tool used to detect inadequate core cooling and uncovering of the core would be coolant temperature measurements taken with core-exit thermocouples (temperature measuring devices) at a point above the active length of the fuel rods. In many cases, a predetermined core-exit thermocouple measurement would be used to signal the time for PWR operators to transition from emergency operating procedures (EOP) to severe accident management guidelines (SAMG). The NRC's Near-Term Task Force report states that "EOPs typically cover accidents to the point of loss of core cooling and initiation of inadequate core cooling (*e.g.*, core exit temperatures in PWRs greater than 649 degrees Celsius [1,200 degrees Fahrenheit])."[199]

Experimental data indicates that core-exit thermocouple measurements would not be an adequate indicator for when to safely transition from EOPs to SAMGs.[200] Two of the main conclusions from experiments are: 1) that core-exit temperature measurements display *in all cases* a significant delay (up to several hundred seconds) and: 2) that core-exit temperature measurements are *always* significantly lower (up to several hundred degrees Celsius) than the actual maximum cladding temperature.[201] In an experiment simulating a severe accident—LOFT LP-FP-2—core-exit temperatures were measured at approximately 800°F when in-core fuel-cladding temeratures exceeded 3300°F.[202]

In a severe accident, plant operators are supposed to implement SAMGs before the onset of the rapid zirconium-steam reaction, which leads to thermal runaway in the reactor core. Clearly, using core-exit thermocouple measurements in order to detect inadequate core cooling or uncovering of the core would be neither reliable nor safe. For example, PWR operators could end up re-flooding an overheated core simply because they did not know the actual condition of the core. Unintentionally re-flooding an overheated core could generate hydrogen, at rates as high as effectiveness."[203]

Core-exit thermocouples are not installed in BWRs. In a severe accident involving this type of reactor, plant operators are supposed to detect inadequate core cooling or uncovering of the core by measuring the water level in the reactor core. However, after the onset of core damage BWR reactor water level measurements are unreliable; and can read erroneously high in low-pressure accidents, like large-break LOCAs, and when there are high drywell temperatures.[204]

In the Fukushima Daiichi accident, plant operators did not know the actual condition of the reactor cores of Units 1, 2, and 3. In a December 2011 article, Saloman Levy—a former GE engineer-manager for BWRs[205]—stated his judgment that in the Fukushima Daiichi accident, plant operators should have recognized that water level measurements were unreliable and that reactor and containment pressures as well as the wetwell water temperature would be superior indicators of the state of the core. According to Levy, "The reactor and the containment pressures will rise faster when hydrogen is produced. Increased reactor and containment pressure rates and wetwell [water] temperature rises confirm accelerated core melt."[206] Yet what Levy recommends is not a solution to the problem of identifying the correct time to transition to SAMGs in a BWR severe accident, because *the rapid zirconium-steam reaction would have already commenced by the time operators confirmed an accelerated core melt.*

MONITORING FOR THE PRESENCE OF OXYGEN AND HYDROGEN

The NRC requires that BWR Mark I and Mark II units operate with oxygen monitors installed in their primary containments in order to confirm that the containment remains inerted during operation. In a severe accident, if a primary containment were to become de-inerted, "severe accident management strategies, such as purging and venting, would need to be considered."[207]

The NRC also requires that all licensed plants operate with the ability to monitor hydrogen concentrations in their containments. However, in 2003, the NRC reclassified hydrogen monitors (and oxygen monitors) as "non-safety-related" equipment,[208, 209] meaning that this equipment does not have to undergo full qualification (including seismic qualification), does not have redundancy, and does not require onsite (standby) power.

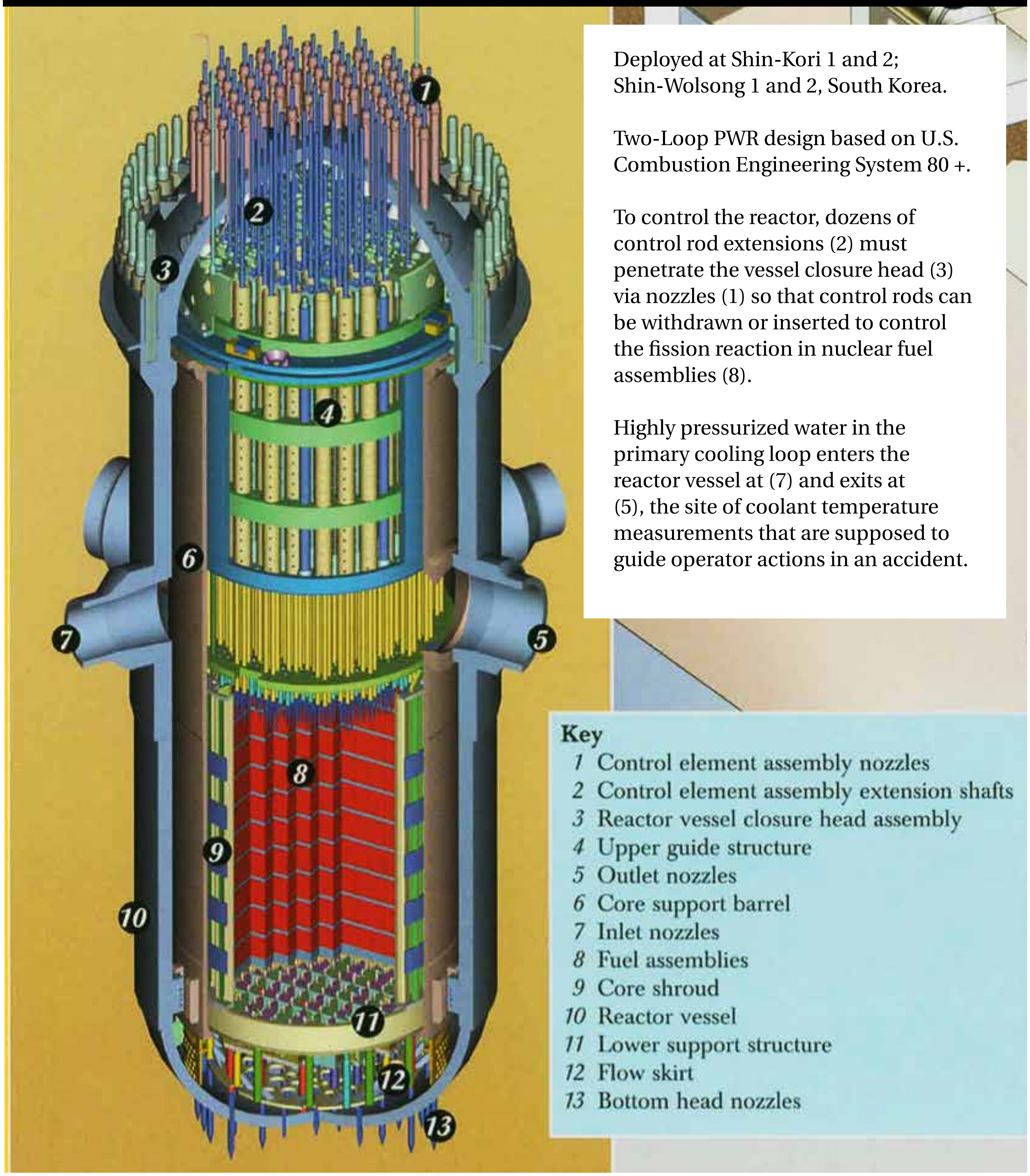
In severe accidents, hydrogen monitors would be used to help assess the degree of core damage that had occurred and to help with accident management. For example, BWR Mark IIIs use hydrogen monitors to help guide emergency operating procedures: Hydrogen igniters would not be used In scenarios in which hydrogen reached concentrations that would threaten containment integrity if the hydrogen were to combust.

BWR Mark I and Mark IIs operate with hydrogen monitors installed in their inerted primary containments yet do not have such monitors in their secondary containments. David Lochbaum of the Union of Concerned Scientists has cautioned that "[t]he inability to monitor hydrogen concentrations could cause [plant] operators to not vent [BWR Mark I and Mark II] reactor buildings, thus leading to ignitions resulting in loss of secondary containment integrity." He states further that without the ability to monitor hydrogen, operators could "preemptively vent the reactor buildings when it was not necessary to do so," which would also cause radioactive releases.[210]

In 1983, the NRC issued an order requiring that in a severe accident, hydrogen monitors function within 30 minutes after coolant water is injected into the reactor vessel; in 1998, the NRC "determined that the 30-minute requirement can be overly burdensome" and imposed a 90-minute requirement, instead.[211] The NRC seems to believe that all severe accidents would be slow-moving station-blackout accidents—a complete loss of grid-supplied and backup onsite alternating current power—like the Fukushima Daiichi accident; it does not consider that fast-moving accidents are also possible.

Despite Fukushima Daiichi's three devastating hydrogen explosions, the NRC has relegated severe-accident hydrogen safety issues to the *least proactive* stage of its post-Fukushima regulatory responses to the accident (termed "Tier 3"). NRDC believes that the NRC should reconsider its approach and promptly address severe accident safety issues involving hydrogen. In this section we outline a number of safety initiatives that the NRC should pursue to reduce the risk of hydrogen explosions in severe accidents.

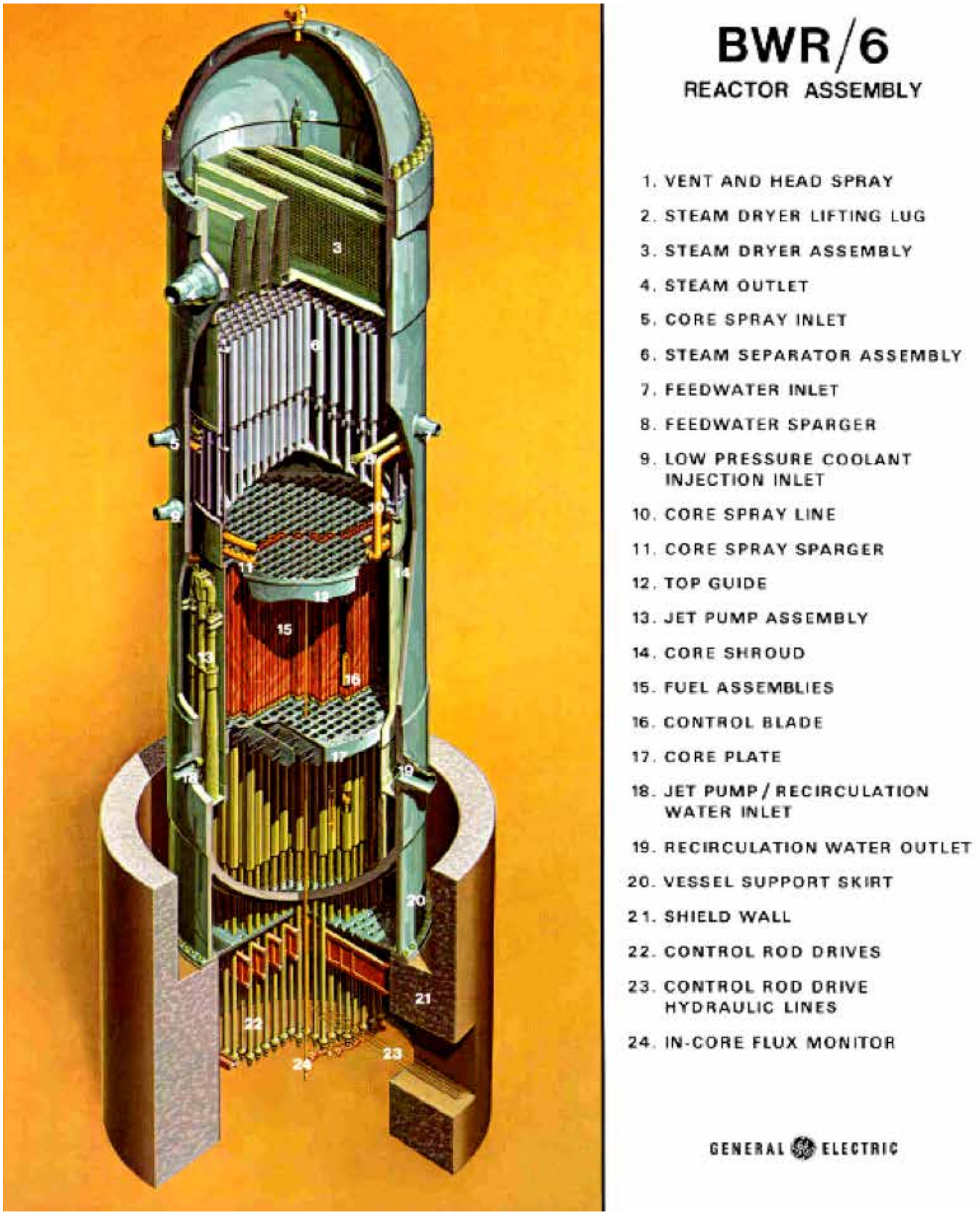
Figure 16: Cutaway View of PWR Pressure Vessel and Core of Korean Standard Nuclear Power Plant Plus (KSNP +)



Source: econtent.unm.edu/cdm/search/collection/nuceng

Figure 17: GE Boiling Water Reactor (BWR) Model 6 Reactor Vessel

Note that control rod blades on the bottom must be hydraulically driven upward into the core, rather than dropping from above as they do in a PWR.



Source: USNR Technical Training Center Reactor Concepts Manual: Boiling Water Reactor (BWR) Systems, www.nrc.gov/reading-rm/basic-ref/teachers/03.pdf

Figure 18

Source: Reactor Concepts Manual, Boiling Water Reactor Systems, USNRC, Technical Training Center, www.nrc.gov/reading-rm/basic-ref/teachers/03.pdf

V. NRDC'S RECOMMENDATIONS FOR REDUCING THE RISK OF HYDROGEN EXPLOSIONS IN SEVERE NUCLEAR ACCIDENTS

A. DEVELOP AND EXPERIMENTALLY VALIDATE COMPUTER SAFETY MODELS THAT WOULD BE CAPABLE OF CONSERVATIVELY PREDICTING RATES OF HYDROGEN GENERATION IN SEVERE ACCIDENTS

The NRC needs to acknowledge that its existing computer safety models underpredict the rates of hydrogen generation that occur in severe accidents. The NRC should conduct a series of experiments with multi-rod bundles of zirconium alloy fuel rod simulators and/or (actual) fuel rods as well as study the full set of existing experimental data. The NRC's objective in this effort should be to develop models capable of predicting with greater accuracy the rates of hydrogen generation that occur in severe accidents.

B. ASSESS THE SAFETY OF EXISTING HYDROGEN RECOMBINERS, AND POTENTIALLY DISCONTINUE THE USE OF PARS UNTIL TECHNICAL IMPROVEMENTS ARE DEVELOPED AND CERTIFIED

Experimentation and research should be conducted in order to improve the performance of PARs so that they would not malfunction and incur ignitions in the elevated hydrogen concentrations that occur in severe accidents. Some experimentation and research has already been conducted; however, the problem of PARs incurring ignitions in elevated hydrogen concentrations remains unresolved.

The NRC and European regulators should also perform safety analyses to determine if existing PARs should be removed from plant containments. It is possible such analyses would find that removing PARs would help improve safety in the event of a severe accident. Until PARs are developed that do not pose a risk of ignitions in elevated hydrogen concentrations, the NRC and European regulators should also review whether to replace PARs with electrically powered thermal hydrogen recombiners. However, this could prove costly, and thermal hydrogen recombiners would not function in a station-blackout accident unless provided with their own independent train of emergency power.

In a severe accident, plant operators would be able to turn off thermal recombiners in order to prevent them from operating in elevated hydrogen concentrations. However, to safely operate thermal recombiners, operators would be required to have instrumentation providing timely information on the local hydrogen concentrations throughout the containment.

C. SIGNIFICANTLY IMPROVE EXISTING OXYGEN AND HYDROGEN MONITORING INSTRUMENTATION

The NRC should reclassify oxygen and hydrogen monitors as safety-related equipment that has undergone full qualification (including seismic qualification), has redundancy, and has its own independent train of emergency electrical power. These recommendations are in accordance with the conclusions of the NRC's Advisory Committee on Reactor Safeguards (ACRS), which stated that "[t]he experience at Fukushima showed that essential reactor and containment instrumentation should be enhanced to better withstand beyond-design basis accident conditions" and that "[r]obust and diverse instrumentation that can better withstand severe accident conditions is needed to diagnose, select, and implement accident mitigation strategies and monitor their effectiveness."[212]

The NRC should require that, after the onset of a severe accident, hydrogen monitors be functional within a time frame that enables timely detection of quantities of hydrogen indicative of core damage and a potential threat to containment integrity. The current requirement that hydrogen monitors be functional within 90 minutes of the injection of coolant water into the reactor vessel is clearly inadequate for protecting public and plant worker safety.

NRDC supports the Union of Concerned Scientists' request to the NRC regarding hydrogen-monitoring instrumentation. The NRC should require that hydrogen monitoring instrumentation be installed in 1) BWR Mark I and Mark II secondary containments, 2) the fuel handling buildings of PWRs and BWR Mark IIIs, and 3) any other plant structure where it would be possible for hydrogen to enter.

D. UPGRADE CURRENT CORE DIAGNOSTIC CAPABILITIES IN ORDER TO BETTER SIGNAL TO PLANT OPERATORS THE CORRECT TIME TO TRANSITION FROM EMERGENCY OPERATING PROCEDURES TO SEVERE ACCIDENT MANAGEMENT GUIDELINES

The NRC should require plants to operate with thermocouples placed at different elevations and radial positions throughout the reactor core to enable plant operators to accurately measure a wide range of temperatures inside the core under both typical and accident conditions. In the event of a severe accident, in-core thermocouples would provide plant operators with crucial information to help them track the progression of core damage and manage the accident—for example, indicating the correct time to transition from EOPs to SAMGs.

E. REQUIRE ALL NUCLEAR POWER PLANTS TO CONTROL THE TOTAL QUANTITY OF HYDROGEN THAT COULD BE GENERATED IN A SEVERE ACCIDENT

The NRC should require all PWRs (with large dry containments, subatmospheric containments, and ice condenser containments) and BWR Mark IIIs to operate with systems for combustible gas control that would effectively and safely control the total quantity of hydrogen that could potentially be generated in different severe accident scenarios (this value is different for PWRs and BWRs). The NRC should also require the same for BWR Mark I and Mark II unless it is demonstrated that venting (without causing significant radiological releases) their inerted containments would effectively and safely control the hydrogen generated in severe accidents. Systems for combustible gas control also need to effectively and safely control the total quantity of hydrogen that could potentially be generated at all times throughout different severe accident scenarios, taking into account the potential rates of hydrogen generation.

Additionally, the NRC should require all PWRs and BWR IIIs to operate with systems for combustible gas control that would be capable of preventing local concentrations of hydrogen in the containment or other structures from reaching levels that would support combustions, deflagrations, or detonations that could cause a loss of containment integrity and/or necessary accident mitigating features.

Furthermore, the NRC should require licensees of PWRs with ice condenser containments and BWR Mark IIIs (and any other nuclear power plants that would operate with hydrogen igniter systems) to perform analyses demonstrating that their hydrogen igniter systems would effectively and safely mitigate hydrogen in different severe accident scenarios. Licensees unable to do so should be ordered to upgrade their systems to adequate levels of performance.

F. REQUIRE THAT DATA FROM LEAK RATE TESTS BE USED TO HELP PREDICT THE HYDROGEN LEAK RATES OF THE PRIMARY CONTAINMENT OF EACH BWR MARK I AND MARK II LICENSED BY THE NRC IN DIFFERENT SEVERE ACCIDENT SCENARIOS

The NRC should require that data from overall leak rate tests and local leak rate tests—already required by Appendix J to Part 50 for determining how much radiation would be released from the containment in a design basis accident—be used to help predict hydrogen leak rates from the primary containment of each BWR Mark I and Mark II licensed by the NRC under different severe accident scenarios. If data from an individual leak rate test indicates that dangerous quantities of explosive hydrogen gas would leak from a primary containment in a severe accident, the plant owner would be required to repair the containment.

NRDC also recommends that the NRC require that overall leak rate tests and local leak rate tests be conducted without allowing repairs to be made immediately before the testing of potential leakage paths, such as containment welds, valves, fittings, and components which penetrate containment.[213]

Additionally, NRDC recommends that the NRC reevaluate its plan to extend the intervals of overall and local leak rate tests to once every 15 years and 75 months, respectively.[214] (There are two types of local leak rate tests; Type B is required at least once every 10 years.) The NRC needs to conduct safety analyses that take into account the relatively greater vulnerability of BWR Mark I and Mark II primary containments to hydrogen leakage. It is probable that the intervals between leak rate tests would need to be shortened rather than extended.

The NRC also needs to consider that in the past it was a common practice to make repairs to valves immediately before conducting "as found" local leak rate tests. Clearly, such tests do not provide accurate assessments of preexisting containment leak rates. The NRC needs to investigate whether repairs have been recently made immediately before conducting "as found" tests. More important, the NRC needs to fully integrate into its regulatory role the fact that in the Fukushima Daiichi accident, hydrogen leaked from the primary containments of Units 1, 2, and 3 and detonated in the secondary containments of Units 1, 3, and 4, causing large radiological releases.

ENDNOTES

1 In this report we frequently refer to "severe" nuclear accidents: i.e., accidents in which there is severe damage to the reactor core—for example, a partial core meltdown. A severe nuclear accident could be caused by a natural disaster, mechanical failure, or plant operator errors. The accidents at Three Mile Island Unit 2, Chernobyl Unit 4, and Fukushima Daiichi Unit 1, 2, and 3 were all severe accidents.

2 As nuclear safety expert David Lochbaum has noted, "Secondary containment is designed to have limited leakage…into the reactor building. The secondary containment leak test entails starting the standby gas treatment system. This system features fans, ductwork, dampers, and filter trains that draw air from the reactor building and refueling floors. This filtered air is discharged via an elevated release point. The filter trains are tested periodically to see if they remove over 99% of the radioactive particles from the discharge stream." Note to author from David L. Lochbaum, nuclear safety expert with the Union of Concerned Scientists, 01-06-2014.

3 Since hydrogen is a noncondensable gas, it will accumulate in the air space above the water surface of the suppression pool. When the differential pressure between the drywell and wetwell gets too great, vacuum breakers open automatically to transport hydrogen gas from the wetwell into the drywell, where it can accumulate or leak out into the surrounding reactor building.

4 Note to author from David L. Lochbaum, nuclear safety expert with the Union of Concerned Scientists, January 6, 2014.

5 This request to the NRC was first made by the Union of Concerned Scientists.

6 Typical operating BWR and PWR coolant pressures are approximately 1000–1050 pounds per square inch (psi) and approximately 2250 psi, respectively. See International Atomic Energy Agency (IAEA), "Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: BWR Pressure Vessels," IAEA-TECDOC-1470, October 2005, p. 7; and IAEA, "Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: PWR Pressure Vessels," IAEA-TECDOC-1120, October 1999, p. 5.

7 The NRC's definition of the reactor coolant system: The system used to remove energy from the reactor core and transfer that energy either directly or indirectly to the steam turbine. See www.nrc.gov/reading-rm/basic-ref/glossary/reactor-coolant-system.html.

8 Typical operating BWR and PWR coolant temperatures are 540°–550°F and 540°–620°F, respectively. See IAEA, "Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: BWR Pressure Vessels," IAEA-TECDOC-1470, October 2005, p. 7; and IAEA, "Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: PWR Pressure Vessels," IAEA-TECDOC-1120, October 1999, p. 5.

9 For consistency, this report will use the term *zirconium* to refer to all the various types of zirconium alloys that make up fuel cladding. Zircaloy, ZIRLO, and M5 are particular types of zirconium alloy fuel cladding. In a LOCA environment, the oxidation behavior of the different fuel cladding materials, with various zirconium alloys, would be similar because of their shared zirconium content.

10 The NRC's definition of a design basis accident: A postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety. See www.nrc.gov/reading-rm/basic-ref/glossary/design basis-accident.html.

11 The NRC states that "beyond design basis accident" is a term "used as a technical way to discuss accident sequences that are possible but were not fully considered in the design process because they were judged to be too unlikely. (In that sense, they are considered beyond the scope of design basis accidents that a nuclear facility must be designed and built to withstand.)" See www.nrc.gov/reading-rm/basic-ref/glossary/beyond-design basis-accidents.html.

12 The coolant water slows down or "moderates" the kinetic energy of the neutrons produced by fission, enabling a self-sustaining fission reaction in the uranium isotope 235U, which makes up about 4 percent of the uranium in the fuel.

13 In a PWR, fuel rod temperatures could exceed 1830°F within 60 seconds; at a BWR, fuel rod temperatures could exceed 1830°F within three minutes.

14 The equation for the reaction is written as $Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$ + energy. The energy (heat) generated by the reaction is about 6.5 megajoules per kilogram (kg) of Zr reacted.

15 Randall O. Gauntt, Sandia National Laboratories, email to Jason Schaperow of NRC, "Re: Cladding Behavior Under Steam and Air Conditions," January 31, 2000, available at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML010680338.

16 In the TMI-2 accident, cooling water was discharged from the pilot-operated relief valve, which was stuck open.

17 Robert E. Henry held research positions at Argonne National Laboratory during the decade leading up to the TMI-2 accident and was associate director of the Reactor Analysis and Safety Division at Argonne when he became involved in the evaluation of the TMI-2 accident, as part of a group formed by the Electric Power Research Institute's Nuclear Safety Analysis Center (NSAC).

18 Robert E. Henry, presentation slides from "TMI-2: A Textbook in Severe Accident Management," 2007 ANS/ENS International Meeting, November 11, 2007; seven of these presentation slides are in Attachment 2 of the transcript from "10 C.F.R. 2.206 Petition Review Board Re: Vermont Yankee Nuclear Power Station," July 26, 2010, available at: ADAMS Documents, Accession Number: ML102140405, Attachment 2.

19 Robert E. Henry, presentation slides from "TMI-2: A Textbook in Severe Accident Management."

20 It is acknowledged that runaway oxidation occurred in the TMI-2 accident; however, the temperature at which it commenced is unknown, because there is no thermocouple data from the hot spots of the fuel assemblies. NRDC does not intend to present Robert E. Henry's postulation that runaway oxidation of zirconium cladding by steam commenced at 1832°F in the TMI-2 accident as evidence that a runaway reaction did in fact commence at 1832°F.

21 Robert E. Henry, presentation slides from "TMI-2: A Textbook in Severe Accident Management."

22 NRC, "Feasibility Study of a Risk-Informed Alternative to 10 CFR 50.46, Appendix K, and GDC 35," June 2001, available at: ADAMS Documents, Accession Number: ML011800519, p. 3-1.

23 Peter Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," *Journal of Nuclear Materials* 270, Nos. 1–2 (April 1, 1999), p. 205.

24 Sherrell R. Greene, Oak Ridge National Laboratory, "The Role of BWR Secondary Containments in Severe Accident Mitigation: Issues and Insights from Recent Analyses," 1988.

25 The regulation 10 C.F.R. § 50.46(b)(i) stipulates that in a postulated design basis accident, "[t]he calculated maximum fuel element cladding temperature shall not exceed 2200°F."

26 E. Bachellerie et al., "Generic Approach for Designing and Implementing a Passive Autocatalytic Recombiner PAR-System in Nuclear Power Plant Containments," *Nuclear Engineering and Design* 221, Nos. 1–3 (April 2003), p. 158 (hereinafter "Designing and Implementing a PAR-System in NPP Containments").

27 Atomic Energy Commission, "Safety Evaluation Report for Indian Point Nuclear Generating Unit No. 3," Docket No. 50-286, September 21, 1973, available at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML072260465, p. 6-10.

28 E. Bachellerie et al., "Designing and Implementing a PAR-System in NPP Containments," p. 158.

29 OECD Nuclear Energy Agency, "State-of-the-Art Report on Flame Acceleration and Deflagration-to-Detonation Transition in Nuclear Safety," NEA/CSNI/R(2000)7, August 2000, available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML031340619, p. 6.38 (hereinafter "Report on FA and DDT").

30 E. Bachellerie et al., "Designing and Implementing a PAR-System in NPP Containments," p. 158.

31 J. Starflinger, "Assessment of In-Vessel Hydrogen Sources," in *Projekt Nukleare Sicherheitsforschung: Jahresbericht 1999* (Karlsruhe: Forschungszentrum Karlsruhe, FZKA-6480, 2000).

32 OECD Nuclear Energy Agency, "In-Vessel Core Degradation Code Validation Matrix: Update 1996-1999," report by an OECD NEA Group of Experts, October 2000, p. 13.

33 IAEA, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," IAEA-TECDOC-1661, July 2011," p. 10 (hereinafter "Mitigation of Hydrogen Hazards in SA").

34 This estimate is based on that fact that large BWR cores and large PWR cores have up to approximately 800 and 200 fuel assemblies, respectively (see NRC, "Boiling Water Reactors" (located at: http://www.nrc.gov/reactors/bwrs.html) and NRC, "Pressurized Water Reactors" (located at: http://www.nrc.gov/reactors/pwrs.html)); and recent designs of BWR and PWR fuel assemblies have up to approximately 190 kg and 480 kg of initial uranium mass per assembly, respectively (see NRC, "Certificate of Compliance No. 1014," Appendix B, "Approved Contents and Design Features for the Hi-Storm 100 Cask System," (available at ADAMS No: ML13351A189), pp. 2.39, 2.44). Hence, large BWR cores and large PWR cores are estimated to have a total of approximately 152,000 kg and 96,000 kg of initial uranium mass, respectively.

35 BWRs and PWRs have up to approximately 800 and 200 fuel assemblies in their cores, respectively. NRC, "Boiling Water Reactors" (located at: http://www.nrc.gov/reactors/bwrs.html) and NRC, "Pressurized Water Reactors" (located at: http://www.nrc.gov/reactors/pwrs.html).

36 Recent designs of BWR and PWR fuel assemblies have on the order of 96 and 264 fuel rods per assembly, respectively. Hence BWR and PWR cores can have up to approximately 76,800 and 52,800 fuel rods per core, respectively; so BWRs cores can have approximately 45 percent more fuel rods. NRC, "Certificate of Compliance No. 1014," Appendix B, "Approved Contents and Design Features for the Hi-Storm 100 Cask System," (available at ADAMS No: ML13351A189), pp. 2.39, 2.44.

37 Yasuo Hirose et al., "An Alternative Process to Immobilize Intermediate Wastes from LWR Fuel Reprocessing," WM'99 Conference, February 28-March 4, 1999.

38 Jae Sik Yoo and Kune Yull Suh, "Analysis of TMI-2 Benchmark Problem Using MAAP4.03 Code," *Nuclear Engineering and Technology* 41, No. 7 (September 2009), p. 949.

39 IAEA, "Mitigation of Hydrogen Hazards in SA," p. 6.

40 Report by Nuclear Energy Agency Groups of Experts, OECD Nuclear Energy Agency, "In-Vessel and Ex-Vessel Hydrogen Sources," NEA/CSNI/R(2001)15, October 1, 2001, Part I: B. Clément (IPSN), K. Trambauer (GRS), and W. Scholtyssek (FZK), Working Group on the Analysis and Management of Accidents, "GAMA Perspective Statement on In-Vessel Hydrogen Sources," p. 15 (hereinafter: "In-Vessel and Ex-Vessel Hydrogen Sources," Part I).

41 IAEA, "Mitigation of Hydrogen Hazards in SA," p. 6.

42 Power Authority of the State of New York, Consolidated Edison Company of New York, "Indian Point Probabilistic Safety Study," Vol. 8, 1982, available at: ADAMS Documents, Accession Number: ML102520201, p. 4.3-10.

43 The volume percent of the carbon monoxide in the containment is the volume of the carbon monoxide in the containment divided by the volume of the containment multiplied by 100.

44 IAEA, "Mitigation of Hydrogen Hazards in SA," p. 47.

45 Report by Nuclear Energy Agency Groups of Experts, OECD Nuclear Energy Agency, "In-Vessel and Ex-Vessel Hydrogen Sources," Part I, p. 9.

46 T.J. Haste et al., Organisation for Economic Co-Operation and Development, "Degraded Core Quench: A Status Report," August 1996, p. 13.

47 L.J. Ott, Oak Ridge National Laboratory, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," 1997, p. 4.

48 LOFT LP-FP-2 was conducted in the Loss-of-Fluid Test Facility at Idaho National Engineering Laboratory in July 1985. The CORA and QUENCH tests were conducted at Karlsruhe Institute of Technology in Germany in the 1980s and 1990s.

49 L. J. Ott, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," p. 4.

50 L. J. Ott, "Advanced BWR Core Component Designs and the Implications for SFD Analysis," p. 4.

51 IAEA, "Mitigation of Hydrogen Hazards in SA," p. 14.

52 OECD Nuclear Energy Agency, "In-Vessel Core Degradation Code Validation Matrix: Update 1996-1999," report by an OECD NEA Group of Experts, October 2000, p. 210.

53 G. Bandini et al., Presentation Slides, "Progress of ASTEC Validation on Circuit Thermal-Hydraulics and Core Degradation," 3rd European Review Meeting on Severe Accident Research September 23-25, 2008, pp. 24, 28 (located at: http://www.sar-net.org/upload/4-5_bandini_ermsar2008_1.pdf).

54 Charles Miller et al., NRC, "Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force Review of Insights from the Fukushima Daiichi Accident," SECY-11-0093, July 12, 2011, available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML111861807, p. 3.

55 Burton Richter et al., "Report of the Fuel Cycle Research and Development Subcommittee of the Nuclear Energy Advisory Committee," June 2012, p. 5.

56 A.P. Ramsey, T. McKrell, and M.S. Kazimi, "Silicon Carbide Oxidation in High Temperature Steam," Advanced Nuclear Power Program, MIT-ANP-TR-139, 2011, abstract.

57 A triplex cladding design consists of three layers of material surrounding the nuclear fuel: an inner layer of dense silicon carbide for fission gas retention, a central composite layer of wound silicon carbide fibers to enhance mechanical performance, and an outer environmental barrier coating to enhance corrosion resistance. See Ken Yueh, David Carpenter, and Herbert Feinroth, "Clad in Clay," Nuclear Engineering International (January 2010), p. 14-15.

58 Ken Yueh, David Carpenter, and Herbert Feinroth, "Clad in Clay," *Nuclear Engineering International* (January 2010), p. 14.

59 A 2011 Idaho National Laboratory report states that the thermal conductivity of silicon carbide "can exceed the value of zirconium before irradiation. Extended irradiation tends to lower the [thermal] conductivity to a value half to one-third that of zirconium." See George Griffith, Idaho National Laboratory, "U.S. Department of Energy Accident Resistant SiC Clad Nuclear Fuel Development," INL/CON-11-23186, October 2011.

60 David M. Carpenter, Gordon E. Kohse, and Mujid S. Kazimi, "An Assessment of Silicon Carbide as a Cladding Material for Light Water Reactors," Advanced Nuclear Power Program, MIT-ANP-TR-132, November 2010, abstract.

61 Electric Power Research Institute, "Silicon Carbide Provides Opportunity to Enhance Nuclear Fuel Safety," EPRI Progress Report, September 2011, mydocs.epri.com/docs/CorporateDocuments/Newsletters/NUC/2011-09/09d.html.

62 Burton Richter et al., "Report of the Fuel Cycle Research and Development Subcommittee of the Nuclear Energy Advisory Committee," June 2012, p. 6.

63 INPO, "Report on the Fukushima Dai-ichi Accident," p. 24.

64 The author is indebted to David Lochbaum of the Union of Concerned Scientists for raising this point.

65 E. Studer et al., Kurchatov Institute, "Assessment of Hydrogen Risk in PWR," [undated], p. 1.

66 Allen L. Camp et al., Sandia National Laboratories, "Light Water Reactor Hydrogen Manual," NUREG/CR-2726, August 1983, p. 4-107.

67 PWR ice condenser and BWR Mark III containments have volumes of approximately 1.2 x 106 cubic feet and 1.3 x 106 cubic feet, respectively; PWR large dry containments have a volume of approximately 2.2 x 106 cubic feet. PWRs with ice condenser containments and BWR Mark IIIs have containment design pressures of approximately 20 psig and 15 psig, respectively; PWR large dry containments have a design pressure of approximately 53 psig. See M.F. Hessheimer et al., "Containment Integrity Research at SNL," NUREG/CR-6906, p. 24.

68 Charles Miller et al., "Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force Review of Insights from the Fukushima Daiichi Accident," p. 42.

69 These analyses were conducted for different PWRs, which have containments with different free volumes and different quantities of fuel cladding (active length) in their cores; the containments of these PWRs also have different design pressures and estimated failure pressures. Therefore, the results of these analyses do not directly apply to all PWRs. However, they do provide a general idea of the magnitude of the pressure spikes that a PWR containment might be expected to incur if an explosion—of the quantity of hydrogen generated from a zirconium-steam reaction of 100 percent of the active fuel cladding length—were to occur in the event of a severe accident.

70 T.G. Colburn, NRC, letter regarding Three Mile Island Unit 1, license amendment from hydrogen control requirements, February 8, 2002, Attachment 2, "Safety Evaluation by the Office of Nuclear Reactor Regulation, Related to Amendment No. 240 to Facility Operating License No. DPR-50, Three Mile Island Unit 1," available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML020100578, p. 5.

71 Kahtan N. Jabbour, NRC, letter regarding Turkey Point Units 3 and 4, exemption from hydrogen control requirements, December 12, 2001, Attachment 2, "Safety Evaluation by the Office of Nuclear Reactor Regulation, Turkey Point Units 3 and 4," available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML013390647, p. 3.

72 "Pounds per square inch gauge" is the value of a given pressure relative to the atmospheric pressure at sea level (14.7 pounds per square inch).

73 "Pounds per square inch absolute" is the value of a given pressure relative to a vacuum (0.0 pounds per square inch).

74 Power Authority of the State of New York, Consolidated Edison Company of New York, "Indian Point Probabilistic Safety Study," Vol. 8, 1982, available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML102520201, p. 4.2-1 and Appendix 4.4.1, p. 14.

75 Power Authority of the State of New York, Consolidated Edison Company of New York, "Indian Point Probabilistic Safety Study," Vol. 8, 1982, available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML102520201, p. 4.3-22, 4.3-23.

76 M.F. Hessheimer et al., "Containment Integrity Research at SNL," NUREG/CR-6906, p. 28; the source of this quote is NRC, "Severe Accident Risks: An Assessment or Five U.S. Nuclear Power Plants," NUREG-1150, Vol. 3, January 1991, Appendix D, "Responses to Comments on First Draft of NUREG-1150," p. D-22.

77 IAEA, "Mitigation of Hydrogen Hazards in SA," p. 61-62.

78 M. F. Hessheimer et al., "Containment Integrity Research at SNL," NUREG/CR-6906, p. 8.

79 IAEA, "Mitigation of Hydrogen Hazards in SA," p. 62.

80 Institute of Nuclear Power Operations (INPO), "Special Report on the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station," INPO 11-005, November 2011, p. 9, 12, 21, 24, 25, 32, 37, 79, 85, 86, 96.

81 Institute of Nuclear Power Operations (INPO), "Special Report on the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station," INPO 11-005, November 2011, p. 9.

82 The volume percent of the hydrogen in the containment is the volume of the hydrogen in the containment divided by the volume of the containment multiplied by 100.

83 IAEA, "Mitigation of Hydrogen Hazards in SA," p. 35.

84 IAEA, "Mitigation of Hydrogen Hazards in SA," p. 63. Containment spray systems are typically located inside the roof dome of PWR containments and are designed to spray cool water to condense the steam and reduce internal gas pressure within the containment. See Figure 8.

85 IAEA, "Mitigation of Hydrogen Hazards in SA," p. 34.

86 IAEA, "Mitigation of Hydrogen Hazards in SA," p. 33.

87 Kahtan N. Jabbour, NRC, letter regarding Turkey Point Units 3 and 4, exemption from hydrogen control requirements, December 12, 2001, Attachment 2, "Safety Evaluation by the Office of Nuclear Reactor Regulation, Turkey Point Units 3 and 4," available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML013390647, p. 4.

88 W. E. Lowry et al., Lawrence Livermore National Laboratory, "Final Results of the Hydrogen Igniter Experimental Program," NUREG/CR-2486, February 1982, p. 4.

89 IAEA, "Mitigation of Hydrogen Hazards in SA," p. 35.

90 OECD Nuclear Energy Agency, "Report on FA and DDT," p. 1.2.

91 IAEA, "Mitigation of Hydrogen Hazards in SA," p. 33.

92 D.W. Stamps et al., Sandia National Laboratories, "Hydrogen-Air-Diluent Detonation Study for Nuclear Reactor Safety Analyses," NUREG/CR-5525, January 1991, available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML071700388, p. 43.

93 OECD Nuclear Energy Agency, "Carbon Monoxide-Hydrogen Combustion Characteristics in Severe Accident Containment Conditions: Final Report," NEA/CSNI/R(2000)10, 2000, p. 18.

94 Helmut Karwat, "Igniters to Mitigate the Risk of Hydrogen Explosions—A Critical Review," *Nuclear Engineering and Design* 118, 1990, p. 267.

95 IAEA, "Mitigation of Hydrogen Hazards in SA," p. 113.

96 Advisory Committee on Reactor Safeguards, 586th Meeting, September 8, 2011, available at: ADAMS Documents, Accession Number: ML11256A117, p. 95.

97 A number of hydrogen combustion experiments have been conducted at Sandia National Laboratories; for example, such experiments were conducted in the 1980s at the FLAME facility—a rectangular channel 100 feet long, 8 feet high, and 6 feet wide. M.P. Sherman et al., Sandia National Laboratories, "FLAME Facility: The Effect of Obstacles and Transverse Venting on Flame Acceleration and Transition to Detonation for Hydrogen-Air Mixtures at Large Scale," NUREG/CR-5275, available at: ADAMS Documents, Accession Number: ML071700076, abstract.

98 Most experiments investigating the lower hydrogen concentration limits at which deflagration-to-detonation transition occurs have been conducted in detonation tubes; such tubes have been 39 to 70 feet long and about 11 to 17 inches in diameter. OECD Nuclear Energy Agency, "Report on FA and DDT," p. 3.5.

99 OECD Nuclear Energy Agency, "International Standard Problem ISP-47 on Containment Thermal Hydraulics: Final Report," NEA/CSNI/R(2007)10, September 2007, p. 7.

100 Westinghouse, "AP1000 Design Control Document," Rev. 19, Tier 2 Material, Chapter 19, "Probabilistic Risk Assessment," Sections 19.41 to 19.54, June 13, 2011, available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML11171A409, p. 19.41-2.

101 Charles Miller et al., "Recommendations for Enhancing Reactor Safety," SECY-11-0093, p. 42.

102 OECD Nuclear Energy Agency, "SOAR on Containment Thermal Hydraulics and Hydrogen Distribution," 1999, p. 18.

103 OECD Nuclear Energy Agency, "Report on FA and DDT," p. 1.6.

104 NRC, "Notice Regarding Eliminating the Hydrogen Recombiner Requirement," *Federal Register* 68, No. 186 (September 25, 2003), p. 55419.

105 E. Bachellerie et al., "Designing and Implementing a PAR-System in NPP Containments," p. 158.

106 Indian Point Energy Center, License Renewal Application, Technical Information, 2.0, "Scoping and Screening Methodology for Identifying Structures and Components Subject to Aging Management Review and Implementation Results," p. 2.3-61.

107 E. Bachellerie et al., "Designing and Implementing a PAR-System in NPP Containments," p. 159.

108 In January 1985, the NRC began requiring plant owners to install hydrogen control systems in the containments of such designs. See NRC Policy Statement, "Combustible Gas Control in Containment," Federal Register, Vol. 68, No. 179, September 16, 2003, p. 54124.

109 PWRs with ice condenser containments and BWR Mark IIIs have containment design pressures of approximately 20 psig and 15 psig, respectively. See M.F. Hessheimer et al., "Containment Integrity Research at SNL," NUREG/CR-6906, p. 24.

110 Allen L. Camp et al., Sandia National Laboratories, "Light Water Reactor Hydrogen Manual," NUREG/CR-2726, August 1983, p. 4-107.

111 NRC Policy Statement, "Combustible Gas Control in Containment," *Federal Register* 68, No. 179 (September 16, 2003), p. 54124.

112 In December 1981, the NRC began requiring plant owners to operate BWR Mark Is and Mark IIs with inerted primary containments. See NRC Policy Statement, "Combustible Gas Control in Containment," Federal Register, Vol. 68, No. 179, September 16, 2003, p. 54123.

113 *Federal Register* 68, No. 179 (September 16, 2003), p. 54141.

114 IAEA, "Mitigation of Hydrogen Hazards in SA," p. 74.

115 BWR Mark I and Mark II primary containments have volumes of approximately 0.28 x 106 cubic feet and 0.4 x 106 cubic feet, respectively; these are about one-eighth and one-sixth the volumes, respectively, of typical PWR large dry containments. See M.F. Hessheimer et al., "Containment Integrity Research at SNL," NUREG/CR-6906, p. 24.

116 NRC, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents," EA-12-050, March 12, 2012, available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML12054A694.

117 See NRC, "Installation of a Hardened Wetwell Vent," Generic Letter 89-16, September 1, 1989, p. 1. Generic Letter 89-16 states that "the Commission has directed the [NRC] staff to approve installation of a hardened vent under the provisions of 10 CFR 50.59 ["Changes, Tests, and Experiments"] for licensees, who on their own initiative, elect to incorporate this plant improvement."

118 NRC, "Installation of a Hardened Wetwell Vent," Generic Letter 89-16, September 1, 1989, p. 1.

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