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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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MEETING OF THE US-APWR SUBCOMMITTEE

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FRIDAY

FEBRUARY 22, 2013

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ROCKVILLE, MARYLAND

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

COMMITTEE MEMBERS:

JOHN W. STETKAR, Subcommittee Chairman

DENNIS C. BLEY, Member

CHARLES H. BROWN, JR. Member

JOY REMPE, Member

STEPHEN P. SCHULTZ, Member

WILLIAM J. SHACK, Member

NRC STAFF PRESENT:

GIRIJA SHUKLA, Designated Federal Official

EDWARD FULLER, NRO

HOSSEIN HAMZEHEE, NRO

TODD HILSMEIER, NRO

KERRI KAVANAGH, NRO

SAMUEL LEE, NRO

EILEEN MCKENNA, NRO

STEPHEN MONARQUE, NRO

LYNN MROWCA, NRO

HANH PHAN, NRO

MARIE POHIDA, NRO

RUTH REYES, NRO

TARUN ROY, NRO

JEFF SCHMIDT, NRO

THEODORE TJADER, NRO

ROBERT VETTORI, NRO

ALSO PRESENT:

ROY KARIMI, ERI

MOHSEN KHATIB-RABHAR, ERI

PRAVIN SAWANT, ERI

RON CARVER, Luminant

TIMOTHY CLOUSER, Luminant

JOHN CONLY, Luminant

ROBERT REIBLE, Luminant
DONALD WOODLAN, Luminant
HIROSHI GODA, MHI
HIROSHI HAMAMOTO, MHI
TAKASHI KURISAKI, MHI
TAKAYUKI NIRASAWA, MHI
FUTOSHI TANAKA, MHI
OSAMI WATANABE, MHI
JAMES CURRY, MNES
SCOTT KIPPER, MNES
KEVIN LYNN, MNES
RON REYNOLDS, MNES
RYAN SPRENGEL, MNES
GEORGE WADKINS, MNES
EDMOND WIEGERT, MNES

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P R O C E E D I N G S

8:32 a.m.

CHAIRMAN STETKAR: The meeting will now come to order. This is the second day of the subcommittee meeting on the US-APWR. I'm John Stetkar, Chairman of the Subcommittee meeting. Members in attendance are Stephen Schultz, Dennis Bley, Bill Shack, Charlie Brown, and Joy Rempe. All of the protocol issues that we discussed yesterday still remain in effect this morning. Please silence your cell phones. Use the microphones. Identify yourself if you have something to say. And since I see MNES sitting up at the front table, I'm assuming that we're going to get some resolution on perhaps some open items from yesterday. Is that --

DR. CURRY: Yes, sir. We have some feedback for the Committee, if you'd like.

CHAIRMAN STETKAR: Let's do that.

DR. CURRY: All right. Let me -- I'm Jim Curry. Dr. Tanaka is here, and we have the same group also that was here yesterday in case there are any follow-ups.

All right. Yesterday, from our notes, we had several items that we agreed with the Committee we'd follow up on. So four of those items relate to questions that were raised by the Committee that we have responded in RAI fashion.

CHAIRMAN STETKAR: Okay.

DR. CURRY: So going down the list, the RAI that discusses room heat-up for the various rooms, we would refer you to RAI 750-5675.

CHAIRMAN STETKAR: 5675?

DR. CURRY: Yes, sir. Question 19-516.

CHAIRMAN STETKAR: Okay.

DR. CURRY: The RAI that relates to the calculation of RC top seal cooling, the one-hour time, we would refer you to RAI 148-1700, Question 19-273.

CHAIRMAN STETKAR: 273?

DR. CURRY: Yes, sir.

CHAIRMAN STETKAR: Thank you.

DR. CURRY: The RAI that responded to the staff's question about alternate containment cooling and how we analyze that, we would refer you to RAI 480-3711. It has an odd question number, but I think that will be straightforward for you.

CHAIRMAN STETKAR: Okay. It's the whole -- okay.

DR. CURRY: And then for the RAI that discusses the probability value for moving from the main control room to the remote shutdown console, we would refer you RAI 744-5668, Question 19-505, and it's Revision 2 of that RAI response.

MEMBER BLEY: I'm sorry. The last thing you said?

DR. CURRY: Revision 2 of that RAI response. So Question 19-505.

CHAIRMAN STETKAR: And we appreciate that. As I've said, in many cases, the amount of material that we receive is daunting, to say the least. And we typically don't request the RAIs for a variety of reasons. Number one, it just increases the volume of material. Number two, if we request them, there's an implicit idea that we'll actually read all of that stuff. So in some cases, the questions that we raise in the subcommittee meeting have already been addressed. And I know it's a bit frustrating for you, but we really appreciate this sort of winnowing down into a real focus. So I do appreciate that.

DR. CURRY: Our pleasure.

MR. SHUKLA: One more, like in that before non-seal testing is not done until the end of this month.

DR. CURRY: Yes, I think, Ryan, you talked about that yesterday.

CHAIRMAN STETKAR: But this RAI

specifically addresses the no-seal leak within an hour because it's a related but slightly separate issue.

DR. CURRY: That's right. The question was the one-hour time.

CHAIRMAN STETKAR: Thank you.

DR. CURRY: All right. The question we were talking about the peer review and whether it explicitly, the peer review explicitly addressed attributes of the PRA in terms of category one and two. We did a look at that, and it looks, at the time that the peer review was done, we didn't, it wasn't appropriate from a standard to compare attribute to category. So it was a process which graded the PRA technical element attributes, and we would refer you to RAI 564-4399, Question 19-426. There was a question about why the difference between the treatment of loss of all component cooling water and partial loss of component cooling water, the loss of all component cooling water was a fault tree treatment versus the point estimate for the partial loss of component cooling water. And our response, based on the generic data in NUREG/CR-6928 is that, in this situation, we felt that we could provide a point estimate value for the partial loss of component cooling water because it was already a partial loss in a very, you know, clear failure mode, as opposed to a complete loss of all component cooling water in which we were interested in evaluating in failure modes. So that's why a fault tree was constructed. If you recall the flooding protection, we partitioned the subsystem, so that was the reason that the fault tree was developed in the formal case.

CHAIRMAN STETKAR: I appreciate that feedback. I still don't understand why you couldn't quantify the partial from the same fault tree because, in the same sense, you have four trains with kind of one and one in each half system normally running. Service water and component cooling water system designs in currently operating plants, regardless of how many trains you may define for licensing bases, vary all over the place. I mean, I've seen plants that have six service water pumps with three normally running, which is more than you have running, you know, for a two train plant for example. I've seen plants that have two trains with one and one normally running, where failure of one and only one pump would be partial loss of cooling water, which is like what you have. But as long as you deal with the model for the total and rely on it, why couldn't you do the same thing? You'd at least use consistent data, consistent failure rates, and those two initiating event frequencies. Regardless of whatever questions one might have in an absolute frequency, it would certainly line up in, you know, relative sense much better than they do now. So that was a main concern. But I appreciate the feedback, and I'll grant you there's some arguments where your partial could look more like generic U.S. plants, but I really haven't seen a generic U.S. plant for service water and component cooling water that is generic. We had five pumps at Zion, and two were normally running, but it was shared between two complete units. So what's partial?

MEMBER BLEY: There was actually an attempt to do that 30 years ago at NRC to build kind of generic models, and they were able to build most systems but electric power and --

CHAIRMAN STETKAR: Component cooling water service water --

MEMBER BLEY: Component cooling water service water were unique, every one.

CHAIRMAN STETKAR: Yes. So, anyway, I at least appreciate kind of the thought process.

DR. CURRY: Yes, and we appreciate the feedback. The question about the common cause failure of the CCW pumps and the source of the data being a single individual, we evaluated that and confirmed that's the case. We went back to the original documentation and the consultant felt that it was appropriate to use that number, rather than a somewhat higher number, based on the judgment that there was never any common cause failure of normally running CCW or service water system. We felt that calculating in a normal method would be too high.

CHAIRMAN STETKAR: And in some sense, I understand that rationale. In the interest of time, I don't want to get into too deep a discussion, but I did a quick calculation using the common cause failure data from -- I'm terrible with NUREG numbers -- from the NUREG that's cited and ran out beta, gamma, and delta. And if I remember, and I'd have to look up my notes, it's about an order of magnitude higher in frequency, but it would also be supported by the fact that you've never seen a complete bus of component cooling water in a plant that has four trains. So just the fact that you haven't seen one yet doesn't necessarily support, you know, one number versus another number. It's something that, you know, there could be different ways of treating it with a broader uncertainty distribution, a broader number of experts providing input to development and that.

And the only reason that it could be

important is, obviously, that initiating event and failures of the component cooling water system itself, post-trip response, that uses the same component, the same common cause failure parameters are an important contributor. So for some reason, that parameter value is underestimated. The importance of component cooling water would be underestimated.

It won't change any of the overall conclusions, and I have to keep saying this, for the purposes of the design certification. Component cooling water is important. It would remain important. Would it increase the core damage frequency to ten to the minus two? No, it would not, certainly. But it's something to be sensitive to, you know, especially because you actually have done a very, very good job in many areas of this PRA. You know, our job is to be critical, but I have to say that. There are a lot of parts of this PRA that are really good, especially the LOCA analyses I think are generally pretty good.

DR. CURRY: Thank you, sir. Do you want to add anything to that, or are you okay?

DR. TANAKA: No.

DR. CURRY: All right. Let's see. The question about the main steam depressurization valves, the main steam relief valves, and whether the depressurization valves could depressurize in a timely manner, we reviewed that and the depressurization valves and the relief valves were the same size.

CHAIRMAN STETKAR: They are? Let me go look up -- again, I don't want to take time here too much. Let me make a note of that because from Chapter, I think it was 10, and I can refer you to the table. I'd need to find it in my notes. I thought they were substantially different in terms of their rated relief capacities. But I'll look up the table. I might --

DR. CURRY: And I should say we were able to check as far as the size was the same, so I'm kind of making a jump that --

CHAIRMAN STETKAR: There is a table, there is a table in Chapter 10 that listed pounds mass per hour relief capacity. Now, I backed that up to rated steam flow to get a fraction of, you know, rated core power. And from that, I thought that the MSRVs were about five, I thought that the rated steam flow was about five times higher. I'll have to go look, but I'm pretty sure it's Chapter 10. I could be wrong.

DR. CURRY: DCD Chapter 10?

CHAIRMAN STETKAR: DCD Chapter 10. Now, I could find the, at the break I'll find the table number and let you know if you don't find it.

DR. CURRY: And it should be straightforward here.

CHAIRMAN STETKAR: Because that was all recent from my question because I saw that what I thought was a large difference in the size. And in the discussions, at least in the DCD, the main steam depressurization valves are characterized primarily as a way of controlled cool-down to get to cold shutdown. They're not really characterized in the DCD as a safety-related, you know, rapid depressurization, that sort of function. They are obviously, I think -- are they safety-related or none?

DR. CURRY: They are safety-related. They're powered from safety buses, if I remember right.

CHAIRMAN STETKAR: MSDVs. MSRVs are not safety. I know they come from non-safety. Okay.

DR. CURRY: I think we can cross-check that --

CHAIRMAN STETKAR: Okay, yes, if you could, because I might have misinterpreted something fundamentally there.

DR. CURRY: We'll cross-check. In this regard to the terminology, we acknowledge the terminology difference, and in the next update of the PRA we will go through it and --

CHAIRMAN STETKAR: And you did confirm at least that the depressurization valves were the ones -- well, if you did the same relief capacity, it doesn't make any difference which one was used.

DR. CURRY: Okay, okay. So we will follow up on that terminology. The action item related to the correlated uncertainties, RiskSpectrum option, the RiskSpectrum option was the state of knowledge correlation was applied.

CHAIRMAN STETKAR: It was good. Excellent.

DR. CURRY: For the question about the HRA numbers that didn't add up, there were really two parts to that. We will evaluate that question, but we know that there are some products, cross products that have to be considered, so it may not be a simple sum.

CHAIRMAN STETKAR: Okay. That, as I mentioned yesterday briefly, I looked at -- it doesn't make any difference why I started looking at the tables. The first table I looked at, the numbers didn't add up. I thought, gee, this is funny. And I tried a few thought experiments, and none of my thought experiments worked. And then I looked at a couple of other tables where they did add up, and that even more confused me. So I gave up on the addition.

DR. CURRY: We'll evaluate, but that may be the reason.

CHAIRMAN STETKAR: If there are cross products or something like that. It's certainly not explained in the table, you know, because, in the table, there's just a single line item on the table that says, I can't remember what it says, task one. It always says task one plus task two plus dot, dot, dot, plus whatever the last task is listed, which implies that they're just added. But maybe the dot, dot, dot has some logic in that isn't really explained.

DR. CURRY: The second part of that question had to do with the fifth percentile, why did we use fifth percentile numbers in some cases. And that was a judgment based on important actions from a risk perspective would require frequent training and detailed operator training, familiarity of the control when operators looked at accident sequence. So in some of those cases, the lower bound of the ATP was applied. That appears to be consistent with NUREG/CR-4772, Page 8-8.

CHAIRMAN STETKAR: You have to be a bit careful with NUREG/CR-4772 because Alan Swain mixed up the quality of procedures and training versus the uncertainty in the error rates. It's really difficult to sort of sort through the guidance in uncertainties on that NUREG because there's a mixture between average, better than average, and worse than average, if I can call it that, quality of procedures in training, whether you use one set of values versus another. And then that sort of gets mixed into an uncertainty analysis, which is really different.

DR. CURRY: The other point we may note just for continuity and translating these insights, they are listed in DCD Table 19.1-119, and we would expect those important actions to be carried forward in the development of procedures and training programs.

CHAIRMAN STETKAR: And they are. You're absolutely correct. They are listed as important actions. It's just in terms of -- again, the mix of contributors, you know, to that risk profile could be biased because, as I said, from those uncertainty distributions, mean value is about a factor of eight times higher than the fifth percentile value, which is, you know, it's not at 800 but it's not insignificant either so . . .

DR. CURRY: Okay. In terms of question about interfacing system LOCA, I think Mr. Bley pointed out the factor of a thousand you pointed us to. So we took a quick look at that, and we agree. So --

MEMBER BLEY: I hope it was a typo, but I think maybe it was because it was carried through to the final product.

CHAIRMAN STETKAR: That's an inside joke.

DR. CURRY: So we will address that issue. I think there was also a question about the pipe failure data.

MEMBER BLEY: Yes, where it came from and its applicability to this particular case.

DR. CURRY: NUREG/CR-6928.

MEMBER BLEY: Just an aside, I'll go look. I think that's kind of a number applying to pipe sitting around anywhere. We're talking about pipe that's suddenly run up to a much higher pressure than it normally sees. I don't know why San Bruno just popped in my mind, but it did. I'll look and see, but I suspect -- it will be interesting to see a justification of why that number the right one to use for this specific case and to consider over a 24-hour period. I think that's what that number is. I'll have to go look to double-check.

DR. CURRY: Okay. And that's our source of data, and you recall the discussion yesterday about the design pressure --

MEMBER BLEY: I do.

DR. CURRY: -- and the fact that it's an open system and the like. The question about the 15-minute assumption for ESW pipe, ESW pipe leak affecting CCW pumps, and, as I think we pointed out yesterday, there's level switch, level indication. There's also notification of the operator by a decrease in the outlet flow from the heat exchanger and/or ESWS header. There's an alarm in the control room, as well. As we talked about yesterday, then the action is again from the control room to just simply turn off the pump.

And I think the last question on our list, which I think the Chairman was going to think about, but we thought about it, as well, was the difference between POS 8 and 4 in low-power shutdown, POS 8 coming after refueling and why that was a bigger contributor, a bigger parent contributor. That really has to do with the alignment, the CCW alignment assumed, not the heat load.

CHAIRMAN STETKAR: Okay. Thanks. And I didn't have a chance to look at it, so thank you very much for pointing me to that. I appreciate that.

MR. SHUKLA: Jim, Dr. Rempe asked a question about the fuel pressure being low. It could be proprietary so . . .

DR. CURRY: Well, you know, we did evaluate that, but I think we also talked about the hydrogen generation, which I thought was where you were headed with that, Ms. Rempe.

MEMBER REMPE: I believe, actually, they did answer and they said they picked something that they thought was conservative, and then they beefed up their hydrogen. I guess it was just a couple of cases that you analyzed where you decreased the hydrogen, but they were considered, that, with the staff's analysis, was enough. They decided to put the hydrogen igniters on, so perhaps it doesn't matter.

There's other things that would be interesting to know about what happens when you make this conservative assumption, but I think we'll talk about it today. It appears MELCOR made the same assumption. We'll see. Again --

MR. SHUKLA: Assumption is not realistic.

MEMBER REMPE: I'm curious on what the basis for it is. And I think MHI said we just tried to be conservative is what they told us, which is an answer.

MR. SHUKLA: I just wanted to make sure that we covered it. Thank you.

MEMBER SCHULTZ: It was conservative and consistent. I don't think that was a selected number but --

MEMBER REMPE: They assumed something that was lower, so it would relocate earlier. And in some areas, that may be conservative. In the way MAAP truncates hydrogen production once they relocate or melts, it may not be conservative. But that's, again, MAAP is still doing, but they've compensated for that.

DR. CURRY: So that's all we have on our list.

CHAIRMAN STETKAR: That's it?

MEMBER BLEY: Can I go back to one of them? Because I'm still a little -- the one about ESW leak. I'm glad to hear you have the alarms on outlet flow and header pressure for ESW. What I still don't understand are why 15 minutes? What size leak did you assume? Is that the biggest leak that could occur? Why assume the HEP is zero rather than doing an HRA analysis for the operator action?

DR. CURRY: So a couple of questions. I'm not sure that we have alarms on the outlet flow.

MEMBER BLEY: Oh, that's what I thought I heard you --

DR. CURRY: We can monitor outlet flow. But I wanted to get across the point that it's not just a level switch, but we can monitor outlet flow and header pressure. I'm not sure where the alarms are, but there's a control on the alarm. I think it's on level, but maybe someone else can --

CHAIRMAN STETKAR: Is it ESW flow at the outlet of the heat exchanger or to the heat exchanger? Because if you have a leak at the heat exchanger and the flow is at the inlet to the heat exchanger, you're never going to see it. You're never going to see it anywhere on the ESW system if the pump is happy.

DR. CURRY: I have in the notes that I have, outlet flow from the heat exchanger.

CHAIRMAN STETKAR: Outlet flow. Okay. If it was a very, very big break, that certainly, you know, would fall.

MEMBER BLEY: If it was a very, very big break, 15 minutes might have everything wiped out.

CHAIRMAN STETKAR: I don't know. I mean, that's --

MEMBER BLEY: I don't know.

MEMBER SCHULTZ: The other question or at least part of the discussion was, given the alarm, what's the operator action?

DR. CURRY: Right. Now, the operator action is just terminate the ESW pump.

MEMBER SCHULTZ: That's clear from the

procedure.

DR. CURRY: Right. Well, we don't have procedures, but that's what the model would have.

MEMBER SCHULTZ: That's for the model. Thank you.

MEMBER BLEY: No attempt to isolate, so you'd take out all that side of ESW?

DR. CURRY: Right. You don't have to do any diagnostics or, you know, go out . . .

MEMBER BLEY: Are there no tumble line effects from wiping out all the ESW in that term? Doesn't it affect the risk model in other places? It just seems . . .

DR. CURRY: Well, I'll let Dr. Tanaka speak to that, but, fundamentally, that's the flood level and we evaluate the components that are lost.

DR. TANAKA: Yes. Even if it's 15 minutes, it's already propagated to the train next to it, so it takes away two --

CHAIRMAN STETKAR: If they do trip it within -- I didn't look at the model. If they successfully stop it within 15 minutes, what is the damage that's modeled from that condition?

DR. TANAKA: It takes away the ECW that has failed, of course.

CHAIRMAN STETKAR: Sure.

DR. TANAKA: And the one next to it.

CHAIRMAN STETKAR: And the other one in that, so you take out that half of the plant?

DR. TANAKA: Yes, half the plant. Yes.

CHAIRMAN STETKAR: That's if they do successfully trip it within 15 minutes?

DR. TANAKA: Correct, yes. If they do, yes, even if they do, it's still --

CHAIRMAN STETKAR: What happens if they don't within 15 minutes? What's the difference? Well, you don't know because you didn't model that. I'm sorry.

DR. TANAKA: From the notes we have, of course the level will increase. The judgment we had is one of the doors would break, which will --

CHAIRMAN STETKAR: Would go over to the other side --

DR. TANAKA: It's the outside, I guess.

CHAIRMAN STETKAR: Yes, that's right. It's hard to get over to the other side of the plant. I'm sorry.

DR. TANAKA: So it goes outside. So, anyway, it does not propagate to the other two trains but go outside. In any event, it will take away two trains.

CHAIRMAN STETKAR: From what he just said, the 15 minutes may not make all that much difference to the impact.

MEMBER BLEY: That may be the case. And then why all the --

CHAIRMAN STETKAR: Right.

MEMBER BLEY: Yes. Personally, I'll have to look at that one some more and really try to understand it. One, maybe it makes no difference. Two, why 15 minutes, and did you look at all the possible leaks? And, third, if you keep it the way it is, why assume absolute guarantee of human success in tripping the pump? You don't do that other places. You do an HRA.

DR. CURRY: I'll look at it some more.

CHAIRMAN STETKAR: Anything else? No.

DR. CURRY: Scott Kipper.

MR. KIPPER: Scott Kipper from MNES. I have one additional piece of information for the main steam relief valves and depressurization valves, Chapter 10. And the capacities for those are given at different pressures. That's why the capacities are listed differently in the table.

MEMBER BLEY: Oh, so the throat sizes are the same.

MR. KIPPER: They are the same size valve, correct.

CHAIRMAN STETKAR: Thank you. Thank you. Anything else?

DR. CURRY: No, sir.

CHAIRMAN STETKAR: Thank you very, very much. That's been very useful. While you're sitting there, Dr. Rempe, I believe, had one more item that she wanted to revisit, correct?

MEMBER REMPE: So yesterday there was a discussion about trying to benchmark the MAAP depressurization characteristics, namely the reactor vessel pressure and water level predictions, against WCOBRA. And I'm well aware that we use MAAP, as well as MELCOR, for other reasons. But there are a lot of assumptions built into those codes that make it even more difficult to benchmark or compare results after you get through the depressurization to top of core. And, in fact, if we look at some of the staff results where they did those comparisons, they did see differences, and I actually think that should be done not only for MAAP but also MELCOR. And, in fact, in the SRP that was issued for passive reactors in September, they kind of state what I'm getting to better than probably I can state it, that the reviewer examines the results of any sensitivity studies performed by the applicant and the choice of thermohydraulic accident analysis codes used to perform such studies. Applicants frequently use the MAAP code for such studies. The staff is aware of thermohydraulic modeling issues with the code that could compromise its ability to confirm the validity of the PRA success criteria involving minimal sets of mitigating equipment. Use of this code is acceptable only if sufficient benchmarking studies have been done, which compare MAAP results with those of the thermohydraulics code the staff has reviewed and approved to show that MAAP is able to capture the important thermohydraulic phenomena and the timing of such phenomena in simulations of accident sequences included in this PRA.

So I'd like to again emphasize the request that you've done the WCOBRA analyses for medium LOCA, small LOCA, etcetera. Can we see some comparisons of that? Okay? And, hopefully, I've been a little more clear this time than I was yesterday. Thank you.

MR. SPRENGEL: I'd like to request a clarification. We'll need to confirm because I think we're getting a little hung up maybe on the discussion specifically with WCOBRA/TRAC.

MEMBER REMPE: Some thermohydraulics model was accepted. You're right. That's what I'm trying to get to.

MR. SPRENGEL: Because of break sizes and application --

MEMBER REMPE: Yes, just a couple of break sizes. Let's get some confidence that your MAAP model is capturing the thermohydraulics phenomena correctly before we start worrying about other things sometimes is a good idea. And, thankfully, I think that should be done with the MELCOR code, too, versus perhaps, I think they used RELAP in this particular application. I'm an equal opportunity reviewer.

MR. SPRENGEL: Okay. Thank you for the clarification.

CHAIRMAN STETKAR: Anything else, Joy?

MEMBER REMPE: Nope, that's . . .

CHAIRMAN STETKAR: Good. That was pleasant. And, again, thanks for the feedback. It's one of the reasons why a two-day meeting is a long meeting, but it does give us this opportunity to get some things, you know, resolved or clarified and makes the process a little bit more efficient anyway. So we really appreciate the homework that all of you did. I'm sure there were people up late last night looking for things, and I appreciate that very much.

With that, I think we'll ask the staff to come up and hear their side of the story on Chapter 19.

MS. REYES: Good morning, everyone. My name is Ruth Reyes. I'm the chapter PM of Chapter 19 for the US-APWR DCD and COL applications. And here with me are Hanh Phan, Marie Pohida, and Todd Hilsmeier. And also in the audience, we have Dr. Ed Fuller. He was the reviewer for the Severe Accident Evaluation before moving to research. And we also have the contractors who helped us on the review.

Before I let the staff start with their presentation, I just wanted to mention something, which also said yesterday. The staff presentation does not include the seismic evaluation. That was not in the SE either, and the reason for that is because that will be part of the, we will present that at the Chapter 3 ACRS meeting in the future.

So having said that, I'm going to let the staff . . . okay.

MR. PHAN: Okay. Good morning, ladies and gentlemen. My name is Hanh Phan. I am the lead reviewer for US-APWR DCD Chapter 19 PRA and severe accident. This is our privilege to be sitting in the same table with you again. You can feel that by the vibration of my hands.

I'd also like to recognize the contribution of Dr. John Lai. He was the reviewer of internal fires and flooding. And also Mr. Nick Soltis. He was the lead reviewer, and he is retired.

So with that, I will go straight to the technical discussion.

CHAIRMAN STETKAR: That would be fine. You're well known to the Subcommittee.

MR. PHAN: Please go to slide three. So in this presentation, the staff would like to go over the PRA quality, internal events PRA, internal fires PRA, internal flooding PRA, external events risk evaluation, low-power and shutdown PRA, Level 2 PRA, and the Severe Accident Evaluation.

This slide provides the overall reviews approach so that you would understand the depth of the reviews that we have performed. In general, the key activities include receive trainings on the US-APWR designs; develop initial risk insight to support all the technical branches; discuss US-APWR designs with other technical branches; perform PRA audits and participate in many public technical discussions; ensured review consistency with other design certifications; performed audit/confirmatory calculations for assessment of specific severe accident/Level 2 PRA issues; and, last, review the application in accordance to the requirements of 10 CFR Part 52, the Commission's goals, the SRP, and PRA standard.

The next slide is 5678 and provides the brief discussion of the open items. Due to the time constraints, I would not read them all. However, those open items identifies with a star that will be discussed in the technical topics of interest in the following slide.

MEMBER BROWN: Can I have one question? Since one of them involves my stuff or stuff I'm interested in that has an asterisk by it, I presume then that when we do Chapter 7 in April that we will be able to evaluate or have a discussion on how the assumption for the application of the I&C failure, common cause failures, and all that stuff, will be included as part of that overall presentation. So is there some coordination between you all and the I&C folks to make sure that happens in that meeting in April 24th and 25th, I think. It's a two-day meeting. I'm presuming based on saying that that's what's going to occur. I'm just trying to confirm that. It's either a yes or no, I think.

MR. PHAN: No, we have not had any

discussion with them after that meeting.

MEMBER BROWN: Oh, okay. So even though it's going to be addressed, nobody is going to be able to talk about it?

MR. PHAN: I will be talking --

MEMBER BROWN: Oh, you will at that time?

MR. PHAN: Yes. No, in this presentation I will reopen that issue.

MEMBER BROWN: Oh, I thought you weren't going to talk about these things, these particular open items today.

CHAIRMAN STETKAR: The star they will.

MR. PHAN: For those with the star.

MEMBER BROWN: Oh, they're going to be done later in the presentation --

CHAIRMAN STETKAR: Yes.

MEMBER BROWN: -- if they have a star? Okay. I didn't understand that.

CHAIRMAN STETKAR: The message is that they will not give us a PRA-related presentation on digital I&C in April. We're going to hear that from the staff's perspective today.

MEMBER BROWN: Okay.

CHAIRMAN STETKAR: On the other hand, in April, because we're an ACRS subcommittee, we can ask questions about anything, so it's certainly possible in April, when we're hearing the design certification presentation on digital I&C, we can always ask questions regarding, gee, please refresh us or give us a little more information about how that was modeled in the PRA. That's certainly a possibility. We've done that for other topics, but this is basically our primary chance to address those issues.

MEMBER BROWN: All right. Well, I did not see, and maybe I went through this too fast, I did not see a specific page for that particular one, and that's why -- now, I might have missed it because I just quickly thumbed through it. That was 515. Other than what I'd call --

CHAIRMAN STETKAR: The fifth bullet has a star on this page here, so I'm assuming that we're going to hear something about it.

MR. PHAN: Yes. On slide 14 and 15, I will be --

MEMBER BROWN: Okay, all right. Thank you.

MR. PHAN: I learned a lesson yesterday. When you're quiet, I will be quick as possible.

CHAIRMAN STETKAR: That's an appropriate strategy.

MR. PHAN: Please turn to slide nine. The first topic of interest is on quality of PRA. In accordance with Reg Guide 1.200, the US-APWR PRA quality is evaluated in terms of scope, level of detail, and technical adequacy. The scope of the US-APWR design-specific PRA includes Level 1 PRA; Level 2 PRA for internal events, including internal flooding, internal fires, at-power, and at-shutdown conditions. Seismic risk was evaluated using PRA-based Seismic Margins Assessment. Other external events, including high winds, external floods, external fires, and so on, they will be addressed by the COL applicant.

The level of details of the US-APWR design-specific PRA are reviewed to ensure that the PRA reflects the design and anticipated operation of practice, to the extent possible, to provide confidence in the results so they can used to support the DCD process. To ensure that level of detail is sufficient, in DCD Section 19.1.2.4, "PRA Maintenance and Upgrade" states that the PRA is placed under configuration control in accordance with ASME/ANS 2009 PRA Standard.

MEMBER SHACK: They have to do the SAMDA analysis. Why doesn't that scope, in fact, include a Level 3 consideration when your reviewing the quality of the PRA?

MR. PHAN: According to the SRP guidance in Chapter 19, there would not be any Level 3 information required in Chapter 19. That's why it's got nothing included.

MEMBER SHACK: But you give the review that you do when you look at the SAMDAs and the environmental report, a comparable review that you do for the Level 1 and Level 2.

MR. HILSMEIER: In the environmental report, they do a Level 3 PRA.

MEMBER SHACK: Right. But I'm asking about the level of review of that PRA and who does it?

MR. PHAN: Ed, could you answer that?

MR. FULLER: This is Ed Fuller. We did review the Level 3 PRA, even though it wasn't officially on the docket, for the expressed purpose of evaluating its use in the SAMDA evaluation. And in the process of so doing, questions came up and it turned out that we asked them to recast the Level 2 evaluation a little bit in order to make it easier to look at the inputs to the offsite consequence analysis that was part of the SAMDA. In other words, we asked them to expand the number of release categories so that you had associated core melt progression and source term releases.

We never considered that we needed a Level

3 PRA for the SAMDA evaluation, but it's hard to get there without having that kind of an offsite consequence evaluation. Am I answering your question, Bill?

MEMBER SHACK: Yes, I just, the question I have is how do you do the SAMDA without the Level 3?

MR. FULLER: You could, in theory, do it, but you absolutely do need source terms and you absolutely need timing. And the only practical way to get that is to do a Level 3 type evaluation. So, you know, what it says on paper and what actually gets done are two different things.

MR. PHAN: PRA technical adequacy. In this section, 19.1.2.3, the applicants stated that the PRA follows the recommendation for why this in Reg Guide 1.200 pertaining to the technical adequacy. The staff reviews the information in the DCDs and issue RAI 6790, Question 19-575, requesting the applicant to provide the basis for the segments in Section 19.1.2.3. First, the PRAs had been developed in accordance with industry consensus standard; and, second, the PRA has been subjected to the peer review process as defined in the ASME/ANS and associated addenda.

Since the PRA technical adequacy is not clearly addressed in the DCD, the staff also requested the applicant to perform a self-assessment for in-house reviews regarding PRA technical adequacy of the US-APWR PRA against the PRA standard and provide the research back to the staff.

Question 19-575 is identified as open items in phase two.

CHAIRMAN STETKAR: So, Hanh --

MR. PHAN: Yes, sir.

CHAIRMAN STETKAR: -- from that, do you expect then, I notice it says self-assessment or in-house review, but it doesn't say peer review. You expect some level of review of the design certification PRA to be provided to you before the design is certified; is that correct? Because when I read through things, it wasn't clearly exactly what level of review or the timing of that review the staff expected to sort of close out this question, so I'm asking --

MR. PHAN: Yes.

CHAIRMAN STETKAR: -- my question in that context.

MR. PHAN: Yes. In the original submittals, the PRAs had been subjected to the peer reviews. So we did ask for the peer review results, so they sent us the findings from that peer review. Based on our evaluation, we recognized that that peer review did not use ASME standard but used NEI 00-02 standard. There are differences between those two process. In the public meetings, we raised the issue, and MHI told us that they withdraw the information regarding the peer reviews. So with that, we requested to conduct an in-house review or another review against the ASME standard and provide that to us. And, yes, we expect them to send us sooner after this meeting.

CHAIRMAN STETKAR: I guess I'm hanging up on the word self-assessment or in-house review versus words that are peer reviewed because I understand that those might be different. I don't know why they would be different, but I understand that they might be. So what -- are you expecting a formal peer review against the ASME/ANS standard to be provided as part of the response to this open item?

MR. PHAN: Yes, sir.

CHAIRMAN STETKAR: A peer review performed according to current guidance for peer review against the ASME standard?

MR. PHAN: Yes. In accordance to the Staff Interim Guidance and will be seen in the SRP, the peer review is not required for the DC applicants. That's why I cannot use the peer reviews here. But, yes, in fact, you used the term peer review or, you know, if

I can't. But, yes, if they can't use, you know, the peer review process using outside evaluators to evaluate their PRA against the standards, that would be sufficient.

MEMBER SCHULTZ: Have you provided a definition that differentiates fairly self-assessment and in-house review?

MR. PHAN: Yes. The peer reviews normally peers conduct by those that not the staff of that company or that nuclear power plant, totally independent from the group who performed the PRA or participate in the development of that design. The term self-assessment or in-house review, it means a staff of that particular company can review the portion of the PRA that they not involved in the development. Even though they involved in the PRAs but they not involved in that particular element, then they can conduct a review for that particular element. In additional, for the peer reviews, the guidance requires the becquerels of years participates in PRAs and others that listed in the NEI's reviews process, and that more tricky than the --

CHAIRMAN STETKAR: So from what you just explained, it's my understanding that you are not requesting a peer review to resolve this open item.

MR. PHAN: No, sir.

CHAIRMAN STETKAR: Okay, thank you.

MR. PHAN: Slide 11. The technical topics of interest on internal events PRAs includes, first, the documentation of key sources of uncertainties --

CHAIRMAN STETKAR: Let me come back to -- you were good. I was silent for 15 seconds, so you thought you'd get out of that. When the staff looks at the quality of the PRA for Chapter 19, in particular the scope and the level of detail, I don't want to repeat my ranting from yesterday, but I'm sure you heard things that I said, examples of things that are not modeled in this PRA that are part of the plant, which, to me, is a scope item. Things that are modeled but to no level of detail, a 0.1 value for the entire main feedwater system, for example, a 0.1 value for the entire gas, you know, things like that. How do you make your determination regarding the fact that that level of detail and scope are adequate? Because, you know, here you've asked for an in-house assessment, but it's easy to ask for other people to give you confidence. How do you do your evaluation? I mean, why haven't you raised questions, or have you? As I've said, we've not seen all of the RAIs, so similar questions about scope and level of detail.

MR. PHAN: For PRA, the staff has three

expectations. First, the PRAs that meet the regulation, that also meet the Commission's goals of 1 to the minus 4 per year CDF and 1 to the minus 6 LRF. For that particular expectation, the conservative PRAs would be acceptable.

The second expectations on the regulations and the use of the PRA to provide the risk insight to improve the design. For that, the PRA need to be more realistic to point out the weakness. With that, the staff believe the PRA is okay to support that task.

For the last expectation that the use of this PRA to support risk-informed decision-making, this PRA not there yet.

CHAIRMAN STETKAR: And that's pretty clear from the SER.

MR. PHAN: Yes.

CHAIRMAN STETKAR: Let me pull you back, though. One of the things that the PRA is used, I keep emphasizing this, is that it is used to inform the Design Reliability Assurance Program list, and that is an output of the design certification. It's something that's adopted in COL going forward. Now, you can talk about expert panels, and we had quite a bit of discussion yesterday that I won't repeat. The scope and level of detail and the balance between what is modeled and what is not modeled in the PRA, the presumed conservatism in some part of a model or data for one purpose, which might be to elevate, let's say, the core damage frequency or a large release frequency, can, in many cases, mask other contributors that are not modeled or are, in a sense, in a relative term, artificially suppressed by these other large contributors.

What happens then is that your core damage frequency may be, I'll pick a number, 10 to the minus 6 with a set of contributors, and those contributors are evaluated according to their relative importance. A more realistic model might have -- and I don't want to throw out numbers, this is just an example -- a core damage frequency of 10 to the minus 7 with a much more balanced set of contributors.

Now, the problem with a more balanced set of contributors is something that's not modeled right now might have a Risk Achievement Worth of a factor of two or three or four to that lower overall total. That piece of equipment right now is not identified as a potentially risk-importance piece of equipment because it's risk achievement worth to the 10 to the minus 6 value would be something on the order of 0.2 or 0.3 or 0.4. And that's sort of the basic concern about getting that balance in there in terms of populating -- and I don't care about the safety-related stuff. The vast majority of equipment that's on the current D-RAP list, although it's 51 pages long, the vast majority of that is safety-related equipment, which is already subject to Appendix B, which is already subject to tech specs, everything else. I care about the non-safety related stuff and whether that has been appropriately identified, whether there's something particular about this design that elevates to importance some non-safety related equipment that we don't know about because we can't see it in the current PRA. And there's not a lot of the evidence from the expert panels population of that D-RAP list that they thought very much about the non-safety equipment. I didn't do a body count. There are some non-safety systems in there, but, as we said yesterday, main feedwater is in there because the whole system is in there and it's modeled as 0.1 in the PRA, so it's Risk Achievement Worth popped up to the top.

So I'm curious does the staff think about that? Because that is another purpose of the PRA in the design certification world, not just for an absolute number do I trip over it 10 to the minus some number.

MR. PHAN: The answer is, yes, we do. You know that I am the lead reviewer for EPR and also Section 17.4. And Mr. Hilsmeier, he's the lead for 17.4 for

APWR. We understand that the PRA would not cover every single component or structures in their models. So we might push it on experience when it deals with the scope of 17.4, not just from the PRA importance rankings. We understand that the ranking is the absolute ranking. However, because the asymmetric issues or other issues, some components may be left out.

So for example, CCW, if one train make the list, the staff would ensure the other train also making the list. We not just look at the ranking. We look at every single component identified in the PRA, even those they included in the models or not. If they mention them, we evaluate from the deterministic, not from the probabilistic, to ensure that if we believe that component or that train, even systems, need to be included in the scope of 17.4, we include that there.

CHAIRMAN STETKAR: Okay. Yesterday I think we asked for a number of RAIs that address sort of that exchange, so I think I'll leave it. We're interested to see those RAIs, they're on those two or three pages of the SER, and see what happened during that exchange.

MR. HILSMEIER: Just to add what Hanh said, the PRA is just one tool for identifying a RAP list.

It can compensate for the limitations of the PRA. You rely on expert panel and engineering judgment. When it rose to the expert panel is to look at those SSCs that were determined to be not risk significant from the PRA, like all the SSCs that have RAWs less than two and Fussell-Vesely less than 0.05. The expert panel who consists of the PRA expert, operations, maintenance, and design and engineering experts, they need to evaluate, okay, yes, the RAW and Fussell-Vesely doesn't meet the threshold criteria, but could it still be risk significant because of limitations of the PRA.

Also, before initial fuel load, the PRA will be updated to the current standards. The standard is in effect either six months or one year. One year? And that's another opportunity which the RAP list would be updated.

CHAIRMAN STETKAR: That's certainly the case. I mean, you know, that's, in some sense, a fallback position that always exists. The problem is, as we discussed yesterday, before fuel load, the equipment is already there, so if that re-evaluation suddenly identifies, and I'll use the example I used yesterday, the heater drain pumps as a potentially risk significant piece of equipment, the heater drain pumps have already been purchased, they've already been installed in the plant, and now the licensee at that time would face the notion that, A, they have to be added to the Maintenance Rule Program because of their risk significance; and, B, they're going to have to convince somebody that, indeed, the pieces of equipment that are there have been purchased and installed and designed and constructed according to the appropriate quality, not Appendix B but some enhanced quality requirements.

It could be really difficult for them. It wouldn't be, it's a surprise that I wouldn't enjoy if I was the owner/operator of a power plant, for example.

MR. HILSMEIER: I completely agree with you. If I was the owner of the plant, I would make sure that the list is as complete as possible before all that equipment --

CHAIRMAN STETKAR: Before I go out for bid specs.

MR. HILSMEIER: Right, exactly.

CHAIRMAN STETKAR: Okay. Everybody agrees it's in everybody's best interest to do that. On the other hand, the law doesn't require me to do that, and I basically follow the law.

MR. HILSMEIER: Right.

CHAIRMAN STETKAR: Anyway, I just wanted

to hear some feedback from the staff while we had you formally up there and grilled. So thank you. And we look forward to seeing those RAIs to see what sort of exchange went on.

MEMBER SCHULTZ: But that's why the process not only asks to look at the information associated with using the best estimate evaluation but also the sensitivities so you can identify some of those key elements of items that could have an impact and should be watched in terms of procurement and construction so surprises don't happen just before start-up.

CHAIRMAN STETKAR: It's just curious that I've seen D-RAP lists from other design centers that have had a fairly, what I'd call a robust combination of safety-related and non-safety related stuff. Although this list is really long, the amount of non-safety related SSCs in this list, which are not modeled in the PRA, is very, very slim, if any. As I said, it's a 51-page table. I haven't studied every last line item. That brings into question how carefully that expert panel thought about things that were not in the PRA.

MR. HILSMEIER: You know, there's a lot of AC power equipment that are not safety related that are in the list. I had to re-look at the list in order to determine, get a feel again for what's not safety related and what's safety related.

CHAIRMAN STETKAR: Okay, thanks.

MR. PHAN: Again, for the internal events, we'd like to cover the documentation, the asymmetric configuration, the digital I&C, and the sensitivity studies.

First, the documentation of the key source of uncertainty, insights, and assumptions. There are two tables that the staff considered to be important in Chapter 19, Table 19.1-38, Key Source of Uncertainties and Key Assumptions; Table 19.1-119, Key Insights and Assumptions.

This table provide key PRA insight assumptions related to the design and operational features with an appropriate disposition. Most of the staff's important findings during this review are documented in these tables, as well, to ensure that the assumptions made in the PRAs will remain valid.

Next, please. Another technical topic of interest, asymmetric configurations. For example, for medium break LOCA initiating event, the PRA assumes the break always occurs at the vessel injection, line A, so that always the impacts on Train A of high injections, accumulators, containment sprays, SRs, are always important than the others.

In Question 19-198, the staff requested the applicant to ensure the PRA are properly adjusted to prevent appropriate conclusion about the risk significance of SSC. In this response, the applicants confronted that. The asymmetric condition due to modeling simplicity have been taken into consideration when reporting PRA results and insight, the use of PRA to support D-RAP.

The applicants also state that PRA will be upgraded before the implementation of risk-informed applications to ensure that the asymmetric additions due to modelings address it. Open items 19.1-Level 1-574, the staff requested the applicant to modify COL's Information Item 19.3(1) to ensure that the asymmetric conditions due to modeling simplicity will be addressed or properly accounted when the PRA is used for decision-making.

Next slide. Digital I&C. The digital I&C in small dose in details in the PRA specifically in the PRA Attachment 6A.13, engineered safety features actuation system, and also in Attachment 6A.12 on reactor trip.

During the staff reviews based on the staff's findings, the I&C fault trees was revised to address the hardware common cause failures, the application software common cause failures, dependency between automatic and manual actuation signals, application software diversity, and to include other failures such as input module power supplies and communications between the RPS trains and so on.

There are two kinds of software common cause failures included in the PRA models. First, the basic software common cause failures. This type of common cause failures is defined as a failure of the MELTAC operation systems which encompasses the common software for all application. This common cause failure was estimated to be 1E minus 7. Second, the application software common cause failure --

CHAIRMAN STETKAR: Let me stop -- well, get to the bottom, and then we'll go back.

MR. PHAN: The second common cause failure on the application software, this failure would result in the loss of all of S-signals and P-signals, and the applicant's estimate 1E minus 5. And based on the findings, the hardware common cause failures also included in the models with 2.1E minus 6.

CHAIRMAN STETKAR: Now let me go back to the third bullet there, and Charlie brought this up yesterday, but I wanted to investigate this a bit with the staff. Twenty million hours is about two-thousand -- I can't remember -- roughly --

MEMBER SHACK: Two hundred and eighty-three.

CHAIRMAN STETKAR: Thank you. I remember, 2,300, but I could have been off by a couple of hundred. It's a little over 2,000 years. Obviously, we've not had MELTAC platforms installed for 2,000 years, I don't think. Now, what we've learned from doing real uncertainty analysis is that the experience of 100 plants that have had 100 trips in a period of ten years is not the same as each plant having one trip every year because there's variability. The actual experience is ten years, ten plants, and what is the operating experience?

Similarly to this, operating, you know, and I'll try not to be overly sexist here, having nine women pregnant for a month each does not produce a baby. So my question is how is the accumulation of bits and pieces of MELTAC operation in many units equivalent of 20 million years, I'm sorry, 20 million hours or 2,000 years of operating experience with a particular system? And how has the staff accounted for that? There are ways to account for this. It's called Bayesian analysis. You look at the evidence available from each plant and account for the plant-to-plant variability. Some plants might have one year of operating experience, some plants might have 15 years of operating experience, none of which has had any failures. That will give you a much different value, I would think, than 1E to the minus 7 per demand. Has the staff really questioned the applicant regarding that value or different approaches to estimate that value?

MR. PHAN: When we received the response from the applicants, we questioned the numbers 20 million hours. We did roughly convert using 8,760 hours per year back so many years. So we just assumed that they have 30 units out there, so if you divide by over 200 years by 30, you would have, you know, even those we don't believe that, you know, all the system would be operating all the times.

But we have another issue here because they providing the numbers to estimate the failures per hours, not the failure per demand. There's no correlation between these two numbers. They give us one thing, and they concludes the other site. I kept that in mind. I did not go back and ask them what the correlation here, how do you convert from failure per hours to failure per demand? Now that you've justified the numbers of 1E minus 7, you need to provide us more details on the database that you collect per demands, including the number of demands, and so on.

But the technical reviewers who responsible for this particular question, he just stop right there and told them put that in the Table 19.1-28, but the key uncertainties, so because he believe that, by the times the plant is constructed, this number may not be --

CHAIRMAN STETKAR: I understand that. And I had sort of a generic notion. In any kind of thorny area that I read in the SER where the staff sort of probed and pushed, this is one example, the component cooling water common cause parameter is another, there are a number of them, those issues seem to be basically punted to that wonderful Table 19.1, whatever it is, 118. This is a key assumption, somebody else go figure it out later down the road. Suppose it's just wrong. You know, isn't that something that ought to be resolved now, rather than just punted down the street and say somebody else go worry about this? Here's something you need to worry about. We've identified it. Go fix it later. That I found so many places. It certainly is a key assumption. Suppose it's wrong.

MR. PHAN: I agree with you that every single item we identify in Tables 38 and 119 need to be addressed during this phase but because --

CHAIRMAN STETKAR: Not during this phase. It says later on.

MR. PHAN: Yes. Right now it's saying that later on.

CHAIRMAN STETKAR: I don't know how to get past this, but there were several things, and you orally this morning just kind of gave an example. You said the reviewer got to a point where he just said, oh, we'll just put it in that table as a key assumption and probably later there might be more data available or there might be other information available or there might be something available; we don't need to worry about it now. And that, to me anyway, is a bit troubling, only because there are ways to better justify some number. We won't ever know what that number is. There's large uncertainty associated. We won't ever know what that number is, but there's certainly better ways than is done in this particular application to estimate what that number might be.

MR. PHAN: Yes. We not try to excuse for ourselves. But you know that during the DC process, many information not infallible, like EOP not infallible, even the correlation between humans and machine interface not infallible. So we must have a list of assumptions, depending on how the reviewers or the readers decide that this key assumption is significant or not during the DC phase or DC stage. That need to be resolved at this point or in the future that different reviewers has different expectations.

CHAIRMAN STETKAR: Yes, I hear what you're saying, and I understand the constraints that you're operating under for the DCD. It's just somehow troubling, only because I've seen a lot of other estimates for those CCF parameters and other things and they range all over the place. They're basically a number that people use so that the results come out okay. And don't laugh. I hear snickering back there, but, quite honestly, that's a sense that I get many times. And I guess we'll just leave it there.

MEMBER SCHULTZ: Well, the assumptions and the math should be checked, at the very least, because it doesn't seem --

CHAIRMAN STETKAR: Well, the problem is I still put myself, I put my potential licensee hat on and think about the issues that I will now need to address when I produce that plant-specific PRA after the COL is issued that meets all of the standards and has to satisfy a, hopefully, very, very rigorous peer review that challenges numbers like this. And that's going to be a very, very significant burden for some of those licensees because they're going to need to grapple individually, in principle, with all of these difficult technical issues. And it's a strategy, you know. I just hope that the licensees, in the PRA area in particular, recognize what effort they may be facing.

MS. MROWCA: Dr. Stetkar?

CHAIRMAN STETKAR: Yes.

MS. MROWCA: Can I just add to that, too? This is very difficult for us, too. I may be shouldn't include myself as a PRA practitioner, but our technical reviewers would love to see everything modeled so that we can have more confidence in the PRA, but it's a balancing act between what do you do during the design certification phase and what you do later. And I think that that's why you see a lot of those sensitivity studies and why Hanh already mentioned the importance of these two tables. So it really is a balancing act. Where do we draw the line?

CHAIRMAN STETKAR: And I appreciate that. It's just that -- well, I'll just leave it. I won't repeat. Thanks.

MS. MROWCA: We understand your concern.

CHAIRMAN STETKAR: Yes.

MEMBER BROWN: Can I ask a question?

CHAIRMAN STETKAR: As I said yesterday, I can't say no so . . .

MEMBER BROWN: Well --

CHAIRMAN STETKAR: And you're free to talk about all these numbers because I would not have gotten it right. That's why when you spoke up, I figured I'd short-circuit you there.

MEMBER BROWN: I guess I tried to focus on something a little bit different. Aside from the numbers again that gets played in the PRA, I tried to look at it from a different level. In some of the hardware CCF, there's a statement made, and this is in the RAI response to the 19-515 that you all provided, the answer to it where they comment that the hardware CCF results in no actuation of all automatic signals in the PSMS. In other words, any common hardware failure is not -- the way I read this, there's four trains of equipment. I'll just pick the reactor protection system, reactor trip system, whatever you want to call it, as an example. There's four trains, and there's two controllers, a digital and MELTAC platforms, at least in each one. Correct me if I'm incorrect, Mitsubishi. And so a common cause failure in all of those, you could say, well, gee, that's eight systems, I can't imagine they'd all fail at the same time. However, you never know how they're going to respond when they're facing the same challenge. If a challenge to one will make one fail, a challenge to the next may make it fail, etcetera, etcetera, on down the line.

The argument on this that can be used that, gee, we've got a diverse actuation system and, therefore, we've compensated for that relative to having this diverse actuation, which is not computer based. And I haven't gone through this one in detail. I haven't read all the paragraphs in the DCD yet, but I'm hoping when I do that that the DCD does say that the diverse actuation system is and gives some definition of what the different technology is so that it's just not another microprocessor-based system. And I didn't find that with a quick review of keywording, which means I probably have to read the whole thing which is going to be laborious.

But it seems contrary to my thought process just to say the purpose of a common cause failure is they all fail when challenged. Whether you want to believe it or not is another issue.

The same thing applies when you get down to application software, for instance. Let's say, in the PSMS, the reactor protection system consists of two separate controllers in each train to achieve defense-in-depth through functional diversity, as explained and described in DCD Section 7.2.1, and it happens to be 7.2.1.2. And there they talk about how they measure two different parameters, one with one controller and one with the other. But it doesn't say what parameters, and it doesn't attach or assign those parameters to specific accident sequences to ensure that those two parameters are two different means of assuring a trip for that particular severe accident design basis accident sequence.

So I was looking for that type of information. When I looked at NUREG-6303, diversity is defined, functional diversity is defined somewhat differently in 6303 relative to this. The IEC standard for this talks about two different parameters. I'm not sure, if my memory is, that if we don't deal with IEC standards, we deal with U.S. standards, if I'm not correct. And 603 doesn't exactly fall into the category for the functional diversity, and that didn't seem to be challenged at all in terms of the discussion.

These are just some higher level, as opposed to the numbers aspects, in terms of looking at this. Like John, I'm going to pass, at least at this point, because I've got to do a little bit more looking, but I suspect I'll be bringing the point up in the Chapter 7 discussions if I can come up with some rationale thought process. I'm trying to look for ways to say this stuff is okay and say that it's acceptable to us as a committee. So that's just a little heads-up in terms of a slightly different take relative to the number crunching that everybody has been doing.

MR. PHAN: Thank you, sir. Thank you for your past.

MEMBER BROWN: I know you're much appreciative of detailed wholistic thought processes.

MR. PHAN: Yes. Throughout the reviews and by the responses, we understand how the models in the software common cause failures and the hardwares, the failures of four trains and the failures of the digital controllers and the backup of that at the systems. But to give you a little confidence in the staff reviews, at least go to the next slides, the last bullet, open items. Right now, we say resolved, but still that provides more definition how they assume in the PRS regarding what the signal actually impact by the common cause failures and which components in that common cause --

MEMBER BROWN: So you're still looking for more information. I gathered you were still waiting for some additional -- I wasn't quite sure how complete this response was. I'd just like to reiterate that all, and maybe my thought process is not valid, but all this is a way of saying, well, gee, this stuff will never really break in a manner in which it's going to compromise our ability to do business, the computers. And, quite frankly, I don't trust a computer any farther than you can throw it, which is farther than you could in the old days. These are pretty light, so I can throw those a lot farther than the giant machines that we had in the past. But, fundamentally, you have to assume that the software fails, period, or that you've got corrupt information going from one train that permeates all four and they all stop. They just lock up, and the plant should shut down under those circumstances. So I will be looking, personally, through the DCD and the other plant description to ensure there's a suitable method that if they all lock up the plant will shut down and how that occurs in some definition with some specificity, not just the higher-level thought, well, gee, it's going to do something, but we want to know how because it needs to be done with analog or non-software based functions once they all lock up.

Anyway, that's just down the line to cover this. That's how I always walk my way past all this stork dance that we do with probabilities and everything else in the PRAs. I'll stop right now, and you can proceed, unless somebody else is --

CHAIRMAN STETKAR: I was going to say, in some sense, we get paid, in a very vague sense, get paid to beat up people. And in some sense, this is the PRA where I've seen the best, by far, models of the whole I&C system. There's a real design, there's a real model. You can trace signals. It's not a single number, as we've seen in other design certifications for some amorphous failure of all of that stuff. So in some sense, actually, MHI is paying a bit of a price because they're getting more questions, I think, in the digital I&C area in this particular PRA because they actually have a real model of a real system that people can look at. And, indeed, the model that they have, there are questions about, there's always going to be questions about the software common cause failures, there's always going to be questions about the boundaries that you define around either chunks of hardware or chunks of software or whatever you do. But in terms of the basic architecture of the system, it's well represented in terms of its dependencies on power supplies, shared signals between DAS and PSMS. That's all in there, you know. It's infinitely better than we've seen in other design certifications, for example. And I think, you know, we certainly, as a subcommittee, recognize that. It's a pity that if you put something in the model you get questions about it, but this is actually a heck of a lot better model of the real plant design than I think any of the other digital I&C systems that we've seen.

One other vendor had a real design, and they did a pretty decent job, I'd say, but not at this level of detail. Others basically didn't model it at all. So that's just something to sort of raise also in the context of our sort of pervasive negativity in attacking things. I was pretty impressed with a lot of it.

MR. SPRENGEL: We appreciate that positiveness. And, Charlie, we'll look forward to additional discussion in Chapter 7.

MEMBER BROWN: I had one other question that I'm just trying to make sure I understand. We tend to focus on the automatic trip functions, reactor protection, and generic safeguards, etcetera, etcetera. But one of the things that is mentioned throughout is the manual, there's an operator there who can take manual actions, if necessary. He can go insert rods. He can exercise or trip the SCRAM breakers, or he can actuate certain safeguards functions, etcetera, etcetera. If you look at the basic system architecture, and all I've got is the big one-line diagrams that are in the DCD and the other plant design document, all of those signals for the manual control go down to the same unit bus and then down to the systems for the actuation of your stuff, controls, as do protection signals go monitoring up into the main control room and others. And they all use the -- then that's a much less diverse network bus, and I haven't seen anything that talks about failures of that bus in the ability to even provide the manual backup functions, if necessary. We've got a remote shutdown console, and then we've got a diverse actuation system. If somebody has to run over and operate the diverse actuation system, then I presume that's hardwired. Does that shake your head up and down? Is that correct? No computers involved? MR. SPRENGEL: That's correct. It's an analog system.

MEMBER BROWN: All right. I was hoping that was the right answer. But on the rest of the stuff, it's almost whether they've got two little network lines that you can run through. That's the least defense-in-depth ability to get control signals back down to the plant to actually actuate some of these safeguards or other protection functions. I don't know whether you've looked at it or not, but I would suggest you might want to at least see what the configuration looks like and how vulnerable is that relative to its manual operations, as opposed to just the automatic stuff. That's it for me on this page.

MR. PHAN: Thank you. Slide 15. We already talking about DAS. One point I'd like to mention here that there's no fault trees developed for DAS. MHI estimates the failures of 1E minus two probability, but there's backup system. Per staff request, that's the common cause failures of software and hardwares. As mentioned in the last bullets, they are included in the DCD analysis in Table 38.

But for the open items of both those that we've been talking here that the staff need more definition, more explanation regarding the I&C hardware and software. And we'd like the applicant to document that in the DCD, clearly document that in the DCD, as well.

Okay. Next. Sensitivity studies. The PRA includes a wide range of sensitivity study. They identify in the Tables 19.1-140. This including a list of study mentioned here. We also documents them in the staff's evaluation. Mostly, those sensitivities testing the CDF and LRF impacts due to the numbers assignments.

Next one, please. Based on these sensitivities, we have some important insights that we'd like to share with you. The CDF is sensitive to several common cause failures particularly. CDF is not very sensitive to an increase in single component failure probability or initiating event sequences. CDF is not significantly sensitive to further reduction in safety system outage times for tests and maintenance. And CDF is not significantly sensitive to further reduction in human error probabilities.

CHAIRMAN STETKAR: This question I brought up about using the fifth percentile of the uncertainty distribution for human reliability. Did you flag that? I didn't see where you flagged that anywhere.

MR. PHAN: I did not. Based on the discussion yesterday, I went back and I could not find anything from that information for the issue you just raised there. So I don't have any information at this point on that particular --

CHAIRMAN STETKAR: In the review you mean?

MR. PHAN: Yes.

CHAIRMAN STETKAR: Okay, thanks. I mean, that doesn't affect the sensitivity study, obviously, because if you fail all of them it doesn't make any difference what number you actually used in there.

MR. PHAN: Yes.

CHAIRMAN STETKAR: Thanks. What I'd like to do now, before we start talking about fires, as long as none of the Committee members have any questions about the internal events review that we've just heard about, I think what I'd like to do is take a break now. It seems like a reasonable break point. And then we'll come back and talk about the remainder of the presentation.

So we will recess until 10:30.

(Whereupon, the foregoing matter went off the record at 10:16 a.m. and went back on the record at 10:35 a.m.)

CHAIRMAN STETKAR: We're back in session. I didn't have my timekeeper to tell me I was late, so I had an excuse. We'll pick up with the staff's presentation.

MR. PHAN: Okay. We would like to continue with the internal fires PRA, and we're going to talk about the fire protection concept, the use of NUREG/CR-6850, the major assumptions, and fire PRA insights. As reported yesterday, you saw that the fire PRA CDF is lower than any operating plant because the PRA is built based on the following concept. First, each of four safety divisions is separated by the physical fire barriers. Second, safety-related components and cables are separated using three-hour fire-rated protections. And, third, US-APWR is designed to be built using all qualified cables.

Next, please. NUREG-6850 methodology. The fire PRA is developed using the guidance provided in NUREG/CR-6850. The PRA includes all tasks, except Task 8, scoping fire modeling. This task has two main objective: first, to screen out the fixed ignition source that do not pass to the components or to the targets; and, second, to assign the severity factors to the ignition source. So by keeping this step, the PRA is conservative in the estimation.

Next, please. PRA documents. The regulation do not require the applicant to submit the PRA. However, MHI voluntarily submits their PRA documents, even those they label as proprietary information. I am listing those files related to the fire PRAs. By reading the titles, you will recognize the depth of the information that they provided to us.

Next, please. Major assumptions. In the staff's Safety Evaluation Report, Section 19.1.4.5.2.1.1., we document all the key assumptions, including in the PRAs. There are 30 of them. These are the key drivers to the CDF and LRF. Table 19.1-119 of the DCD also documents the key insights and assumptions that the COL's applicant validates and verifies during the COL stage. COL's information items in DCD Section 19.3 would ensure that these key assumptions will remain valid for the as-built, as-operated plant. To ensure that the key assumptions would remain valid during the DC stage, not the COL stage but during the DC stage, Section 19.1.2.4 state that any changes to the assumptions relevant to the internal fire events will be incorporated into the PRA as part of the PRA maintenance process.

Fire PRA insights. There are a number of fire PRA insights, but the key ones are on this slide and also on the next slide. First, the models. The model does not credit any mitigation functions of the fire detection/suppression and fire brigade. The most significant fire areas are the LOOP due to switchyard fires that has the highest CCDP (conditional core damage probability) and the turbine-bypass valve due to turbine building compartment fires. This scenario contributes about 53 percent to the total fire CDF. Third, the --

CHAIRMAN STETKAR: I didn't have a chance to look at those scenarios. I was going to last night, and I got sidetracked. Are those turbine building scenarios simply modeled as a steam line break outside containment with everything else in the steam line break model operating? Do you remember, or do you have the reviewer --

MR. PHAN: Oh, would you please give me one second?

CHAIRMAN STETKAR: Or maybe MHI might remember. I was just curious that that particular effect from a fire in the turbine building is highlighted as the biggest single contributor to the fire CDF. Simplified models of the turbine building, you know, in the past, I'm sort of familiar with. If you have fairly large compartments in the turbine building, they tend to show up as important. But in most cases, they show up as important as loss of main feedwater, not stuck-open turbine-bypass valves. And that's the aspect of that fire scenario that I was curious about, and I was wondering whether you had, your reviewers had delved into it very much.

MR. PHAN: Based on my understandings, the turbine building's importance ranking high because they assume there's a large amount of ignition source in there.

CHAIRMAN STETKAR: True.

MR. PHAN: Not just because --

CHAIRMAN STETKAR: No, the frequency is high, and, indeed, turbine building fires happen in the real world. Not large ones. Occasionally, they're large. But I'm more interested in the assigned consequences from that fire, in particular the stuck-open turbine-bypass valves. And the only reason I'm interested in that is that the other internal events models do not include the turbine-bypass valves, so they have no chance of sticking open, except, apparently, in a fire. And I'm curious about whether there's something in that fire model, some additional effect from the fire, that enhances that particular failure mode's contribution to overall risk compared to, you know, a plant trip scenario where the turbine-bypass valves stick open. So if you don't have it, I just thought you might because it's the largest contributor.

MR. PHAN: May I take that as actions --

CHAIRMAN STETKAR: Sure. That would be great. You know, we'll be here this afternoon. If you can find something quickly, noontime, I'd appreciate that. If you can't, that's fine, also. Thanks.

MR. PHAN: Thank you. The next one, the third bullet up there on the hot short, the applicants assume of 1.0 always failures for hot short. We raised the issue in our RAIs, and they conduct sensitivities.

And based on their study, they say the hot short is low with that. They not adjust to the 0.3 that's recommended in the 6850 NUREG/CR.

CHAIRMAN STETKAR: Although it's hard for me to think about how the turbine-bypass valves would open spuriously without hot shorts since their design to fail is closed. So when you say the contribution from hot shorts is low, I'm not sure what that 53 percent then comes from. Low perhaps in an absolute numerical sense but . . .

MR. PHAN: My assumption for now is it's not included in the models. That's why the impact is not included there. That's why the conclusions say hot shorts will not be an issue.

CHAIRMAN STETKAR: Okay. Anyway, maybe we'll get some resolution this afternoon. Thanks.

MR. PHAN: Yes. All fire compartments, except the containments and the switchyard, to be composed of the fire resistant for all four ESF trains. They are all individually separated.

Slide 24. The fire PRA identifies no significant multi-compartment fire scenarios. Based on the CFAST simulations, fires in any fire compartments in the containments would not spread to the adjacent compartments. Electrical room in turbine building is separated into two compartments that also have the reductions in the fire risk. The operator actions at the remote shutdown consoles during the main control room's evacuation are the only new actions added to the fire PRA. A sensitivity analysis, assuming the failures of the probability of 1.0 show an increase of the CDF, the fire CDF. The most significant fire actions relevant to fire events is the connection of Class 1E bus to the alternate AC in case of four Class 1 gas turbine diesel is unavailable.

MEMBER BLEY: I'm a little curious, given what you had on the previous page. I didn't get a chance to really study the fire PRA. If the most significant fire scenarios are the switchyard fire and this fire that causes the turbine-bypass valves to open that somehow isn't a hot short, then why does failure to move out of the main control room double the fire CDF? Because those two -- well, I guess whatever fires open the turbine-bypass valves might be associated with the control room, no?

CHAIRMAN STETKAR: No, that's a turbine building fire.

MEMBER BLEY: Then how do we double the fire CDF if those are the dominant contributors? And failing to evacuate the control room, I don't know how that affects those two --

CHAIRMAN STETKAR: Well, it could if the, let's say a main control room fire with initiating event frequency was, I'll pick a number, 10 to the minus 4, and you said that they're going to successfully, you're going to successfully recover that at 10 to the minus 5. You know, if you made that 10 to the minus 5 one, that main control room fire would suddenly show up as a big contributor. I think that's what they're trying to say. Although the frequency of that main control room fire might be a lot lower than the turbine building fire, accounting for very good success -- it does show, you know, question the value that's assigned for successful abandonment and control from the remote shutdown area.

MEMBER BLEY: Okay.

MR. PHAN: Any more questions on the fire PRA before we get to the internal flood PRA? Please turn to slide 25. For internal flood PRA, we will cover the flood protection concept, the methodologies, the major assumptions, and the PRA insights.

The internal flood PRA is based on the following concept. Prevent the flood propagation to multiple mitigation systems (more than two out of four trains) by: first, separation of the reactor buildings into two areas, east and west sides; installation of water-tight doors for safety-related SSC areas, safety-related I&C rooms, and main control rooms; and, third, isolation of essential service water pump to prevent inflow from reactor buildings. Another concept that prevents inflow to the reactor buildings from other buildings and also install the flood relief panels on the turbine building's external walls to drain water from the circulating water system to the yard.

Twenty-seven, please. The internal floods PRA using both qualitative and quantitative analyses. The qualitative analysis includes: step one, identify independent flood areas and SSC; step two, identify flood sources and flood mechanisms; perform plant walkdown during the DC stage, the tabletop examinations were performed instead of the actual plant walkdown; and, step four, perform qualitative screening.

Next slide, please. The quantitative analysis includes: first, develop flood scenarios for each flood source; step two, perform flood-induced initiating event analysis; step three, evaluate the impact on equipment; step four, evaluate flood mitigation and perform human reliability analysis; step five, develop the PRA model; and, step six, quantify the model.

MEMBER BLEY: Hanh, this might be as good a place as any to ask you. I think you heard the discussions we had with the applicant about that fire scenario with the emergency service water rupture affecting component cooling water system, and if you could, maybe you can make me a little more comfortable with their analysis about why that 15-minute time period is reasonable, what kind of leaks they assumed, and did it consider all the possible leaks, and, you know, the assumption that there's perfect operator action within 15 minutes.

MR. PHAN: If I remember correctly, in one of the staff discussion with the applicants, they explained to you why they came up with the 15 minutes. Please, give me one second and let me put together my thought here. In the flooding PRA, the main feedwater is not relevant for any mitigations proposals, only as the initiating events, the pipe rupture. So the mitigation of that pipe rupture would reduce the flood sequences but not in the modelings. So with that, there's no modelings regarding the, in the flooding PRAs, but not from the EFW.

MEMBER BLEY: I don't think that quite works for me, so I'm going to have to wait until we can look a little harder.

MR. PHAN: Yes.

MEMBER BLEY: We talked to them about, you know, what kind of ruptures did they consider, why 15 minutes, you know, what kind of flooding rates were there, what happens in 15 minutes, and then how the operators really determine what's leaking and what they'll turn off the particular pump within 15 minutes with probability failure of zero for that. All of those things were things I didn't quite follow, and I was wondering if you dug into those at all and can explain them.

MR. PHAN: Yes. For EFWs, the applicants told us that there are indication from the main control rooms for any floodings or any water spilling and other indication due to pump failures also indicated in the control rooms, and they told us that the control rooms would not dispute that going to isolate the rupture but another room would be handling for that. So they say, with that, because two different of people with buildings with the floods. That's why they have a low probability but still -- I don't think I answered your concern here.

MEMBER BLEY: I don't think so. In my experiences, a sump alarm going off isn't something that is the highest order of attention immediately when it happens.

CHAIRMAN STETKAR: You'd probably send somebody out there to see whether there's water in the room first.

MEMBER BLEY: Usually, yes. And the perfect human performance, feeling it all out and responding, and even if the 15 minutes is reasonable, all of those things were concerns. So we'll look a little harder ourselves. I don't think you did.

MR. PHAN: May I take that as an action to go back to see any staff --

MEMBER BLEY: I'd appreciate it.

MR. PHAN: -- discussion in those particular areas.

MEMBER BLEY: Thank you.

MR. PHAN: Thank you. Okay. Next slide. Here are the files for flooding PRA and the information to support the PRA's development. Okay. Slide 30. We documented all of the major assumptions in the safety evaluation. We identified 37 key assumptions relative to the internal flooding, and those are the drivers of the PRA and the CDF estimation. The COL information items would ensure that these key assumptions will remain valid for the as-built and as-operated plant. And like fire PRA, DCD Section 19.1.2.4 would ensure that any change to the assumptions will be evaluated and incorporated into the flooding PRA.

Slide 31, please. The key internal flooding PRA insights are summarized on this slide. First, the most significant areas are the second floor corridors of the reactor buildings where EFW piping is located. And, second, the steam generator radiation monitor room and the turbine EFW pumps room are also important.

The most significant system contributing to the internal flooding sequences are the emergency feedwater system, the main feedwater system, the main steam system, and the circulating water system. The most significant system contributing to the internal flooding risk is EFW, and the most significant operator action contributing to the internal flood risk is to perform EFW switching.

On slide 32, I just quickly summarized the external events risk evaluation. Like mentioned, the staff's evaluation on the seismic, the PRA-based seismic margin assessment will be provided to you later. All of the external events will be addressed by the COL applicant. To ensure that the COL applicant will be addressing all of these external events in the application, the COL information items 19.1.3 is developed to reprise the COL applicant to address the external events from the screening of the PRA from the quantitative assessment.

With that, I'd like to turn it over to Ms. Marie POHIDA. She will be presenting to you the low-power and shutdown PRA.

MS. POHIDA: May I begin? All right. On slide 33, we have the outline of what I'd like to discuss this morning. And the first item is shutdown tech specs or, rather, lack of in Modes 5 and 6. And what I'm concerned about here is the availability of standby RCS injection and containment closure before boiling and reduced inventory operation.

The second item is containment closure, and what I'm looking at is actually the feasibility and probability of successfully closing containment or re-closing it before boiling during reduced inventory operation. The third item is the omission of draindown events during POSes 5, 6, and 7, and that's when the refueling cavity is flooded.

The next item is the auto isolation of letdown and the initiation of vortexing in hotleg. And this design has automated isolation of letdown when RCS level reaches a certain set point in the hotleg to protect the arch R pumps from air ingestion. And what we're concerned about is where that point of vortexing initiates in the hotleg and just to make sure that set point is appropriately placed. The last topic I'm going to be discussing is hotleg level instrumentation during reduced inventory operation.

Okay. The first topic is shutdown tech specs. And if I may begin with the regulations. According to 10 CFR 50.36(c)(2), a technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criterion. And I'll direct your attention to Criterion 4, a structure, system, or component which operating experience or PRA has shown to be significant to public health and safety.

Slide 35, please. During my review, I found that there were no tech specs for standby RCS injection and containment closure during reduced inventory operation. And what I'm talking about is standby injection. I'm talking about the pumps, a path of pumped injection that is in addition to the pumps as part of the normal decay heat removal function. And there was no tech specs for containment closure during reduced inventory operation.

If you take the MHI PRA and you remove credit for standby RCS injection and containment closure, the Commission's goals for new reactors are exceeded.

MEMBER BLEY: Marie?

MS. POHIDA: Yes.

MEMBER BLEY: I'm sure that most of our operating plants have committed to close the containment during mid-LOOP operations, but is that not a tech spec at other places? Is it just a practice? Do you know?

MS. POHIDA: There would be a practice if they're following the guidance of Generic Letter 88-17. That gives guidance to operating PWRs the need to close containment before boiling so that you can close it before containment conditions become intolerable. But, no, it --

MEMBER BLEY: So that's the only --

MS. POHIDA: It's not a tech spec in current plants.

MEMBER BLEY: So plants that are doing that, it's just a plant policy?

MS. POHIDA: It's a voluntary initiative.

MEMBER BLEY: Okay, thanks.

MS. POHIDA: Okay. Going back to my previous bullet, you take the MHI PRA and you remove the capability or the availability of standby injection and containment closure, the Commission goals, by my calculations, my calculations are exceeded. MHI did a sensitivity study to analyze the same thing, and they also found the Commission goals to be exceeded.

Containment closure is not credited in the PRA. And I just want to note that standby RCS injection and containment closure before boiling during reduced inventory operations, these actions are identified as expeditious actions in Generic Letter 88-17. And that generic letter was written to improve the PWR's ability to mitigate extended losses of RHR during reduced inventory operation and mid-LOOP, which is a subset of reduced inventory operation.

The Tech Spec Branch asked MHI in an RAI on how Criterion 4 was applied of 50.36 and what tech specs were added. And in that RAI response, MHI agreed that the lack of safety injection did not meet Commission's goals. However, they proposed administrative controls in lieu of tech specs. And based, and what we've concluded is that options for tech spec LCOs for safety injection and containment closure are required under Criterion 4 of 50.36.

We're in the process, the staff is in a process of drafting a letter to document our position to MHI. This is a rather old RAI. This topic was discussed in the PRA audit in May of 2011. We had numerous public phone calls on the issue. The last one that I participated in was April of 2012. So that's the current status of that issue.

Containment closure. What we're doing now is evaluating whether the manual actions to close containment before boiling are feasible and reliable. And what I mean by -- I'm sorry?

MEMBER BLEY: May I ask you something about that?

MS. POHIDA: Sure.

MEMBER BLEY: Because I'm reflecting back many years, 20 - 30 years ago. People, as soon as they were down, would take off the hatch, and they'd actually run temporary piping and cabling and everything else through their -- in the past even send off that hatch for refurbishing. So when they're answering this, have you made sure that they have the capability to remove any temporary cabling and piping that might be running through the hatch and the estimate of how long it will take them to close it up?

MS. POHIDA: Anything that could impede hatch closure would need to go into our assessment of whether this action is feasible before boiling because time to boiling is under a half an hour.

MEMBER BLEY: Maybe this comes up under the COLA. I'm not sure because that's a practice during maintenance outages that, you know, isn't spelled out anywhere, I don't think, in the application.

MS. POHIDA: Can you repeat that?

MEMBER BLEY: Okay. And I don't know what current practice is or what they've committed to. But in the past, people would run temporary piping and cables --

MS. POHIDA: Oh, yes, we've seen that --

MEMBER BLEY: -- for that hatch.

MS. POHIDA: -- in the ROP process.

MEMBER BLEY: And sometimes it takes hours to get that stuff out of the way. So if you're claiming you can close it up in 15 minutes or an hour, it might not be feasible, unless there's some controls in place to make sure that those things are easily removed.

MS. POHIDA: I agree. Based on my experience in the ROP process, evaluating performance deficiencies during shutdown at operating plants, you know, we've had issues where people have had to install rail track --

MEMBER BLEY: Yes --

MS. POHIDA: -- closure. You know, so we've seen a multitude of issues on why people would not be able to close the hatch before boiling. So, yes, that would go into our assessment. What they would need is to keep very careful track on all containment penetrations are open. You know, you would need to have pre-staging with people, equipment, communications. There would have to be infrastructure to justify that you could close the containment so quickly.

MEMBER BLEY: I guess that's what I'm getting at. I mean, you do an analysis now, but in 15 years somebody is doing an outage, what's there to ensure that it's still true and for the site inspectors to have some guidance on what to look for?

MS. POHIDA: Well, other design centers have tech specs for this.

MEMBER BLEY: Oh, is that right? Okay.

MS. POHIDA: So it becomes the part of the licensing basis for the --

MEMBER BLEY: Yes, if it's in the tech specs --

MS. POHIDA: Yes. So that helps a lot to guarantee the infrastructure is there.

MEMBER BLEY: Okay.

MS. POHIDA: Okay. Does that help you?

MEMBER BLEY: Yes, it helps me a lot.

MS. POHIDA: Okay, great. So when we're talking about re-closure, what we're talking about here is containment closure consistent with Generic Letter 88-17. And it's basically a barrier to the postulated release of fission products following a severe accident shutdown. We also have a side RAI on the igniters and whether they're needed to keep the containment intact once it's closed to control hydrogen.

Okay. So also, based on staff RAIs, MHI will implement a design change to use the alternate AC generators to power the equipment hatch hoist, in addition to off-site power, which helps, but in Chapter 8 of the DCD it states that power from these alternate AC sources can be restored within 60 minutes. And you might run into a snag if your time to boiling is under a half an hour. So we need to ensure that containment closure is feasible, and we're going to be drafting supplemental RAIs on this issue. There's still work to be done.

Okay. Omission of draindown events. Draindown events when the refueling cavity was flooded was omitted from the PRA, and during my review of the PRA I was concerned about draindown events when the cavity is flooded, particularly when temporary fuel racks in the refueling cavity are used. There's two racks in the refueling cavity, and each rack can contain three fuel bundles, so that's a total of six bundles.

So what I'm evaluating is potential drain paths and the availability of level indications and alarms when these temporary fuel racks are used and that someone, if there's a draindown path that's created, that conditions are acceptable for the operator to go and terminate the drain path. Now, MHI, we asked, MHI RAIs on this issue, and they judged that the risk was small, given the largest draindown path, and the probability of failure that the operator fails to isolate the drain path is small when these racks are used.

CHAIRMAN STETKAR: Marie?

MS. POHIDA: Yes.

CHAIRMAN STETKAR: When you quizzed them about that, was the risk smaller than 10 to the minus 13 times the general transient initiating event frequency, which is a quantified value for failure to SCRAM in their PRA? Was it small compared to that? Because small is relative. Is it small compared to the large LOCA risk?

MS. POHIDA: I agree. I'm concerned because, you know, small is small for this design, and I need to ensure that this risk is small given that the internal shutdown CDF is 2E minus 7.

CHAIRMAN STETKAR: Right.

MS. POHIDA: So we have to make sure it's really small. We're still evaluating this.

CHAIRMAN STETKAR: Okay. So they didn't come, they just came back with a qualitative response that it's small, or did they try to quantify it --

MS. POHIDA: I'm going to need to go back and check.

CHAIRMAN STETKAR: -- a little bit more? Okay. If they tried to quantify it, at least, you know, you try to get a handle on what is small. Qualitative statements about small contributors don't mean much, as you said, in the context of these types of --

MS. POHIDA: It's small.

CHAIRMAN STETKAR: -- PRAs.

MS. POHIDA: I'm trying to think about the actual RAI response. You know, what was done was evaluate individual drain paths and, you know, looking at drain path size, if they're eight inch or four inches, looking at the valves, you know, locked, manual closed and they have to be opened. But the problem is, you know, based on our review of shutdown experience, you know, when people go to manipulate the plant, you know, people open up locked closed manual valves to establish drain paths, you know --

CHAIRMAN STETKAR: You and I know about a plant that lives in the state of Tennessee, I believe, where people did that.

MS. POHIDA: And water moved very quickly.

CHAIRMAN STETKAR: Water moved very quickly.

MS. POHIDA: Very, very, very, very quickly.

CHAIRMAN STETKAR: No, this was Sequoyah. They actually --

MS. POHIDA: I was actually thinking of another one.

CHAIRMAN STETKAR: Another one.

MS. POHIDA: That actually was during hot shutdown, and water moved, because you had driving head, water moved very, very, a lot of water moved very, very fast. So, anyway, because a valve is in locked, you know, locked manual closed, you can't presume that it's not going to be opened because those events do happen with some regularity. But, anyway, yes, so I need to ensure that the risk is small, given that internal CDF value is so small. Also, the other issue is the racks were initially referenced in the DCD, so I'm working with the Radiation Protection Branch who shares the same concerns that I do to make sure that these issues are sorted out.

Okay. May I turn to slide 38, please?

This is the auto isolation of letdown, the initiating of vortexing. And I wanted to draw your attention to the overdrain frequency. It's, you know, 4E minus 6 per year. And what this frequency is, it's the frequency that the operators are going to overdrain or basically overshoot the draindown to achieve mid-LOOP, and they're going to have a loss of RHR suction.

MEMBER BLEY: Now, do they have one of these designs where that tap comes in pretty close to mid-LOOP, such that you don't have to overshoot very far before you can lose suction?

MS. POHIDA: Are you talking about the RHR dropline?

MEMBER BLEY: Yes. Some of them come off low, some come off high, which is --

MS. POHIDA: Oh, okay. You're talking if it comes off the direct --

MEMBER BLEY: Off the reactor --

MS. POHIDA: -- as opposed to at an angle?

MEMBER BLEY: Yes, which is when you get the vortexing easily.

MS. POHIDA: You know, that was a subject of our vortexing audit, and I just can't remember.

CHAIRMAN STETKAR: They can probably tell us. They're here.

MR. SCHMIDT: This is Jeff Schmidt from the staff. It comes off at a 45-degree angle. It's not straight vertically below. It comes off at an angle, and the nominal is about 0.47 feet above the top of that nozzle. So you have a 45-degree angle off of it. Off the top of that pipe is about 0.47 feet above is the nominal value.

MEMBER BLEY: When you're at mid-LOOP?

MR. SCHMIDT: Yes.

MEMBER BLEY: Okay.

MR. SCHMIDT: Yes, these are all --

MEMBER BLEY: So you have over four feet?

MR. SCHMIDT: You have 0.4 --

MEMBER BLEY: Six inches before you start to uncover that.

MR. SCHMIDT: Right. That's the nominal value.

MEMBER BLEY: Yes, okay.

MS. POHIDA: Thank you, Jeff. Okay. So what we're doing, I was reviewing -- I'm sorry.

CHAIRMAN STETKAR: And where, at what level does the automatic isolation kick in?

MS. POHIDA: Oh, gees. I think --

CHAIRMAN STETKAR: In other words, if you're draining level really fast, will the valves go closed in enough time?

MS. POHIDA: Okay. I believe it's 0.47 inches above hotleg mid-pipe. But since MHI is here, I defer to them.

MR. KIPPER: Scott Kipper from MNES. That is correct, 0.47 feet above the center line of the main coolant pipe is where, that is where the interlock actuates. Correct.

MS. POHIDA: Okay, thank you.

CHAIRMAN STETKAR: I'm hearing two 0.47's here, which is sort of curious. So if I have a pipe and I'll call that the hotleg, and in the middle of that hotleg I draw a line, the RHR suction line comes off at some elevation below that mid point; is that correct? And you said that's 0.47 --

MR. SCHMIDT: No, what Scott said is correct. It's 0.47 feet above the midline of the hotleg.

CHAIRMAN STETKAR: Unfortunately, this doesn't come through on the transcript all that well. But if you can look at this cross-section of the hotleg, and if this is mid-plane on the hotleg, where along this quarter circle does the RHR suction line come from?

MR. SCHMIDT: Forty-five degrees.

CHAIRMAN STETKAR: Forty-five degree, for example, here?

MR. SCHMIDT: Yes.

CHAIRMAN STETKAR: And that's 0.47 feet below the midline?

MR. SCHMIDT: No.

CHAIRMAN STETKAR: Okay.

MR. KIPPER: The water level is 0.47 feet above the center line, so there is, in excess of 0.47 feet, there's the additional height between the center line and the nozzle.

CHAIRMAN STETKAR: And how far is that distance?

MEMBER BLEY: Below the center line.

CHAIRMAN STETKAR: Below the center line.

MEMBER BLEY: To the top of that nozzle.

MEMBER SHACK: It's 45 degrees. It's half of the . . .

CHAIRMAN STETKAR: Okay. All I'm trying to figure out is if the signal comes in at whatever it is, five or six inches above the mid-plane, it's going to take some time to close the valves. If it's a rapid draindown event, I don't know whether the valves get closed by the time the level gets down below this tap. If it's a really slow draindown event, you'd probably have enough time, or if the valves really close pretty quickly. That's why I'm trying to ask about these relative geometries because valves don't close instantaneously, typically.

MS. POHIDA: Oh, I agree. That was the subject of RAI 19-568.

CHAIRMAN STETKAR: Never mind.

MR. SCHMIDT: Yes. We also have an RAI in phase four.

CHAIRMAN STETKAR: Okay.

MEMBER BLEY: That will help because this is a really small number for things that happened quite often in the past.

MS. POHIDA: I agree. It's very low compared to operating plants, but you have to remember that that frequency includes two things. It includes the failure of auto isolation, and that was given as somewhere around E to minus 3-ish; and failure for the operator to manually stop the draining, and that was also given at E minus 3, and that's also a topic of more review.

MEMBER BLEY: And the first one is related to the fastest draindowns you might have, as well.

MS. POHIDA: And that set point of auto isolation and at what level that sits at at the hotleg. Okay, great. The next initiating event frequency that

I reviewed was the failure to maintain level, and that's where, basically, you're operating in mid-LOOP conditions and you, for some reason, you're unable to control level and you have a loss of RHR pump suction. So, anyway, those are very low compared to operating, very low compared to operating PWRs. One reason, of course, is the automatic set point, and that's why it doesn't appear in the dominant cut sets.

CHAIRMAN STETKAR: Yes. I mean, these operators will not be better, on average, than average operators. So the operator performance should be consistent with the current operating fleet or current operating fleet today, let's say, not necessarily 30 years ago. So, basically, what's saving them is the auto isolation feature.

MEMBER BLEY: And compared to when this used to happen a lot, the level indication systems --

CHAIRMAN STETKAR: Yes, that's --

MEMBER BLEY: They use ultrasonic level indication?

MS. POHIDA: That's RAI 19-568.

MEMBER BLEY: Thank you. We don't know yet.

MS. POHIDA: Well, it's the next slide. My presentation is rather brief. But, yes -- I beg your pardon?

MEMBER BLEY: I said we're extending that a bit so . . .

MS. POHIDA: Thank you. May I continue?

MEMBER BLEY: Please.

MS. POHIDA: Okay. This auto isolation function is risk significant. Of course, you know that overdrains during mid-LOOP are not appearing in the dominant sequences. I did some scrap paper calculations, and if you remove the auto isolation function it starts approaching, you know, the Commission goals, the removal of this automatic set point.

During my review of Chapter 19 and then I went back and looked at 547 of the DCD, and that covers RHR operation during mid-LOOP operation, I was concerned about where this auto isolation set point sat versus, you know, where does vortexing initiate in the hotleg for the highest anticipated operational flow rate because, with vortexing, it's driven by two factors. One is your hotleg level; and, of course, the lower it goes the more likely you're to ingest air in the pumps. And the second is your flow rate. If your RHR flow rate is rather high, that's also going to, that's going to also aid in ingesting air into those pumps.

CHAIRMAN STETKAR: When you say RHR flow

rate, I don't know how they control cooling during shutdown on this plant. A lot of plants, that flow rate is constant. I mean, it is the full flow rate. You just split flow, whether it's going through a heat exchanger or bypassing a heat exchanger. And I think two of the loops have the capability to do that or something like that. So, you know, that's not a variable. We ought to know what that flow rate is, unless, for some reason, they --

MEMBER BLEY: Unless it's different --

CHAIRMAN STETKAR: -- unless they do it differently.

MR. SCHMIDT: This is Jeff Schmidt from the NRC. There is a range in the DCD. The top number is 2650. I'm not sure I remember the bottom number --

CHAIRMAN STETKAR: So there is a range, though? Okay.

MR. SCHMIDT: Yes. And the DCD 5.4.7, there is a range.

CHAIRMAN STETKAR: Okay. So maybe they control it differently. Thanks.

MS. POHIDA: Okay. The other thing that's noteworthy, I guess, is that there's no indication of RHR pump motor amperage in the control room. And that would be one of your first indications that, you know, if you see oscillating pump motor amps, that would be one of your first indication that your pumps are ingesting air. So this --

CHAIRMAN STETKAR: When you say no, it means that the digital display systems they have don't have pump motor amps?

MS. POHIDA: As my reading of RAI 19-568 goes, and that's the next slide, that's what I understand.

MEMBER BLEY: That's very unusual.

CHAIRMAN STETKAR: That's really unusual.

MEMBER BLEY: I guess the other question I'd have, and maybe this is coming up next, how are they fixed for the ability to vent out all of the high spots in that system should they ingest air?

MS. POHIDA: Perhaps I defer to our Chapter 5 reviewer.

MEMBER BLEY: That's been really tough in some plants in the past.

MS. POHIDA: Oh, I understand. With the concept of LOOP seals, yes.

MR. SCHMIDT: Again, this is Jeff Schmidt from the NRC. I mean, what we're really trying to do is obviously minimize air ingestion --

MEMBER BLEY: Yes, but should it happen.

MR. SCHMIDT: They've committed to basically running pipe slopes, you know, so you have the inlet, say, at the bottom and the outlet at the top to try to minimize. There's an NEI guidance document. I can't remember the number off the top of my head. But if you look at the staff's safety evaluation for Chapter 5 in 547, there's a bunch of items in there which try to deal with gas accumulation and all the best practices. So, you know, they have --

MEMBER BLEY: And they've committed to --

MR. SCHMIDT: They've committed to the NEI guidance of, you know, basically having pipe slopes, vents at the high points, for example, in the as-built condition. There was also an information notice out fairly recently that they've also committed to, not committed to but, you know, they've recognized the importance of gas accumulation and tried to address that in an information notice.

MEMBER BLEY: Okay. Thanks.

MS. POHIDA: Thank you, Jeff. Okay. As you probably noticed, this issue is being resolved as part of the Chapter 5 review, the issue concerning, you know, pump operability during mid-LOOP. And once that becomes resolved, then I'll be able to go back and review these initiating event frequencies to make sure that they're consistent with the design.

Once again, this is a rather old RAI. It was discussed at the US-APWR audit in May 2011. We've had numerous phone calls on the issue, and the staff audited their vortexing calculations in October of 2012.

May I go to the next slide, please? Thank you. Okay. The hotleg level instrumentation. Failure of the operator to start RCS injection is risk significant. If you look at the risk values for POSes 4-3 and 8-1, the Fussell-Vesely values are 0.5. There's no automated RCS injection in this design, so when the RCS is open and steam generator cooling is not viable anymore, you know, manual injection is the sole mitigation path to prevent core damage, given that you have an extended loss of the RHR function.

If you review the PRA, the probability of starting RCS injection, and that includes failure to start from the operator, charging, and SI, is approximately 1E minus 4. So we asked an RAI, and it's 19-568, and it asked for a lot of details on the hotleg level instrumentation. And we got back the response, and the hotleg level indication, the sensors, I believe, are stage-related, but the indication is not.

Also, we have concerns about the validity of this level indication during boiling. And this is how the hotleg level instrumentation is tapped. You've got your bottom tap on the bottom of the crossover piping, and you've got your upper tap, it's connected to the pressurizer. And, of course --

MEMBER BLEY: So this is a DP?

MS. POHIDA: Yes, that's exactly it. So what you're doing is you're taking a DP measured between the void space in the pressurizer and the bottom of the crossover pipe. Well, if you read Generic Letter 88-17 and NUREG-1410, that was the IIT report on the loss of DHR at mid-LOOP at Vogtle. You know, if you're situated with an open RCS, you've got vents opened up in your pressurizer, the head is on, if you're at high decay heat, okay, if you're at high decay heat and you have a loss of RHR, when the RCS is going to be boiling, you're going to have surge line flooding effect where you're going to be, you know, with steam, it's going to be sweeping water into the pressurizer, and it's going to be entrained into the pressurizer. Well, the problem is is that the level indication is going to be looking at that back pressure of water in the pressurizer, and you could have indicated level much greater than actual level.

MEMBER BLEY: These are the kind of

instruments that have caused trouble in the past.

MS. POHIDA: Yes.

MEMBER BLEY: Now, some of them were rigged to tied-on tubing, which caused even more trouble. But to have had such good success with the ultrasonics, they are not using an ultrasonic level detector; is that right?

MS. POHIDA: Not based on the response of this RAI. Now, in other advanced PWRs, what they have is they have taps on the bottom of the hotleg and the top of the hotleg, so it kind of removes this pressurizer phenomena that's going on.

So this issue is taking a lot of work. We're working with the human performance people because they need to get involved in the man/machine interface aspects. And with the indication, we're also working with reactor systems in NRR, and, you know, we'll be developing supplemental RAIs on this.

And that concludes my presentation. If you have any questions . . .

MEMBER SCHULTZ: So just to clarify, you provided the RAI and there's a lot of pieces to that.

MS. POHIDA: Yes.

MEMBER SCHULTZ: And there's been a complete response to it.

MS. POHIDA: Yes.

MEMBER SCHULTZ: And there's a listing of different issues that are still outstanding, and additional RAIs are going to be provided.

MS. POHIDA: Yes, there's a lot of issues that we have to, there's still a lot of evaluation that needs to be done. And it's going to be, and it's to be coordinated with the different branches, with human factors, you know, and reactor systems. And then we'll be issuing supplemental RAIs.

Thank you for your time. I guess I turn it over to Todd.

MR. HILSMEIER: Do we want to start now or break for lunch?

CHAIRMAN STETKAR: No, we want to start now.

MR. HILSMEIER: Okay.

CHAIRMAN STETKAR: Some people have airplanes to catch. We're motivated today.

MR. HILSMEIER: I want to begin with the review of the US-APWR Level 2 severe accidents analysis. The evaluation was performed by Dr. Ed Fuller and support from his contractors, ERI. And it's been a four-year effort. And Ed Fuller recently joined Office of Research, and we miss him dearly. And, therefore, I'll be presenting the presentation on Severe Accident Evaluation in Level 2. And even though I'm not a Level 2 severe accident expert, I know enough to be dangerous.

CHAIRMAN STETKAR: Even though you're not an expert and didn't do the review.

MR. HILSMEIER: Right.

CHAIRMAN STETKAR: That's okay. Ed's here. We'll beat him up.

MR. HILSMEIER: So I'm going to thank Ed and the contractors for being here today to address any questions. I did stay at a Holiday Inn, though.

Regarding the outline, I will be discussing the staff's review of the applicant's Level 2 PRA Severe Accident Evaluation. And this presentation first begins with an overview of the applicant's Level 2 PRA and Severe Accident Evaluation. This is necessary to support the detailed discussion of the topics that we want to go into detail on, which is ex-vessel steam explosion, hydrogen generation and control, core debris coolability, and risk metrics.

Next slide, please. This slide provides a flow diagram for the Level 2 PRA. Basically, the output from the Level 1 PRA are the accident classes, and the accident classes are fed into the Level 2 PRA containment system event trees. And the containment system event trees model systems and functions that prevent containment failure and mitigate the consequences of severe accident. It's basically a bridge tree between the Level 2 containment phenomena event tree and the Level 2 PRA.

So the output from the containment system event trees are accident sequences that are grouped in two plant damage states. Each plant damage states contain details about core damage status and the availability of mitigation features.

CHAIRMAN STETKAR: And, Todd, before you flip that slide, I want to make sure that I -- I asked the applicant yesterday, but it's my understanding that the containment systems event trees are linked directly to the sequences from the Level 1 PRA model. I mean, in RiskSpectrum parlance, they're consequence trees.

MR. HILSMEIER: Correct.

CHAIRMAN STETKAR: So, in effect, that intermediate accident class list doesn't really exist except to define the structure of perhaps different branching boundary conditions in the containment systems event tree.

MR. HILSMEIER: I guess.

CHAIRMAN STETKAR: Okay, okay.

MR. HILSMEIER: And there's some systems

that are not modeled in the containment system event tree, like core spray system and --

CHAIRMAN STETKAR: Alternate containment cooling.

MR. HILSMEIER: Yes.

CHAIRMAN STETKAR: Because they're in the front end. They're in the white box on this slide.

MR. HILSMEIER: Exactly. Next slide, please. And then the core damage frequency, the core damage frequencies from the plant damage state are fed into the Level 2 containment phenomena event trees. And the containment phenomena event trees model the physical phenomena in the containment that influences containment failure, such as ex-vessel steam explosion, hydrogen combustion, in-vessel steam explosion. And we'll be discussing that in a few slides.

And this slide presents the top events for the containment system event tree for station blackout, which is the top events. And then the top events in the bottom figure is for all the other accident sequences. As you can see, it models the systems and functions that help mitigate severe accidents.

MEMBER BLEY: The switches, the models in the containment event tree depend on conditions in the Level 1 event tree. Do they somehow reset those -- are they sequence, or do they have a class of Level 1 sequences that get the same treatment in the containment event tree, something like the old plant damage states? How do they handle that?

MR. HILSMEIER: Because containment system event trees use the same software. I believe it's RiskSpectrum.

MEMBER BLEY: Yes.

MR. HILSMEIER: It's all linked together, so if there's a --

MEMBER BLEY: So they have some kind of that pick up characteristics of the --

MR. HILSMEIER: Correct. So it's like if an accident sequence has a loss of AC power, it would be reflected in the containment system event trees.

MEMBER BLEY: Okay. So they, they've, and they condition them maybe on electric power presence or not. Okay.

CHAIRMAN STETKAR: If it's fully linked, they don't even need to condition it. I mean, it's, you know, the containment isolation valve will have an electric power fault tree, and when you solve that the same basic event will kill everything. The only thing that I've seen in these models is that oftentimes there are, and I always get the jargon wrong, so I'll just call them boundary conditions, there are boundary conditions set on specific sequences that will toggle perhaps success criteria or timing. In other words, they'll toggle in -- the success criteria, for example, in one sequence might be two out of two and in another sequence it might be one out of two, for example. And those are set by boundary conditions. I mean, that's not solution to the fault tree. It's which fault tree you actually toggle in.

And I don't know. I didn't study the containment systems event tree enough or think about all of those transitions to know if those types of modeling techniques are used here. That has been a source of problem in the past, and I think that's more of what Dennis was talking about, not just linking the thing and making sure that failure of this circuit breaker in one part of the model is failure to the same circuit breaker elsewhere. It's toggling those boundary conditions.

MR. HILSMEIER: Right, exactly.

CHAIRMAN STETKAR: Do you know, did they use that technique or --

MR. HILSMEIER: I can't speak for this. I can only speak for my past experiences at other plants. The toggling of the flags, so to speak --

CHAIRMAN STETKAR: Yes.

MR. HILSMEIER: -- can be very --

CHAIRMAN STETKAR: It's a source of error.

MR. HILSMEIER: Right.

MEMBER BLEY: Necessary but a source of error.

CHAIRMAN STETKAR: Does anyone, does anyone from MHI, is there anyone here who's -- this is not a phenomenological or systems modeling. It's an actual quantification. You know, the person who runs the model would know this, and I don't know if you have anyone here with that --

MR. HILSMEIER: We can take that as an ACRS action item to address --

CHAIRMAN STETKAR: It's a question. As I said, if they use flags, or whatever the appropriate terminology is, to toggle in different, essentially, parts of the model, you either negate a part of a general model or toggle in, however they do it. Certainly, examination of those flags and making sure that they're set appropriately has been an identifiable source of error. And many times it's difficult, it's easy to find the error if something strange boils up to the surface. You say, oh, my God, I got that flag wrong, I need to correct that. It's really hard to find if it somehow artificially suppresses something.

MR. HILSMEIER: Right. Ed, do you know if the flags were looked at?

MR. FULLER: No, I do not.

MR. HILSMEIER: Okay.

CHAIRMAN STETKAR: Okay.

MR. GODA: Excuse me. Hiroshi Goda from MHI. You're talking about something, a dependency between containment system and those --

CHAIRMAN STETKAR: Not dependency in the sense of electric power or cooling water or signal or any of those things. What I'm talking about is -- and the problem is this is speculation because I didn't look at the model close enough, and I'm not even sure the information is in there. In some models, when you link an event tree together, this could even be in the same event tree, under some scenarios a success criterion may require two of two, and in a different sequence the success criterion might require one of two. And you'll have either two fault trees or a general logic for a fault tree with house events, for example.

MR. GODA: We have bunch of fault trees, depending on that -- we have two between middle one. We call that the accident classes. And in US-APWR, we totally 28 accident classes that we developed 28 CSETs

--

CHAIRMAN STETKAR: Now, is there a one -- for each CSET, is there a uniquely-defined set of fault trees for that CSET?

MR. GODA: That's right, yes.

CHAIRMAN STETKAR: So there isn't any toggling within the CSET.

MR. GODA: Yes.

CHAIRMAN STETKAR: So it's simply then the linking of those 28 CSETs to the correct sequence from the Level 1 tree?

MR. GODA: That's right.

CHAIRMAN STETKAR: Okay, okay. So it's that linking process then, rather than the toggling. What I was thinking about, you know, in RiskSpectrum you can set a boundary condition that says, you know, I used boundary condition one on sequence A, and I used boundary condition two on sequence B. You didn't use that, from my understanding. It's more the what is the correct CSET tree linked to sequence number three compared to sequence number, let's say 18.

MR. GODA: That's right, yes. That's right.

CHAIRMAN STETKAR: Did anyone in the staff look at that process? Because that's what I was getting to that I didn't think much about the, whatever you call them, ACLs because I read that the thing was wired together. So I didn't pay much attention to what is the definition of ACL, you know, whatever because it really didn't mean anything. But making sure that the right one of that set of 28 is linked to sequence number three is something that I would have hoped the staff would have looked at.

MR. KARIMI: John? I'm sorry. Roy Karimi from ERI. Actually, I looked at the fault trees that affected the top events. The fault trees are static fault trees. There's no attributes in there, except for when we have SBO issues. Loss of offsite power recovery is available or not, but it changes the CCW operability. That's the only one I saw --

CHAIRMAN STETKAR: But that's ubiquitous throughout the whole model. If you look at the Level 1 model, it's got the same type of thing in it. I'm aware of that. What I'm literally talking about, though, is MHI just said that they developed 28 of these CSETs.

MR. KARIMI: No, they have the same set of the CSET applying to different ACLs. For AEV, they're using the bottom CSET. For the SBO, SEV prime, which is the SBO, they're using the top CSET. The only thing that's different is they have this RSB over there. The RSB is the CCW recovery for containment cooling, whereas in a case when there's no SBO, that is not required. That's the only distinction between the various ACA, they're called ACL coming into the CSV. When you look at the fault trees in PRA 6A, you will see there's nothing there that says that this is only applicable to AED or AEC, you know, specific ACL. But because, as you say, because of the common faults that they're being modeled in a CSET and those in the Level 1, then when you use the same model combining the CSET with the ACL in RiskSpectrum, those common elements will not be recounted again. And then you have results that mostly apply to the PDSes.

Now, another issue that comes out from this review was that when you look at this result, you saw that the sum of the PDS frequencies were more than the sum of the CDF frequencies. So they had to do some adjustment for making sure the results are going through the --

CHAIRMAN STETKAR: But that's just RiskSpectrum because it doesn't take the compliment --

MR. KARIMI: Exactly.

CHAIRMAN STETKAR: It looks like an event tree, but it really isn't. That's the rare event fault tree approximation, especially with some of these big -- you know, it's bizarre, but it's well known. I was more concerned, I thought I was hearing that there were 28 separate CSETs, and some human being decided that CSET number one is attached to sequence number seven in the steam line break outside containment and it's attached to sequence number, you know, 36 in the station blackout, and some human being made that decision.

MEMBER BLEY: That's what I thought I heard.

CHAIRMAN STETKAR: But that's different from what I'm hearing Roy say. I'm hearing Roy say that there are two CSETs, one that has an RSB and one that doesn't.

MEMBER BLEY: And that the fault trees are identical.

CHAIRMAN STETKAR: And that the fault trees are identical. So it's curious, if that's the case, why do I have a large number of ACL designators in the Level 1-2 model. They mean different things.

MR. HILSMEIER: I understand that. Each accident class, and correct me if I'm wrong, Roy, goes, let's say non-station blackout, goes through its own CSET tree.

MR. KARIMI: I know, but the CSET, when you say its own, it's because there are component failures in there, not the specific ACL, which, if it's failed over there, it's already failed in here. That's, essentially, that's why it become individual set. You may have one or two ACL going through the same fault tree because they have a common failure.

CHAIRMAN STETKAR: Let me see if I can, because I'm certainly confused. People who don't speak RiskSpectrum don't know what we're talking about, so that's fine. If I look at -- and we're okay on time, so I'm going to belabor this a bit. If I look at the large LOCA event tree model for Level 1 PRA, I see, for example, consequence states called ALC. I see AEI. I see AEIHS. I see AES and so forth. There are a number of these. There are more than two. That's my whole point.

Now, my question, first basic fundamental question is is there a different containment systems event tree branching logic structure, event tree logic structure assigned for ALC and AEI? I'll just take two.

MEMBER BLEY: You can answer in principle.

CHAIRMAN STETKAR: In principle. Anybody.

MR. KARIMI: When you look at the event tree, yes, because of the --

CHAIRMAN STETKAR: Okay.

MR. KARIMI: What you have to do here --

CHAIRMAN STETKAR: Thank you. That's all. Now, I understand that there are 28 of these logic structures. I don't care about the fault trees. The fault trees will take care of themselves. Okay. Now, my question now from a review perspective is did the staff's review do what I started to talk about here? Did the staff's review confirm that the logic structure for ALC was correctly linked to sequences in the large LOCA model, for example, number three, number six, and number eight, and that it was not inappropriately linked, for example, to sequence nine? In other words, who checked to make sure that the thing was wired together correctly? Did you do that?

MR. HILSMEIER: Me, personally, no.

MR. KARIMI: We did not really. What it is actually, if you look at what they have provided and what they did provide, as MHI said, they have 28 different ACL and there are 28 different of the CSETs that they become PDSes. We only look at the results. We did not go to --

CHAIRMAN STETKAR: Right. And that's, that's --

MR. KARIMI: -- we did not look at

RiskSpectrum. We did not look at any of the calculation --

CHAIRMAN STETKAR: Okay. That's --

MR. KARIMI: -- results.

CHAIRMAN STETKAR: That's the answer. I have the answer that I was asking for. I'm not necessarily happy with it. I now understand the mechanics of how the model was wired together.

MEMBER BLEY: And you only had the paper, you didn't have the model to play with.

CHAIRMAN STETKAR: Yes. And you can't -- well, RiskSpectrum, you can't tell how it's wired together. And my experience, as I mentioned earlier, just looking at the results, I will tell you that the people running this model, if they found something wired incorrectly because something was coming out to be, you know, ridiculously high, they fixed those. They may not have fixed the ones that were artificially suppressing numbers because you had the wrong event logic attached to a sequence because that requires you to look at things, in many cases, that you can't see because of the truncation frequencies or that are so low in a list of cut sets that, you know, you really get bored silly and say, you know, why isn't this thing higher than I would have expected it? So that process, that actual linking process, whether it's done this way in kind of a hardwired sense or whether it's done the other way with flags and setting toggles, has been a problem in the past. And you really can't check that simply by looking at the paper trail. I mean, you can see the paper trail that says this tree ought to be used for ALC, and this tree ought to be used for AEW, but you actually can't confirm and you didn't look at that in any of your, from what I'm hearing, any of the audits because the only way you could do it is in an audit. You have to actually look at how the model is wired together.

Todd, you need to now tell me, because you know the presentation, when it's a good place to break for lunch.

MR. HILSMEIER: When we complete the overview.

CHAIRMAN STETKAR: Okay. That's fine.

MR. HILSMEIER: Regarding the containment system, the CSET, again, the containment spray system alternate containment cooling is not reflected in there because it's reflected in Level 1 PRA. Also, no credit is taken for in-vessel retention, core debris by external reactor vessel cooling or water injection. It's assumed that reactor vessel melt-through occurs.

Also, equipment survivability is not in the top of the event because it's confirmed separately. We confirmed that the equipment that's used to mitigate severe accidents can survive the harsh environment of a severe accident.

Next slide, please. This slide prevents the plant damage states, and there are about 72 plant damage states, and how they were categorized. The plant damage states are categorized by reactor coolant system pressure at the time of reactor vessel melt-through and the reactor cavity flooding status at the time of reactor vessel melt-through and the condition of the containment isolation before core damage and the condition of igniters, containment spray systems, and containment cooling.

Reactor pressure is important because it determines if high pressure melt injection occurs or temperature-induced steam generator tube rupture occurs. Again, reactor vessel flooding status is important because it impacts ex-vessel steam explosion. Each plant damage state contains a unique set of parameters which influences the likelihood the magnitude of the phenomena in the containment phenomena event tree, which is discussed next.

Next slide. And this slide provides the

containment phenomena event tree used for each plant damage state. So the entry point into the containment phenomena event tree is core damage frequency from the plant damage state. And there's, basically, two end states for the containment phenomena event tree, and one is the intact containment which is called release category six and then large release. Those are release categories one through five. And the large release includes all sequences that involve containment failure, and it's independent of the time of the failure.

Next slide, please. The severe accident mitigation features associated with each top event in the containment phenomena event tree is provided on the next three slides. And I think I have time to go quickly through the three slides.

The first entry in the table is to minimize a potential of temperature-induced hotleg rupture and temperature-induced steam generator tube rupture after core damage. The mitigation feature is to reduce reactor coolant system pressure through the depressurization valves. And there's the safety depressurization valves, and also there's severe accident-dedicated depressurization valves.

The next entry is to minimize the potential reactor vessel failure, which is another top event in the containment phenomena event tree. The reactor vessel failure after core damage, to minimize the potential reactor vessel failure after core damage, water can be injected into the reactor vessel per severe accident procedures or reactor vessel can be externally cooled by the reactor cavity water. And, again, in-vessel retention is not credited in the Level 2 PRA model due to uncertainty about its effectiveness.

MEMBER REMPE: Okay. So yesterday I was asking about the insulation presence, which isn't modeled or considered. And there will be some, if you start flooding up the vessel, there will be some steam generated. And if you had considered the in-vessel retention, people in the past have worried about the structural integrity of it. And so let's just assume a worst case, and so this stuff just starts falling off and it goes down in the cavity and you've got a lot of junk in the cavity. Is that a problem? There's no sump down there or pump, right? So maybe it isn't a problem, but did you guys look at the pressurization analysis that MHI produced to make sure that you felt comfortable that it was okay?

MR. HILSMEIER: I'll need to defer ERI or Ed to answer that question.

MR. KHATIB-RABHAR: Mohsen Khatib-Rabhar from ERI. We did not look into the retention of the melt debris in the lower head, either from the standpoint of ex-vessel flooding, because that was not credited. And we did not look into the pressurization issue. I think what you're referring to is issues such as what were considered for the other plants in the past where, even though you may not be able to retain the melt inside the reactor vessel by external cooling, you may generate steam because of the heat transfer, which you may not be possible to vent it into the upper region of the containment. Is that your concern, or your concern is related to whether the specifics -- go ahead.

MEMBER REMPE: Unintended consequences of the fact that you've flooded out the picture, and maybe it's just a schematic. It shows water going up to, you know, to surround the lower head.

MR. KHATIB-RABHAR: Right.

MEMBER REMPE: So, really, it doesn't surround the lower head, it surrounds some insulation and things like that.

MR. KHATIB-RABHAR: Sure. Design of insulation --

MEMBER REMPE: Okay. Let's assume that that insulation, because you didn't consider it, just falls off because it's not able to withstand the chugging from steam.

MR. KHATIB-RABHAR: Right.

MEMBER REMPE: And what about unintended consequences because of this that you've got a bunch of junk in your cavity and the pressurization from, you know --

MR. KHATIB-RABHAR: What is the concern in terms of unintended consequences? I didn't understand the question, to be honest with you. I don't know what you're driving at.

MEMBER REMPE: We worry about GSI 191 and debris, but there's no sump down there --

MR. KHATIB-RABHAR: Exactly. There's no sump --

MEMBER REMPE: But has anyone just stopped and said is this a problem or --

MR. KHATIB-RABHAR: No, because this can't be any worse than core debris going into the cavity.

MEMBER REMPE: Okay. I just am wondering.

MR. KHATIB-RABHAR: Yes, so it's not really an issue.

MEMBER REMPE: Okay.

MEMBER SHACK: But you've verified there is a path to relieve any pressurization.

MR. KHATIB-RABHAR: The general path for pressurization are two things. One is the cavity which goes out and the area around the pressure vessel. Now, if you're getting significant heat transfer, potentially you can cool the lower head. If you're going to a dry-out, you're not going to get a lot of heat transfer, so I don't think that's going to be much of an issue. And I think there's plenty of path. Maybe MHI can address the areas. I don't remember the numbers, but I doubt if there will be an issue in terms of pressurizing the cavity.

Furthermore, even if you pressurize the cavity, so what? It's under containment. So I don't believe that's a severe accident issue to begin with. It's not worse than steam explosions, in other words. Let's put it that way.

MR. HILSMEIER: Any further questions?

MEMBER REMPE: That's good.

MR. HILSMEIER: Regarding in-vessel steam explosion, it's considered negligible based on NUREG-1524. And, therefore, no mitigation features are provided to address the in-vessel steam explosion. However, it is considered in the Level 2 PRA. And regarding ex-vessel steam explosion, we will be discussing this in detail later, so we'll talk about that in more detail a few slides from now. Also, hydrogen combustion and control and core debris cooling will also be discussed in detail later in this presentation.

To minimize the potential applied pressure melt injection which could lead to direct containment heating or rocket-mode reactor vessel failure, the reactor coolant system pressure is reduced through depressurization valves. Also, there's a debris trap in the reactor cavity, as well as no direct pathway to the upper containment which would reduce the likelihood of direct --

MEMBER BLEY: Did you do any analysis or anything to convince yourself that the debris trap, how effective it could be under different blow down modes here? This was released yesterday a little bit, and I think what I heard from MHI was it looked like it would keep the stuff in there to them and not much more than that, from what I heard.

MEMBER REMPE: But they said there was no testing done, and then I just was wondering if you'd ever get some pile-up occurring. But I don't think I saw either in the MELCOR or the MAAP analyses anybody trying to model that.

MR. FULLER: Is this on?

CHAIRMAN STETKAR: Yes.

MR. FULLER: This is Ed Fuller. MHI did not use MELTSPREAD, but they used another code, FLOW something, FLOW-3D. And the MAAP 4.06 code was used to calculate the accident progression, and when the vessel fails in the MAAP approach you have very high temperature core debris that is essentially liquid that flows quickly, and MHI calculated that you spread that debris out very uniformly. And then if there was water in there beforehand, you had, you know, FCI calculation if water came in later. They depended on that water to keep the debris cool and to avert base-spent melt-through.

So in terms of some of this kind of debris accumulating at this ledge or whatever it is, I never could quite figure out what it was during the review, but they called it a trap. It seemed to me, in the context of the model that was being used, you were just looking at sloshing waves of molten material that settled back into a uniform sea of it, so to speak.

MEMBER SHACK: Well, again, that sounds like what happens when you depressurize and you get the failure. Did they actually do calculations where they assumed that the depressurization didn't succeed and they had a high pressure injection and what those flow paths look like?

MEMBER BLEY: Well, that's what they're talking about here. Rocket-mode? That is where you're getting a high-pressure injection, right?

MEMBER SHACK: No, this is more like a direct -- yes, I'm thinking more the direct containment heating where I'm throwing the stuff up, yes.

MEMBER BLEY: Which comes from that.

MEMBER SHACK: Yes.

MEMBER BLEY: Well, they're saying there's no direct path. I looked at the picture. It's kind of hard without a 3D model to really --

MEMBER SHACK: I mean, the picture sort of says there's no direct path, but it's only a cartoon.

MEMBER BLEY: A cartoon, yes.

MR. HILSMEIER: I was looking at the applicant's analysis. First, if the reactor coolant system pressure is about 250 psi or greater, then you get a high-pressure melt injection. The containment peak pressure is calculated for postulated direct containment heating phenomena, assuming debris dispersal of five percent, the peak pressure is about 100 psi. And the containment ultimate capacity is like 216.

MEMBER SHACK: But they just assumed that debris dispersal in.

MR. HILSMEIER: Of five percent. I'm not sure where they get the five percent, although the staff did agree with the analysis.

MEMBER BLEY: If we don't know why we assume that, how much difference would it make if it's off a bit? I mean, the more stuff you get up there, the more direct containment heating you get. And I haven't heard anything yesterday or today that tells me any engineering basis for that assumption. I did hear somebody yesterday say, well, it looked to us like you couldn't get much up there.

MEMBER REMPE: They said their experts also thought it would be better with it, versus without it, but I don't know what basis of the MHI experts was to come up with that conclusion to --

MR. KHATIB-RABHAR: Again, Mohsen Khatib-Rabhar. Let me talk about this a little bit. Direct containment heating. This issue was resolved by the NRC for operating plants. If you look at containment failure pressure for this plant compared to the plants for which NRC analyzed, and, in fact, some of them are a lot more dispersive than this particular cavity that they have here. The condition of containment failure probability was very small.

So even if you used those sets of conservative analyses, which were done for Zion and other U.S. plants with much lower containment fragility, you're not going to get a very high likelihood of containment failure. Now, look at the containment fragility for this plant compared to the others. If you believe the analysis, this one should be greater. So even if you assume dispersal levels as the same as the others, this is not going to be a major issue. I think it can be resolved from that point of view. It's very simple.

MEMBER REMPE: It's not one of my hotter items, I guess, of concern, but it just seems like an RAI asking about the debris ledge and the basis for its inclusion and possible effects and why they're not important might be warranted. But I'm not going to get uptight about this one. There's other things I would rather be more concerned about.

MR. HILSMEIER: One thing, the RAIs, that's already been issued. I'll see if it's been addressed, to some degree.

MEMBER BLEY: That would be helpful. I mean, the argument that was just made might be a good one, but it's not made in this analysis. This analysis has --

MEMBER REMPE: It just says it's good, man. That's why I'm wondering.

MR. KHATIB-RABHAR: From a review point of view --

CHAIRMAN STETKAR: Mohsen, come up to the microphone.

MR. KHATIB-RABHAR: I speak loud enough --

CHAIRMAN STETKAR: No, that's okay. That isn't -- there's a transcript so you need to --

MR. KHATIB-RABHAR: From a review point of view, that's how you convince yourself. If they're not presented, you feel comfortable enough not to ask the question.

MEMBER REMPE: There's one other question I had. Again, perhaps I needed to be more educated, but I ran out of time reading. But, apparently, there's instrumentation that was coded as being used to keep the flooding level in the cavity at a certain value because of hydrogen generation. Could you kind of explain the way that's occurring and what's being done there? It's discussed in the SER, also. And I'd just better like to understand that process. I think that they decided instrumentation wasn't needed, you could rely on evaluating how much water went in and control the water level, but is it really that important?

MR. HILSMEIER: This is for the hydrogen control in the RWSP.

MEMBER REMPE: Right.

MR. HILSMEIER: From my understanding, and then Mohsen can describe it a lot more than I can, but that artifact of the water level is just in order to generate hydrogen concentrations in the RWSP.

CHAIRMAN STETKAR: We've got more slides, actually, this afternoon on hydrogen. So --

MEMBER REMPE: Okay. Just don't let me forget because I just was curious about that when I was --

CHAIRMAN STETKAR: It's not my job to make you not forget. Let him know. What I'm trying to do is let Todd get through his --

MR. KHATIB-RABHAR: John, just a short response, I think I will address that. Just by knowing the flow rate of higher water, they know what the level is. That's why they indicate --

MEMBER REMPE: How important is it, though? I just was curious about that. I don't know of other plants that are doing that.

MR. KHATIB-RABHAR: We can address that later when we get there.

CHAIRMAN STETKAR: Let's see if we can get through the overview, and let Todd finish his sort of introduction.

MR. HILSMEIER: Next slide. I think there's just one more slide.

CHAIRMAN STETKAR: Yes, that's what I was trying to get.

MR. HILSMEIER: In early and late containment failure through overpressure is mitigated by large and high-strength containment. Also, active cooling, containment cooling is provided through the containment spray system and alternate containment cooling through containment fan-cool units. Also, firewater system can be aligned to the containment spray system.

CHAIRMAN STETKAR: Todd, I didn't -- here's what I'll ask. I didn't look far enough ahead. Are you going to talk more about the alternate containment cooling in the context of post-core melt conditions in the containment?

MR. HILSMEIER: Not really.

CHAIRMAN STETKAR: You're not. Okay. Because, you know, I asked about it yesterday in terms of what analyses were done under, let's call it a clean containment environment. You just heat in the -- and I'm not sure how different it might be in a not so clean containment environment or even whether the models, I'm sure the models do take credit for it, you know.

MR. HILSMEIER: You're referring to natural re-circulation from the containment fan-cooling --

CHAIRMAN STETKAR: I don't care about getting the water in and out of the coils. I care about heat transfer into the coils from things that might be fouling the coils, for example.

MR. HILSMEIER: Right.

CHAIRMAN STETKAR: Stuff that might be coming out, you know, and fouling those coils.

MR. HILSMEIER: Yes, I won't be addressing that later. Maybe we should discuss it now.

CHAIRMAN STETKAR: No. I think we all want to go to lunch now.

MR. HILSMEIER: Okay.

CHAIRMAN STETKAR: I will remember that one for this afternoon.

MR. HILSMEIER: Okay. That's fine. It will give us time to --

CHAIRMAN STETKAR: Think about it a little bit, yes.

MEMBER REMPE: I have another question on this slide, too, but you can think about it and talk about it later. But it's the equipment survivability assessment, and, apparently, that's stemming from a couple of SECYs that are for the advanced light water reactor designs. And in it, what, of course, piqued my interest was the instrumentation survivability. They identified particular sensors that needed to withstand the severe accident conditions to help the operators diagnose what was going on in the plant and mitigate it, and, of course, that's of interest after Fukushima. And I was just wondering how rigorous an assessment was done, how many scenarios were considered, and then what parameters and conditions were deemed necessary and what sensors? Because after TMI the radiation in the containment died. The thermocouples didn't, they didn't obtain readings with sufficient short enough times, and so how did the staff decide what should and shouldn't be included in that list of equipment that survives?

MR. HILSMEIER: Okay. We'll discuss that later, right?

MEMBER REMPE: Okay. I'd like to because I didn't see it in any of their slides, so I picked that slide to pick on you for it. Thanks.

MR. HILSMEIER: Thank you.

CHAIRMAN STETKAR: Anything else for the

staff?

(Whereupon, the foregoing matter went off the record at 12:16 p.m. and went back on the record at 1:01 p.m.)

CHAIRMAN STETKAR: Let's come back in session and continue with the staff's presentation.

MS. REYES: I'm sorry for the interruption. I wanted to ask you if you want, we've got two actions items from this morning. Do you want the staff to discuss those action items now or after --

CHAIRMAN STETKAR: Sure. No, let's do it now because, otherwise, we'll forget what they were.

MR. PHAN: Thank you. Well, this is the shortest lunch of my life. I have two action items, the first one on internal fires. I went back and checked the document on turbine building fires, and they have multiple fire-induced initiating event for those fires compartments, including, like, transient, loss of feedwaters, and the third one they call SLBOs, and that stand for the -- let me read the language -- steam line break downstream of MSIB turbine size, so which mean that outside containments and that the outside. Is that answer your questions?

CHAIRMAN STETKAR: No. Here's my question, and I looked at some of the cut sets at lunch.

That model, and I have to be careful about words, the impact of that fire is at least one turbine bypass valve is stuck open. That's the way it's modeled. I don't know how it's quantified because I can't find any numbers, but it's modeled as at least one turbine bypass valve is stuck open.

MR. PHAN: Yes.

CHAIRMAN STETKAR: Due to the fire. I don't know how due to the fire. I don't know whether it's presumed to be a hot short, a spurious actuation. I don't know. I don't care at the moment. I don't care how they quantified it. My question is that if a fire -- and I looked at the cut sets, and the cut sets do include failures of the main steam isolation valves to close. So the fire does not disable the main steam isolation valves. The fire, it doesn't make any difference whether it disables main feedwater because main feedwater is not included in that model and a safety injection signal which closes the main steam isolation valves also trips and isolates main feedwater. So main feedwater impacts from the fire are not an issue.

This simply seems to be a transient event with one or more open turbine bypass valves that then goes through what looks exactly like the steam line break outside containment event tree, I checked the event tree structures, and results in 53 percent I think is what you said of the fire core damage frequency. My question is in the Level 1 internal event PRA, MHI does not model operation of any steam relief valves. And when I say any, I mean not the turbine bypass valves, not the main steam relief valves except for rapid cooldown scenarios. They model the main steam depressurization valves, but those are initiated manually, and they don't model the main steam safety valves. There is no model for steam relief. There is no model for steam relief. Because there's no model for steam relief, there is zero probability, precisely zero, that any steam relief path can stick open.

MR. PHAN: When you say not, you mean not modeled as the initiator or --

CHAIRMAN STETKAR: I mean if I look at a fault tree, there is no basic event that says this valve fails to open, and there is no basic event that says if it's open it fails to re-close. It is not modeled. It is not in the model. It is ignored. It is presumed that that function is 100-percent absolutely guaranteed to always be successful that you get enough steam relief. And it's presumed that if things open they're 100-percent successful closed. That's in the Level 1 model.

Now, the question is if spurious opening of a turbine bypass valve, because of a fire, is potentially so important to core damage and risk from fire events, why isn't it also, why couldn't it also be potentially important to risk from a normal transient, any plain vanilla transient? Turbine trip, reactor trip, loss of offsite power, maybe not loss of offsite power because you need power to open these valves but loss of main feedwater flow, any of those other events for which they're not modeled.

And the reason I was asking about the fire in particular was I didn't know if there were any other fire-induced failures that would disable functions to protect against that stuck-open valve. And I can't find any, at least in what I looked at, so that was basically what I was asking you. And when I looked at the cut sets, I didn't see any functions. They said the fires could affect main feedwater, but that's irrelevant. That function is not relevant to that model. The main steam isolation valves can work because their failures show up in the cut sets. The event tree looks exactly the same as the other event tree. It's got high-pressure injection, the feed and bleed cooling, and, you know, emergency feedwater and all that. So it was curious to me why that fire-induced effect was so important to the fire risk and, yet, it's invisible, not numerically invisible, it's assumed to be guaranteed success for everything else in the PRA. Not modeled. Not --

MR. PHAN: They have one big assumption that for any component inside containment would not be impacted by the fire, only for those that outside containment.

CHAIRMAN STETKAR: And that's fine. It's not relevant because this is a turbine building fire.

MR. PHAN: Yes, yes.

CHAIRMAN STETKAR: You know, the fire scenario is a fire in the turbine building --

MR. PHAN: So for those inside containments, they say they would be --

CHAIRMAN STETKAR: That's fine, yes. And it's fine because, as best as I can tell, once you have this fire-induced failure, the rest of the model seems to be exactly the same as the steam line break outside containment initiating event for the Level 1 internal event PRA. The event sequence model looks exactly the same, the functions that they look at. And the best as I can tell, the numerical values are the same, although I didn't check all of that.

So I'm curious, unless you know about some other fire failure as a result of a fire that would affect any of the mitigation functions from that stuck-open turbine bypass valve, and I couldn't find any, but I couldn't find any description and I only had a few minutes to look at it. I'm really curious about that. First of all, I'm really curious about why it's so important for the fires, and if it's legitimately that important for the fires I'm curious why it's not been examined for anything else in the PRA.

MR. PHAN: One point I'd like to mention here that this particular fire area is identified as important because the fire sequences that were E minus 2. That's mostly higher than the other compartment, and the condition of frequency is E minus 6. So for those two combination, E minus 8, E minus 8 that was making the list at the high importance.

CHAIRMAN STETKAR: But do we know the frequency of all transient events for which the turbine bypass valves should normally open and the conditional probability that one or more would stick open after that? Total frequency of transients is about once a year on this plant. That's a lot. That's about a factor of 70 times higher than I think the turbine building fire frequency that they used.

MR. PHAN: Yes.

CHAIRMAN STETKAR: So we've got a factor of 70 there to deal with.

MR. PHAN: Yes.

CHAIRMAN STETKAR: And I don't know whether they assumed that the turbine bypass valve stuck open or whether it was actually quantified. There's a fault tree that says, you know, a big OR gate with a bunch of basic events, but I can't find a value for any of the basic events. It says, you know, turbine bypass number one sticks open or turbine bypass number two or number three or number four, but I can't find any values for those so I don't know whether they were assumed to be one or whether they were quantified with some numerical value because I couldn't find those. Anyway, I think that's sort of a little more --

MR. PHAN: We would reading more and try to find more related information.

CHAIRMAN STETKAR: I mean, from the fire perspective, I'm mostly interested to see how they, what numerical values or assumptions were made regarding the conditional probability that a turbine bypass valve sticks open. Was it just assumed that it would because of some spurious actuation signal, or was some probability assigned to the fact that, given this event, one valve would stick open? And I couldn't find that.

I couldn't find anything from the fire perspective where the fire disabled any of the other mitigation systems in that model, but I might have missed something. I mean, if the fire also disabled some of those mitigation systems or partially, that would increase the conditional core damage probability from that model compared to a normal transient.

So those are sort of my concerns. And if there isn't any of that fire-induced impact, then I'm really curious about why it was modeled for fires but not modeled for anything else.

MR. PHAN: The next action items on the internal flooding within 15 minutes. That is an assumption in the internal flooding regarding the mitigation of the waters within 15 minutes. In there, they mention that the number is so low, but they did not provide any values associated --

MEMBER BLEY: Yes, I saw the assumption, but I didn't see any basis or any analysis based on --

CHAIRMAN STETKAR: Does that mean the number is lower than 10 to the minus 13 per year?

MR. PHAN: Pardon me?

CHAIRMAN STETKAR: Does that mean that the number is lower than 10 to the minus 13 per year?

MR. PHAN: I hope it's higher, but I have no clue. That's why we issue an RAI asking for them what is your number. So in RAI 53-956, questions 19-101, we asked them three questions. The first one is what is the failure probabilities of the detection device for flooding because they say the number is low because the failure probabilities of the protection is so low. So we asked them for that.

The second question asking them, given you isolate the rupture, what are the consequences that in number two? The number three is not related. So in their responses, first they say they're using the IEEE standard 500, and they give us the numbers of 1.4E minus 6 per hours.

CHAIRMAN STETKAR: Per hour.

MR. PHAN: Per hours.

CHAIRMAN STETKAR: Now, the last I checked, if I fill up a component area with water, that per hour failure rate, if I use a standby failure rate model, is the time between the times that I fill up that compartment. So if they never fill up that compartment, even if it's a low per hour failure rate, once I accumulate something on the order of 500,000 hours, it's guaranteed to be failed. So I don't know how frequently they actually test that level switch. Do they have a test standard? Okay. So an hourly failure rate doesn't make any sense at all for that level switch.

MR. PHAN: That is correct.

CHAIRMAN STETKAR: Okay.

MR. PHAN: We did not ask them any further question on that numbers because, to us, it doesn't make sense. However, when they respond to the second question regarding the consequences, they assumed that, in either way, the flooding would only impact one side of the reactor buildings. The worst case of non-success to isolate the rupture would be more water get into the higher levels, but also the impact or the consequences remitted to the one side of the RB. That's why successful or failures to isolate the rupture would have minor impact on the calculation.

CHAIRMAN STETKAR: Because they're taking out that half of the building anyway.

MR. PHAN: Yes, yes.

CHAIRMAN STETKAR: See, that might be the saving grace.

MEMBER BLEY: The saving grace. It doesn't explain the analysis.

CHAIRMAN STETKAR: By the way, IEEE 500 is something that, I've got to be careful, I, as an individual, would never recommend anyone to use for data. It is, there is operating experience in the U.S.

industry, a lot of operating experience that shows that, essentially, all of the failure rates in that reference are numerically optimistic. The failure rates in that reference were derived by what's called a Delphi method, which is asking a bunch of people what they thought the failure rate might be. Our actual experience shows that real failure rates are a lot higher because the experts weren't asked the right questions.

MEMBER BLEY: And it was done in the mid 70s.

CHAIRMAN STETKAR: And it was done in the mid 70s also when nobody really thought about this stuff. So IEEE, any numerical values that are derived from IEEE 500 you almost have to presume are numerically optimistic. That's just kind of pretty well known in the data field these days. Some of them are a little less or more numerically optimistic, but it's very, very rare to find numbers there that are supported by actual operating experience.

MEMBER BLEY: I guess the other thing you touched on, you said they did say on their assumptions the probability of ESWS flood without isolation in 15 minutes is judged to be very small. Considering the flood frequency, looking at the flood frequency is really small, you know, unexpected events make human response less likely, I think, rather than the other way around. So that one confused me just to start with. Anyway, if it doesn't matter, it might not matter. But the analysis --

CHAIRMAN STETKAR: Well, the question is, you know, if it doesn't matter --

MEMBER BLEY: Why is it there?

CHAIRMAN STETKAR: -- then why do the analysis?

MEMBER BLEY: Yes.

CHAIRMAN STETKAR: Just say that flood takes out half the reactor building and --

MEMBER BLEY: No matter what --

CHAIRMAN STETKAR: -- you either isolate it or you don't, and that's it.

MEMBER SCHULTZ: It's better to go back and take out the assumptions if the analysis, in fact, is not being used to justify the conclusion because the conclusion is derived differently from the analysis. Otherwise, it appears that the 15-minute assumption, for example, has some validity, and it doesn't. It hasn't been justified, but it still sits there because the consequences, it doesn't make any difference to the consequence. The assumption is that the trains are out of service. But if it sits there, it can be used for other purposes. Somebody picks up the analysis later and thinks it's justifiable, but it's not.

MR. PHAN: Yes, thank you. Yes.

MEMBER SCHULTZ: So it's important to clarify.

MR. PHAN: Yes. In the next revision of our safety evaluation, we will consider your advice to clean up those assumptions that not relevant to the model. Thank you.

MEMBER SCHULTZ: Good approach. Thank you.

CHAIRMAN STETKAR: You were pretty happy that you were off the hook, weren't you, Todd?

MR. HILSMEIER: Off the hook?

CHAIRMAN STETKAR: For going back and presenting the stuff that you didn't review and aren't an expert in.

MR. HILSMEIER: Oh, yes.

CHAIRMAN STETKAR: Yes. Is that it?

MR. PHAN: Am I answering your question? Do you have anymore --

MEMBER BLEY: You've told me what you know about it, yes. Thank you.

MR. PHAN: Thank you, sir.

MR. HILSMEIER: I was just thinking about the IEEE that you mentioned. That's very interesting.

CHAIRMAN STETKAR: I mean, people who deal in the data area don't particularly -- it was really good. Back in the 70s, it was about -- once we started collecting actual data on circuit breakers and transformers and switches and relays and all of that stuff, we found, gee, they're pretty optimistic.

MR. HILSMEIER: Yes, I'll keep that in mind. So am I next?

CHAIRMAN STETKAR: You are.

MR. HILSMEIER: Okay. So the first technical topic of interest is ex-vessel steam explosion. In a severe accident leading to core melting through the reactor vessel into a flooded reactor cavity, potential exists for ex-vessel steam explosion due to the fuel-coolant interaction leading to highly-energetic impulse loads on the containment structure. And this is modeled as event ESX in the containment phenomena event tree.

And no mitigation features are provided to minimize the potential for ex-vessel steam explosion. Rather, design approach relies on a robust reactor cavity and robust reactor coolant system piping that are strong enough to withstand the pressure loads created by ex-vessel steam explosion.

Next slide. The applicant predicted the dynamic pressures inside containment induced by ex-vessel steam explosion using a modified version of a TEXAS-V code, and the applicant predicted the containment structural response due to the dynamic pressures induced by ex-vessel steam explosion using the LS-DYNA code. And the applicant's evaluation showed that both the reactor cavity wall and reactor coolant system piping structures can withstand with sufficient margin the shockwave pressure load generated by ex-vessel steam explosion. And this is what the applicant severe accident progression analysis showed. So, therefore, the applicant concludes that the containment can withstand the loads generated by potential ex-vessel steam explosions and, therefore, the probability of containment failure due to ex-vessel steam explosion is judged to be very unlikely.

Next slide. The staff performed confirmatory calculations using the original TEXAS-V code, and the results were considerably different from the applicant's results. Differences, in part, were caused by differences between the original and modified TEXAS-V code, the effects of nodalization in uncertainties in modeling parameters.

Some of the differences in results included the staff's confirmatory calculations and the applicant's. The staff calculated a 50-percent higher peak explosive shockwave load. Also, the impulse loads showed considerable dependence in their sensitivity on the selected fragmentation model parameter, while the applicant's analysis showed that the peak shockwave load was rather insensitive to the selected fragmentation model parameter. So considering the differences in results between the staff's calculations and the applicant's calculations, the peak explosive shockwave load predicted by the original TEXAS-V code may lead to significantly lower margin between the calculated containment plastic strain and the maximum allowable strain. Therefore, the staff issued RAI 19-521 requesting the applicant to investigate the implications of larger uncertainties in the peak explosive shockwave loads associated with ex-vessel steam explosion.

Next slide. In response to staff's RAI 19-521, the applicant performed several finite element structural analyses for the reactor coolant system pipes and reactor cavity, assuming a larger peak explosive shockwave load for ex-vessel steam explosion. And from their analysis, the applicant determined that reactor coolant system piping structure has sufficient capacity to withstand the challenges from ex-vessel steam explosions over the greater range of uncertainty. However, the reactor cavity structural integrity cannot be assured under the higher end of uncertainty. It kind of depends on the reactor cavity wall model assumed in the finite element structural analysis.

So as such, the LRF sensitivity analysis was performed to determine the impact of reactor cavity failure on LRF. And so in this sensitivity analysis, the probability of containment failure due to ex-vessel steam explosion was conservatively increased for plant damage states where reactor cavity is flooded. And the sensitivity analysis showed that the estimated LRF for all initiators, including low-power shutdown, is below the NRC guideline of 1E minus 6.

MEMBER BLEY: So you weren't able to determine exactly why the two calculations that led to this were, got such different results or which one was correct? I assume they're using the same physics, it's just the way the code is laid out. Have you figured anything out about that?

MR. HILSMEIER: You mean between the difference in TEXAS-V code --

MEMBER BLEY: Yes.

MEMBER SCHULTZ: The modification that was done.

MR. KHATIB-RABHAR: Mohsen Khatib-Rabhar. The TEXAS-V code was developed by Mike Corradini, your colleague here.

MEMBER BLEY: Too bad he's not here.

MR. KHATIB-RABHAR: He's not here, yes. The analyses done by MHI used a version of the TEXAS-V code which had changed the fragmentation model in the code compared to the original model that Mike had put together. And for those who do these calculations, they know the uncertainties are huge. They showed that by changing the model they were able to match the tests which were done on the OECD sponsorship a few years ago, keeping in mind there's only two data points that they were trying to match. Even though the original TEXAS code also, with similar parametrics, could match the same data.

So given the uncertainties in these types of calculations, the first one cannot rely on one versus the other. So we asked them to go to the higher loads and try to estimate what's the impact on containment failure. Having said that, it's important to note, even though you fail the cavity, it doesn't necessarily mean you fail the containment.

MEMBER BLEY: I was going to ask how did

the cavity fail? What was the failure mode? What was breaking? I've seen a picture of it, but that's all I've seen.

MR. KHATIB-RABHAR: You're going to crack the cavity. That's what it's going to do. But keep in mind the cavity is away from the containment structure wall, so in all these analyses, typically, these Level 2 PRAs, I think we've done about 30 or 40 of them, you always put the screening value for ex-vessel steam explosion, even though the most likely failure mode is the following: you have an explosion, it vibrates the reactor pressure vessel, it possibly causes the pipes which are penetrating the containment to perhaps create a leak. That's, in my opinion, the most likely failure mode. Otherwise, failure of a cavity, in my personal opinion, cannot fail containment.

MEMBER BLEY: That's what it looked like looking at the picture.

MR. KHATIB-RABHAR: But, of course, you know, it's very difficult to do these calculations to actually --

MEMBER BLEY: The calculations just show that you develop a crack.

MR. KHATIB-RABHAR: Well, that's, presumably, what, you know -- we haven't seen the actual calculation. That's what I presume this will probably do. But, again, I have not looked at it. The NRC staff have probably have looked at it. We have not. This was outside of the scope of our review.

MEMBER BLEY: Okay. And staff looked at them. What did they . . .

MR. HILSMEIER: The staff still needs to verify the finite element analysis. Pending that, that's a confirmatory item in the SER. Depending on that confirmatory item, the staff concurs that the report analyses and results that demonstrates overall challenges to containment integrity from ex-vessel steam explosions are small.

MEMBER SCHULTZ: Todd, you mentioned nodalization as a difference, as well. Was the difference in results determined, given the applicant's nodalization and the nodalization that was used in the review calculation? Was that difference a large number compared to the 50-percent increase or a large component of the 50-percent increase, or was that determined?

MR. SAWANT: Pravin Sawant from ERI. Using original TEXAS code, we did some nodalization sensitivities. Applicant also did some nodalization sensitivities. So both these nodalization sensitivities did resolve the difference between the calculation, and so it was this 50 percent, approximately 50-percent higher estimation of load.

MEMBER SCHULTZ: So it was mostly attributable to the fragmentation?

MR. SAWANT: Yes, fragmentation model which is different in the modified approach.

MEMBER SCHULTZ: Okay.

MR. KHATIB-RABHAR: Just let me make one more point here. If the Committee is interested, there's a paper which was published by the authors of the modified TEXAS code, which shows a difference in the fragmentation model. We can give you a copy of it. It's a publically-available paper. So for those who are interested -- I'm sure Mike Corradini would be interested to see that

CHAIRMAN STETKAR: Thank you.

MR. HILSMEIER: Next slide, please. I already went through the conclusion. Next slide, please. The next topic is hydrogen generation and control. In a severe accident leading to core melt, hydrogen would be generated due to oxidation of fuel rod cladding, MCCI, oxidation of other core structures. Therefore, the potential exists for hydrogen combustion leading to containment failure.

Mitigation features to minimize containment failure due to hydrogen combustion include an open and large containment that allow good containment atmospheric mixing and to prevent excessive combustible gas accumulation. In addition, the containment vessel provides sufficient strength to withstand pressure loads generated by most hydrogen burns. And, lastly, the original design, US-APWR design consisted of 20 AC-powered hydrogen igniters. However, subsequent of NRC analyses showing potential of hydrogen concentrations exceeding 10 percent inside the RWSP, the design was modified to provide DC power to 11 of the 20 igniters with backup dedicated 24-hour batteries. And we'll get into the analysis more in the next slide.

Next slide, please. The applicant's severe accident progression analysis using GOTHIC included that localized hydrogen burns could be initiated by igniters in compartments near release points, that global burns in the dome and deflagration to detonation transition, or DDT, is not expected since igniters control hydrogen concentration below 10 percent. Also, the peak static pressure from hydrogen burn would be below 70 psi, which is well below the containment ultimate pressure capability of 216 psi. Also, a flammable atmosphere in the RWSP is predicted for a medium LOCA scenario, but the hydrogen concentration remains below 10 percent.

So overall conclusion is that there is no DDT potential during severe accidents, that the containment atmosphere is well mixed. That was the applicant's original severe accident progression analysis.

CHAIRMAN STETKAR: I need help from my people who understand how water boils and hydrogen explodes. That peak static pressure of 70 psia, if I'm in this alternate containment heat removal cooling mode, that's been evaluated as being initiated when containment pressure reaches the design pressure, which is about 85 psig, let's say. And if I look at the traces from the analyses that were done, pressure doesn't come down very fast. It kind of stays around 85 and slowly tails off. I don't know what it's going to be, you know, during core melt accident, so should I add this now, 70 pounds to the 85 pounds or so, and get about 155 pounds, which is somewhat closer to my 215 pounds? And then what question do I, what question do I, you know, if I'm in this alternate containment heat removal mode, what pressure do I really have in there?

So just saying if I have a completely depressurized containment and a 70-pound pulse, you know, that's quite a bit of margin. I don't know what uncertainties there are about these things, but the margin might not quite be what's indicated on that third bullet if I'm in this alternate containment heat removal mode. I'll grant you that if I'm cooling the RWSP through the RHR heat exchanger somehow, the pressure is going to be pretty low because that's --

MR. KHATIB-RABHAR: Mohsen Khatib-Rabhar here, John. You don't just add up pressures. That's not how it works.

CHAIRMAN STETKAR: Okay.

MR. KHATIB-RABHAR: You have to see what is the steam concentration in containment when you're trying to burn hydrogen. If you're above 55 percent --

CHAIRMAN STETKAR: Yes. And in this case, you'd be at a high steam concentration --

MR. KHATIB-RABHAR: So you have steam concentration, you're not going to burn, you're not going to have --

CHAIRMAN STETKAR: Yes, you're right, you're right, you're right. Thanks, Mohsen.

MR. KHATIB-RABHAR: Sure. Hopefully, that clarifies.

CHAIRMAN STETKAR: That does. Thank you.

You're right.

MEMBER REMPE: But I know when I read that, I was a little puzzled. Usually, we talk about the design pressure, but if we're getting, it seems like in the document, if we're getting close to the design pressure, let's just go ahead and bump it up to the ultimate pressure. You've reduced your margin. Is that readily accepted by the staff now? I mean, that's why we have ultimate in design pressures is we have that --

CHAIRMAN STETKAR: Yes, that's okay. This is a PRA. This is not a, this is a risk assessment, so using the ultimate pressure --

MEMBER REMPE: Okay.

CHAIRMAN STETKAR: -- capacity, if it's justified, is fine.

MEMBER REMPE: Okay.

CHAIRMAN STETKAR: I mean, this is not a design licensing calculation.

MEMBER REMPE: Right. I just -- sometimes, even though in the PRA, we go back to the design.

MR. FULLER: This is Ed Fuller. The way we evaluate it is we look to see how the core melt progression analysis, in this case with MAAP 4, calculates what happens when you have conditions for a burn. Whatever the temperature and pressure and molar concentrations are in the containment at the time, as Mohsen indicated, pretty much determine what kind of a burn you're going to get.

So, typically, you know, you would get a burn that adds, I don't know, 40 - 50 psi to what's already in the containment. The containment is not at atmospheric pressure when this happens, okay? So you look to see what happens and how close are you to detonation conditions when it goes off, what is the hydrogen concentration when it goes off, and then you just decide whether or not there's any way to get DDT or confined spaces in your containment.

So, you know, these containments have to be designed to take 100 percent of all of the hydrogen you could produce from cladding oxidation, and I stress the word cladding oxidation. And they have to show that, well, two things. First, that they cannot exceed 10 percent mole fraction of hydrogen; and, second of all, that the burn that would happen would not fail the containment. You have to maintain containment integrity for 24 hours as part of the regulation here.

So in terms of that particular bullet, you know, the only way you're going to get to 216 psia or anywhere close to it is to have a detonation. Now, what does that mean in terms of this design? Just to turn the page a little bit, in the process of the review we found that there were conditions in the RWSP, refueling water storage pit, where you could exceed 10 percent. And, you know, in the process of going through the review, one of the things that happened was, as Todd already said, we now see that they're putting igniters on half of the, half of the igniters on DC power for 24 hours. So I just wanted to lay out the land for you, so you understand the context of the issue.

MEMBER REMPE: Since we're talking about the RWSP, is this a good time to revisit my question about the controlling of the water in the cavity and exactly how that is done? And it is to control hydrogen, and they're right. And could you just elaborate a little bit about the process?

MR. HILSMEIER: Yes. I've discussed with Ed Fuller and the contractors during lunch; and, basically, in the hydrogen progression analysis, water level is modeled. The water level is a modeling assumption to preclude MCCI and also to push the hydrogen through the RWSP. This maximizes hydrogen concentration in the RWSP, but in the actual severe accident it's not necessary to maintain the water level.

It's just a modeling assumption made --

MEMBER REMPE: Okay. I guess from what I was reading in the SER, I didn't get that. I got that you didn't need to have the instrumentation, they could control it another way, but I didn't hear the final bottom line or perhaps I was about to fall asleep at that point.

MR. HILSMEIER: It is mentioned in the SER, but it's maybe not as clear as it should be. And we'll clarify that.

MEMBER REMPE: Okay.

MR. FULLER: One other point. This is Ed Fuller again. Although the hydrogen concentration in the containment always remains below 10 percent by design, in the RWSP it can go higher because you are in a confined space with water and that, in turn, condenses whatever steam comes in along with the hydrogen. So you end up with a situation where you're no longer steam inerted.

CHAIRMAN STETKAR: Ed, how confined is that RWSP space?

MR. FULLER: Well, you know, I took a look at it, and it looks to me like, with the water level typically where it's supposed to be, you've got yourself the equivalent of a parfait layer on the top of atmosphere, probably about, probably less than a foot, but I don't know. I think MHI could enlighten us further on that. But if you look at the geometry, it was definitely one where you could get flame acceleration.

MR. KIPPER: This is Scott Kipper from MNES. I just wanted to point out that the, basically this RWSP hydrogen was only occurring during an extended loss of AC power when we not only lost containment spray and safety injection but also power to the igniters. So that's the event tree which would result in the high RWSP hydrogen levels. Basically, what you have there is the containment spray and safety injection are not drawing down the RWSP water, so you have a sub-cooled water volume which is increasing the steam condensation there. And so our solution for that was, instead of trying to deal with this, we tried to prevent it by increasing the reliability of having the igniters available through battery power.

MR. KHATIB-RABHAR: Mohsen Khatib-Rabhar. I disagree with that position. It actually turned out, even if you credit all the igniters, it doesn't get away the issue of reaching concentrations inside the RWSP, which is significantly higher than what you expect them to be. And the problems that Ed Fuller described that you can get a lot of steam and hydrogen going there and the steam is going to condense and hydrogen becomes very high in terms of concentration. So, yes, having additional AC power helps, but it doesn't eliminate the problem. We have analysis to show that, by the way. And I think that's been submitted.

MEMBER REMPE: So I guess I'm a little confused. I thought the staff told me they don't need to worry about controlling the hydrogen anymore, right?

MR. KHATIB-RABHAR: Yes, let me tell you how that --

MEMBER REMPE: And you're telling me --

MR. KHATIB-RABHAR: -- disposition came about.

MEMBER REMPE: Oh, good.

MR. KHATIB-RABHAR: This disposition came about probabilistically. The MHI did a sensitivity study. They increased the likelihood of containment failure due to detonation, and they showed the LERF value would not be exceeded. So we accepted that.

MEMBER REMPE: So the bottom line is what the staff is true?

MR. KHATIB-RABHAR: Precisely.

MEMBER REMPE: That's what I want to hear.

MR. KHATIB-RABHAR: Precisely. But, deterministically, the problem does not go away.

MEMBER REMPE: Okay.

MEMBER BLEY: I'm just curious, is that, the RWSP area down tube? I don't know how the 3D organization of that space is. I just saw the sketch. And what I'm wondering is, once you get all the steam in there and the hydrogen coming in, does the air that was in there get moved out, or is there still --

MR. KIPPER: Well, we do have five pairs of air vents around the RWSP for pressure equalization. How they are set up is that at each pair they are offset, and one pair normally stays below the RWSP water level. That's to limit evaporation losses during normal operation. And when the RWSP water level is actually drawn down, that then allows both of those, that venting and air flow path. When the RWSP is not drawn down, when we don't have containment spray or safety injection, then the one pair of vent pipes stays below the water level, and so that's when we had, that's when we had additional accumulation within that air space.

MEMBER BLEY: Okay.

MR. HILSMEIER: Any other questions?

MEMBER REMPE: We'll let you go to another slide.

MR. HILSMEIER: Actually, the next slide is pretty much what all the questions that have been going on covers. So the staff performed confirmatory analyses using MELCOR code and confirmed the applicant's findings, with the exception of detonatable hydrogen mixture in the RWSP. And the staff's confirmatory analysis predicted hydrogen concentrations exceeding 10 percent in RWSP during long-term station blackout scenarios. So in response to RAI 19-449, the applicant's analysis also showed a potential for hydrogen concentrations exceeding 10 percent in the RWSP.

MEMBER REMPE: Just for later on for questions to confirm, I believe this MELCOR calculation assumed 100 percent or hydrogen that would equal 100 percent of what you would get if all the in the core oxidized, right? And whether that hydrogen came from steel structures oxidizing or cladding is irrelevant, but that's the mass of hydrogen, right?

MR. HILSMEIER: Yes, yes, it is.

MEMBER REMPE: That's true, right?

MR. HILSMEIER: Yes.

MEMBER REMPE: Okay.

MEMBER SCHULTZ: That's not what I heard Ed say earlier.

MEMBER REMPE: How much hydrogen did they assume? I thought that your GOTHIC or your hydrogen calculation was, I thought, assuming that.

MR. KHATIB-RABHAR: I hate to be at the microphone only two seconds, but yes. If you do a deterministic analysis using MELCOR, you can get it higher than 100 percent if you oxidize steel under certain conditions. But you are absolutely correct. For the analysis that we did here, we use 100-percent equivalent to demonstrate that. But that's not the absolute limit.

MEMBER REMPE: Right.

MR. HILSMEIER: So in response to RAI 19-560, the applicant proposed a design change to provide dedicated batteries out of the 20 igniters. The dedicated batteries will have the capacity for at least 24 hours following onset of station blackout and also alternate AC, and these DC power igniters are strategically located near potential hydrogen release locations. And in addition, as part of the SAMGs, the reactor cavity will be flooded by diesel-driven firewater system to provide core debris cooling to prevent MCCI.

And the applicant also performed a GOTHIC calculation for long-term station blackout with the proposed DC-powered igniter configuration and showed that hydrogen concentration inside containment, including RWSP, remains below 10 percent. Also, containment integrity is maintained for 24 hours after the accident.

MEMBER SCHULTZ: Remains below 10 percent for the accident? What does that mean? Below 10 percent or is it above 10 percent later when the igniters are --

MR. HILSMEIER: Throughout the accident remains below 10 percent.

MEMBER SCHULTZ: Throughout the accident, if the igniters are available for 24 hours --

MR. HILSMEIER: Correct.

MEMBER SCHULTZ: -- with DC power? In other words, having igniters available 24 hours with DC power and loss of outside power, then the hydrogen concentration will not exceed 10 percent?

MR. HILSMEIER: Correct.

MEMBER SCHULTZ: For the duration of the accident?

MR. HILSMEIER: Is that correct, Mohsen? I see your head shaking.

MR. FULLER: Steve, the other piece of this is that, for the line return after 24 hours, you need to make sure you don't get any non-condensable gas generation from the core-concrete interaction. So that SAMG act of getting water in through a diesel-driven firewater pump is absolutely essential to prevent that from happening.

MEMBER SCHULTZ: So it's through that combination? I just wanted to make sure we picked the words properly. Thank you.

MR. HILSMEIER: Next slide.

MEMBER REMPE: Okay. So this is where I'd like to take a detour. I looked through the remainder of your slides, and I don't see anything that really discusses the effort that's described in the document you sent me that the staff did for their MELCOR calculations, and I just would like to be educated a little bit about some of the assumptions made. The staff regularly talks about, you know, that they did some sort of checks with MELCOR, and I'd like to talk about some of the assumptions that I think I saw in the MELCOR analyses last night, if that's okay with you, sir. Okay. I didn't see a no, so let's just plow ahead.

First of all, it was MELCOR 1.83, so the staff did not use the latest version of MELCOR; is that correct?

MR. FULLER: That's probably correct. The chronology goes somewhat as follows. The confirmatory assessment report that you're referring to was done by the Office of Research, and I guess it was done in around 2010, 2009 - 2010. I can't remember the date. And that pre-dated those ERI reports by a couple of years.

Now, the work was done in the Office of Research. It was not done by ERI. And, unfortunately, the person who did the work is not here today. I didn't think to ask him to come down, but he could talk about whatever assumptions were made.

MEMBER REMPE: Maybe you or the staff can answer a couple of questions. What was the intent of the calculations? Was it just to try and match the results that MAAP got? So, for example, if I look at the --

MR. FULLER: I can speak to that, yes.

MEMBER REMPE: Okay.

MR. FULLER: The idea was to take what we considered the most typical set of accident scenarios performed by MHI and run as close as possible the same scenarios with MELCOR. And a MELCOR deck was put together based on information provided by MHI and by looking at other, you know, the MAAP parameter file that they have, for example, and things like that. And I believe that, generally speaking, the only assumptions that might be made would have to do with treating some boundary conditions, perhaps, or something. But the intent was to run MELCOR the way we would normally run it, not necessarily to tune it to the way MAAP is run.

MEMBER REMPE: Okay. So I didn't see a table that lists assumptions like I did see in the MAAP report or the PRA. But if I look at plots like fuel temperature in various rings, if it showed relocation occurring, it appears that at 2500 K, which is approximately 2200 C, is where you're melting the fuel because the fuel drops out of the core and goes down.

MR. FULLER: Which is a lower temperature than MAAP does.

MEMBER REMPE: Actually, they assume 2200 C, and your plots indicate you had 2500 K, which is pretty darn close, within 23 degrees or something like that, right?

MR. FULLER: Well --

MEMBER REMPE: And, again, I don't have a table --

MR. FULLER: -- typically, MAAP will get you up to 3,000 Kelvin in the core.

MEMBER REMPE: That's true, but remember yesterday our friends from MHI said they picked a conservative temperature for fuel melting.

MR. FULLER: That must have been before I arrived. I didn't hear that.

MEMBER REMPE: Actually, it was after you left.

MR. FULLER: After I left?

MEMBER REMPE: It was after you left, yes. And during that discussion, and you can probably --

MR. FULLER: So they changed some of the model parameters?

MEMBER REMPE: Those were input parameters, right?

MR. FULLER: I was not aware of that.

MEMBER REMPE: And, actually, then, in the old days, because, again, I was just aware of what was in the other codes, but it used to be MAAP stopped hydrogen production when you had fuel melting. So even though it may be conservative for one thing, it may not be conservative for another thing.

MR. FULLER: Well, not quite that way. MAAP doesn't ever stop hydrogen production, provided steam can get to the metal being oxidized. What happens in MAAP is you have this TMI crucible model of melt formation and progression, so when it hits, in a PWR, when it hits the core support plate, it just stops and you form a crust, just like was observed in TMI.

MEMBER REMPE: Well, actually, I think --

MR. FULLER: And then, at that point, you get a blockage, steam goes around this and doesn't hit the unoxidized zircaloy.

MEMBER REMPE: In the old days, about AP 600, that was when it would stop hydrogen production. And I was aware of it. Mohsen behind you shaking his head saying, yes, she's right this time. So, anyway, that was my understanding. But where I'm going to is that, as you acknowledge, well, 2200 C, that's pretty darn low for fuel melting from a MELCOR analyses, and I'm guessing that the staff tried to run the MELCOR code in a way that they could check the MAAP results, and they picked that in, you know --

MR. FULLER: I'm not so sure. I wasn't trying to say it was low from a MELCOR point of view. I was saying from a MAAP point of view.

MEMBER REMPE: Oh, anyway, it looks like, you know how MELCOR has these plots and there's fuel in the cores, and it gives you a temperature of that fuel, and suddenly it drops to zero because it's left the core? And it looks like, to me, that what the staff did was assumed a fuel melting temperature of about 2500 K. And, okay, so Mohsen seems to --

MR. KHATIB-RABHAR: Let me explain to you how MELCOR works. You're looking at two different things. In MELCOR, there's a relocation temperature. The default value is 2,800 degrees K.

MEMBER REMPE: Well, this time it's 2500.

MR. KHATIB-RABHAR: Just one second, please. Bear with me. You can relocate fuel before you reach 2800 degrees if the supports go. So there's another mechanism for relocating if the core supports are heated up to high enough temperatures that the fuel would relocate.

MEMBER REMPE: What's the material in those supports?

MR. KHATIB-RABHAR: One more thing. I'll address that question last. You can also -- sometimes, the plotting does not show that the temperature reaches 2800 degrees because the way you're plotting the results. Sometimes, you know, you just put a plot out, and 2500 or 2800 degrees occurred in a very short time period and you missed the peak. So you got to be very careful when, you have to actually look into the details of the MELCOR calculation to find out what was the cause of relocation and whether you reached 2800 degrees or relocated before reaching that.

The supports are typically steel, which is causing the -- for instance, if you have circulation, you can heat up the lower core area to high enough temperatures. Once your supports fail in MELCOR it can relocate. So it's not just the temperature criteria in MELCOR which causes relocation, so you've got to be careful how you use that parameter.

MEMBER REMPE: Okay. So I'm looking at levels two through nine for about five different rings the fuel temperature as a function of time, and every single time you get to 2500 K and the fuel goes down. And then later we can look at the plots at the core plate, but, anyhow, where I'm going to is did the, there used to be user defined parameters in MELCOR that could help you determine what you wanted to have occur. And I don't know what was done in these particular calcs but --

MR. KHATIB-RABHAR: Well, first of all, we haven't done those calculations, so I cannot speak with RES. But the way currently things are done, NRC has a number of what's called blessed default parameters that have come about based on a CERCLA study, and those are the ones that are currently being used for default. In the older days, you had a lot more, you know, variability in what you used for the parameters. You're absolutely right. You can play around with these parameters and get whatever you want.

MEMBER REMPE: Yes.

MR. KHATIB-RABHAR: No question about it. I've done this for 40 years.

MEMBER REMPE: I know.

MR. KHATIB-RABHAR: So the bottom line is that you have to actually look at the results to see how they came about, whether the relocation was due to lower melting temperature, due to the fact that you may have had failures in support structures, or many other aspects. So my suggestion to you is do not just focus on the melting temperature. You need to ask the question from RES exactly what was the cause of relocation.

MEMBER REMPE: Well, what I'd like to know is what was the objective of the calculations because it seems like they were trying to get some confidence in the MAAP results. And if that's the case, dinking with the default parameter might not have been a bad, you know, might have been reasonable. And I think we don't have the people here today to answer that question.

MR. FULLER: Well, I can tell you what the overall objective was because I was the one that actually did the calculation in the first place. The objective was to see how close we could match the two accident progression models for the accident scenarios in question regarding timing of events, pressure temperatures, fission product releases. And then I don't know if you've made it all the way through the end of that report, but there's a section they actually compare the MAAP and MELCOR results.

MEMBER REMPE: And I'm looking at one of those tables now. They did it for every scenario.

MR. FULLER: Yes. And you'll probably see that when it comes to the melt progression part, things don't look too different. When it comes to source terms, sometimes they look quite different.

MEMBER REMPE: Well, actually, I guess I've seen vessel failures one time in one case that was 7 hours with MAAP and 14 hours with MELCOR. I guess where I'm leading to eventually is that I understand, too, that once you get past top of core, there's a lot of assumptions in MAAP and MELCOR that have some basis experimentally, but they sure differ and you get different results. And so an overall saying, well, these results differ, but maybe things don't change too much isn't maybe so bad for looking at severe accident phenomena, but when you're talking about success criteria and you see an order, you know, a factor of ten difference on where the top of core and the timing in a scenario, I get a little more concerned. And that's why today earlier I was saying to MHI I really would like to have a little more confidence in their MAAP model because that's something, thermohydraulics ought to be something you could benchmark against other approved NRC codes. And so I'm going to ask the same question, too, with the MAAP model. Maybe that should be done, too, because we'd like to have a little confidence, before you get into the things that get more fuzzy in the thermohydraulics capability of this particular application of MELCOR. I realize MELCOR has been and a bunch of other things, but it's a different model.

MR. FULLER: Yes, I think we understand the differences of the sort you're talking about, and it has directly to do with the melt progression treatments.

MEMBER REMPE: No, it's talking about just depressurization.

MR. FULLER: I'm sorry?

MEMBER REMPE: On water level in the vessel, and it's just the thermohydraulics modeling. That's something we ought to be able to have --

MR. FULLER: Well, in my experience, that always looks pretty close.

MEMBER REMPE: I remember some benchmarks against the tests with MELCOR and MAAP and SCDAP, and things weren't so close. And, again, this is a different plant model, and just before we start getting into the fuzzy stuff with severe accidents, you've got RELAP analysis, you did for a medium LOCA, small LOCAs. And, okay, if you can't use MAAP for a large LOCA, let's just see how bad it is. Let's just see some thoughts comparing water level in vessel and pressure in the vessel until you get to the top of the core.

MR. FULLER: Okay. I think maybe the best course forward is, with respect to this particular meeting today, is for us to get some information to you regarding what some of the criteria were that the analysts actually developed and followed in the course of the confirmatory assessment.

MEMBER REMPE: That would be helpful, but, again, I'd like to just see some benchmarks of what MAAP predicted against SCDAP, and I would hope the staff, because, I mean, that's just maybe curiosity because of your using MELCOR, but when you get to success criteria with the MAAP analysis, I'd hope the staff would also ask for something like that from MHI.

MR. FULLER: The only comparisons I'm aware of of MAAP against SCDAP is in the context of induced steam generator tube rupture that was done about ten years ago where Mark Kempton did the MAAP calculations and Karen Bera, when she was at Purdue, did the SCDAP RELAP 5 calculations. And I think that's published, but we can get you, certainly get you a copy of the documentation of that.

MEMBER REMPE: But, again, I'm concerned with this particular plant model. Everybody makes mistakes. Let's just get some confidence in the thermohydraulics. And so just because you have an analysis for some other plant that was done or I might have seen some analyses for an AP600 that was done, it's just, that's kind of how we always started off. Let's just see if we can do the thermohydraulics, and then let's move forward.

MR. FULLER: Okay. I wish I could help you more but . . .

MEMBER REMPE: I'm sorry. I think I've belabored it enough. I think that's the point I wanted to make with going through all this.

MEMBER SHACK: Well, before you leave, Ed, let me ask a question. When I looked at these analyses, what struck me was that six out of the six analyses there was no containment overpressure failure from the MELCOR calculations, and five out of the six MAAP calculations said I would fail containment by overpressure.

MR. FULLER: This is the APWR?

MEMBER SHACK: Yes.

MR. FULLER: Oh.

MEMBER REMPE: And, actually, there's a lot of text that says that the MAAP analysis is conservative for this case. There's no going into why is it different.

MR. FULLER: The question I would ask is in the MELCOR calculations was the containment pressure still going up at the end of the calculation?

MEMBER SHACK: I'm only looking at comparison tables.

MR. FULLER: Okay. I'd be surprised if they had fundamentally different conclusions.

MEMBER SHACK: Okay. You think somehow that the stop-off of the timing was just different, and it would have gotten there? Yes, we got these last night. All I had time to do was look at the tables.

MR. FULLER: And I apologize to the Committee for not being astute enough to bring the person in the Office of Research down here today.

MEMBER REMPE: But, actually, again, there was a lack of trying to understand the differences in the two codes in the analysis, that we just say, well, MAAP was conservative for this, and so that would be a good question to follow up on, too.

CHAIRMAN STETKAR: Remember: interpret silence as an opportunity to speak.

MR. HILSMEIER: The staff performed a confirmatory analysis and verified the applicant's results regarding the effectiveness of DC-powered igniters in controlling hydrogen concentration during long-term station blackout scenarios.

MEMBER SHACK: But I thought your consultant said that wasn't true?

MR. HILSMEIER: Yes, I don't know.

MEMBER BLEY: Time for a break.

CHAIRMAN STETKAR: No, I really want to go through the next four pages, if we can. There are folks from Texas who need to go home tonight.

MR. PHAN: We last on the external --

CHAIRMAN STETKAR: I know. I'm aware of that.

MR. HILSMEIER: There may be more open items after all of these discussions. But officially, as of now, an open item and it's related to hydrogen generation control, how it's modeled in the fault trees. It wasn't clear whether the new hydrogen control top event that modeled DC-powered igniters was used in the fault tree modeling. And per a recent telecom, the applicant clarified to us how the DC igniters were modeled in the fault trees, and we're currently reviewing the response on that.

MEMBER SCHULTZ: They provided a written response following the phone call?

MR. HILSMEIER: Yes, and, actually, we just received that written response as of, like, I think earlier this week. Next slide. The next technical topic of interest is core debris coolability. In a severe accident leading to core melting through reactor vessel into the reactor cavity, the potential exists for containment failure through MCCI. The potential exists for containment failure through MCCI if the molten debris in the reactor cavity is not sufficiently cooled, and this is event EVC in the containment phenomena event tree.

The applicant's design approach to mitigate this severe accident type is flooding the reactor cavity to cool the debris using the containment spray system or the firewater injection system. Another mitigation feature is the actual design geometry of the reactor cavity to enhance spreading of the corium to ensure adequate coolability. Basically, the reactor cavity has a wide-open floor over 970 square feet and a reactor cavity floor thickness of 36 inches.

Next slide. The applicant performed a

severe accident progression analysis for core debris coolability, and this slide presents the results of the applicant's analysis. For severe accidents where molten debris dropped into a flooded reactor cavity, the debris appropriately cooled and no basemat erosion occurred. Where molten debris dropped into a dry reactor cavity and then the cavity was flooded, the debris cooled and there was slight basemat erosion, about 0.1 inches. And where molten debris dropped into a dry reactor cavity and it was never flooded, the basemat melt-through occurred after 28 hours and containment pressure within 24 hours remained below the ultimate containment pressure. Also, a sensitivity analysis shows that basemat melt-through and the containment overpressurization failure did not, are not expected to occur within 24 hours.

Next slide. The applicant also showed that the molten core debris spreads very well over the entire reactor cavity floor. Molten core depth over most of the floor is less than ten inches. The ten inches is an acceptance criteria in Generic Letter 88-20. Also, molten core debris accumulation in a very limited area could exceed ten inches, and this is a small area adjacent to the reactor cavity wall. This potential was treated probabilistically in the Level 2 PRA.

Next slide. So in NRC's evaluation, we felt the applicant used the latest versions of MAAP and FLOW-3D, realizing that the applicant started the design back in 2006. And they've provided sufficient information to show that these codes are adequate to support the conclusions in molten core debris spreading and coolability. The applicant also used recent data and studies on MCCI and core debris cooling and demonstrated how it's applicable to the US-APWR design. And the applicant understood that the research on debris cooling remains incomplete and subject to inherent uncertainties. Therefore, the applicant performed a variety of sensitivity studies. Such sensitivity studies included the amount of corium involved, the heat transfer coefficient between the molten core and coolants, use of limestone sand concrete versus basaltic concrete basemats. And the staff also found that the applicant's methodology and assumptions are suitable for evaluating core debris spreading and coolability. And, lastly, the staff performed a confirmatory analysis using MELCOR for several severe accident scenarios where debris cooling was assumed unavailable, and results showed that basemat melt-through occurs later than 24 hours.

The staff concludes that containment

integrity is likely to be maintained for more than 24 hours after onset of core damage. And, also, the staff concludes that the acceptance criteria regarding core debris cooling and MCCI defined -- the staff concludes that the acceptance criteria defined in SECY-93-087 and Generic Letter 88-20 on core debris cooling and MCCI are met.

And the next slide is the risk metrics. As you see, this slide presents a lot of, presents a CDF and LRF for at-power shutdown and also the containment, conditional containment failure probability. However, due to the unresolved open items, the staff cannot make any final conclusions on how the US-APWR containment performance compares to the Commission goals.

And that's all I have to say, unless there's any questions.

MEMBER REMPE: There was one that I mentioned before lunch. If we're running short of time, maybe -- some of it's been documented in Chapter 15, but it's the survivability assessment. In particular, I have a few questions about instrumentation and why certain sensors were picked and why they had to survive certain conditions and why radiation monitors weren't selected.

MR. HILSMEIER: Right. The methodology for evaluating equipment survivability first included identifying the time frames for equipment survivability. There was three time frames identified. There's T02122. T0 is before core damage, T1 is from core damage to vessel melt-through, and T3 is after vessel melt-through. And an equipment survivability assessment only considers time frames T2 and T3, so, basically, from core damage to after vessel melt-through. And the applicant identified about five SSCs needed for equipment survivability: containment penetrations, hydrogen igniters, depressurization valves, and the containment pressure sensor.

MEMBER REMPE: They don't have severe accident management guidelines yet, so how did they identify those things needed by the operators?

MR. HILSMEIER: I would imagine, from the Level 2 severe accident analyses, they were able to identify the equipment. Ed probably can shed more light.

MR. FULLER: This is Ed Fuller. I'll offer you my perception of what might be happening. In the JLD activities, one of the items is to deal with recommendation eight of the Fukushima Near-Term Task Force so that severe accident management guidelines and training, etcetera, are adequately done and auditable in the future. The industry has just completed an update of their severe accident management technical basis report, which is now with the Owner's Groups.

With respect to how this particular one, this design might go, I believe that Mitsubishi would probably have to attach or, rather, say Comanche Peak would have to attach themselves to an Owner's Group and make sure that they work to get the insights from all of this activity into their severe accident management technical basis and, going forward, make sure that all of the relevant actions can be taken care of. So I believe that this particular design and COL activity process is probably in the same boat as the operating plants are right now.

MEMBER REMPE: Well, actually, again, I'm reading your SER because I wasn't aware of this because I'm still sort of new to ACRS, but there were two SECYs that apply to the advanced light water reactor designs, and that's where the requirement to ask them to do this came from, right? And so, apparently, this was done with some of the other design certification activities, and I have not been party to it, but I just was wondering with what rigor and, again, they maybe only have identified a pressure sensor, does the staff interact with the applicant and say, well, jeepers, at TMI we thought the radiation sensors might have been a good thing to know when you're having a problem? Is there any give and take, or you just take what they say and say okay? Again, at TMI moisture was a big thing and the pressure shock from the hydrogen burn was a big thing for damaging sensors. And I just was kind of wondering maybe this is something to take as an action item? Part of it is for my own education, but it would help me to better understand what's in the SER if I could have a little more background on what occurred. Thanks.

MR. FULLER: Well, you know, in this one, as well as the other design cert applications, we asked for commitment by the vendor to provide the technical basis for the SAMGs to the COL applicants. And that should be all in place, you know, before fuel load at the various plants. And we suggested they do it in the same manner as done for the existing plants, through the Owner's Groups.

You know, today, even today it's all voluntary initiative on the part of the industry to comply with what they said they would do. And going forward, as I was trying to indicate, I think it's going to be more than just a voluntary initiative.

MEMBER REMPE: I think, again, the SECY,

it was required they do something, but I just was wondering how much was required right now versus what will happen in the future.

MR. HILSMEIER: Right. SECYs, like RAP is through a SECY, but it's really not a requirement.

MEMBER REMPE: Okay. But you evaluate it and say this --

MR. HILSMEIER: I mean, it's a requirement of staff, but not to industry.

MEMBER REMPE: Right.

MR. HILSMEIER: But industry has complied.

MEMBER SCHULTZ: It's an area that's evolving, obviously, with the lessons learned and the actions coming from Fukushima going forward are in progress.

MEMBER REMPE: Right now, the design --

MEMBER SCHULTZ: And so issues, as we've discussed over the last few hours, could certainly be affected by that. And so decisions like amount of time that one would want to have available, hydrogen igniters available, for example, might be affected by future decisions associated with outcomes of Fukushima evaluations. So I would, we're dealing with time frames that were derived back when at this point in time.

MEMBER REMPE: But right now we don't

require the current plants to have things that survive as much as we're requiring these that are going through design certification, so I just would like to --

MEMBER SCHULTZ: I understand that.

MEMBER REMPE: -- have a better feel for what we have required and are requiring because of what's going on.

MEMBER SCHULTZ: Right.

CHAIRMAN STETKAR: Any other questions for the staff? I think we've sort of got most of our issues out on the table. We may have to wade through some transcript to sort things out, but I think we have.

As I always do, I'll ask if there are any members of the public who have any questions or comments regarding Chapter 19 of the design certification and the staff's review? Anyone? I doubt there will be, but, if not, what I'd like to do is take a 15-minute break, and we are now just slightly ahead of schedule. So let's recess until 2:30.

(Whereupon, the foregoing matter went off the record at 2:26 p.m. and went back on the record at 2:42 p.m.)

CHAIRMAN STETKAR: We are back in session, and we'll hear first from Luminant about their part of Chapter 19. That will be back to Don, I guess.

MR. WOODLAN: All right. Thank you very much. Good afternoon. And as stated, Luminant is now going to present Chapter 9 of the COLA FSAR, 19 of the COLA FSAR. It's been a long two days.

Well, we'll talk about the Comanche Peak 3 and 4 PRA and severe accident evaluation. And I have with me today Hitoshi Tanaka and I think we're also going to have Jim Curry. I don't think he's back in the room yet, but I do think he's going to come up here. Oh, here he comes now.

MEMBER BLEY: He was out in the hallway with his computer.

MR. WOODLAN: They felt I needed two doctors, one on each side, to get me through this presentation. Here's pretty much our standard agenda, which we'll follow. We'll start with an introduction, talk about the SER open items, the SER confirmatory item. We'll have a big discussion about risk-informed tech specs and Surveillance Frequency Control Program. We've covered it quite a bit yesterday, but we'll reexamine again what's in our methodology. And then we'll talk about the site-specific aspects, which is what's in the FSAR itself.

For the FSAR, as with all our sections, we follow the IBR approach in incorporating by reference the standard design. We've taken no departure from the US-APWR DCD. All COL items that are in the DCD have been addressed in the FSAR.

We have seven SER open items. We have one SER confirmatory item, and there's no contentions pending before the ASLB.

So let's take a brief look at the open items we have in Chapter 19. The first item, describe how FSAR will fully address all COL action items listed in DCD Section 19.3. This came from an RAI, and the reason it occurred is that the DCD, in the process of doing the review, have created additional COL action items, and they weren't in the FSAR that the NRC had to review. It was just a matter of timing. So once we got the RAI, we went back, we have gathered all the COL action items, and we have submitted a response that addresses all those. It's under review.

19.2, identify and describe use of PRA and risk-informed applications in accordance with Reg Guide 1.206 guidance. This was a matter of us having presented, I believe, all the information, but it was kind of scattered in the Chapter 19 and in other chapters of the FSAR. So to make it clearer, we developed a table. The table actually is a list of all the programs that use PRA, as well as some information in the text. And, again, this came to us in the form of an RAI. We have responded to that RAI and submitted that table, and it's under review by the staff.

Number three, revise FSAR to address plant-specific PRA technical adequacy, including justification that the PRA is sufficient to support the COLA. This was a matter of having a very good description in the standard plant of technical adequacy and not having anything in the FSAR that specifically addressed the plant-specific aspects. So we did add those words. We linked it to the guidance in Reg Guide 1.200, and we confirmed that we are complying with the reg guide with respect to quality and technical adequacy of the PRA, the plant-specific portion.

CHAIRMAN STETKAR: But you're not going to have a peer review, and I'll use that in the sense of peer review, performed prior to the issuance of the COLA; is that correct?

MR. WOODLAN: That's correct.

CHAIRMAN STETKAR: Okay.

MR. WOODLAN: Number four, and correct me if I'm wrong on anything here, external hazards risk evaluation. Since the review of FSAR Chapter 2 and 3 is ongoing, staff is unable to finalize its conclusions regarding acceptability of external hazards assessments. This pertains to seismic and hydrology. We had a number of issues in both of those categories. We have closure plans that we are currently implementing. Both of those plans are going to be finished over the next four to six months, I would say.

CHAIRMAN STETKAR: Don, is it only seismic and hydrology, or is it seismic and I'll call it meteorology, including hydrology? Because I thought that there were also questions regarding high winds and tornado analysis that would be --

MR. WOODLAN: We have questions in that area, but it's not in that group of open items that we're covering in our hydrology closure plan.

CHAIRMAN STETKAR: It is not. Okay.

MR. WOODLAN: Yes. And so as we complete those closure plans and we complete those chapters, there is a potential that that's going to impact the PRA work. I don't expect that Chapter 2 will. Most of those activities screen out, as you're probably already aware, looking at our material. And we'll cover it in a later slide. Chapter 3, because it's the seismic that will really be addressed, just like on the standard plant, that will be addressed at that point in time.

Next slide. Number five, document that

extreme winds -- this is maybe the one you were thinking of.

CHAIRMAN STETKAR: Oh, okay.

MR. WOODLAN: Document that extreme winds do not contribute more than ten percent of the full-power CDF or shutdown CDF compared to the US-APWR. This was probably a hole in the FSAR in that we did address hurricanes, we did address tornados. This is kind of the extreme winds that don't fall in those two categories. We did go back. We used the guidance that is available but which I consider to be very conservative for our site, considering where it's located. But we did follow the guidance, and we followed the values in there. We evaluated the scenarios where this is important, and we came up, as we present in our RAI supplemental response, that the frequency, the CDFs are less than 1 times 10 to the minus 7.

Next slide. Number six, update screening discussions in FSAR 19.1.5 to be consistent with Reg Guide 1.200, Section 1.2.5, and use site-specific data in the external flooding screening. This was a matter of clarity, actually. We did not explain real well how the screening criteria we used linked up with the reg guide, so we did add words in the FSAR to make that clarification and we specifically referenced back to

Chapter 2 and the PMP data, the probable maximum precipitation -- I actually said that -- and we confirmed that the data being used in Chapter 2 is the same data that were used in the screening activities. And, again, this is a response that we have submitted, and it's under staff review.

Open item number seven, and the last open item, clarify how each cost component is of the averted costs were determined for internal events with a 7-percent and 3-percent discount rate. This came into focus in that we had data in our environmental report and data in our FSAR, and they were different. They didn't match up, and there was reasons for that. We had used different, because of the timing, we had used different versions of the guidance that explained how to do these calculations.

MEMBER SHACK: Just let me go back to that one slide about the PMP. I was just curious. I see the external flooding screening, so that means that, are you going to take the exemption that you've used the PMP according to the SRP and, therefore, you don't have to do a probabilistic flooding analysis? Is that what this means, or that's -- you're looking at it blankly. So you are going to do a probabilistic flooding analysis?

MR. WOODLAN: We are not going to do it.

MEMBER SHACK: You're not?

MR. WOODLAN: It's screened out.

MEMBER SHACK: It's screened out.

MR. WOODLAN: So the data in the environmental report in the FSAR were different. We had used different versions of the regulatory guidance because of timing, and we used different time values for the money. We had different years of time values. So we went back. We brought them both up to the latest and the same version of the guidance. We put them both in the same year money/time value. We updated the environmental report and the FSAR, including addressing both the 7-percent and the 3-percent discount rate. And, again, that information has been submitted with the NRC for review.

CHAIRMAN STETKAR: Don?

MR. WOODLAN: Yes.

CHAIRMAN STETKAR: I can't keep everything straight. Has Comanche Peak Units 1 and 2 submitted or received a license renewal?

MR. WOODLAN: No.

CHAIRMAN STETKAR: They have not yet. Okay.

MR. WOODLAN: I'm not sure we've even filed yet. I know we're --

CHAIRMAN STETKAR: Okay.

MR. WOODLAN: -- we haven't filed.

CHAIRMAN STETKAR: That's the appropriate answer. I was just -- thanks.

MR. WOODLAN: Okay. That concludes the open items. We do have one confirmatory item, and this is to address three items with respect to NEI 04-10 Rev 1 and 06-09 Rev 0. These were, again, in the form of an RAI. We did provide the responses to address all three of those items to the staff. The staff has reviewed them, but we haven't yet filed the next version of the methodology. So it's in the confirmatory status until we do that.

CHAIRMAN STETKAR: And that's basically improvements or changes that are necessary because those documents don't explicitly address new reactors; is that right?

MR. WOODLAN: Yes. And we had included that in Rev 1, which is the current standing revision, but in the, yes, in the staff's review, they found a few areas that they thought needed to be clarified, and we fixed it.

Okay. I'm going to briefly now talk a little bit about the risk manage tech specs and

Surveillance Frequency Control Program methodologies. The risk manage tech specs is controlled by the CRMP, Configuration Risk Management Program -- I knew that -- which is --

CHAIRMAN STETKAR: By its own abbreviations. I'll wait until the third bullet.

MR. WOODLAN: Well, I was just going to kind of summarize it. What it comes down to is we have both programs, the Configuration Risk Management Program (CRMP) and the Surveillance Frequency Control Program. Both of those are controlled by tech specs, administrative tech specs, 5.5.18 for the CRMP and 5.5.19 for the SFCP. In both cases, they rely on the methodology, and the methodology, as we talked about yesterday, is directly linked to the two NEI documents, as revised, to bring those up to date for new plants.

A little bit of a summary. Again, I'm going to go lightly over this slide because we did talk about a lot of this yesterday. This is just a summary of some of the details out of the CRMP, out of the methodology with respect to the CRMP that covers how we're going to do it. It's going to be contained in a site-specific procedures. The procedures themselves will implement the administrative tech specs. It will include things like identifying responsibilities in training requirements. It will include many of the requirements related to PRA, such that the PRA must match the as-built plant. It must be of, and we've talked about this a lot the last two days, of sufficient detail or granularity that you can evaluate the risks for the tech spec components that are of concern. There is a section in the methodology that discusses the PRA specifically, and the program itself, the procedures will include and discuss the CRM, the configuration risk management tool, the actual tool that's used when you have to enter the program and try to adjust your outage times.

CHAIRMAN STETKAR: And, Don, just to belabor the point yet one more time, the PRA which models the as-built plant in, as you've characterized it, sufficient granularity needs a lot of work. It can't have a basic event that says main feedwater system. It can't assume that recovery of offsite power magically re-energizes every bus in the plant because that doesn't happen. Circuit breakers have to work. It can't assume that if offsite power is recovered between 3 hours and 24 hours that you can re-energize a bus because you don't have any DC power. So it's got to account for that stuff. There's a lot of work to be done, and I honestly really hope you'll appreciate the amount of work that needs to be done on this PRA to satisfy those simple bullets because --

MR. WOODLAN: I don't know that I can say I appreciate it --

CHAIRMAN STETKAR: But I'd just like to kind of alert you to that, that there is quite a bit of work that needs to be done.

MR. WOODLAN: Yes.

CHAIRMAN STETKAR: Otherwise, when you use the PRA for this application, you can draw the wrong conclusions, you know, in both directions. Either it wouldn't be favorable to you in terms of extending allowed outage times for surveillance frequencies, or, in the other direction, it might otherwise too strongly constrain you.

MR. WOODLAN: I believe Luminant, as a company, understands that. First of all, we had quite a few public meetings in order to develop this with the NRC staff, and at one of those meetings was a very lengthy discussion about how that was going to work.

CHAIRMAN STETKAR: That's good. That's --

MR. WOODLAN: And the people that were in the program came out of that meeting and they said you really ought to document that, and we did. In addition to that, you know, we went through a lot of this as we rolled up the maintenance rule on Units 1 and 2, and we had to take our PRA that we had at that time and enhance it to support the maintenance rule and to support what we call the risk rule, risk assessment tool that we use even on Unit 1 now. Everyday, they assess the current risk, and when we go through an outage they assess the risk at various stages. And you had to have a lot of this detail in order to do that.

And we also explored adopting 50.69, which we may do in the future. But one of the challenges we recognized there was we were going to have to take our PRA even to another step, and we haven't really figured out all that stuff yet so we haven't done it yet. But I think that shows that we have looked at this, we know what's going on. We did get a question from the NRC staff about how are you going to do all this, and we answered that with a kind of a very large block plan that showed six or seven of the major activities and how those would be scheduled between licensing and fuel load so that we would have it done.

CHAIRMAN STETKAR: And this was just a curiosity, do you plan at the time of fuel load to actually have in place all of the supporting analyses to allow you to implement, you know, the risk-informed technical specifications? So as soon as you pull rods --

MR. WOODLAN: Yes, that is the plan.

CHAIRMAN STETKAR: Okay.

MEMBER SCHULTZ: I'm glad you're thinking about what opportunities you have with Units 1 and 2 in the analyses that are done there to develop, as you said, lift up that calculational capability to give you even greater capability for the site, rather than to have two different programs that you're trying to manage. You've got some big thinking to do about opportunities in getting two units to four units.

DR. CURRY: In a lot of different areas.

MEMBER SCHULTZ: Yes, in a lot of different areas. Thank you.

MR. WOODLAN: Okay, next slide. Now, this is very similar to the previous slide, except it's for the Surveillance Frequency Control Program. Again, it will be controlled by plant procedures and include things like responsibilities in training. It addresses the PRA and the requirements that we have a current PRA in order to support the program. And, again, I identify the fact that the requirements for the PRA are identified in the methodology. I should really say the methodology, as well as the NEI documents. They have a lot to say about that, too.

Next slide. Okay. Now I'm going to move into some of the material that's addressed right in the

FSAR. In essence, what you have in the FSAR is the updates to the standard plant PRA that are necessary to support the site-specific designs and the site-specific equipment. The PRA is being used, and here's kind of a summary for some of the areas where we're going to apply PRA to the design. Again, like I said, we put that table in the FSAR to make it clear.

The PRA will have to be updated to support the risk-informed tech specs. There will be a program in order to maintain and upgrade the PRA, including the fact that it needs to be upgraded or updated, I'm not sure of the right word, on a periodic basis. I think it's every three years. Four years.

So that's a requirement. But in addition to that requirement, you have to evaluate any ongoing events or changes in the plant to see if something should be upgraded on a more urgent basis instead of waiting for the four years, especially with the way we're using it for things like risk-informed tech specs. And the only component that was added to our list, as we talked about yesterday when we did 17.4, is the vent fans for the ultimate heat sink cooling towers.

CHAIRMAN STETKAR: Don, I've forgotten and I should have looked it up, is there anything special about your switchyard configuration for Units 3 and 4 that would require different treatment than what's in the certified design PRA? I'm thinking about either in terms of frequency of loss of offsite power, reliability of recovery of offsite power, or, as we heard earlier today, I believe they said the switchyard was subdivided into nine fire areas, which sounds like it presumes some sort of configuration.

MR. WOODLAN: I believe the answer to that is no. It's a standard, I think you've seen it, it's a standard dual bus --

CHAIRMAN STETKAR: Well, yes, I've seen it. I just couldn't recall it, and I didn't --

MR. WOODLAN: Yes. And, you know, my thoughts are going back to 1 and 2 because we use a very similar design on 1 and 2, and what I know of the rest of the industry it's typically either the ring bus or the dual bus, and we use the dual bus approach and follow the normal requirements. And I know that, in assessing this, because it has come up on a few things, that we do have to, I think it came up on some of the high-wind evaluations on whether or not you were going to have offsite power because those wooden values are kind of in the range where now you begin to wonder is the wind going to affect offsite power? And so we did evaluate that based on data available for that kind of design, as well as what happens in Texas. MEMBER SCHULTZ: This is on the last slide, Don, on the fourth bullet. Based upon John's comment really before but in addition to everything we've been talking about the last day and a half, on the fourth bullet, wouldn't that need to be evaluated to determine what PRA maintenance and upgrade is required? In other words, I don't think there's any question there would be an upgrade needed in order to support operation, unless you're trying to perform this differently.

MR. WOODLAN: Yes, we do have a --

CHAIRMAN STETKAR: Oh, that's the evolving new --

MR. WOODLAN: Yes. So we're going to have to do the upgrade, and we're going to have to match it to the as-built plant prior to fuel load.

CHAIRMAN STETKAR: I got you.

MR. WOODLAN: And then, after that, there will be an ongoing maintenance activity.

MEMBER SCHULTZ: Good. I understand. Thank you.

MR. WOODLAN: This one, this slide covers several additional items that are in the FSAR. The first one talks about the screening of external events, and this takes up back to Chapter 2. We did look at all the events. The only one that didn't clearly screen out was the tornados. We do have quite a few tornados in Texas. So that one was fully evaluated. The results are on this slide here and are in the FSAR and did not end up -- as you can see, it came out to 8 times 10 to the minus 8. And the tornado, although it ended up being evaluated, it does not have a significant contribution to risk. I believe the FSAR says that it's like less than one percent.

CHAIRMAN STETKAR: Are those scenarios, though, now in the PRA, in your PRA, or haven't you done anything?

MR. WOODLAN: It was a specific assessment.

DR. TANAKA: Oh, so does the question mean --

CHAIRMAN STETKAR: If I asked you what specific plant damage states come out of the tornado analysis, could you tell me those? Plant damage states. I'm talking about Level 2 PRA output frequencies. That's an easy way to answer my question. The answer is either yes or no.

The question was have you actually put this into your PRA, or have you just simply said it has a low frequency so we're not going to quantify it?

DR. TANAKA: We do have the event tree.

CHAIRMAN STETKAR: Okay.

DR. TANAKA: And we calculate the sequence, yes, the frequency of the sequence --

CHAIRMAN STETKAR: So is that now in the PRA for the COL?

DR. TANAKA: Yes.

CHAIRMAN STETKAR: It is. So there's a different model for the COL than for the DCD; is that correct?

DR. TANAKA: Okay. So it's an additional model, additional event --

CHAIRMAN STETKAR: Have you linked it to the Level 2 models?

DR. TANAKA: No.

CHAIRMAN STETKAR: Okay, thank you. That will need to be done.

MR. WOODLAN: In addition, we evaluated the site-specific systems and structures, which are fairly limited, primarily the ultimate heat sink and the portion of the ESWS that's linked in the ultimate heat sink, determined that they did not have a discernible effect on the internal fire, internal flooding, or low-power shutdown PRA results. And I think we've already talked about it, we talked about it yesterday, the need for the peer review, which will be done or be completed within more than one year prior to fuel load, to support our risk-managed tech specs.

Section 19.2 of the FSAR talks about severe accident management. We've talked a little bit about survivability assessment. This portion of the program will fall on us, as the applicant, that when we actually procure the equipment we'll have to confirm it's either already qualified or we'll have to do something to qualify that equipment to the scenarios that are developed primarily by the standard plant.

The accident management programs, which are closely linked or overlaps the emergency planning efforts that are necessary, will need to be in place and we'll need to have training completed for those programs prior to fuel load. Evaluation process for SAMDAs, we talked a little bit on the earlier slides where we did the comparisons and the averted cost calculations, and there were no design-related SAMDAs that were identified as being warranted.

There are two appendices that we, again, reference, we IBR. One of them we currently IBR, the design basis aircraft impact assessment, and one that we will IBR which is the PSMS reliability analysis, which we talked about earlier and I'm sure we'll talk about that when we discuss Chapter 7.

CHAIRMAN STETKAR: Don, I'm sorry. I'm going to be completely ignorant here. I know what the beyond design basis aircraft impact assessment is, and we discussed a little bit about that yesterday. What is the beyond design basis PSMS reliability analysis? I'm not sure that we've run into that before.

MEMBER BROWN: We talked about it in -- I take that back. I'm not so sure they talked about the beyond design basis in the document --

CHAIRMAN STETKAR: That's what -- I understand what a PSMS reliability analysis is, but what is the beyond design basis PSMS? The reason I'm asking is is that it's apparently a different other analysis.

DR. CURRY: I think you're just referencing 19B, right? It's 19B, the PSMS sensitivity studies.

CHAIRMAN STETKAR: Okay, okay, okay. I'm just hanging up on the beyond design basis.

MEMBER BROWN: No, I just remember a statement in one of the things where there was an assumption made some place about the entire PSMS failed, and I didn't know, I'm not even sure I remember where it was right now. So that's what hit me when I saw the beyond design basis, that you would assume the whole thing failed and you had to, you know, manual operator actions for doing whatever you needed. So I don't know. I'll quit talking.

MR. WOODLAN: And this next slide, which is really the last slide, is just to identify the fact that the large area fire is a Chapter 19 review activity. It is a withheld document, and it is part of our COLA, which we have docketed.

MEMBER BLEY: I have a process question. Most of how all this works for the COLA I'm starting to get my arms around, but overviewing the slides yesterday and today and at the bottom, no contentions pending before the ASLB, but you don't have a certified design yet. Am I right in assuming that, once the design is certified, then it will still be possible for people to file contentions?

MR. WOODLAN: Actually, they can file contentions at any time during the application process.

MEMBER BLEY: Up until you get the COL?

MR. WOODLAN: Up until we get the COL.

MEMBER BLEY: Okay. I didn't realize that. I thought there was some time limits.

MR. WOODLAN: And then even after you get a COL, when you file license amendments, they can ask for --

MEMBER BLEY: Yes.

MR. WOODLAN: -- hearings, which is almost the same thing.

MR. MONARQUE: Yes, I can elaborate a little more. There's always a possibility for a member of the public to file a petition for a contention for new information, but the threshold for accepting that is much higher than for when the Federal Register notice was issued in 2009. So there was --

MEMBER BLEY: I guess that's what's curious to me since, in 2009, there wasn't a certified design. How could anybody --

MR. MONARQUE: Well, we issued a Federal Register notice in spring of 2009, I believe, and we gave the members of the public a limited amount of time and ASLB gave them a limited amount of time to file contentions, petitions for contention. And they could still do it now based on new information, but there's a higher threshold for acceptance.

MR. WOODLAN: It would be a late filing, and the rules are tougher for a late filing than if they had filed on time.

MR. MONARQUE: So the answer to your question is yes, but there's a higher threshold.

MEMBER BLEY: Okay.

MEMBER BROWN: Was the Appendix B stuff beyond design basis? You said it's going to be Rev 4.

MR. WOODLAN: It will be Rev 4 for us. Our Rev 4 comes out --

MEMBER BROWN: It's probably not done then. It's not done right now; is that correct?

MR. WOODLAN: Assuming Steve approves my exemption request, Rev 4 will go in in November.

CHAIRMAN STETKAR: Well, but, I mean, in the DCD, Rev 4, it will appear in Rev 4 of the DCD, right?

MR. WOODLAN: Oh, you think it's already in there? That already exists.

CHAIRMAN STETKAR: It exists, but we don't have it yet. When I say we don't have it, I mean I don't have it in the version of the DCD that I have on my computer. That is, in some sense, the royal we. I don't know whether -- does the staff have Rev 4 of the DCD. Not COL FSAR, DCD.

DR. CURRY: Actually, I think we're talking about living DCD, so you may not have that.

CHAIRMAN STETKAR: Yes.

MEMBER SHACK: We have a Rev 3, but it does no, there is no 19B.

CHAIRMAN STETKAR: We got a copy of what's called an interim Rev 4 of Chapter 9 because there were a lot of changes for that, so it's apparently in that.

MR. MONARQUE: The official DCD is coming in July for Rev 4.

CHAIRMAN STETKAR: Then we should expect to see that in Rev 4, the 19B, should we expect to see that appear in Rev 4?

MR. SPRENGEL: Yes, it will be there and it's coming in August, the end of August.

CHAIRMAN STETKAR: Calendar time I've learned --

MR. SPRENGEL: It will be in --

CHAIRMAN STETKAR: The next version we see it will appear in. Okay. I got the answer. That's fine. It's just something obviously that some of us would be interested in looking at, just to kind of organize our timing, because, Charlie, in some sense, you know, we need to be cognizant of whatever that might be when we look at Chapter 7. I mean, it's not design, but whatever they have in there for their reliability assessment might --

MEMBER BROWN: In the Chapter 7 part?

CHAIRMAN STETKAR: It won't be in Chapter 7.

MEMBER BROWN: No, obviously, not based on our last discussions.

CHAIRMAN STETKAR: It certainly won't. Okay. Anything else for Luminant? If not, thank you. Appreciate it. That was pretty painless. And I'll ask the staff to come up, noting that we are now 30 minutes ahead of schedule.

MR. MONARQUE: Okay. My name is Steve Monarque. I want to go ahead and introduce the tech staff. Hanh Phan, Marie POHIDA, and Todd Hilsmeier, and Bob Tjader in support. And with that, we want to present Chapter 19, discussion of the safety evaluation probabilistic risk assessments severe accident safety evaluation.

With that, I'll turn it over to tech staff.

MR. PHAN: Good afternoon, ladies and gentlemen. We are back. We are back to present you our review of the Comanche Peak COL application for Unit 3 and 4, FSAR Chapter 19 PRA and severe accident evaluation. In this presentation, we will cover the COL action items, the open items, and the technical topics of interest.

Next, please. Before going to the technical discussions, we'd like to take a few minutes to present you the approach that we have taken to review the Comanche Peak Chapter 19. First, we verified the applications against the corresponding sections in the DCD to ensure that the combined information of the DCD and the FSAR represents the complete scope of the Chapter 19. Second, we discussed the plant-specific information with other technical branches. We ensured review consistency with other COL applications, and we ensured the review consistency with the analysis documents in COL FSAR Chapter 2, Chapter 3, and Chapter 16.

There are seven open items identified at to review. Next slides, please. With these seven open items 19-4 on external hazards, items 19-5 on high winds, and the last one, items 19-7, on discrepancy in adverted costs between COL FSAR and COL environmental report. These items will be discussed in the technical topics of interest.

Next, please. The first topic is on the external hazards risk evaluation. As described in the COL FSAR, most of the external events were screened out from the PRA using the five preliminary screening criteria of ASME/ANS standard that are endorsed by Reg Guide 1.200. Those identified in the supporting requirements EXT-B2. In Reg Guide 1.200, Section 1.2.5, states that it is recognized that, for those new reactor designs with lower risk profiles, the quantitative screening values should be adjusted according to the relative baseline risk value.

With that, the Comanche Peak Units 3 and 4, from the quantitative perspective, the applicant screens out an event if it can be shown its frequency is less than 1E minus 7 per year. In addition, if an event frequency is higher than 1E minus 7 per year, it can only be screened out if the bounding analysis show that its contribution to the total CDF is insignificant, which means it's less than one percent of the total CDF.

The staff's review found that the list of the external events analyzed in Chapter 19 is consistent with the list of the external hazards identified in the ASME/ANS standard, Appendix 6-1, list of external hazards requiring consideration in the last bullet. However, staff acceptance of the Chapter 19 external hazards will be conditioned, in part, of the completion of the reviews of FSAR Chapter 2 and Chapter 3.

Next, please. With that, I'd like to turn it over to Marie POHIDA. She will talk about the high winds issue.

MS. POHIDA: I just have one slide. Thank you, Hanh. My technical topic of interest was high winds, and that's other than tornados because they could have a greater frequency of occurrence. If you look in Chapter of the FSAR where they discuss meteorology, they give site-specific extreme wind speeds, you know, other than tornados. And it's documented in the FSAR table as 96 miles per hour in one in a hundred years, and that's based on the wind speed maps in this document from the American Society of Civil Engineers. It's minimum design loads for buildings and other structures.

MEMBER BROWN: Does one in a hundred years means it only occurs once in a hundred years? Is that what -- you said 96 miles per hour in one slash a hundred years. Does it only happen once within a hundred years that you have --

MS. POHIDA: As I understand this document, the 100-year wind speed is 96 miles per hour.

MEMBER BROWN: Occurring one time? What's with the one?

MS. POHIDA: I'm sorry?

MEMBER BROWN: I understand 96 miles per hour, but how many times? Does the one have some meaning relative to the 100 years?

MS. POHIDA: It's expected to occur once per a hundred-year period.

MEMBER BROWN: Oh, okay. That's what I --

MS. POHIDA: Oh, I'm sorry. Okay. And then I went and looked, and in Chapter 3.3 of the FSAR, for design wind velocities, it says that none-safety related equipment and structures, including the switchyard, is designed to the site-specific extreme wind speed, which is 96 miles per hour. So what we wanted to, what I wanted to confirm is that, at beyond design basis wind speeds, for instance --

CHAIRMAN STETKAR: Ninety-seven miles an hour.

MS. POHIDA: Well, one in a 125-year wind speed, the one in a 150, the one in 200, that the contribution to risk, you know, from extreme winds is still less than 10 percent for full-power and shutdown. And on reviewing the latest RAI response, it came in in December of 2012, in this latest RAI response I learned that the alternate AC power generators and all supporting equipment will be housed in CAT 1 and CAT 2 structures, and that should help reduce the risk but I'm still evaluating the RAI response.

CHAIRMAN STETKAR: Marie, and just for your information, if you're not aware, we, as the ACRS Subcommittee, have not yet seen either Chapter 2 or Chapter 3 of the FSAR or the DCD. So, you know, we're --

MEMBER BLEY: In a meeting.

CHAIRMAN STETKAR: In a meeting. I mean, you know, we have them, but we haven't actually formally reviewed them. In their RAI response, did they give you a set of wind, high-wind exceedance curves? You know, an exceedance curve that shows frequency of exceedance as a function of wind speed with uncertainty?

MS. POHIDA: No. I'm still reviewing the RAI response. I haven't reviewed it in a while, but no.

CHAIRMAN STETKAR: Okay. Because I'm curious, without that, how they're going to answer your question about, you know, what I hear you saying you're asking.

MS. POHIDA: We've had, for other COL applicants, what they've done is they've basically looked at, you know, the one in a 150-year wind speed, just right at, you know, just at design basis. And what they did is they looked at the conditional core damage probability, assuming that your non-safety related equipment, you know, was inoperable or unavailable to get a sensitivity study to understand the risk. That's what another COL applicant did.

MEMBER BLEY: So they just took out all the non-safety related --

MS. POHIDA: Yes, yes.

CHAIRMAN STETKAR: Essentially, 60 to the minus 2 per year.

MS. POHIDA: So I'm still evaluating this response.

CHAIRMAN STETKAR: Which my guess isn't going to work because a fire in the turbine building generates 53 percent of the fire risk, which is a third of the total core damage frequency. So if you're looking for something that's less than 10 percent, I mean, just do some calculations, you can kind of see where that's going to go because the turbine building fire pretty much takes out secondary --

MS. POHIDA: Oh, okay, okay, okay.

CHAIRMAN STETKAR: -- unless they don't assume that a high wind event can cause those turbine bypass valves to open.

MS. POHIDA: Okay, okay.

CHAIRMAN STETKAR: Be careful when they do those comparisons. If they take stuff out cleanly, it may be different than if they take it out dirty.

MS. POHIDA: Okay.

CHAIRMAN STETKAR: Fires take things out dirty, but I don't know how high winds take things out.

MS. POHIDA: Okay. I appreciate that. Thank you. I'll keep that in mind. Thank you. Okay. Does anybody else have any questions? That was my single slide. Okay. Then I'm going to turn the tables over to Todd.

MEMBER SCHULTZ: Excuse me.

MS. POHIDA: Oh, I'm sorry.

MEMBER SCHULTZ: That's part of the answer, but you're just looking at non-safety equipment that fails --

MS. POHIDA: Because --

MEMBER SCHULTZ: -- at that speed? Because you're talking about an exceedance wind speed, and that's going to have an impact on safety-related equipment, as well. I don't know what you have in terms of information related to capability of safety-related equipment at 150 miles an hour, for example, straight wind speed.

MS. POHIDA: You know, usually, what we've done, we haven't done this very often, for the other COL applicant, what we assumed is is that anything housed in a CAT 1 structure is designed for tornados, and it should survive a tornado and, therefore, it should be fine during an extreme wind event. But, no, we don't have fragilities, and, no, we don't have exceedance frequencies for winds, no. We just had these data points, that single data point. Does that answer your question?

MEMBER SCHULTZ: Yes, it does.

MS. POHIDA: Okay, thank you.

MEMBER SCHULTZ: So you had the opportunity to do the straight calculation?

MS. POHIDA: I'm sorry?

MEMBER SCHULTZ: You had the opportunity to do just the straight calculation associated --

MS. POHIDA: Yes, as a sensitivity. Yes. Okay. All right, Tom. Oh, I'm sorry. Oh, okay, I'm sorry.

MR. PHAN: Before going to slide number nine here, I'd like to say one thing. I apologize for my presentation not clear to you regarding the contribution to fire CDF. The 53 percent of the total fires that contribute from the switchyard in the turbine buildings, the number one contributor to the CDF, fire CDF is the switchyard.

CHAIRMAN STETKAR: Oh, okay.

MR. PHAN: So if you roughly estimate the contribution from the turbine building less than 20 percent --

CHAIRMAN STETKAR: Okay. So it's knocked

down a factor of two, but it's, it's still a fairly high fraction of what is about a third of the total. So it's about a --

MEMBER SCHULTZ: I looked at the chart. That helped somewhat, but it's still important the way it's modeled.

CHAIRMAN STETKAR: Okay. Thanks. That, at least, puts things in a little better numerical perspective.

MR. PHAN: Yes. This topic is on the plant-specific information. In the FSAR, the applicants identified the site-specific and the plant-specific information that has a potential effect on PRA. The plant-specific design and operational changes or departures from the DC as described in the FSAR Tables 1.8-1R, significant site-specific interface with the standard US-APWR designs.

The applicant concluded that all plant-specific changes or deviations listed in this table would have no potential impact on the PRA, except for those related to the access of service water and the ultimate heat sink. The staff reviews the information and issue RAI 19-4, requested the applicants to conduct a systematic search for the site-specific or plant-specific factors, such as LOOP frequency, recovery of offsite powers, maximum ambient temperatures used in the HVAC design calculation.

In their response, the applicant provides the justification for the staff request. In addition, the applicant also provided the results of its reviews of the site-specific interface and the potential impacts on the DC PRA. These are documented in the staff evaluation, Table 19-2.

CHAIRMAN STETKAR: Just out of curiosity, and this is a question for Luminant, actually, have you ever had a loss of offsite power at Units 1 and 2?

MR. WOODLAN: No. We have lost single --

CHAIRMAN STETKAR: Single lines. Okay.

MR. WOODLAN: We've lost one total.

CHAIRMAN STETKAR: Thank you.

MR. PHAN: Slide 10. This topic is on risk-informed tech spec. Luminant requested NRC approval to implement NEI Topical Report 06-09, "Risk-Managed Tech Specs Initiative 4b," and NEI's Topical Report 04-10, "Risk-Informed Tech Spec Initiative 5b, Risk-Informed Method for Control of Surveillance Sequences." It should be noted that the NRC issued its safety evaluations that approve NEI's Topical Report 06-09 and NEI's 04-10 in 2007, in May and September 2007.

Next slide. The issue here is that the applicant is seeking for the approval of the risk-informed tech spec implementation at the COL stage, which means before the developments of the as-built, as-to-be operated plant-specific PRA model. The staff realized that the PRA required per 10 CFR 50.71(h)(1), which will be used to support risk-informed tech spec implementation, will not be available at the time of the COL issuance. In addition, the application-specific infrastructure, such as the procedures, the training, the software, the programs used during the operation, will not be available at the time of the COL's issuance.

With that, the staff has conducted many public meetings on the risk-informed tech spec and are listed on this slide. In the last bullet, on October 20, 2011, the staff provides a presentation to the ACRS on the Comanche Peak Nuclear Power Plant's COL's risk-informed tech spec for reviews of the risk-informed tech spec approach.

Next, please. Based on the staff's findings and based on the discussions, the applicant developed the methodologies referenced in the tech spec that provides the necessary changes to the information in the NEI Topical Reports 06-09 and 04-10 for application to Comanche Peak Units 3 and 4. That report is available in ADAMS, and I have an ML number listed on this slide.

Next, please. To ensure that the PRA is sufficient to support the Comanche Peak risk-informed tech spec, in the response to question 19-3, the applicant committed to update and upgrade the PRA, specifically the second bullet there. The site-specific models will be included in the first series of the PRA upgrade. Emergency operating procedures and detailed design information will be reflected in the PRA during the second series of the PRA upgrade. Uncertainties on PRA models will be identified and addressed during the PRA upgrades, and peer reviews will be performed and the findings will be resolved before the initial fuel load.

In Table 19-1 of the safety evaluation, this table documents the Comanche Peak PRA updates and upgrade activities for the risk-informed tech spec, including the internal events, fire, flooding, seismic, external events Level 2, and the peer review.

Next, please. From the PRA quality perspective, Luminant is committed to its response to the staff question 19-3. The PRA for risk management tech spec must basically meet Capability 2 for the supporting requirements of ASME/ANS internal events at-power PRA standard. The scope of the PRA model would include Level 1 CDF, plus large early release frequency. Contribution from external events, internal flooding events, internal fire events must also be considered. The PRA for RMTS will be updated to satisfy the PRA technical adequacy described in the NEI guideline and will be available one year prior to the fuel load.

CHAIRMAN STETKAR: Can I ask for some clarification? That last sentence, I read that last sentence to say that the PRA will be available one year prior to fuel load. Is that what the applicant is saying?

MR. PHAN: Yes, sir.

CHAIRMAN STETKAR: Okay. Okay. That's a bit, I just wanted to make sure that I was reading that because I'm aware that the PRA before fuel load has to satisfy whatever guidance and standards are in place one year prior to fuel load, so I wanted to make sure that they're actually saying they're going to, the PRA will be ready for prime time one year prior to fuel load.

MEMBER SCHULTZ: But the peer review could follow this. If you go back one slide. No, go back one slide.

CHAIRMAN STETKAR: Well, it says satisfy the PRA technical -- is that right?

MEMBER SCHULTZ: Go to the previous slide for what was said in response to this question, the last bullet.

CHAIRMAN STETKAR: That's interesting.

MEMBER SCHULTZ: So they'll have a PRA, but it may not be peer reviewed until and peer review findings resolved. They've got a year to do that, according to the sequence, if I'm getting that right.

CHAIRMAN STETKAR: Well, but how do I know that it satisfied the technical adequacy in the NEI guideline without that peer review? I mean, what confidence do I have that -- in the next slide says it will be updated to satisfy the PRA technical adequacy described in the NEI guideline, which refers to Reg Guide 1.200.

MEMBER SCHULTZ: So I would turn to Don.

MR. WOODLAN: I mean, I'm looking at the schedule that we provided in response to an RAI when the NRC had similar questions, how you're going to do all this in the times that you have. And the schedule was laid out from fuel load going backwards and the various activities. We currently have planned that the peer review will start about three years prior to fuel load --

CHAIRMAN STETKAR: Okay.

MR. WOODLAN: -- and should be available one year, as complete. That means --

CHAIRMAN STETKAR: So that would be on the next slide, on slide 15. The thing that's delivered one year prior to fuel load will be a peer review --

MR. WOODLAN: Peer review PRA.

CHAIRMAN STETKAR: Great, thank you.

MR. PHAN: Thank you. Next slide, please. With that, I would like to turn over to Todd Hilsmeier.

MR. HILSMEIER: Thank you, Hanh. Information item 19-3.4 states that the probabilistic risk assessment in severe accident evaluation is updated as necessary to assess site-specific information, result of this COL information item. The SAMA analysis, in the SAMA analysis the applicant updated the maximum averted cost for 7-percent and 3-percent discount rates using site-specific information. The applicant concluded from their analysis that there's no cost effective design SAMAs. However, it was not clear to the staff on how the averted costs in the SAMA were determined. Myself and some other staff tried to reproduce the averted costs. There is assumptions made. So we issued RAI 19-23 to clarify how each cost component in the averted costs were computed, and we received a response and we're evaluating that response now. And so that's open item 19-7.

CHAIRMAN STETKAR: The only question I had, and I think we covered it yesterday, it's sort of this nebulous issue of how will the treatment of flooding, for example, in response to the Fukushima lessons learned be closed as far as the COL is concerned? I think we addressed that. You know, it's kind of contingent on the timing of rules, right? Yes, okay.

MR. HILSMEIER: I wasn't there during that part of the presentation.

CHAIRMAN STETKAR: Okay.

MR. MONARQUE: But I'll say we're aware of Fukushima and the implication it may have on the environmental report, as well as --

CHAIRMAN STETKAR: It comes into a little --

MR. MONARQUE: -- chapters.

CHAIRMAN STETKAR: -- a little bit of what Bill was asking about, you know, screening and probabilistic maximum precipitation and how those flooding issues will be resolved. Obviously, seismic --

MR. MONARQUE: Chapter 2 flooding and Chapter 19. And we're waiting for, you know, the staff is developing their review standards, and we have a task force addressing how to move forward on this.

CHAIRMAN STETKAR: Okay.

MR. MONARQUE: Fukushima. Okay.

MEMBER SCHULTZ: Do you have a rough schedule as to when you feel you might close this last bullet?

MR. HILSMEIER: I'm planning to work on it in March because we have the response.

MEMBER SCHULTZ: Within the next few months then?

MR. HILSMEIER: Yes.

MEMBER SCHULTZ: Good, thank you.

MR. HILSMEIER: I need to re-look at Mitsubishi's, MHI's SAMA, SAMDA and compare it to Luminant's SAMA analysis.

MEMBER BLEY: Let me go back to the scheduling talk about the PRA. Can I dig a little further? The peer review is going to start three years prior. When will there be a simulator and all the procedures available and actually operators in training? Will that be before that point in time or after that point in time?

MR. WOODLAN: I don't have that on a chart.

MEMBER BLEY: Oh, you don't. Okay. I thought that might have been on the same time line you were looking at.

MR. WOODLAN: I have to speak from memory, and, unfortunately, Tim Clouser, who gave the presentation yesterday on tech specs, would be the expert for us in this area. But we have schedules laid out similar to what we have for the development of PRA for our procedure development and training of personnel. And as you would imagine, they're very much integrated and overlapping.

MEMBER BLEY: Well, yes, that's why it's troubling. I don't see how a PRA can be finished unless you've already got your procedures and, you know, operators to include in the --

MR. WOODLAN: I mean, you could do it, but to finish it and call it done you've got to have the EOPs and PRAs.

MEMBER BLEY: So there may be some iterations here.

MR. WOODLAN: With the beyond design basis, the SAMGs and those that have to go in there, too. So it will be some iterations as we go, yes.

MEMBER BLEY: Okay. Thanks.

MR. WIEGERT: Ed Wiegert, MNES. The peer review process allows you to review the majority of the technical items against the standard, and you can say that parts were not reviewed and then can be reviewed later. And there's been no clean peer reviews that came out perfect, so there's substantial re-work.

MEMBER BLEY: Okay, fair enough.

CHAIRMAN STETKAR: Are there -- any members have any further questions for the staff? If not, I will thank the staff again very much. I'll ask if there's anyone in the public who has any comments they'd like to make? And if not, thank you very much.

We are not completely finished. What I'd like to do is two things. It's been a long two days. We've covered a lot of topics, but, usually, at the close of a subcommittee meeting, I like to go around the table and ask any of the members if they have any final thoughts or comments that they'd like to make. So I will do that, starting with Joy.

MEMBER REMPE: Oh, okay. Well, I think I highlighted my concerns already, and I can reiterate them if it helps. But, again, I appreciate everybody's presentation and willingness to address questions and provide information to us, especially Todd gets a gold star for yesterday providing me some documents.

I think I've emphasized that I'd really like to see an RAI MHI to see some comparisons of the vessel water level and pressure until you get to the top of the fuel. And the staff and I have set up a tentative time for a meeting to talk about a similar type of activity that might be performed by them and to explore some of the assumptions that might have been changed in MELCOR and to try and address some of the questions that I raised. And so I'm looking forward to that.

I mentioned during the meeting about the instrumentation survivability assessment, and I would really like to see more details on that. Not questioning, unless there is some questions that come up, but also it's just for my history to understand better what has occurred in the past in these types of interactions with the design certification.

And I have said already that, you know, I don't think the debris trap is a problem, but I have, from personal experience in our laboratory, seen cases where people in an experiment will put something in because it seems like a good idea and maybe not fully thought through and we end up cleaning up stuff afterwards that can be kind of pricey. It doesn't seem like it would hurt to put an RAI out and just ask MHI to clarify what the experts' thoughts were on the use of that debris trap and why it is beneficial. I guess, after hearing that the staff said they weren't quite even sure what it looked like, I think it's a good idea.

CHAIRMAN STETKAR: Is that it?

MEMBER REMPE: That's it, sir.

CHAIRMAN STETKAR: Thank you. Mr. Brown, sir?

MEMBER BROWN: There was one question I asked yesterday. It was an information one that was relative to the -- oh, here it is -- the frequency for the calibration checks in COTs, channel operational tests, which are 24 months. And it was relative to the SFCP program to change those periodicities based on the PRAs and stuff. And I just asked what do the plants do today for those particular checks. Was 24 months kind of a standard for that? And it's not something that's relative to the PRA. It was just an information question. So if somehow that can be fed back to Girija just so he can feed it back again. That's just an information item.

The other item I had mentioned was on the Design Reliability Assurance Program, other than all the I&C stuff I mentioned, which I won't try to reiterate, was the MILTAC basic platform not being incorporated, included in the Design Reliability

Assessment Program as a major component within the digital I&C systems, as valves are included for every, I mean, every little piece of thing that's included in all the blacksmith technology systems. That's a fairly critical piece of equipment with a potential for software changes and operating system tweaks and all that other kind of stuff. How do you assure that that maintains its reliability capability?

There's no answer for that right now. It's just it wasn't there, and how is it or not going to be addressed?

MEMBER SHACK: There were two days of very good presentations. I think I've learned a lot. I have no further comments, though.

CHAIRMAN STETKAR: Dr. Bley?

MEMBER BLEY: Yes, I, too, appreciated all the presentations and discussions and comments. In the short time we reviewed the PRA, I think we've communicated with you there's enough little gaps and inconsistencies and problems. I can't say it won't cover what you need for a design cert, but it's a long way from the PRA you need to use for risk management purposes. I still struggle with that, but I guess that's the way we are doing with all of the design certs, so I don't think we have a way out of that. But I just want to make sure people know there's a lot of things that need to get cleaned up before it's a usable tool, but there's a lot of time before you're operating this plant, too. That's all.

CHAIRMAN STETKAR: Steve?

MEMBER SCHULTZ: Yes. Just following on Dennis' thought, I was pleased to hear the schedule that Luminant put forth in terms of what their expectations are and plan is to get into a PRA that will be applicable for operational use. And I believe that recognizes the amount of work that needs to be done and the tie-in with all the pieces that need to be tied together in order to support risk-informed operation here.

The general comment I would make, based on the discussions that we've had over the last two days, is just an emphasis that both Luminant, as well as MHI, pay close attention to what is ongoing with respect to not just the lessons learned but also the changes that are in play with regard to response to Fukushima, both with regard to immediate regulatory changes, as well as what we expect to see in terms of longer-term regulatory changes. A new design should take full advantage of the information that has been learned from Fukushima so that proactive changes can be made in the design phase and then in the pre-operational phase associated with the development and implementation of a new plant.

I really do appreciate the presentations that have been made, and I feel that they demonstrated very thorough work that has been done both by the applicant, the staff, Luminant, associated with all of the activities we've heard about in the last two days. And I do appreciate that both the staff, as well as the applicant, came forward with a lot of good information based on the discussions we had yesterday and this morning and answered a lot of the questions very thoroughly for us. And I'm sure there will be more to follow, so thank you very much.

MEMBER BROWN: John, I did miss one. I made the comment, it was under -- sorry, I didn't have my glasses on. The other thing that wasn't included was the failure modes of the unit bus, which is, if you look at their architecture picture, it's not included in either the PSMS or in the MCR. It's just a line of communication where everything coming from the plant up to the main control room, TSC, and the rest, and any control signal that goes down, it's a connecting piece in between, has no failure analysis or didn't appear to be any failure mode analysis of that bus included in the overall PRA.

CHAIRMAN STETKAR: Okay. Is that it? I don't have anything to add. I've obviously been very vocal over the last few days, and I'm not going to reiterate any of the things I've said. I would like to very much again thank both MHI and Luminant and the staff for covering an awful lot of material. I actually was somewhat concerned that we wouldn't get through all of this, and I really appreciate all of the effort that everybody has put into this.

Again, as always, feedback from MHI, you know, this morning was really, really helpful, I think getting a few things resolved. So I'd like to express my appreciation to everyone for that.

One last thing now we do have to cover, and I'll do this online just so that we have it for the record. We have a full committee meeting right now scheduled for the US-APWR in April. This is the reason I wanted to bring it up is this is our last opportunity as a group with the staff, MHI, and Luminant present to discuss what topics will be covered in that full committee meeting. And the reason I wanted to bring it up is, in my mind, there's some uncertainty about what we should cover.

Now, let me, for everyone's memory, refresh where we are. In the last full committee meeting we had on the APWR was in September of last year, and, at that point, what we've had subcommittee meetings on from that full committee until through today were Chapter 15 for both DCD and COL; Chapter 13, which is conduct of operations for the COL; Chapter 4, fuels for DCD and COL with two of the three topical reports for that chapter. We've not yet reviewed the Fines Topical Report. And then the material that we covered in the last two days, so, basically, 17, 19 for both DCD and COL and Chapter 16 for the COL.

What I'd like, a little bit of feedback, from the subcommittee members in particular, because of Chapters 15 and 4 is I think it's important -- well, first of all, should we have the full committee briefing, given where we are?

MEMBER SHACK: I'm sure we've got pieces of those --

CHAIRMAN STETKAR: Well, the only thing we have complete is what we basically covered over the last two days. That's sort of complete.

MEMBER SHACK: Right. We're done.

CHAIRMAN STETKAR: The reason I wanted to discuss this is Chapters 15, we're in sort of an interim state, but this is all interim. And Chapter 4 we don't have quite everything but some open questions. So let me just ask should we have the full committee meeting in April, given what we have available?

MEMBER BLEY: Well, the purpose of a full committee meeting, as opposed a letter, seems to be just to give a little status report before we sit along the way.

CHAIRMAN STETKAR: Exactly.

MEMBER BLEY: I don't think we have any showstopper pressing issues that we've got to get on the table. It could be more complete if you wait, but it just seems, it's not unreasonable to go ahead and have one and make a status report.

CHAIRMAN STETKAR: At this stage in the review, that's exactly right. The purpose of a full committee meeting is, basically, to brief the rest of the committee, ACRS, on what has been covered over the last however many months it is here, seven or so, and if the committee feels, the full committee feels that there are any, as Dennis characterized it, showstopper issues that we feel rise to a level that we want to formally communicate to the staff, that's our opportunity. And if there isn't, that's fine. If we don't think there are any, that's also valuable information to the staff.

So I'm inclined to agree with Dennis that it's useful to have the meeting. The question is should we cover, you know, should that meeting basically cover all of the chapters that we have?

MEMBER SHACK: Status report on 4 and 15--

CHAIRMAN STETKAR: Status report on 4 and 15.

MEMBER SHACK: Kind of a little more complete.

CHAIRMAN STETKAR: Right.

MR. SHUKLA: John, staff has already indicated that Chapter 13 for COLA also could come --

CHAIRMAN STETKAR: Yes, I mentioned that. It's in here. We had a meeting on that already.

MR. SHUKLA: Right. And there are five topical reports.

CHAIRMAN STETKAR: Right. Those I kind of cover under the associated chapters because those topical reports are -- the other topical report that we have not reviewed is the Fines methodology. It's 7034. That was, it was coming in in January, and then it got pushed off. So in terms of topical reports associated with either Chapter 4 or Chapter 15, the 7034, we've yet to see that. You know, we can address that.

MEMBER SCHULTZ: John, I think it's more

important to brief the full committee on a timely basis, as we've developed the intermediate reviews, and indicate what has been done and what has not been done, what still needs review, and the staff can identify that, along with the applicant. It would be very helpful for the full committee to get that briefing.

CHAIRMAN STETKAR: All right. Then let's -- what I'm hearing is we'll plan on, as scheduled, having the full committee briefing in the April full committee meeting. And we'll cover everything that I mentioned, which I'll reiterate it so everybody can take notes. For the DCD, it will be Chapters 4, 15, 17, and 19, Topical Reports 7008, 7009, 7010, 7011, and 7013. We don't need presentations on every last bit of that detail but just to make sure what we're talking about. And for the COL, it will be Chapters 4, 13, 15, 16, 17, and 19, and that will include, if you want to say anything, the large loss of area which you may not want to say anything if we have to close the meeting for that. You know, be careful. There wasn't much material there, but, in principle, that's covered also.

That's a lot of material to cover. Our schedule right now for the April full committee meeting is a little bit light, so we can probably allocate two or maybe two and a half hours. We'll have to work that out in our planning process. So we're not very time constrained at the moment, but I'll urge everyone to try to identify significant issues from your own notes that you feel is important to present to the full committee, not just tables of open items but discuss things that have come up, you know, either in the interactions between the applicants and the staff or things that have come up, in addition, in our discussions at the subcommittee meetings. I haven't put together kind of a list, a hit list myself yet, but I'd like to make sure that we address thorny issues and not just address programmatic lists of tables of RAIs. So I'd encourage you all to keep that in mind. And unless anyone has any --

MEMBER REMPE: I have a question. It's related to concerns for Dr. Banerjee, not myself, of course. But there are some questions that he and I transmitted to MHI. Will we have results from those or responses to those prior to this April meeting?

MR. SPRENGEL: I think so because I think they're coming in end of March. My intention would be, at the meeting, to more focus on the update portion, rather than the resolution, because you would have just received them.

CHAIRMAN STETKAR: That's right. Thank

you.

MEMBER REMPE: Okay. So you said let's focus on thornier issues or issues that might be questions, and so should we say, yes, we had these questions?

CHAIRMAN STETKAR: The purpose of the full committee briefing is to bring the entire committee up to date on where the whole process is, not from a process standpoint but from a technical issues standpoint. And if we, as a subcommittee, and, more importantly, the full committee feel that any of the technical issues that have been discussed, you know, in the subcommittee meetings rise to a level that we want to formally alert the staff to those issues, that's our opportunity to do that as a full committee.

MEMBER REMPE: So it may be we identified these issues, and we're evaluating --

CHAIRMAN STETKAR: These are, these are -- I'm not going to write the letter. I don't --

MEMBER REMPE: I know. But I just --

CHAIRMAN STETKAR: Not all of it.

MEMBER REMPE: That could be a response --

CHAIRMAN STETKAR: You know, Dr. Banerjee is one of 13. This is a full committee meeting, so, you know, the full committee engages.

MEMBER REMPE: What I'm trying to get to is that we may not have some of these things --

CHAIRMAN STETKAR: I understand that. We don't have answers to a very large number of questions. The issue is not do we have answers to the questions. Do we feel, from where we are, that anything that we don't have issues could be really, really significant? That's what we're trying to communicate in these interim letters. If we have a lot of questions that need answering before the final SER with no open items is issued, and the ACRS blesses that process, you know, we will get answers to those questions. We have another chance at writing a letter later on. This is just to, essentially, alert the staff and management, the Commission --

MEMBER BLEY: If there's anything you think we aren't going to get a response on that's going to be really important, then we ought to --

CHAIRMAN STETKAR: Then we absolutely must put it in our letter. It is incumbent on us to alert everyone that we feel that it could be a, we don't see a path to resolution, essentially. Okay? Any other questions? With not -- yes?

MR. PHAN: Just one last thing. On behalf of the technical staff, we would like to thank the ACRS members, all of you, for the opportunities for us to share with you our review and findings on US-APWR. Thank you very much.

CHAIRMAN STETKAR: Thank you. And with that, Texans, go run for your airplanes. We will adjourn.

(Whereupon, the foregoing matter was concluded at 4:09 p.m.)

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Protecting People and the Environment

Presentation to the ACRS Subcommittee

US-APWR Design Certification Application Review

Safety Evaluation Report with Open Items

Chapter 19: PROBABILISTIC RISK ASSESSMENT & SEVERE ACCIDENT EVALUATION

February 21-22, 2013

Staff Review Team

- **Technical Staff**
  - **Hanh Phan** (Lead), Senior PRA Analyst
    PRA and Severe Accidents Branch
  - **Marie Pohida,** Senior PRA Analyst
    PRA and Severe Accidents Branch
  - **Todd Hilsmeier,** PRA Analyst
    PRA and Severe Accidents Branch
- **Project Managers**
  - **Jeff Ciocco**
  - **Ruth Reyes**

Presentation Outline

Chapter 19 - Probabilistic Risk Assessment and Severe Accident Evaluation

1) PRA Quality
2) Internal Events PRA At-Power

   Internal Fires PRA At-Power

   Internal Floods PRA At-Power

   External Events Risk Evaluation
3) Low-Power and Shutdown PRA
4) Level 2 PRA
5) Severe Accident Evaluation

Review Approach

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- Received training on US-APWR design
- Developed initial risk insights to support other technical branches
- Discussed US-APWR designs with other technical branches
- Performed PRA audit at MHI's facility and participated in many public technical discussions
- Ensured review consistency with other design certifications
- Performed audit/confirmatory calculations for assessment of specific severe accident/Level 2 PRA issues
- Reviewed the application in accordance with requirements (10 CFR Part 52), Commission's goals, SRP, PRA standard

Description of SE Open Items

- **Open Item 19.1-LEVEL1-574 * (RAI 898-6275, Questions 19-507, 19-509, 19-559, and 19-564) -** Completion of COL information items provided in Section 19.3

- **Open Item 19.1-LEVEL1-512 * (RAI 750-5675, Question 19-512) -** Identification and documentation of important design features in DCD Table 19.1-119

- **Open Item 19.1-LEVEL1-513 * (RAI 40-610, Questions 19-97 and 19-98, RAI 423-2710, Question 19-364) -** Systematic investigation to demonstrate the robustness of the assumed PRA success criteria for all "success" sequences

- **Open Item 19.1-LEVEL1-514 (RAI 750-5675, Question 19-514) -** Operator action to equalize primary and secondary pressures

- **Open Item 19.1-LEVEL1-515 * (RAI 750-5675, Question 19-515) -** Treatment of I&C hardware and software CCFs

(* Open items will be discussed in Technical Topics of Interest)

Description of SE Open Items (continued)

- **Open Item 19.1-LEVEL1-516 (RAI 750-5675, Question 516) -** Modeling of HVAC failures in the PRA

- **Open Item 19.1-LEVEL1-575 * (RAI 967-6790, Question 19-575) -** Verification of PRA technical adequacy in accordance with PRA standards

- **Open Item 19.1-FIRE-573 (RAI 967-6790, Question 19-573) -** Transferring of plant control from MCR to RSC should be included in DCD Table 19.1-119

- **Open Item 19.1-LEVEL2-560 * (RAI 871-6121, Question 19-560) -** Address hydrogen build-up in the RWSP

- **Open Item 19.1-LPSD-546 (RAI 783-5855, Question 19-546) -** Impact on LPSD risk should a COL applicant decide to drain RCS in POS 4-1

Description of SE Open Items (continued)

- **Open Item 19.1-LPSD-565 * (RAI 899-6281, Question 19-565) -** Omission of LOCAs during POSs 5, 6, and 7 from the LPSD PRA

- **Open Item 19.1-LPSD-506 (RAI 749-5651, Question 19-506) -** Failure of the SG nozzle dams due to a postulated RCS re-pressurization

- **Open Item 19.1-LPSD-495 * (RAI 681-5257, Question 19-495) -** Auto-isolation function of RCS letdown on low hot leg level and prevention of vortexing in the hot leg

- **Open Item 19.1-LPSD-568 * (RAI 924-6352, Question 19-568) -** Auto-isolation of letdown on low hot leg level, manual isolation of letdown, and RCS hot leg indication which affects the calculated OVDR and FLML frequency

- **Open Item 19.1-LPSD-494 * (RAI 669-5219, Question 19-494) -** Lack of shutdown technical specifications

Description of SE Open Items (continued)

- **Open Item 19.1-LPSD-567 (RAI 899-6281, Question 19-567) -** Lack of automation for standby RCS injection given the risk significance of manual RCS injection
- **Open Item 19.1-LPSD-570 (RAI 924-6352, Question 19-570) -** Key sources of uncertainty and key assumptions identified in the LPSD PRA
- **Open Item 19.1-LPSD-566 * (RAI 899-6281, Question 19-566) -** Information regarding containment closure consistent with staff guidance in GL 88-17
- **Open Item 19.1-LPSD-66 (RAI 39-548, Question 19-66) -** Risk insight to be added to the risk insights Table
- **Open Item 19.2-SE-569 * (RAI 924-6352, Question 19-569) -** Clarify whether operability of the hydrogen igniters and other severe accident design features are necessary for the containment to remain intact

Technical Topics of Interest
Quality of PRA

- **PRA Scope**
  - Level 1 PRA and Level 2 PRA for internal events (including internal floods and internal fires) at power and shutdown conditions
  - PRA-based SMA
  - Other external events (i.e., high winds, external floods, external fire, etc.) will be addressed by the COL applicant
- **Level of Detail**
  - DCD Section 19.1.2.4, “PRA Maintenance and Upgrade,” states that the PRA is placed under configuration control in accordance with PRA Standard ASME/ANS RA-Sc-2009

Technical Topics of Interest
Quality of PRA (continued)

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- **PRA Technical Adequacy**
  - **Open Item 19.1-LEVEL1-575, Question 19-575 -** Clarify the following statements:
    - *“The PRA has been developed in accordance with industry consensus standards as described in Section 19.0.”*
    - *“The PRA … has been subjected to a peer review process as defined in ASME/ANS RA-S-2008 and associated addenda.”*
  - Self assessment or in-house review regarding PRA technical adequacy is needed

Technical Topics of Interest
Internal Events PRA At-Power

- **Outline**
  - Documentation of Key Sources of Uncertainty, Insights, and Assumptions
  - Asymmetric Configuration
  - Digital I&C
  - Sensitivity Studies

Technical Topics of Interest

Documentation of Key Sources of Uncertainty, Insights, and Assumptions

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- DCD Table 19.1-38 – “Key Sources of Uncertainty and Key Assumptions”
- DCD Table 19.1-119 – “Key Insights and Assumptions”
- Provide key PRA insights and assumptions related to design and operational features with an appropriate disposition
- Ensure the assumptions made in the PRA will remain valid

Technical Topics of Interest
Asymmetric Configuration

- Asymmetric conditions due to modeling simplicity have been taken into consideration when reporting PRA results and insights (e.g., providing input to D-RAP)
- PRA will be upgraded before the implementation of risk-informed applications, to ensure that asymmetric conditions due to modeling simplicity are addressed
- **Open Item 19.1-LEVEL1-574** - COL Information Item 19.3(1) must be modified to ensure that asymmetric conditions due to modeling simplicity will be addressed or properly accounted when the PRA is used for decision making

Technical Topics of Interest
Digital I&C

- PRA Attachment 6A.13 “Engineered Safety Feature Actuation System”
- I&C fault trees were revised to address hardware CCF, application software common mode failure between parallel controllers, dependency between automatic and manual actuation signals, application software diversity, and to include several other failures (e.g., input module power supply, communication between RPS trains, digital part of power interface module)
- **Basic software CCF** - The operating system MELTAC provides basic functions for the application software. MELTAC has been used in Japanese nuclear industry over 20 million hours with no CCF - 1.0E-07/d
- **Application software CCF** - Generates the S-signal (ECCS actuation signal) and P-signal (containment spray actuation signal) - 1.0E-05/d
- **Hardware CCF** - Failure of all digital systems that use the same hardware - 2.1E-06/d

Technical Topics of Interest
Digital I&C (continued)

- Diverse actuation system (DAS) - Installed as a counter-measure against CCF of I&C software. Consists of conventional equipment that is totally diverse and independent from the MELTAC platform - 1E-2
- CCF of I&C software and hardware and DAS are identified as a significant source of uncertainty in the DCD (included in DCD Tables 19.1-38 & 19.1-119)
- **Open Item 19.1-LEVEL1-515 -** Provide the definition of I&C hardware and software CCFs modeled in the PRA, in terms of diversity assumptions and what signals are impacted by the failure
- **Open Item 19.1-LEVEL1-512 (Resolved) -** Include in the DCD "Key Insights and Assumptions," the assumptions made regarding hardware and software diversity along with the appropriate disposition

Technical Topics of Interest
Sensitivity Studies

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- DCD Table 19.1-140 “Impact on PRA Associated with Key Sources of Uncertainty and Key Assumptions”
  - Extent of “On-line” Maintenance
  - Human Error Probabilities
  - Digital I&C Software Reliability
  - TS Requirements for I&C Systems
  - GTG Reliability
  - EFW Pit Capacity
  - Operation of EFW Pump Discharge Tie-Line Valves
  - CCF of Sump Screens
  - Test Intervals of Valves
  - Others in support of RAI responses

Technical Topics of Interest
Sensitivity Studies (continued)

The important insights gained from sensitivity studies:

- CDF is sensitive to several CCF probabilities (divisional separation, diversity of redundant components)
- CDF is not very sensitive to an increase in single component failure probability or initiating event frequency
- CDF is not significantly sensitive to further reduction in safety system outage time for test and maintenance
- CDF is not significantly sensitive to further plausible reduction in human error probabilities

Technical Topics of Interest
Internal Fires PRA At-Power

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- **Outline**
  - Fire Protection Concept
  - Use of NUREG/CR-6850, "Fire PRA Methodology for Nuclear Power Facilities" and PRA documentation
  - Major Assumptions
  - Fire PRA Insights

Technical Topics of Interest

Fire Protection Concept

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- Each of four safety divisions is separated by a physical fire barrier to protect its safety function and to prevent fire propagation
- Safety-related equipment and cables are separated using three-hour fire-rated protection
- Qualified cables

Technical Topics of Interest
NUREG/CR-6850 Methodology

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Task	Description
1	Plant Boundary Definition and Partitioning
2	Fire PRA Component Selection
3	Fire PRA Cable Selection
4	Qualitative Screening
5	Plant Fire-Induced Risk Model
6	Fire Ignition Frequency
7	Quantitative Screening
8	Scoping Fire Modeling
9	Detailed Circuit Failure Analysis
10	Circuit Failure Mode Likelihood Analysis
11	Detailed Fire Modeling
12	Post-Fire HRA
13	Seismic Fire Interactions
14	Fire Risk Quantification
15	Uncertainty and Sensitivity Analyses

Technical Topics of Interest
PRA Documentation

File #	Index	Description
163	23	CHAPTER 23 INTERNAL FIRE RISK ASSESSMENT
164	23A0	CONTENTS : INTERNAL FIRE PRA RESULTS
165	23A	LIST OF FIRE AREAS AND FIRE COMPARTMENTS
166	23B	LIST OF FIRE PRA COMPONENTS
167	23C	FAULT TREE MODEL FOR INITIATING EVENTS
168	23D	MATRIX OF FIRE PRA COMPONENTS DEPENDENCIES
169	23E	LIST AND DRAWING OF CABLE ROUTES OF FIRE PRA COMPONENTS
170	23F	LIST OF FIRE PRA COMPONENTS IN FIRE COMPARTMENT
171	23G	LIST OF FIRE PRA CABLES IN FIRE COMPARTMENT
172	23H	FIRE PRA MODEL AT POWER
173	23J	LIST OF FIRE FREQUENCY FOR FIRE COMPARTMENT
174	23K	FIRE-INDUCED CIRCUIT FAILURE ANALYSIS
175	23L	SPURIOUS ACTUATION OF FIRE PRA COMPONENTS
176	23M	SINGLE COMPARTMENT FIRE SCENARIOS
177	23N	FIRE SCENARIOS AND FIRE-INDUCED CIRCUIT FAILURE ANALYSIS
178	23P	MULTIPLE COMPARTMENT FIRE SCENARIO
179	23Q	INSIDE CONTAINMENT FIRE SCENARIO CFAST ANALYSIS
180	23R	ANALYSIS AND DATA OF FIRE PRA AT POWER (LEVEL 1)
181	23S	ANALYSIS AND DATA OF FIRE PRA AT POWER (LEVEL 2)
182	23T	FIRE PRA MODEL AT SHUTDOWN
183	23U	ANALYSIS AND DATA OF FIRE PRA AT SHUTDOWN

Technical Topics of Interest
Major Assumptions

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- SER Section 19.1.4.5.2.1.1 “Major Assumptions” - 30 key assumptions
- Table 19.1-119 “Key Insights and Assumptions”
- COL information item would ensure that the key assumptions will remain valid for the as-built, as-operated plant
- DCD Section 19.1.2.4, “PRA Maintenance and Upgrade” - Any changes to the assumptions relevant to the internal fire events will be incorporated into the PRA as part of PRA maintenance process

Technical Topics of Interest

Internal Fires PRA Insights

- No credit is taken for mitigation function of fire detection/suppression and fire brigade.
- The most significant fire scenarios are: (1) LOOP due to switchyard fire (highest CCDP), and (2) turbine-bypass valve spurious open (SLBO) due to T/B compartment FA6-101-01 fire (large amount of combustible materials), contributing about 53% of the total fire CDF.
- The cable hot short probability was conservatively set to 1.0. The sensitivity study showed that the contribution from hot short is low.
- All fire compartments, except for the containment and switchyard, to be composed of fire resistant wall, floor, and ceiling, therefore, all four ESF trains are separated individually.

Technical Topics of Interest

Internal Fires PRA Insights (continued)

- The internal fires PRA identifies no significant multi-compartment fire scenarios
- A fire in any fire compartment in the containment would not spread to the adjacent compartments as demonstrated by CFAST simulation
- Electrical room in T/B is separated into two compartments resulting in a reduction of fire risk
- Operator actions at RSC during MCR evacuation are the only new actions added to fire PRA. A sensitivity analysis, assuming a failure probability of 1.0, showed an increase of twice the base fire CDF.
- Significant operator action relevant to fire event is the connection of Class 1E bus to the AAC in case of all four Class 1E GTGs unavailable

Technical Topics of Interest

Internal Floods PRA At-Power

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- **Outline**
  - Flood Protection Concept
  - Methodology and PRA Documentation
  - Major Assumptions
  - Insights

Technical Topics of Interest

Flood Protection Concept

- Prevent the flood propagation to multiple mitigation systems (more than two out of four trains) by:
  - Separation of R/B into two areas of east and west sides
  - Installation of water-tight doors for the safety-related SSC areas, safety-related I&C rooms, and main control room
  - Isolation of essential service water pump to prevent inflow into R/B
- Prevent inflow to R/B from adjoining buildings (i.e., T/B and A/B) by installation of water-tight doors
- Install flood relief panels on T/B exterior walls to drain flood water from the circulating water system to the yard

Technical Topics of Interest

Internal Floods PRA Methodology

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Internal Floods PRA includes both qualitative and quantitative analyses

<u>Qualitative Analysis</u>

Step 1 – Identify independent flood areas and SSCs

Step 2 – Identify flood sources and flood mechanisms

Step 3 – Perform plant walkdown (alternately, a table-top examination has been performed at DC stage)

Step 4 – Perform qualitative screening by considering flood sources and modes, and flood propagation pathways

Technical Topics of Interest
Methodology (continued)

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Quantitative Analysis

Step 1 – Develop flood scenarios for each flood source

Step 2 – Perform flood-induced initiating events analysis

Step 3 – Evaluate the impact on equipment, including failures by submergence, spray, jet impingement, pipe whip, humidity, condensation, and temperature

Step 4 – Evaluate flood mitigation strategies and perform human reliability analysis

Step 5 – Develop probabilistic risk model

Step 6 – Quantify flood-induced accident sequences

Technical Topics of Interest
Internal Floods PRA Documentation

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File #	Index	Description
157	22	CHAPTER 22 INTERNAL FLOOD RISK ASSESSMENT
158	22A	INTERNAL FLOOD AND FLOOD PROPAGATION SCENARIOS
159	22B	CONDITIONAL CORE DAMAGE PROBABILITY OF INTERNAL FLOOD AT POWER
160	22C	CONDITIONAL CORE DAMAGE PROBABILITY OF INTERNAL FLOOD AT LPSD
161	22D	ANALYSYS DATA OF INTERNAL FLOOD PRA AT POWER (LEVEL 2)
162	22E	PROCESS ON HOW TO CALCULATE THE INTERNAL FLOOD SCENARIOS

Technical Topics of Interest
Major Assumptions

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- SER Section 19.1.4.5.3.1.2, “Major Assumptions” - 37 key assumptions
- COL information item would ensure that the key assumptions will remain valid for the as-built, as-operated plant
- DCD Section 19.1.2.4, “PRA Maintenance and Upgrade” - Any changes to the assumptions relevant to the internal flood event will be incorporated into the PRA as part of the PRA maintenance process

Technical Topics of Interest
Internal Floods PRA Insights

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- The most significant areas are:
  - Second floor corridors (FA2-321-01 and FA2-320-01) of R/B where EFW piping is located (assumed to propagate to lower areas in the R/B east or west side and fail two safety-related systems)
  - SG radiation monitor room (FA2-507-02) and T/D EFW pump rooms (FA2-102-01 and FA2-108-01) (due to numerous water sources and potential failure of two safety-related systems)
- The most significant systems contributing to internal flood frequencies are: emergency feedwater system, main feedwater system, main steam system, and circulating water system
- The most significant system contributing to internal flood risk is EFW
- The most significant operator action contributing to internal flood risk is to perform EFW switching

Technical Topics of Interest

External Events Risk Evaluation

- Staff's evaluation of seismic risk will be provided later
- Site-specific external events (i.e., high winds, external flooding, etc.,) will be addressed by COL applicant
- COL Information Item 19.3(4):

  "The Probabilistic Risk Assessment and Severe Accident Evaluation is updated as necessary to assess specific site information and all associated potential site-specific external hazards (both natural and man-made hazards) that may affect the facility are screened out or subjected to analysis."

Technical Topics of Interest
Low-Power and Shutdown (LPSD) PRA

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- **Outline**
  - Shutdown TS in Modes 5 and 6
  - Containment Closure
  - Omission of Draindown Events during POSs 5, 6, and 7
  - Auto-Isolation of Letdown & Initiation of Vortexing in Hotleg
  - Hotleg Level Instrumentation

Technical Topics of Interest

Shutdown TS (Open Item 19.1-LPSD-494)

- According to 10 CFR 50.36(c)(2)
  - (ii) A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria.
  - (D) *Criterion 4.* A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Technical Topics of Interest
Shutdown TS (continued)

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- No TS for standby RCS injection and containment closure during reduced inventory operation
- Without RCS injection and containment closure, Commission's goals may be exceeded
- Containment closure not credited in PRA
- Standby RCS injection and containment closure before RCS boiling during reduced inventory operations identified as expeditious actions in Generic Letter 88-17
- MHI agreed lack of safety injection (SI) did not meet Commission's goals
- MHI proposed administrative controls in lieu of TS
- Staff concludes options for TS LCO(s) for SI and containment closure required under 10 CFR 50.36(c)(2)(ii)(D), Criterion 4

Technical Topics of Interest

Containment Closure
(Open Item 19.1-LPSD-566)

- Staff evaluating whether manual actions for containment re-closure before boiling feasible
- Staff questions whether igniters needed (RAI 19-569) to keep containment intact once closed
- MHI will implement a design change to use AACs to power the equipment hatch hoist in addition to offsite power

Technical Topics of Interest
Omission of Draindown Events
Open Item 19.1-LPSD-565

- Draindown events when refueling cavity flooded omitted from the PRA
- Staff concerns regarding draindown events especially when temporary fuel racks in refueling cavity used
- Staff evaluating potential drain down paths and availability of level indication/alarms when these temporary fuel racks used

Technical Topics of Interest

Auto-Isolation of Letdown & Initiation of Vortexing (Open Item 19.1-LPSD-495)

- Overdrain Frequency (OVDR) = 3.7E-6/yr POS 8-1
- Failure to Maintain Level (FLML) = 5.7E-7/yr
- Auto-isolation function risk significant
- Staff concerns regarding auto-isolation of letdown setpoint versus initiation of vortexing in hotleg for highest anticipated operational RHR flow rate
- No indication of RHR pump motor amperage (RAI 19-568)
- Issue being resolved as part of Chapter 5 review

Technical Topics of Interest

Hotleg level Instrumentation (RAI 19-568)

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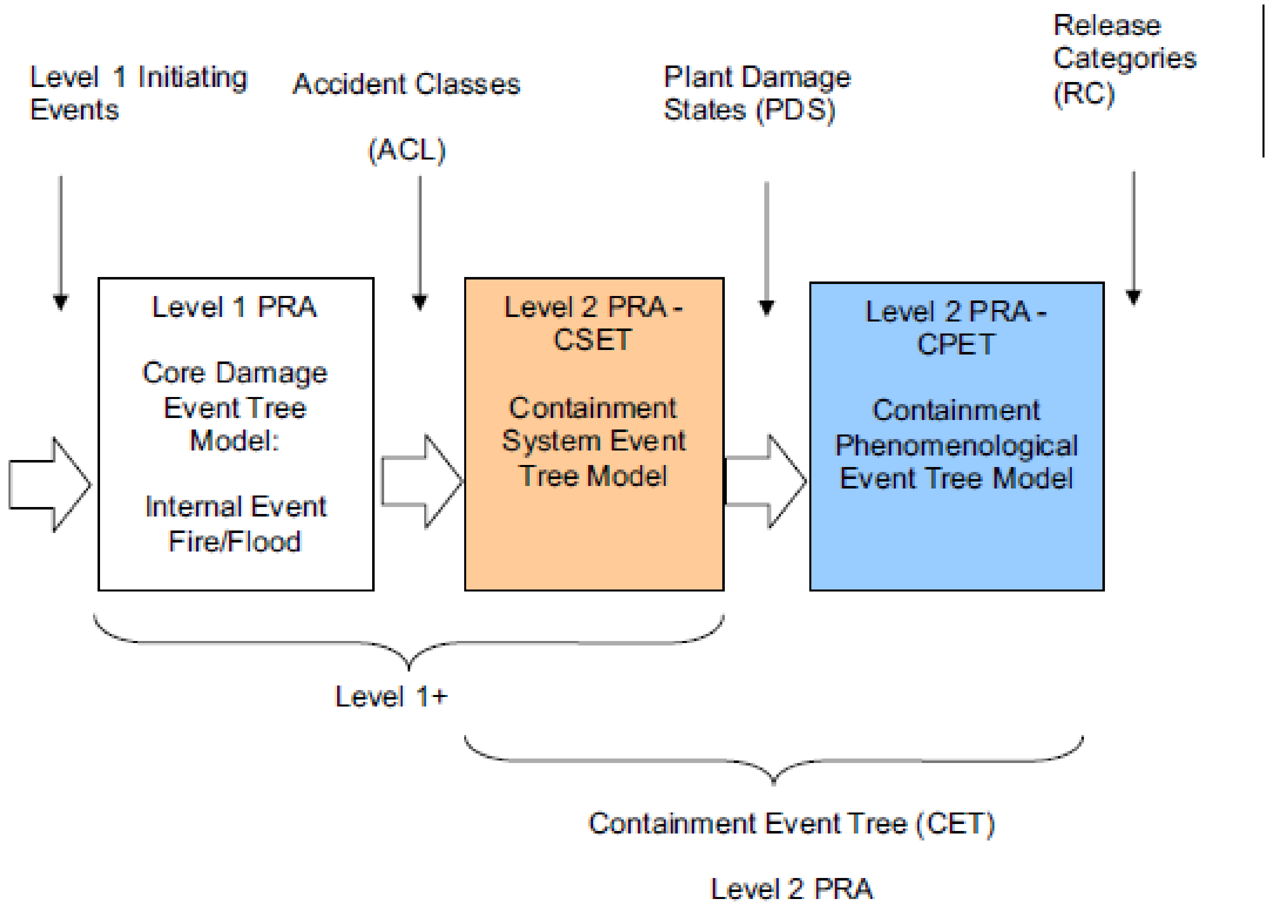
- Failure of operator to start RCS injection is risk significant
- No automated RCS injection
- Failure probability of RCS injection (charging and SI) by operator approximately 1E-4
- Hot leg level indication not safety-related
- Staff's concern regarding validity of level indication during RCS boiling

Outline

- Overview of Level 2 PRA and Severe Accident (SA) Evaluation
- Technical Topics of Interest
  - Ex-Vessel Steam Explosion
  - Hydrogen Generation and Control
  - Core Debris Coolability
  - Risk Metrics

Level 2 PRA and SA Evaluation

Overview of Level 2 PRA and SA Evaluation



Overview of Level 2 PRA and SA Evaluation (continued)

- Containment System Event Tree (CSET) for SBO and other Accident Classes

CSET	CSET	CSET	CSET	CSET		
Containment isolation	RCS depressurization	Hydrogen control	Reactor cavity flooding	Recovery CSS and CS/RHR HX(SBO)	No.	PDS
CI	FD	IG	CF	RSB		

CSET	CSET	CSET	CSET		
Containment isolation	RCS depressurization	Hydrogen control	Reactor cavity flooding	No.	PDS
CI	FD	IG	CF		

Level 2 PRA and SA Evaluation

Overview of Level 2 PRA and SA Evaluation (continued)

- Plant damage states

Primary System Pressure	Reactor Cavity Flooding Status		C/V Isolated								C/V not Isolated		C/V failure before core damage	C/V Bypass (SGTR)
			Igniter Functional				Igniter not Functional							
			CSS Injected		CSS not Injected		CSS Injected		CSS not Injected		CSS Injected	CSS not Injected		
			C/V Cooled	C/V not Cooled	C/V Cooled	C/V not Cooled	C/V Cooled	C/V not Cooled	C/V Cooled	C/V not Cooled				
			A	B	C	D	E	F	G	H	I	J	K	L
Low	Not Flooded	1	NA	NA	1C	1D	NA	NA	1G	1H	NA	1J	1K	NA
	Flooded after RV Failure	2	2A	2B	2C	2D	2E	2F	2G	2H	2I	2J		
	Flooded before RV Failure	3	3A	3B	3C	3D	3E	3F	3G	3H				
Medium	Not Flooded	4	NA	NA	4C	4D	NA	NA	4G	4H	NA	4J	4K	4L
	Flooded after RV Failure	5	5A	5B	5C	5D	5E	5F	5G	5H	5I	5J		
	Flooded before RV Failure	6	6A	6B	6C	6D	6E	6F	6G	6H				
High	Not Flooded	7	NA	NA	7C	7D	NA	NA	7G	7H	NA	7J	NA	NA
	Flooded after RV Failure	8	8A	8B	8C	8D	8E	8F	8G	8H	8I	8J		
	Flooded before RV Failure	9	9A	9B	9C	9D	9E	9F	9G	9H				

Number of PDS= 72

-NA means combination has no possibility.

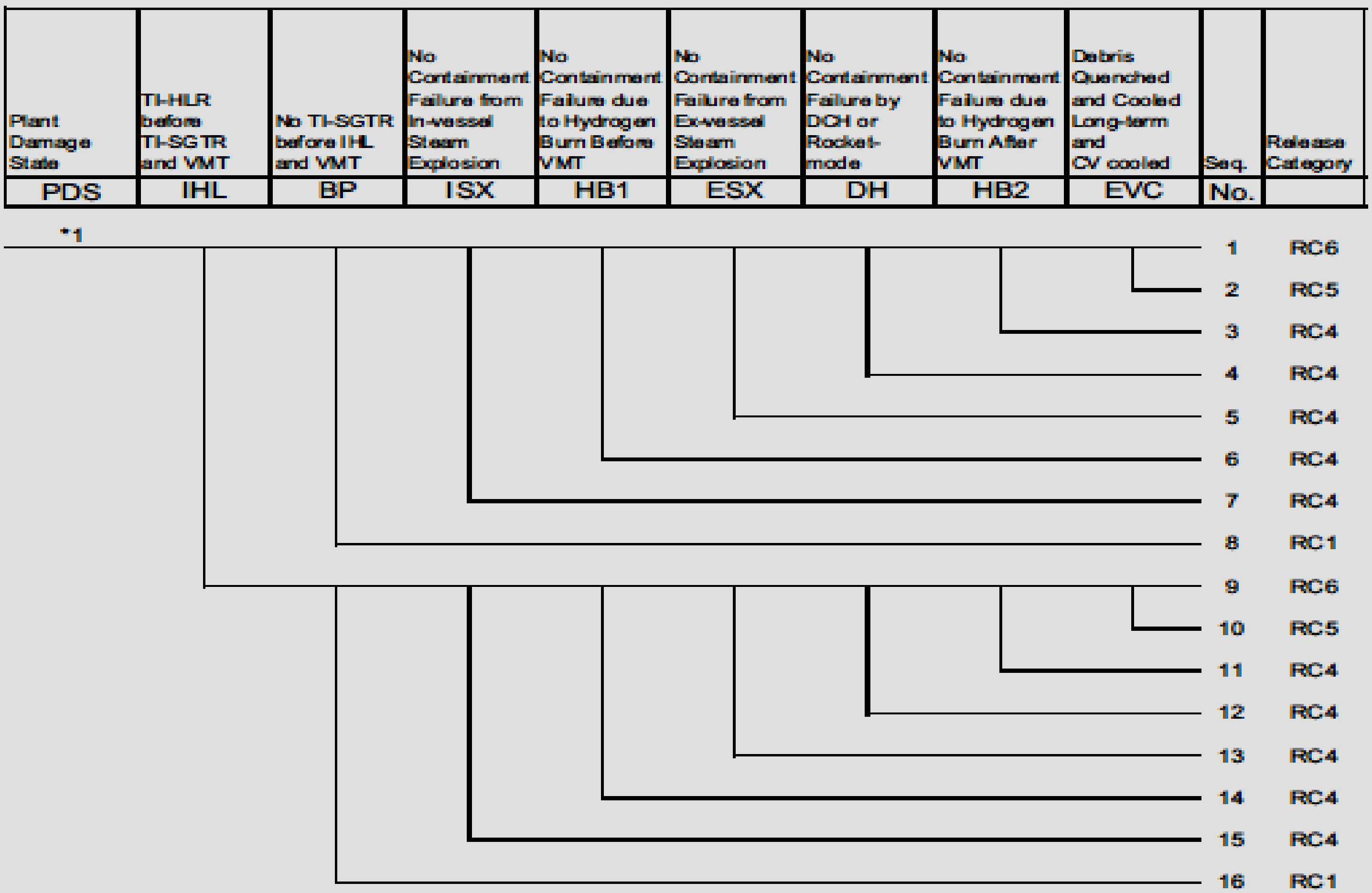
(light blue): Reactor Cavity is flooded when CS success.

(green): As water is not injected into RV in high pressure sequences, C/V failure before core damage never occurs. Injection with charging pumps is not considered.

(orange): The primary system pressure during SGTR accident sequences is equivalent to medium one.

Level 2 PRA and SA Evaluation

Overview of Level 2 PRA and SA Evaluation (continued)



*1 CPET for release categories is developed only PDSs(<1-9><A-H>). Because all other PDS are directly assigned to release category.

Level 2 PRA and SA Evaluation

Overview of Level 2 PRA and SA Evaluation (continued)

Physical Phenomena in CPET	SA Mitigation Features
Temperature-induced hot leg or surge line creep rupture (CPET event IHL) Temperature -induced SGTR (CPET event BP)	Reduce RCS pressure after core damage through depressurization valves (SDVs and severe accident dedicated DVs)
RV failure (CPET event BP)	• Water injection into RV per SA procedure manual • External RV cooling by reactor cavity water In-vessel retention is not credited in the Level 2 PRA due to uncertainty about its efficacy
In-vessel steam explosion (CPET event ISX)	No mitigation features are provided to address in-vessel steam explosion. [The potential for containment failure due to in-vessel steam explosion is considered negligible (NUREG-1524), therefore, this issue is addressed in the PRA]
Ex-vessel steam explosion (CPET event ESX)*	No mitigation features are provided to minimize the potential for ex-vessel steam explosion. [The design approach relies on a robust reactor cavity and RCS piping to withstand the pressure load of an ex-vessel steam explosion]

* Further discussion of topic is provided under technical topics of interest

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Overview of Level 2 PRA and SA Evaluation (continued)

Physical Phenomena in CPET	SA Mitigation Features
Hydrogen mixing and combustion (CPET events HB1 and HB2) *	• 20 strategically located hydrogen igniters, 11 of which are dc-powered and backed-up by dedicated batteries • Open and large containment volume • Containment vessel provides sufficient strength to withstand pressure loads generated by most hydrogen burns
Core debris coolability and molten core concrete interaction, MCCI (CPET event EVC) *	• Flooding reactor cavity to cool debris (CSS, firewater injection) • Design geometry of reactor cavity to enhance spreading of core debris to ensure adequate coolability
High pressure melt ejection (direct containment heating and rocket-mode reactor vessel failure (CPET event DH)	• Reduce RCS pressure after core damage through depressurization valves (SDVs and severe accident dedicated DVs) • Debris trap in reactor cavity, as well as no direct pathway to the upper compartment of containment

* Further discussion of topic is provided under technical topics of interest

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Overview of Level 2 PRA and SA Evaluation (continued)

Physical Phenomena in CPET	SA Mitigation Features
Early (release categories RC1 through RC4) and late containment failure modes including overpressure failure (release category RC5) (CPET event EVC)	Containment overpressure protection is provided through: •Large, high-strength containment •Active containment cooling using CSS, and alternate containment cooling using containment fan coolers and/or fire water system to promote steam condensation
	Equipment survivability (not considered a top event as it is confirmed separately): •The COL applicant is responsible for completing the equipment survivability assessment of the as-built equipment required to mitigate severe accidents to provide reasonable assurance that they will operate in the environmental conditions resulting from the SA for which they are intended, and over the time span for which they are needed (COL Action Item 19.3(7))

Technical Topics of Interest
Ex-Vessel Steam Explosion

- If core debris and water come into contact after RV breach, potential exists for fuel-coolant interaction to cause ex-vessel steam explosion leading to highly energetic impulse loads on structures (CPET event ESX)
- No mitigation features are provided to minimize the potential for ex-vessel steam explosion. Design approach relies on a robust reactor cavity and RCS piping to withstand the pressure loads resulting from ex-vessel steam explosions

Technical Topics of Interest

Ex-Vessel Steam Explosion (continued)

- Applicant’s SA progression analyses conclude:
  - Peak explosive shockwave load due to ex-vessel steam explosion is 1.2×10^{4} psia (evaluated using a modified TEXAS-V code under the most severe conditions in terms of both the possibility and magnitude of steam explosions)
  - Structural capability analysis shows that both reactor cavity wall and RCS piping structures can withstand this shockwave pressure load with sufficient margin (evaluated using finite element analysis employing LS-DYNA code with time-dependent pressure from TEXAS-V code)
  - Applicant concludes that containment can withstand the loads generated by potential ex-vessel steam explosions. Hence, probability of a containment failure due to an ex-vessel steam explosion is judged to be "Very Unlikely," and assigned a CCFP of 0.01

Technical Topics of Interest

Ex-Vessel Steam Explosion (continued)

- Staff's confirmatory calculations using the original TEXAS-V code shows considerably different results:
  - Peak explosive shockwave load is 50% higher than that estimated by the applicant
  - Impulse load shows considerable dependence on the selected fragmentation model parameter
  - Considering the noted differences between the results of the original and the applicant's modified TEXAS-V code calculations, use of the pressure history predicted by the original TEXAS-V code in the US-APWR cavity structural analysis may lead to a significantly lower margin between the calculated plastic strain and the maximum allowable strain
  - Staff issued RAI 19-521, requesting applicant investigate the implications of larger uncertainties in the calculated peak pressure associated with ex-vessel steam explosions

Technical Topics of Interest
Ex-Vessel Steam Explosion (continued)

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- Applicant's response to RAI 19-521:
  - RCS pipe structures have sufficient capacity to withstand challenges from ex-vessel steam explosions over the wider range of uncertainties (based on two finite-element structural analyses, FESAs, for both the RCS pipes and the reactor cavity that assumed a range of 10 percent and 50 percent increase in calculated peak pressure associated with ex-vessel steam explosions)
  - However, the reactor cavity structural integrity cannot be assured under the higher end of the explosions loads (depending on the reactor cavity wall model used in the FESAs). Therefore, a sensitivity analysis is performed to determine the impact of reactor cavity failure on LRF (probability of containment failure due to ex-vessel steam explosion is conservatively increased from 0.01 to 0.1 for the PDSs when the reactor cavity is flooded before vessel melt through at low RCS pressure)
  - Sensitivity analysis shows that the estimated LRF value for all initiators, including LPSD modes, is below the Commission's goal of 1.0E-6/yr

Technical Topics of Interest

Ex-Vessel Steam Explosion (continued)

- Conclusion:

  Pending the staff's verification of the FESAs discussed in the response to RAI 19-521 (Confirmatory Item 19.1-LEVEL2-521), the applicant provided appropriate information on the loads generated by a shock wave from an ex-vessel steam explosion sufficient to address the structural response to ex-vessel steam explosions, and considers the ex-vessel steam explosion issue resolved.

Technical Topics of Interest
Hydrogen Generation and Control

- In a SA, hydrogen would be generated due to oxidation of fuel rod cladding, MCCI, and oxidation of other core and upper plenum structures. Therefore, potential exists for hydrogen combustion leading to containment failure (CPET events HB1 and HB2)
- Mitigation features to minimize containment failure due to hydrogen combustion include:
  - Original design consisted of 20 ac-powered igniters
  - Subsequent to NRC analyses showing potential for hydrogen concentration exceeding 10% inside RWSP, design was modified by providing dc-power to 11 of 20 igniters (with back-up by dedicated 24-hr batteries) (RAI 19-560)
  - Open and large containment volume
  - Containment vessel provides sufficient strength to withstand pressure loads generated by most hydrogen burns

Technical Topics of Interest

Hydrogen Generation and Control (continued)

United States Nuclear Regulatory Commission
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- Applicant's SA progression analysis using GOTHIC conclude:
  - Localized hydrogen burns could be initiated by the igniters in compartments near the release points
  - Global burns in the dome and deflagration to detonation transition (DDT) is not expected, since igniters control hydrogen concentration below 10%
  - Peak static pressures would be below 70 psia, which is well below the containment ultimate pressure capability of 216 psia
  - Flammable atmosphere in RWSP is predicted for MLOCA scenario, but hydrogen concentration remains below 10%
  - Overall conclusion, there is no DDT potential during SAs and that the containment atmosphere is well mixed

Technical Topics of Interest

Hydrogen Generation and Control (continued)

United States Nuclear Regulatory Commission
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- With the exception of potential detonable hydrogen mixture in RWSP, staff's confirmatory analysis using MELCOR code confirms the applicant's findings, with no significant change in containment failure probability due to hydrogen combustion
- However, staff's confirmatory analysis predicted hydrogen concentrations exceeding 10% in the RWSP compartment (due to condensation of steam) during long-term SBO scenarios
- In response to RAI 19-449, the applicant's analysis also showed a potential for hydrogen concentrations exceeding 10% in RWSP. A sensitivity calculation (assuming containment failure due to detonation) showed that total LRF exceeded the Commission's goals

Technical Topics of Interest

Hydrogen Generation and Control (continued)

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- In response to RAI 19-560:
  - Applicant proposes a design change to provide dedicated batteries to 11 out of 20 igniters that will have a capacity for at least 24 hours following onset of SBO and loss of AAC. These igniters are strategically located near potential hydrogen release locations. In addition, as part of SAMGs, the reactor cavity will be flooded by diesel-driven firewater system to provide core debris cooling and prevent MCCI
  - Applicant shows, for a long-term SBO with the proposed dc-powered igniter configuration, that hydrogen concentration inside containment (including RWSP) remains below 10%. Also, containment integrity is maintained for 24 hours after accident
  - Applicant revises Level 2 PRA to reflect this design modification, including modifying the fault tree for the hydrogen control top event in the CSET (which reduces unavailability of igniters for damage states where ac igniters are not functional after SBO (PDS 5E))
  - Level 2 results clearly show significant reductions in LRF for PDS 5E

Technical Topics of Interest

Hydrogen Generation and Control (continued)

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- Staff's confirmatory analysis verifies applicant's results and concurs that the regulatory requirements for meeting the hydrogen combustion challenge is satisfactorily met

- **Open Item 19.1-Level2-560 (RAI 871-6121, Question 19-560)**:
  - It is not clear in the response to RAI 19-560 how the revised hydrogen control top event in the CSET (that modeled the 11 dc-powered igniters) was applied in the Level 2 PRA
  - Applicant provided clarification in a telecom, and staff awaiting final revised response

Technical Topics of Interest

Core Debris Coolability

- In a SA leading to core melting through RV, potential exists for containment failure if molten debris is not sufficiently cooled, e.g., MCCI (CPET event EVC)
- Mitigation features to mitigate this SA type include:
  - ♦ Flooding reactor cavity to cool debris (CSS, firewater injection)
  - ♦ Design geometry of reactor cavity to enhance spreading of core debris to ensure adequate coolability:
    - reactor cavity floor area > 970 ft^2
    - reactor cavity concrete floor thickness > 36 inches

Technical Topics of Interest
Core Debris Coolability (continued)

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- Applicant's SA progression analysis using MAAP 4.0.6 code:
  - For SAs where molten debris dropped into a <u>flooded reactor cavity</u> - debris appropriately cooled and no basemat erosion occurred
  - For SAs where molten debris dropped into a <u>dry reactor cavity, then flooded</u> - debris appropriately cooled and slight basemat erosion occurred (0.1 inches)
  - For SAs where molten debris dropped into a <u>dry reactor cavity, and not flooded</u> - basemat melt through occurred after 28 hours and containment pressure within 24 hours remained below ultimate containment pressure
  - Sensitivity analysis shows basemat melt-through, and containment over-pressurization failure are not expected to occur within 24 hours

Technical Topics of Interest
Core Debris Coolability (continued)

- Molten core debris spreads very well over entire reactor cavity floor:
  - Molten core depth over most of the floor area < 10 inches (prescribed in GL 88-20)
  - Molten core debris accumulation in a very limited area (much less than 1% of cavity floor adjacent to reactor cavity wall) could exceed 10 inches
  - Potential for a non-coolable geometry (i.e., molten core debris accumulation exceeding 10 inches) is treated probabilistically in the Level 2 PRA (CPET event EVC)

Technical Topics of Interest
Core Debris Coolability (continued)

- Staff's confirmatory analysis using MELCOR:
  - For several SA scenarios where debris cooling was assumed to be unavailable, basemat melt through occurs later than 24 hours (assuming uniform and complete spreading)
  - Staff concludes that containment integrity is likely to be maintained for more than 24 hours after onset of core damage. The acceptance criteria regarding core debris cooling and MCCI issues defined in SECY-93-087 and GL 88-20 are satisfactorily met

Technical Topics of Interest

Risk Metrics

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- CDF at-power = 2.8E-6/yr

  (IE CDF = 1.03E-6/yr; Fires CDF = 8.6E-7/yr; Floods CDF = 8.9E-7/yr)

- CDF at LPSD = 2.9E-7/yr
- LRF at-power = 4.6E-7/yr

  (IE LRF = 1.07E-7/yr; Fires LRF = 1.9E-7/yr; Floods LRF = 1.6E-7/yr)

- LRF at LPSD = 2.9E-7/yr
- CCFP = 0.1 (internal events at-power), = 0.16 (at-power)
- Containment integrity maintained for 24 hours following core damage for the more likely SA challenges
- Staff cannot make any final conclusions on how the US-APWR design containment performance compares to the Commission's goals before all open items are resolved

Questions?

ACRONYMS

AAC - alternate alternating current
A/B - auxiliary building
ac - alternating current
ACL - accident class
ACRS - Advisory Committee on Reactor Safeguards
AICC - Adiabatic Isochoric Complete Combustion
ANS - American Nuclear Society
APWR - advanced pressurized water reactor
ASME - American Society of Mechanical Engineers
CCDP - conditional core damage probability
CCF - common-cause failure
CCFP - conditional containment failure probability
CCW - component cooling water
CDF - core damage frequency
CET - containment event tree
CFR - Code of Federal Regulations
COL - combined license
CPET - containment phenomenological event tree
CSET - containment system event tree
CSS - containment spray system
CVCS - chemical and volume control system
CWS - circulating water system
D-RAP - design reliability assurance program
DAS - diverse actuation system
dc - direct current
DC - design certification
DCD - design control document
DCH - direct containment heating
DDT - deflagration-to-detonation transition
DV - depressurization valve
ECCS - emergency core cooing system
EFW - emergency feedwater
EFWS - emergency feedwater system
ESF - engineered safety features

ACRONYMS

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FESA - finite-element structural analyses

FLML - loss of RHR because of failure to maintain water level

GL - Generic Letter

GTG - gas turbine generator

HRA - human reliability assessment

HVAC - heating, ventilation, and air conditioning

HX - heat exchanger

I&C - instrumentation and control

IE - initiating event

LCO - limiting conditions for operation

LOCA - loss of coolant accident

LOOP - loss of offsite power

LPSD - low-power and shutdown

LRF - large release frequency

MCCI - molten core concrete interaction

MCR - main control room

MFWS - main feedwater system

MHI - Mitsubishi Heavy Industries, Ltd

MLOCA - medium loss of coolant accident

MSIV - mainsteam isolation valve

MSS - main steam system

NRC - Nuclear Regulatory Commission

OVDR - loss of RHR because of over-drain

PDS - plant damage state

POS - plant operating states

PRA - probabilistic risk assessment

PWR - pressurized water reactor

R/B - reactor building

RAI - request for additional information

RC - release category

RCS - reactor coolant system

RG - regulatory guide

RHR - residual heat removal

RPS - reactor protection system

RSC - remote shutdown console

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RV - reactor vessel

RWSP - refueling water storage pit

SA - severe accident

SAMDA - severe accident mitigation design alternatives

SAMGs - severe accident management guidelines

SBO - station blackout

SDV - safety depressurization valve

SE - safety evaluation

SER - safety evaluation report

SG - steam generator

SGTR - steam generator tube rupture

SI - safety injection

SLBO - steam line break downstream of MSIV

SMA - seismic margin assessment

SRP - Standard Review Plan

SSC - system, structure, and component

T/B - turbine building

T/D - turbine driven

TR - topical report

TS - technical specifications

yr - year

LUMINANT GENERATION COMPANY

Comanche Peak Nuclear Power Plant, Units 3 and 4

ACRS US-APWR Subcommittee

FSAR Chapter 19 – Probabilistic Risk Assessment and Severe Accident Evaluation

February 22, 2013

Agenda

- Introduction
- SER Open Items
- SER Confirmatory Item
- RMTS and SFCP Methodology
- Site-Specific Aspects

Introduction

- FSAR uses IBR methodology
- No departures from US-APWR DCD
- All COL Items addressed in FSAR
- 7 SER Open Items
- 1 SER Confirmatory Item
- No contentions pending before ASLB

SER Open Items

19-1 RAI 268-6913 Question 19-24

Describe how FSAR will fully address all COL action items listed in DCD Section 19.3

Proposed Resolution – FSAR revised to reflect updated COL action items

SER Open Items (cont'd)

19-2 RAI 268-6913 Question 19-25

Identify and describe use of PRA and risk-informed applications in accordance with RG 1.206 guidance

Proposed Resolution – Cross-references to specific programs and risk-informed applications delineated in FSAR Table 19.1-207

SER Open Items (cont'd)

19-3 RAI 268-6913 Question 19-26

Revise FSAR to address plant-specific PRA technical adequacy including justification that the PRA is sufficient to support the COLA

Proposed Resolution – New FSAR Subsection 19.1.2.3 added

SER Open Items (cont'd)

19-4 External Hazards Risk Evaluation

Since review of FSAR Chapters 2 and 3 is ongoing, staff is unable to finalize its conclusion regarding acceptability of external hazards assessment

Proposed Resolution – To be submitted as part of Luminant's Integrated Seismic Closure and Integrated Hydrology Closure Plans

SER Open Items (cont'd)

19-5 RAI 264-6877 Question 19-21

Document that extreme winds do not contribute more than 10% to the full-power CDF or shutdown CDF compared to the US-APWR

Proposed Resolution – FSAR revised - screening assessment shows that CDFs due to extreme winds is less than 1.0E-7 per year

SER Open Items (cont'd)

19-6 RAI 264-6877 Question 19-22

Update screening discussion in FSAR 19.1.5 to be consistent with RG 1.200 Section 1.2.5 and use site specific PMP data for external flooding screening

Proposed Resolution – FSAR Subsection 19.1.5 revised to address screening process and Table 19.1-205 updated to screen external flooding based on site specific data in FSAR Chapter 2

SER Open Items (cont'd)

19-7 RAI 267-6907 Question 19-23

Clarify how each cost component of the averted cost-risks were determined for internal events with a 7% and 3% discount rate

Proposed Resolution – FSAR Subsection 19.2.6.6 revised to reference more recent cost-risk values in ER Rev 3 Section 7.3

SER Confirmatory Item

19-1 RAI 259-6441 Question 19-20

Address three items with respect to NEI 04-10 Rev 1 and NEI 06-09 Rev 0

Proposed Resolution – "Technical Specifications Methodology for Risk-Managed Technical Specifications and Surveillance Control Program" revised to address all three items

RMTS and SFCP Methodology

- RMTS is controlled by CRMP
- Methodology addressed in “Comanche Peak Nuclear Power Plant, 3&4, Technical Specification Methodology for Risk-Managed Technical Specifications and Surveillance Frequency Control Program” which is adopted by TS 5.5.18 and 5.5.19
- CRMP (TS 5.5.18) IBRs NEI 06-09 Rev 0 and SFCP (TS 5.5.19) IBRs NEI 04-10 Rev 1 with changes to make the NEIs applicable to pre-operational plants

RMTS and SFCP Methodology - CRMP

- Contained in CPNPP procedure which complies with NEI 06-09 Rev 0 as modified and must be implemented before TS 5.5.18 is applied
- Basic program elements in a procedure that designates responsibilities and identifies training requirements
- Program and supporting PRA match as-built plant
- PRA is updated to assess combined risk of unit in current and projected configurations
- Program states how PRA is modified to support CRMP
- Procedure fully describes CRM tool to be used

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RMTS and SFCP Methodology - SFCP

- Contained in CPNPP procedure which complies with NEI 04-10 Rev 1 as modified and must be implemented before TS 5.5.19 is applied
- Basic program elements in a procedure that designates responsibilities and identifies training requirements
- Program and supporting PRA match as-built plant
- PRA is updated to assess combined risk of unit in current and projected configurations
- Program states how PRA is modified to support SFCP

Site-Specific Aspects

19.1 Probabilistic Risk Assessment

- PRA updated to assess site-specific information and external events using systematic process
- PRA to be used during operations to support HFE, SAM, MR, reactor oversight, PM, and reliability programs
- PRA to be updated to reflect RITS, RMTS, SFCP
- Changes to PRA inputs/new information evaluated to determine if PRA maintenance/upgrade needed
- Only site-specific UHS design has potential effect on level 1 and level 2 PRAs, but it is very small

19.1 PRA (cont'd)

- ASME/ANS RA-Sa-2009 has screening criteria for external events
  - CPNPP 3&4 uses E-07 CDF to screen (advanced LWR)
  - CPNPP 3&4 performs bounding analysis for frequency > E-07 to confirm each power operation and LPSD external event CDF < E-07
  - Tornadoes (probability ~E-07) only events not screened
- Total CDF by tornado strike at power < 8E-08/RY and does not contribute > 10% of total shutdown CDF
- Tornado during LPSD does not have significant contribution to risk

19.1 PRA (cont'd)

- Based on site-specific UHS/ESWS design, there is no discernible effect on internal fire, internal flooding, or LPSD PRA results
- PRA for RMTS, SFCP, and peer review available 1 year prior to fuel load

19.2 Severe Accident Evaluation

- Survivability assessment of SAM equipment not already tested will be performed prior to fuel load
- Accident management program, procedures, and training will be developed. Training complete prior to fuel load.
- Evaluation process for SAMAs limited to demonstrating that CPNPP is bounded by DCD analysis and determining magnitude of changes that would be cost-effective
  - Maximum averted cost-risk is so low that no additional design changes are cost-effective
  - Further evaluation of design-relates SAMAs not warranted

DCD Ch 19 Appendices

- Appendix A – US-APWR Beyond Design Basis Aircraft Impact Assessment (IBR in COLA Rev. 3)
- Appendix B – US-APWR Beyond Design Basis PSMS Reliability Analysis (will be IBR in COLA Rev.4)

COLA Part 9 “Withheld Information”

Loss of Large Areas of the Plant due to Explosion or Fire (SRI)

- NEI 06-12 Rev. 3 (SRI)
- ISG-016 Rev. 0 (SRI)

Acronyms

ASLB	Atomic Safety and Licensing Board
ASME/ANS	American Society of Mechanical Engineers/American Nuclear Society
COL	Combined license
CDF	Core Damage Frequency
COLA	Combined license application
CPNPP	Comanche Peak Nuclear Power Plant
CRM	Configuration Risk Management
CRMP	Configuration Risk Management Program
DBE	Design basis event
DCD	Design Control Document
ER	Environmental Report
ESWS	Essential service water system
FSAR	Final Safety Analysis Report
HCLPF	High confidence of low probability of failure
HFE	Human factors engineering
IBR	Incorporated by reference
LPSD	Low-power and shutdown
LWR	Light water reactor
MR	Maintenance rule

Acronyms (cont'd)

- NEI — Nuclear Energy Institute
- OI — Open Item
- PM — Preventive Maintenance
- PMF — Probable maximum flood
- PMP — Probable maximum precipitation
- PRA — Probabilistic Risk Assessment
- PSMS — Protection and Safety Monitoring System
- RAI — Request for Additional Information
- RG — Regulatory Guide
- RITS — Risk-Informed Technical Specifications
- RMTS — Risk-Managed Technical Specifications
- RY — Reactor-year
- SAM — Severe accident mitigation
- SAMA — Severe accident mitigation alternative
- SER — Safety Evaluation Report
- SFCP — Surveillance frequency control program
- SMA — Seismic Margin Analysis
- SRI — Security related information
- SSE — Safe-shutdown earthquake
- UHS — Ultimate heat sink
- US-APWR — United States Advanced Pressurized Water Reactor

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Presentation to the ACRS Subcommittee

Comanche Peak Nuclear Power Plant, Units 3 and 4
COL Application Review

Safety Evaluation Report with Open Items

Chapter 19: PROBABILISTIC RISK ASSESSMENT & SEVERE ACCIDENT EVALUATION

February 21-22, 2013

Staff Review Team

- **Technical Staff**
  - **Hanh Phan** (Lead), Senior PRA Analyst
    PRA and Severe Accidents Branch
  - **Marie Pohida,** Senior PRA Analyst
    PRA and Severe Accidents Branch
  - **Todd Hilsmeier,** PRA Analyst
    PRA and Severe Accidents Branch
- **Project Managers**
  - **Stephen Monarque**
  - **Ruth Reyes**

Presentation Outline

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Section 19.1 - Probabilistic Risk Assessment

- COL Information Items
  1) Open Items
  2) Technical Topics of Interest

Section 19.2 - Severe Accident Evaluation

- COL Information Item
  1) Open Item
  2) Technical Topics of Interest

Review Approach

- Verified application against the corresponding sections in DCD to ensure that the combined information of the DCD and FSAR represents a complete scope of Chapter 19
- Discussed plant-specific information with other technical branches
- Ensured review consistency with other COL applications
- Ensured review consistency with the analyses documented in COL FSAR (e.g., Chapter 2 “Site Characteristics,” Chapter 3, “Design of Structures, Systems, Components and Equipment,” and Chapter 16 “Technical Specifications”)

Description of SE Open Items

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- **Open Item 19-1** (RAI 6913, Question 19-24) - Revise FSAR to fully address all COL information items listed in US-APWR DCD Section 19.3 in light of the US-APWR DC RAI 6790, Question 19-574, dated October 9, 2012

- **Open Item 19-2** (RAI 6913, Question 19-25) - Identify and describe the use of PRA and risk-informed applications during the COL application phase and construction phase

- **Open Item 19-3** (RAI 6913, Question 19-26) - Provide the supplemental information in FSAR to address plant-specific PRA technical adequacy including the justification that the PRA is sufficient to support the CPNPP 3&4 COLA

Description of SE Open Items (continued)

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- **Open Item 19-4 * -** The staff's acceptance of Chapter 19 external hazards will be contingent, in part, on the completion of the review of FSAR Chapters 2 and 3
- **Open Item 19-5 *** (RAI 6877, Question 19-21) - Modify full-power and shutdown extreme wind analysis and submit the updated PRA results
- **Open Item 19-6** (RAI 6877, Question 19-22) - Update the screening discussions on external flooding described in Section 19.1.5 of the COLA FSAR to be consistent with RG 1.200 screening criteria
- **Open Item 19-7 *** (RAI 6907, Question 19-23) - Address the discrepancy in averted cost between COL FSAR and COL environmental report

(* Open items will be discussed in Technical Topics of Interest)

Technical Topics of Interest
External Hazards Risk Evaluation

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- RG 1.200, Section 1.2.5 “Screening and Conservative Analysis of Other External Hazards Technical Elements”

  “It is recognized that for those new reactor designs with substantially lower risk profiles (e.g., internal events CDF below 1E-6/yr), the quantitative screening value should be adjusted according to the relative baseline risk value.”

- Consistent with the list of external hazards identified in ASME/ANS RA-Sa-2009, Appendix 6-1 “List of External Hazards Requiring Consideration”

- Staff's acceptance of Chapter 19 external hazards will be contingent, in part, on the completion of the review of FSAR Chapters 2, “Site Characteristics” and 3, “Design of Structures, Systems, Components, and Equipment”

Technical Topics of Interest
High Winds other than Tornadoes

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- Site specific extreme wind speed (other than tornado) documented in FSAR (Table 2.0-1R, Page 2.0-2) as 96 mph in 1/100 years
- Non-safety related equipment and structures (including switchyard) designed to site specific extreme windspeed
- Staff to confirm at beyond site specific wind speed, CDF not greater than 10% of full power and shutdown operation
- Staff reviewing latest RAI response 12/2012
- AAC generators and all supporting equipment will be housed in Category 1/Category 2 structures

Technical Topics of Interest
Plant-Specific Information

- Plant-specific design and operational changes or departures from the certified design are described in FSAR Table 1.8-1R, “Significant Site-Specific Interfaces with the Standard US-APWR Design”
- Requested a systematic search for site-specific or plant-specific factors, i.e., LOOP frequency, offsite power recovery probability, the maximum ambient temperature used in HVAC design calculations (RAI 3214, Question 19-4)

Technical Topics of Interest
Risk-Informed Technical Specifications

- Luminant requested NRC approval to implement NEI Topical Report 06-09, “Risk Managed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines,” and NEI Topical Report 04-10, “Risk Informed Technical Specifications Initiative 5b, Risk Informed Method for Control of Surveillance Frequencies”
- NRC issued its SEs that approved NEI Topical Reports 06-09, Rev. 0 (ML071200238) and NEI 04-10, Rev. 1 (ML072570267) on May 17, 2007 and September 19, 2007, respectively

Technical Topics of interest

Risk-Informed Technical Specifications (continued)

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- Approval of risk-informed technical specifications (RITS) implementation at the COLA stage (before the development of an as-built, as-to-be-operated plant-specific PRA model)
  - PRA required per 10 CFR 50.71(h)(1) (which will be used for RITS implementation) not available at the time of COL issuance
  - Application-specific infrastructure, such as procedures, training, software, and programs used during operations not available at the time of COL issuance

Technical Topics of Interest

Risk-Informed Technical Specifications (continued)

- February 18, 2009, public meeting to discuss implementation of risk metrics for new reactor risk-informed applications
- April 2, 2009, public meeting between NRC, Luminant, and MHI on RITS
- Public meetings between NRC and Luminant on RMTS
  - January 11, 2011
  - March 30, 2011
  - June 30, 2011
- October 20, 2011, presentation to ACRS on CPNPP COL RITS

Technical Topics of interest

Risk-Informed Technical Specifications (continued)

- Methodology referenced in the TS provides necessary changes to information in NEI Topical Reports 06-09 and 04-10 (also referenced in the TS) for application to CPNPP 3 & 4
  - Applicant submitted “Comanche Peak Nuclear Power Plant, 3 and 4, Technical Specification Methodology for Risk-Managed Technical Specifications and Surveillance Frequency Control Program” (ML1118232229)

Technical Topics of Interest

Risk-Informed Technical Specifications (continued)

- PRA upgrades and updates to be performed by the applicant to support RITS (response to RAI 3287, Question 19-3):
  - Site-specific models will be included in the first series of PRA upgrades
  - Emergency operating procedures and detailed design information will be reflected in the PRA (during the second series of PRA upgrades)
  - Uncertainties on PRA model will be identified and addressed (during the PRA upgrades)
  - Peer review will be performed and findings will be resolved prior to initial fuel load

Technical Topics of Interest
Risk-Informed Technical Specifications (continued)

- In its response to RAI 3287, Question 19-3, the applicant provided the following statements/commitments:

  "The PRA for RMTS must basically meet Capability Category 2 for the supporting requirements of the ASME/ANS internal events at power PRA standard. The scope of the PRA model must include Level 1 (CDF) plus large early release frequency (LERF). Contributions from external events, internal flooding events, and internal fire events must also be considered. The PRA for RMTS will be updated to satisfy the PRA technical adequacy described in the NEI guideline and will be available one year prior to fuel load."

Technical Topics of Interest

Severe Accident Mitigation Alternatives

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- *"The Probabilistic Risk Assessment and Severe Accident Evaluation is updated as necessary to assess specific site information and associated site-specific external events …"*
- In SAMA analysis, applicant updated the maximum averted cost for 7% and 3% discount rates using site-specific information
- Applicant concluded that there are no cost-effective design SAMAs
- RAI 19-23 requests applicant to clarify how each component of the averted cost in SAMA was determined for 7% and 3% discount rates
- The staff has not completed its evaluation of the applicant's response to RAI 19-23 (**Open Item 19-7**)

ACRONYMS

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AAC - alternate alternating current
ANS - American Nuclear Society
APWR - advanced pressurized water reactor
ASME - American Society of Mechanical Engineers
CDF - core damage frequency
COL - combined license
COLA - combined license application
CP - Comanche Peak
CPNPP - Comanche Peak nuclear power plant
DC - design certification
DCD - design control document
FSAR - final safety analysis report
gpm - gallons per minute
HVAC - heating, ventilation, and air conditioning
LERF - large early release frequency
LOOP - loss of offsite power
MHI - Mitsubishi Heavy Industries, Ltd
mph - miles per hour
NEI - Nuclear Energy Institute
NRC - Nuclear Regulatory Commission
PRA - probabilistic risk assessment
RAI - request for additional information
RG - regulatory guide
RI - risk-informed
RITS - risk-informed technical specifications
RMTS - risk-managed technical specification
SA - severe accident
SAMA - severe accident mitigation alternatives
SER - safety evaluation report
SFCP - surveillance frequency control program
SRP - Standard Review Plan
TS - technical specifications
yr - year